

Davis Besse 1LOT21 NRC Written Exam Rev. 1

1. Initial Plant Conditions:

- The Reactor was tripped due to Main Turbine High Vibrations
- DB-OP-02000, Section 4.0 "Supplemental Actions" are in progress

Current Plant Conditions:

- Steam Generator 1 Pressure 975 psig and stable
- Security Reports steam issuing from a MSSV located on the West side of the Aux Building roof
- NNI X AC Power Available Light is OFF
- Annunciator 14-2-D "ICS/NNI 118V AC PWR TRBL" is in alarm
- Annunciator 6-5-C "SEAL INJ FLOW LO" is in alarm
- Control Rod 3-1 has indications of being stuck out at 100%

Based on Current Plant Conditions, which of the following actions will be completed NEXT?

- A. Emergency Borate the Reactor Coolant System refer to DB-OP-02516 "CRD Malfunctions"
- B. Place #1 Atmospheric Vent Valve in HAND and lower Steam Generator 1 Pressure refer to DB-OP-02525 "Steam Leaks"
- C. Initiate AND Isolate Steam Feed Rupture Control System (SFRCS) refer to DB-OP-02532 "Loss of NNI/ICS Power"
- D. Start the STBY Makeup Pump refer to DB-OP-02512 "Makeup and Purification System Malfunctions"

Davis Besse 1LOT21 NRC Written Exam Rev. 1

2. Initial Conditions:
- 100% power
 - Makeup Pump 1 is in service
 - Component Cooling Water Pump 1 is in service

A plant transient occurs which causes the following Annunciators:

- 6-5-C "SEAL INJ LO"
- 6-6-C "SEAL INJ TOTAL FLOW"
- 1-3-D "BUS C1 L/O"
- 4-2-E "PZR LVL LO"

All systems responded as expected and all Immediate Actions have been completed. Which of the following actions is required NEXT IAW DB-OP-02512 "Makeup and Purification System Malfunctions" to mitigate this event?

- A. Monitor RCP Seal performance. REFER TO DB-OP-02515 "Reactor Coolant Pump and Motor Abnormal Operation"
- B. Verify CCW is being supplied to the RCPs. REFER TO Attachment 1 "Verification Of CCW Flow To Reactor Coolant Pumps"
- C. Trip the Reactor, GO TO DB-OP-02000 "RPS, SFAS, SFRCS Trip, or SG Tube Rupture"
- D. Trip the Reactor, Stop ALL RCPs, GO TO DB-OP-02000 "RPS, SFAS, SFRCS Trip, or SG Tube Rupture"

Davis Besse 1LOT21 NRC Written Exam Rev. 1

3. Following an ATWS event the Reactor was shutdown by momentarily Deenergizing 480-volt Unit Substations E2 AND F2.

Based on the above event, complete the following statements:

This event will be classified as an _____ (1) _____.

After the rods drop into the core, Absolute Position Indication for Groups 1 through 7 will _____ (2) _____ when power is restored to E2 and F2.

REFERENCE PROVIDED

- A. (1) Unusual Event
(2) indicate 0%
- B. (1) Unusual Event
(2) NOT be available
- C. (1) Alert
(2) indicate 0%
- D. (1) Alert
(2) NOT be available

Davis Besse 1LOT21 NRC Written Exam Rev. 1

4. The plant is operating at 100% power.

The breaker for MS107 “Steam Generator 2 to Auxiliary Feed Pump Turbine 2” trips open and cannot be reset.

Which of the following describes the required action?

The (1) Limiting Condition for Operation must be restored within (2).

- A. (1) Steam and Feed Rupture Control System Actuation Logic
(2) 72 hours
- B. (1) Emergency Feedwater System
(2) 7 days
- C. (1) Steam and Feed Rupture Control System Actuation Logic
(2) 7 days
- D. (1) Emergency Feedwater System
(2) 72 hours

Davis Besse 1LOT21 NRC Written Exam Rev. 1

5. Initial conditions
- Plant shutdown in progress
 - Reactor Power is 12%
 - Motor Driven Feedwater Pump is lined up in the Main Feedwater mode

The following occurs:

- Loss of Offsite Power
- Loss of all Feedwater

Current conditions:

- AFW Pump 2 is restored and is feeding Steam Generator 2
- AFW Pump 1 cannot be restored and is still unavailable
- Steam Generator 1 has boiled dry
- Steam Generator 1 Tube to Shell differential temperatures are within limits
- Station Blackout Diesel Generator has been started
- AF3871 "Auxiliary Feed Pump 2 to Steam Generator 1" is failed closed

The Command SRO directs the Reactor Operator to restore feedwater to Steam Generator 1, using DB-OP-02000 "RPS, SFAS, SFRCS TRIP, OR SG Tube Rupture" Attachment 5 "Guidelines for Restoring Feedwater".

- (1) Which Section of Attachment 5 will the Command SRO direct the Reactor Operator to perform?

AND

- (2) Which of the following actions will be required to restore flow to Steam Generator 1?

- A. (1) Section A: Motor Driven Feedwater Pump in the Main Feedwater Mode
(2) Block AND Reset SP7B using both HIS SP7AB and HIS SP7CB on panel C5792N, MSIV/MFW Control Valve Reset Switch Panel
- B. (1) Section A: Motor Driven Feedwater Pump in the Main Feedwater Mode
(2) Block AND Reset SP7B using both HIS SP7AB and HIS SP7CB on panel C5712, CTRM Right Console (MFW Control) Panel
- C. (1) Section B: Emergency Feedwater Pump via the Auxiliary Feedwater header
(2) Throttle EFW Flow to Steam Generator 1 using HCEF8-2, EFWP DISCHARGE FLOW VALVE on panel C5732, EFW Control Panel
- D. (1) Section B: Emergency Feedwater Pump via the Auxiliary Feedwater header
(2) Throttle EFW Flow to Steam Generator 1 using HCEF8-2, EFWP DISCHARGE FLOW VALVE on panel C5706, CTRM Center Console (AFW Control) Panel

Davis Besse 1LOT21 NRC Written Exam Rev. 1

6. Plant Conditions:
- The Reactor tripped due to a loss of Off-site power
 - Both Essential Busses C1 AND D1 remain de-energized AND attempts to start the SBODG and Both EDGs IAW Specific Rule 6 actions have failed

Based on the Plant Conditions, what procedure will be utilized and how will Steam Generator 2 level be initially controlled?

- A. DB-OP-02521 "Loss of AC Bus Power Sources" will direct Steam Generator 2 to be controlled at the required level by adjusting HIS521A "AFPT 2 GOVERNOR".
- B. DB-OP-02521 "Loss of AC Bus Power Sources" will direct closing AF599 "AFW TO SG 2 LINE STOP" to terminate an AFW Overcooling event.
- C. DB-OP-02704 "Extended Loss of AC Power DC Load Management" will direct Steam Generator 2 to be controlled at the required level by adjusting HIS521A "AFPT 2 GOVERNOR".
- D. DB-OP-02704 "Extended Loss of AC Power DC Load Management" will direct closing AF599 "AFW TO SG 2 LINE STOP" to terminate an AFW Overcooling event.

Davis Besse 1LOT21 NRC Written Exam Rev. 1

7. A planned power reduction from 100% power, with ICS in full Automatic was in progress to remove #1 Condensate Pump from service for maintenance. When at 95% power, the following indications were received:
- (5-1-E) CRD SYSTEM FAULT
 - (5-2-E) CRD ASYMMETRIC ROD
 - Tave lowered to approximately 579°F
 - NI 5 indicates 89%
 - NI 6 indicates 96%
 - NI 7 indicates 88%
 - NI 8 indicates 95%

Based on the indications above, complete the following statements.

DB-OP-02516, CRD MALFUNCTIONS will be entered for indications of a (1).

IAW DB-OP-002516, CRD MALFUNCTIONS, reactor power will be lowered to (2) RTP.

- A. (1) Stuck Rod
(2) 60%
- B. (1) Stuck Rod
(2) 50%
- C. (1) Dropped Rod
(2) 60%
- D. (1) Dropped Rod
(2) 50%

Davis Besse 1LOT21 NRC Written Exam Rev. 1

8. A plant startup-up is in progress with the following conditions:
- Reactor power, as indicated on NI 3 and NI 4 (intermediate range detectors), is 1×10^{-8} amps
 - All systems are in normal alignment for this condition

A fuse in the power supply to the NI 3 detector blows (detector supply voltage is zero).

(1) How will the NI 1 and NI 2 (source range detectors) respond to this blown fuse?

(2) IAW Technical Specifications, what actions are required?

REFERENCE PROVIDED

- A. (1) re-energize
(2) within 2 hours, Reduce neutron flux to $\leq 1E-10$ amp
- B. (1) re-energize
(2) once per 12 hours, Verify SDM is within the limits specified in the COLR
- C. (1) remain de-energized
(2) within 2 hours, Reduce neutron flux to $\leq 1E-10$ amp
- D. (1) remain de-energized
(2) once per 12 hours, Verify SDM is within the limits specified in the COLR

Davis Besse 1LOT21 NRC Written Exam Rev. 1

9. Given the following conditions:

- The crew has entered DB-OP-02518 “High Condenser Pressure”
- Reactor power is at 34%
- Condenser Pressure is at 5.6 inches HgA and rising slowly
- Turbine Load is at 235 MWe

IAW DB-OP-02518 “High Condenser Pressure” which ONE of the following actions is required next?

- A. Reduce Reactor power to maintain Condenser pressure less than or equal to 5.0 inches HgA. REFER TO DB-OP-02504 “Rapid Shutdown”.
- B. Trip the Turbine. REFER TO DB-OP-02500 “Turbine Trip”.
- C. Trip the Reactor AND GO TO DB-OP-02000 “RPS, SFAS, SFRCS Trip, or SG Tube Rupture”.
- D. Trip the Reactor AND Initiate AND Isolate SFRCS THEN GO TO DB-OP-02000 “RPS, SFAS, SFRCS Trip, or SG Tube Rupture”.

Davis Besse 1LOT21 NRC Written Exam Rev. 1

10. The plant is operating at 100% power in a normal system alignment.

The following plant conditions are noted:

- Annunciator 4-1-B SUBCOOL MARGIN LO comes into alarm.
- The plant remains steady at 100% power.
- It is determined that Ch 2 SCM Meter will NOT respond as required.

Which of the following is required due to a failed RC System Subcooling Margin Monitor?

REFERENCE PROVIDED

- A. Only TNC 8.3.7, Post Accident Monitoring (PAM) Instrumentation is NOT met, comply with Nonconformance A.
- B. Only LCO 3.3.17, Post Accident Monitoring (PAM) Instrumentation is NOT met, comply with Condition A.
- C. Both TNC 8.3.7 and LCO 3.3.17 are NOT met, comply with TNC 8.3.7 Nonconformance A and LCO 3.3.17 Condition A.
- D. Both TNC 8.3.7, Post Accident Monitoring (PAM) Instrumentation and LCO 3.3.17, Post Accident Monitoring (PAM) Instrumentation are met, no action is required.

Davis Besse 1LOT21 NRC Written Exam Rev. 1

11. The plant is operating at 100% power with all systems in a normal alignment.

At 0800, a reactor trip occurs. SFAS Actuates on Low RCS Pressure, Low-Low RCS Pressure and High Containment Pressure.

At 0815, The Zone 3 Equipment Operator reports HPI Pump 1 discharge pressure is approximately 150 psig and has an abnormal running noise.

At 0830, BWST level is 39 feet and lowering and level will reach 9 feet at 1630.

At 0845, LPI Train 1 AND 2 indicate 0 gallons per minute.

At 0900, Incore temperatures have stabilized at approximately 480 °F with RCS pressure at 500 psig.

Which ONE (1) of the following DB-OP-02000 Attachments provides the required actions that mitigate these plant events?

- A. Attachment 11, HPI Flow Balancing.
- B. Attachment 12, Establishing Long Term Boron Dilution.
- C. Attachment 14, Establishing HPI Alternate Minimum Recirc Flowpath.
- D. Attachment 22, Cross Connect LPI Pump Discharge.

Davis Besse 1LOT21 NRC Written Exam Rev. 1

12. Which of the following transients or failures is the Reactor Protection System credited for to ensure the Reactor Coolant System Pressure Safety Limit is not exceeded, per Technical Specification Bases?
- A. Make-up flow transient
 - B. Pressurizer Spray Valve failure
 - C. Pressurizer Pilot Operated Relief Valve (PORV) failure
 - D. Rod withdrawal transient from subcritical condition

Davis Besse 1LOT21 NRC Written Exam Rev. 1

13. Reactor Coolant Pump 1-1 was stopped at 70% power IAW Attachment 1 of DB-OP-02515 "Reactor Coolant Pump and Motor Abnormal Operation" due to high vibration.

The BOP Reactor Operator checks for proper Feedwater (FW) flow ratios and reports SP6A "SG 2 MFW Control Valve" is not responding to FW Loop Demand.

Which of the following statements describes the impact of the failed MFW Control Valve and the appropriate direction from the CSRO?

- A. SG2 will experience an overfeed condition, the CSRO will direct the BOP Reactor Operator to place Both FW Loop Demands, SP6A and SP7A in Manual to control FW Flow
- B. SG2 will experience an overfeed condition, the CSRO will direct the ATC Reactor Operator to Trip the Reactor and Initiate AND Isolate SFRCS
- C. SG2 will experience an underfeed condition, the CSRO will direct the BOP Reactor Operator to place Both FW Loop Demands, SP6A and SP7A in Manual to control FW Flow
- D. SG2 will experience an underfeed condition, the CSRO will direct the ATC Reactor Operator to Trip the Reactor and Initiate AND Isolate SFRCS

Davis Besse 1LOT21 NRC Written Exam Rev. 1

14. The following plant conditions exist:

- The plant is DEFUELED
- C1 electrical bus is tagged out for maintenance
- EDG 2 is operable
- The SBODG is functional

Which one of the following conditions would require a declaration of an ALERT if the event was expected to last for more than 15 minutes?

REFERENCE PROVIDED

- A. A D1 electrical bus lockout
- B. A loss of the SBODG and EDG 2
- C. A loss of offsite power concurrent with a loss of EDG 2
- D. A loss of offsite power concurrent with a loss of the SBODG

Davis Besse 1LOT21 NRC Written Exam Rev. 1

15. Initial conditions:

- Plant is at 100% Power
- Emergency Diesel Generator (EDG) 1 is paralleled to the grid during testing

The following occurs:

- Loss of Offsite Power (LOOP)
- EDG 2 trips

Which of the following correctly states the response of EDG 1 to the LOOP and which procedure which will be entered NEXT?

- A. EDG 1 output breaker, AC 101, will open immediately due to 13.8 Kv Bus Undervoltage Relay
GO TO DB-OP-02521, Loss of AC Bus Power Sources
- B. EDG 1 will remain loaded until an undervoltage condition occurs and isolates the essential bus
GO TO DB-OP-02521, Loss of AC Bus Power Sources
- C. EDG 1 output breaker AC 101, will open immediately due to 13.8 Kv Bus Undervoltage Relay
GO TO DB-OP-02000, RPS, SFAS, SFRCS Trip OR SG Tube Rupture
- D. EDG 1 will remain loaded until an undervoltage condition occurs and isolates the essential bus
GO TO DB-OP-02000, RPS, SFAS, SFRCS Trip OR SG Tube Rupture

Davis Besse 1LOT21 NRC Written Exam Rev. 1

16. The plant is operating at 100% power with all systems in normal alignment EXCEPT:
- Maintenance is being performed in Reactor Protection System (RPS) Channel (CH) 1 to replace the power supply to NI 6.
 - Limiting Conditions for Operations (LCO) 3.3.1 Reactor Protection System (RPS) Instrumentation is not met. Condition A has been entered for RPS CH 1 being inoperable and required action is complete.

While maintenance is being performed in RPS CH 1, RPS CH 4 spuriously trips on High RCS Pressure and the reactor does NOT trip.

What is the Required Action for Limiting Conditions for Operations (LCO) 3.3.1 Reactor Protection System (RPS) Instrumentation?

- A. Enter Condition A for RPS CH 4, a Separate Condition entry is allowed for each Function
- B. Enter Condition B for RPS CH1 and CH4, place CH 4 in Bypass within one hour
- C. Enter Condition B, for RPS CH1 and CH4, verify CH 4 is tripped within one hour
- D. Enter Condition A for RPS CH4 and Condition B for RPS CH 1 and CH 4, be in MODE 3 within 6 hours

Davis Besse 1LOT21 NRC Written Exam Rev. 1

17. Plant conditions:

- A large break LOCA has occurred
- DB-OP-02000 “RPS, SFAS, SFRCS TRIP, or SG Tube Rupture” Section 5.0 “Lack of Adequate Subcooling Margin” is in progress
- PAM Channel 2 is NOT available

Which of the following indications provide confirmation that routing to Section 9 “Inadequate Core Cooling” is required?

- A. P/T plot is in Region 2 “RCS Superheated” and trending parallel to the saturation curve
- B. Two working incore detectors are displaying a NEG MARGIN
- C. P/T plot is in Region 2 “RCS Superheated” and trending toward Region 3
- D. RCS Pressure is less than 150 psig and Channel 1 Tsat meter is blank

Davis Besse 1LOT21 NRC Written Exam Rev. 1

18. DB-SS-04150, MAIN TURBINE STOP VALVE TEST and DB-SS-04159, ONLINE ELECTRICAL TRIP DEVICE TEST are BOTH scheduled to be performed during the shift.

Complete the following statements.

During the performance of _____ (1) _____ the ARTS TURBINE-GEN Bypass Switch is required to be placed in the BYPASS position.

When the ARTS TURBINE-GEN Bypass Switch is placed in the BYPASS position, LCO 3.3.16 Anticipatory Reactor Trip System (ARTS) Instrumentation Condition A _____ (2) _____ met.

- A. (1) DB-SS-04159, ONLINE ELECTRICAL TRIP DEVICE TEST
(2) is
- B. (1) DB-SS-04159, ONLINE ELECTRICAL TRIP DEVICE TEST
(2) is not
- C. (1) DB-SS-04150, MAIN TURBINE STOP VALVE TEST
(2) is
- D. (1) DB-SS-04150, MAIN TURBINE STOP VALVE TEST
(2) is not

Davis Besse 1LOT21 NRC Written Exam Rev. 1

19. The plant is at 100% power at minimum staffing levels.

During turnover the oncoming At The Controls (ATC) Reactor Operator notifies the Unit Supervisor that he has started taking a new blood pressure medication.

Upon further investigation it has been determined that the Medical Review Officer has not reviewed the change in medication.

As the Unit Supervisor, what action is required to ensure minimum staffing levels are met?

- A. Allow the oncoming ATC RO to take the watch with additional oversight and notify the MRO of the change of medication.
- B. Hold over the current ATC RO within requirements of fatigue rule and immediately call out for a replacement.
- C. Allow the current ATC RO to leave when a relief has been contacted since the relief will arrive within 2 hours.
- D. The Medical Review Officer has 30 days to complete a review of the change in medical status, no further action is required.

Davis Besse 1LOT21 NRC Written Exam Rev. 1

20. The crew has entered DB-OP-02531 "Steam Generator Tube Leak" for a Steam Generator Tube Leak on #1 Steam Generator. During the plant shutdown, the ATC Crew Updates "Steam Generator Tube leak rate is now greater than 50 gpm".

With these current conditions per NOP-OP-1002, Conduct of Operations, complete the following statements.

1. The _____ shall clearly announce to the crew when transitioning to DB-OP-02000 "SFAS, RPS, SFRCS Trip or SG Tube Rupture".
 2. A _____ should be used for this announcement.
-
- A.
 1. Shift Manager
 2. Crew Update
 - B.
 1. Shift Manager
 2. Transient Crew Brief
 - C.
 1. Command SRO
 2. Crew Update
 - D.
 1. Command SRO
 2. Transient Crew Brief

Davis Besse 1LOT21 NRC Written Exam Rev. 1

21. The Plant is operating at 100%
- Technical Specification LCO 3.4.11 requires the PORV and associated block valve shall be OPERABLE.
 - Surveillance DB-PF-03811, MISCELLANEOUS VALVES TEST, is performed to meet this requirement

While performing DB-PF-03811, MISCELLANEOUS VALVES TEST

- RC11, PRESSURIZER RELIEF POWER ISO VALVE CLOSES but WILL NOT OPEN.
- Maintenance finds a bad power supply breaker to the MOV and replaces the entire breaker assembly at the MCC.
- ALL of their required work package instructions have been completed.
- The tagout has been lifted, RC11 is ENERGIZED and CLOSED.
- RC11 is ready for operations' post-maintenance testing.

For these conditions:

What MINIMUM post-maintenance testing will be REQUIRED to verify compliance with Technical Specification LCO 3.4.11?

(For each of the below actions, assume all valve stroke times and indications are within acceptable limits)

- A. Open RC11, PRESSURIZER RELIEF POWER ISO VALVE, no other actions required.
- B. Open RC11, PRESSURIZER RELIEF POWER ISO VALVE; then Close; then re-open.
- C. Cycle RC2A, PRESSURIZER POWER OPERATED RELIEF VALVE through one complete cycle, then open RC11.
- D. Cycle RC2A, PRESSURIZER POWER OPERATED RELIEF VALVE through one complete cycle, then open RC11; then Close; then re-open.

Davis Besse 1LOT21 NRC Written Exam Rev. 1

22. An overhead Annunciator Alarm in the Control Room is not operating properly.

To avoid nuisance alarms, the Operations Manager has determined that the Annunciator will be disabled by removing the affected Annunciator Point Card.

Which of the following documents must be completed to remove this point card to disable the affected annunciator alarm?

1. Annunciator System Operating Procedure
 2. Work Order for point card removal
 3. 10CFR50.59 RAD, Screen, and/or Evaluation
 4. Engineering Change Package
 5. Maintenance Information Tag
 6. Clearance and Red Danger Tags
- A. 1 and 3
- B. 2, 4, and 5
- C. 1 and 6
- D. 2, 3 and 6

Davis Besse 1LOT21 NRC Written Exam Rev. 1

23. The Miscellaneous Waste Monitor Tank (MWMT) has been prepared for batch discharge.

The following radiation monitors and flow elements are out of service and Non-Functional

- Miscellaneous RE 1878A
- Miscellaneous RE 1878B
- Clean RE 1770B
- FE 4687 Storm Sewer Flow

All other instrumentation is OPERABLE.

Based on these conditions, what Offsite Dose Calculation Manual (ODCM) actions will be required in order to discharge this tank?

REFERENCE PROVIDED

- A. The system/process flow rate is estimated at least once per 4 hours during the actual release.
- B. At least two independent samples of the tank's content are analyzed and at least two independent verifications of the release rate calculations and discharge valve lineups are performed AND the system/process flow rate is estimated at least once per 4 hours during the actual release.
- C. Grab samples are collected, at least once per 12 hours, and analyzed, at least once per 12 hours, for gross radioactivity (beta or gamma) at a lower limit of detection no greater than 1.0^{-07} $\mu\text{Ci/ml}$ or a gamma isotopic analysis meeting the LLD Requirement of Table 2-3.
- D. At least two independent samples of the tank's content are analyzed and at least two independent verifications of the release rate calculations and discharge valve lineups are performed.

Davis Besse 1LOT21 NRC Written Exam Rev. 1

24. Plant Conditions:

- The Plant is in a refueling outage.
- Refueling Canal water level is lowering unexpectedly.

Sequence of Events:

- Time= 1110: An unplanned rise in area radiation levels is indicated by RE8426 SFP Area.
- Time= 1115: The Shift manager declares the event as an UNUSUAL EVENT.
- Time= 1119: Prior to state and local notifications being made for the UNUSUAL EVENT, conditions change such that the Shift Manager upgrades the classification to an ALERT.
- The Shift Manager informs the Communicator to make notifications of the ALERT, ONLY.

Given the above information, which one of the following identifies:

- (1) The latest time that the State and County notifications must be made, and
- (2) What action is required to stop the clock?

- A. (1) 1130
(2) State and County notifications initiated
- B. (1) 1130
(2) FENOC Nuclear Power Plant Initial Notification Form (DBEP-010) telefaxed
- C. (1) 1134
(2) State and County notifications initiated
- D. (1) 1134
(2) FENOC Nuclear Power Plant Initial Notification Form (DBEP-010) telefaxed

Davis Besse 1LOT21 NRC Written Exam Rev. 1

25. A Reactor Trip due to a Large Break LOCA occurred coincident with some fuel damage.
- The Shift Manager/Emergency Director declared a General Emergency on FG1 10 minutes after the Reactor Trip
 - The initial notification has NOT been made at this time
 - Containment Radiation monitors RE4596A and B indicate $1.20E+4$ R/hr
 - An unisolable gaseous release is in progress, from a failed containment penetration
 - The expected duration of the leakage is ~ 40 minutes
 - Wind direction is from 18°
 - Dose projections at 5 miles are 0.5 Rem TEDE and 1.5 Rem Child Thyroid CDE
 - There is no Hostile Action or Known Impediment to Evacuation


Based on these conditions, what Protective Action Recommendation (PAR) is required?

REFERENCE PROVIDED

- A. Shelter is N/A
Evacuate 2 mile radius & 5 mile downwind subarea 1, 2, 6, 12
- B. Shelter is N/A
Evacuate 2 mile radius & 5 mile downwind subareas 1, 2, 12
- C. Shelter is N/A
Evacuate 2 mile radius & 10 mile downwind subareas 1, 2, 4, 5, &12
- D. Shelter 2 mile radius & 5 mile downwind subareas 1, 2
Evacuate 2 mile radius & 5 mile downwind subarea 12

DAVIS-BESSE REFERENCE MATERIAL		Number: DBRM-EMER-1500B	
Title:	Davis-Besse EAL Wallboards	Revision: 02	Page 1 of 3

Davis-Besse EAL Wallboards

APPROVED BY: 
Emergency Response Manager

6-22-17
Date

Effective date: 6/29/17

DAVIS-BESSE REFERENCE MATERIAL		Number: DBRM-EMER-1500B	
Title:	Davis-Besse EAL Wallboards	Revision: 02	Page 2 of 3

The purpose of this reference material is to provide the Davis-Besse Emergency Action Level (EAL) wallboards. EAL wallboards are a display of the emergency action levels as described in RA-EP-01500, Emergency Classification procedure. The intent of the wallboards is to provide an aid to be used by emergency response personnel who are evaluating plant conditions to determine if the event is classifiable.

1. The wallboards are comprised of two separate boards:
 - a. Hot Mode EALs - which provides as a single display those EALs that are applicable in operation Modes 1, 2, 3, and 4
 - b. Cold Mode EALs – which provides as a single display those EALs that are applicable in Modes 5 and 6
2. EALs that are associated with Hazards and Other Conditions Affecting Plant Safety, Dry Fuel Storage Facility (DFSF), and Abnormal Rad Levels / Rad Effluent are common to both the Hot Mode EALs and the Cold Mode EALs wallboards.
3. EAL wallboards are primarily found in the Control Room, Control Room Simulator, Technical Support Center and the Emergency Operations Facility.

TO OPEN HOT EAL OR COLD EAL PRESS CTRL AND CLICK ON DESIRED EAL.

(HOT)

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
1 Normal				
2 Normal				
3 Normal				
4 Normal				
5 Normal				
6 Normal				
7 Normal				
1 Normal				
2 Normal				
3 Normal				
4 Normal				
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3.3 INSTRUMENTATION

3.3.9 Source Range Neutron Flux

LCO 3.3.9 Two source range neutron flux channels shall be OPERABLE.

-----NOTE-----
High voltage to detector may be de-energized with neutron flux
> 1E-10 amp on intermediate range channels.

APPLICABILITY: MODES 2, 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One source range neutron flux channel inoperable with neutron flux $\leq 1E-10$ amp on the intermediate range neutron flux channels.	A.1 Restore channel to OPERABLE status.	Prior to increasing neutron flux
B. Two source range neutron flux channels inoperable with neutron flux $\leq 1E-10$ amp on the intermediate range neutron flux channels.	B.1 -----NOTE----- Plant temperature changes are allowed provided the temperature change is accounted for in the calculated SDM. ----- Suspend operations involving positive reactivity changes.	Immediately
	<u>AND</u> B.2 Initiate action to insert all CONTROL RODS. <u>AND</u>	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.3 Open CONTROL ROD drive trip breakers.	1 hour
	<u>AND</u> B.4 Verify SDM is within the limits specified in the COLR.	1 hour <u>AND</u> Once per 12 hours thereafter
C. One or more source range neutron flux channels inoperable with neutron flux > 1E-10 amp on the intermediate range neutron flux channels.	C.1 Initiate action to restore affected channel(s) to OPERABLE status.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.9.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.9.2	-----NOTE----- Neutron detectors are excluded from CHANNEL CALIBRATION. ----- Perform CHANNEL CALIBRATION.	18 months

3.3 INSTRUMENTATION

3.3.10 Intermediate Range Neutron Flux

LCO 3.3.10 Two intermediate range neutron flux channels shall be OPERABLE.

APPLICABILITY: MODE 2,
MODES 3, 4, and 5 with any CONTROL ROD drive (CRD) trip breaker in
the closed position and the CRD System capable of rod withdrawal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 Reduce neutron flux to $\leq 1E-10$ amp.	2 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> Two channels inoperable.	B.1 -----NOTE----- Plant temperature changes are allowed provided the temperature change is accounted for in the calculated SDM. ----- Suspend operations involving positive reactivity changes. <u>AND</u> B.2 Open CRD trip breakers.	Immediately 1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.10.1 Perform CHANNEL CHECK.	12 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.3.10.2 -----NOTE----- Neutron detectors are excluded from CHANNEL CALIBRATION. ----- Perform CHANNEL CALIBRATION.	18 months

3.3 INSTRUMENTATION

3.3.17 Post Accident Monitoring (PAM) Instrumentation

LCO 3.3.17 The PAM instrumentation for each Function in Table 3.3.17-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one required channel inoperable.	A.1 Restore required channel to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 -----NOTE----- Only applicable to Functions other than Functions 13, 14, and 15. ----- Initiate action in accordance with Specification 5.6.5.	Immediately
	<u>OR</u> B.2 -----NOTE----- Only applicable to Functions 13, 14, and 15. ----- Enter the Condition referenced in Table 3.3.17-1 for the channel.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more Functions with two required channels inoperable.	C.1 Restore one channel to OPERABLE status.	7 days
D. Required Action and associated Completion Time of Condition C not met.	D.1 Enter the Condition referenced in Table 3.3.17-1 for the channel.	Immediately
E. As required by Required Action B.2 or D.1 and referenced in Table 3.3.17-1.	E.1 Be in MODE 3.	6 hours
	<u>AND</u> E.2 Be in MODE 4.	12 hours
F. As required by Required Action D.1 and referenced in Table 3.3.17-1	F.1 Initiate action in accordance with Specification 5.6.5.	Immediately

SURVEILLANCE REQUIREMENTS

-----NOTE-----

These SRs apply to each PAM instrumentation Function in Table 3.3.17-1 except where identified in the SR.

SURVEILLANCE	FREQUENCY
SR 3.3.17.1 Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	31 days

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.3.17.2 -----NOTE----- Neutron detectors are excluded from CHANNEL CALIBRATION. ----- Perform CHANNEL CALIBRATION for Functions 1, 11, 12, 14, 15, 16, and 17.	18 months
SR 3.3.17.3 Perform CHANNEL CALIBRATION for Functions 2, 3, 4, 5, 6, 7, 8, 9, 10, and 13.	24 months

Table 3.3.17-1 (page 1 of 1)
Post Accident Monitoring Instrumentation

FUNCTION	REQUIRED CHANNELS	CONDITIONS REFERENCED FROM REQUIRED ACTION B.2 or D.1
1. Wide Range Neutron Flux	2	E
2. Reactor Coolant Loop Outlet Temperature	2 per loop	E
3. Reactor Coolant Loop Pressure	2 per loop	E
4. Reactor Coolant Hot Leg Level (Wide Range)	2	E
5. Containment Water Level (Wide Range)	2	E
6. Containment Pressure (Wide Range)	2	E
7. Penetration Flow Path Containment Isolation Valve Position	2 per penetration flow path ^{(a)(b)}	E
8. Containment High Range Radiation	2	F
9. Pressurizer Level	2	E
10. Steam Generator Startup Range Level	2 per SG	E
11. Incore Thermocouples	2 per core quadrant	E
12. Auxiliary Feedwater Flow Rate	2 per SG	E
13. Steam Generator Outlet Steam Pressure	1 per SG	E
14. High Pressure Injection Flow	1 per injection line	E
15. Low Pressure Injection (Decay Heat Removal) Flow	1 per train	E
16. Borated Water Storage Tank Level	2	E
17. Neutron Flux (Source Range)	2	E

(a) Not required for isolation valves whose associated penetration is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.

(b) Only one position indication channel is required for penetration flow paths with only one installed control room indication channel.

8.3 INSTRUMENTATION

8.3.7 Post Accident Monitoring (PAM) Instrumentation

TECHNICAL NORMAL CONDITIONS

TNC 8.3.7 The PAM instrumentation for each Function in Table 8.3.7-1 shall be FUNCTIONAL.

APPLICABILITY: MODES 1, 2 and 3.

CONTINGENCY MEASURES

NONCONFORMANCE	CONTINGENCY MEASURES	RESTORATION TIME
A. One or more Functions with one required channel Nonfunctional.	A.1 Restore required channel to FUNCTIONAL status.	30 days
B. Contingency Measure and associated Restoration Time of Nonconformance A not met.	B.1 Initiate action to evaluate failure to meet TNC per TRM Section 7.3.	Immediately

TECHNICAL VERIFICATION REQUIREMENTS

-----NOTE-----
The below TVRs apply to each PAM instrumentation Function in Table 8.3.7-1.

TVR	VERIFICATION	FREQUENCY
8.3.7.1	Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	31 days
8.3.7.2	Perform CHANNEL CALIBRATION.	24 months

Table 8.3.7-1 (page 1 of 1)
Post Accident Monitoring Instrumentation

FUNCTION	REQUIRED CHANNELS
1. RC System Subcooling Margin Monitor	1
2. PORV Position Indicator	1
3. PORV Block Valve Position Indicator	1
4. Pressurizer Safety Valve Position Indicator	1/valve
5. Containment Normal Sump Level	1

BASES

8.3.7 Post Accident Monitoring (PAM) Instrumentation

The primary purpose of the PAM instrumentation is to display unit variables that provide information required by the control room operators during accident situations. This information provides the necessary support for the operator to take the manual actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for Design Basis Events.


The FUNCTIONALITY of the accident monitoring instrumentation ensures that there is sufficient information available on selected unit parameters to monitor and to assess unit status and behavior following an accident.

The availability of accident monitoring instrumentation is important so that responses to corrective actions can be observed, and so that the need for and magnitude of further actions can be determined. These essential instruments are identified by UFSAR Section 7.13 (Ref. 1) addressing the recommendations of Regulatory Guide 1.97 (Ref. 2) as required by Supplement 1 to NUREG-0737 (Ref. 3).

Only those instruments monitoring Type A and Category 1 variables are required to be included in Technical Specifications. The instruments in this Technical Requirement did not meet the criterion for inclusion into Technical Specifications.

DAVIS-BESSE REFERENCE MATERIAL		Number: DBRM-EMER-1500B	
Title:	Davis-Besse EAL Wallboards	Revision: 02	Page 1 of 3

Davis-Besse EAL Wallboards

APPROVED BY: 
Emergency Response Manager

6-22-17
Date

Effective date: 6/29/17

DAVIS-BESSE REFERENCE MATERIAL		Number: DBRM-EMER-1500B	
Title:	Davis-Besse EAL Wallboards	Revision: 02	Page 2 of 3

The purpose of this reference material is to provide the Davis-Besse Emergency Action Level (EAL) wallboards. EAL wallboards are a display of the emergency action levels as described in RA-EP-01500, Emergency Classification procedure. The intent of the wallboards is to provide an aid to be used by emergency response personnel who are evaluating plant conditions to determine if the event is classifiable.

1. The wallboards are comprised of two separate boards:
 - a. Hot Mode EALs - which provides as a single display those EALs that are applicable in operation Modes 1, 2, 3, and 4
 - b. Cold Mode EALs – which provides as a single display those EALs that are applicable in Modes 5 and 6
2. EALs that are associated with Hazards and Other Conditions Affecting Plant Safety, Dry Fuel Storage Facility (DFSF), and Abnormal Rad Levels / Rad Effluent are common to both the Hot Mode EALs and the Cold Mode EALs wallboards.
3. EAL wallboards are primarily found in the Control Room, Control Room Simulator, Technical Support Center and the Emergency Operations Facility.

TO OPEN HOT EAL OR COLD EAL PRESS CTRL AND CLICK ON DESIRED EAL.

(HOT)

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
H 1 2 3 4 5 6 7	1			
	2			
	3			
	4			
	5			
	6			
	7			
F 1 2 3	1			
	2			
	3			
E 1				

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT																																																
H 1 2 3 4 5 6 7 8 9	1																																																			
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Table F-1 Fission Product Barrier Matrix																																																				
<table border="1"> <thead> <tr> <th rowspan="2">Category</th> <th colspan="2">Fuel Clad (FC) Barrier</th> <th colspan="2">Reactor Coolant System (RCS) Barrier</th> <th colspan="2">Containment (CT) Barrier</th> </tr> <tr> <th>Loss</th> <th>Penetration Loss</th> <th>Loss</th> <th>Penetration Loss</th> <th>Loss</th> <th>Penetration Loss</th> </tr> </thead> <tbody> <tr> <td>A</td> <td>None</td> <td>None</td> <td>None</td> <td>None</td> <td>None</td> <td>None</td> </tr> <tr> <td>B</td> <td>None</td> <td>None</td> <td>None</td> <td>None</td> <td>None</td> <td>None</td> </tr> <tr> <td>C</td> <td>None</td> <td>None</td> <td>None</td> <td>None</td> <td>None</td> <td>None</td> </tr> <tr> <td>D</td> <td>None</td> <td>None</td> <td>None</td> <td>None</td> <td>None</td> <td>None</td> </tr> <tr> <td>E</td> <td>None</td> <td>None</td> <td>None</td> <td>None</td> <td>None</td> <td>None</td> </tr> </tbody> </table>					Category	Fuel Clad (FC) Barrier		Reactor Coolant System (RCS) Barrier		Containment (CT) Barrier		Loss	Penetration Loss	Loss	Penetration Loss	Loss	Penetration Loss	A	None	None	None	None	None	None	B	None	None	None	None	None	None	C	None	None	None	None	None	None	D	None	None	None	None	None	None	E	None	None	None	None	None	None
Category	Fuel Clad (FC) Barrier		Reactor Coolant System (RCS) Barrier			Containment (CT) Barrier																																														
	Loss	Penetration Loss	Loss	Penetration Loss	Loss	Penetration Loss																																														
A	None	None	None	None	None	None																																														
B	None	None	None	None	None	None																																														
C	None	None	None	None	None	None																																														
D	None	None	None	None	None	None																																														
E	None	None	None	None	None	None																																														
Modes: 1 2 3 4 5 6 DEF																																																				

(COLD)

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
H 1 2 3 4 5 6	1			
	2			
	3			
	4			
	5			
	6			
F 1 2 3	1			
	2			
	3			
E 1				

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
H 1 2 3 4 5 6	1			
	2			
	3			
	4			
	5			
	6			
DEFINITIONS				
<p>The following definitions apply to the Emergency Action Levels (EALs) and are intended to provide a common understanding of the terms used in the EAL Wallboard. The definitions are provided for informational purposes only and are not intended to be used as a substitute for the EAL Wallboard or other documents.</p> <p>ALERT: A condition that requires the operator to take immediate action to prevent a potential accident. The operator should be prepared to execute the Emergency Action Level (EAL) procedures if the condition persists.</p> <p>UNUSUAL EVENT: A condition that requires the operator to take immediate action to prevent a potential accident. The operator should be prepared to execute the Emergency Action Level (EAL) procedures if the condition persists.</p> <p>GENERAL EMERGENCY: A condition that requires the operator to take immediate action to prevent a potential accident. The operator should be prepared to execute the Emergency Action Level (EAL) procedures if the condition persists.</p> <p>SITE AREA EMERGENCY: A condition that requires the operator to take immediate action to prevent a potential accident. The operator should be prepared to execute the Emergency Action Level (EAL) procedures if the condition persists.</p>				
Modes: 1 2 3 4 5 6 DEF				

Table 2-1
RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

<u>INSTRUMENT</u>	<u>REQUIRED CHANNELS</u>	<u>APPLICABILITY</u>	<u>ACTION</u>
1. Gross Radioactivity Monitors Providing Alarms and Automatic Termination of Release			
a. Liquid Radwaste Effluent Line (either Miscellaneous (RE 1878A, B) or Clean (RE 1770A, B), but not both simultaneously)*	1	(1)	A
2. Flow Rate Measurement Devices			
a. Liquid Radwaste Effluent Line	1	(1)	B
b. Dilution Flow to Collection Box	1	(1)	B
c. FE 4687 Storm Sewer	1	(1)	B
3. Gross Beta or Gamma Radioactivity Monitors Providing Alarm But Not Providing Automatic Termination of Release			
a. Storm Sewer Drain (RE 4686)	1	(1)	C

* Only one release (either MWM or CWMT) at a time can be in progress.

Table 2-1 (continued)

TABLE NOTATION

(1) During radioactive releases via this pathway

ACTION A With less than the number of required channels FUNCTIONAL, effluent releases may be resumed, provided that prior to initiating a release:

1. At least two independent samples are analyzed in accordance with Table 2-3 for analyses performed with each batch;
2. At least two independent verification of the release rate calculations are performed;
3. At least two independent verifications of the discharge valving are performed;

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION B With less than the number of required channels FUNCTIONAL, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases. Pump curves may be used to estimate flow.

ACTION C With less than the number of required channels FUNCTIONAL, or if high alarm is locked in on RE, effluent releases via this pathway may continue provided that during effluent releases, grab samples are collected, at least once per 12 hours, and analyzed, at least once per 12 hours, for gross radioactivity (beta or gamma) at a lower limit of detection no greater than 1.0E-07 $\mu\text{Ci/ml}$ or a gamma isotopic analysis meeting the LLD Requirement of Table 2-3.

Table 2-2
RADIOACTIVE LIQUID EFFLUENT MONITORING
INSTRUMENTATION VERIFICATION REQUIREMENTS

<u>INSTRUMENT</u>	<u>CHANNEL CHECK</u>	<u>SOURCE CHECK</u>	<u>CHANNEL CALIBRATION</u>	<u>CHANNEL FUNCTIONAL TEST</u>
1. Gross Beta or Gamma Radioactivity Monitors Providing Alarm and Automatic Isolation, if applicable.				
a. Liquid Radwaste Effluents Lines	D ⁽¹⁾	P	E ⁽³⁾	Q ⁽²⁾
b. Storm Sewer Discharge Line	D ⁽⁴⁾	M	E ⁽³⁾	Q ⁽²⁾
2. Flow Rate Monitors				
a. Liquid Radwaste Effluent Lines	D ⁽⁴⁾	N/A	E	Q
b. Dilution Flow to Collection Box	D ⁽⁴⁾	N/A	E	Q
c. Storm Sewer		N/A		

Table 2-2 (continued)

TABLE NOTATION

- (1) During releases via this pathway.
- (2) If applicable, the CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if the instrument indicates measured levels above the alarm/trip setpoint.
- (3) The initial CHANNEL CALIBRATION for radioactivity measurement instrumentation shall be performed using one or more of the reference standards certified by the National Institute of Standards and Technology (NIST) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NIST. These standards should permit calibrating the system over its intended range of energy and rate capabilities. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration should be used, at intervals of at least once per eighteen months. For high range monitoring instrumentation, where calibration with a radioactive source is impractical, an electronic calibration may be substituted for the radiation source calibration.
- (4) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once daily on any day on which continuous, periodic, or BATCH RELEASES are made.

- (D) At least once per 24 hours.
- (M) At least once per 31 days.
- (P) Prior to each release.
- (E) At least once per 18 month (550 days).
- (Q) At least once per 92 days.
- (R) At least once per 24 months (730 days)

DAVIS-BESSE REFERENCE MATERIAL		Number: DBRM-EMER-1500B	
Title:	Davis-Besse EAL Wallboards	Revision: 02	Page 1 of 3

Davis-Besse EAL Wallboards

APPROVED BY: 
Emergency Response Manager

6-22-17
Date

Effective date: 6/29/17

DAVIS-BESSE REFERENCE MATERIAL		Number: DBRM-EMER-1500B	
Title:	Davis-Besse EAL Wallboards	Revision: 02	Page 2 of 3

The purpose of this reference material is to provide the Davis-Besse Emergency Action Level (EAL) wallboards. EAL wallboards are a display of the emergency action levels as described in RA-EP-01500, Emergency Classification procedure. The intent of the wallboards is to provide an aid to be used by emergency response personnel who are evaluating plant conditions to determine if the event is classifiable.

1. The wallboards are comprised of two separate boards:
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 - b. Cold Mode EALs – which provides as a single display those EALs that are applicable in Modes 5 and 6
2. EALs that are associated with Hazards and Other Conditions Affecting Plant Safety, Dry Fuel Storage Facility (DFSF), and Abnormal Rad Levels / Rad Effluent are common to both the Hot Mode EALs and the Cold Mode EALs wallboards.
3. EAL wallboards are primarily found in the Control Room, Control Room Simulator, Technical Support Center and the Emergency Operations Facility.

TO OPEN HOT EAL OR COLD EAL PRESS CTRL AND CLICK ON DESIRED EAL.

(HOT)

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
H Normal	1 Normal			
	2 Normal			
	3 Normal			
	4 Normal			
	5 Normal			
	6 Normal			
	7 Normal			
F Fuel	1 Normal			
	2 Normal			
	3 Normal			
E Emergency	1 Normal			

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
H Normal	1 Normal			
	2 Normal			
	3 Normal			
	4 Normal			
	5 Normal			
	6 Normal			
	7 Normal			
	8 Normal			
	9 Normal			
F Fuel	1 Normal			
	2 Normal			
	3 Normal			
	4 Normal			
	5 Normal			
E Emergency	1 Normal			

Category	Fuel Clad (FC) Barrier		Reactor Coolant System (RCS) Barrier		Containment (CT) Barrier	
	Loss	Penetration Loss	Loss	Penetration Loss	Loss	Penetration Loss
A	None	None	None	None	None	None
B	None	None	None	None	None	None
C	None	None	None	None	None	None
D	None	None	None	None	None	None
E	None	None	None	None	None	None

Modes: 1 2 3 4 5 6 DEF
DAVIS-BESSE, Unit 1
 COLD CONDITIONS
 RCS - 2007

(COLD)

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
H Normal	1 Normal			
	2 Normal			
	3 Normal			
	4 Normal			
	5 Normal			
	6 Normal			
C Control	1 Normal			
	2 Normal			
	3 Normal			
R Reactor	1 Normal			
	2 Normal			
	3 Normal			
E Emergency	1 Normal			

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
H Normal	1 Normal			
	2 Normal			
	3 Normal			
	4 Normal			
	5 Normal			
	6 Normal			
C Control	1 Normal			
	2 Normal			
	3 Normal			
R Reactor	1 Normal			
	2 Normal			
	3 Normal			
E Emergency	1 Normal			

Modes: 1 2 3 4 5 6 DEF
DAVIS-BESSE, Unit 1
 COLD CONDITIONS
 RCS - 2007

Davis-Besse Nuclear Power Station

EMERGENCY PLAN IMPLEMENTING PROCEDURE

RA-EP-02245

Protective Action Guidelines

REVISION 09

Prepared by: G. S. Van Wey

Procedure Owner: Emergency Response Manager

Effective Date: 11/14/19

**LEVEL OF USE:
IN-FIELD REFERENCE**

TABLE OF CONTENTS

	<u>Page</u>
1.0 PURPOSE	3
2.0 REFERENCES.....	3
3.0 DEFINITIONS	4
4.0 RESPONSIBILITIES.....	6
5.0 INITIATING CONDITIONS	6
6.0 PROCEDURE	7
6.1 Onsite Protective Actions.....	7
6.2 Offsite Protective Actions	8
7.0 FINAL CONDITIONS.....	9
8.0 RECORDS	9
ATTACHMENT 1 Flowchart for Offsite Protective Action Recommendation Determination	11
ATTACHMENT 2 Comparison of Offsite Sectors and Subareas	13
COMMITMENTS.....	14

1.0 PURPOSE

- 1.1 This procedure defines specific guidelines for determining protective action recommendations for emergencies involving abnormal releases of radioactivity at the Davis-Besse Nuclear Power Station (DBNPS).

2.0 REFERENCES

2.1 Developmental

- 2.1.1 Davis-Besse Nuclear Power Station Emergency Plan.
- 2.1.2 NUREG-0654 FEMA REP-1, Rev. 1, Supplement 3, Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants, November 2011
- 2.1.3 NEI 12-10 (Rev. 1), Guideline for Developing a Licensee Protective Action Recommendation Procedure Using NUREG-0654, Supplement 3, August 2014
- 2.1.4 EPA-400-R-92-001, May 1992, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents
- 2.1.5 KLD Associates, Inc., “Davis-Besse Nuclear Power Station Development of Evacuation Time Estimates”, KLD TR-482, Rev. 2, October 2012
- 2.1.6 U.S. Food and Drug Administration, Federal Register, Vol. 47, No. 205, Oct. 22, 1982
- 2.1.7 Anno, George, Dore, and Michael: The Effectiveness of Sheltering as a Protective Action Against Nuclear Accidents Involving Gaseous Releases, EPA 520/I-78- 001A, April 1978.
- 2.1.8 SAND 77-1725, Public Protection Strategies for Potential Nuclear Reactor Accidents – Sheltering Concepts with Existing Public and Private Structures.
- 2.1.9 NRC IE Information Notice No. 83-28: Criteria for Protective Action Recommendations for General Emergencies, dated May 4, 1983.
- 2.1.10 Regulatory Information Summary (RIS) 2002-16, Current Incident Response Issues
- 2.1.11 Regulatory Information Summary (RIS) 2003-12, NRC Regulatory Issue Summary 2003-12: Clarification of NRC Guidance for Modifying Protective Actions
- 2.1.12 Regulatory Information Summary (RIS) 2004-13, Consideration of Sheltering in Licensee's Range of Protective Action Recommendations
- 2.1.13 NEI Position Paper, Range of Protective Actions for Nuclear Power Plant Incidents, July, 2004.

2.1.14 NRC Regulatory Issue Summary 2005-08, Endorsement of Nuclear Energy Institute (NEI) Guidance “Range of Protective Actions for Nuclear Power Plant Incidents”, June 6, 2005

2.2 Implementation

2.2.1 RA-EP-02110, Emergency Notification

2.2.2 NOP-LP-5022, Davis-Besse Unified RASCAL Interface (URI) Dose Assessment Software.

2.2.3 RA-EP-02520, Assembly and Accountability

2.2.4 RA-EP-02530, Evacuation

2.2.5 RA-EP-02620, Emergency Dose Control and Potassium Iodide Distribution

2.2.6 RA-EP-02720, Recovery Organization

3.0 DEFINITIONS

- 3.1 ALARA – As Low As Reasonably Achievable, means making every reasonable effort to maintain exposures to radiation as far below the dose limits in 10CFR20 as is practical and consistent with the purpose for which the licensed activity is undertaken.
- 3.2 COMMITTED DOSE EQUIVALENT (CDE) – The dose equivalent to organs or tissues that will be received from an intake of radioactive material by an individual during the 50-year period following the intake.
- 3.3 EMERGENCY PLANNING ZONE (EPZ) – The two zones that are established around a nuclear power station in which predetermined protective actions plans are needed.
- 3.3.1 The first zone has an approximate radius of 10 miles for the plume exposure pathway.
- 3.3.2 The second zone has an approximate radius of 50 miles for the ingestion exposure pathway.
- 3.4 EVACUATION DOSE – The dose that a potential evacuee would receive if he or she were openly exposed during the evacuation.
- 3.5 EVACUATION EXPOSURE PERIOD – The period during which those people being evacuated are exposed to the radioactive plume.
- 3.6 EXPOSURE TIME – That period of time during which the offsite population will be exposed to radiation as a result of an airborne radioactive release.
- 3.7 HOSTILE ACTION – An act toward DBNPS 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 DBNPS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the OWNER CONTROLLED AREA.

- 3.8 KI FOR THE GENERAL PUBLIC - Recommending potassium iodide (KI) for the general public is the responsibility of the State of Ohio Department of Health. The station will recommend to the State administrating KI in accordance with State procedures upon declaration of a General Emergency.
- 3.9 KNOWN IMPEDIMENTS TO EVACUATION – Those conditions that may affect the ability to evacuate a large portion of the population within the Emergency Planning Zone (such as inclement weather, roadway/bridge damage following a natural event). State or counties must have provided prior knowledge of these impediments to the utility prior to the Protective Action decision-making.
- 3.10 LAKE BREEZE – A meteorological condition that may occur on clear, sunny days. During a lake breeze, a radioactive release can travel inland, rise, reverse course in an overhead return flow, and then return to land in a convoluted path.
- 3.11 MINIMUM RADIOACTIVE RELEASE PROTECTIVE ACTION RECOMMENDATIONS (PAR) – At a minimum a PAR will be issued for Subarea 1, Subarea 12, and affected downwind subareas within five miles AND advise the general public to take KI in accordance with the Ohio Plan for Response to Radiation Emergencies at Licensed Nuclear Facilities.
- 3.12 MONITOR AND PREPARE – Guidance provided to the population within the Emergency Planning Zone, typically for those Sheltering in Place. This advices the public to maintain an awareness of the event through media broadcasts and subsequent Emergency Alert System messages. The public should make preparations for evacuation (or other protective actions) as directed.
- 3.13 OFFSITE – Any area outside the Owner Controlled Area surrounding Davis-Besse Nuclear Power Station.
- 3.14 RAPIDLY ESCALATING SEVERE ACCIDENT – This describes an accident in which there has been significant degradation of the reactor core, and because of the accident, the integrity of Containment is challenged. It is postulated that this accident will result in a significant release of radioactive material and subsequent dose to the public.
- 3.15 RELEASE - A release is defined as a radiological release attributable to the emergency event. Two levels of radiological release exist:
 - a. A minor unplanned release below levels that require offsite Protective Action Recommendations.
 - b. A release that requires offsite Protective Action Recommendations.
- 3.16 SAFETY PARAMETER DISPLAY SYSTEM (SPDS) – The SPDS is a group of graphic displays developed to assist with monitoring plant operations.

- 3.17 SECTOR – One of the 16 areas bounded by radii 22½ degrees apart into which the 10-mile EPZ is divided. Sectors are designated by the Letters A through P, excluding I and O. Sector A is north, E is East, J is south, and N is west.
- 3.18 SHELTERING – The use of a structure for radiation protection from an airborne plume and from deposited radioactive material. A wood frame home without a basement is the assumed structure for sheltering in the Davis-Besse EPZ.
- 3.19 SHORT-TERM (PUFF) RELEASE – A controlled venting of radioactive gases (usually from Containment), sometimes referred to as a “puff” release. To be considered a short-term release, it must meet all of the following criteria:
- Initiated by the operator, not the accident.
 - Performed in a controlled manner and able to be secured, if necessary
 - Monitored by effluent radiation monitors (e.g., Station Vent radiation monitors)
 - Short-term (≤ 1 hour)
 - Doses are less than EPA guidance
- 3.20 TOTAL EFFECTIVE DOSE EQUIVALENT (TEDE) – The sum of the deep-dose equivalent (for external exposure) and the Committed Effective Dose Equivalent (for internal exposure).

4.0 RESPONSIBILITIES

- 4.1 The Emergency Director is responsible for directing protective actions for Station personnel and recommending protective actions to offsite officials for the Plume Exposure Pathway (10-mile EPZ).
- 4.2 The Dose Assessment Coordinator is responsible for collecting and analyzing offsite dose assessment data used to provide the basis for protective action recommendations.

5.0 INITIATING CONDITIONS

Initiate this procedure when a declared emergency has the potential for an abnormal release of radioactivity.

6.0 PROCEDURE

Steps in this procedure may be performed simultaneously.

6.1 Onsite Protective Actions

- 6.1.1 The Emergency Director shall initiate the necessary actions to protect DBNPS personnel.
 - a. Evacuate personnel in accordance with RA-EP-02530, Evacuation.
 - b. Account for personnel in accordance with RA-EP-02520, Assembly and Accountability.
 - c. Distribute potassium iodide in accordance with RA-EP-02620, Emergency Dose Control and Potassium Iodide Distribution.
 - d. If Onsite Protective Actions are required due to a HOSTILE ACTION, refer to DB-OP-02544, Security Events Or Threats.

- 6.1.2 All supervisors shall ensure that appropriate safety and ALARA precautions are implemented.

NOTE 6.2

- ANY CONDITION THAT JUSTIFIES ISSUING AN OFFSITE PROTECTIVE ACTION REQUIRES A GENERAL EMERGENCY DECLARATION.
- Offsite Protection Action Recommendations shall be made with initial notification of a General Emergency.
- Davis-Besse will always recommend EVACUATION of Subarea 12 (Lake Erie) and when appropriate Subarea 10 (Wildlife area) due to lack of shelters in these areas.
- A SHELTERING PAR will **NOT** be issued for any subarea in which an EVACUATION PAR has already been recommended.

6.2 Offsite Protective Actions

6.2.1 Refer to Attachment 1 to determine recommended Protective Action Recommendation.

CAUTION 6.2.2

Protective Action Recommendations once issued start in motion a sequence of events in the 10-mile emergency planning zone that, if modified, have the potential to cause confusion that may hamper the orderly implementation of protective actions for the general public.

6.2.2 Notify offsite agencies and the NRC of the PARs and the affected subareas using RA-EP-02110, Emergency Notification, and Initial Notification Form, DBEP-010.

- a. **IF** these are revised PARs
THEN DO NOT downgrade a previously issued PAR for a specific subarea until the conditions that caused the PAR to be issued are fully under control. The new PAR should include those subareas that were previously evacuated or sheltered and any new subareas.
- b. **IF** a lake breeze is occurring, the wind direction is unknown, or the wind direction is from between 162° and 277°,
THEN inform the NRC that the release may enter Canadian territory.
- c. **IF** TEDE doses are $\geq 1,000$ mrem or thyroid dose $\geq 5,000$ mrem are projected beyond 10 miles,
THEN coordinate with state and county officials to determine appropriate PAR.
- d. **IF** an Initial PAR was to Shelter a Subarea(s), and consultation with the State and County Emergency Management Agencies determine that it is safe to do so,
THEN change the PAR from Shelter to Evacuate.

6.2.3 As Radiation Monitoring Team (RMT) data becomes available, compare it to dose projections and verify that Protective Action Recommendations are adequate.

- 6.3 **IF** the General Emergency condition still exists,
THEN continue to monitor radiological and meteorological conditions,
AND repeat Steps 6.1 and 6.2 to ensure current PARs are adequate.

7.0 FINAL CONDITIONS

Terminate this procedure when the Emergency Director, and offsite agencies determine that dose assessment and protective actions are no longer necessary.

NOTE: 8.0

All logs, documents and forms generated during the implementation of this procedure are to be retained and processed in accordance with RA-EP-02720, Recovery Organization.

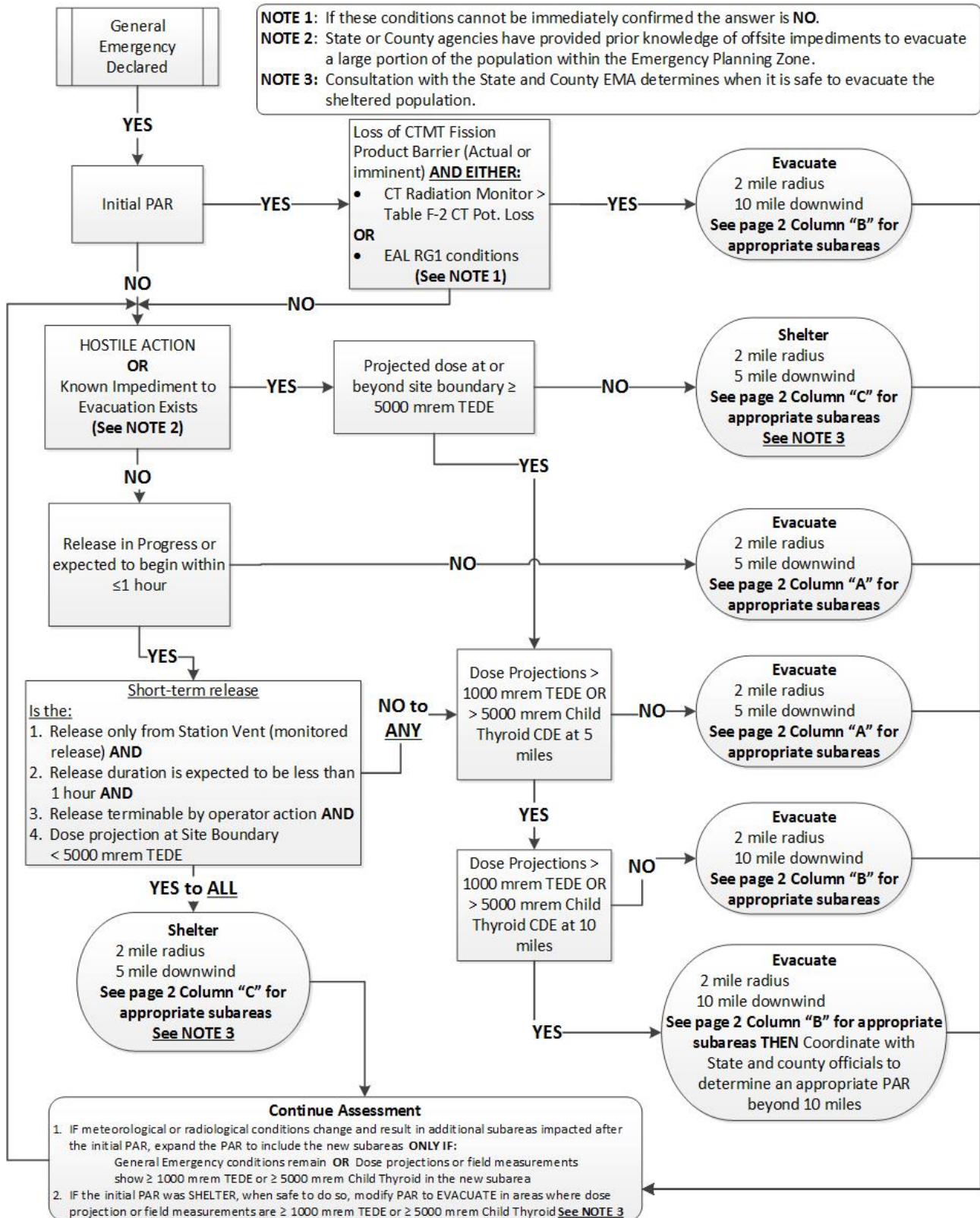
8.0 RECORDS

- 8.1 The following records that are completed by this procedure shall be processed and retained as part of the Event Package in accordance with RA-EP-02720, Recovery Organization.

8.1.1 None

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**ATTACHMENT 1: FLOWCHART FOR
OFFSITE PROTECTIVE ACTION RECOMMENDATION DETERMINATION**
Page 1 of 2



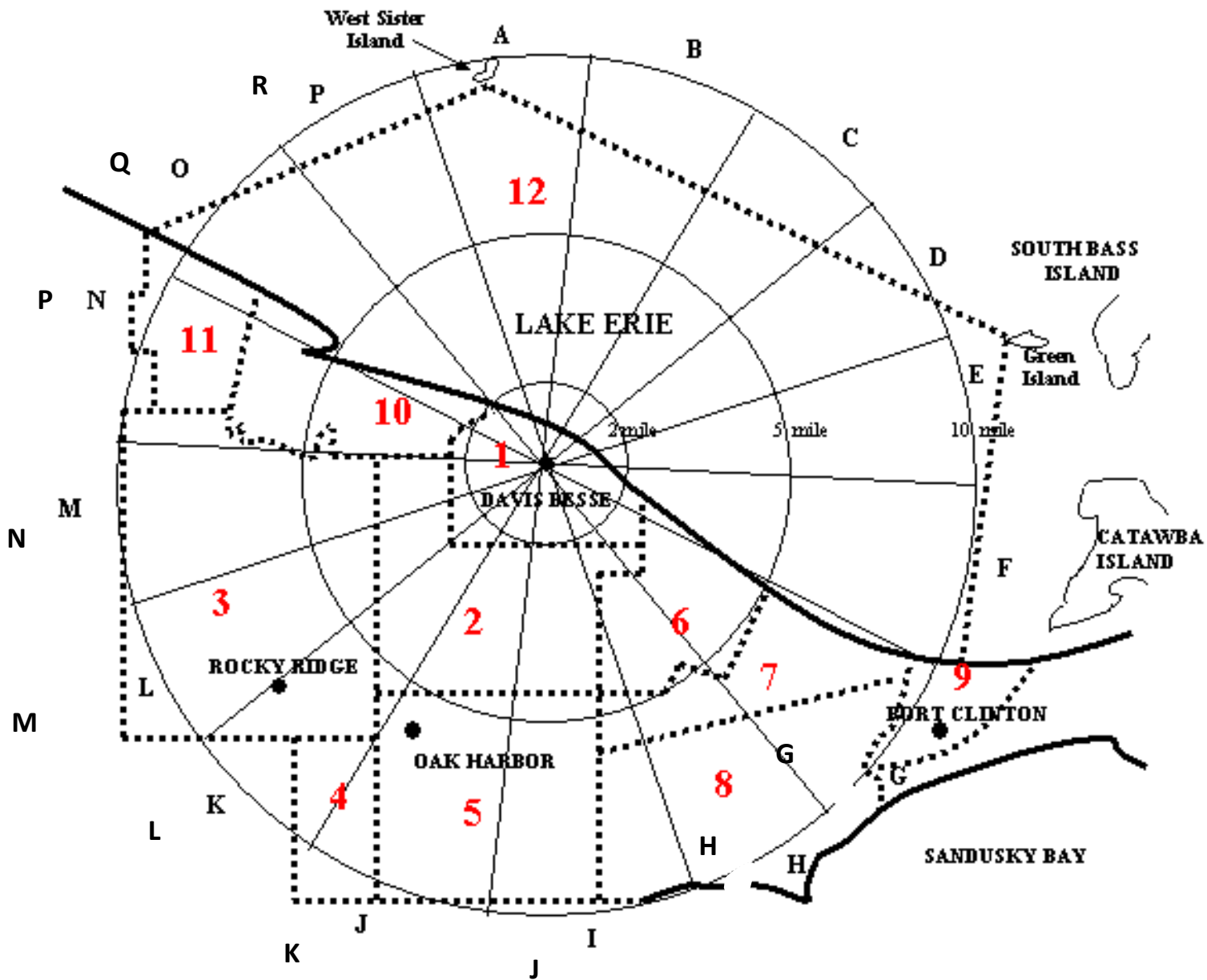
ATTACHMENT 1: FLOWCHART FOR
OFFSITE PROTECTIVE ACTION RECOMMENDATION DETERMINATION

Page 2 of 2

Wind Direction From	Protective Action	A - Evacuate	B - Evacuate	C – Shelter/Evacuate
		2-Mile Radius & 5-Miles Downwind	2-Mile Radius & 10-Miles Downwind	2-Mile Radius & 5-Miles Downwind
Unknown or Lake Breeze	Shelter	N/A	N/A	1, 2, 6
	Evacuate	1, 2, 6, 10, 12	ALL Subareas	10, 12
141° to 278°	Shelter	N/A	N/A	1
	Evacuate	1, 12	1, 12	12
279° to 286°	Shelter	N/A	N/A	1, 6
	Evacuate	1, 6, 12	1, 6, 7, 9, 12	12
287° to 293°	Shelter	N/A	N/A	1, 6
	Evacuate	1, 6, 12	1, 6, 7, 8, 9, 12	12
294° to 330°	Shelter	N/A	N/A	1, 2, 6
	Evacuate	1, 2, 6, 12	1, 2, 6, 7, 8, 9, 12	12
331° to 005°	Shelter	N/A	N/A	1, 2, 6
	Evacuate	1, 2, 6, 12	1, 2, 5, 6, 7, 8, 12	12
006° to 013°	Shelter	N/A	N/A	1, 2, 6
	Evacuate	1, 2, 6, 12	1, 2, 4, 5, 6, 7, 8, 12	12
014° to 020°	Shelter	N/A	N/A	1, 2
	Evacuate	1, 2, 12	1, 2, 4, 5, 12	12
021° to 065°	Shelter	N/A	N/A	1, 2
	Evacuate	1, 2, 12	1, 2, 3, 4, 5, 12	12
066° to 072°	Shelter	N/A	N/A	1, 2
	Evacuate	1, 2, 12	1, 2, 3, 4, 12	12
073° to 078°	Shelter	N/A	N/A	1, 2
	Evacuate	1, 2, 10, 12	1, 2, 3, 10, 12	10, 12
079° to 117°	Shelter	N/A	N/A	1, 2
	Evacuate	1, 2, 10, 12	1, 2, 3, 10, 11, 12	10, 12
118° to 122°	Shelter	N/A	N/A	1
	Evacuate	1, 10, 12	1, 3, 10, 11, 12	10, 12
123° to 140°	Shelter	N/A	N/A	1
	Evacuate	1, 10, 12	1, 10, 11, 12	10, 12

Once the PAR and subareas are selected **GO TO** Step 6.2.2

ATTACHMENT 2: COMPARISON OF OFFSITE SECTORS AND SUBAREAS
Page 1 of 1



COMMITMENTS

None