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May 7, 2001
L-01-066

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

Subject: Beaver Valley Power Station, Unit No. 1 and No. 2
BV-1 Docket No. 50-334, License No. DPR-66
BV-2 Docket No. 50-412, License No. NPF-73
Beaver Valley Power Station Emergency Preparedness Plan and Implementing Procedures (Volumes 1, 2 and 3)

In accordance with 10 CFR Part 50.4, this letter forwards recent revisions of the Beaver Valley Power Station Emergency Preparedness Plan and Implementing Procedures (Volumes 1, 2 and 3) to the Nuclear Regulatory Commission. The changes do not decrease the effectiveness of the Plan and the Plan, as changed, continues to meet the requirements of Appendix E of 10 CFR 50. Therefore, 10 CFR Part 50.54(q) requires that these changes be submitted for information only.

If there are any questions on this submittal, please contact Mr. Thomas S. Cosgrove, Manager, Regulatory Affairs at 724-682-5203.

Sincerely,

Lew W. Myers

Lew W. Myers

Enclosures

c: Mr. L. J. Bürkhart, Project Manager (w/o enclosures)
Mr. D. M. Kern, Sr. Resident Inspector (w/o enclosures)
Mr. H. J. Miller, NRC Region I Administrator (2 copies)

A045

ENCLOSURE I

The following summary of changes for the BVPS Emergency Preparedness Plan and Implementing Procedures are being provided:

Emergency Preparedness Plan (Volume 1):

Emergency Preparedness Plan – Section 4 Emergency Conditions

Emergency Preparedness Plan – Section 5 Emergency Organization

Emergency Preparedness Plan – Section 6 Emergency Measures

Emergency Preparedness Plan – Section 7 Emergency Facilities and Equipment

Emergency Preparedness Plan – Appendix A Letters of Agreement

Emergency Preparedness Plan – Appendix G References

Emergency Preparedness Plan – Table of Contents

Emergency Preparedness Implementing Procedures (Volumes 2 and 3):

EPP/IP Effective Index

EPP/I-1a Unit #1 Recognition and Classification of Emergencies (EALs)

EPP/I-1b Unit #2 Recognition and Classification of Emergencies (EALs)

EPP/IP 1.1 Notifications

EPP/IP 1.2 Communication and Dissemination of Information

EPP/IP 1.5 Emergency Support Center (OSC/ROC) Activation, Operation and Deactivation.

EPP/IP 2.6 Environmental Assessment and Dose Projection Controlling Procedure.

EPP/IP 4.1 Offsite Protective Actions

EPP/IP 6.2 Termination of the Emergency and Recovery

EPP/IP 7.1 Emergency Equipment Inventory and Maintenance Procedure

EPP/IP Annex B DELETED

EPP/IP Annex C Major Injury Involving Radioactive Contamination For The Medical Center, Beaver

Primary revision summary:

Emergency Action Levels (EALs)

- Single procedure split into separate EPP/I-1a UNIT #1 and EPP/I-1b UNIT #2 procedures
- New CRITERION and INDICATOR terminology
- EAL 2.4 Fuel Clad Degradation - reworded to refer to TS 3.8.4
- EAL 4.2 Explosion - added refererence to Security EAL

Protective Action Recommendations (PARs)

- PAR Flowchart revised to use as form for documentation
- Clarified decision blocks for PAR based on Plant Conditions or Dose Projection

Miscellaneous

- Removal of UPMC, Beaver (Aliquippa Hospital)

For a detailed listing, please refer to Enclosure II, BVPS Emergency Preparedness Plan Changes.

ENCLOSURE II
BVPS Emergency Preparedness Plan Changes

The following is a brief recap of the changes made to the Beaver Valley Power Station Emergency Plan and EPP/Implementing Procedures.

Emergency Preparedness Plan – Vol. 1

E-PLAN, SECTION 4 CHANGES – REV. 13

PAGE	SECTION	CHANGE	REASON
4-1	4.1	Deleted "recognition".	Terminology change to be consistent with NRC EPPOS #2 and NEI 99-02 Guidance.
4-1	4.1	Deleted "EPP/I-1" and added"EPP/I-1a/b".	Procedures are now EPP/I-1a (Unit #1) and EPP/I-1b (Unit #2). Ease of use and revisions per Unit.
4-1	4.1.1	Deleted "EPP/I-1" and added"EPP/I-1a/b".	Procedures are now EPP/I-1a (Unit #1) and EPP/I-1b (Unit #2). Ease of use and revisions per Unit.
4-2	4.1.1.1	Deleted "recognizes the initiating condition" and Added "for the initiating condition".	Terminology change to be consistent with NRC EPPOS #2 and NEI 99-02 Guidance.
4-2	4.1.1.2	Deleted "recognizes the initiating condition" and Added "for the initiating condition".	Terminology change to be consistent with NRC EPPOS #2 and NEI 99-02 Guidance.
4-5	4.1.2	Deleted "from the recognition of the indicator" and added "of sufficient indications being available to Control Room operators that an Emergency Action Level (EAL) has been exceeded."	Terminology change to be consistent with NRC EPPOS #2 and NEI 99-02 Guidance.

E-PLAN, SECTION 4 CHANGES – REV. 13

PAGE	SECTION	CHANGE	REASON
4-5	4.1.2	Added "is available via instrumentation, calculations, procedure Entry (AOPs, EOPs, etc.), operator knowledge of plant conditions (pressure, temperatures, etc.) in the Control Room, or reports received from plant personnel, whichever is most limiting," and deleted "a specific instrumentation reading, a physical condition, a report by plant personnel,".	Terminology change to be consistent with NRC EPPOS #2 and NEI 99-02 Guidance and provide additional guidance to Control Room personnel.
		Added "occurrence" and deleted "recognition".	Terminology change to be consistent with NRC EPPOS #2 and NEI 99-02 Guidance.
4-5	4.1.2	Deleted "from recognition of the indicator." And added "of indications being available to Control Room operators that an Emergency Action Level (EAL) has been exceeded."	Terminology change to be consistent with NRC EPPOS #2 and NEI 99-02 Guidance.
4-5	4.1.3	Deleted "EPP/I-1" and added"EPP/I-1a/b".	Procedures are now EPP/I-1a (Unit #1) and EPP/I-1b (Unit #2). Ease of use and revisions per Unit.
4-6	4.1.3.2	Added "occurs" and deleted "is recognized"	Terminology change to be consistent with NRC EPPOS #2 and NEI 99-02 Guidance.
	4.1.3.3	Added "classified" and deleted "recognized".	Terminology change to be consistent with NRC EPPOS #2 and NEI 99-02 Guidance.

E-PLAN, SECTION 4 CHANGES – REV. 13

PAGE	SECTION	CHANGE	REASON
4-7	4.1.3.4	Deleted period (.).	Typo
		Added "indications being available to Control Room operators that an Emergency Action Level (EAL) has been exceeded." and deleted "point of recognition or report of one or more indicators".	Terminology change to be consistent with NRC EPPOS #2 and NEI 99-02 Guidance.
4-9	4.2.1	Added "is available via instrumentation, calculations, procedure Entry (AOPs, EOPs, etc.), operator knowledge of plant conditions (pressure, temperatures, etc.) in the Control Room, or reports received from plant personnel, whichever is most limiting," and deleted "a specific instrumentation reading, a physical condition, a report by plant personnel,".	Terminology change to be consistent with NRC EPPOS #2 and NEI 99-02 Guidance and provide additional guidance to Control Room personnel.
		Added "occurrence" and deleted "recognition".	Terminology change to be consistent with NRC EPPOS #2 and NEI 99-02 Guidance.
4-15	4.4	Deleted "EPP/I-1" and added"EPP/I-1a/b".	Procedures are now EPP/I-1a (Unit #1) and EPP/I-1b (Unit #2). Ease of use and revisions per Unit.
4-62	TAB 2.4 UE	Deleted "indicates (a or b)" and added "exceeds Technical Specification 3.4.8". Deleted Step a and b.	Reference the applicable Tech. Spec. instead of providing the Tech. Spec. value.
4-64	TAB 2.6 UE	Deleted "OST 1.6.2 (2.6.2A)" and added "OST 1.6.2 or 1.6.2.A (2.6.2 or 2.6.2A) results"	Computer based procedure.
4-104	TAB 4.6 UE	Deleted "Duquesne Light Company" and added "BVPS".	Transitional Change.

E-PLAN, SECTION 4 CHANGES – REV. 13

PAGE	SECTION	CHANGE	REASON
4-131	TAB 6.2 UE	Deleted "temporary" and added LI-1RC-480 or LI-1RC-482C (2RCS-LI-102, LR-102)" and added "less than"	Typo. Use full Mark Number and no longer temporary equipment.
4-147	TAB 7.3 ALERT	Added "Unit 1" and "Unit 2".	Human factoring.
4-150	TAB 7.4 ALERT	Added "Unit 1" and "Unit 2".	Human factoring.
4-152	TAB 7.4 UE	Added "Unit 1" and "Unit 2".	Human factoring.
Various		Added Unit designation to appropriate Mark Numbers.	Site Standard.
Various		Corrected formatting errors	Formatting.
Various		"CV-3 (Unit 1/2 Cable Tunnel)" changed to "U1/U2 Cable Tunnel (CV3)".	Standardized reference.
Various		"Cable Vault & Rod Control Bldg." changed to "Rod Control Cable Vault Bldg.".	Standardized reference.

EMERGENCY PREPAREDNESS PLAN
SECTION 5– REV 15 - CHANGES

PAGE	SECTION	CHANGE	REASON
5-7	5.2.4	Deleted "the Assistant Nuclear Shift Supervisor" and replaced with "an opposite unit Senior Reactor Operator, as available.".	Ability to use personnel from opposite unit.
5-9	5.2.6	Deleted "Director" and replaced with "Supervisor".	Title change.
5-15	5.2.15	Deleted "Security Shift Supervisor" and replaced with "Supervisor, Nuclear Shift Security".	Title change.
		Added "The Security Coordinator reports directly to the Emergency Director.[C12]"	Correction CR# 01-0246
5-23	5.3.9	New Step.	CR# 01-0246
5-25	5.5.2	Added "Nuclear".	Correct Title.
5-26	5.5.5	Deleted "The University of Pittsburgh Medical Center, Beaver Valley".	Scheduled to close.
5-36	Table 5.1	Deleted "Chemist" and replaced with "Chemistry[C13]".	CR# 01-1168
5-49	Figure 5.5	Added "Contact" and [C12]" to EMA Representative (3) block. Deleted "Telecomm. Rep." Block and moved Logistics Coordinator block up to show Engr. Comm. Rep., Administrative Support, and Security reporting to Logistics Coordinator.	CR# 01-0246

EMERGENCY PREPAREDNESS PLAN
SECTION 6 – REV. 14 CHANGES

PAGE	SECTION	CHANGE	REASON
6-6	.4	Deleted "Director" and replaced with "Supervisor".	Title change.
6-7	6.3.1	Deleted "The University of Pittsburgh Medical Center, Beaver Valley. Added "Presbyterian University Hospital".	Scheduled to close. Corrected name.
6-30	.7	Added "monitoring" and deleted "/processing".	Processing of TLD's no longer done at BVPS.
6-41	6.8.2	Deleted "First aid personnel are trained with the Red Cross Multi media training materials." and replaced with "The qualified individuals are trained in First Aid/CPR."	First Aid course change and additional training.
6-42	6.8.4	Deleted "The University of Pittsburgh Medical Center, Beaver Valley.	Scheduled to close.
6-43	6.8.4	Deleted "The University of Pittsburgh Medical Center, Beaver Valley.	Scheduled to close.

EMERGENCY PREPAREDNESS PLAN
SECTION 7 – REV. 14 CHANGES

PAGE	SECTION	CHANGE	REASON
7-5	.3	Deleted "TLD processing equipment of" and "capacity" and replaced with "TLD availability".	Processing no longer done at BVPS.
	.7	Deleted "First Aid Room" and replaced with "Medical Services".	New terminology.
7-6	7.2	Deleted "the University of Pittsburgh Medical Center, Beaver County".	Scheduled to close.
7-26	Table 7.1	Deleted reference to CAS having Radcon Circuit.	Circuit does not exist.

EMERGENCY PREPAREDNESS PLAN
APPENDIX A– REV 12 - CHANGES

PAGE	SECTION	CHANGE	REASON
A-ii	List	Deleted "University of Pittsburgh Medical Center, Beaver Valley".	Scheduled to close.
A-ii	List	Deleted "Pennsylvania Power Company".	No longer required. PPCO is part of FirstEnergy.
A-iii	List	Deleted "Teledyne Brown Engineering Environmental Services	No longer required. Covered by Purchase Order with another Vendor.
A-iii	List	Deleted "Dobbs International Services".	No longer required. Covered by Purchase Order with another Vendor.

EMERGENCY PREPAREDNESS PLAN
APPENDIX G– REV 3 - CHANGES

PAGE	SECTION	CHANGE	REASON
App-G-1	Reference Section	Added new references. C13, C14	CR #01-1168 CR #00-3939

EMERGENCY PREPAREDNESS PLAN
TABLE OF CONTENTS– MAY 1, 2001 - CHANGES

PAGE	SECTION	CHANGE	REASON
		Replaced Revision Numbers	Noted revision number changes.

Emergency Preparedness/Implementing Procedures – Volume 2 and 3

EPP/I-1a CHANGES – REV. 0

PAGE	SECTION	CHANGE	REASON
All		Converted procedure from Pagemaker to MSWORD. Formatting.	Improve ability to update and view procedure on LAN.
Cover, i, ii, iii		Split one procedure (EPP/I-1a/b) into two (2) separate procedures Unit specific. Added Unit number.	Procedures are now EPP/I-1a (Unit #1) and EPP/I-1b (Unit #2). Ease of use and revisions per Unit.
1	B. References	Added NRC Emergency Preparedness Position Paper (EPPOS) #2, NEI 99-02, and Condition Report references.	Updated references.
	ALL EAL pages	Deleted "WXEPPI1A.DOC".	File reference not necessary.
	ALL EAL pages	Deleted "AXXXX.pm4".	File reference not necessary.

EPP/I-1a CHANGES – REV. 0

PAGE	SECTION	CHANGE	REASON
4	3.1.4.1	Deleted "from the recognition of the indicator" and added "of indications being available to Control Room operators that an Emergency Action Level (EAL) has been exceeded."	Terminology change to be consistent with NRC EPPOS #2 and NEI 99-02 Guidance.
	3.1.4.2	Replaced "a specific instrumentation reading, a physical condition, a report by plant personnel," with "is available via instrumentation, calculations, procedure Entry (AOPs, EOPs, etc.), operator knowledge of plant conditions (pressure, temperatures, etc.) in the Control Room, or reports received from plant personnel, whichever is most limiting,". Deleted "recognition" and replaced with "occurrence".	Terminology change to be consistent with NRC EPPOS #2 and NEI 99-02 Guidance and provide additional guidance to Control Room personnel.
	3.1.4.3	New Step created from last paragraph of 3.1.4.2. Deleted "from the recognition of the indicator" and replaced with "of sufficient indications being available to Control Room Operators that an Emergency Action Level (EAL) has been exceeded." Renumbered.	Terminology change to be consistent with NRC EPPOS #2 and NEI 99-02 Guidance and provide additional guidance to Control Room personnel.
5	3.3.3	Added "occurs" and deleted "is recognized".	Terminology change to be consistent with NRC EPPOS #2 and NEI 99-02 Guidance.
6	3.4.4	Deleted "recognized" and added "classified".	Terminology change to be consistent with NRC EPPOS #2 and NEI 99-02 Guidance.

EPP/I-1a CHANGES – REV. 0

PAGE	SECTION	CHANGE	REASON
6	3.5	Added "indications being available to Control Room operators that an Emergency Action Level (EAL) has been exceeded." Deleted "point of recognition or report of one or more INDICATORs."	Terminology change to be consistent with NRC EPPOS #2 and NEI 99-02 Guidance.
7	3.5.1.3	Deleted.	To be consistent with NRC EPPOS #2 and NEI 99-02 Guidance.
	3.7.1	Added the word "bold".	Clarification.
8	NOTE	Deleted "recognition" and added "occurrence".	Terminology change to be consistent with NRC EPPOS #2 and NEI 99-02 Guidance.
	NOTE	Deleted "If the required duration is exceeded, OR is unknown, when the condition is reported or recognized THEN the assessment time shall be limited to 15 minutes from the time of recognition."	Terminology change to be consistent with NRC EPPOS #2 and NEI 99-02 Guidance.
10	NOTE	Deleted "recognition" and added "occurrence".	Terminology change to be consistent with NRC EPPOS #2 and NEI 99-02 Guidance.
TABs	All	Formatting. Changed TAB numbers from gray to black and used inverse box (white on black) to designate current TAB.	Human factoring and copying.
		Corrected EAL 6.3 Modes 1-4 to Modes 5 & 6.	Correction. Actual EALs correct.
		Deleted Copyright by Duquesne Light Co.	Transition to FirstEnergy.

EPP/I-1a CHANGES – REV. 0

PAGE	SECTION	CHANGE	REASON
Defin.	Criterion, Indicator	Same as page 3-4	Terminology change to be consistent with NRC EPPOS #2 and NEI 99-02 Guidance.
	Strike Action	Replaced "Duquesne Light Company" with "BVPS".	Transition to FirstEnergy.
EAL's	ALL	Added "1" to Mark Numbers.	To show Unit specific for clarity.
EAL	TAB 1 CRITERIA UE 2.4	Replaced "RCS Specific Activity" with "Fuel Clad Degradation (RCS Specific Activity >LCO)".	Reworded to Tab title and UE Criterion.
EAL	TAB 2.4 UE	Step 2. Reworded to read "Radiochemistry analysis exceeds Technical Specification 3.4.8" and deleted sub-steps a and b.	Reference the applicable Tech. Spec. instead of providing the Tech. Spec. value.
EAL	2.6 UE	Added "or 1.6.2A"	Computer based procedure.
EAL	TAB 4.2 ALERT	Added "*Refer to Tab 4.6 "Security""*".	Human factoring
	TAB 4.2 UE	Added "*Refer to Tab 4.6 "Security""*".	Human factoring
EAL	Figure 4-B	Added "Owner Controlled Property".	Clarification
EAL	Figure 4-C	Added "Owner Controlled Property".	Clarification
	Table 4-2	Deleted "Water Treatment Building".	Water Treatment area no longer functional. Chemical previously used in that area no longer used.
EAL	6.2 SAE	Corrected "-482C" to read "LI-1RC-482C".	Typo.
	6.2 UE	Corrected "-482C" to read "LI-1RC-482C".	Typo.
EAL	Figure 7-A	Added "Owner Controlled Property".	Clarification

EPP/I-1b CHANGES – REV. 0

PAGE	SECTION	CHANGE	REASON
All		Converted procedure from Pagemaker to MSWORD. Formatting.	Improve ability to update and view procedure on LAN.
Cover, i, ii, iii		Split one procedure (EPP/I-1a/b) into two (2) separate procedures Unit specific. Added Unit number.	Procedures are now EPP/I-1a (Unit #1) and EPP/I-1b (Unit #2). Ease of use and revisions per Unit.
1	B. References	Added NRC Emergency Preparedness Position Paper (EPPOS) #2, NEI 99-02, and Condition Report references.	Updated references.
	ALL EAL pages	Deleted "WXEPPI1B.DOC".	File reference not necessary.
	ALL EAL pages	Deleted "AXXXX.pm4".	File reference not necessary.
4	3.1.4.1	Deleted "from the recognition of the indicator" and added "of indications being available to Control Room operators that an Emergency Action Level (EAL) has been exceeded."	Terminology change to be consistent with NRC EPPOS #2 and NEI 99-02 Guidance.
	3.1.4.2	Replaced "a specific instrumentation reading, a physical condition, a report by plant personnel," with "is available via instrumentation, calculations, procedure Entry (AOPs, EOPs, etc.), operator knowledge of plant conditions (pressure, temperatures, etc.) in the Control Room, or reports received from plant personnel, whichever is most limiting,". Deleted "recognition" and replaced with "occurrence". Deleted "from the recognition of the indicator" and replaced with "of indications being available to Control Room Operators that an Emergency Action Level (EAL) has been exceeded."	Terminology change to be consistent with NRC EPPOS #2 and NEI 99-02 Guidance and provide additional guidance to Control Room personnel.

EPP/I-1b CHANGES – REV. 0

PAGE	SECTION	CHANGE	REASON
	3.1.4.3	New Step created from last paragraph of 3.1.4.2. Deleted "from the recognition of the indicator" and replaced with "of sufficient indications being available to Control Room Operators that an Emergency Action Level (EAL) has been exceeded." Renumbered.	Terminology change to be consistent with NRC EPPOS #2 and NEI 99-02 Guidance and provide additional guidance to Control Room personnel.
5	3.3.3	Added "occurs" and deleted "is recognized".	Terminology change to be consistent with NRC EPPOS #2 and NEI 99-02 Guidance.
6	3.4.4	Deleted "recognized" and added "classified".	Terminology change to be consistent with NRC EPPOS #2 and NEI 99-02 Guidance.
7	3.5	Added "indications being available to Control Room operators that an Emergency Action Level (EAL) has been exceeded." Deleted "point of recognition or report of one or more INDICATORs."	Terminology change to be consistent with NRC EPPOS #2 and NEI 99-02 Guidance.
	3.5.1.3	Deleted.	To be consistent with NRC EPPOS #2 and NEI 99-02 Guidance.
	3.7.1	Added the word "bold".	Clarification.
8	NOTE	Deleted "recognition" and added "occurrence".	Terminology change to be consistent with NRC EPPOS #2 and NEI 99-02 Guidance.
9	NOTE	Deleted "If the required duration is exceeded, OR is unknown, when the condition is reported or recognized THEN the assessment time shall be limited to 15 minutes from the time of recognition."	Terminology change to be consistent with NRC EPPOS #2 and NEI 99-02 Guidance.

EPP/I-1b CHANGES – REV. 0

PAGE	SECTION	CHANGE	REASON
10	NOTE	Deleted "recognition" and added "occurrence".	Terminology change to be consistent with NRC EPPOS #2 and NEI 99-02 Guidance.
TABs	All	Formatting. Changed TAB numbers from gray to black and used inverse box (white on black) to designate current TAB.	Human factoring and copying.
		Corrected EAL 6.3 Modes 1-4 to Modes 5 & 6.	Correction. Actual EALs correct.
		Deleted Copyright by Duquesne Light Co.	Transition to FirstEnergy.
Defin.	Criterion, Indicator	Same as page 3-4	Terminology change to be consistent with NRC EPPOS #2 and NEI 99-02 Guidance.
	Strike Action	Replaced "Duquesne Light Company" with "BVPS".	Transition to FirstEnergy.
EAL's	ALL	Added "1" or "2" to Mark Numbers.	To show Unit specific for clarity.
EAL	TAB 1 CRITERIA UE 2.4	Replaced "RCS Specific Activity" with "Fuel Clad Degradation (RCS Specific Activity >LCO)".	Reworded to Tab title and UE Criterion.
EAL	TAB 2.4 UE	Step 2. Reworded to read "Radiochemistry analysis exceeds Technical Specification 3.4.8" and deleted sub-steps a and b.	Reference the applicable Tech Spec. instead of providing the Tech. Spec. value.
EAL	TAB 4.2 ALERT	Added "*Refer to Tab 4.6 "Security""*.	Human factoring.
	TAB 4.2 UE	Added "*Refer to Tab 4.6 "Security""*.	Human factoring.

EPP/I-1b CHANGES – REV. 0

PAGE	SECTION	CHANGE	REASON
EAL	Figure 4-B	Added "Owner Controlled Property".	Clarification
EAL	Figure 4-C	Added "Owner Controlled Property".	Clarification
	TABLE 4.1	Added "RWST 2QSS-TK21".	Enhancement to list previously omitted.
EAL	Figure 7-A	Added "Owner Controlled Property".	Clarification

EPP/IP 1.1 CHANGES – REV. 25

PAGE	SECTION	CHANGE	REASON
11	Att. 2 #8	Deleted "C.P. Hynes" and added "Albert Hartner and phone number (724-378-2639)".	Personnel change.
13	Att. 2	Replaced Site Rep contact.	Westinghouse Site Rep changed.
16	Att. 2	Deleted No. 33 – University of Pittsburgh Medical Center-Beaver Valley	Scheduled to close.
79	Note	Changed "Step 8.0" to "Step 9.0".	Typo.

IP 1.2 – REV. 16 CHANGES

PAGE	SECTION	CHANGE	REASON
4	1.7.3	Deleted "University of Pittsburgh Medical Center Beaver Valley,".	Scheduled to close.
7	Att. 1	Deleted reference to CAS having Radcon Circuit.	Circuit does not exist in CAS.
10	Att. 2	Added Phone numbers to NRCOC.	Supplied by NRC.
23	Att. 5 Step 1.2	Added "for ERO purposes".	Clarification.

IP 1.5 – REV. 12 CHANGES

PAGE	SECTION	CHANGE	REASON
1	5.0	New Reference	CR-00-2206
5	2.2.4	New Step.	Provided by HP.
8	G.	New Attachment	New Attachment.
21	Att. 7	New	Per CR-00-2206

IP 2.6 – REV. 13 CHANGES

PAGE	SECTION	CHANGE	REASON
5	5.2.2	Deleted "2.6.1" and replaced with "2.6.2".	Typo.
16	26.1	Deleted "REAP 5.512, Performing and with guidance from REAP 5.511, "Class B Model, Run Menu Option Selection". Added "a" and made "Dose Projection" lower case.	REAP 5.512 deleted per OSC-67-00 (10/31/00). Guidance not required with new computers.
19	31.1.1	Deleted "FRMAP" and replaced with "FRMAC".	Typo.

IP 4.1 – REV. 14 CHANGES

PAGE	SECTION	CHANGE	REASON
1	9.0	Added Reference to Condition Reports.	Reference CR# 00-2221 Reference CR# 00-2343
12-13	Att. 1	Added spaces to write in wind speed and 150/500' wind directions.	Provide area to document meteorological conditions used for PAR determinations.
		Added in first Decision Block: 1) "at least" in title, 2) "(or unavailable)" at first and third bullet, 3) underlined "difference, 4) added "(opposite wind directions)" in third bullet, and 5) added "(within one hour)"in fourth bullet.	Emphasis and clarification. 1) As soon as one bullet is identified as TRUE, can answer YES and go on to next Decision Block. 2) Clarifies if wind speed or wind direction data is unavailable, may not know if plume is "puddling" or direction it is moving, so a downwind wedge can not be determined, 3) emphasis, 4) Clarify wind directions are approximately 180 degrees apart, 5) Clarified imminent.
		Added information from Step D.3.1 to Downwind Wedge Determination Decision Block.	Incorporated information from procedure into Flowchart and reworded for human factoring.
		Added "(FSAR, monitor data, etc.)" to dose projection results available Decision Block (two locations in Flowchart).	Clarify information used for dose projections and PARs. If FSAR default calculation condition similar to current plant condition, FSAR dose projection information should be factored into decision.
		Added Decision Block for dose at EAB >1 Rem TEDE or 5 Rem CDE, associated arrows and "NO" (two locations in Flowchart).	Added information from step D.1.2 into Flowchart. Clarifies that no PAR is provided if based on dose projections less than the EPA Protective Action Guides. A PAR shall be provided based on plant conditions as a minimum.
		Bolded "EAB" in all locations.	Emphasis and human factoring.

IP 4.1 – REV. 14 CHANGES

PAGE	SECTION	CHANGE	REASON
		Added information on upgrading PARs to "Continue Assessment" Block.	Reminder that an upgraded PAR requires a new Initial Notification Form, but does not change the emergency classification from a General Emergency.
12-13	Att 1	Added "FINAL PAR APPROVAL" Block.	Provide signature approval location upon final determination of a PAR. This documents methodology if multiple PARs necessary.
		Added A5.715GP Record Type List number on both pages.	New Form number.
15	Att. 2 Step 1.0	Replaced "envelope from the EOF Emergency Cabinet" with "from the Assistant to the E/RM Workbook in the EOF".	Information for activating Conference call no longer kept in cabinet.
16	4.0 & 4.1	Reworded and added Step 4.1.	Clarification for introductions on Conference Call.
17	5.0 & 6.0	Reversed Steps and renumbered.	Clarification and human factoring.

EPP/IP 6.2 – REV. 10 CHANGES

PAGE	STEP	CHANGE	REASON
10	Att. 1, Step 16	Deleted	Organizational Title change.

IP 7.1 – REV. 12 CHANGES

PAGE	SECTION	CHANGE	REASON
1	A.	Replaced "Director, Emergency Preparedness" with "Manager, Emergency Preparedness".	Organizational Title change.
	C.	Deleted "Department" after Health Physics and replaced "EP Department" with "Emergency Preparedness".	Organizational change.
	D.1.1	Replaced "Director, Emergency Preparedness" with "Manager, Emergency Preparedness".	Organizational Title change.
2	1.2	Replaced "Director, Emergency Preparedness" with "Manager, Emergency Preparedness".	Organizational Title change.
	2.3	Deleted "Department".	Organizational change.
	2.4	Replaced "Director, Emergency Preparedness" with "Manager, Emergency Preparedness".	Organizational Title change.
4	NOTE	Replaced "Director, EP" with "Manager, EP".	Organizational Title change.
	5.3	Deleted "the EP Department" and replaced with "Emergency Preparedness".	Organizational change.
7	Att. 1	Deleted "University of Pittsburgh Medical Center Beaver Valley,".	Scheduled to close.
	NOTE	Deleted "Department".	Organizational change.

EPP/IP ANNEX B CHANGES

DELETED

EPP/IP ANNEX C – REV 9 - CHANGES

PAGE	SECTION	CHANGE	REASON
2	II.	Deleted "Duquesne Light" in two places and replaced with "BVPS".	Transitional change.
2	III.A.3	Deleted "Room C-4 (Critical Care 4)" and added "designated Treatment Room".	Per hospital designation and response area available.
3	III.B.1	Added "designated" and deleted "across from Room C-4".	Per hospital designation and response area available.
3	III.B.2	Added "the designated Treatment Room", "and/or" and deleted "Room C-4".	Per hospital designation and response area available.
3	III.B2.a.	Deleted "canvas" and added "J-Flex".	New material being used.
3	III.B2.b.	Deleted "Room C-4" and added "designated Treatment Room".	Per hospital designation and response area available.
4	III.B2.d	Deleted "C-4" and added "of the Treatment Room.	Per hospital designation and response area available.
4	III.B.3	Deleted "C-4" and added "designated Treatment Room.	Per hospital designation and response area available.
4.	III.B.3.b	Added "A large enough area to handle the number of patients and/or the".	Per hospital designation and response area available.
4	III.B.3.f.1)	Deleted "Room C-3" and added "the adjacent room" and Deleted "Room C-4" and added "designated Treatment Room".	Per hospital designation and response area available.

EPP/IP ANNEX C – REV 9 - CHANGES

PAGE	SECTION	CHANGE	REASON
5	III.B.4.	Deleted "Room C-4" and added "designated Treatment Room".	Per hospital designation and response area available.
5	III.B.5.	Deleted "Room C-4" and added "designated Treatment Room".	Per hospital designation and response area available.
5	III.B.5.	Deleted "Duquesne Light" and replaced with "BVPS".	Transitional change.
5	III.C.1.	Deleted "Room C-4" and added "designated Treatment Room".	Per hospital designation and response area available.
6	III.C.4	Deleted "Room C-4" and added "designated Treatment Room".	Per hospital designation and response area available.
6	III.C.2., 3., 4., 5., and 6.	Deleted "Duquesne Light" and replaced with "BVPS".	Transitional change.
7	III.D.2.d., and e.	Deleted "Duquesne Light" and replaced with "BVPS".	Transitional change.
8	III.D.2.j.	Deleted "Duquesne Light" and replaced with "BVPS".	Transitional change.
8	IV.A.1. and 2.	Deleted "Duquesne Light" and replaced with "BVPS".	Transitional change.
13	Appendix B, II.	Deleted "Duquesne Light" and replaced with "FirstEnergy".	Transitional change.
14	Appendix B, II. First paragraph	Deleted "Aliquippa Hospital" and reworded.	Scheduled to close.
14	Appendix B, II. second paragraph	Deleted "Aliquippa Hospital" and replaced with The Medical Center, Beaver".	Scheduled to close.

EPP/IP ANNEX C – REV 9 - CHANGES

PAGE	SECTION	CHANGE	REASON
14	Appendix B, II. third paragraph	Replaced "TMC" with "The Medical Center, Beaver".	Corrected name.
14	Appendix B, II. third paragraph	Deleted "Room C-4" and added "designated Treatment Room".	Per hospital designation and response area available.
18	Title	Replaced "Duquesne Light Company" with "FirstEnergy Nuclear Operating Company".	Transitional Change.
19	Title	Replaced "Duquesne Light Company" with "FirstEnergy Nuclear Operating Company"	Transitional Change.
19	(3)	Replaced "Duquesne Light" with "BVPS'.	Transitional Change.
19	(4)	Deleted "Room C-4".	Per hospital designation and response area available.
19	(2)	Replaced "Duquesne Light" with "BVPS'.	Transitional Change.
20	Title	Replaced "Duquesne Light Company" with "FirstEnergy Nuclear Operating Company"	Transitional Change.
20	(3)	Replaced "Duquesne Light" with "BVPS'.	Transitional Change.
	(4)	Replaced "Duquesne Light" with "BVPS'.	Transitional Change.
21	Title	Replaced "Duquesne Light Company" with "FirstEnergy Nuclear Operating Company"	Transitional Change.

B.V.P.S. -- E.P.P.

REVIEW AND REVISION OF THE EMERGENCY PREPAREDNESS PLAN

E.P.P. APPROVAL

CHAPTER 57

VOLUME 1, EMERGENCY PREPAREDNESS PLAN

Rev. No.	Pages Issued	OSC Approval Date	APPROVAL		Effective Date
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Rev. 5	ALL	N/A	Ernest Chaffee	11/22/93	1/1/94
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Rev. 6A	SECTION 5 PAGES 5-27, 5-28, 5-29 & 5-30 SECTION 6 PAGES 6-5 & 6-6	N/A	Ernest Chaffee	5/30/95	6/9/95
Rev. 7	ALL	N/A	[signature]	12/15/95	12/22/95
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Rev. 11	Sections 1,2,3,5,6,7,8 9,10, Appendix A,B,C,D,E F	N/A	Robert E. Keiser	12/11/98	12/30/98
Rev. 0	Appendix G				

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REVIEW AND REVISION OF THE EMERGENCY PREPAREDNESS PLAN

E.P.P. APPROVAL

Chapter 57

Volume 1, Emergency Preparedness Plan

Rev. No.	Pages Issued	OSC Approval Date	APPROVAL		Effective Date
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Rev. 12	ALL (Except Appendix A and E) (Appendix G, Rev. 1)	N/A	Robert E. Fenker	11/24/99	12/2/99
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Emergency Preparedness Plan

Beaver Valley Power Station

EMERGENCY CONDITIONS

Section 4

EMERGENCY CONDITIONS

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Section 4

EMERGENCY CONDITIONS

Table of Contents

EMERGENCY CONDITIONS

4.0 EMERGENCY ACTION LEVEL BASES

4.1 CLASSIFICATION OF EMERGENCIES

Emergency conditions are classified into one of four categories covering the spectrum of postulated accidents from those events which indicate a potential degradation of the level of plant safety or result in a radiological emergency ranging from a single location in-plant to those involving large numbers of people offsite. Emergency planning is based primarily on the minimization of any potential or resultant radiation exposure to individuals onsite and offsite. Specific criteria are provided for the classification, and[CI4] declaration of each of the emergency classes. The scheme provides for notification of appropriate emergency response organizations and for implementation of actions immediately applicable to a specific condition. Provisions are included for a graded scale of response to conditions within each classification, and for upgrading, downgrading, or terminating the emergency classification in the event of a change in the severity of the emergency condition.

This section describes the scope and identifies events which comprise each of the four emergency classifications. Emergency Action Levels "EALs" based on the criteria, and the specific plant parameters to which the EALs refer and the instrument(s) on which that parameter is indicated are specified in EPP/I-1a/b, *Recognition and Classification of Emergencies.* Action statements referring the operator to the Emergency Implementing Procedures are incorporated, where appropriate, in the Beaver Valley Power Station Operating Procedures. To the extent feasible, the EALs are based on readily available information such as Control Room instrumentation readings which, if exceeded, will initiate assessment measures. Immediate actions to be taken in response to conditions involving plant parameters, such as Technical Specification Limiting Conditions for Operation (LCO), are detailed in the Beaver Valley Power Station alarm response procedures, Abnormal Operating Procedures, and Emergency Operating Procedures. Other immediate actions and follow-up actions are identified in Section 6 of this Plan and are described in detail in applicable Emergency Implementing Procedures, listed in Appendix C.

The emergency classification scheme is coordinated with state and local agencies, and was reviewed by the Nuclear Regulatory Commission. Periodic training is conducted (see Section 8 of the Plan) on the classification scheme. These activities ensures that the classification scheme is compatible with the scheme used by those agencies.

4.1.1 Classification Categories

The emergency classification system is described in detail in EPP/I-1a/b, *Recognition and Classification of Emergencies.* The bases of this scheme are addressed in Section 4.2 of the Plan. The classification scheme is based on four emergency classifications:

4.1.1.1 Unusual Event

Events within this classification meet the following definition:

Unusual events are in process or have occurred which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

Such events characterize abnormal plant conditions which, by themselves, do not constitute significant emergency conditions, but are considered to be potential precursors to more severe conditions. In this use, a precursor is a condition that could, if appropriate action were not taken, escalate to a more severe condition. The purpose of this classification is to ensure that the plant operating staff, takes appropriate action for the initiating condition, such as [C14] assessment and verification, and comes to a state of readiness to respond in the event that the condition becomes more severe. Offsite authorities are notified of this classification within 15 minutes, however, with the possible assistance by local support groups such as fire companies or medical facilities, no offsite response is expected.

4.1.1.2 Alert

Events within this classification meet the following definition:

Events are in process or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of the EPA protective action guideline exposure levels.

Such events characterize plant conditions that warrant activation of the site emergency response organization and augmentation of onsite emergency resources. The purpose of this classification is to ensure that the plant operating staff takes appropriate action for the initiating condition, such as assessment and verification, and [C14] activates the emergency response organization. Offsite authorities are notified of this classification within 15 minutes. Some offsite agencies may place their respective emergency organizations on standby.

4.1.1.3 Site Area Emergency

Events within this classification meet the following definition:

Events are in process or have occurred which involve actual or likely major failures of plant functions 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 outside the Exclusion Area Boundary

Such events characterize plant conditions that warrant activation of the site emergency response organization, augmentation of onsite emergency resources, and constitute the lowest level where offsite emergency response may be necessary. Offsite emergency response organizations activate in anticipation of the need to implement offsite protective actions should the condition degrade.

4.1.1.4 General Emergency

Events within this classification meet the following definition:

Events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA protective action guidelines exposure levels outside the Exclusion Area Boundary

At this classification, total activation of the onsite and offsite emergency response organizations is required. The onsite organization shall recommend offsite protective actions to designated offsite agencies. These offsite organizations, following evaluation of the onsite recommendation, will implement appropriate offsite protective actions.

4.1.2 Classification Scheme

The classification scheme is comprised of a number of emergency action levels, arranged by severity of the event and by the type of condition. There are two general types of emergency action levels included in this procedure:

- Barrier-Based EALs: These EALs address conditions that represent potential losses, or losses, of one or more of the Fuel Clad, RCS, or Containment fission product barriers. Indicators of these conditions include Critical safety function status, fundamental indications such as subcooling or reactor vessel water level, or auxiliary indications such as containment radiation monitor readings. Classifications are based on the number of barriers lost or potentially lost.

- Event-Based EALs: These EALs address discrete conditions or events that are generally precursors to fission product barrier degradation, or are otherwise degradations in the level of safety of the plant. Events may be external (*e.g., severe weather, earthquakes, loss of offsite power*) internal (*e.g., fires, explosions, instrumentation failure*) or may involve radioactivity releases.

The EALs are grouped by recognition category as follows:

Section 1	Fission Product Barrier Matrix
Section 2	System Degradation
Section 3	Loss of Power
Section 4	Hazards and ED Judgment
Section 5	Destructive Phenomena
Section 6	Shutdown Systems Degradation
Section 7	Radiological

Each of the EAL sections includes one or more columns, or Tabs, that address one initiating condition (*e.g., fires*). Each tab provides EALs for each of the four emergency classifications, as applicable. A notation adjacent to each EAL identifies the plant operating mode(s) for which the EAL is applicable.

Each EAL is comprised of a Criterion, printed in bold type, and one or more Indicators. The purpose of each is as follows:

- CRITERION: identifies the emergency condition and any numeric values which define that condition (*i.e., the basis of the declaration*) All classifications are based on an assessment *(i.e., determination that the condition is valid)* by the Emergency Director that the criterion has been met or exceeded. Implicit in this protocol is the necessity for these assessments to be completed within 15 minutes (unless otherwise noted) of sufficient indications being available to Control Room operators that an Emergency Action Level (EAL) has been exceeded. [C14]

- INDICATOR: is available via instrumentation, calculations, procedure Entry (AOPs, EOPs, etc.), operator knowledge of plant conditions (pressure, temperatures, etc.) in the Control Room, or reports received from plant personnel, whichever is most limiting, or other evidence that the associated [C14] criterion may be exceeded. Upon occurrence of one or more indicators, the Emergency Director performs an assessment against the criterion. Depending on the particular condition, this assessment may be as simple as a review of the criterion, an instrument channel check, or a detailed calculation as in the case of a radioactivity release. Inherent in this protocol is the necessity for these assessments to be completed within 15 minutes (unless otherwise noted) of indications being available to Control Room operators that an Emergency Action Level (EAL) has been exceeded. [C14]

  The indicators were selected with the objective of providing unambiguous guidance to assist with assessment of the criterion. There may be other indicators not envisioned by the writers of this scheme that, in the judgment of the Emergency Director, correspond to the criterion. In these cases, the Emergency Director should base the declaration on engineering judgment, using the supplied indicators as examples of the severity of the condition.

4.1.3 Implementation of the Classification Scheme

This section addresses how the scheme is implemented. Complete instructions are provided in EPP/I-1a/b, *Recognition and Classification of Emergencies.*

4.1.3.1 Events Affecting Both Units

If an event occurs such that both reactor units are affected, e.g., tornado, toxic gas offsite, etc., the senior Nuclear Shift Supervisor makes the appropriate classification and assumes the role of Emergency Director. If the common plant condition results in a higher emergency classification at one reactor unit, the Nuclear Shift Supervisor from that unit makes the appropriate classification and assumes the role of Emergency Director.

4.1.3.2 Mode Applicability

The plant operating mode that existed at the time that the event occurred, prior to any protective system or operator action initiated in response to the condition, is compared to the mode applicability of the EALs. If an event occurs, and a lower or higher plant operating mode is reached before the classification can be made, the classification is based on the mode that existed at the time that the event occurred. The fission product barrier matrix is applicable only to those events that occur at mode 4 or higher. An event that occurs in modes 5 or 6 is not classified using the fission product [C14] barrier matrix, even if mode 4 is entered due to subsequent heatup. In these cases, Tab 6, Shutdown Systems Degradation, is used for classification.

4.1.3.3 Transient Events

For some EALs the existence of the event, without regard to duration, is sufficient to warrant classification. In these cases, the appropriate emergency classification is declared as soon as the Emergency Director assessment concludes that the criterion is exceeded. However, some EALs specify a duration of occurrence. For these EALs the classification is made when Emergency Director assessment concludes that the specified duration is exceeded or will be exceeded (*i.e., condition can not be reasonably rectified before the duration elapses*), whichever is sooner. In many cases, the plant operating staff will be able to take actions to correct the abnormal condition before a classification is made. These situations are handled as follows:

- If the plant condition exceeding an EAL criterion is rectified before the specified duration time is exceeded, then the event is not classified by that EAL. Lower severity EALs shall be reviewed for applicability.

- If the plant condition exceeding an EAL criterion is not classified at the time of occurrence, but is identified well [C14] after the condition has occurred (e.g., as a result of routine log or record review) and the condition no longer exists, an emergency is not declared. However, reporting under 10 CFR 50.72 may be required. Such a condition could occur, for example, if a follow-up evaluation of an abnormal condition was more severe than earlier believed.

- If an emergency classification was warranted, but the plant condition has been rectified (such that the CRITERION is no longer exceeded) prior to declaration and notification, the following guidance applies:

  For transient events that would have been declared as Unusual Events, no emergency is declared. However, the event shall be reported to those local, state, and Federal agencies designated to receive the initial notifications. These agencies shall be told that the Unusual Event condition was rectified upon detection and no emergency is being declared.

  For transient events that would have been declared as an Alert or higher, the event shall be declared and the emergency response organization activated.

4.1.3.4 Declaration Timing and Assessment

Emergency conditions are classified as soon as the Emergency Director assessment of the indicators shows that the criterion is exceeded. The assessment time starts from the indications being available to Control Room operators that an Emergency Action Level (EAL) has been exceeded.[CI4] The assessment time is limited to 15 minutes unless the EAL specifies a duration (*e.g., release exceeds T/S for one hour*). In this case, the assessment time runs concurrently with the required duration and is the same length (*e.g., in this example, one hour*). If the assessment cannot be completed within the specified period, then the event is declared on the basis of indicators that cannot be reasonably discounted.

4.2 EAL Bases

The Beaver Valley Power Station emergency action levels were based on the guidance contained in NUMARC/NESP-007, *Methodology for Development of Emergency Action Levels*, Rev 2, 1/92. USNRC Regulatory Guide 1.101, *Emergency Planning and Preparedness for Nuclear Power Reactors*, Rev 3, 8/92. This section identifies the NUMARC/NESP-007 Initiating Condition, the corresponding EAL at BVPS, and the status of implementation. With regard to this latter item, the term "deviation" appears adjacent to the BVPS reference if the BVPS EAL differs in intent from the NUMARC guidance. In this use, a change from the original guidance is considered an intent change if, as a result of difference, the threshold for a classification is modified such that the BVPS EAL will result in a different classification than the NUMARC guidance for the same event. Similarly, omissions of EALs specified by the NUMARC guidance are marked as deviations.

Minor changes from the NUMARC guidance, such as terminology changes, format, re-wording that does not change intent, and other similar site specific adaptation are not considered as intent changes and are not marked as deviations.

Justification for each of the deviations was documented separately and was made available during the regulatory review of these EALs.

4.2.1 Generic Terminology Changes

The table below compares terminology changes from the NUMARC guidance that are generic to all BVPS EALs.

NUMARC TERM	CORRESPONDING BVPS TERM	DISCUSSION
Initiating Condition	CRITERION	In the BVPS EALs, the CRITERION identifies the emergency condition and any numeric values which define that condition *(I.e., the basis of the declaration)* All classifications are based on an assessment (i.e., determination the condition is VALID) by the Emergency Director that the CRITERION has been met or exceeded.
Example EAL	INDICATOR	In the BVPS EALs, the INDICATOR is available via instrumentation, calculations, procedure Entry (AOPs, EOPs, etc.), operator knowledge of plant conditions (pressure, temperatures, etc.) in the Control Room, or reports received from plant personnel, whichever is most limiting, or other evidence that the associated CRITERION may be exceeded. Upon occurrence of one or more INDICATORs, the Emergency Director performs an assessment against the CRITERION. C14
Recognition Category	Recognition Category	The BVPS EALs are separated into seven recognition categories, each of which is section. There are seven sections: (1) Fission Product Barrier Matrix, (2) System Degradation, (3) Loss of Power, (4) Hazards and ED Judgment, (5) Destructive Phenomena, (6) Shutdown System Degradation, and (7) Radiological. These seven sections are further sub-divided into two or more TABs that address a particular type of event. For example, "Loss of AC", and "Loss of DC" are TABs in the 'Loss of Power' Section. There are 36 TABs.
n/a	EAL	The term EAL refers to the CRITERION and INDICATOR(s) for a particular classification and TAB.

4.3 EAL Matrix

4.3.1 NUMARC/NESP-007 Abnormal Rad Levels/Radiological Effluent

NUMARC/NESP-007 Reference		BVPS Reference	
AU1	Gaseous or Liquid Effluent	7.1.U 7.2.U	Gaseous effluents liquid effluents
AU2	Plant Radiation Levels	7.3.U 7.4.U 6.5.U	Addresses example EAL#4 Addresses example EAL #1,3 Addresses example EAL #1,3
AA1	Gaseous or Liquid Effluent	7.1.A 7.2.A	Gaseous effluents Liquid effluents
AA2	Fuel Damage/Loss of Water Level	7.4.A 6.5.A	 Addresses example EAL #1,2
AA3	Plant Radiation Levels	7.3.A	
AS1	Gaseous Effluent	7.1.S	Deviation
AG1	Gaseous Effluent	7.1.G	Deviation

4.3.2 NUMARC/NESP-007 Fission Product Barrier Degradation

NUMARC/NESP-007 Reference		BVPS Reference
FU1	Loss or Potential Loss of CNMT	FPM
FA1	Loss or Potential Loss of either RCS/Fuel	FPM
FS1	Loss or Potential Loss of both RCS/Fuel	FPM Modification
FG1	Loss of Two and Potential Loss of Third	FPM
Fuel	Indicator 1	1.1.1
Fuel	Indicator 2	1.1.4
Fuel	Indicator 3	1.1.2
Fuel	Indicator 4	1.1.3
Fuel	Indicator 5	1.1.6
Fuel	Indicator 6	1.1.5 Addition
Fuel	Indicator 7	1.1.7
RCS	Indicator 1	1.2.1
RCS	Indicator 2	1.2.3
RCS	Indicator 3	1.2.4 Modification 1.3.4
RCS	Indicator 4	1.2.5
RCS	Indicator 5	1.2.2 Addition
RCS	Indicator 6	1.2.6
CNMT	Indicator 1	1.3.1
CNMT	Indicator 2	1.3.2
CNMT	Indicator 3	1.3.3
CNMT	Indicator 4	1.3.4
CNMT	Indicator 5	1.3.5
CNMT	Indicator 6	1.3.1 2.2.G Addition
CNMT	Indicator 7	1.3.4 Modification & Addition
CNMT	Indicator 8	1.3.6

4.3.3 NUMARC/NESP-007 Hazards and Other Conditions Affecting Plant Safety

NUMARC/NESP-007 Reference		BVPS Reference	
HU1	Destructive Phenomena in Protected Area	5.1.U	(Addresses example EAL #1)
		5.2.U	(Addresses example EAL #2)
		5.3.U	(Addresses example EAL #4)
		4.2.U	(Addresses example EAL #5)
		2.9.U	(Addresses example EAL #6)
		5.6.U	(Addresses example EAL #4)
		5.4.U	(Addresses example EAL #7)
HU2	Fire	4.1.U	
HU3	Flammable or Toxic Gases	4.3.U	(Flammable)
		4.4.U	(Toxic gas)
HU4	Security	4.6.U	
HU5	Emergency Director Judgment	4.7.U	
		2.10.U	(Uncontrolled cooldown)
HA1	Destructive Phenomena in Vital Area	5.1.A	(Addresses example EAL #1)
		5.2 A	(Addresses example EAL #2)
		5.3.A	(Addresses example EAL #5)
		2.9.A	(Addresses example EAL #6)
		5.4.A	(Addresses example EAL #7)
		5.5.A	(Addresses example EAL #7)
HA2	Fire/Explosion Affecting Safety Systems	4.1.A	(Fire)
		4.2.A	(Explosion)
HA3	Toxic/Flammable Jeopardizes	4.3.A	(Flammable Gas)
		4.4.A	(Toxic Gas)
HA4	Security Event in Protected Area	4.6.A	
HA5	Control Room Evacuation	4.5.A	
HA6	ED Judgment	4.7.A	
HS1	Security Event in Plant Vital Area	4.6.S	
HS2	Control Room Evacuation	4.5.S	Also 4.1.S (App. R Procedure)
HS3	ED Judgment	4.7.S	
HG1	Security Event / Loss of Ability to S/D	4.6.G	
HG2	ED Judgment	4.7.G	Also 4.1.G (App. R Procedure w/ failures)

4.3.4 NUMARC/NESP-007 System Malfunction

NUMARC/NESP-007 Reference		BVPS Reference
SU1	Loss of Offsite Power	3.1.U (Power Ops) 3.2.U Addition - (Shutdown) 6.3.U Addition - (Shutdown)
SU2	T/S Shutdown	2.7.U 2.8.U Addition
SU3	Loss of Annunciators	2.1.U
SU4	Fuel Clad Degradation	2.4.U
SU5	RCS Leakage	2.5.U Modification - (Unidentified) 2.6.U Addition - (Identified)
SU6	Loss of Communication	2.2.U
SU7	Loss of Required DC during S/D	3.3.U Addition 6.4.U
SA1	Loss of Offsite and Onsite AC-S/D	3.2.A 6.3.A Addition
SA2	Failure to Scram - Manual Trip Successful	2.3.A
SA3	Inability to Maintain Cold Shutdown	2.2.A Modified 6.1.U 6.1.A
SA4	Loss of Annunciators	2.1.A
SA5	AC Power Degraded	3.1.A
SS1	Loss of All AC Power	3.1.S
SS2	Failure to Trip - Manual Trip Unsuccessful	2.3.S
SS3	Loss of Vital DC Power	3.3.S
SS4	Loss of Function to Achieve Hot S/D	2.2.S
SS5	Loss of Water Level Uncovering Fuel	6.2.S
SS6	Inability to Monitor Transient	2.1.S
SG1	Prolonged Loss of All AC Power	3.1.G
SG2	Failure to Trip/Challenge to Core	2.3.G

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4.4 Individual EAL Basis Descriptions

In the section that follows, each EAL is described and the bases are provided.

> NOTE
> This section may be referenced for guidance in understanding an EAL, particularly those events involving ED judgment. However, emergency classifications shall be made from EPP/I-1-1a/b, *Recognition and Classification of Emergencies*, the information in which has precedence over the information in this section.

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4.5 SPECTRUM OF POSTULATED ACCIDENTS

The classification of accidents and corresponding protective actions required relative to off-normal and significant emergency conditions are based on operational conditions and projected dose commitment. Methods are described in this Plan and in Emergency Implementing Procedures for projecting, measuring, and evaluating those dose commitments. In nearly all cases, the proper response to an unusual event or emergency condition requires a considerable degree of judgment by the Emergency Director, based on experience and knowledge of the details pertaining to the condition. This requirement is exemplified in this discussion of specific postulated accidents.

The discrete accidents addressed in this section are described in the Beaver Valley Power Station Unit #1 and Unit #2 Final Safety Analysis Report (FSAR). Discussion of these postulated accidents identifies the instrumentation and other mechanisms which will be employed for prompt detection of an event and continued assessment of the consequences and plant status and describes how each accident is encompassed within the emergency classification system of this Plan.

The postulated offsite doses from these events are documented in the UFSARs for both Units. These analyses are performed using conservative worst case assumptions.

Since the offsite dose from an actual event will likely be different, dose assessments performed at the time of the event are used to classify the event and, as necessary, make Protective Action Recommendations.

The manpower needed to take immediate action to minimize damage to the plant equipment, and to initiate protective measures for onsite and offsite individuals is provided by the normal shift operating crew. The composition of this around-the-clock crew, the emergency assignments for these individuals, and arrangements for augmentation with emergency support personnel, are described in Section 5.

4.5.1 Core and Coolant Boundary Accidents

The Beaver Valley Power Station FSAR identifies several core and coolant boundary accidents primarily related to unintentional changes in plant conditions which lead to changes in core temperature, pressure, and/or reactivity. These accident analyses show that there should be minimal damage to the core and no expected release of radioactivity to the environment. The accidents are accommodated with, at most, a reactor shutdown with the unit being capable of returning to operation after a corrective action. The accidents analyzed are:

.1 Uncontrolled Rod Cluster Control Assembly (RCCA) bank withdrawal from subcritical

.2 Uncontrolled RCCA bank withdrawal from power

.3 RCCA misalignment

.4 Uncontrolled boron dilution

.5 Partial loss of forced reactor coolant flow

.6 Startup of an inactive reactor coolant loop

.7 Loss of external electric load and/or turbine trip

.8 Loss of normal feedwater

.9 Excessive heat removal due to feedwater system malfunctions

.10 Excessive load increase accident

.11 Loss of offsite power (station blackout to the unit auxiliaries)

.12 Turbine-generator accidents

.13 Accidental depressurization of the main steam system

.14 Accidents due to external environmental causes

.15 Accidental depressurization of the reactor coolant system

These conditions, by themselves, do not constitute significant emergency conditions. However, these off-normal conditions do indicate a potential degradation in the level of plant safety and could escalate to a more severe condition if appropriate action is not taken.

4.5.2 Fuel Handling Accident

The fuel handling accident as described in the BV-1 FSAR is postulated to involve dropping a single fuel assembly (264 fuel rods) during handling such that all rods are damaged. It is assumed that 100% of the noble gas gap inventory and 1% of the halogen gap inventory would be released to the fuel handling building. Of this, all of the noble gas and 5% of the halogens would be released to the environment.

A fuel handling accident, per BV-2 FSAR, is defined as the dropping of one spent fuel assembly onto another fuel assembly in the spent fuel storage area. The accident is postulated to cause damage to all of the fuel rods in the dropped assembly plus an additional 50 rods in the struck fuel assembly with subsequent release of all the activity in the fuel rod gap. The gap activity in the core fuel assemblies consists of 10 percent of the core noble gas and iodine activities, except for Kr-85, which is taken as 30 percent of the Kr-85 core activity at the time of the accident.

Initial assessment of this accident includes the performance of dose projections in accordance with Emergency Implementing Procedures. Dose projections utilize data from the Reactor Containment Building effluent monitors (Reactor Building and SLCRS Vent), area radiation monitors, meteorological instrumentation, and direct environmental radiation measurements.

Protective actions would be based on the projected dose to the public and to plant personnel.

4.5.3 Accidental Release of Waste Liquid

Accidents have been postulated to occur to components and piping that would result in spillage of waste liquids within the facility. Design features are provided to contain and collect spillage such that there are no offsite consequences.

Initial assessment of this type of accident involves determining the source and the extent of the spillage, and determining area dose rates from area radiation monitors or portable survey instruments. As it is unlikely that there would be offsite consequences, protective actions may involve normal radiological controls and, perhaps, local and plant evacuations.

4.5.4 Accidental Release of Waste Gases

Accidental releases of waste gases are postulated to involve the sudden rupture of the Volume Control Tank (VCT), or the Gas Surge Tank (GST), with subsequent release of the radioactive gas inventories to the environment. It is assumed in this analysis that there is failed fuel (Unit 1-0.11%, Unit 2-1.0%), and that charcoal delay beds would remove essentially all iodines.

Initial assessment of this accident includes the performance of dose projections in accordance with Emergency Implementing Procedures. Dose projections utilize data from the Reactor Containment Building effluent monitors (Reactor Building and SLCRS Vent), meteorological instrumentation, and direct environmental radiation measurements.

4.5.5 Steam Generator Tube Rupture

The steam generator tube rupture accident is postulated to involve the complete severance of a single steam generator tube with the plant at power with failed fuel (Unit 1-0.11%, Unit 2-1.0%). With offsite power available, the affected steam generator can be isolated on the secondary side within approximately 30 minutes and all gaseous activity diverted to the containment upon a high radiation signal from the condenser air ejector vent monitor.

If offsite power is not available, the affected steam generator may still be isolated on the secondary side within approximately 30 minutes; however, volatile activity may be released to the environment via safety valves.

Initial assessment of this accident includes the performance of dose projections in accordance with Emergency Implementing Procedure. Dose projections utilize data from the condenser air ejector monitor and meteorological instrumentation, and/or direct environmental radiation measurements.

4.5.6 Main Steam Line Break Within Containment

The main steam line break accident, within the containment, is postulated to involve the rupture of a main steam line upstream of the main steam isolation valves. It is assumed that there would be a 10 gpm primary to secondary leak, and 1% failed fuel. It is postulated that the release would continue for 8 hours, the period of time necessary for the primary system to reach atmospheric pressure, thereby halting the primary to secondary leak.

Initial assessment of this accident includes the performance of dose projections in accordance with Emergency Implementing Procedures. Dose projections utilize data from Reactor Building and supplementary leak collection and release system vent effluent monitor (atop Reactor Containment Building), meteorological instrumentation, and/or direct environmental radiation measurements.

4.5.7 Main Steam Line Break Outside Containment

This accident is postulated under the same conditions as the main steam line break within containment, except that the steam break occurs downstream of the Main Steam Isolation Valves (MSIV). It is postulated that a release of activity would continue for 5 seconds, the time required for the MSIVs to close.

Due to the short duration and the direct release to the environment, there would be no feasible mechanism to monitor the actual release. An estimate of the resultant doses can be made, however, by comparison of the actual primary to

secondary leak rate and actual percentage of the failed fuel to the values of these parameters used in the accident analysis (10 gpm and 1% respectively) and ratioing the postulated dose accordingly. Dose estimates and corresponding protective actions could be projected on the basis of measurements made in the plant environs. It should be noted that under most meteorological conditions, the short duration of the release would preclude measurements in the environs necessary for implementing protective actions. Because of this, the emergency condition classification system provides action criteria based on plant process parameters rather than radioactive effluent monitors.

4.5.8 Major Rupture of a Main Feedwater Pipe

This accident is postulated to involve the rupture of a main feedwater pipe such that it impairs the ability to supply main feedwater to the steam generator. The accident analysis indicates that the auxiliary feedwater system capacity is sufficient to remove decay heat, to prevent primary system over pressure, and prevent uncovering the core.

4.5.9 Rod Cluster Control Assembly Ejection

This accident postulates the effects of a mechanical failure of a control rod drive mechanism (CRDM) housing resulting in the ejection of a rod cluster control assembly and drive shaft. The consequence of this accident is a rapid reactivity insertion and a small LOCA. The accident analysis postulates that there would be less than 10% fuel failure in the hot channel and that there is no danger of sudden fuel dispersal into the coolant. The accident analysis is limited to the effects of a reactivity insertion. Because of the small LOCA, there is a possibility for an offsite release. See paragraph 4.2.13.

Initial assessment of this accident includes the performance of dose projections in accordance with Emergency Implementing Procedures. Dose projections utilize data from the Reactor Building and supplementary leak collection and Release System Vent effluent monitors (atop Reactor Containment Building), containment area radiation monitors, meteorological instrumentation, and/or direct environmental radiation measurements.

4.5.10 Single Reactor Coolant Pump Locked Rotor

This accident analysis postulates the effects of a rapid reduction in reactor coolant flow resulting in a reactor trip, and core pressure and temperature transient. The accident analysis assumes that the peak reactor coolant pressure and temperature do not result in damage to the fuel or primary coolant boundary.

4.5.11 Complete Loss of Forced Reactor Coolant Flow (pumps coast down)

This accident analysis postulates a complete loss of flow from a loss of all power supplies to all reactor coolant pumps, and which would result in an increase in coolant temperature. Reactor trips would occur on reactor coolant pump power busses, low reactor coolant loop flow, or a pump circuit breaker opening, which would prevent core damage or a release of fission products.

4.5.12 Single RCCA Withdrawal at Full Power

A single RCCA withdrawal may occur in the unlikely event of simultaneous electrical failures, or as a result of operator error. Rod deviation, rod control failure, and rod position indicators and alarms would provide warning to the operator. Because of the localized nature of this condition, the ensuing reactor trip (high temperature) may not occur fast enough to prevent damage in these core location. It is postulated that 5% of the total number of core fuel rods would be subjected to high temperatures.

4.5.13 Loss of Coolant Accident

The loss of coolant accident (LOCA) is defined as a rupture of the reactor coolant system piping. The reactor coolant make-up system is capable of maintaining pressurizer level against an 0.375 inch diameter hole. In the case of breaks up to 1.0 square feet, Safety Injection Systems (SIS), initiated by the decreasing pressurizer pressure, would be capable of maintaining core clad temperature within limits. These two conditions are considered as small LOCAs.

The double ended rupture of the largest pipe in the reactor coolant system, although not expected to take place, is postulated because its consequence would include the potential for the release of significant amounts of radioactive material to the environment. The double ended rupture concurrent with a loss of offsite power and/or failure of one train of the Engineered Safeguards System is the design basis accident (DBA) upon which the engineered safeguards system and the containment were designed.

Initial assessment of this accident includes the performance of dose projections in accordance with Emergency Implementing Procedures. Dose projections utilize data from Reactor Containment Building and Supplementary Leak Collection and Release System Vent effluent monitors (atop Reactor Containment Building), containment area radiation monitors, meteorological instrumentation, or direct radiation measurements in the environment.

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
Tab	Not Applicable
EAL	Not Applicable
Mode	1,2,3,4
Indicator(s)	Not Applicable
Basis	In the section to follow, the bases of the Fission Product Barrier Matrix are presented. The section is divided into two sub-sections. The first provides the bases for each of the 'Potential Loss' and 'Loss' INDICATORs. In this section Unit 1 INDICATORs are provided then followed by the Unit 2 INDICATORs in parentheses. The second sub-section provides the bases for the four CRITERION that apply to the Fission Product Barrier Matrix. (Since the use of the terms INDICATOR and CRITERION will be obvious from the context, the terms will not be capitalized herein.) In reviewing these bases, and in using the matrix for classification, it is important to keep in mind that the indicators should not be viewed as discrete events. There is extensive synergy between the indicators for the three barriers. Some of this is obvious, some is not. For example, consider indicator 1.3.1: "Actions of FR-C.1 (RED PATH) are INEFFECTIVE". One could conclude that such an event represented an Unusual Event (i.e., Potential Loss of Containment Barrier). This would appear to be inconsistent with the similarly worded first indicator for EAL 2.2.G, a General Emergency. However, indicator 1.1.1 considers a Core Cooling CSF RED PATH to be a loss of the Fuel Clad Barrier. This is now two barriers challenged – a Site Area Emergency. Under the RCS Barrier, indicators address loss of subcooling and reactor vessel level. In as much as a Core Cooling CSF RED PATH could not exist without a loss of subcooling or reduced inventory, we would conclude that all three barriers were challenged.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev 2, 1/92 per USNRC Regulatory Guide 1.101

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
Tab 1.1	**FUEL CLAD BARRIER**
EAL 1.1.1	**Critical Safety Function Status**
Mode	1,2,3,4
Indicator(s)	**LOSS:** Core Cooling CSF RED PATH **Potential LOSS** Core Cooling CSF ORANGE PATH OR Heat Sink CSF RED PATH
Basis	**LOSS** The 'Loss' Indicator addresses the condition of inadequate Core Cooling. If the Emergency Operating Procedure CSF status trees indicate a RED PATH the condition must be considered to be an extreme challenge to the safety function needed to ensure protection of the public. A RED PATH terminus for Core Cooling indicates significant superheating and core uncovery and is considered to indicate a 'Loss' of the Fuel Clad Barrier. Clad failure is probable in a very short time period after core uncovery. Core melting will follow if level cannot be restored. **Potential LOSS:** The "Potential Loss" Indicator addresses the condition where an inadequate Core Cooling situation can develop. If the Emergency Operating Procedure status trees indicate an orange path, the conditions must be considered to be a severe challenge to the safety function. Core Cooling CSF ORANGE PATH indicates subcooling has been lost and that some clad damage may occur. Heat Sink CSF RED PATH indicates the heat sink function is under extreme challenge and thus either of these two items indicate a "Potential Loss" of the Fuel Clad Barrier. Either condition would escalate to a 'Loss' if function restoration procedures do not correct the condition.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev 2, 1/92 per USNRC Regulatory Guide 1.101 FR-C.1 Inadequate Core Cooling FR-C.2 Degraded Core Cooling FR-H.1 Loss of Heat Sink

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
TAB 1.1	**FUEL CLAD BARRIER**
EAL 1.1.2	**Five Hottest CETCs (Three Max CETCs)**
Mode	1,2,3,4
Indicator(s)	**LOSS:** Greater Than 1200°F (1200°F) **Potential LOSS** Greater Than 719°F (729°F)
Basis	**LOSS** The "Loss" Indicator uses a reading of 1200°F (1200°F) which corresponds to a Core Cooling CSF RED PATH condition on the EOP status trees. A reading of this magnitude corresponds to significant superheating of the reactor coolant and clad heating which results in a "Loss" of Fuel Clad Barrier. This indicator is intentionally redundant to Indicator 1.1.1 and is included to cover situations in which status tree monitoring has not yet been started. **Potential LOSS:** The "Potential Loss" Indicator uses a reading of 719°F (729°F) which (in conjunction with Indicator 1.1.3) corresponds to a Core cooling CSF ORANGE PATH Condition on the EOP status trees. A reading of this magnitude corresponds to a loss of RCS subcooling. This indicator is intentionally redundant to Indicator 1.1.1 and is included to cover situations in which status tree monitoring has not yet been started. This condition will escalate to a 'Loss' if temperature continues to rise.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev 2, 1/92 per USNRC Regulatory Guide 1.101 FR-C.1 Inadequate Core Cooling FR-C.2 Degraded Core Cooling

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
TAB 1.1	**FUEL CLAD BARRIER**
EAL 1.1.3	**Reactor Vessel Water Level**
Mode	1,2,3,4
Indicator(s)	**LOSS:** Not Applicable **Potential LOSS** VALID RVLIS Full Range Level <40% (40%) (No RCP running)
Basis	**LOSS** There is no "Loss" Indicator corresponding to this item because it is covered by the other Fuel Clad Barrier "Loss" indicators. **Potential LOSS** The "Potential Loss" Indicator is defined by a RVLIS full range indication less than <40% (40%) level with no reactor Coolant pumps running. This corresponds (in conjunction with Indicator 1.1.2) to an Core Cooling CSF RED PATH terminus. This condition indicates that considerable Clad heating and loss of RCS subcooling has occurred. This indicator is intentionally redundant to Indicator 1.1.1 and 1.2.2 and is included to cover situations in which status tree monitoring has not yet been started.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev 2, 1/92 FR-C.2 Degraded Core Cooling

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
TAB 1.1	**FUEL CLAD BARRIER**
EAL 1.1.4	**Primary Coolant Activity Level**
Mode	1,2,3,4
Indicator(s)	**LOSS:** RCS sample activity is Greater Than 300 µCi/gm dose equivalent Iodine-131 **Potential LOSS** Not Applicable
Basis	**LOSS** The "Loss" Indicator addresses the condition of high RCS activity. RCS activity >300 µ Ci/gm is above expected iodine spikes (limited by T/S to 60 µCi/gm), and well above steady state iodine concentrations (limited by T/S to 1 µCi/gm). RCS sample activities greater than this indicate failure of some (approximately 2-5%) fuel cladding. **Potential LOSS** There is no "Potential Loss" Indicator associated with this item. TAB 2.4, 'Fuel Clad Degradation' serves as a precursor to the 'Loss' indicator.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev 2, 1/92

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
TAB 1.1	**FUEL CLAD BARRIER**
EAL 1.1.5	**Letdown Monitor Indication**
Mode	1,2,3,4
Indicator(s)	**LOSS:** RM-CH101 A or B (2CHS-RQ101 A/B) VALID reading greater than 3.5E5 cpm (300uC/ml) with unisolated letdown. **Potential LOSS** Not Applicable
Basis	**LOSS** The "Loss" Indicator addresses the condition of high RCS activity. The reading specified equates to an RCS activity of 300 µCi/gm. This concentration is above expected iodine spikes (limited by T/S to 21 µCi/gm (60 µCi/gm)), and well above steady state iodine concentrations (limited by T/S to 0.35 µCi/gm (1 µCi/gm)). RCS sample activities greater than this indicate failure of some (approximately 2-5%) fuel cladding. This indicator is not applicable if letdown is isolated since the monitor isolates with letdown. As such, this indicator would be useful only in those events (e.g., RCP locked rotor) in which safety injection and containment isolation do not actuate. **Potential LOSS** There is no "Potential Loss" Indicator associated with this item. TAB 2.4, 'Fuel Clad Degradation' serves as a precursor to the 'Loss' indicator.
Escalation	Not Applicable
References	NUMARC/NESP-007, (addition) Rev 2, 1/92

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
TAB 1.1	**FUEL CLAD BARRIER**
EAL 1.1.6	**Containment Radiation Monitors**
Mode	1,2,3,4
Indicator(s)	**LOSS:** VALID reading exceeds: (table of RM-219A/B and RM-201 readings versus time since S/D) 2RMR-RQ202 A/B, 2RMR-RQ206 or 207) **Potential LOSS** Not Applicable
Basis	**LOSS** The monitor readings listed in the table for this indicator are intended to indicate the release of reactor coolant, with elevated activity indicative of fuel damage, into the containment. Thus, this indicator indicates a 'Loss' of the Fuel Clad Barrier and the RCS Barrier. The reading assumes the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 300 μ Ci/gm dose equivalent I-131 into the containment atmosphere. Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations (including iodine spiking) allowed within technical specifications and are therefore indicative of fuel damage (approximately 2 - 5% clad failure depending on core inventory and RCS volume). For the specified concentration, these are worst case assumptions. The existence of VALID monitor readings of these magnitudes is a certain indicator of fuel clad damage. There could, however, be conditions (e.g., high RCS activity with a small RCS leak, gas stratification in CNMT) for which a lower monitor reading would equate to the same amount of fuel damage. Thus, the absence of monitor readings of these magnitudes should not be taken as evidence of Fuel Clad integrity if other indicators of damage are present. **Potential LOSS:** There is no "Potential Loss" Indicator associated with this item. The uncertainties in determining the monitor readings would render the distinction between 'Loss' and 'Potential Loss' meaningless.
Escalation	If the radiation level increases further, indicating about 20% clad damage, the CNMT barrier is considered potentially lost. Since this will result in the loss of two barriers, and the potential loss of the third, a General Emergency is declared.
References	NUMARC/NESP-007, Rev 2, 1/92

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
TAB 1.1	**FUEL CLAD BARRIER**
EAL 1.1.7	**Emergency Director Judgment**
Mode	1,2,3,4
Indicator(s)	Any condition that, in the judgment of the NSS/ED, indicates Loss or Potential Loss of the Fuel Clad Barrier comparable to the conditions listed above.
Basis	This Indicator gives the ED the latitude to use his judgment in determining if the Fuel Clad Barrier is or will be in a "Loss" or "Potential Loss" condition. This situation is usually considered when plant conditions are present that require the monitoring of CSFs or performance of EOP corrective actions. Specific cases where ED judgment may be required are the loss of instrumentation needed to monitor the CSFs and the loss of all AC power. Although the majority of the Indicators provide very specific thresholds, the Emergency Director must remain alert to events or conditions that lead to the conclusion that exceeding the Indicator threshold is imminent. If, in the judgment of the Emergency Director, an imminent situation is at hand with no viable success path available, the classification should be made as if the thresholds have been exceeded. While this is particularly prudent at the higher emergency classes (as the early classification may provide for more effective implementation of protective measures), it is nonetheless applicable to all emergency classes.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev 2, 1/92

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
TAB 1.2	**RCS BARRIER**
EAL 1.2.1	**Critical Safety Function Status**
Mode	1,2,3,4
Indicator(s)	**LOSS** Not Applicable **Potential LOSS** RCS Integrity CSF RED PATH OR Heat Sink CSF RED PATH
Basis	**LOSS** There is no "Loss" Indicator associated with this item. The CSFs related to RCS Barrier, while appropriate as 'Potential Losses', are deemed long-term with regard to an actual loss of the barrier. **Potential LOSS:** The "Potential Loss" Indicator is defined by a RCS Integrity CSF RED PATH or a Heat Sink CSF RED PATH terminus. In the case of RCS Integrity (PTS), consideration is given to a failure of the reactor vessel resulting in a loss of coolant accident (LOCA). Heat Sink is identified since an inability to remove core heat could lead to a vessel or RCS failure. Also, in the case of loss of heat sink, it may become necessary to cool the core by bleed and feed with safety injection. Although this is deliberate action, the open PORV is a breech of the RCS Barrier that would allow fission products to be released to containment.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev 2, 1/92 FR-P.1 Pressurized Thermal Shock FR-H.1 Loss of Heat Sink

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
TAB 1.2	**RCS BARRIER**
EAL 1.2.2	**Reactor Vessel Water Level**
Mode	1,2,3,4
Indicator(s)	**LOSS** VALID RVLIS Full Range level < 40% (40%) (No RCP Running) **Potential LOSS** Not Applicable
Basis	**LOSS** The "Loss" Indicator is defined by RVLIS Full Range level less than 40% (40%) with no RCP's running. A reduction in RCS volume of this magnitude during modes 1, 2, 3, and 4, indicates a significant breech in the RCS Barrier since no intentional valving configuration would result in such a decrease. The inability to maintain reactor vessel water level is the fundamental indication that the RCS Barrier has been lost. **Potential LOSS** There is no "Potential Loss" Indicator associated with this item.
Escalation	Not Applicable
References	NUMARC/NESP-007, (addition) Rev 2, 1/92 FR-C.2 Degraded Core Cooling

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
TAB 1.2	**RCS BARRIER**
EAL *1.2.3*	**RCS Leak Rate**
Mode	1,2,3,4
Indicator(s)	**LOSS** RCS Leak results in Loss of RCS subcooling **Potential LOSS** Unisolatable RCS leak that requires an additional charging pump be started with letdown isolated. OR RCS leak causes safety injection actuation indicated by direct entry into EOP E-1 required by EOP E-0.
Basis	**LOSS** The "Loss" Indicator addresses conditions where leakage from the RCS is greater than available makeup capacity such that a loss of subcooling has occurred. The loss of subcooling is the fundamental indication that the makeup systems are inadequate in maintaining RCS pressure and inventory against the mass loss through the leak. Such a situation would involve a significant breech of the RCS Barrier. **Potential LOSS:** The "Potential Loss" Indicator is based on the inability to maintain normal liquid inventory within the Reactor Coolant System (RCS) by operation of one centrifugal charging pump discharging to the charging header with letdown isolated. This condition would be exceeded by an operator manually starting a second charging pump in response to decreasing RCS volume. It is important to note that the indicator involves an unisolable RCS leak. Starting a second charging pump in response to a RCS volume decrease associated with a main steam line break would not be classified by this indicator (refer to 2.10 Steam/Feed Line Break). The second 'Potential Loss' indicator is similar to the first indicator, but addresses automatic safety injection actuation. The reference to the direct transition from E-0 to E-1 discounts safety injection actuations associated with non-LOCA events.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev 2, 1/92 E-1 Loss of Reactor or Secondary Coolant

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
TAB 1.2	**RCS BARRIER**
EAL 1.2.4	**Primary-to-Secondary Leak**
Mode	1,2,3,4
Indicator(s)	**LOSS** SGTR that results in a safety injection actuation OR Entry into E-3 required by EOPs **Potential LOSS** Not Applicable
Basis	**LOSS** The "Loss" Indicator addresses conditions where a steam generator tube rupture (SGTR) exists and the RCS flow into the steam generator is such that pressurizer level and pressure cannot be maintained. This results in a safety injection actuation. For redundancy, entry into EOP E-3 as required by EOPs is provided as a alternate indicator. This wording precludes a classification if E-3 is optionally referenced during a tube leak. The activation of safety injection represents the threshold rupture size. Smaller leaks will be classified on the basis of Tab 2.6. This "Loss" Indicator in conjunction with the CNMT Barrier "Loss" Indicator #4 addresses the situation where the S/G that is ruptured and also Faulted. This "Loss" of two barriers requires an event classification of Site Area Emergency. This structure inherently recognizes that a SGTR can lead to a failure of two fission product barriers. **Potential LOSS:** There is no "Potential Loss" Indicator associated with this item.
Escalation	Not Applicable
References	NUMARC/NESP-007, (addition) Rev 2, 1/92 E-3 Steam Generator Tube Rupture

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
TAB 1.2	**RCS BARRIER**
EAL 1.2.5	**Containment Radiation Monitors**
Mode	1,2,3,4
Indicator(s)	**LOSS:** VALID reading exceeds: (table of RM-202 and RM-201 (2RMR-RQ201 or 202) readings versus time since S/D **Potential LOSS** Not Applicable
Basis	**LOSS** The monitor readings listed in the table for this indicator are intended to indicate the release of reactor coolant, with normal RCS activity, into the containment. This indicator indicates a 'Loss' of the RCS Barrier. The reading assumes the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with concentration of 0.35 µCi/gm dose equivalent I-131 (i.e., U1/U2 C6 T/S RCS activities) into a containment atmosphere. The release and dispersal assumptions are worst case. The existence of VALID monitor readings of these magnitudes is a certain indicator of RCS leakage. There could, however, be conditions (e.g., high RCS activity with a small RCS leak, gas stratification in CNMT) for which a lower monitor reading would equate to the same amount of leakage. Thus, the absence of monitor readings of these magnitudes should not be taken as evidence of RCS Barrier integrity if other indicators of leakage are present. **Potential LOSS:** There is no "Potential Loss" Indicator associated with this item. The uncertainties in determining the monitor readings would render the distinction between 'Loss' and 'Potential Loss' meaningless.
Escalation	The numeric values for this indicator are less than those specified for the Fuel Clad Barrier in indicator 1.1.6. If the readings increase to the levels specified in indicator 1.1.6, then the Fuel Clad Barrier is also affected.
References	NUMARC/NESP-007, Rev 2, 1/92 *Unit 1 Technical Specification Amendment 205* *Unit 2 Technical Specification Amendment 101* C6

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
TAB 1.2	**RCS BARRIER**
EAL 1.2.6	**Emergency Director Judgment**
Mode	1,2,3,4
Indicator(s)	Any Condition that, in the Judgment of the NSS/ED, indicates Loss or Potential Loss of the RCS Barrier comparable to the conditions Listed Above.
Basis	This Indicator gives the ED the latitude to use his judgment in determining if the RCS Barrier is or will be in a "Loss or Potential Loss" condition. This situation is usually considered when plant conditions are present that require the monitoring of CSFs or performance of EOP corrective actions. Specific cases where ED judgment may be required are the loss of instrumentation needed to monitor the CSFs and the loss of all AC power. Although the majority of the EALs provide very specific threshold, the ED must remain alert to events or conditions that lead to the conclusion that exceeding the EAL threshold is imminent. If, in the judgment of the ED, an imminent situation is at hand with no viable success path available, the classification should be made as if the thresholds have been exceeded. While this is particularly prudent at the higher emergency classes (as the early classification may provide for more effective implementation of protective measures), it is nonetheless applicable to all emergency classes.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev 2, 1/92

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
TAB 1.3	**CNMT BARRIER**
EAL 1.3.1	**Critical Safety Function Status**
Mode	1,2,3,4
Indicator(s)	**LOSS:** Not Applicable **POTENTIAL LOSS:** Containment CSF RED PATH OR Actions of FR-C.1 (RED PATH) are INEFFECTIVE
Basis	**LOSS:** There is no "Loss" Indicator associated with this item since CSF containment monitoring is designed to detect conditions that would fail containment, rather than conditions that indicate that containment has failed. **Potential LOSS:** The first "Potential Loss" Indicator is defined by a RED PATH on the Containment status tree. A RED PATH indicates an extreme challenge to the safety function derived from appropriate instrument readings and/or sampling results, and thus represents a potential loss of CNMT Barrier. Conditions leading to a containment RED PATH result from RCS Barrier and/or Fuel Clad Barrier Loss. Thus, this Indicator is primarily a discriminator between the Site Area Emergency and General Emergency representing a potential loss of the third barrier. The second "Potential Loss" Indicator is defined by a RED PATH on the core cooling status tree with FR-C.1 INEFFECTIVE. In this Indicator, the functional restoration procedures are those emergency operating procedures that address the recovery of the core cooling critical safety functions. The procedure is considered INEFFECTIVE if the temperature is not decreasing or if the vessel water level is not increasing within 15 minutes of implementation. The conditions identified in this potential loss Indicator represent an imminent melt sequence which, if not corrected, could lead to vessel failure and an increased potential for containment failure. In conjunction with the core exit thermocouple Indicators in the Fuel barrier column and the loss of subcooling indicators in RCS Barrier column, this Indicator would result in the declaration of a General Emergency – loss of two barriers and the potential loss of a third. If the functional restoration procedures are INEFFECTIVE, there is no "success" path.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev 2, 1/92 FR-Z.1 High Containment Pressure FR-C.1 Inadequate Core Cooling

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
TAB 1.3	**CNMT BARRIER**
EAL 1.3.2	**Containment Pressure/Hydrogen Concentration**
Mode	1,2,3,4
Indicator(s)	**LOSS:** Rapid unexplained decrease in pressure following initial increase OR Containment pressure or sump level response NOT consistent with LOCA conditions **Potential LOSS** Pressure greater than 45 PSIG OR Containment Hydrogen increases to >4% OR Pressure greater than 8 PSIG (8 PSIG U-2) with less than one full train of containment sprays
Basis	**LOSS** The first "Loss" Indicator addresses a rapid unexplained loss of pressure (i.e., not attributable to containment spray effects) following an initial pressure increase indicating a loss of containment integrity as a result of the event. The second 'Loss' indicator addresses the condition in which containment pressure and sump levels do not increase as a result of the mass and energy release into containment from a LOCA. The lack of pressure increase indicates a pre-incident failure of containment integrity, or a LOCA outside of containment. **Potential LOSS:** The first "Potential Loss" Indicator is identical to the first 'Potential Loss' in indicator 1.3.1, and is included to address situations in which CSF status tree monitoring has not yet begun. The second 'Potential Loss' indicator addresses the existence of an explosive mixture of hydrogen and oxygen in the containment, which if ignited, would be a challenge to the CNMT Barrier. The third "Potential Loss" Indicator represents a potential loss of CNMT Barrier in that the containment heat removal/depressurization system is either lost or performing in a degraded manner, as indicated by containment pressure greater than the cnmt depressurization equipment actuation setpoint, 8 PSIG, (8 PSIG U-2) at which the equipment should have actuated. (Con't)

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
TAB 1.3	**CNMT BARRIER**
EAL 1.3.2	**Containment Pressure/Hydrogen Concentration (Con't)**
Basis: (Con't)	These "Potential Loss" Indicators are primarily discriminators between the Site Area Emergency and General Emergency representing a potential loss of the third barrier.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev 2, 1/92 FR-Z.1 High Containment Pressure

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
TAB 1.3	**CNMT BARRIER**
EAL 1.3.3	**Containment Isolation Status**
Mode	1,2,3,4
Indicator(s)	**LOSS:** Containment Isolation is Incomplete creating a direct release path to the environment when required. [C6] **Potential LOSS** Not Applicable
Basis	**LOSS** The 'Loss' Indicator is intended to address incomplete containment isolation that allows a direct release to the [C6] environment when required. It represents a loss of the CNMT Barrier. **Potential LOSS:** There is no "Potential Loss" indicator associated with this item.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev 2, 1/92

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
TAB 1.3	**CNMT BARRIER**
EAL 1.3.4	**Containment Bypass**
Mode	1,2,3,4
Indicator(s)	**LOSS:** RUPTURED S/G is also FAULTED outside of CNMT OR P-S leakrate >T/S with approx. 4 to 8 hours steam release from affected S/G via-nonisolable MSSV, SGADV, or MSLB outside of CNMT **Potential LOSS:** Unexplained VALID increase in area or ventilation monitors in contiguous areas with known LOCA OR Hi-Hi Alarm on RM-RW-100 A, B, C, or D (HIGH 2SWS-RQ100 A,B,C,D) and affected HX is NOT isolated
Basis	**LOSS:** The first "Loss" Indicator addresses a non-isolable secondary side release from a ruptured steam generator. This allows a direct release of radioactive fission and activation products to the environment, a containment bypass. Note that this condition also meets RCS Barrier indicator 1.2.4. Thus, such an event would be classified as a Site Area Emergency at a minimum. The UFSAR postulates doses exceeding the General Emergency threshold for such an event. However, the UFSAR analysis incorporates several conservative assumptions that are not deemed appropriate in an EAL. Nonetheless, needed escalation to a General Emergency would occur if fuel damage is indicated, or on the basis of dose assessments. The second "Loss" Indicator addresses a prolonged steam release from the secondary side outside of the containment from a steam generator having primary to secondary leakage greater than T/S. This indicator addresses main steam line breaks (MSLB), feedwater line breaks, and failed open relief valves or atmospheric dump valves. The duration of 'prolonged' is left to Emergency Director judgment but should typically be on the order of 4 to 8 hours in duration. It is not the intent of this indicator to address MSLBs downstream of the MSIVs if the MSIVs isolate the break within a short period, or for other similar transient events. Steam releases via the main condenser air ejectors should be declared on the basis of dose assessments rather than the Fission Product Barrier Matrix. The air ejectors should not be considered a prolonged steam release path. (Con't)

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
TAB 1.3	**CNMT BARRIER**
EAL 1.3.4	**Containment Bypass (continued)**
Mode	1,2,3,4
Basis (continued)	**Potential LOSS:** The first "Potential Loss" Indicator addresses an increase in area or ventilation radiation monitors located in areas contiguous to the containment. With a LOCA in progress, such increases could be due to penetration leakage. Other causes for increases could be interfacing system LOCAs involving systems (e.g., LHSI) located in these areas, and leakage from systems recirculating containment sump water. All of these conditions are associated with a 'known LOCA' and are indicative of a potential loss of the CNMT Barrier. Increases in monitor readings without a LOCA should be classified in accordance with TAB 7. The second "Potential Loss" Indicator addresses the situation of a leak in one of the recirculation spray heat exchangers. Such a leak would allow containment sump water to be released to the environment. At Unit 1 background radiation can increase the monitor response. Due to the location of these monitors adjacent to the outer containment wall, background can be expected to increase significantly post-LOCA with core melt. The *Difference* between readings on the four monitors is more significant than the *absolute reading* on any one monitor.
Escalation	Not Applicable
References	NUMARC/NESP-007, (Modification) Rev 2, 1/92 E-2 Faulted Steam Generator Isolation

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
TAB 1.3	**CNMT BARRIER**
EAL 1.3.5	**Significant Radioactivity in Containment**
Mode	1,2,3,4
Indicator(s)	**LOSS:** Not Applicable **Potential LOSS** VALID reading exceeds: (table of RM-219A/B and RM-201 readings versus time since S/D) (2RMR-RQ202, 206, or 207)
Basis	**LOSS** There is no "Loss" Indicator associated with this item. The uncertainties in determining the monitor readings would render the distinction between 'Loss' and 'Potential Loss' meaningless. **Potential LOSS** This reading indicates significant fuel damage well in excess of the indicators associated with both loss of Fuel Clad and loss of RCS Barriers. Thus, if this indicator is met, the indicators for the other two barriers are also met, resulting in a General Emergency declaration. The reading assumes the instantaneous release and dispersal of 20% of the clad inventory of noble gas and iodine into the containment atmosphere. 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 CNMT Barrier, such that a General Emergency declaration is warranted. The 20% clad inventory threshold is based on NUREG-1228, "Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents", which indicates that a major release of radioactivity requiring offsite protective actions from core damage is not likely at fuel failures releasing less than 20% clad inventory from the core into the reactor coolant. It is important to note that containment failures may not be necessary to achieve offsite doses exceeding protective action guides. Depending on meteorological conditions, the amount of core damage, and the containment pressure transient, leakage comparable to the T/S containment leak rate may be sufficient to cause offsite protective actions. The BVPS UFSAR postulated over 250 rem thyroid at the EAB in the first hour of the accident with only T/S leakage.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev 2, 1/92

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
TAB 1.3	**CNMT BARRIER**
EAL 1.3.6	**Emergency Director Judgment**
Mode	1,2,3,4
Indicator(s)	Any condition that, in the judgment of the NSS/ED, indicates Loss or Potential Loss of the CNMT Barrier comparable to the conditions listed above.
Basis	This Indicator gives the ED the latitude to use his/her judgment in determining if the CNMT Barrier is a "Potential Loss" or "Loss". This situation is usually considered when plant conditions are present that require the monitoring of CSFs or performance of EOP corrective actions. Specific cases where ED judgment may be required are the loss of instrumentation needed to monitor the CSFs and the loss of all AC power. Although the majority of the Indicators provide very specific thresholds, the ED must remain alert to events or conditions that lead to the conclusion that exceeding the EAL threshold is imminent. If, in the judgment of the ED, an imminent situation is at hand with no viable success path available, the classification should be made as if the thresholds have been exceeded. While this is particularly prudent at the higher emergency classes (as the early classification may provide for more effective implementation of protective measures), it is nonetheless applicable to all emergency classes.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev 2, 1/92

Section	**FISSION PRODUCT BARRIER MATRIX**
TAB	**Not Applicable**
Classification	**GENERAL EMERGENCY**
Mode	1,2,3,4
Criterion(s)	LOSS of any two barriers and Potential LOSS of third barrier OR LOSS of all three barriers
Basis	Definition: Events are in process or have occurred which involve Actual or Imminent Substantial Core Degradation or Melting with Potential for Loss of Containment integrity. Releases can be reasonably expected to exceed EPA Plume Protective Action Guidelines Exposure Levels outside the EXCLUSION AREA BOUNDARY. The main differentiation between the Site Area and General Emergency classification is whether or not the EPA PAG plume exposure levels are expected to be exceeded outside the site boundary. This threshold, in addition to dynamic dose assessment considerations, addresses NRC and offsite emergency response agency concerns as to timely declaration of a General Emergency. The main objective of the General Emergency is to determine whether evacuation or sheltering of the general public is indicated based on EPA PAGs, and therefore should be interpreted to include radionuclide release regardless of cause. Consideration must be given to failures of systems and or structures that provide fission product barrier integrity which is the primary method of preventing uncontrolled radionuclide releases. In terms of fission product barriers, the loss of two barriers with potential loss of the third barrier constitutes a General Emergency.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev 2, 1/92

Section	**FISSION PRODUCT BARRIER MATRIX**
TAB	**Not Applicable**
Classification	**SITE AREA EMERGENCY**
Mode	1,2,3,4
Criterion(s)	LOSS or Potential LOSS of any two barriers OR LOSS of one barrier and a Potential LOSS of a second barrier
Basis	Definition: Events are in process or have occurred which involve Actual or Likely Major Failures of Plant Functions needed for the Protection of the Public. Any releases are not expected to result in Exposure Levels which Exceed EPA Plume Protective Action Guideline Exposure Levels outside the Exclusion Area Boundary. It is considered to be a challenge to plant functions necessary for the protection of the public if the integrity of any two of the three fission product barriers has or has the potential of being degraded. This approach is more conservative than USNRC Regulatory Guide 1.101 in that the CNMT Barrier is not weighted less significant than the other two barriers. Thus a "Loss" or "Potential Loss" of any two barriers is a Site Area Emergency. This approach also simplifies the Site Area Emergency classification from the fission product barrier matrix.
Escalation	Escalation would be based on Actual or Imminent Substantial Core Degradation
References	NUMARC/NESP-007, (modified) Rev 2, 1/92

Section	**FISSION PRODUCT BARRIER MATRIX**
TAB	**Not Applicable**
Classification	**ALERT**
Mode	1,2,3,4
Criterion(s)	Any LOSS or Potential LOSS of Fuel Clad Barrier OR Any LOSS or Potential LOSS of RCS Barrier
Basis	Definition Events are in process or have occurred which involve an Actual or Potential Substantial Degradation of the Level of Safety of the Plant. Any releases are expected to be limited to small fractions of the EPA Plume Protective Action Guideline Exposure Levels. The "Loss" or "Potential Loss" of either the Fuel Clad Barrier or RCS Barrier is considered to be an actual or potential substantial degradation of the level of safety of the plant. The Alert classification resulting from potential degradation of the fuel clad or RCS integrity also addresses the operation staff's need for help by staffing the Technical Support Center (TSC), independent of whether an actual decrease in plant safety is determined. This increased monitoring can then be used to better determine the actual plant safety state, whether escalation to a higher emergency class is warranted, or whether de-escalation or termination of the emergency class declaration is warranted. Dose consequences from these events are small fractions of the EPA PAG plume exposure levels, i.e., about 10 millirem to 100 millirem. The CNMT Barrier is not addressed at the Alert classification. A challenge of the CNMT Barrier, without a concurrent challenge to either the Fuel Clad or RCS Barriers, is not deemed as significant as a challenge to innermost barriers. A challenge to the CNMT Barrier is addressed as an Unusual Event.
Escalation	Escalation would be based on Actual or Likely Major Failures of Plant Functions needed to Protect the Public.
References	NUMARC/NESP-007, Rev 2, 1/92

Section	**FISSION PRODUCT BARRIER MATRIX**
TAB	**Not Applicable**
Classification	**UNUSUAL EVENT**
Mode	1,2,3,4
Criterion(s)	LOSS or Potential LOSS of Containment Barrier See also EALs 2.4, 2.5, 2.6
Basis	Definition: Unusual Events are in process or have occurred which indicate a Potential Degradation of the Level of Safety of the Plant. No releases of Radioactive Material requiring Offsite Responses or Monitoring are expected unless further degradation of Safety Systems occurs. In these EALs, Unusual Events are treated as precursors to more significant events. TABs 2.4, 2.5, and 2.6 address events that are precursors to the Fuel Clad and RCS Barrier challenges. The 'Potential Loss' or 'Loss' of either the Fuel Clad or RCS Barriers individually is an ALERT. The "Loss or "Potential Loss" of the CNMT Barrier alone is not considered to be substantial degradation of the level of safety of the plant (i.e., ALERT) when the other two fission product barriers are intact. However, since there is a potential for substantial degradation if another condition develops, hence, the Unusual Event classification.
Escalation	Escalation would be based on Actual or Potential Substantial Degradation of the Level of Safety of the Plant.
References	NUMARC/NESP-007, Rev 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
TAB 2.1	**LOSS OF INSTRUMENTATION**
EAL 2.1.S	**SITE AREA EMERGENCY**
Mode	1,2,3,4
Description	**Inability to monitor a SIGNIFICANT TRANSIENT in progress** *(1 and 2 and 3 and 4)* 1. Loss of most (>75%) of annunciators or indications for >15 Minutes 2. **SIGNIFICANT TRANSIENT** in progress 3. Loss of SER and SPDS (deleted for Unit 2) 4. Inability to directly monitor any of the following CSFs: Subcriticality Vessel Integrity Core Cooling Containment Heat Sink
Basis	This EAL is intended to recognize the inability of the control room staff to monitor the plant response to a transient. When the loss of annunciators or Control Room indications is complicated with a significant unplanned power change as well as loss of non-alarming compensatory indications, such as, SPDS and SER (for Unit 1 only), and those Control Room indications needed to monitor Plant Critical Safety Functions, a Site Area Emergency exists. This declaration is prudent since the control room staff cannot monitor safety functions needed for protection of the public. No discrimination between "safety system" and "non-safety system" annunciators is immediately practical. All annunciators are powered from uninterruptable and redundant power supplies. Additionally, the "safety system" annunciators are interspersed throughout the annunciator panels. For these reasons, no separation of annunciator types is made in the EAL. For the purposes of quantification "most" is approximated as greater than 75%. Losses in excess of this indicates an increased risk that a degraded plant condition could go undetected. It is not intended that a detailed count of the instrumentation be performed but only a rough approximation be used to determine the severity of the condition. SIGNIFICANT TRANSIENT involves an UNPLANNED event involving one or more of the following: (1) An automatic turbine runback > 25% thermal reactor power; (2) Electrical load rejection >25% full electrical load; (3) Reactor Trip; or (4) Safety Injection System Activation. Due to the limited number of safety systems in operation during cold shutdown and refueling modes, no initiating conditions are indicated during these modes of operation. (Con't)

Section 2.0	**SYSTEM DEGRADATION**
TAB 2.1	**LOSS OF INSTRUMENTATION**
EAL 2.1.S	**SITE AREA EMERGENCY (continued)**
Mode	1,2,3,4
Basis (Con't)	The (15 minute) time duration was selected to exclude transient or momentary power losses.
Escalation	Escalation will be based on "Fission Product Barrier Matrix".
References	NUMARC/NESP-007, (SS6), Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
TAB 2.1	**LOSS OF INSTRUMENTATION**
EAL 2.1.A	**ALERT**
Mode	1,2,3,4
Description	**UNPLANNED loss of most annunciators or indications for >15 Minutes with either a SIGNIFICANT TRANSIENT in progress or a loss of non-alarming compensatory indications** *(1 and 2 and 3)* 1. **UNPLANNED** loss of most (>75%) annunciators or indications for >15 Minutes 2. NSS judgment that additional personnel (beyond normal shift complement) are required to monitor the safe operation of the unit. 3. *(a or b)* a. SIGNIFICANT TRANSIENT in progress b. Loss of SER and SPDS (delete SER for Unit 2)
Basis	This EAL indicates that a loss of annunciators complicated with either the loss of SPDS and SER (if applicable) or a plant transient indicates a deterioration of the level of plant safety has occurred and an Alert should be declared. Fifteen minutes was selected as a threshold value to exclude momentary power losses or transients. No discrimination between "safety system" and "non-safety system" annunciators is immediately practical. All annunciators are powered from uninterruptable and redundant power supplies. Additionally, the "safety system" annunciators are interspersed throughout the annunciator panels. For these reasons, no separation of annunciator types is made in the EAL. NSS judgment is intended to recognize the need for additional resources and ensure adequate resources are available. SIGNIFICANT TRANSIENT involves an UNPLANNED event involving one or more of the following: (1) An automatic turbine runback > 25% thermal reactor power; (2) Electrical load rejection >25% full electrical load; (3) Reactor Trip; or (4) Safety Injection System Activation. Unplanned loss of annunciators excludes scheduled maintenance and testing activities. For the purposes of quantification "most" is approximated as greater than 75%. Losses in excess of this indicates an increased risk that a degraded plant condition could go undetected. It is not intended that a detailed count of the instrumentation be performed but only a rough approximation be used to determine the severity of the condition. (Con't)

Section 2.0	**SYSTEM DEGRADATION**
TAB 2.1	**LOSS OF INSTRUMENTATION**
EAL 2.1.A	**ALERT (continued)**
Mode	1,2,3,4
Basis (con't)	Due to the limited number of safety systems in operation during cold shutdown and refueling modes, no initiating conditions are indicated during these modes of operation
Escalation	Escalation of this event will be based on the inability of the operating crew to monitor a transient in progress.
References	NUMARC/NESP-007, (SA4), Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
TAB 2.1	**LOSS OF INSTRUMENTATION**
EAL 2.1.U	**UNUSUAL EVENT**
Mode	1,2,3,4
Description	**UNPLANNED loss of most annunciators or indications for >15 Minutes** *(1 and 2)* 1. Unplanned loss of most (>75%) annunciators or indications for >15 Minutes 2. NSS judgment that additional personnel (beyond normal shift complement) are required to monitor the safe operation of the unit.
Basis	For this EAL, if annunciators or indications are partially or completely lost it is still possible to use other systems to indicate plant conditions (e.g., SER or SPDS). However, it is prudent to declare an Unusual Event since there is a greater risk that a degraded condition could go undetected. Fifteen minutes was selected as a threshold value to exclude momentary power losses or transients. For the purposes of quantification "most" is approximated as greater than 75%. Losses in excess of this indicates and increased risk that a degraded plant condition could go undetected. It is not intended that a detailed count of the instrumentation be performed but only a rough approximation be used to determine the severity of the condition. No discrimination between "safety system" and "non-safety system" annunciators is immediately practical. All annunciators are powered from uninterruptable and redundant power supplies. Additionally, the "safety system" annunciators are interspersed throughout the annunciator panels. For these reasons, no separation of annunciator types is made in the EAL. Unplanned loss of annunciators excludes scheduled maintenance and testing activities. NSS judgment is intended to recognize the need for additional resources and ensure adequate resources are available. Due to the limited number of safety system in operation during cold shutdown, refueling and defueled modes, no initiating conditions are indicated during these modes of operation.
Escalation	Escalation of this event would be based on loss of annunciators complicated by the loss of SPDS and plant computer or a transient in progress.
References	NUMARC/NESP-007, (SU3), Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
TAB 2.2	**LOSS OF FUNCTION**
EAL 2.2.G	**GENERAL EMERGENCY**
Mode	1,2,3,4
Description	**Inability to cool the core** *(1 or 2)* 1. Actions of FR-C.1 (RED PATH) are **INEFFECTIVE** 2. *(a and b)* a. Five hottest CETCs (three max CETCs) >1200°F (>1200°F); or CETCs >719°F (>729°F) with no RCPs running and RVLIS full range <40% (<40%). b. Actions taken have NOT resulted in a rising trend in RVLIS level or a dropping trend in core exit thermocouple temperatures within 15 minutes of initiation of restoration actions
Basis	The basis for a General Emergency is redundant to the declaration using the fission product barrier matrix. It is included here to permit rapid assessment of a predominant path through the matrix. Refer to the Fission Product Barrier Matrix basis for additional detail.
Escalation	Not Applicable
References	NUMARC/NESP-007, (FPM-addition),Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
TAB 2.2	**LOSS OF FUNCTION**
EAL 2.2.S	**SITE AREA EMERGENCY**
Mode	1,2,3,4
Description	**Loss of function needed to achieve or maintain hot shutdown** *(1 or 2)* 1. Ops personnel report a CSF status tree RED PATH terminus for core cooling or heat sink exists 2. Five hottest (three max) core exit thermocouples >1200 F; (>1200°F) or core exit thermocouples >719°F (>729°F) with NO RCPs running and RVLIS full range <40% (40%)
Basis	This EAL addresses loss of functions, including core cooling and heat removal required for hot shutdown with the reactor at pressure and temperature. Concerns for reactivity control are appropriately addressed in EAL 2.3 "Failure of Reactor Protection." Under these conditions, there is an actual major failure of a functions intended for protection of the public. Thus, declaration of a Site Area Emergency is warranted. This is also consistent with the Fission Product Barrier Matrix.
Escalation	Escalation will be based on "Fission Product Barrier Matrix" or 2.2.G.
References	NUMARC/NESP-007, (SS4), Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
TAB 2.2	**LOSS OF FUNCTION**
EAL 2.2.A	**ALERT**
Mode	1,2,3,4
Description	**Complete loss of function needed to achieve Cold Shutdown when Shutdown required by Tech Specs** *(1 and 2 and 3)* 1. Loss of decay heat removal capability (RHR, CCR, or RPRW) / (RHS, CCP, SWS) 2. Inability to remove heat via the condenser 3. Shutdown to mode 5 required by T/S
Basis	For this EAL the inability to achieve Cold Shutdown when it is required, refers to unplanned actions, equipment malfunctions or operator error that prevents achievement of Cold Shutdown This condition could result from a loss of RHR capability, service water to the RHR, heat exchange or equipment failure with the RHR system or AC/DC power loss to the RHR and or reactor plant river water components (i.e., CCR, RPRW) The combination of this and the loss of ;the secondary heat sink to the condenser for cooldown indicates a degradation of the level of plant safety and warrants the declaration of an Alert. This is more serious than the concern expressed for a shutdown in excess of shutdown action statement time requirements within 2.7.U. In this situation attainment of cold shutdown (Mode 5) is more than delayed, it is currently not obtainable.
Escalation	Escalation of this event would be based on complete loss of functions needed to achieve or maintain Hot Shutdown.
References	NUMARC/NESP-007, (SA3-modified)

Section 2.0	**SYSTEM DEGRADATION**
TAB 2.2	**LOSS OF FUNCTION**
EAL 2.2.U	**UNUSUAL EVENT**
Mode	All
Description	**UNPLANNED loss of communication** *(1 or 2)* 1. In-plant *(a and b and c)* a. **UNPLANNED** loss of all PAX phones b. **UNPLANNED** loss of all Gaitronics (Page/Party) c. **UNPLANNED** loss of all Radios (handie-Talkies) 2. Offsite *(a and b and c)* a. **UNPLANNED** loss of ENS b. **UNPLANNED** loss of Bell Lines c. **UNPLANNED** loss of Radios to Offsite
Basis	The purpose of this EAL is to recognize a loss of communications capability that either defeats the plant operations staff's ability to perform routine tasks necessary for plant operations or the ability to communicate problems with offsite authorities. Onsite communications loss must encompass the loss of all means of routine communications (i.e., phones, page party system and radio/walkie talkies). The loss of offsite communications ability is expected to be significantly more comprehensive than those addressed by 10 CFR 50.72. Offsite communications loss must encompass the loss of all means of communications with offsite authorities. This EAL is intended to be used only when extraordinary means are being utilized to make communications possible (i.e., individuals being sent to offsite locations to establish communications).
Escalation	Escalation of this event will involve the loss of other plant functions.
References	NUMARC/NESP-007, (SU6), Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
TAB 2.3	**FAILURE OF RX PROTECTION**
EAL 2.3.G	**GENERAL EMERGENCY**
Mode	1,2
Description	**Rx power >5% after VALID trip signal(s) and loss of core cooling capability** *(1 and 2)* 1. Ops personnel report FR-S.1 has been entered and subsequent actions do NOT result in a reduction of power to <5% and decreasing 2. *(a or b)* a. Ops personnel report CSF status tree **RED PATH** terminus exists for core cooling or heat sink b. Five hottest core exit thermocouples (three max) >1200 F (>1200 F); or five hottest core exit thermocouples (three max) >719°F (729°F) with NO RCPs running and RVLIS full range <40% (40%)
Basis	Under the conditions of this EAL, the efforts to bring the reactor to less than five percent power 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. FR-S.1 lists actions intended to shutdown the reactor. This includes actions in the control room and in other areas of the plant. FR-S.1 is utilized within the EAL to discriminate between those situations in which immediate manual reactor trip was not possible from the control room. The BVPS Unit 1 control room has two trip control locations on the main control board. Both are within immediate access for the reactor operator. If both fail to result in a reactor trip EOP E-0 directs the operator to FR-S.1. There are additional capabilities (i.e., emergency boration) to bring the plant under control. The indication of a Core Cooling Red is used to indicate these capabilities are not effective. The existence of inadequate core cooling thus indicates that sufficient heat is not being removed from the core., which is a core melt sequence. Similarly, the challenge to the Steam Generators in the early stages of the event (i.e., RED PATH terminus for Heat Sink) indicates insufficient feed water flow to remove heat and is a precursor for a core melt sequence. In either situation, if these challenges exist at a time that the reactor has not been brought below 5% power, core degradation can occur rapidly and a core melt sequence is considered to exist. For this reason, the General Emergency declaration is intended to be consistent with the Fission Product Barrier Matrix declaration to permit maximum offsite intervention time.
Escalation	Not Applicable
References	NUMARC/NESP-007, SG2, Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
TAB 2.3	**FAILURE OF RX PROTECTION**
EAL 2.3.S	**SITE AREA EMERGENCY**
Mode	1,2
Description	**Reactor trip failure after VALID Trip signal(s) with reactor power >5% and attempts to cause a manual trip from the control room are unsuccessful.** *(1)* 1. Ops personnel report FR-S.1 has been entered and manual reactor trip from the control room did not result in reduction of power to <5% and decreasing
Basis	This EAL indicates a failure of the automatic and control room manual signals to trip the reactor with reactor power above 5%. 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 EAL may be viewed as anticipatory to the Fission Product Barrier Degradation EAL, its inclusion is necessary to better assure timely recognition and emergency response. FR-S.1 lists actions intended to shutdown the reactor. This includes actions in the control room and in other areas of the plant. FR-S.1 is utilized within the EAL to discriminate between those situations in which immediate manual reactor trip was not possible from the control room. The BVPS Unit 1 control room has two trip control locations on the main control board. Both are within immediate access for the reactor operator. If both fail to result in a reactor trip EOP E-0 directs the operator to FR-S.1.
Escalation	Escalation of this event would be based on the inability to trip the RX and indications of Heat Sink Red or Core Cooling Red.
References	NUMARC/NESP-007, (SS2), Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
TAB 2.3	**FAILURE OF RX PROTECTION**
EAL 2.3.A	**ALERT**
Mode	1,2
Description	**Automatic reactor trip did not occur after VALID trip signal and manual trip from the control room was successful** *(1 and 2* 1. **VALID** reactor trip signal received or required 2. Manual reactor trip from the control room was successful and power is <5% and decreasing
Basis	This EAL indicates failure of the Reactor Protection System (RPS) to automatically trip the reactor. This condition is a potential degradation of a safety system in that a primary front line automatic protection system did not function in response to a plant transient or condition requiring system actuation. This is an immediate threat to the fuel clad barrier. The declaration of an Alert will increase plant staff awareness of an RPS failure and expedite the post trip review which ensures a comprehensive and systematic investigation of the cause of the failure, verification of fuel clad status, and subsequent equipment repairs. This is consistent with the definition of an Alert.
Escalation	Escalation of this event would be based on the reactor power not being reduced to less than five percent by actions of FR-S.1 or via the Fission Product Barrier Matrix.
References	NUMARC/NESP-007, (SA2 - Deviation), Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
TAB 2.3	**FAILURE OF RX PROTECTION**
EAL 2.3.U	**UNUSUAL EVENT**
Mode	1,2
Description	Not Applicable
Basis	Not Applicable
Escalation	Not Applicable.
References	Not Applicable

Section 2.0	**SYSTEM DEGRADATION**
TAB 2.4	**FUEL CLAD DEGRADATION**
EAL 2.4.U	**UNUSUAL EVENT**
Mode	1,2,3,4,5
Description	**Reactor Coolant System specific activity exceeds LCO (Refer to BVPS technical specification 3.4.8)** *(1 or 2* 1. **VALID** high alarm on RM-CH-101A or B (2CHS-RQ101 A/B) reactor coolant letdown monitor 2. Radiochemistry analysis exceeds Technical Specification 3.4.8
Basis	This EAL is included as an Unusual Event since it indicates a potential degradation in the level of safety of the plant and a potential precursor to more serious problems. This level of cladding degradation is escalated via the Fission Product Barrier Matrix, so no escalation exists within TAB 2.4. INDICATOR #1 addresses the high alarm on CVCS letdown liquid which would provide indication of the loss of fuel clad integrity. This permits rapid indication of the need for additional assessment/confirmation of the monitors validity. It is not intended to require full sample analysis. INDICATOR #2 addresses the results of coolant sample analysis that may not be preceded by a high alarm. In both cases, the level is intended to be higher than the activity expected as the result of an Iodine spike resulting from a routine transient. The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) accident. The LCO contains specific activity limits for both Dose Equivalent I-131 and gross specific activity. The allowable levels are intended to limit the 2-hour dose at the site boundary to a small fraction of the 10 CFR 100 dose guideline values.
Escalation	Escalation will be based on "Fission Product Barrier Matrix".
References	NUMARC/NESP-007, (SU4), Rev. 2, 1/92 T.S. 3.4.8 RCS Specific Activity Unit 1 Technical Specification Amendment 205 Unit 2 Technical Specification Amendment 101 C6

Section 2.0	**SYSTEM DEGRADATION**
TAB 2.5	**RCS UNIDENTIFIED LEAKAGE**
EAL 2.5.U	**UNUSUAL EVENT**
Mode	1,2,3,4,5 (Applies to Mode 5 if RCS Pressurized)
Description	**Unidentified or pressure boundary RCS leakage >10 GPM** 1. Unidentified or pressure boundary leakage (per T/S) >10 GPM as indicated below *(a or b)* a. OST 1.6.2 results (2.6.2A) b. With RCS temp. and PZR level stable, VCT level dropping at a Rate >10 GPM (>1%/min indicated on LI-CH-115 (2CHS-LI115) with no VCT makeup in progress)
Basis	This EAL is included as an Unusual Event because it may be a precursor of more serious conditions and, as a result, is considered to be 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 and it is above the value associated with the Technical Specification required shutdown. This is consistent with the definition of the Unusual Event. . Only operating modes in which there is fuel in the reactor coolant system and the system is pressurized are specified. An additional annotation is included for Mode 5 to clarify this consideration.
Escalation	Escalation will be based on "Fission Product Barrier Matrix".
References	NUMARC/NESP-007, (SU5 - Modification), Rev. 2, 1/92 T.S. 3.4.6.2 RCS Operational Leakage T.S. Definitions 1.14a and b OST-1.62

Section 2.0	**SYSTEM DEGRADATION**
TAB 2.6	**RCS IDENTIFIED LEAKAGE**
EAL 2.6.U	**UNUSUAL EVENT**
Mode	1,2,3,4,5 Applies to Mode 5 if RCS Pressurized
Description	**Identified RCS leakage >25 GPM** 1. Identified RCS leakage (as defined by Technical Specifications) >25 GPM *(a or b)* a. OST 1.6.2 or 1.6.2.A (2.6.2 or 2.6.2A) results b. **UNPLANNED** level rise in excess of 25 GPM total into PRT, DG-TK-1, and DG-TK-2 / (PRT, 2DGS-TK-21 and 2DGS-TK-22)
Basis	This EAL is included as an Unusual Event because it may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. The 25 gpm value for the identified leakage was selected as it is observable with normal control room indications and it is above the value associated with the Technical Specification required shutdown. The threshold for this EAL is set at a higher value than unidentified leakage due to the reduced significance of identified leakage. This is true since the leakage is collected and of known quantity. Only operating modes in which there is fuel in the reactor coolant system and the system is pressurized are specified. An additional annotation is included for Mode 5 to clarify this consideration.
Escalation	Escalation will be based on "Fission Product Barrier Matrix".
References	NUMARC/NESP-007, (SU5 - Modified), Rev. 2, 1/92 T.S. 3.4.6.2 RCS Operational Leakage T.S. Definitions 1.14a and b

Section 2.0	**SYSTEM DEGRADATION**
TAB 2.7	**TECHNICAL SPECIFICATION**
EAL 2.7.U	**UNUSUAL EVENT**
Mode	1,2,3,4
Description	**Inability to reach required Shutdown within Technical Specification limits** *(1 and 2)* 1. A Technical Specification action statement, requiring a mode reduction, has been entered 2. The unit has NOT been placed in the required mode within the time prescribed by the action statement
Basis	Limiting Conditions of Operation (LCO) action statements require the plant to be brought to a required shutdown mode when the Technical Specification required configuration cannot be restored within an appropriate time frame. Specific time durations are included to permit an orderly shutdown of the unit to progress in these circumstances. The initiation of plant shutdown required by the site Technical Specifications requires a one hour report under 10 CFR 50.72 (b) Non-emergency events. The plant is within its safety envelope when being shut down within the allowable action statement time in the Technical Specifications. An immediate declaration of an Unusual Event is required when the plant is not or will not, for whatever reason, be brought to the required operating mode within the allowable action statement time in the Technical Specifications.
Escalation	Not Applicable
References	NUMARC/NESP-007, (SU2), Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
TAB 2.8	**SAFETY LIMIT**
EAL 2.8.U	**UNUSUAL EVENT**
Mode	1,2,3,4,5
Description	**Safety Limit has been Exceeded** *(1 or 2)* 1. The combination of thermal power, RCS temperature, and RCS pressure greater than safety limit as determined from BVPS Technical Specifications Figure 2.1-1 "Reactor Core Safety Limit" 2. RCS/pressurizer pressure exceeds safety limit (>2735 psig)
Basis	This EAL considers concerns with exceeding specified safety limits. The restrictions of these safety limits prevent overheating of the fuel and cladding, as well as possible cladding perforation that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady-state peak linear heat rate (LHR) below the level at which centerline fuel melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding-surface temperature is slightly above the coolant-saturation temperature. Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat-transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding-water (zirconium-water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant. It is intended that this escalation be recognized via the Fission Product Barrier Matrix. This EAL is consistent with the definition of an Unusual Event as a potential precursor to fission product barrier degradation and thus warrants the classification.
Escalation	Not Applicable
References	NUMARC/NESP-007, (SU2 - Addition), Rev 2 1/92

Section 2.0	**SYSTEM DEGRADATION**
TAB 2.9	**TURBINE FAILURE**
EAL 2.9.A	**ALERT**
Mode	1,2,3
Description	**Turbine failure generated missiles cause penetration of a missile shield wall of any area containing safety related equipment** 1. Plant personnel report missiles generated by turbine failure with casing penetration also results in a through-wall penetration of a missile shield wall listed in Table 5-2
Basis	This EAL is intended to address the threat to safety related equipment imposed by missiles generated by main turbine rotating component failures. Shield walls are incorporated into the design of the areas of concern. To permit a rapid assessment of the potential for damage to safety related equipment, an assessment of these shield walls is appropriate. If no through wall penetration is observed, equipment should not be jeopardized. The list of areas provided includes all areas containing safety-related equipment, their controls, and their power supplies. This EAL is, therefore, consistent with the definition of an ALERT. Unit 1 Table 5-2 Plant Areas Associated With Shield Wall Penetration EAL Control Room — Electrical Switchgear — Safeguards 1WT-TK-10 — Diesel Generator Bldg — Cable Tray Mezz Containment — Primary Aux. Building Unit 2 Plant Areas Associated With Shield Wall Penetration EAL Main Steam Valve Room — 2FWE-TK210 Diesel Generator Bldg — Containment Service Bldg. 745' and 760' — Primary Aux. Building Emergency Switchgear 730
Escalation	Escalation of this event will be based on "Fission Product Barrier Matrix".
References	NUMARC/NESP-007, (HA1 example #6), Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
TAB 2.9	**TURBINE FAILURE**
EAL 2.9.U	**UNUSUAL EVENT**
Mode	1,2,3
Description	**Turbine failure results in casing penetration** 1. Plant personnel report a turbine failure which results in penetration of the turbine casing or damage to main generator seals (with evidence of significant hydrogen or seal oil leakage)
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 main turbine generator. Of major concern is the potential for damage to non-safety related equipment or the leakage of combustible fluids, lubricating oils and gases (hydrogen) to the plant environs. Actual fires and flammable gas build up are appropriately classified via other events. 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	Escalation of this event would be based on potential damage done by turbine PROJECTILES to safety related equipment.
References	NUMARC/NESP-007, (HU1 example # 6), Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
TAB 2.10	**STEAM/FEED LINE BREAK**
EAL 2.10.U	**UNUSUAL EVENT**
Mode	1,2,3,4
Description	**UNPLANNED rapid depressurization of the Main Steam System resulting in a rapid RCS cooldown and Safety Injection initiation** *(1 and 2)* 1. Ops personnel report rapid depressurization of Main Steam System that causes SLI (<510 psig) (SLI <500 PSIG) 2. Ops personnel report Safety injection has actuated
Basis	For this EAL a rapid depressurization could be caused by a Main Steam line break or feed line break which results in rapid RCS cool down and safety injection. This EAL is therefore consistent with the definition of an Unusual Event and warrants declaration whether SLI and/or SI are initiated by automatic or manual initiation in response to the depressurization. UNPLANNED is included in the EAL to preclude the declaration of an emergency as a result of planned maintenance activities.
Escalation	Escalation of this event will be based on "Fission Product Barrier Matrix".
References	NUMARC/NESP-007, (HU5), Rev. 2, 1/92

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Section 3.0	**LOSS OF POWER**
Tab 3.1	**LOSS OF AC (Power Ops)**
EAL 3.1.G	**GENERAL EMERGENCY**
Mode	1,2,3,4
Description	**Prolonged loss of offsite and onsite AC Power** *[1 and 2]* 1. AE and DF 4KV buses not energized from Unit 1 (2) source for >15 minutes 2. *[a or b or c]* a. Ops personnel report CSF status tree **RED PATH** or **ORANGE PATH** terminus exists for core cooling b. Restoration of either AE or DF 4KV bus is not likely from any source within 3 hours of loss c. Five hottest core exit thermocouples (three max) >1200 F (>1200 F); or five hottest core exit thermocouples (three max) >719°F (>729 F) with NO RCPs running and RVLIS full range<40% (<40%)
Basis	Loss of all AC power compromises all plant safety systems requiring electric power including ECCS, Containment Depressurization, and Containment Heat Removal. Prolonged loss of all AC power will lead to loss of fuel clad, RCS, and containment. This is due to the inability to add inventory to the RCS. Additionally, inventory is lost from the RCS at an increasing rate via the reactor coolant pump seals. Loss of AC is defined in INDICATOR #1 identically to ECA 0.0, as both emergency buses de-energized. This permits achieving this EAL even though offsite power may be available to the normal 4KV buses. This is appropriate, since the charging pumps are powered only from the emergency buses. The 15 minute time duration, selected to exclude transient or momentary power losses, allows for re-energization within a timely manner if the normal buses remain energized. INDICATOR #2 considers three indications of event degradation. Both a. and c. include concern for actual indication of degrading core cooling capability. This is placed at the CSF RED or ORANGE PATH terminus for Core Cooling. This is appropriate and consistent with the Fission Product Barrier Matrix, without an allowance for 15 minutes of response in FR-C.1. This too, is appropriate since no AC power exists in this event to take actions in FR-C.1. The three hours to restore AC power allotted by INDICATOR #2.b., was based on a site blackout coping analysis of 4 hours performed in conformance with 10 CFR 50.63 and Regulatory Guide 1.155, "Station Blackout." An appropriate allowance of one hour is included for the initiation of offsite emergency response. It is intended that the 4 hour time designation be used as a default value. While analysis indicates there is reason to believe that core cooling can be adequately maintained for several (3) hours, real time indications may indicate that this is not true. Although this EAL is redundant to the Fission Product Barrier Degradation it 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 (Con't)

Section 3.0	**LOSS OF POWER**
TAB 3.1	**LOSS OF AC (Power Ops)**
EAL 3.1.G	**GENERAL EMERGENCY (continued)**
Mode	1,2,3,4
Basis (Con't)	reasonable assessment of the event trajectory. This permits time to initiate offsite intervention actions. It is also noteworthy, that under these conditions, fission product barrier monitoring capability may be degraded. Manual electrical cross-tie capability should be considered to constitute restoration of a single emergency power supply and eliminate the necessity to declare a General Emergency due to the 3 hour time allotment in 2.b. Monitoring for and manual operation of equipment is necessary to avoid inadequate core cooling situations. This, too, prevents the necessity to declare a General Emergency due to the constraints of 2a. and 2c.
Escalation	Not Applicable
Reference	NUMARC/NESP-007, (SG1), Rev 2, 1/92

Section 3.0	**LOSS OF POWER**
TAB 3.1	**LOSS OF AC (Power Ops)**
EAL 3.1.S	**SITE AREA EMERGENCY**
Mode	1,2,3,4
Description	**Loss of offsite and onsite AC power for >15 Minutes** 1. AE and DF 4KV buses not energized from Unit 1 (2) source for >15 minutes
Basis	The Loss of all AC power compromises all plant safety systems requiring electric power including ECCS, Containment Depressurization, and Containment Heat Removal. Prolonged loss of all AC power will cause core uncovering and loss of containment integrity. This is due to the inability to add inventory to the RCS. Additionally, inventory is lost from the RCS at an increasing rate via the reactor coolant pump seals. Loss of AC is defined in INDICATOR #1 identically to ECA 0.0, as both emergency buses de-energized. This permits achieving this EAL even though offsite power may be available to the normal 4KV buses. This is appropriate, since the charging pumps are powered only from the emergency buses. The 15 minute time duration, selected to exclude transient or momentary power losses, allows for re-energization within a timely manner if the normal buses remain energized. The AC power tie-line between Unit 1 and Unit 2 is not credited as a source of onsite power in this EAL as the need to power the safety systems in the affected unit from the companion unit is deemed to represent major failures of functions necessary for the protection of the public -- a Site Area Emergency. The configuration of the tie-line is such that it cannot be placed in operation within 15 minutes. The tie-line could, however, maintain CSFs and prevent an escalation to a General Emergency.
Escalation	Prolonged loss of all offsite power and prolonged loss of all onsite power will, when combined with inadequate core cooling, result in an escalation of this event.
References	NUMARC/NESP-007 (SS1), Rev. 2, 1/92

Section 3.0	**LOSS OF POWER**
TAB 3.1	**LOSS OF AC (Power Ops)**
EAL 3.1.A	**ALERT**
Mode	1,2,3,4
Description	**AC power to emergency buses reduced to a single source of power such that any additional failure will result in the de-energization of both buses** *[1 and 2]* 1. Either AE or DF 4KV bus is de-energized for >15 minutes 2. The energized AE or DF 4KV bus has only one source of power *[a or b]* a. Emergency diesel generator b. 1A or 1D 4KV normal bus (2A or 2D)
Basis	The condition indicated by this EAL is the degradation of the offsite and onsite power systems such that any additional single failure would result in a station blackout. This condition could occur due to a loss of offsite power with a concurrent failure of one emergency diesel generator to supply power to its emergency busses. The (15 minute) time duration was selected to exclude transient or momentary power losses. INDICATOR #2 includes the four normal means of supplying power to the two emergency buses. The loss of any three of the four constitutes this INDICATOR and thus the Alert declaration.
Escalation	Prolonged Loss of all offsite power and prolonged Loss of all onsite power will escalate this event.
References	NUMARC/NESP-007,(SA5), Rev. 2, 1/92

Section 3.0	**LOSS OF POWER**
TAB 3.1	**LOSS OF AC (Power Ops)**
EAL 3.1.U	**UNUSUAL EVENT**
Mode	1,2,3,4
Description	**Loss of Offsite Power for >15 Minutes** *[1 and 2]* 1. 1A and 1D (2A or 2D) 4KV normal buses de-energized for >15 minutes 2. Each diesel generator is supplying power to its respective emergency bus
Basis	Prolonged loss of offsite AC power reduces required redundancy to the class 1E electrical distribution system 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). This is consistent with the definition of an Unusual Event. Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.
Escalation	Loss of one additional power supply to the shutdown boards will escalate this event.
References	NUMARC/NESP-007 (SU1), Rev. 2, 1/92

Section 3.0	**LOSS OF POWER**
TAB 3.2	**LOSS OF AC (Shutdown)**
EAL 3.2.A	**Alert**
Mode	5,6, defuel
Description	**UNPLANNED loss of offsite and onsite AC power for >15 minutes** 1. AE and DF 4KV buses not energized from Unit 1 (2) source for >15 minutes
Basis	A loss of all AC power compromises all plant safety systems that require AC power including RHR, spent fuel pool cooling, and the river water systems. At modes 1-4, this event would be classified as Site Area Emergency. A lower classification is justified here due to the reduced decay heat. 15 minutes is specified so as to exclude momentary power losses. Note however, that this event is bounded by EAL 6.2.S if the loss continues such that core boiling has or will uncover fuel in the reactor vessel, a Site Area Emergency would be declared. INDICATOR #1 encompasses the CRITERION in that the AE and DF buses are fed from either offsite or onsite sources. Thus, having both buses de-energized indicates a failure of both sources. This EAL is intentionally redundant to 6.3 Loss of AC (Shutdown).
Escalation	Escalation would occur if the RCS temperature increased above 200°F due to a loss of RHR caused by the loss of power
References	NUMARC/NESP-007 (SA1), Rev 2, 1/92,

Section 3.0	**LOSS OF POWER**
TAB 3.2	**LOSS OF AC (Shutdown)**
EAL 3.2.U	**Unusual Event**
Mode	5,6, defuel
Description	**UNPLANNED loss of offsite AC power for >15 minutes** *(1 and 2)* 1. 1A and 1D (2A or 2D) 4KV buses de-energized for >15 minutes 2. Either diesel generator is supplying power to its respective emergency bus
Basis	A prolonged loss of offsite AC power reduces power source redundancy and potentially degrades the level of safety of the plant by rendering the plant more vulnerable to a complete loss of AC power. 15 minutes is specified so as to exclude momentary power losses. This EAL is similar to EAL 6.2.U, except that the phrase UNPLANNED was added to exclude classifications that could result from offsite power bus outages scheduled and controlled by maintenance work activities. INDICATOR #1 are the buses that would be de-energized in the event of a loss of offsite power. INDICATOR #2 establishes that at least one train of onsite power is available. This EAL is intentionally redundant to 6.3 Loss of AC (Shutdown).
Escalation	Escalation would occur if onsite AC power was lost.
References	NUMARC/NESP-007 (SU1), Rev 2, 1/92

Section 3.0	**LOSS OF POWER**
TAB 3.3	**LOSS OF DC**
EAL 3.3.S	**SITE AREA EMERGENCY**
Mode	1,2,3,4
Description	**Loss of all vital DC Power for >15 minutes** 1. Voltage <110.4 VDC on DC buses 1-1 and 1-2 and 1-3 and 1-4 (2-1 and 2-2 and 2-3 and 2-4) for >15 minutes Also Refer to the "Fission Product Barrier Matrix", "Loss of Function", and "Loss of Instrumentation" and "Loss of Shutdown Systems"
Basis	Loss of all DC power compromises the ability to monitor and control plant safety functions. Prolonged loss of all DC power will cause core uncovering and loss of containment integrity when there is significant decay heat and sensible heat in the reactor system. Fifteen minutes is specified to exclude momentary power losses. In INDICATOR #1, the specified voltage is the minimum voltage specified in the UFSAR at which DC loads will perform reliably.
Escalation	Escalation would occur through the Fission Product Barrier Matrix Degradation or Loss or Function
References	NUMARC/NESP-007, (SS3), Rev. 2, 1/92

Section 3.0	**LOSS OF POWER**
TAB 3.3	**LOSS OF DC**
EAL 3.3.U	**UNUSUAL EVENT**
Mode	1,2,3,4
Description	**UNPLANNED Loss of one Train of DC power for >15 Minutes** *[1 or 2]* 1. Voltage <110.4 VDC on DC Buses 1-1 and 1-3 (2-1 and 2-3) for >15 Minutes 2. Voltage <110.4 VDC on DC Buses 1-2 and 1-4 (2-2 and 2-4) for >15 Minutes
Basis	The purpose of this EAL is to recognize a loss of DC power compromising the ability to monitor and control the plant. This EAL is in addition to the concerns for loss of annunciation or indication identified in EAL 2.1. The loss of one train of DC power while operating in modes 1,2,3 or 4 is consistent with the definition of an Unusual Event for BVPS. The 110.4 volt Bus Voltage is the minimum bus voltage necessary for the operation of safety related equipment. This voltage value should incorporate a margin of at least 15 minutes of operation before the onset of inability to operate those loads. The fifteen minute threshold is utilized to exclude a transient or momentary power losses.
Escalation	The event will escalate if indications are lost and a transient occurs per 2.1.S
References	NUMARC/NESP-007, (SU7 - addition), Rev. 2, 1/92

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Section 4.0	**HAZARDS AND ED JUDGMENT**
TAB	**ALL**
EAL	Not applicable
Mode	Not applicable
Description	Not applicable
Basis	*This discussion applies generically to all EALs in Section 4* TAB 4.7 provides the generic definitions for the four emergency classifications. All of the specific EALs were developed to correspond to these four definitions. The Emergency Director may find these definition useful in classifying an event that isn't adequately addressed by a specific EAL. The other TABs in this section address events that have the potential to affect plant operations. In this section, generally it is the event and its potential for impact on the operation of the plant that is addressed. As a general protocol, UNUSUAL EVENTS are categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern. Areas identified in the EALs define the location of the event based on the potential for damage of equipment contained therein. Depending on the event, the magnitude is established on the basis of the duration of the event (*e.g., FIRE lasting longer than 15 minutes*) or on other definable values (*e.g., flammable gas exceeding 25% LEL*). Escalation to an ALERT generally occurs when the magnitude of the event is sufficient to result in damage to the equipment contained in the specified location. In these cases, the reference to damage of systems is used to identify the magnitude of the event. References to areas and systems are used to locate the event in areas where the event could lead to a substantial degradation in the level of safety of the plant. The significance here is not that a particular system was degraded, but rather, the event was of sufficient magnitude to cause this degradation. The system malfunction that might have occurred is addressed by EALs in other sections Escalation to a SITE AREA EMERGENCY occurs when the system damage is sufficient enough to represent a loss of a function necessary for the protection of the public. This typically occurs based on EALs in other sections (e.g., fission product matrix, system malfunction). EALs for SITE AREA EMERGENCY are provided in this section for some events deemed significant enough to warrant an anticipatory declaration. There are two GENERAL EMERGENCY EALs provided in this section. These address events significant enough to cause concern regarding core melt sequences or loss of control of the plant. They are classified in this section to provide for an anticipatory declaration and offsite protective actions.
Escalation	Not applicable
References	Not applicable

Section 4.0	**HAZARDS AND ED JUDGMENT**
TAB 4.1	**FIRE**
EAL 4.1.G	**GENERAL EMERGENCY**
Mode	1,2,3,4
Description	**FIRE in the Instrument and Relay Room (CB-1), Cable Spreading Room (CB-2), Control Room (CB-3), West [C6] Communications Room (CB-6), or Cable Tunnel (CB-1) resulting in an evacuation of the control room per 1.56C.4 (2.56C.A) *"Alternate Safe Shutdown"* and loss of any required equipment resulting in an uncontrolled RCS heatup.** *(1 and 2 and 3)* 1. 1.56C.4 (2.56C.4) *"Alternate Safe Shutdown"* entered 2. Ops personnel report inability to operate at least one of each (any) of the following components of the available train (equipment required by 2.56C.4): Unit 1 Charging Pump — AFW pump — Diesel generator RPRW pump — BIP — Steam relief path Unit 2 2CHS-P21A — 2CCP-P21A EGS-EG2-1 — 2FWE-P23A & 2FWE-P22 2SAS-C21A — Alternate S/D Panel 2SWS-P21A — 2RHS-P21A Black D/G 3. Uncontrolled RCS heatup lasting longer than 15 minutes.

Section 4.0	**HAZARDS AND ED JUDGMENT**
TAB 4.1	**FIRE**
EAL 4.1.G	**GENERAL EMERGENCY (Con't)**
Mode	1,2,3,4
Basis	See generic bases at the beginning of this section. The EAL considers the degradation associated with the implementation of OM 1(2).56C.4 "Alternate Safe Shutdown". The procedure is designed to permit a small operating crew to shutdown and cooldown the unit without the use of the control room or alternate shutdown panel (Unit 2 Areas: Instrument and Relay Room (CB-1), Cable Spreading Room [C6] (CB-2), Control Room (CB-3), West Communications Room (CB-6), or Cable tunnel (CB-1)). The procedure is entered when there is a fire in the control room, cable tray mezzanine, or process control room. These areas carry cabling and equipment controls that can affect safety systems significantly. The cable separation is such that a fire in any one of these areas will not eliminate both trains of equipment capability. To achieve unit shutdown and cooldown without fire induced spurious activations and failures, only select components of a single available train are utilized. This intentionally reduces the normal redundancy of safety related equipment and thus necessitates that all equipment identified operate as required. INDICATOR #2 recognizes that if one of the components performing each of the identified functions is not operating properly, plant control cannot be ensured. For the Unit 1 charging and reactor plant river water systems this can be accomplished with the available train pump or the swing "C" pump. For the AFW (FWE) system this can be accomplished by the use of the available motor driven pump or the turbine driven pump. Any available steam path is acceptable, (atmospheric dump valves or residual heat release valve). The loss of this equipment under these conditions will lead to a core melt sequence. INDICATOR #3 is included to recognize the RCS heatup toward a core melt sequence and prevent an overly conservative declaration due to momentary losses of equipment functions. When the loss of functions leads to an uncontrolled heatup the situation constitutes a General Emergency.
Escalation	Not Applicable
References	NUMARC/NESP-007 (addition consistent w/ HG2) Rev. 2, 1/92 OM 1.56C.4

Section 4.0	**HAZARDS AND ED JUDGMENT**
TAB 4.1	**FIRE**
EAL 4.1.S	**SITE AREA EMERGENCY**
Mode	1,2,3,4
Description	**FIRE in the Instrument and Relay Room (CB-1), Cable Spreading Room (CB-2), Control Room (CB-3), West [C6] Communications Room (CB-6), or Cable Tunnel (CB-1) resulting in an evacuation of the control room per 1.56C.4 (2.56C.4)"Alternate Safe Shutdown"** 1. 1.56C.4 (2.56C.4) "Alternate Safe Shutdown" entered
Basis	See generic bases at the beginning of this section. The EAL considers the degradation associated with the implementation of OM 1.56C.4 "Alternate Safe Shutdown". The procedure is designed to permit a small operating crew to shutdown and cooldown the unit without the use of the control room or alternate shutdown panel(Unit 2 Areas: Instrument and Relay Room (CB-1), Cable Spreading Room (CB-2), Control Room (CB-3), West Communications Room (CB-6), or Cable tunnel (CB-1)). (Unit 2 Areas: Instrument and [C6] Relay Room (CB-1), Cable Spreading Room (CB-2), Control Room (CB-3), West Communications Room (CB-6), or Cable tunnel (CB-1)).. The procedure is entered when there is a fire in the control room, cable tray mezzanine, or process control room. These areas carry cabling and equipment controls that can affect safety systems significantly. The cable separation is such that a fire in any one of these areas will not eliminate both trains of equipment capability. To achieve this unit shutdown and cooldown without fire induced spurious activations and failures, only select components of a single available train are utilized. This intentionally reduces the normal redundancy of safety related equipment. This reduction in available equipment coupled with the fire in progress and the limitations associated with instrumentation constitutes a Site Area Emergency.
Escalation	Escalation would be based on 4.1.G due to loss of necessary equipment to perform OM 1.56C.4
References	NUMARC/NESP-007 (addition consistent w/ HS2) Rev. 2, 1/92 OM 1.56C.4

Section 4.0	**HAZARDS AND ED JUDGMENT**
TAB 4.1	**FIRE**
EAL 4.1.A	**ALERT**
Mode	All
Description	**FIRE in any of the areas listed in Table 4-1 that is affecting safety related equipment** *(1 and 2)* 1. **FIRE** in any of the areas listed in Table 4-1 2. *(a or b)* a. Ops personnel report **VISIBLE DAMAGE** to permanent structure or equipment in specified area due to **FIRE** b. Control Room indication of degraded system or component (within specified areas) response due to **FIRE**
Basis	See generic bases at the beginning of this section. Fires that are likely to affect the plant's safety systems represent a degraded plant condition. The fire may have damaged equipment or damage is likely due to the proximity of heat, or flame to the systems required for safe shutdown. The likelihood of damage is subjective but is based on fire location, intensity and duration without performance of a detailed damage assessment prior to classification. The determination of the safety and supporting systems necessary for safe shutdown during the applicable operating mode and the assessment of the impact of the fire on the performance of those systems will be determined by the Emergency Director. For this reason, no time duration is designated to quantify the fire. This EAL is predicated on the existence and magnitude of the fire, not on the loss of equipment due to the fire. This is due to a desire to avoid reliance on an extensive damage assessment and to recognize the timely concern for hidden damage. Verification of the fire requires evidence of VISIBLE DAMAGE or degradation of system or component performance. This is included in INDICATORs #2a. and b. This acts to quantify the fire. In all cases, verification should be accomplished within 15 minutes. The verification of a containment fire alarm (with containment subatmospheric) should be through the reset of the alarm at the local panel. If this fails, the use of equipment response degradation addition to redundant area fire alarms and/or containment temperature indications should be used. Unit 1 Table 4-1 Plant Structures Associated with Fire and Explosion EALs Control Room — AE/DF Switchgear — U1/U2 Cable Tunnel (CV3) Cable Tray Mezzanine — Demin Water (1WT-TK-10) — D/G Fuel Oil (Con't)

Section 4.0	**HAZARDS AND ED JUDGMENT**
TAB 4.1	**FIRE**
EAL 4.1.A	**ALERT (Con't)**
Mode	All
Basis (con't)	Process Control Room Relay Room Rod Drive/MG set Room RWST (1QS-TK-1) RW Valve Pit Containment Building Primary Auxiliary Building Safeguards Building Diesel Generator Room Fuel Building Intake Structure Cubicles CO2 Stor./PG Pump Room Unit 2 Control Room Emer. Switchgear W. Comm. Rm 707' Penetrations Area Diesel Gen. Bldgs. Intake Structure Cub. Rod Control Cable Vault Bldg. Relay Room Cbl Spreading Room 725' Service Bldg. Cable Tunnel 735' PAB Containment Bldg. Inst. and Relay Rm. 707' Safeguards Bldg. Cable Tunnel 712'. Main Stm Valve Rm. Fuel Bldg. U1/U2 Cable Tunnel (CV-3) ERF Substation & ERF Diesel Bldg. FIRE is combustion characterized by heat and light. Source of smoke such as slipping drive belts or overheated electrical components do not constitute fires. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed. VISIBLE DAMAGE is damage to equipment that is readily observable without measurements, testing, or analyses. Damage is sufficient enough to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering. Surface blemishes (e.g., paint chipping, scratches) should NOT be included.
Escalation	Escalation would be based on Fission Product Barrier Matrix or Control Room Evacuation
References	NUMARC/NESP-007, (HA2), Rev. 2, 1/92 Figure 4-A Protected Area and Site Perimeter

Section 4.0	**HAZARDS AND ED JUDGMENT**
TAB 4.1	**FIRE**
EAL 4.1.U	**UNUSUAL EVENT**
Mode	All
Description	**FIRE in or adjacent to those areas listed in Table 4-1 not extinguished within the 15 minutes from the time of control room notification or verification of control room alarm**

Basis

See generic bases at the beginning of this section.

This EAL addresses confirmed fires that occur in selected areas of the plant that house safety systems It also covers verified fires outside of these areas that may impact structures that contain safety systems due to the proximity of the fire. In either case these fires may be potentially significant precursors to damage of safety systems or may impact structures that contain safety systems. The initiating condition excludes fires that occur outside these key buildings, such as the warehouses, or other small fires that do not potentially affect safety systems. The 15 minute time limit has been established to exclude small fires that can be controlled by the Emergency Squad resources. This EAL is predicated on the existence and magnitude of the fire, not on the loss of equipment due to the fire. This is due to a desire to avoid reliance on an extensive damage assessment and to recognize the timely concern for hidden damage.

Verification of the fire in this EAL is either by direct communication with plant personnel confirming that a fire exists or the action taken by the Control Room personnel to determine that a fire annunciator received in the Control Room is not due to a spurious signal. Implicit in this is the need for timely verification of the alarm. In all cases, verification should be accomplished within 15 minutes. The verification of a containment fire alarm (with containment subatmospheric) should be through the reset of the alarm at the local panel. If this fails, additional area fire alarms and/or containment temperature indications should be used.

Unit 1

Table 4-1 Plant Structures Associated with Fire and Explosion EALs

Control Room	AE/DF Switchgear	U1/U2 Cable Tunnel (CV3)
Cable Tray Mezzanine	Demin Water (1WT-TK-10)	D/G Fuel Oil
Process Control Room	RW Valve Pit	Diesel Generator Room

(Con't)

Section 4.0	**HAZARDS AND ED JUDGMENT**
TAB 4.1	**FIRE**
EAL 4.1.U	**UNUSUAL EVENT**
Mode	All
Basis (con't)	Relay Room Rod Drive/MG set Room RWST (1QS-TK-1) Containment Building Primary Auxiliary Building Safeguards Building Fuel Building Intake Structure Cubicles CO2 Stor./PG Pump Room Unit 2 Control Room Emer. Switchgear W. Comm. Rm 707' Penetrations Area Diesel Gen. Bldgs. Intake Structure Cub. Rod Control Cable Vault Bldg. Relay Room Cbl Spreading Room 725' Service Bldg. Cable Tunnel 735' PAB Containment Bldg. Inst. and Relay Rm. 707' Safeguards Bldg. Cable Tunnel 712'. Main Stm Valve Rm. Fuel Bldg. U1/U2 Cable Tunnel (CV3) ERF Substation & ERF Diesel Bldg. FIRE is combustion characterized by heat and light. Source of smoke such as slipping drive belts or overheated electrical components do not constitute fires. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.
Escalation	Escalation of this event is based on the Fire affecting plant safety related equipment required to establish or maintain safe shutdown.
References	NUMARC/NESP-007, (HU2-addition), Rev. 2, 1/92

Section 4.0	**HAZARDS AND ED JUDGMENT**
TAB 4.2	**EXPLOSIONS**
EAL 4.2.A	**ALERT**
Mode	All
Description	**EXPLOSION in any of the areas listed in Table 4-1 that is affecting safety related equipment** *(1 and 2)* 1. **EXPLOSION** in any of the areas listed in Table 4-1 2. *(a or b)* a. Ops personnel report **VISIBLE DAMAGE** to permanent structure or equipment in specified area b. Control Room indication of degraded system or component (within listed areas) response due to the **EXPLOSION**
Basis	See generic bases at the beginning of this section. EXPLOSIONS include those that are of sufficient magnitude to damage permanent structures or equipment within the plant vital area. As used here, an EXPLOSION is a rapid, violent, unconfined combustion, or a catastrophic failure of pressurized equipment, that potentially imparts significant energy to near-by structures and material. VISIBLE DAMAGE is damage to equipment that is readily observable without measurements, testing, or analyses. Damage is sufficient enough to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering. Surface blemishes (e.g., paint chipping, scratches) should NOT be included. The "Report of VISIBLE DAMAGE" should not be interpreted as requiring a lengthy damage assessment prior to classification. The observation of damage to a structure is sufficient to make a declaration. The declaration of the Alert and the activation of the TSC is warranted and will provide the Emergency Director with resources necessary to perform damage assessment. Unit 1 Table 4-1 Plant Structures Associated with Fire and Explosion EALs Control Room — AE/DF Switchgear — U1/U2 Cable Tunnel (CV3) Cable Tray Mezzanine — Demin Water (1WT-TK-10) — D/G Fuel Oil Process Control Room — RW Valve Pit — Diesel Generator Room
Escalation	Escalation will be based on "Fission Product Barrier Matrix".
References	NUMARC/NESP-007, (HA2), Rev 2, 1/92

Section 4.0	**HAZARDS AND ED JUDGMENT**
TAB 4.2	**EXPLOSIONS**
EAL 4.2.A	**ALERT (Con't)**
Mode	All
Description	**EXPLOSION in any of the areas listed in Table 4-1 that is affecting safety related equipment** *(1 and 2)* 1. **EXPLOSION** in any of the areas listed in Table 4-1 2. *(a or b)* a. Ops personnel report **VISIBLE DAMAGE** to permanent structure or equipment in specified area b. Control Room indication of degraded system or component (within specified areas) response due to the **EXPLOSION**

Basis (Con't)

Relay Room	Containment Building	Fuel Building
Rod Drive/MG set Room	Primary Auxiliary Building	Intake Structure Cubicles
RWST (1QS-TK-1)	Safeguards Building	CO2 Stor./PG Pump Room

Unit 2

Control Room	Relay Room	Inst. and Relay Rm. 707'
Emer. Switchgear	Cbl Spreading Room 725'	Safeguards Bldg.
W. Comm. Rm 707'	Service Bldg.	Cable Tunnel 712'.
Penetrations Area	Cable Tunnel 735'	Main Stm Valve Rm.
Diesel Gen. Bldgs.	PAB	Fuel Bldg.
Intake Structure Cub.	Containment Bldg.	U1/U2 Cable Tunnel (CV3)
Rod Control Cable Vault Bldg.		ERF Substation & ERF Diesel Bldg.

Escalation	Escalation will be based on "Fission Product Barrier Matrix".
References	NUMARC/NESP-007, (HA2), Rev 2, 1/92

Section 4.0	**HAZARDS AND ED JUDGMENT**
TAB 4.2	**EXPLOSIONS**
EAL 4.2.U	**UNUSUAL EVENT**
Mode	All
Description	**UNPLANNED EXPLOSION in areas adjacent to those areas listed in Table 4-1** 1. **UNPLANNED EXPLOSION** in or adjacent to those areas listed in Table 4-1
Basis	See generic bases at the beginning of this section. This EAL considers explosions in areas adjacent to the areas listed in Table 4-1. This is consistent with the Unusual Event definition. Unit 1 Table 4-1 Plant Structures Associated with Fire and Explosion EALs Control Room Cable Tray Mezzanine Process Control Room Relay Room Rod Drive/MG set Room RWST (1QS-TK-1) AE/DF Switchgear Demin Water (1WT-TK-10) RW Valve Pit Containment Building Primary Auxiliary Building Safeguards Building U1/U2 Cable Tunnel (CV3) D/G Fuel Oil Diesel Generator Room Fuel Building Intake Structure Cubicles CO2 Stor/PG Pump Room Unit 2 Control Room Emer. Switchgear W. Comm. Rm 707' Penetrations Area Diesel Gen. Bldgs. Intake Structure Cub. Rod Control Cable Vault Bldg. Relay Room Cbl Spreading Room 725' Service Bldg. Cable Tunnel 735' PAB Containment Bldg. Inst. and Relay Rm. 707' Safeguards Bldg. Cable Tunnel 712'. Main Stm Valve Rm. Fuel Bldg. U1/U2 Cable Tunnel (CV3) ERF Substation & ERF Diesel Bldg. (Con't)
Escalation	Escalation of this event would be based on EXPLOSION damage to a structure or equipment causing a degradation in the performance of equipment.
References	NUMARC/NESP-007, (HU2), Rev 2, 1/92

Section 4.0	**HAZARDS AND ED JUDGMENT**
TAB 4.2	**EXPLOSIONS**
EAL 4.2.U	**UNUSUAL EVENT (Con't)**
Mode	All
Description	**UNPLANNED EXPLOSION in areas adjacent to those areas listed in Table 4-1** 1. **UNPLANNED EXPLOSION** in or adjacent to those areas listed in Table 4-1
Basis (Con't)	See generic bases at the beginning of this section. As used here, an EXPLOSION is a rapid, violent, unconfined combustion, or a catastrophic failure of pressurized equipment, that potentially imparts significant energy to near-by structures and material. For this event classification, the occurrence of the EXPLOSION is sufficient to make the declaration without making a lengthy assessment of the damage. UNPLANNED is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities.
Escalation	Escalation of this event would be based on EXPLOSION damage to a structure or equipment causing a degradation in the performance of equipment.
References	NUMARC/NESP-007, (HU2), Rev 2, 1/92

Section 4.0	**HAZARDS AND ED JUDGMENT**
TAB 4.3	**FLAMMABLE GAS**
EAL 4.3.A	**ALERT**
Mode	All
Description	**Release of flammable gas within a facility structure containing safety related equipment or associated with power production.** 1. Plant personnel report the average of three readings taken in an approximately10ft triangular area is > 25% LEL (Lower Explosive Limit) within any building listed in Table 4-2
Basis	See generic bases at the beginning of this section. Report or detection of flammable gases within plant vital structures in concentrations that are approaching the lower explosive limit is a degradation of the level of safety of the plant and warrants the declaration of an Alert. The potential for substantial equipment damage exists with the ignition of such a gas concentration. Table 4-2 Plant Structures Associated with Toxic or Flammable Gas EALs Unit 1 Containment Bldg Safeguards Bldg Primary Aux. Bldg Fuel Handling Bldg Gaseous Waste Valve Room CO2 Storage/PG Pump Room Turbine Building Demin. Water Sto. (1WT-TK-10) Water Treatment Building Main Intake Structure Diesel Generator Building Service Bldg. (incl FW Reg Vlv Rm) Unit 2 Control Building* Emer. Switchgear Service Bldg. Penetrations Area Diesel Gen. Bldgs. Pri Intake Structure Rod Control Cable Vault Bldg. (incl. MSVR) Fuel Handling Bldg. Safeguards Bldg. PAB Containment Bldg. Demin. Water Sto (2FWE-TK210) U1/U2 Cable Tunnel (CV3) Turbine Bldg. RWST (2QSS-TK21) A 10ft triangular area was chosen to ensure any reading obtained was representative of the general area concentration. This prevents a declaration due to a reading very near the source of a minor gas leak
Escalation	Escalation will be based on "Fission Product Barrier Matrix".
References	NUMARC/NESP-007, HA3, Rev 2, 1/92

Section 4.0	**HAZARDS AND ED JUDGMENT**
TAB 4.3	**FLAMMABLE GAS**
EAL 4.3.U	**UNUSUAL EVENT**
Mode	All
Description	*(A or B)* **A. UNPLANNED release of flammable gas within the SITE PERIMETER.** 1. Plant personnel report the average of three readings taken in an approximately 10ft triangular area is > 25% **LEL** (Lower Explosive Limit) within the **SITE PERIMETER** (Refer to Figure 4-A) **B. Confirmed report by local, county, or state officials That an offsite flammable gas release has occurred within one mile of the site with potential to enter the SITE PERIMETER in concentrations >25% of LEL (Refer to Figure 4-A & 4-B)**
Basis	See generic bases at the beginning of this section. Two EALs are specified to account for the potential source of flammable gas being either onsite or offsite. Report or detection of flammable gases in concentrations within the site or near the site that will affect the health of plant personnel or affect the safe operation of the plant (i.e., tanker truck accident releasing flammable gases, etc.) constitutes an Unusual Event. EAL A. acts to support EAL B. in the event that an offsite situation is not reported as having the capacity to affect conditions onsite. Unplanned is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities. SITE PERIMETER encompasses all owner controlled areas in the immediate site environs as shown on Figure 4-B. Additionally, a one mile radius is included with distinctive landmarks to aid in determining location relative to the site.
Escalation	Escalation is based on flammable gases entering a plant area that jeopardizes safety related equipment or power production.
References	NUMARC/NESP-007, (HU3), Rev 2, 1/92 Figure 4-B One Mile Radius/Site Perimeter

Section 4.0	**HAZARDS AND ED JUDGMENT**
TAB 4.4	**TOXIC GAS**
EAL 4.4.A	**ALERT**
Mode	All
Description	**Release of TOXIC GAS within a facility structure which prohibits safe operation of systems required to establish or maintain cold S/D** (1 and 2) 1. Plant personnel report **TOXIC GAS** within any building listed in Table 4-2 2. Plant personnel would be unable to perform actions necessary to establish and maintain cold shutdown while utilizing appropriate personnel protection equipment
Basis	See generic bases at the beginning of this section. Report or detection of toxic gases within plant vital structures in concentrations that are life threatening to plant personnel and affect the ability to achieve or maintain the plant in a cold shutdown condition is a degradation of the level of safety of the plant and warrants the declaration of an Alert. Allowance is made for the use of protective equipment in INDICATOR #2. If such equipment is unavailable or ineffective and access to the area is required for station shutdown to mode 5, the declaration should be made. Table 4-2 Plant Structures Associated with Toxic or Flammable Gas EALs Unit 1 Containment Bldg — Gaseous Waste Valve Room — Main Intake Structure Safeguards Bldg — CO2 Storage/PG Pump Room — Diesel Generator Building Primary Aux. Bldg — Turbine Building Service Bldg. (incl FW Reg Vlv Rm) Fuel Handling Bldg — Demin. Water Sto. (1WT-TK-10) — Water Treatment Building Unit 2 Control Bldg* — Fuel Handling Bldg. — Turbine Bldg. Emer. Swgr — Safeguards Bldg. — RWST (2QSS-TK21) Service Bldg. — PAB Penetrations Area — Containment Bldg. Diesel Gen. Bldgs. — Demin. Water Sto (2FWE-TK210) Pri Intake Structure — U1/U2 Cable Tunnel (CV3) Rod Control Cable Vault Bldg. (incl. MSVR) TOXIC GAS is a gas that is dangerous to life or health by reason of inhalation or skin contact (e.g., chlorine).
Escalation	Escalation will be based on "Fission Product Barrier Matrix".
References	NUMARC/NESP-007, (HA2), Rev 2, 1/92 Figure 4-B One Mile Radius/Site Perimeter

Section 4.0	**HAZARDS AND ED JUDGMENT**
TAB 4.4	**TOXIC GAS**
EAL 4.4.U	**UNUSUAL EVENT**
Mode	All
Description	(A or B) **A. Normal operation of the plant impeded due to access restrictions caused by UNPLANNED TOXIC GAS concentrations within a facility structure listed in Table 4-2** OR **B. Confirmed report by local, county, or state officials that an offsite TOXIC GAS release has occurred within one mile of the site with potential to enter the SITE PERIMETER in concentrations > than the Lower Toxicity Limit (LTL) (Refer to Figure 4-A & 4-B)** *Refer to AOP 1/2.44A.1 "Chlorine/Toxic Gas Release", Attachment 4 for a list of chemicals stored, produced, or transported near BVPS and their toxicity limits*
Basis	See generic bases at the beginning of this section. Report or detection of a release of toxic gases in concentrations within the site or near the site perimeter that will affect the health of plant personnel or that could lead to an effect on the safe operation of the plant (i.e., tanker truck accident releasing toxic gases, etc.) constitutes an Unusual Event. TOXIC GAS is a gas that is dangerous to life or health by reason of inhalation or skin contact (e.g., chlorine). SITE PERIMETER encompasses all owner controlled areas in the immediate site environs as shown on Figure 4-A. Additionally, a one mile radius is included with distinctive landmarks to aid in determining location relative to the site. Table 4-2 Plant Structures Associated with Toxic or Flammable Gas EALs Unit 1 Containment Bldg Gaseous Waste Valve Room Main Intake Structure Safeguards Bldg CO2 Storage/PG Pump RoomDiesel Generator Building
Escalation	Escalation to this event will be based on toxic gases entering a plant area that jeopardizes life or impacts cold shutdown capability
References	NUMARC/NESP-007, HU3, Rev 2, 1/92 DOT Emergency Response Guide for Hazardous Materials Figure 4-B One Mile Radius/Site Perimeter

Section 4.0	**HAZARDS AND ED JUDGMENT**
TAB 4.4	**TOXIC GAS**
EAL 4.4.U	**UNUSUAL EVENT (Con't)**
Mode	All
Description	(A or B) **A. Normal operation of the plant impeded due to access restrictions caused by UNPLANNED TOXIC GAS concentrations within a facility structure listed in Table 4-2** OR **B. Confirmed report by local, county, or state officials that an offsite TOXIC GAS release has occurred within one mile of the site with potential to enter the SITE PERIMETER in concentrations > than the Lower Toxicity Limit (LTL) (Refer to Figure 4-A & 4-B)** *Refer to AOP 1/2.44A.1 "Chlorine/Toxic Gas Release", Attachment 4 for a list of chemicals stored, produced, or transported near BVPS and their toxicity limits*
Basis (Con't)	Primary Aux. Bldg Fuel Handling Bldg Turbine Building Demin. Water Sto. (1WT-TK-10) Service Bldg. (incl FW Reg Vlv Rm) Water Treatment Building Unit 2 Control Bldg* Emer. Swgr Service Bldg. Penetrations Area Diesel Gen. Bldgs. Pri Intake Structure Fuel Handling Bldg. Safeguards Bldg. PAB Containment Bldg. Demin. Water Sto (2FWE-TK210) U1/U2 Cable Tunnel (CV3) Turbine Bldg. RWST (2QSS-TK21) Rod Control Cable Vault Bldg. (Incl. MSVR)
Escalation	Escalation to this event will be based on toxic gases entering a plant area that jeopardizes life or impacts cold shutdown capability
References	NUMARC/NESP-007, HU3, Rev 2, 1/92 DOT Emergency Response Guide for Hazardous Materials Figure 4-B One Mile Radius/Site Perimeter

Section 4.0	**HAZARDS AND ED JUDGMENT**
TAB 4.5	**CONTROL ROOM EVACUATION**
EAL 4.5.S	**SITE AREA EMERGENCY**
Mode	All
Description	**Evacuation of the control room has been initiated and control of all necessary equipment has not been established within 15 minutes of manning the Shutdown Panel** *(1 and 2)* 1. AOP-1.33.1 (2.33.1A)*"Control Room Inaccessibility"* has been entered 2. Inability to transfer any single component listed in Table 4-3 within 15 minutes of manning the shutdown panel
Basis	Evacuation of the control room and relocation to the shutdown panel results in a significant reduction in available instrumentation and control. INDICATOR #1 considers the evacuation of the control room through the entry into AOP 1.33.1 (2.33.1A) "Control Room Inaccessibility". INDICATOR #2 further considers the inability to control specified pieces of equipment that are intended to protect the Critical Safety Functions and fission product barriers. Each of these equipment items is redundant, with the exception of FCV-1CH-122, (2CHS*FCV122) and it is only intended that one of the redundant train pieces of equipment be transferred and under operator control to meet the requirement for the INDICATOR. If transfer of these safety system components has not been performed in an expeditious manner protection of the CSFs and fission product barriers is reduced. This condition warrants the declaration of a Site Area Emergency. Table 4-3 Equipment Required at Shutdown Panel includes: One Auxiliary Feedwater Pump — One Boric Acid Pump(and boration valve) One Atmospheric Steam Dump — FCV-1CH-122 One Charging Pump — (2CHS*FCV122) The 15 minute time limit for transfer of control is based on a reasonable time period for personnel to leave the control room, arrive at the Shutdown Panel area, and reestablish plant control to preclude core uncovery and/or core damage per AOP 1.33.1 (2.33.1A) "Control Room Inaccessibility".
Escalation	Escalation will be based on "Fission Product Barrier Matrix".
References	NUMARC/NESP-007, (HS2), Rev 2, 1/92 AOP 1.33.1 "Control Room Inaccessibility"

Section 4.0	**HAZARDS AND ED JUDGMENT**
TAB 4.5	**CONTROL ROOM EVACUATION**
EAL 4.5.A	**ALERT**
Mode	All
Description	**Evacuation of the control room is required** 1. AOP 1.33.1 (2.33.1A) "Control Room Inaccessibility" has been entered
Basis	Evacuation of the control room and relocation to the shutdown panel results in a significant reduction in available instrumentation and control. INDICATOR #1 considers the evacuation of the control room through the entry into AOP 1.33.1 (2.33.1A) "Control Room Inaccessibility". This is consistent with the definition of an Alert. Additionally, support from the Technical Support Center is advisable.
Escalation	Escalation of this event would be based on the inability to establish plant control from outside the Control Room within 15 minutes.
References	NUMARC/NESP-007, (HA5), Rev 2, 1/92 AOP 1.33.1 "Control Room Inaccessibility

Section 4.0	**HAZARDS AND ED JUDGMENT**
TAB 4.6	**SECURITY**
EAL 4.6.G	**GENERAL EMERGENCY**
Mode	All
Description	**Security event resulting in loss of control of the systems necessary to establish or maintain cold shutdown** *(1 or 2)* 1. Hostile armed force has taken control of the control room or the remote shutdown panel 2. Hostile armed force has taken control of plant equipment such that Ops personnel report the inability to operate equipment necessary to maintain the following functions *(a or b or c):* a. Subcriticality b. Core Cooling c. Heat Sink
Basis	This event represents a condition where a hostile force has taken control of the Control Room or vital areas within the plant that are required to reach and maintain a cold shutdown. This loss could be due to physical loss of control or by the damage of essential equipment. This situation leaves the plant in a very unstable condition with a high potential of multiple barrier failures. Further degradation remains a possibility and can lead rapidly to a core melt sequence. The declaration permits time for offsite intervention as deemed appropriate and permits additional resources to be focused on the site problems. No separation is afforded to permit avoiding the declaration of a General emergency based on the location of the transfer switches at the shutdown panel.
Escalation	Not Applicable
References	NUMARC/NESP-007, (HG1), Rev 2, 1/92

Section 4.0	**HAZARDS AND ED JUDGMENT**
TAB 4.6	**SECURITY**
EAL 4.6.S	**SITE AREA EMERGENCY**
Mode	All
Description	**Security event has or is occurring which results in actual or likely failures of plant functions needed to protect the public** *(1 or 2)* 1. **VITAL AREA**, other than the control room, has been penetrated by a hostile armed force 2. Suspected **BOMB** detonates within a **VITAL AREA.**
Basis	This event represents a significant threat to the safety of the plant since there has been a hostile intrusion into the areas of the plant that contain equipment important to maintaining the plant in a safe condition. A confirmed security event is satisfied when physical evidence of a hostile intrusion exist. VITAL AREA is any area within the PROTECTED AREA which contains equipment, systems, devices, or material, the failure, destruction, or release of which could directly or indirectly endanger the public health and safety by exposure to radiation.
Escalation	Escalation of this event would be based on loss of plant control, (control room or remote shutdown panel).
References	NUMARC/NESP-007, (HS1), Rev 2, 1/92

Section 4.0	**HAZARDS AND ED JUDGMENT**
TAB 4.6	**SECURITY**
EAL 4.6.A	**ALERT**
Mode	All
Description	**Security event which indicates an actual or potential substantial degradation in the level of safety of the plant** *(1 or 2 or 3)* 1. **BOMB** discovered within a **VITAL AREA** 2. **CIVIL DISTURBANCE** ongoing within the **PROTECTED AREA** 3. **PROTECTED AREA** has been penetrated By a hostile armed force Refer to Figure 4-A for a drawing of PROTECTED AREA
Basis	These class of Security events represent a threat to the level of safety of the plant. A confirmed report is satisfied if physical evidence supporting the hostile intrusion or Bomb is discovered in the specified area. The identification of a bomb within a VITAL AREA is designated as an Alert. This is consistent with the explosion EAL, in that the BOMB creates a potential for safety degradation. This should escalate to a Site Area Emergency if the BOMB detonates within a VITAL AREA. BOMB refers to an explosive device. A CIVIL DISTURBANCE exists when there is a group of ten (10) or more persons violently protesting station operations or activities at the site. PROTECTED AREA encompasses all owner controlled areas within the security protected area fence as shown on Figure 4-A.
Escalation	Escalation of this event would be based on hostile intrusion into plant vital areas.
References	NUMARC/NESP-007, (HA4), Rev 2, 1/92 Figure 4-A PROTECTED AREA/SITE PERIMETER

Section 4.0	**HAZARDS AND ED JUDGMENT**
TAB 4.6	**SECURITY**
EAL 4.6.U	**UNUSUAL EVENT**
Mode	All
Description	**Security event which indicates a potential degradation in the level of safety of the plant** *(1 or 2)* 1. **BOMB** discovered within the **PROTECTED AREA** 2. Security Shift Supervisor reports one or more of the events listed in Table 4-4
Basis	A security threat that is identified as being directed towards the Station which represents a potential degradation in the level of safety of the plant warrants declaration of an Unusual Event. A confirmed report is satisfied if physical evidence supporting the threat exists, information independent from the actual threat message exists or a specific group claims responsibility for the threat. Examples of security events are provided in Table 4-4 Security Events a. SABOTAGE/INTRUSION has or is occurring within the PROTECTED AREA b. HOSTAGE/EXTORTION Situation that threatens to interrupt Plant Operations c. CIVIL DISTURBANCE ongoing between the SITE PERIMETER and PROTECTED AREA d. Hostile STRIKE ACTION within the PROTECTED AREA which threatens to interrupt Normal Plant Operations (judgment based on behavior of Strikers and/or intelligence received) In addition, BVPS uses a trained security organization and an approved physical security plan and procedures. External events which may result in a security threat would be reported to the duty Nuclear Shift Supervisor (NSS) by the Security Shift Supervisor. If in the NSS's judgment these events constitute an actual threat, they would be reported and a declaration made. BOMB refers to an explosive device. A HOSTAGE is a person(s) held as leverage against the station to ensure that demands will be met by the station. PROTECTED AREA encompasses all owner controlled areas within the security protected area fence as shown on Figure 4-A. SABOTAGE is deliberate damage, mis-alignment, or mis-operation of plant equipment with the intent to render the equipment inoperable. (Con't)

Section 4.0	**HAZARDS AND ED JUDGMENT**
TAB 4.6	**SECURITY**
EAL 4.6.U	**UNUSUAL EVENT**
Mode	All
Basis (continued)	A CIVIL DISTURBANCE exists when there is a group of twenty (20) or more persons violently protesting station operations or activities at the site. A STRIKE ACTION is a work stoppage within the PROTECTED AREA by a body of workers to enforce compliance with demands made on BVPS. The STRIKE ACTION must threaten to interrupt normal plant operations. EXTORTION is an attempt to cause an action at the station by threat of force. An INTRUSION/INTRUDER is a suspected hostile individual(s) present in a protected area without authorization.
Escalation	Escalation of this event would be based on hostile intrusion into the plant Protected Area.
References	NUMARC/NESP-007, (HU4), Rev 2, 1/92

Section 4.0	**HAZARDS AND ED JUDGMENT**
TAB 4.7	**EMERGENCY DIRECTOR JUDGMENT**
EAL 4.7.G	**GENERAL EMERGENCY**
Mode	All
Description	**Events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA Plume Protective Action Guidelines exposure levels outside the EXCLUSION AREA BOUNDARY. Refer to Figure 4-C**
Basis	This event classification provides the Shift Supervisor/Emergency Director, the flexibility to declare a General Emergency if in their judgment unanticipated conditions not explicitly covered elsewhere warrant declaration of an emergency. The declaration of a General Emergency indicates that there is a very high probability that the fuel has been damaged and the loss of containment integrity is possible or other conditions exist that may result in a release to the environment that may be greater than the EPA Protective Action Guides.
Escalation	Not Applicable
References	NUMARC/NESP-007, (HG2), Rev 2, 1/92

Section 4.0	**HAZARDS AND ED JUDGMENT**
TAB 4.7	**EMERGENCY DIRECTOR JUDGMENT**
EAL 4.7.S	**SITE AREA EMERGENCY**
Mode	All
Description	**Events are in process or have occurred which involve actual or likely major failures of plant functions needed for the protection of the public. Any releases are NOT expected to result in exposure levels which exceed EPA Plume Protective Action Guideline exposure levels outside the EXCLUSION AREA BOUNDARY. Refer to Figure 4-C**
Basis	This event classification provides the Shift Supervisor/Emergency Director, the flexibility to declare a Site Area Emergency if in their judgment unanticipated conditions not explicitly covered elsewhere warrant declaration. The declaration of a Site Area Emergency indicates high probability of major failures of plant functions needed to protect the public.
Escalation	Escalation of this event would be based on actual or imminent substantial core degradation.
References	NUMARC/NESP-007, (HS2), Rev 2, 1/92

Section 4.0	**HAZARDS AND ED JUDGMENT**
TAB 4.7	**EMERGENCY DIRECTOR JUDGMENT**
EAL 4.7.A	**ALERT**
Mode	All
Description	**Events are in process or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of the EPA Plume Protective Action Guideline exposure levels.**
Basis	This event classification provides the Shift Supervisor/Emergency Director, the flexibility to declare an Alert if, in their judgment, unanticipated conditions not explicitly covered elsewhere warrant declaration of an Alert emergency.
Escalation	Escalation of this event would be based on actual or likely failures in plant functions needed to protect the public.
References	NUMARC/NESP-007, (HA6), Rev 2, 1/92

Section 4.0	**HAZARDS AND ED JUDGMENT**
TAB 4.7	**EMERGENCY DIRECTOR JUDGMENT**
EAL 4.7.U	**UNUSUAL EVENT**
Mode	All
Description	**Unusual events are in process or have occurred which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.**
Basis	This event classification provides the Shift Supervisor/Emergency Director the flexibility to declare an Unusual Event if, in his judgment, unanticipated conditions not explicitly covered elsewhere warrant declaration of an emergency.
Escalation	Escalation of this event would be based on actual or potential degradation of plant safety systems.
References	NUMARC/NESP-007,(HU5), Rev 2, 1/92

Section 5.0	**DESTRUCTIVE PHENOMENA**
TAB 5.1	**EARTHQUAKE**
EAL 5.1.A	**ALERT**
Mode	All
Description	**Earthquake greater than 0.06g acceleration occurs ((1 and 2) for Unit 2)** 1. Analysis of Accelelorgraph Recording System data indicate ground acceleration > 0.06g in accordance with AOP 1/2.75.3 "*Acts of Nature - Earthquake*" *Unit 2 only* 2. (a and b) a. One or more alarm lamps and horn energized on the Seismic Warning panel (2ERS-ANN-1 b. Review of the printout on 2ERS-RSA-1 Response Spectrum Analyzer reveals an acceleration > 0.06g has occurred (see OM 2.45.4F) " Seismic Instrumentation Central Control Cabinet (2ERS-CCC-1) Running").
Basis	A seismic event of this level can cause damage to safety related systems. Plant seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required. This magnitude of acceleration is therefore consistent with the definition of an Alert.
Escalation	Escalation of this event will be based on " Fission Product Barrier Matrix".
References	NUMARC/NESP-007, (HA1), Rev. 2, 1/92 AOP 1/2.75.3 "Acts of Nature - Earthquake"

Section 5.0	**DESTRUCTIVE PHENOMENA**
TAB 5.1	**EARTHQUAKE**
EAL 5.1.U	**UNUSUAL EVENT**
Mode	All
Description	**Earthquake detected by site seismic instrumentation, >0.01g acceleration** *(1 and 2)* 1. Ann. A11-59 (A10-5h) "Seismic Accelerograph Operation" indicates initiation of the Accelerograph Recording System 2. *(a or b)* a. Ground motion sensed by plant personnel b. Unit 2 (Unit 1) reports seismic event detected on unit instrumentation
Basis	A seismic event of this level can cause some minor damage to plant structures or systems but it is not expected to have any impact on overall plant safety functions. There is a potential for degradation, however, and this is consistent with the definition of an Unusual Event. Plant seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required.
Escalation	Escalation of this event will be based on the magnitude of the ground acceleration.
References	NUMARC/NESP-007, (HU1), Rev. 2, 1/92

Section 5.0	**DESTRUCTIVE PHENOMENA**
TAB 5.2	**TORNADO**
EAL 5.2.A	**ALERT**
Mode	All
Description	**Tornado or high wind strikes any structure listed in Table 5-1 and results in structural damage** *(1 and 2)* 1. Tornado or high winds strikes any structure listed in Table 5-1 2. *(a or b)* a. Confirmed report of any **VISIBLE DAMAGE** to specified structures b. Control room indications of degraded safety system or component response within listed structures due to event
Basis	Tornados or high winds striking the structures listed in Table 5-1 can cause damage to plant structures or systems needed for Safe Shutdown of the Plant. Tornadoes are a phenomena whose occurrence cannot be specifically predicted. INDICATOR #1 includes both tornados and high wind. No magnitude or duration is specified to define high wind. This is due to the current limitation of the met instrumentation (50 mph) and the reliance on the observation of VISIBLE DAMAGE. Winds of sufficient magnitude and duration to cause damage to safety structures are of concern. The presence of VISIBLE DAMAGE to the specified structures identified in INDICATOR #2, indicates a potential for damage to the equipment contained within that structure. A second INDICATOR is used to avoid a missed declaration when actual equipment degradation is noted. In these cases, the damage is consistent with the declaration of an Alert. A magnitude and duration for high winds is not specified since the resultant damage and it's impact or potential impact on safety systems is addressed. Unit 1 Table 5-1 Plant Structures Associated With Tornado/Hi Wind and Aircraft EALs Containment Building — RWST (1QS-TK-1) — Diesel Generator Building Safeguards Building — CO2 Storage/PG Pp Rm — Main Intake Structure Primary Aux. Building — Service Bldg (incl. FW Reg Vlv Rm) Fuel Handling Building — Demin. Water Sto. (1WT-TK-10)
Escalation	Escalation of this event will be based on Fission Product Barriers.
References	NUMARC/NESP-007, (HAI), Rev. 2, 1/92

Section 5.0	**DESTRUCTIVE PHENOMENA**
TAB 5.2	**TORNADO**
EAL 5.2.A	**ALERT (Con't)**
Mode	All
Description	**Tornado or high wind strikes any structure listed in Table 5-1 and results in structural damage** *(1 and 2)* 1. Tornado or high winds strikes any structure listed in Table 5-1 2. *(a or b)* a. Confirmed report of any **VISIBLE DAMAGE** to specified structures b. Control room indications of degraded safety system or component response within listed structures due to event
Basis (Con't)	Unit 2 Table 5-1 Plant Structures Associated With Tornado/Hi Wind and Aircraft EALs Main Stm Vlv Rm. — Containment Building — Safeguards Bldg. RWST (2QSS-TK21) — Diesel Generator Building — 24 Ton CO2 Unit Main Intake Structure — Primary Aux. Building Service Bldg (incl. FW Reg Vlv Rm) — Fuel Handling Building Demin. Water Sto. (2FWE-TK210) — Control Bldg. Rod Control Cable Vault Bldg. VISIBLE DAMAGE is intended to be indicative of observed physical degradation. This damage has to affect plant safety systems or functions required to establish or maintain cold shutdown.
Escalation	Escalation of this event will be based on Fission Product Barriers.
References	NUMARC/NESP-007, (HAI), Rev. 2, 1/92

Section 5.0	**DESTRUCTIVE PHENOMENA**
TAB 5.2	**TORNADO**
EAL 5.2.U	**UNUSUAL EVENT**
Mode	All
Description	**Tornado within the SITE PERIMETER** 1. Plant personnel report a tornado has been sighted within the **SITE PERIMETER** (Refer to Figure 5-A)
Basis	A tornado touchdown within the Site Protected Area may have the potential to damage plant structures containing systems required for Safe Shutdown of the plant. This is consistent with the definition of an Unusual Event. SITE PERIMETER encompasses all owner controlled areas in the immediate site environs as shown on Figure 5-A.
Escalation	Escalation of this event will be based on the tornado striking plant structures or high sustained winds within the protected area.
References	NUMARC/NESP-007, (HUI), Rev. 2, 1/92

Section 5.0	**DESTRUCTIVE PHENOMENA**
TAB 5.3	**AIRCRAFT CRASH/PROJECTILE**
EAL 5.3.A	**ALERT**
Mode	All
Description	**Aircraft or PROJECTILE impacts (strikes) any plant structure listed in Table 5-1 resulting in structural damage** *(1 and 2)* 1. Plant personnel report aircraft or **PROJECTILE** has impacted any structure listed in Table 5-1 on previous page 2. *(a or b)* a. Confirmed report of any **VISIBLE DAMAGE** to specified structures b. Control Room indications of degraded safety system or component response(within listed area) due to event.
Basis	Aircraft or PROJECTILEs striking the structures listed in Table 5-1 can cause damage to plant structures or systems needed for Safe Shutdown of the Plant. The presence of VISIBLE DAMAGE to the specified structures identified in INDICATOR #2, indicates a potential for damage to the equipment contained within that structure. A second INDICATOR is used to avoid a missed declaration when actual equipment degradation is noted. In these cases, the damage is consistent with the declaration of an Alert. Unit 1 Table 5-1 Plant Structures Associated With Tornado/Hi Wind and Aircraft EALs Containment Building — RWST (1QS-TK-1) — Diesel Generator Building Safeguards Building — CO2 Storage/PG Pp Rm — Main Intake Structure Primary Aux. Building — Service Bldg (incl. FW Reg Vlv Rm) Fuel Handling Building — Demin. Water Sto. (1WT-TK-10) Unit 2 Table 5-1 Plant Structures Associated With Tornado/Hi Wind and Aircraft EALs Main Stm Vlv Rm. — Containment Building — Safeguards Bldg. RWST (2QSS-TK21) — Diesel Generator Building — 24 Ton CO2 Unit Main Intake Structure — Primary Aux. Building Service Bldg (incl. FW Reg Vlv Rm) — Fuel Handling Building Demin. Water Sto. (2FWE-TK210) — Control Bldg Rod Control Cable Vault Bldg.
Escalation	Escalation to this event will be based on "Fission Product Barriers Matrix".
References	NUMARC/NESP-007, (HA1, HA2), Rev. 2, 1/92

Section 5.0	**DESTRUCTIVE PHENOMENA**
TAB 5.3	**AIRCRAFT CRASH/PROJECTILE**
EAL 5.3.A	**ALERT (Con't)**
Mode	All
Description	**Aircraft or PROJECTILE impacts (Strikes) any plant structure listed in Table 5-1 resulting in structural damage** *(1 and 2)* 1. Plant personnel report aircraft or **PROJECTILE** has impacted any structure listed in Table 5-1 2. *(a or b)* a. Confirmed report of any **VISIBLE DAMAGE** to specified structures b. Control Room indications of degraded safety system or component response within listed structures due to event.
Basis (Con't)	VISIBLE DAMAGE is intended to be indicative of observed physical degradation. This damage has to affect plant safety systems or functions required to establish or maintain cold shutdown. PROJECTILE is intended to include any object that is ejected, thrown, or launched towards a plant structure. The object must be of sufficient size or mass to potentially inflict damage sufficient to cause concern regarding the integrity of the affected structure or the operability of the safety equipment contained within the structure.
Escalation	Escalation to this event will be based on "Fission Product Barriers Matrix".
References	NUMARC/NESP-007, (HA1, HA2), Rev. 2, 1/92

Section 5.0	**DESTRUCTIVE PHENOMENA**
TAB 5.3	**AIRCRAFT CRASH/PROJECTILE**
EAL 5.3.U	**UNUSUAL EVENT**
Mode	All
Description	**Aircraft crash or PROJECTILE impact within the SITE PERIMETER** 1. Plant personnel report aircraft crash or **PROJECTILE** impact within the **SITE PERIMETER** (Refer to Figure 5-A)
Basis	Aircraft or PROJECTILE Impacts within the SITE PERIMETER are off normal events that can indicate a potential degradation of the level of safety of the plant. This is consistent with the definition of an Unusual Event. SITE PERIMETER encompasses all owner controlled areas in the immediate site environs as shown on Figure 5-A. PROJECTILE is intended to include any object that is ejected, thrown, or launched towards a plant structure. The object must be of sufficient size or mass to potentially inflict damage sufficient to cause concern regarding the integrity of the affected structure or the operability of the safety equipment contained within the structure.
Escalation	Escalation to this event will be based on an Impact on plant structures.
References	NUMARC/NESP-007, (HU1), Rev. 2, 1/92

Section 5.0	**DESTRUCTIVE PHENOMENA**
TAB 5.4	**RIVER LEVEL HIGH**
EAL 5.4.A	**ALERT**
Mode	All
Description	**River water level > 705 mean sea level** *(1 or 2)* 1. 1LR-CW-101, if accessible, indicates >705 mean sea level 2. National Weather Bureau (412-644-2882) or Montgomery Lock (724-643-8400) reports Montgomery Lower Pool stage height >52.48 Ft. *Note: Mean Sea Level = stage height + 652.52 Ft*
Basis	The requirements for flood protection ensures that facility protective actions will be taken and operation will be terminated in the event of flood conditions. A river level of >705 mean sea level is consistent with the elevation of the main transformer pad. This river level will permit flooding to occur within the turbine building. While no safety related equipment is expected to be affected at this elevation, the height is sufficient to warrant declaration of an Alert.
Escalation	Escalation of this event will be based on "Fission Product Barriers Matrix".
References	NUMARC/NESP-007, (HA1), Rev. 2, 1/92

Section 5.0	**DESTRUCTIVE PHENOMENA**
TAB 5.4	**RIVER LEVEL HIGH**
EAL 5.4.U	**UNUSUAL EVENT**
Mode	All
Description	**River water level > 700 mean sea level** *(1 or 2)* 1. 1LR-CW-101, if accessible, indicates >700 mean sea level 2. National Weather Bureau (412-644-2882) or Montgomery Lock (724-643-8400) reports Montgomery Lower Pool stage height >47.48 Ft. *Note: Mean Sea Level = stage height + 652.52 Ft*
Basis	The requirements for flood protection ensures that facility protective actions will be taken and operation will be terminated in the event of flood conditions. A river level of >700 mean sea level is below the level of the main transformer pad but above the level requiring shutdown per Technical Specifications. This is indicative of a potential degradation in the level of safety of the plant and thus is consistent with the definition of an Unusual Event.
Escalation	Escalation of this event will be based on "Fission Product Barriers Matrix".
References	NUMARC/NESP-007, (HU1), Rev. 2, 1/92

Section 5.0	**DESTRUCTIVE PHENOMENA**
TAB 5.5	**RIVER LEVEL LOW**
EAL 5.5.A	**ALERT**
Mode	All
Description	**River water level <648.6 Ft Mean Sea Level** *(1 or 2)* 1. 1LR-CW-101, if accessible, indicates < 648.6 Ft mean sea level 2. National Weather Bureau (412-644-2882) or Montgomery Lock (724-643-8400) reports Montgomery Lower Pool stage height < -3.92 Ft. *Note: Mean Sea Level = stage height + 652.52 Ft*
Basis	A level of < 648.6 Ft mean sea level was selected for this EAL. This river level will result in reduction/loss of suction to the intake structure pumps. Two methods of obtaining the information is included in the EAL. This precludes reliance on a single instrument.
Escalation	Escalation to this event will be based on "Fission Product Barrier Matrix."
References	NUMARC/NESP-007, (HA1 example #7), Rev. 2, 1/92

Section 5.0	**DESTRUCTIVE PHENOMENA**
TAB 5.6	**WATERCRAFT CRASH**
EAL 5.6.U	**UNUSUAL EVENT**
Mode	All
Description	**Watercraft strikes primary intake structure and results in a flow reduction of Reactor Plant or Turbine Plant River Water flow** *(1 and 2)* 1. Plant personnel report a watercraft has struck the primary intake structure 2. *(a or b* a. RPRW (SWS)flow reduction indicated by sustained pressure reduction <20 (<30) psig on PI-1RW-113A and/or 113B. (2SWS-PI113A and/or B) b. TPRW flow reduction indicated by sustained pressure reduction (Ann A6-118 "RAW Water Pump Disch Press Low" <15 psig) / (n/a for Unit 2)
Basis	This EAL is included to consider the potential degradation of plant safety due to a large watercraft striking the main intake structure. Actual degradation in flow is included as INDICATOR #2. Sustained pressure reduction is intended to allow the starting of the standby pump. Actual flow degradation is used at the Unusual Event level since the intake structure is supported by a redundant structure. The Alternate intake structure is located upstream of the main intake structure and has capability of replacing the Reactor Plant River Water pumps. The absence of active rail spurs and rail traffic within the Beaver Valley Power Station property eliminates the need to consider structural damage resulting from a train derailment.
Escalation	Escalation would be based on "Fission Product Barrier Matrix".
References	NUMARC/NESP-007, (SU4), Rev. 2, 1/92

Section 6.0	**SHUTDOWN SYSTEMS DEGRADATION**
TAB 6.1	**LOSS OF SHUTDOWN SYSTEMS**
EAL	**All**
Mode	5,6
Description	Not applicable
Basis	*This discussion applies generically to all EALs in TAB 6.1:* The EALs in this TAB address concerns raised by Generic Letter 88-17, "*Loss of Decay Heat Removal*", SECY-91-283, "*Evaluation of Shutdown and Low Power Risk Issues.*", NUREG-1449, "*Shutdown and Low Power Operation at Commercial Nuclear Power Plants in the United States*", and NUMARC 91-06, "*Guidelines for Industry Actions to Assess Shutdown Management*". A number of plant conditions such as initial vessel level (e.g., mid-loop, reduced level/flange level, normal, or cavity filled), RCS venting strategy, decay heat removal system design, vortexing pre-disposition, steam generator U-tube draining, and level instrumentation problems can have a significant impact in causing a loss of decay heat removal, or acerbating the consequences of such a loss. NRC analyses show that some specific sequences shortly after shutdown can result in core uncovery in 15 to 20 minutes and severe core damage within an hour after decay heat removal is lost. The progression and severity of shutdown events, and the magnitude of potential radioactivity releases that result, depends on numerous factors. The primary factors affecting progression and severity are (1) time since shutdown (i.e., magnitude of decay heat), (2) RCS inventory (including flooded cavity as applicable), and (3) availability of heat sink. For radioactivity releases, the primary factors are (1) time since shutdown, and (2) integrity of fission product barriers. All of these factors are variables in shutdown events. Unlike events which occur at power, the "starting point' for shutdown events can vary significantly, as can the availability of redundant means of heat removal, release mitigation features, and instrumentation. This situation makes assessment difficult. The EALs in this TAB are a compromise between potential over-conservatism in declarations for events that occur under the best of circumstances (e.g., late in outage, RCS and containment intact), and the need for anticipatory action for events that occur under the worst of circumstances (e.g., mid-loop operations early in outage). (Con't)

Section 6.0	**SHUTDOWN SYSTEMS DEGRADATION**
TAB 6.1	**LOSS OF SHUTDOWN SYSTEMS**
EAL	**All**
Mode	5,6
Description	Not applicable
Basis (Con't)	*This discussion applies generically to all EALs in TAB 6.1:* The ability to assess shutdown events is contingent on the availability of RCS temperature indication. There may be, during certain phases of an outage (e.g., head lifts), extended periods during which the core exit temperature instrumentation is totally dependent on RTDs exposed to RHR forced flow. If RHR is lost, so is the ability to monitor the parameter most significant to assessment. In order to address this, the EALs refer first to temperature increases on instrumentation and then, as an backup, to fixed time frames or other physical evidence reported by plant personnel.

Section 6.0	**SHUTDOWN SYSTEMS DEGRADATION**
TAB 6.1	**LOSS OF SHUTDOWN SYSTEMS**
EAL	**All**
Mode	5,6
Description	Not applicable
Basis	Not applicable
Escalation	Not applicable
References	Generic Letter 88-17, "*Loss of Decay Heat Removal* SECY-91-283, "*Evaluation of Shutdown and Low Power Risk Issues.*" NUREG-1449, *'Shutdown and Low Power Operation at Commercial Nuclear Power Plants in the United States'* NUMARC 91-06, "*Guidelines for Industry Actions to Assess Shutdown Management*".

Section 6.0	**SHUTDOWN SYSTEMS DEGRADATION**
TAB 6.1	**LOSS OF SHUTDOWN SYSTEMS**
EAL 6.1.S	**Site Area Emergency**
Mode	Not applicable
Description	Not applicable"
Basis	Not applicable
Escalation	Not applicable
References	Pending (NUMARC SS7P)

Section 6.0	**SHUTDOWN SYSTEMS DEGRADATION**
TAB 6.1	**LOSS OF SHUTDOWN SYSTEMS**
EAL 6.1.A	**Alert Emergency**
Mode	5,6
Description	**Inability to maintain unit in cold shutdown** *(1 and 2)* 1. **UNPLANNED** Loss of RHR or CCR or RPRW (RHS or CCP or SWS) 2. *(a or b or c)* a. Core exit thermocouples (CETCs)(if available) indicate the temperature has increased >10°F and has exceeded 200°F b. (w/ RHR (RHS) in service) RHR (RHS) inlet temp has increased >10°F and has exceeded 200°F. c. (w/o CETCs or RHR (RHS)) Loss has exceeded 30 minutes or there is evidence of boiling in the Rx vessel
Basis	See generic basis for this Tab. This EAL is intended to establish the escalation threshold for the declaration of a Alert Emergency. This Alert Emergency declaration is consistent with the need to rapidly correct the problem through the augmentation of onsite personnel and the need to inform offsite authorities. Continued degradation can result in fuel uncovery and severe damage with resultant releases of a significant fraction of the gap activity. This event escalates to a Site Area Emergency via 6.2 RCS Inventory (Shutdown) or 7.1 Gaseous Effluents. The specification of a 10°F temperature increase precludes Alert Emergency declaration for a momentary controllable loss that occurs at a temperature very near 200°F. The 10°F increase also ensures that the declaration is made prior to the onset of boiling where temperature may temporarily stabilize. The EAL provides for classification based on core exit temperature indication. To address conditions in which core exit temperature indication is not available (e.g., CETCs disconnected, loss of RHR flow past RTDs), 30 minutes is allotted. Physical evidence of boiling is also included. The 30 minute time duration is expected to conservatively encompass nearly all initial conditions.
Escalation	Escalation to Site Area Emergency would occur via 6.2 RCS Inventory (Shutdown), or as indicated by Tab 7.1 Gaseous Effluent EALs
References	Pending (NUMARC SA3P)

Section 6.0	**SHUTDOWN SYSTEMS DEGRADATION**
TAB 6.1	**LOSS OF SHUTDOWN SYSTEMS**
EAL 6.1.U	**Unusual Event**
Mode	5,6
Description	**UNPLANNED loss of any function needed for cold shutdown that results in a core exit temperature increase of more than 10°F** *(1 and 2)* 1. **UNPLANNED** Loss of RHR or CCR or RPRW (RHS or CCP or SWS) 2. *(a or b or c)* a. Core exit thermocouples (CETCs)(if available) indicate the temperature has increased >10°F b. (w/ RHR (RHS) in service) RHR (RHS) inlet temp has increased >10°F. c. (w/o CETCs or RHR (RHS)) Loss has exceeded 15 minutes.
Basis	See generic basis for this Tab. This EAL addresses events in which there is an unplanned loss of any function needed for maintaining cold shutdown. In this EAL, the fundamental parameter of RCS exit temperature is used as a basis for classification. This EAL keys on function, rather than specific pieces of equipment. This EAL establishes the classification threshold at a temperature rise of 10°F. A temperature rise of this magnitude is not expected as a result of normal operation and is beyond normal instrument fluctuations. The phrase 'unplanned' is specified to preclude the declaration of an emergency for circumstances in which decay heat removal is intentionally placed out-of-service and is controlled within the requirements of the T/S. Continued degradation can result in fuel uncovery and severe damage with resultant releases of a significant fraction of the gap activity. The EAL provides for classification based on core exit temperature indication. To address conditions in which core exit temperature indication is not available (e.g., CETCs disconnected, loss of RHR flow past RTDs), 15 minutes is allotted. This time duration is expected to be a conservative default value for nearly all initial conditions.
Escalation	Escalation to Alert Emergency would occur if temperature increased to above 200°F as a result of the 10°F increase, or as indicated by Tab 7.1 Gaseous Effluent EALs
References	Pending (NUMARC SU9P)

Section 6.0	**SHUTDOWN SYSTEMS DEGRADATION**
TAB 6.2	**RCS INVENTORY - SHUTDOWN**
EAL	**All**
Mode	5,6
Description	Not applicable
Basis	*This discussion applies generically to all EALs in TAB 6.2:* The EALs in this TAB address concerns raised by Generic Letter 88-17, *"Loss of Decay Heat Removal"*, SECY-91-283, *"Evaluation of Shutdown and Low Power Risk Issues."*, NUREG-1449, *"Shutdown and Low Power Operation at Commercial Nuclear Power Plants in the United States"*, and NUMARC 91-06, *"Guidelines for Industry Actions to Assess Shutdown Management"*. A number of plant conditions such as initial vessel level (e.g., mid-loop, reduced level/flange level, normal, or cavity filled), RCS venting strategy, decay heat removal system design, vortexing pre-disposition, steam generator U-tube draining, and level instrumentation problems can have a significant impact in causing a loss of decay heat removal, or acerbating the consequences of such a loss. NRC analyses show that some specific sequences shortly after shutdown can result in core uncovery in 15 to 20 minutes and severe core damage within an hour after decay heat removal is lost. The progression and severity of shutdown events, and the magnitude of potential radioactivity releases that result, depends on numerous factors. The primary factors affecting progression and severity are (1) time since shutdown (i.e., magnitude of decay heat), (2) RCS inventory (including flooded cavity as applicable), and (3) availability of heat sink. For radioactivity releases, the primary factors are (1) time since shutdown, and (2) integrity of fission product barriers. All of these factors are variables in shutdown events. Unlike events which occur at power, the "starting point" for shutdown events can vary significantly, as can the availability of redundant means of heat removal, release mitigation features, and instrumentation. This situation makes assessment difficult. Similarly, the development of EALs is made difficult. The EALs in this TAB are a compromise between potential over-conservatism in declarations for events that occur under the best of circumstances (e.g., late in outage, RCS and containment intact), and the need for anticipatory action for events that occur under the worst of circumstances (e.g., mid-loop operations early in outage). Note that BVPS administrative controls ensure containment closure prior to mid-loop operation. The ability to assess the shutdown events in this TAB is contingent on the availability of reactor vessel level indication. There may be, during certain phases of an outage, extended periods during which the level instrumentation is not available. In order to address this, the EALs refer first to level indications on instrumentation and then, as an backup, to other confirmed indications of fuel uncovery.

Section 6.0	**SHUTDOWN SYSTEMS DEGRADATION**
TAB 6.2	**RCS INVENTORY - SHUTDOWN**
EAL	**All**
Mode	Not applicable
Description	Not applicable
Basis	*(Continued)* Not applicable
Escalation	Not applicable
References	Generic Letter 88-17, "*Loss of Decay Heat Removal* SECY-91-283, "*Evaluation of Shutdown and Low Power Risk Issues.*" NUREG-1449, *'Shutdown and Low Power Operation at Commercial Nuclear Power Plants in the United States'* NUMARC 91-06, "*Guidelines for Industry Actions to Assess Shutdown Management*".

Section 6.0	**SHUTDOWN SYSTEMS DEGRADATION**
TAB 6.2	**RCS INVENTORY - SHUTDOWN**
EAL 6.2.G	**General Emergency**
Mode	Not applicable
Description	**Not applicable**
Basis	Not applicable
Escalation	Not applicable
References	Pending (NUMARC SG3P)

Section 6.0	**SHUTDOWN SYSTEMS DEGRADATION**
TAB 6.2	**RCS INVENTORY - SHUTDOWN**
EAL 6.2.S	**Site Area Emergency**
Mode	5,6
Description	**Loss of water level in the reactor vessel that has or will uncover fuel in the reactor vessel with containment closure established** *(1 and 2)* 1. (a or b) a. Loss of RHR or CCR or RPRW (RHS or CCP or SWS) b. Loss of RCS Inventory with inadequate makeup 2. *(a and b)* a. Ops personnel report LI-1RC-480, 482C (2RCS-LI-102, LR-102) RCS level instrumentation in the Control Room indicates a level drop to 0 inches (if available) b. Other confirmed indications of fuel uncovery
Basis	See generic bases for this TAB This EAL is intended to establish the escalation threshold for the declaration of a Site Area Emergency. This declaration is consistent with the need to rapidly correct the problem through the augmentation of onsite personnel and the need to inform offsite authorities. This event progresses from a loss of RHR event such that bulk boiling occurs in the reactor vessel. If RCS inventory cannot be maintained, for whatever cause, the boiling will result in fuel uncovery. Clad damage will occur prior to the onset of core melt due to stresses on the clad. The potential for significant releases from the fuel exists. A Site Area Emergency classification is warranted in that there have been failures of systems necessary for the protection of the public. The EAL provides for classification based on reactor vessel level indication. To address conditions in which reactor vessel level indication is not available, other confirmed indications of fuel uncovery is utilized. This should include local observation, indication of bulk boiling, or significant radiation level increases associated with an inventory loss.
Escalation	Escalation to General Emergency would occur if containment closure was not established with the RCS not intact resulting in direct release to the environs as indicated by Tab 7.1 Gaseous Effluent EALs
References	NUMARC/NESP-007 (SS5), Rev 2, 1/92

Section 6.0	**SHUTDOWN SYSTEMS DEGRADATION**
TAB 6.2	**RCS INVENTORY - SHUTDOWN**
EAL 6.2.U	**Unusual Event**
Mode	5,6
Description	**Loss of Reactor Coolant System Inventory with inadequate make-up** *(1 and 2)* 1. Ops personnel report LI-1RC-480 or LI-1RC-482C (2RCS-LI-102, LR-102) RCS level instrumentation in the Control Room indicates a level drop to less than 14.5 inches 2. Ops personnel report inability to make-up RCS inventory
Basis	See generic bases for this TAB This EAL is intended to serve as a precursor to loss of RHR (RHS). The loss of RCS inventory could be the result of failure of temporary piping or temporary barriers (e.g., steam generator dams, freeze seals). The potential for such events increases during shutdown due to the accelerated maintenance activity that occurs during these periods. In addition to creating the potential for loss of inventory, this maintenance activity, removes equipment from service that could restore inventory to mitigate the consequences of the loss. A sudden loss of inventory could result in a loss of decay heat removal due to RHR (RHS) pump suction vortexing or preemptory operator pump manual shutdowns, as could a smaller leak that cannot be isolated. TABs 2.5 and 2.6 address RCS leakage. Although the mode applicability includes mode 5, it is limited to mode 5 with the RCS pressurized. There are no EALs that address RCS leakage in mode 5 with the RCS depressurized, or in mode 6. Further, those EALs identify a specific numeric leak rate, which is not appropriate to shutdown conditions. This EAL does not specify a numeric leak rate in that the conditions surrounding the leak and the systems available to make-up losses can depend on ongoing maintenance activities. There are no make-up systems required by T/S in shutdown modes.
Escalation	Escalation to higher classifications would occur if (1) the core becomes uncovered, or (2) if the RHR (RHS) loss results in core exit temperature increase in excess of 10 F and exceeds 200 F
References	Pending (NUMARC Shutdown EALs consistent w/ NUMARC/NESP-007 HU5)

Section 6.0	**SHUTDOWN SYSTEM DEGRADATION**
TAB 6.3	**LOSS OF AC (Shutdown)**
EAL 6.3.A	**Alert**
Mode	5,6, defuel
Description	**UNPLANNED loss of offsite and onsite AC power for >15 minutes** 1. AE and DF 4KV buses not energized from Unit 1 (2) source for >15 minutes
Basis	A loss of all AC power compromises all plant safety systems that require AC power including RHR, spent fuel pool cooling, and the river water systems. At modes 1-4, this event would be classified as Site Area Emergency. A lower classification is justified here due to the reduced decay heat. 15 minutes is specified so as to exclude momentary power losses. Note however, that this event is bounded by EAL 6.2.S if the loss of AC results in fuel uncovery. INDICATOR #1 encompasses the CRITERION in that the AE and DF buses are fed from either offsite or onsite sources. Thus, having both buses de-energized indicates a failure of both sources.
Escalation	Escalation would occur if the loss of power results in fuel uncovery per 6.2.S.
References	NUMARC/NESP-007 (SA1 - addition), Rev 2, 1/92

Section 6.0	**SHUTDOWN SYSTEM DEGRADATION**
TAB 6.3	**LOSS OF AC (Shutdown)**
EAL 6.3.U	**Unusual Event**
Mode	5,6, defuel
Description	**UNPLANNED loss of offsite AC power for >15 minutes** *(1 and 2)* 1. 1A and 1D (2A and 2D) 4KV normal buses de-energized for >15 minutes 2. Either diesel generator is supplying power to its respective emergency bus
Basis	A prolonged loss of offsite AC power reduces power source redundancy and potentially degrades the level of safety of the plant by rendering the plant more vulnerable to a complete loss of AC power. 15 minutes is specified so as to exclude momentary power losses. This EAL is similar to EAL 3.2.U, except that the phrase UNPLANNED was added to exclude classifications that could result from offsite power bus outages scheduled and controlled by maintenance work activities. INDICATOR #1 are the buses that would be de-energized in the event of a loss of offsite power. INDICATOR #2 establishes that at least one train of onsite power is available.
Escalation	Escalation would occur if onsite AC power was lost.
References	NUMARC/NESP-007 (SU1 - addition), Rev 2, 1/92

Section 6.0	**SHUTDOWN SYSTEM DEGRADATION**
TAB 6.4	**LOSS OF DC (Shutdown)**
EAL 6.4.U	**Unusual Event**
Mode	5,6, defueled
Description	**UNPLANNED loss of the required train of DC power for >15 minutes** *(1 or 2)* 1. Voltage <110.4 VDC on DC buses 1-1 and 1-3 (2-1 and 2-3) for >15 minutes if train A is the priority train 2. Voltage <110.4 VDC on DC buses 1-2 and 1-4 (2-2 and 2-4) for >15 minutes if train B is the priority train
Basis	The significance of this EAL rests with the impact that a loss of DC power could have on monitoring and controlling decay heat removal during shut down modes. At modes 1-4, this event would be classified as Site Area Emergency if both trains were lost. A lower classification is justified here due to the reduced decay heat. 15 minutes is specified so as to exclude momentary power losses. In INDICATOR #1 and INDICATOR #2, the specified voltage is the minimum voltage specified in the UFSAR at which DC loads will perform reliably.
Escalation	Escalation would occur if RHR loss occurs.
References	NUMARC/NESP-007 (SU7 - addition), Rev 2, 1/92

Section 6.0	**SHUTDOWN SYSTEM DEGRADATION**
TAB 6.5	**FUEL HANDLING**
EAL 6.5.A	**Alert**
Mode	All
Description	**Major damage to irradiated fuel; or loss of water level that has or will uncover irradiated fuel outside the reactor vessel** *(1 and 2)* 1. **VALID** Hi-Hi Alarm on RM-RM-203 or RM-RM-207 or RM-VS-103 A/B or RM-VS-104A/B (High on 2RMF-RQ202(1031), 301A/B (1032/2032), 2HVR-RQ104A/B (1024/1028), or 2RMR-RQ203(1025)) 2. (a or b) a Plant personnel report damage of irradiated fuel sufficient to rupture fuel rods b. Plant personnel report water Level drop has or will exceed makeup capacity such that irradiated fuel will be uncovered
Basis	The major concern of the EAL is a fuel handling accident or loss of water covering spent fuel. Events away from the reactor vessel (e.g., in the cavity, transfer tube, or spent fuel pool) are addressed. Events within the vessel are classified in accordance with TABs 6.1 and 6.2. Events of this type could cause an increase in radioactivity readings and potentially a release to the environment. The magnitude of these releases is dependent on the amount of damage, depth of water above damage, and available filtration systems. Design basis fuel handling accident doses could exceed the EPA PAG, warranting a General Emergency classification. However, as with all UFSAR analyses, there is extensive conservatism in the analysis. Thus, an Alert Emergency is deemed justified. This declaration would result in augmentation of onsite personnel to support assessment of the release and restorative actions to stabilize the condition. With regard to the loss of water level, design features and administrative controls limit the possible fuel uncovery to a single element. Analyses performed in response to IE Bulletin 84-03, showed that the clad on a fuel assembly suspended in air would begin to melt at about 60 minutes, assuming an ambient air temperature of 105 °F, which is conservative. This time period provides for event-specific assessments. Escalation of the classification would be based on the results of these assessments.
Escalation	Escalation would on the basis of TAB 7.1, Gaseous Effluents
References	NUMARC/NESP-007 (AA2 example # 1,3), Rev 2, 1/92 ltr dtd 10/24/84, JJCarey to TEMurley USNRC RI ltr ND1SCA:0095 dtd 9/17/84, MYLee to KDGrada

Section 6.0	**SHUTDOWN SYSTEM DEGRADATION**
TAB 6.5	**FUEL HANDLING**
EAL 6.5.A	**Alert (Con't)**
Mode	All
Description	(Con't)
Basis (Con't)	INDICATOR #1 verifies the reports discussed in INDICATOR #2 by noting the increase in radiation levels, and/or airborne activity in the affected areas. An increase on the ventilation monitors signifies the release of radioactivity in the fuel gap, whereas, an increase on area radiation monitors is indicative of reduced shielding due to the decrease in water level.
Escalation	Escalation would on the basis of TAB 7.1, Gaseous Effluents
References	NUMARC/NESP-007 (AA2 example # 1,3), Rev 2, 1/92 ltr dtd 10/24/84, JJCarey to TEMurley USNRC RI ltr ND1SCA:0095 dtd 9/17/84, MYLee to KDGrada

Section 6.0	**SHUTDOWN SYSTEM DEGRADATION**
TAB 6.5	**FUEL HANDLING**
EAL 6.5.U	**Unusual Event**
Mode	All
Description	**UNPLANNED loss of water level in spent fuel pool or reactor cavity or transfer canal with fuel remaining covered** *(1 and 2 and 3)* 1. Plant personnel report water level drop in spent fuel pool or reactor cavity or transfer canal 2. **VALID** Hi-Hi Alarm on RM-RM-203 or RM-RM-207 (2RMR-RQ203 (1025) or 2RMF-RQ202 (1031)) 3. Fuel remains covered with water.
Basis	The major concern of the EAL is a loss of water covering spent fuel. Events away from the reactor vessel (e.g., in the cavity, transfer tube, or spent fuel pool) are addressed. Events within the vessel are classified in accordance with TABs 6.1 and 6.2. Events of this type could cause an increase in radioactivity readings and potentially a release to the environment. The magnitude of these releases is dependent on the amount of damage, depth of water above damage, and available filtration systems. However, even without a release, elevated dose rates in adjacent areas could create access limitations. (See TAB 7.3) The design of fuel handling equipment and administrative controls on activities involving spent fuel maintains water above the fuel during normal handling. Should there be a loss of water level, such as that associated with a failure of the reactor cavity seal, fuel elements could be exposed to air in three locations: (1) in the manipulator mast, in the RCCA change fixture, and suspended from the fuel pool bridge crane. Analyses performed in response to IE Bulletin 84-03, showed that the clad on a fuel assembly suspended in air would begin to melt at about 60 minutes, assuming an ambient air temperature of 105 °F, which is conservative. The additional heat transfer afforded by the water assumed in this EAL would extend this time to several hours. This time period provides for event-specific assessments. Escalation of the classification would be based on the results of these assessments. INDICATOR #2 verifies the reports discussed in INDICATOR #1 by noting the increase in radiation levels in the affected areas. An increase on area radiation monitors is indicative of reduced shielding due to the decrease in water level. INDICATOR #3 is the discriminator between the Unusual Event and the Alert.
Escalation	Escalation would on the basis of TAB 7.1, Gaseous Effluents, or TAB 7.3, Radiation Levels
References	NUMARC/NESP-007 (AU2 example # 1,2), Rev 2, 1/92 ltr dtd 10/24/84, JJCarey to TEMurley USNRC RI ltr ND1SCA:0095 dtd 9/17/84, MYLee to KDGrada

Section 6.0	**SHUTDOWN SYSTEM DEGRADATION**
TAB 6.6	**INADVERTENT CRITICALITY**
EAL 6.6.A	**Alert Emergency**
Mode	3, 4, 5, 6
Description	**Inadvertent reactor criticality** 1. Nuclear instrumentation indicates unanticipated sustained positive startup rate
Basis	This EAL addresses situations in which inadvertent criticalities occur. Improper rod withdrawals are included but limited in application to Modes 3,4,5, and 6. It is not intended that this Alert apply to a premature criticality during a planned reactor startup. In this situation the plant has been prepared for the reactor to be brought critical and procedural control dictate appropriate action. This situation is therefore not consistent with the declaration of an emergency. This EAL also addresses events (e.g., inadvertant dilution, failure of loop dams) that result in dilution of RCS boron concentration. It has been postulated that localized criticality could occur in the reactor vessel due to such a failure with RCS temperature cold. Such a criticality would cease once in-vessel mixing re-established negative reactivity in the affected region of the core. Since this sequence would likely be less than the recognition and assessment time, the INDICATOR calls for a sustained positive startup rate.
Escalation	Escalation would on the basis of the failure of RHR to remove the heat of fission, resulting in a heat-up.
References	Pending (NUMARC Shutdown EALs consistent w/ NUMARC/NESP-007 HA6)

Section 7.0	**RADIOLOGICAL / FUEL HANDLING**
TAB 7.1	**GASEOUS EFFLUENTS**
EAL	**All**
Mode	All
Description	*The following apply generically to the gaseous effluent Tab:* The Radiological / Fuel Handling TAB is structured with CRITERION and INDICATORs as with the previous tabs (except Tab 1). The CRITERION establishes the numeric values for the offsite dose (General, Site Area), or release rate (Alert, UE). The INDICATORs specify monitor readings that serve as thresholds for performing particular dose assessments -- the results from which are then compared to the CRITERION, and appropriate declarations made. Declarations are not made on the basis of exceeding the INDICATOR threshold alone unless the specified assessment cannot be completed within 15 minutes (60 minutes for UE) of recognition. The radiation monitor readings that serve as INDICATORs for the General Emergency and the Site Area Emergency were calculated using accident source terms based on the UFSAR of Unit 2, design release flow rates, and annual average meteorology. As such, these INDICATORs are expected to provide an upper boundary on the offsite consequences associated with the INDICATOR. However, in an actual accident situation, the actual values of the above parameters (particularly meteorology) are likely to be different, potentially resulting in an over-classification or under-classification. It is for this reason that these EALs are based on the results of timely assessments rather than on the monitor reading itself. Assessments are performed using ARERAS or the EPP/IP-2.6.x series hand procedures. Note that while the monitor thresholds are based on annual average meteorology, the dose assessments are performed with actual meteorology. For the Alert and Unusual Events, a similar protocol is used. In these cases the INDICATORs are based on the methodology of the Offsite Dose Calculation Manual (ODCM) which utilizes an expected nuclide mix and annual average meteorology. The use of the ODCM as a basis provides a desirable linkage to the Radiological Effluent Technical Specifications (RETS) and the Radioactive Waste Discharge Authorizations (RWDA). Assessments are performed using the abnormal gaseous assessment procedures in the Health Physics Manual (HPM) for an Unusual Event and ARERAS or the EPP/IP-2.6.x series hand procedures for an Alert. Assessment using actual meteorology is not required for the Unusual Event due to the several orders of magnitude difference between the UE CRITERION and the EPA PAG. The EXCLUSION AREA BOUNDARY (EAB) referred to in these EALs are shown on EAL Figure 7-A. The EAB is shown as a 2000' circle centered on the Unit 1 RBC. This is consistent with the Unit 1 UFSAR. The Unit 2 UFSAR shows the Unit 2 EAB as being encompassed by the Unit 1 EAB except for areas over the Ohio River. For these EALs, the two EABs are shown as one as the dose projection methods determine X/Q at the EAB radius in all directions.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev 2, 1/92

Section 7.0	**RADIOLOGICAL / FUEL HANDLING**
TAB 7.1	**GASEOUS EFFLUENTS**
EAL 7.1.G	**General Emergency**
Mode	All
Description	**EAB dose resulting from an actual or imminent release of gaseous radioactivity that exceeds 1000 mR TEDE or 5000 mR child thyroid CDE for the actual or projected duration of the release** *(1 or 2 or 3)* 1. A **VALID** rad monitor reading exceeds the values in Column 4 of Table 7-1 for >15 minutes, unless assessment within this period confirms that the **CRITERION** is NOT exceeded 2. Field survey results indicate EAB dose >1000 mR β-γ for the actual or projected duration of the release 3. EPP dose assessment results indicate EAB dose >1000 mR TEDE or >5000 mR child thyroid CDE for the actual or projected duration of the release
Basis	See generic bases for this Tab The CRITERION is based on the current EPA Protective Action Guidelines (PAG) for the plume exposure pathway, which call for offsite evacuations if the projected dose exceeds 1 rem TEDE or 5 rem child thyroid CDE. As such, the CRITERION is consistent with the fundamental definition of a General Emergency. The child thyroid is specified here for consistency with the PAG protocol agreed upon by the states within the BVPS EPZ INDICATOR #1 refers to a set of monitor readings that, based on annual average meteorology and assumed default source terms, correspond to the CRITERION. The time duration is included to discount momentary monitor reading spikes. This time duration runs concurrently with the maximum assessment period. INDICATOR #2 addresses field survey results at the EAB. This INDICATOR is included to address reports received from field surveys initiated at lower emergency classifications. The INDICATOR is specified in terms of dose, i.e., the observed dose rate multiplied by the actual or projected release duration. INDICATOR #3 addresses results obtained from dose assessments performed with ARERAS or EPP/IP-2.6.x hand procedures. These assessments are initiated at lower classifications in response to elevated monitor readings. If the actual meteorology is more restrictive than that used to establish the monitor readings in Table 7-1, INDICATORs for lesser classifications could result in a classification under this EAL.
Escalation	Not Applicable
References	NUMARC/NESP-007, (AG1-Deviation) Rev 2, 1/92

Section 7.0	**RADIOLOGICAL / FUEL HANDLING**
TAB 7.1	**GASEOUS EFFLUENTS**
EAL 7.1.S	**Site Area Emergency**
Mode	All
Description	**EAB dose resulting from an actual or imminent release of gaseous radioactivity that exceeds 100 mR TEDE or 500 mR child thyroid CDE for the actual or projected duration of the release** *(1 or 2 or 3)* 1. A **VALID** rad monitor reading exceeds the values in Column 3 of Table 7-1 for >15 minutes, unless assessment within this period confirms that the **CRITERION** is NOT exceeded 2. Field survey results indicate EAB dose >100 mR β-γ for the actual or projected duration of the release 3. EPP dose assessment results indicate EAB dose >100 mR TEDE or >500 mR child thyroid CDE for the actual or projected duration of the release
Basis	See generic bases for this TAB The 100 mR integrated dose in the CRITERION is consistent with the 10 CFR 20.1301(a)(1) limit on the total effective dose equivalent to individual members of the public. The value is also one order of magnitude less than the CRITERION for the General Emergency which is an appropriate fraction of the EPA PAG and is consistent with the order of magnitude gradient between the General Emergency, Site Area Emergency, and Alert (i.e., 10-100-1000 mR). The 500 mR value for the thyroid was established in consideration of the 1:5 ratio of the EPA PAGs for whole body and thyroid. The child thyroid is specified here for consistency with the PAG protocol agreed upon by the states within the BVPS EPZ. INDICATOR #1 refers to a set of monitor readings that, based on annual average meteorology and assumed default source terms, correspond to the CRITERION. The time duration is included to discount momentary monitor reading spikes. INDICATOR #2 addresses field survey results at the EAB. This INDICATOR is included to address reports received from field surveys initiated at lower emergency classifications. The INDICATOR is specified in terms of dose, i.e., the observed dose rate multiplied by the actual or projected release duration INDICATOR #3 addresses results obtained from dose assessments performed with ARERAS or EPP/IP-2.6.x hand procedures. These assessments are initiated at lower classifications in response to elevated monitor readings. If the actual meteorology is more restrictive than that used to establish the monitor readings in Table 7-1, this INDICATOR could result in a higher classification than the monitor reading would otherwise indicate.
Escalation	Increases in release rate, or increases in X/Q, by a factor of 10 would escalate event.
References	NUMARC/NESP-007, (AS1-Deviation) Rev 2, 1/92

Section 7.0	**RADIOLOGICAL / FUEL HANDLING**
TAB 7.1	**GASEOUS EFFLUENTS**
EAL 7.1.A	**Alert Emergency**
Mode	All
Description	**Any UNPLANNED release of gaseous radioactivity that exceeds 200 times the radiological effluent technical specifications/Offsite Dose Calculation Manual for 15 minutes** *(1 or 2 or 3)* 1. A **VALID** rad monitor reading exceeds the values in Column 2 of Table 7-1 for >15 minutes, unless assessment within this period confirms that the **CRITERION** is NOT exceeded 2. Field survey results indicate >10 mR/hr β-γ at the EAB for 15 minutes 3. EPP dose assessment results indicate EAB dose >10 mR TEDE for the actual or projected duration of the release
Basis	See generic bases for this TAB The significance of this CRITERION is primarily related to loss of control of radioactive material that has allowed the release to continue unabated for 15 minutes. It is this aspect rather that the magnitude of the release that establishes *"...a potential substantial degradation in the level of safety of the plant..."* -- the fundamental definition of an Alert. The numeric value in the CRITERION is based on the Offsite Dose Calculation Manual (ODCM) and/or the Radiological Effluent Technical Specifications (RETS). For the Alert, the threshold is 200 times the RETS. The instantaneous dose limit (T/S 3.11.2.1.a) is 500 mR/year (0.057 mR/hr). This CRITERION equates to 200 x 0.057, or about 10 mR/hr. This value is one order of magnitude less than the CRITERION for the Site Area Emergency. INDICATOR #1 refers to monitor readings that exceed 200 times (200x) the HHSP identified on the Radioactive Waste Discharge Authorization. In order to address releases not controlled by an RWDA, column 2 Table 7-1 provides values representing 200 times the default HHSPs established in the ODCM. INDICATOR #2 addresses field survey results at the EAB. This INDICATOR is included to address reports received from field surveys initiated at lower emergency classifications. The INDICATOR is specified in terms of dose rate for the specified duration. INDICATOR #3 addresses results obtained from dose assessments performed with ARERAS or EPP/IP-2.6.x hand procedures. These assessments are initiated at lower classifications in response to elevated monitor readings. If the actual meteorology is more restrictive than that used to establish the monitor readings in Table 7-1, this INDICATOR could result in a higher classification than the monitor reading would otherwise indicate.
Escalation	Increases in release rate, or increases in X/Q, would escalate event.
References	NUMARC/NESP-007, (AA1) Rev 2, 1/92

Section 7.0	**RADIOLOGICAL / FUEL HANDLING**
TAB 7.1	**GASEOUS EFFLUENTS**
EAL 7.1.U	**Unusual Event**
Mode	All
Description	**Any UNPLANNED release of gaseous radioactivity that exceeds 2 times the radiological effluent technical specifications/Offsite Dose Calculation Manual for 60 minutes** *(1 or 2 or 3)* 1. A **VALID** rad monitor reading exceeds the values in Column 1 of Table 7-1 for >60 minutes, unless assessment within this period confirms that the **CRITERION** is NOT exceeded 2. Field survey results indicate >0.1 mR/hr β-γ at the EAB for >60 minutes 3. EPP dose assessment results indicate EAB dose >0.1 mR TEDE for the actual or projected duration of the release
Basis	See generic bases for this TAB The significance of this CRITERION is primarily related to loss of control of radioactive material that has allowed the release to continue unabated for 60 minutes. It is this aspect rather that the magnitude of the release that establishes *"...a potential degradation in the level of safety of the plant..."* -- the fundamental definition of an Unusual Event. The numeric value in the CRITERION is based on the Offsite Dose Calculation Manual (ODCM) and/or the Radiological Effluent Technical Specifications (RETS). The threshold is 2 times the RETS. The instantaneous dose limit (T/S 3.11.2.1.a) is 500 mR/year (0.057 mR/hr). This CRITERION equates to 2 x 0.057, or about 0.1 mR/hr. Releases less than 2x T/S are not reportable under 10 CFR 50.72. INDICATOR #1 refers to monitor readings that exceed 2 times (2x) the HHSP identified on the Radioactive Waste Discharge Authorization. In order to address releases not controlled by an RWDA, column 1 Table 7-1 provides values representing 2 times the default HHSPs established in the ODCM. INDICATOR #2 addresses field survey results at the EAB. This INDICATOR is included to address reports received from field surveys initiated at lower emergency classifications. The INDICATOR is specified in terms of dose rate for the specified duration. INDICATOR #3 addresses results obtained from dose assessments performed with ARERAS or EPP/IP-2.6.x hand procedures. If the actual meteorology is more restrictive than that used to establish the monitor readings in Table 7-1, this INDICATOR could result in a higher classification than the monitor reading would otherwise indicate.
Escalation	Increases in release rate, or increases in X/Q, would escalate event.
References	NUMARC/NESP-007 (AU1), Rev 2, 1/92

Section 7.0	**RADIOLOGICAL / FUEL HANDLING**
TAB 7.2	**LIQUID EFFLUENTS**
EAL	**All**
Mode	All
Description	*The following apply generically to the liquid effluent Tab:* The Radiological / Fuel Handling TAB is structured with CRITERION and INDICATORs as with the previous tabs (except Tab 1). The CRITERION establishes the numeric values for the release rate. The INDICATORs specify monitor readings that serve as thresholds for performing particular release assessments -- the results from which are then compared to the CRITERION, and appropriate declarations made. Declarations are not made on the basis of exceeding the INDICATOR threshold alone unless the specified assessment cannot be completed within 15 minutes (60 minutes for UE) of recognition. The radiation monitor readings that serve as INDICATORs for the Alert and Unusual Events, were calculated using the methodology of the Offsite Dose Calculation Manual (ODCM) which utilizes an expected nuclide mix. The use of the ODCM as a basis provides a desirable linkage to the Radiological Effluent Technical Specifications (RETS) and the Radioactive Waste Discharge Authorizations (RWDA). Assessments are performed using the liquid release assessment procedures the EPP.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev 2, 1/92

Section 7.0	**RADIOLOGICAL / FUEL HANDLING**
TAB 7.2	**LIQUID EFFLUENTS**
EAL 7.2.A	**Alert Emergency**
Mode	All
Description	**Any UNPLANNED release of liquid radioactivity that exceeds 200 times the radiological effluent technical specifications/Offsite Dose Calculation Manual for 15 minutes** *(1 or 2)* 1. A **VALID** rad monitor reading exceeds the values in Column 2 of Table 7-1 for >15 minutes, unless assessment within this period confirms that the **CRITERION** is NOT exceeded 2. Sample results exceed 200 times the radiological effluent technical specification/ Offsite Dose Calculation Manual value for an unmonitored release of liquid radioactivity >15 minutes in duration
Basis	See generic bases for this TAB The significance of this CRITERION is primarily related to loss of control of radioactive material that has allowed the release to continue unabated for 15 minutes. It is this aspect rather that the magnitude of the release that establishes "...a potential substantial degradation in the level of safety of the plant..." -- the fundamental definition of an Alert. The numeric value in the CRITERION is based on the Offsite Dose Calculation Manual (ODCM) and/or the Radiological Effluent Technical Specifications (RETS). INDICATOR #1 refers to monitor readings that exceed 200 times (200x) the HHSP identified on the Radioactive Waste Discharge Authorization. In order to address releases not controlled by an RWDA, column 2 Table 7-1 provides values representing 200 times the default HHSPs established in the ODCM. INDICATOR #2 addresses results of analyses performed on samples taken in response to unmonitored releases of liquid radioactivity. Classification in these cases will generally have to await sample results due to the lack of effluent monitoring.
Escalation	Not applicable
References	NUMARC/NESP-007, (AA1) Rev 2, 1/92

Section 7.0	**RADIOLOGICAL / FUEL HANDLING**
TAB 7.2	**LIQUID EFFLUENTS**
EAL 7.2.U	**Unusual Event**
Mode	All
Description	**Any UNPLANNED release of liquid radioactivity that exceeds 2 times the radiological effluent technical specifications/Offsite Dose Calculation Manual for 60 minutes** *(1 or 2)* 1. A **VALID** rad monitor reading exceeds the values in Column 2 of Table 7-1 for >60 minutes, unless assessment within this period confirms that the **CRITERION** is NOT exceeded 2. Sample results exceed 2 times the radiological effluent technical specification/Offsite Dose Calculation Manual value for an unmonitored release of liquid radioactivity >60 minutes in duration
Basis	See generic bases for this TAB The significance of this CRITERION is primarily related to loss of control of radioactive material that has allowed the release to continue unabated for 60 minutes. It is this aspect rather that the magnitude of the release that establishes "...a potential degradation in the level of safety of the plant..." -- the fundamental definition of an Unusual Event. The numeric value in the CRITERION is based on the Offsite Dose Calculation Manual (ODCM) and/or the Radiological Effluent Technical Specifications (RETS). INDICATOR #1 refers to monitor readings that exceed 2 times (2x) the HHSP identified on the Radioactive Waste Discharge Authorization. In order to address releases not controlled by an RWDA, column 1 Table 7-1 provides values representing 2 times the default HHSPs established in the ODCM. INDICATOR #2 addresses results of analyses performed on samples taken in response to unmonitored releases of liquid radioactivity. Classification in these cases will generally have to await sample results due to the lack of effluent monitoring.
Escalation	Increases in release rate would escalate event.
References	NUMARC/NESP-007, (AA1), Rev 2, 1/92

Section 7.0	**RADIOLOGICAL / FUEL HANDLING**
TAB 7.3	**RADIATION LEVELS**
EAL 7.3.A	**Alert**
Mode	All
Description	**UNPLANNED increases in radiation levels within the facility that impedes safe operations or establishment or maintenance of cold shutdown (1 or 2)** Unit 1 1. VALID area radiation monitor readings or survey results exceed 15 mR/hr in the Control Room or PAF (on U2 DRMS) for >15 minutes 2. (a and b) a. VALID area radiation monitor readings or survey results exceed values listed in Table 7-2 for >15 minutes b. Access restrictions impede operation of systems necessary for safe operation or the ability to establish or maintain cold shutdown. Unit 2 1. VALID area radiation monitor readings or survey results exceed 15 mR/hr in the Control Room 2RMC-RQ201/202 [1069/1072] or PAF 2RMS-RQ223 [1071] for >15 minutes 2. (a and b) a. VALID area radiation monitor readings or survey results exceed values listed in Table 7-2 for >15 minutes b. Access restrictions impede operation of systems necessary for safe operation or the ability to establish or maintain cold shutdown.
Basis	This EAL addresses conditions in which elevated radiation levels impede necessary access to operating stations, or other areas containing equipment that must be operated manually, in order to maintain safe operation or perform a safe shutdown. The significance of this EAL is with the 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 EAL. However, the Emergency Director must consider the source or cause of the increased radiation levels and determine if any other EAL may be involved. As used here "impede" includes hindering or interfering provided that the interference or delay is sufficient to significantly threaten the safe operation of the plant. Thus, for necessary actions that need to be taken within a few minutes, the need to process a radiation work permit and/or wear protective clothing would be considered as "impeding". The phrase "UNPLANNED" is specified in order to exclude anticipated, transient increases due to planned events (e.g., incore detector movement, radwaste container movement, depleted resin transfers, etc.). Con't

Section 7.0	**RADIOLOGICAL / FUEL HANDLING**
TAB 7.3	**RADIATION LEVELS**
EAL 7.3.A	**Alert**
Mode	All
Basis (Con't)	In INDICATOR #1, the 15 mR/hr value for the control room is derived from the General Design Criterion 19 value of 5 rem in 30 days with adjustment for expected occupancy times. In INDICATOR #2, the monitor readings were selected on the following basis (1) Only areas that contain systems that must be operated manually, or require local surveillances to assure reliable support of safe plant operation, are addressed. Areas having equipment that must be operated locally during an accident, and areas along the pre-designated access routes (REOPs) to those areas are specifically included. (2) For areas not normally High Radiation Areas, the threshold is 100 mR/hour. This change in dose rate designates the area as a High Radiation Area. As such, low rad area general inspection RWPs are no longer applicable. Increased survey and/or dosimetry requirements apply to High Radiation Areas. (3) For areas that are normally High Radiation Areas, the threshold is 5 R/hr. Access to areas with dose rates of this magnitude will be limited due to stay time controls.
Escalation	Not applicable
References	NUMARC/NESP-007 (AA3), Rev 2, 1/92

Section 7.0	**RADIOLOGICAL / FUEL HANDLING**
TAB 7.3	**RADIATION LEVELS**
EAL 7.3.U	**Unusual Event**
Mode	All
Description	**UNPLANNED increases in radiation levels within the facility** 1. **VALID** area radiation monitor readings increase by a factor of 1000 over normal levels for >15 minutes
Basis	This EAL addresses conditions in which there has been a degradation in the control of radioactive material, and hence, a reduction in the level of safety of the plant. The cause and/or magnitude of the increase in radiation levels is not a concern of this EAL. However, the Emergency Director must consider the source or cause of the increased radiation levels and determine if any other EAL may be involved. The phrase "UNPLANNED" is specified in order to exclude anticipated, transient increases due to planned events (e.g., incore detector movement, radwaste container movement, depleted resin transfers, etc.).
Escalation	Escalation would occur per EAL 7.3.A if the increase in radiation level results in impeded operations of equipment necessary for safe operation.
References	NUMARC/NESP-007 (AU2), Rev 2, 1/92

Section 7.0	**RADIOLOGICAL / FUEL HANDLING**
TAB 7.4	**FUEL HANDLING**
EAL 7.4.A	**Alert**
Mode	All
Description	**Major damage to irradiated fuel; or loss of water level that has or will uncover irradiated fuel outside the reactor vessel** *(1 and 2)* Unit 1 1. **VALID** HI-HI Alarm on RM-RM-203 or RM-RM-207 or RM-VS-103 A/B or RM-VS-105A/B 2. (a or b) a Plant personnel report damage of irradiated fuel sufficient to rupture fuel rods b. Plant personnel report water Level drop has or will exceed makeup capacity such that irradiated fuel will be uncovered Unit 2 1. **VALID** HI-HI Alarm on 2RMR-RQ203 [1025] or 2RMF-RQ202 [1031] or 2RMF-RQ301A/B [1032/2032] or 2HVR-RQ104A/B [1024/1028] 2. (a or b) a Plant personnel report damage of irradiated fuel sufficient to rupture fuel rods b. Plant personnel report water Level drop has or will exceed makeup capacity such that irradiated fuel will be uncovered
Basis	The major concern of the EAL is a fuel handling accident or loss of water covering spent fuel. Events away from the reactor vessel (e.g., in the cavity, transfer tube, or spent fuel pool) are addressed. Events within the vessel are classified in accordance with TABs 6.1 and 6.2, or the Fission Product Barrier Matrix. Events of this type could cause an increase in radioactivity readings and potentially a release to the environment. The magnitude of these releases is dependent on the amount of damage, depth of water above damage, and available filtration systems. Design basis fuel handling accident doses could exceed the EPA PAG, warranting a General Emergency classification. However, as with all UFSAR analyses, there is extensive conservatism in the analysis. Thus, an Alert Emergency is deemed justified. This declaration would result in augmentation of onsite personnel to support assessment of the release and restorative actions to stabilize the condition. Con't

Section 7.0	**RADIOLOGICAL / FUEL HANDLING**
TAB 7.4	**FUEL HANDLING**
EAL 7.4.A	**Alert**
Mode	All
Basis (Con't)	With regard to the loss of water level, design features and administrative controls limit the possible fuel uncovery to a single element. Analyses performed in response to IE Bulletin 84-03, showed that the clad on a fuel assembly suspended in air would begin to melt at about 60 minutes, assuming an ambient air temperature of 105 °F, which is conservative. This time period provides for event-specific assessments. Escalation of the classification would be based on the results of these assessments. INDICATOR #1 verifies the reports discussed in INDICATOR #2 by noting the increase in radiation levels, and/or airborne activity in the affected areas. An increase on the ventilation monitors signifies the release of radioactivity in the fuel gap, whereas, an increase on area radiation monitors is indicative of reduced shielding due to the decrease in water level.
Escalation	Escalation would on the basis of TAB 7.1, Gaseous Effluents
References	NUMARC/NESP-007 (AA2), Rev 2, 1/92

Section 7.0	**RADIOLOGICAL / FUEL HANDLING**
TAB 7.4	**FUEL HANDLING**
EAL 7.4.U	**Unusual Event**
Mode	All
Description	**UNPLANNED loss of water level in spent fuel pool or reactor cavity or transfer canal with fuel remaining covered** *(1 and 2 and 3)* Unit 1 1. Plant personnel report water level drop in spent fuel pool or reactor cavity or transfer canal 2. **VALID** Hi-Hi Alarm on RM-RM-203 or RM-RM-207 3. Fuel remains covered with water. Unit 2 1. Plant personnel report water level drop in spent fuel pool or reactor cavity or transfer canal 2. **VALID** Hi-Hi Alarm on 2RMR-RQ203 [1025] or 2RMF-RQ202 [1031] 3. Fuel remains covered with water.
Basis	The major concern of the EAL is a loss of water covering spent fuel. Events away from the reactor vessel (e.g., in the cavity, transfer tube, or spent fuel pool) are addressed. Events within the vessel are classified in accordance with TABs 6.1 and 6.2. Events of this type could cause an increase in radioactivity readings and potentially a release to the environment. The magnitude of these releases is dependent on the amount of damage, depth of water above damage, and available filtration systems. However, even without a release, elevated dose rates in adjacent areas could create access limitations. (See TAB 7.3) The design of fuel handling equipment and administrative controls on activities involving spent fuel maintains water above the fuel during normal handling. Should there be a loss of water level, such as that associated with a failure of the reactor cavity seal, fuel elements could be exposed to air in three locations: (1) in the manipulator mast, in the RCCA change fixture, and suspended from the fuel pool bridge crane. Analyses performed in response to IE Bulletin 84-03, showed that the clad on a fuel assembly suspended in air would begin to melt at about 60 minutes, assuming an ambient air temperature of 105 °F, which is conservative. The additional heat transfer afforded by the water assumed in this EAL would extend this time to several hours. This time period provides for event-specific assessments. Escalation of the classification would be based on the results of these assessments. Con't

Section 7.0	**RADIOLOGICAL / FUEL HANDLING**
TAB 7.4	**FUEL HANDLING**
EAL 7.4.U	**Unusual Event**
Mode	All
Basis (Con't)	INDICATOR #2 verifies the reports discussed in INDICATOR #1 by noting the increase in radiation levels in the affected areas. An increase on area radiation monitors is indicative of reduced shielding due to the decrease in water level. INDICATOR #3 is the discriminator between the Unusual Event and the Alert.
Escalation	Escalation would on the basis of TAB 7.1, Gaseous Effluents, or TAB 7.3, Radiation Levels
References	NUMARC/NESP-007 (AU2), Rev 2, 1/92

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