ENERGY NORTHWEST

INTEROFFICE MEMORANDUM

DATE: May 10, 2002

TO: Distribution

FROM: Procedure Control, Administrative Services, (927A) Wanita Deleon

SUBJECT: **PLANT PROCEDURES MANUAL - VOLUME 13**
Distribution Package: 2002 - 255

REFERENCE:
The following Procedure(s) have been revised/approved and are to be inserted in your controlled copy of the Manual and the superseded revisions are to be removed and destroyed:

Procedure	Rev.	Title/Comments
13.1.1	31	CLASSIFYING THE EMERGENCY
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13.1.1

ENERGY NORTHWEST COLUMBIA GENERATING STATION PLANT PROCEDURES MANUAL		USE CURRENT REVISION
PROCEDURE NUMBER *13.1.1	APPROVED BY JEW - Revision 31	DATE 05/10/02
VOLUME NAME EMERGENCY PLAN IMPLEMENTING PROCEDURES		
SECTION EMERGENCY CLASSIFICATION		
TITLE CLASSIFYING THE EMERGENCY		

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1.0 PURPOSE

NOTE: Refer to PPM 13.1.1.A for the Technical Bases for this procedure.

The purpose of this procedure is to provide guidelines for identifying conditions for which specific emergency classifications must be made. This procedure should be referred to whenever conditions at or near the Plant are out of the ordinary.

2.0 REFERENCES

2.1 Letter GI2-94-349, NRC to Energy Northwest, dated December 9, 1994, Emergency Action Level (EAL) Changes For Energy Northwest Nuclear Project No. 2 (WNP-2)(TAC No. M88504)

2.2 10CFR50 Appendix E, IV.B, Assessment Actions {R5727}

2.3 Internal NRC Letter, William Travors to W.D. Shafer, 6/6/88, Recovery vs. Downgrading

2.4 NUREG-1022, Rev. 1, Event Reporting Systems

2.5 NEI 99-01, Methodology for Development of Emergency Action Levels, Revision 4

2.6 Columbia Generating Station Emergency Plan, Section 4

2.7 PPM 13.1.1.A, Classifying The Emergency - Technical Bases

2.8 PPM 13.2.2, Determining Protective Action Recommendations

2.9 PPM 13.4.1, Emergency Notifications

2.10 PPM 13.10.1, Control Room Operations and Shift Manager Duties

2.11 PPM 13.10.2, TSC Manager Duties

2.12 PPM 13.11.1, EOF Manager Duties

2.13 PPM 13.13.2, Emergency Event Termination and Recovery Operations

2.14 PPM 13.13.4, After Action Reporting

3.0 RESPONSIBILITIES

3.1 Emergency Classification Responsibilities

3.1.1 Emergency Director Responsibilities

Maintain the sole responsibility for timely classification and declaration of any Columbia Generating Station emergency situation utilizing guidance specified in this procedure and the recommendations of the Operations Manager, Shift Manager, Control Room Supervisor, Shift Technical Advisor, Radiological Protection Manager, or other cognizant support personnel.

Given abnormal conditions at or near the Plant, utilize Attachment 5.1, Emergency Classification Table or the Emergency Classification Chart for guidance in determining the proper emergency classification.

The Shift Manager is responsible for the initial emergency classification and immediately becomes the Emergency Director. The TSC Manager or EOF Manager can then relieve the Shift Manager of the Emergency Director responsibilities once the Technical Support Center or Emergency Operations Facility is operational. Ultimately the Emergency Director responsibilities will reside with the EOF Manager. These responsibilities are presented in more detail in PPM 13.10.1 for the Shift Manager, PPM 13.10.2 for the TSC Manager, and PPM 13.11.1 for the EOF Manager.

3.1.2 Shift Manager Responsibilities

Function as the Emergency Director until relieved.

Maintain primary responsibility for monitoring the status of plant parameters and other initiating conditions upon which emergency classification depends.

Recommend an appropriate emergency classification to the Emergency Director, i.e., TSC Manager or EOF Manager, for any observed Columbia Generating Station emergency conditions. Utilize guidance specified in this procedure, and the recommendations of the Control Room Supervisor, Shift Technical Advisor, and Reactor Operators.

3.1.3 Control Room Supervisor, Shift Technical Advisor, Reactor Operator Responsibilities

Monitor the status of Plant parameters and other initiating conditions upon which the emergency classification depends, and inform the Shift Manager if any parameter approaches or exceeds emergency action levels as specified in this procedure. Refer to Attachment 5.1, Emergency Classification Table, or Emergency Classification Chart for guidance in determining the appropriate classification. Additional information describing the basis of each EAL may be found in PPM 13.1.1A.

3.1.4 Technical Support Center (TSC) and Emergency Operations Facility (EOF) Staff Responsibilities

Recommend an emergency classification to the Emergency Director based upon plant conditions and the guidance provided in this procedure.

3.2 Use of Plant Instruments and Indications {R5727}

Plant instrumentation described in each Emergency Action Level in Attachment 5.1 is the primary instrumentation to be used. This does not preclude use of other instruments as alternate indication, as appropriate, to properly classify the emergency.

All conditions defined within the Emergency Action Levels are to be evaluated based on the existence of valid indications.

An indication or reported condition is considered to be valid when it is conclusively verified by:

- An instrument channel check; or
- Indications on related or redundant indicators; or
- By direct observation by plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's truth is removed.

Implicit in this definition is the need for timely assessment.

3.3 Emergency Class Description

The following is a description of the four classes of emergency:

UNUSUAL EVENT: Unusual events are in progress 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 occur.

ALERT: Events are in progress 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.

SITE AREA EMERGENCY: Events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels except near the site boundary.

GENERAL EMERGENCY: Events are in progress 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 Guideline exposure levels offsite for more than the immediate site area.

3.4 Recovery vs. Downgrading Emergency Classifications

Emergency classifications should never be downgraded to a less severe classification.

It is appropriate to terminate an emergency classification of Unusual Event or Alert, and exit the classification process because the emergency condition no longer exists and the plant is stable.

When remaining as a Site Area Emergency or a General Emergency is no longer necessary, it is appropriate to terminate the classification and enter the Recovery Phase process described in 13.13.2.

3.5 Mode Applicability

The operational conditions (modes) in which Emergency Action Levels are applicable are indicated by a series of boxes as follows:

1	2	3	4	5	Def

where the numbers indicate operational conditions as defined in Technical Specifications and Def indicates "Defueled" or all fuel removed from the reactor vessel.

Mode 1 = Power Operations
Mode 2 = Startup
Mode 3 = Hot Shutdown, GT 200°, head bolts tensioned
Mode 4 = Cold Shutdown, LTE 200°
Mode 5 = Refueling

3.6 Emergency Classification Chart

Classifications can be made from either of two sources, the Emergency Classification Table, Attachment 5.1, or the Emergency Classification Chart. In order to maintain this capability, the Emergency Classification Chart must be controlled (i.e., verified, validated, and distributed) in a similar manner as the procedure, thus it is considered to be a part of the procedure. Due to its size, however, the actual chart is not physically attached to the procedure, but is instead issued to those locations that would actually need it during implementation of the procedure. As a minimum, controlled distribution of the Emergency Classification Chart is specified in Attachment 5.2.

If using the Emergency Classification Chart to classify an emergency situation, it is advisable to ensure it is the most current version by comparing the revision number on the chart to the revision number of the procedure.

3.7 Transitory Event Classification

A transitory event classification should be made whenever it is discovered that a condition had existed which met the emergency classification criteria of Attachment 5.1, but where no emergency had been declared and the basis for which no longer exists. This situation could occur due to a rapidly concluded event, an oversight in emergency classification made during the event, or through further assessments made during a post event review. Discoveries of this condition within 90 days following an event are:

a) considered to fall into this category and require the transitory event notifications to be made in accordance with PPM 13.4.1.

b) are to be included in the event's After Action Report in accordance with PPM 13.13.4. Transitory event classification discoveries made beyond 90 days of an event should only be included in the event's After Action Report. If the final After Action Report has already been completed, the discovery should be documented in an addendum to this report.

4.0 PROCEDURE

NOTE: The Emergency Action Levels described in this procedure are NOT intended to be used during maintenance and/or testing situations where abnormal temperature, pressure, or equipment status is expected. They are also mode dependent and classifications should only be made if the Plant's mode of operation at the time of the abnormal occurrence was in the range indicated for the particular initiating condition.

4.1 Initial Classification/Control Room Actions

NOTE: The Control Room Supervisor may perform the following steps if the Shift Manager is NOT in the Control Room or is incapable of performing these duties.

4.1.1 When indications of abnormal occurrences are received by the Control Room staff, the Shift Manager shall:

a. Verify the indications of the offnormal event or reported sighting.

b. Ensure that immediate actions in accordance with the Emergency Operating Procedures, Abnormal Operating Procedures, and Alarm Response Procedures are taken for the safe and proper operation of the Plant.

NOTE: Decisions should be based on conservative principles, definitions, and purposes for event classification.

c. Compare the abnormal conditions with the nine categories listed in the Emergency Classification Table, Attachment 5.1, or the Emergency Classification Chart and determine which category the event falls into.

d. Compare the information available from valid indications or reports to the Emergency Action Levels in Attachment 5.1 for the appropriate category and classify the event, using the highest level which is supported by the available information.

e. When an event is classified, the Shift Manager shall assume the Emergency Director responsibilities until properly relieved.

f. Determine whether public protective action recommendations are needed per PPM 13.2.2.

NOTE: Offsite notifications to state and local officials are required to be completed within 15 minutes of event classification. Nuclear Regulatory Commission notification should be made as soon as possible following the state and local notifications, but no longer than 60 minutes after event classification.

g. Make notifications per PPM 13.4.1.

h. Take additional actions per PPM 13.10.1.

4.2 Continued Classification/Emergency Director Actions

4.2.1 Continually evaluate the plant conditions to ensure the proper emergency classification is being utilized.

NOTE: Decisions should be based on conservative principles, definitions, and purposes for event classsification.

4.2.2 If Plant conditions change, compare the current conditions with the nine categories listed in the Emergency Classification Table, Attachment 5.1, or the Emergency Classification Chart and determine which category the conditions are applicable to.

4.2.3 Compare the information available from valid indications or reports to the Emergency Action Levels in the appropriate category and reclassify the event, using the highest level which is supported by the available information. Refer to section 3.4 for additional guidance.

4.2.4 Determine whether the public protective action recommendations need to change by checking the criteria in PPM 13.2.2.

4.2.5 Perform follow-up notifications in accordance with PPM 13.4.1.

4.3 Terminating The Emergency Classification

NOTE: PPM 13.13.2 allows direct termination of a Site Area Emergency or General Emergency classification without the formation of the Recovery Task Force, if no significant property damage has occurred.

4.3.1 If existing conditions appear to be below the minimum criteria of the Emergency Action Levels, terminate the Emergency Classification using the guidance in PPM 13.13.2.

4.3.2 Ensure reports are performed in accordance with PPM 13.13.4.

4.3.3 Ensure followup notification of offsite agencies is performed per PPM 13.4.1.

4.4 Transitory Event Classification Notification

4.4.1 If it is discovered that a classifiable event has occurred, but the basis for the classification (i.e., the Emergency Action Level condition) no longer exists; OR that a previously classified event was misclassified and the basis for the classification no longer exists, then perform the following:

a. Do NOT declare the missed classification or implement the Emergency Plan response associated with that classification.

b. Perform a transitory event classification notification to state, local and federal agencies in accordance with PPM 13.4.1.

4.4.2 For EALs with time limits associated with the condition:

a. Upon discovery of the condition, the start time for the time limit for classification begins at the time of discovery.

b. If the condition cannot be corrected within this time limit, then classification must be made.

c. If the condition can be corrected in the time limit since discovery, but the condition existed for a period of time that exceeded the time limit, then a transitory event must be declared.

4.4.3 Ensure reports are performed in accordance with PPM 13.13.4.

5.0 ATTACHMENTS

5.1 Columbia Generating Station Emergency Classification Table

5.2 Columbia Generating Station Emergency Classification Chart Distribution

CATEGORY	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
1 Reactor Fuel 1.1 Coolant Activity	Fuel Clad Degradation **1.1.U.1** 1 | 2 | 3 | 4 RCS sample activity GT Technical Specification 3.4.8 for GT LCO Action Statement time.	Loss OR potential loss of fuel clad **1.1.A.1** 1 | 2 | 3 Coolant activity GT 300 μCi/gm dose equivalent iodine.	Loss or potential loss of any two fission product barriers **1.1.S.1** 1 | 2 | 3 Coolant activity GT 300 μCi/gm dose equivalent iodine AND ANY of the following: • <u>Any</u> RCS Loss Indicators, Table 1 • Total RCS leakage GT 30 gpm inside PC or EDR-FRS-623, Pen 2 upscale high • <u>Any</u> PC Loss Indicators, Table 2 • Cannot maintain plant parameters within SRVTPLL, or PSP	A loss of any two fission product barriers and loss or potential loss of the third **1.1.G.1** 1 | 2 | 3 Coolant activity GT 300 μCi/gm dose equivalent iodine AND Drywell pressure response not consistent with LOCA conditions **1.1.G.2** 1 | 2 | 3 Coolant activity GT 300 μCi/gm dose equivalent iodine AND Total RCS leakage GT 30 gpm inside PC or EDR-FRS-623, Pen 2, upscale high AND <u>Any</u> PC Loss Indicators, Table 2 **1.1.G.3** 1 | 2 | 3 Coolant activity GT 300 μCi/gm dose equivalent iodine AND <u>Any</u> RCS Loss Indicators, Table 1 AND either of the following: <u>Any</u> PC Loss Indicators, Table 2 OR Cannot maintain plant parameters within HCTL, SRVTPLL, or PSP

Attachment 5.1

CATEGORY	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
1 Reactor Fuel 1.2 Radiation Monitors	Fuel Clad Degradation **1.2.U.1** 1 | 2 | 3 SJAE CONDSR OUTLET RAD HI-HI alarm (P602)	Loss OR potential loss of RCS **1.2.A.1** 1 | 2 | 3 Containment Radiation Monitor CMS-RIS-27E and CMS-RIS-27F reading GT 70 R/hr	Loss or potential loss of any two fission product barriers **1.2.S.1** 1 | 2 | 3 Containment Radiation Monitor CMS-RIS-27E and CMS-RIS-27F reading GT 3,600 R/hr	A loss of any two fission product barriers and loss or potential loss of the third **1.2.G.1** 1 | 2 | 3 Containment Radiation Monitor CMS-RIS-27E and CMS-RIS-27F reading GT 14,000 R/hr **1.2.G.2** 1 | 2 | 3 Containment Radiation Monitor CMS-RIS-27E and CMS-RIS-27F reading GT 3,600 R/hr AND ANY of following: • Cannot maintain plant parameters within HCTL, SRVTPLL, or PSP. • Drywell pressure response not consistent with LOCA conditions. • Any PC Loss Indicators, Table 2

CATEGORY	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
1 Reactor Fuel 1.3 Refueling Incidents	Unexpected decrease in water covering irradiated fuel assemblies **1.3.U.1** 1 2 3 4 5 def Uncontrolled water level decrease in the reactor cavity or SFP below the level of the weirs with all irradiated fuel assemblies remaining covered by water	Major damage to irradiated fuel OR loss of water level that has resulted or will result in the uncovering of irradiated fuel outside the RPV **1.3.A.1** 1 2 3 4 5 def HIGH alarm on ARM-RIS-1 (Fuel Pool ARM) resulting from an uncontrolled irradiated fuel handling process		
		Loss of water level that has or will result in the uncovering of irradiated fuel outside the reactor vessel **1.3.A.2** 1 2 3 4 5 def Water level, when not intentionally lowered, observed to be below the top of the gate sill separating the reactor cavity and the SFP **1.3.A.3** 1 2 3 4 5 def Report of visual observation of irradiated fuel uncovered or uncovering imminent.		

Attachment 5.1

CATEGORY	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
2 RPV 2.1 RPV Water Level	RCS Leakage 2.1.U.1 1 2 3 Valid unidentified leakage GE 10 gpm or upscale high indicated on recorder EDR-FRS-623, Pen 1 (P632) (Non RCC) OR Valid identified leakage GE 25 gpm indicated on recorder EDR-FRS-623, Pen 2 (P632)	Loss OR potential loss of RCS 2.1.A.1 1 2 3 Total RCS leakage GT 30 gpm inside PC or EDR-FRS-623, Pen 2 upscale high	Loss or potential loss of any two fission product barriers 2.1.S.1 1 2 3 4 5 RPV level LT -161 inches (for ATWS conditions, RPV level LT -192") or cannot be determined	A loss of any two fission product barriers and loss of potential loss of the third 2.1.G.1 1 2 3 Entry into Severe Accident Guidelines 2.1.G.2 1 2 3 RPV level LT -161 inches, (for ATWS conditions, RPV level LT -192") or cannot be determined AND ANY of following: • Rapid unexplained decrease of PC pressure following an initial increase. • Drywell pressure response not consistent with LOCA conditions • Failure of containment isolation valves (LCS Table 1.6.1.3-1) in any one line to close following auto or manual initiation AND downstream pathway outside primary containment exists

Attachment 5.1

CATEGORY	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
2 RPV 2.2 Reactivity Control	Inadvertent Criticality **2.2.U.1** 2 3 4 5 An extended and unplanned sustained positive period observed on NIs, while NOT performing a reactor startup.	Failure of Reactor Protection System (RPS) instrumentation to complete or initiate a reactor scram AND manual scram was successful. **2.2.A.1** 1 2 <u>Any</u> RPS setpoint (including manual) has been exceeded per Technical Specification 3.3.1.1 AND RPS actuation failed to result in a control rod pattern which alone always assures reactor shutdown under all conditions AND Manual Actions (mode switch in shutdown, manual push buttons and ARI) result in reactor power LE 5%.	Failure of RPS instrumentation to complete or initiate an automatic reactor scram once a RPS setpoint has been exceeded AND manual scram was NOT successful **2.2.S.1** 1 2 <u>Any</u> RPS setpoint (including manual) has been exceeded per Technical Specification 3.3.1.1 AND RPS actuation failed to result in a control rod pattern which alone always assures reactor shutdown under all conditions AND either: Reactor power GT 5% or unknown OR Wetwell temperature GT 110°F	Failure of the RPS to complete an automatic scram AND manual scram was NOT successful AND there is indication of an extreme challenge to the ability to cool the core. **2.2.G.1** 1 2 <u>Any</u> RPS setpoint (including manual) has been exceeded per Technical Specification 3.3.1.1 AND RPS actuation failed to result in a control rod pattern which alone always assures reactor shutdown under all conditions AND Wetwell temperature cannot be maintained LT the HCTL

Attachment 5.1

CATEGORY	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
3 Primary Containment 3.1 Primary Containment Pressure	Loss or potential loss of primary containment **3.1.U.1** 1 2 3 Rapid unexplained decrease of PC pressure following an initial increase	Loss or potential loss of either fuel clad or RCS **3.1.A.1** 1 2 3 Drywell pressure GT 1.68 psig with indications of RCS leakage inside drywell.	Loss or potential loss of any two fission product barriers **3.1.S.1** 1 2 3 Drywell pressure response not consistent with LOCA conditions **3.1.S.2** 1 2 3 Rapid unexplained decrease of PC pressure following an initial increase AND ANY of the following: • Total RCS leakage GT 30 gpm inside PC or EDR-FRS-623, Pen 2 upscale high • Containment Radiation Monitor CMS-RIS-27E and CMS-RIS-27F reading GT 70 R/hr • Drywell pressure GT 1.68 psig with indications of RCS leakage inside drywell.	A loss of any two fission product barriers and loss or potential loss of the third **3.1.G.1** 1 2 3 PC pressure exceeds PCPL
3.2 Wetwell Temperature/ Level	Loss or potential loss of primary containment **3.2.U.1** 1 2 3 Cannot maintain plant parameters within SRVTPLL		Loss or potential loss of any two fission product barriers **3.2.S.1** 1 2 3 RPV pressure and wetwell temperature cannot be maintained below HCTL	

CATEGORY	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
3 Primary Containment 3.2 Wetwell Temperature/Pressure			Loss or potential loss of any two fission product barriers **3.2.S.2** 1 | 2 | 3 Cannot maintain plant parameters within SRVTPLL or PSP AND ANY of the following: • Total RCS leakage GT 30 gpm inside PC or EDR-FRS-623, Pen 2 upscale high • Containment Radiation Monitor CMS-RIS-27E and CMS-RIS-27F reading GT 70 R/hr • Drywell pressure GT 1.68 psig with indications of RCS leakage inside drywell	

CATEGORY	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
3 Primary Containment 3.3 Combustible Gas Concentration				A loss of any two fission product barriers and loss or potential loss of the third 3.3.G.1 1 2 3 PC H_2 and O_2 concentrations GT 6% H_2 and 5% O_2
3.4 Containment Isolation Status	Loss or potential loss of primary containment 3.4.U.1 1 2 3 Failure of containment isolation valves (LCS Table 1.6.1.3-1) in any one line to close following auto or manual initiation AND downstream pathway outside primary containment exists	Main steam line break outside containment with isolation 3.4.A.1 1 2 3 Indications of a MSL break AND MSIV closure <u>has isolated</u> the break	Loss or potential loss of any two fission product barriers 3.4.S.1 1 2 3 Failure of containment isolation valves (LCS Table 1.6.1.3-1) in any one line to close following auto or manual initiation AND downstream pathway outside primary containment exists. AND ANY of following: • Total RCS leakage GT 30 gpm inside PC or EDR-FRS-623, Pen 2 upscale high • Containment Radiation Monitor CMS-RIS-27E and CMS-RIS-27F reading GT 70 R/hr • Drywell pressure GT 1.68 psig with indications of RCS leakage inside drywell 3.4.S.2 1 2 3 Indications of a MSL break AND MSIV closure <u>has not isolated</u> the break	A loss of any two fission product barriers and loss or potential loss of the third 3.4.G.1 1 2 3 Intentional venting of Primary Containment to prevent failure.

Attachment 5.1

CATEGORY	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
4 Secondary Containment 4.1 Reactor Building Temperature/ Radiation Levels			Loss or potential loss of any two fission product barriers 4.1.S.1 1 | 2 | 3 | | | Unisolable primary system discharging outside PC <u>resulting in</u> any area temperature or radiation level above Maximum Safe Operating Values (as defined in the Emergency Operating Procedures).	A loss of any two fission product barriers and loss of potential loss of the third 4.1.G.1 1 | 2 | 3 | | | Unisolable primary system discharging outside PC <u>resulting in</u> any area temperature or radiation level above Maximum Safe Operating Values (as defined in the Emergency Operating Procedures). AND ANY of the following: • Containment Radiation Monitor CMS-RIS-27E and CMS-RIS-27F reading GT 3,600 R/hr • RPV level LT -161 inches (for ATWS conditions, RPV level LT -192 inches) or cannot be determined • Coolant activity GT 300 μCi/gm dose equivalent iodine

CATEGORY	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
5 Radioactivity Release 5.1 Offsite Release	Any unplanned release of gaseous or liquid radioactivity to the environment that exceeds two times ODCM limits for 60 minutes or longer 5.1.U.1 1 2 3 4 5 def	Any unplanned release of gaseous or liquid radioactivity to the environment that exceeds 200 times the radiological specifications for 15 minutes or longer 5.1.A.1 1 2 3 4 5 def	Offsite dose resulting from an actual or imminent release of gaseous radioactivity that exceeds 100 mrem TEDE OR 500 mrem thyroid CDE for the actual OR projected duration of the release 5.1.S.1 1 2 3 4 5 def	Offsite dose resulting from an actual or imminent release of gaseous radioactivity that exceeds 1000 mrem total effective dose equivalent OR 5000 mrem thyroid committed dose equivalent for the actual OR projected duration of the release using actual meteorology 5.1.G.1 1 2 3 4 5 def
	A valid reading exists which exceeds or <u>is expected</u> to exceed Table 3 column "UE" for GT 60 minutes 5.1.U.2 1 2 3 4 5 def	A valid reading exists which exceeds or <u>is expected</u> to exceed Table 3 column "Alert" for GT 15 minutes 5.1.A.2 1 2 3 4 5 def	A valid reading exists which exceeds or <u>is expected</u> to exceed Table 3 column "Site Area" for GT 15 minutes 5.1.S.2 1 2 3 4 5 def	A valid reading exists which exceeds or <u>is expected</u> to exceed Table 3 column "General" for GT 15 minutes 5.1.G.2 1 2 3 4 5 def
	Offsite dose calculations indicate offsite dose rates GT Table 4 column "UE" OR Sample analysis indicates release GT 2 times ODCM 6.2.1.1 or 6.2.1.2 limits for GT 60 minutes	Offsite dose calculations indicate dose rates GT Table 4 column "Alert" OR Sample analysis indicates release GT 200 times ODCM 6.2.1.1or 6.2.1.2 limits for GT 15 minutes	Offsite dose calculations indicate doses or dose rates GT Table 4 column "Site Area" OR Field survey or survey sample analysis indicates offsite dose rates GT Table 4 column "Site Area"	Offsite dose calculations indicate doses or dose rates GT Table 4 column "General" OR Field survey or survey sample analysis indicates offsite dose rates GT Table 4 column "General".

CATEGORY	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
5 Radioactivity Release 5.2 Area Radiation	Unexpected increase in plant radiation levels **5.2.U.1** 1 | 2 | 3 | 4 | 5 | def Valid reading GT 5E3 mR/hr on ANY of the following ARMs: • ARM-RIS-4 thru ARM-RIS-18 • ARM-RIS-20 thru ARM-RIS-30 • ARM-RIS-32 thru ARM-RIS-34 (High Range)	Release of radioactive material or increases in radiation levels within the facility that impedes operation of systems required to maintain safe operation or to establish or maintain cold shutdown **5.2.A.1** 1 | 2 | 3 | 4 | 5 | def Valid reading GT 15 mR/hr on ARM-RIS-19 (CR) OR Valid reading GT 1E4 mR/hr on ANY of the following ARMs: • ARM-RIS-4 thru ARM-RIS-18 • ARM-RIS-23 • ARM-RIS-24 • ARM-RIS-32 thru ARM-RIS-34 (High Range)		

Attachment 5.1

CATEGORY	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
6 Electrical Failures 6.1 AC Power Loss	Loss of all offsite power to critical AC busses for greater than 15 minutes 6.1.U.1 | 1 | 2 | 3 | 4 | 5 | def | Power is unavailable to SM-7 and SM-8 from offsite AC sources GT 15 minutes	Loss of all offsite power and loss of all onsite power to critical AC busses for greater than 15 minutes 6.1.A.1 | | | | 4 | 5 | def | Complete loss of <u>all</u> AC power to SM-7 and SM-8 GT 15 minutes Power capability to critical AC busses reduced to a single power source for greater than 15 minutes such that any additional single failure would result in station blackout. 6.1.A.2 | 1 | 2 | 3 | | | | Available emergency bus AC power has been reduced to only one of the following sources for GT 15 min • TR-N1 (SM-7 and/or SM-8) • TR-S (SM-7 and/or SM-8) • TR-B (SM-7 and/or SM-8) • DG-1 (SM-7) • DG-2 (SM-8)	Loss of all offsite power and loss of all onsite power to critical AC busses for greater than 15 minutes 6.1.S.1 | 1 | 2 | 3 | | | | Complete loss of <u>all</u> AC power to SM-7 and SM-8 GT 15 minutes	Prolonged loss of all offsite power and prolonged loss of all onsite power to critical AC busses 6.1.G.1 | 1 | 2 | 3 | | | | Complete loss of <u>all</u> AC power to SM-7 and SM-8 AND either of the following: In the judgement of the Emergency Director, AC power to either SM-7 or SM-8 is not likely to be restored within 4 hours OR RPV level LT -161 inches (for ATWS conditions, RPV level LT -192 inches)
6.2 DC Power Loss	Degradation of all critical DC power for greater than 15 minutes. 6.2.U.1 | | | | 4 | 5 | def | Degradation of <u>both</u> Division 1 and Division 2 critical DC voltage as indicated by bus voltage LT 110 VDC on <u>both</u> 125 V Dist. Panels S1-1 and S1-2 voltmeters (Bd. C) for GT 15 minutes		Degradation of all critical DC power for greater than 15 minutes 6.2.S.1 | 1 | 2 | 3 | | | | Degradation of <u>both</u> Division 1 and Division 2 critical DC voltage as indicted by bus voltage LT 110 VDC on <u>both</u> 125 V Dist. Panels S1-1 and S1-2 voltmeters (Bd. C) for GT 15 minutes	

CATEGORY	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
7 Equipment Failures 7.1 System Failures	Inability to reach required shutdown within technical specification limits **7.1.U.1** 1 2 3 Plant is not brought to required operating mode within T.S. LCO action statement time **7.1.U.2** 1 2 3 4 5 def Uncontrolled flooding in a safe shutdown building, Table 5, that has the potential to affect safety related equipment needed for the current operating mode	Inability to maintain plant in cold shutdown **7.1.A.1** 4 5 Inability to restore and maintain reactor coolant temp LT 200°F **7.1.A.2** 1 2 3 4 5 def Report by plant personnel confirming the occurrence of plant uncontrolled internal flooding in a safe shutdown building, Table 5 AND Affected safe shutdown system parameters indicate degraded performance		
7.2 Control Room Evacuation		Control room evacuation has been initiated **7.2.A.1** 1 2 3 4 5 The decision to evacuate the Control Room has been made.	Control room evacuation has been initiated, but plant control CANNOT be established **7.2.S.1** 1 2 3 4 5 CR evacuation initiated AND Control of plant equipment needed to maintain adequate core cooling cannot be established at either the Remote Shutdown Panel or Alternate Remote Shutdown panel within 15 minutes of the SRO in charge of the CR physically leaving the CR	

CATEGORY	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
7 Equipment Failures 7.3 Loss of Indications/ Communications	Unplanned loss of most or all safety system annunciators or indication in the control room for greater than 15 minutes 7.3.U.1 1 2 3 Unplanned loss of <u>most</u> or <u>all</u> annunciators on P601, P602, P603, and Bd. C associated with safety related equipment GT 15 minutes	Unplanned loss of most or all safety system annunciators or indications in the control room with EITHER: 1) a significant transient in progress; OR 2) Compensatory non-alarming indicators are unavailable 7.3.A.1 1 2 3 Unplanned loss of <u>most</u> or <u>all</u> annunciators on P601, P602, P603, and Bd C associated with safety related equipment GT 15 minutes AND either of the following: A significant plant transient is in progress OR Compensatory non-alarming indications are <u>not</u> available (plant computer system and GDS)	Inability to monitor a significant transient in progress. 7.3.S.1 1 2 3 Loss of <u>most</u> or <u>all</u> annunciators on P601, P602, P603 and Bd. C associated with safety related equipment AND Compensatory nonalarming indications are unavailable (process computer system and GDS) AND Significant transient in progress AND Loss of indications needed to monitor ANY of the following plant critical safety parameters: • Reactor power • RPV level • RPV pressure • Drywell pressure • Drywell temperature • Wetwell pressure • Wetwell/Drywell H2/O2 Concentrations • Wetwell level • Wetwell temperature • Radioactive Gaseous Effluents	
	Significant loss of onsite OR offsite communications capabilities 7.3.U.2 1 2 3 4 5 def Unplanned loss of ALL of the following onsite communications capability: • Plant Public Address (PA) System • Plant Telephone System • Plant Radio System Operations and Security Channels			

CATEGORY	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
7 Equipment Failures 7.3 Loss of Indications/ Communications	Significant loss of onsite or offsite communications capabilities 7.3.U.3 1 | 2 | 3 | 4 | 5 | def Unplanned loss of ALL of the following offsite communications capability: • State/County Notification (CRASH) System • Offsite calling capability from the Control Room via direct telephone and fax lines • Long distance calling capability on the Plant ("2000") Switch and Kootenai (Plant Support Facility)/Deschutes (Plant Engineering Center) ("8000") Switch			

8 Independent Spent Fuel Installation 8.1 ISFSI Operations	Unexpected increase in ISFSI radiation. **8.1.U.1** **1 | 2 | 3 | 4 | 5 | def** Valid radiation reading for irradiated spent fuel in dry storage GT 2 times the ISFSI Technical Specification limits. Damage to a loaded cask confinement boundary **8.1.U.2** **1 | 2 | 3 | 4 | 5 | def** Any of the following conditions: 1) Natural phenomena events affecting a loaded cask confinement boundary: Fire, Tornado Flood, Earthquake Explosion, Lightning Complete SFSC air inlet blockage Burial under debris Extreme environmental temperature OR 2) Accident conditions affecting a loaded cask confinement boundary: Cask handling accident (e.g., drop) Cask tip-over OR 3) Any condition, in the opinion of the Emergency Director, that indicates a loss of loaded storage cask confinement boundary			

8 Independent Spent Fuel Installation 8.1 ISFSI Operations	Confirmed security event with potential loss of level of safety of the ISFSI **8.1.U.3** **1 | 2 | 3 | 4 | 5 | def** Security event as identified by the Physical Security Plan and confirmed by on shift security supervision.			

CATEGORY	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
9 Hazards 9.1 Security Threats	Confirmed security event which indicates a potential degradation in the level of safety of the plant 9.1.U.1 1 2 3 4 5 def Bomb device discovered within plant protected area <u>but</u> outside a Safe Shutdown Building, Table 5 OR Confirmed report of an attempted entry, sabotage or security threat that cannot be properly compensated for within 10 minutes 9.1.U.2 1 2 3 4 5 def Security events as defined by the Physical Security Plan AND reported by on-shift security supervision 9.1.U.3 1 2 3 4 5 def Credible notification of a security threat to Columbia Generating Station	Security event in a Plant Protected Area 9.1.A.1 1 2 3 4 5 def Confirmed report of an intrusion by a hostile force into the Plant Protected Area	Security event in a Plant Vital Area 9.1.S.1 1 2 3 4 5 def Bomb device discovered or detonated within a Safe Shutdown Building, Table 5 OR Confirmed report of intrusion by a hostile force into a Safe Shutdown Building, Table 5	Security event resulting in loss of ability to reach and maintain cold shutdown 9.1.G.1 1 2 3 4 5 def Loss of physical control of the CR due to security event OR Loss of physical control of the remote shutdown capability due to security event

CATEGORY	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
9.2 Fire/Explosion Caused by Equipment Failure	Fire within the Protected Area Boundary not extinguished within 15 minutes of detection OR an explosion within Protected Area Boundary **9.2.U.1** **1 | 2 | 3 | 4 | 5 | def** Fire within or adjacent to any Safe Shutdown Building, Table 5, which is <u>not</u> extinguished within 15 minutes of either CR notification by plant personnel or receipt of an alarm OR Report by plant personnel of an unplanned explosion within the Protected Area boundary resulting in visible damage to permanent structures or equipment	Fire or explosion affecting the operability of plant safety systems required to establish or maintain safe shutdown **9.2.A.1** **1 | 2 | 3 | 4 | 5 | def** Confirmed fire or explosion in a safe shutdown building, Table 5 AND either of the following: Affected safe shutdown system parameters indicate degraded performance OR Report by plant personnel of visible damage to the affected safe shutdown building or equipment contained within the affected safe shutdown building		

CATEGORY	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
9 Hazards 9.3 Man-Made Events	Natural and destructive phenomena affecting the Protected Area Boundary 9.3.U.1 1 2 3 4 5 def Vehicle crash into or projectile which impacts a Safe Shutdown Building, Table 5 9.3.U.2 1 2 3 Turbine failure resulting in casing penetration or damage to turbine or generator seals	Natural and destructive phenomena affecting Safe Shutdown Buildings 9.3.A.1 1 2 3 4 5 def Vehicle crash or projectile impact which impedes access to or damages equipment in a Safe Shutdown Building, Table 5 9.3.A.2 1 2 3 Missiles generated from a turbine failure have resulted in visible structural damage to or penetration of a safe shutdown building, Table 5		
	Release of toxic or flammable gases affecting the Protected Area Boundary deemed detrimental to safe operation of the plant. 9.3.U.3 1 2 3 4 5 def Report or detection of toxic or flammable gases that could enter or have entered within the Protected Area Boundary in amounts that could affect the health of plant personnel or safe plant operation OR Report by local, county or state officials for evacuation or shelter of site personnel based on offsite event	Release of toxic or flammable gases within a facility structure which jeopardizes operation of systems required to maintain safe operations or to establish or maintain cold shutdown. 9.3.A.3 1 2 3 4 5 def Report or detection of toxic or flammable gases within a safe shutdown building, Table 5, in concentrations that will be life threatening to plant personnel or impede access to equipment needed for safe plant operation.		

CATEGORY	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
9 Hazards 9.4 Natural Events	Natural and destructive phenomena affecting the Protected Area Boundary	Natural and destructive phenomena affecting Safe Shutdown Buildings		
	9.4.U.1 1 | 2 | 3 | 4 | 5 | def MINIMUM SEISMIC EARTHQUAKE alarm (H13-P851-S1-2.5) AND CR receives report from plant personnel who have felt an earthquake	**9.4.A.1** 1 | 2 | 3 | 4 | 5 | def OPERATING BASIS EARTHQUAKE alarm (H13-P851-S1-5.1) AND CR receives report from plant personnel who have felt an earthquake		
	9.4.U.2 1 | 2 | 3 | 4 | 5 | def Weather Service projected winds GT 80 mph OR CR measured winds GT 66 mph (5 minute average at 33 ft) OR Report by plant personnel confirming the occurrence of a tornado striking within the Protected Area Boundary	**9.4.A.2** 1 | 2 | 3 | 4 | 5 | def Weather Service projected winds GT 100 mph OR CR measured winds GT 76 mph (5 minute average at 33 ft) OR Report by plant personnel confirming the occurrence of a tornado striking a plant safe shutdown building, Table 5		
	9.4.U.3 1 | 2 | 3 | 4 | 5 | def Range fires near the plant which threaten to reduce the level of safety	**9.4.A.3** 1 | 2 | 3 | 4 | 5 | def Ash fallout from volcanic activity is severe enough to warrant plant shutdown		

Attachment 5.1

CATEGORY	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
9 Hazards 9.4 Natural Events	Natural and destructive phenomena affecting the Protected Area Boundary 9.4.U.4 1 | 2 | 3 | 4 | 5 | def Visible ash fallout from volcanic activity 9.4.U.5 1 | 2 | 3 | 4 | 5 | def River level increase which threatens to flood the river pumphouse	Natural and destructive phenomena affecting Safe Shutdown Buildings 9.4.A.4 1 | 2 | 3 | 4 | 5 | def Report by plant personnel of an event causing visible structural damage to a safe shutdown building, Table 5		

CATEGORY	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
10 Other 10.1 Other	Other conditions existing which, in the judgement of the Emergency Director, warrant declaration of an Unusual Event. **10.1.U.1** 1 | 2 | 3 | 4 | 5 | def In the judgement of the Emergency Director, events are in progress or have occurred, which indicate a potential degradation of the level of safety of the plant	Other conditions existing which, in the judgement of the Emergency Director, warrant declaration of an Alert. **10.1.A.1** 1 | 2 | 3 | 4 | 5 | def In the judgement of the Emergency Director, events are in progress or have occurred which indicate actual or potential substantial degradation of the level of safety of the plant	Other conditions existing which, in the judgement of the Emergency Director, warrant declaration of a Site Area Emergency **10.1.S.1** 1 | 2 | 3 | 4 | 5 | def In the judgement of the Emergency Director, events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public	Other conditions existing which, in the judgement of the Emergency Director, warrant declaration of a General Emergency **10.1.G.1** 1 | 2 | 3 | 4 | 5 | def In the judgement of the Emergency Director, other conditions exist which indicate either of the following: Actual or imminent substantial core degradation or melting with the potential for loss of containment integrity OR Potential for uncontrolled radionuclide releases which can reasonably be expected to exceed EPA PAG plume exposure levels outside the site boundary
	Loss OR potential loss of Primary Containment **10.1.U.2** 1 | 2 | 3 <u>Any</u> event, in the judgement of the Emergency Director, that could lead to or has led to a loss or potential loss of primary containment as indicted by Fission Product Barrier Degradation Table, Table 6	Loss OR potential loss of fuel clad or RCS **10.1.A.2** 1 | 2 | 3 <u>Any</u> event, in the judgement of the Emergency Director, that could lead or has led to a loss or potential loss of either fuel clad or RCS barrier as indicted by Fission Product Barrier Degradation Table, Table 6	Loss or potential loss of any two fission product barriers **10.1.S.2** 1 | 2 | 3 <u>Any</u> event, in the judgement of the Emergency Director, that could lead or has led to a loss or potential loss of any two fission product barriers as indicted by Fission Product Barrier Degradation Table, Table 6	A loss of any two fission product barriers and loss or potential loss of the third **10.1.G.2** 1 | 2 | 3 <u>Any</u> event, in the judgement of the Emergency Director, that could lead or has led to a loss of any two fission product barriers and loss or potential loss of the third as indicated by Fission Product Barrier Degradation Table, Table 6

COLUMBIA GENERATING STATION EMERGENCY CLASSIFICATION TABLE

TABLE 6 FISSION PRODUCT BARRIER DEGRADATION TABLE

Fuel Clad Loss	Fuel Clad Potential Loss	RCS Loss	RCS Potential Loss	PC Loss	PC Potential Loss
Coolant activity GT 300 μCi/gm dose equivalent iodine	RPV level LT -161 inches (for ATWS conditions, RPV level LT -192 inches)	Containment Radiation Monitor CMS-RIS-27E and CMS-RIS-27F reading GT 70 R/hr	Total RCS leakage GT 30 gpm inside PC or EDR-FRS-623, Pen 2 upscale high	Rapid unexplained decrease of PC pressure following an initial increase	Containment Radiation Monitor CMS-RIS-27E and CMS-RIS-27F reading GT 14,000 R/hr
Containment Radiation Monitor CMS-RIS-27E and CMS-RIS-27F reading GT 3,600 R/hr		RPV level LT -161 inches (for ATWS conditions, RPV level LT -192 inches)	Unisolable primary system discharging outside PC resulting in any area temperature or radiation level above Maximum Safe Operating Values (PPM 5.3.1, "Secondary Containment Control")	Drywell pressure response not consistent with LOCA conditions	PC H_2 and O_2 concentrations GT 6% H_2 and 5% O_2
Entry into Severe Accident Guidelines		Drywell pressure GT 1.68 psig with indications of RCS leakage inside drywell		Failure of containment isolation valves (LCS Table 1.6.1.3-1) in any one line to close following auto or manual initiation AND downstream pathway outside primary containment exists OR Unisolable primary system discharging outside PC resulting in any area temperature or radiation level above Maximum Safe Operating Values (PPM 5.3.1,"Secondary Containment Control")	Entry into Severe Accident Guidelines
					Loss of pressure suppression function
					Cannot maintain plant parameters within HCTL or SRVTPLL
				Intentional venting per PPM 5.2.1, "Primary Containment Control"	Wetwell pressure exceeds PSP
					PC pressure exceeds PCPL
Any event, in the judgement of the Emergency Director, that could lead or has led to a loss or potential loss of the fuel clad barrier		Any event, in the judgement of the Emergency Director, that could lead or has led to a loss or potential loss of the RCS barrier		Any event, in the judgement of the Emergency Director, that could lead to or has led to a loss or potential loss of primary containment barrier	

Table 1 RCS Barrier Loss Indicators
• Containment Radiation Monitor CMS-RIS-27E and CMS-RIS-27F reading GT 70 R/hr
• RPV level LT -161 inches (for ATWS conditions, RPV level LT -192 inches)
• Drywell pressure GT 1.68 psig with indications of RCS leakage inside drywell

Table 2 PC Barrier Loss Indicators
• Rapid unexplained decrease of PC pressure following an initial increase
• Failure of containment isolation valves (LCS Table 1.6.1.3-1) in any one line to close following auto or manual initiation AND downstream pathway outside primary containment

TABLE 3

Table 3 Effluent Monitor Classification Thresholds				
Monitor	UE	Alert	Site Area	General
NOTE: If a dose projection cannot be performed, and the monitor reading is sustained for longer than the specified time then the declaration must be made based on the valid reading.				
	60 minutes	15 minutes	15 minutes	15 minutes
PRM-RE-1B Reactor Bldg. Exhaust Inter.	1.35E4 cps	N/A	N/A	N/A
PRM-RE-1C Reactor Bldg. Exhaust Hi	N/A	1.14E3 cps	9.65E3 cps	9.35E4 cps
TEA-RIS-13 Turbine Bldg. Exhaust. Low	1.7E4 cpm	4.4E4 cpm	4.4E5 cpm	N/A
TEA-RIS-13A Turbine Bldg. Exhaust, Int.	N/A	N/A	N/A	11 PMU
WEA-RIS-14 Rad Waste Bldg. Exhaust	1.2E5 cpm	1.7E5 cpm	1.7E6 cpm	N/A
WEA-RIS-14A Rad Waste Bldg. Int.	N/A	N/A	N/A	29 PMU
TSW-RIS-5 TSW Effluent	3.9E3 cpm	3.9E5 cpm	N/A	N/A
FDR-RIS-606 Rad. Waste Effluent	2 x Hi-Hi alarm	200 x Hi-Hi alarm	N/A	N/A
SW-RIS-604 SW 'A' Process	2.0E2 cps	2.0E4 cps	N/A	N/A
SW-RIS-605 SW 'B' Process	2.0E2 cps	2.0E4 cps	N/A	N/A

TABLES 4 & 5

cps = counts per second   cpm = counts per minute   PMU = panel meter units   N/A = not applicable (outside of meter range)

Table 4 Offsite Dose Calculation/Field Survey Sample Analysis Classification Thresholds at 1.2 miles

	UE	Alert	Site Area	General
TEDE	N/A	N/A	100 mrem	1000 mrem
CDE Thyroid	N/A	N/A	500 mrem	5000 mrem
TEDE rate	0.1 mrem/hr	10 mrem/hr	100 mrem/hr (projected GT 60 min)	1000 mrem/hr (projected GT 60 min)
CDE Thyroid rate	0.3 mrem/hr	50 mrem/hr	500 mrem (for GT 1 hr inhalation)	5000 mrem (for GT 1 hr inhalation)

Table 5 Safe Shutdown Buildings

- Vital portions of the RadWaste/Control Building
- Reactor Building
- Vital portions of the Turbine Building
- Standby Service Water Pump Houses
- Diesel Generator Building
- Diesel Generator Fuel Oil Storage Area

COLUMBIA GENERATING STATION EMERGENCY CLASSIFICATION CHART DISTRIBUTION

NOTE: The Emergency Classification Chart is provided in a separate, controlled distribution to the following locations:

Location	No. Of Copies
Control Room	2
Technical Support Center	2
Emergency Operations Facility	2
Control Room Simulator	2
Remote Shutdown Room	1
Simulator Remote S/D Room	1

Attachment 5.2

13.1.1A

ENERGY NORTHWEST COLUMBIA GENERATING STATION PLANT PROCEDURES MANUAL	USE CURRENT REVISION

PROCEDURE NUMBER	APPROVED BY	DATE
*13.1.1A	JEW - Revision 9	05/10/02

VOLUME NAME

EMERGENCY PLAN IMPLEMENTING PROCEDURES

SECTION

EMERGENCY CLASSIFICATION

TITLE

CLASSIFYING THE EMERGENCY - TECHNICAL BASES

TABLE OF CONTENTS

1.0 PURPOSE

The purpose of this procedure is to provide Plant Emergency Response Organization personnel tasked with classifying the emergency the technical bases for the Emergency Action Levels (EAL) listed in PPM 13.1.1, "Classifying The Emergency".

It is not necessary to refer to this procedure to classify the emergency. The emergency classification may be determined by utilizing the guidance contained within PPM 13.1.1. However, Plant Emergency Response Organization personnel may refer to this procedure for supplemental information or clarification of EAL interpretation.

2.0 REFERENCES

2.1 Columbia Generating Station Technical Specifications

2.2 Columbia Generating Station Offsite Dose Calculation Manual

2.3 Columbia Generating Station FSAR Section 1.5.2, SBO Coping Study

2.4 Columbia Generating Station FSAR Section 8.3.2.1, Batteries

2.5 Columbia Generating Station FSAR Table 9.5-8, Location of Emergency Lighting

2.6 Columbia Generating Station FSAR Chapter 3.2, Classification of Structures, Components and Systems

2.7 Calculation CE-02-93-16 (Wind Speed Triggers for UE/Alert Declaration)

2.8 Calculation 2.05.01 (Battery Size Calc)

2.9 NEI 99-01, rev. 4, "Methodology for Development of Emergency Action Levels"

2.10 NUMARC/NRC, "Questions & Answers", June 1993

2.11 NUREG/CR-4982, “Severe Accident in Spent Fuel Pools in Support of Generic Safety Issue 82”

2.12 NUREG-0737, “Clarification of TMI Action Plan Requirements”

2.13 10CFR20, Standards for Protection Against Radiation

2.14 10CFR50, Domestic Licensing of Production and Utilization Facilities

2.15 EPA 400, “Manual of Protective Action Guidelines and Protective Actions for Nuclear Incidents”

2.16 FSAR, Chapter 13.3, Columbia Generating Station Emergency Plan, Section 6

2.17 NUREG-0654/FEMA-REP-1, Rev. 1, Appendix 1 ("Basis For Emergency Action Levels For Nuclear Power Facilities")

2.18 Columbia Generating Station Safeguards Contingency Plan

2.19 ABN-CR-EVAC, Control Room Evacuation and Remote Cooldown

2.20 ABN-FLOODING, Flooding

2.21 ABN-WIND, Tornado/High Winds

2.22 PPM 5.0.10, EOP Flowchart Training Manual

2.23 PPM 5.1.2, RPV Control-ATWS

2.24 PPM 5.7.1, RPV & Primary Containment Flooding Severe Accident Guidelines

3.0 PROCEDURE

3.1 Plant Emergency Response Personnel Responsibilities

Plant Emergency Response Personnel may refer to the Technical Bases in this procedure for emergency classification and after-the-fact review of emergency action levels.

3.2 Use Of Plant Instruments And Indications

Plant instrumentation described in each EAL in Attachment 4.1 is the primary instrumentation to be used. This does not preclude use of other instruments as alternate indication, as appropriate, to properly classify the emergency.

All conditions defined within the EALs are to be evaluated based on the existence of valid indications. An indication or reported condition is considered to be valid when it is conclusively verified by:

- An instrument channel check; or
- Indications on related or redundant indicators; or
- By direct observation by plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's truth is removed.

Implicit in this definition is the need for timely assessment.

3.3 Emergency Class Description

The following is a description of the four classes of emergency:

UNUSUAL EVENT: 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.

ALERT: 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.

SITE AREA EMERGENCY: Events are in process or have occurred which involve actual or likely major failures of plant functions needed for protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels except near the site boundary.

GENERAL EMERGENCY: 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 Guideline exposure levels offsite for more than the immediate site area.

3.4 Mode Applicability

The operational conditions (modes) in which Initiating Conditions are applicable are indicated by a series of boxes as follows:

Operating Conditions	1	2	3	4	5	def

where the numbers indicate operational conditions as defined in Technical Specifications and Def indicates "Defueled" or all fuel removed from the reactor vessel.

4.0 ATTACHMENTS

4.1 EAL Technical Bases

4.2 Fission Product Barrier Degradation Table Bases

1 REACTOR FUEL

The reactor fuel cladding serves as the primary fission product barrier. Over the useful life of a fuel bundle, the integrity of this barrier should remain intact as long as fuel cladding integrity limits are not exceeded.

Should fuel damage occur (breach of the fuel cladding integrity) radioactive fission products are released to the reactor coolant. The magnitude of such a release is dependent upon the extent of the damage as well as the mechanism by which the damage occurred. Once released into the reactor coolant, the highly radioactive fission products can pose significant radiological hazards in plant from reactor coolant process streams. If other fission product barriers were to fail, these radioactive fission products can pose significant offsite radiological consequences.

The following parameters/indicators are indicative of possible fuel failures:

- Coolant Activity: During normal operation, reactor coolant fission product activity is very low. Small concentrations of fission products in the coolant are primarily from either the fission of tramp uranium in the fuel cladding or minor perforations in the cladding itself. Any significant increase from these base-line levels is indicative of fuel failures.

- Radiation Monitors: As with coolant activity, any fuel failures will release fission products to the reactor coolant. Those products which are gaseous or volatile in nature will be carried over with the steam and eventually be detected by the air ejector off-gas radiation monitors. Although not a direct indication or measurement of fuel damage, exceeding predetermined limits on containment high range radiation monitors under LOCA conditions is indicative of possible fuel failures. In addition, this indicator is utilized as an indicator of RCS loss and potential containment loss.

- Refueling Incidents: Both area and process radiation monitoring systems designed to detect fission products during refueling conditions as well as visual observation can be utilized to indicate loss or potential loss of spent fuel cladding integrity .

1 Reactor Fuel 1.1 Coolant Activity

1.1.U.1 Unusual Event

NUMARC IC: SU4 - Fuel Clad Degradation.

APPLICABILITY:

Operating Conditions

1	2	3	4		

EMERGENCY ACTION LEVEL:

RCS sample activity GT Technical Specification 3.4.8 for GT LCO Action Statement time.

BASES:

This Initiating Condition is considered to be a potential degradation in the level of safety of the plant and a potential precursor of more serious problems.

Coolant activity in excess of allowable Technical Specifications reflects a degraded or degrading core condition and represent a decrease in plant safety.

REFERENCE(S):

Columbia Generating Station Technical Specifications

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Unusual Event SU4

Columbia Generating Station Plant Specific EAL Guideline, SU4.2

1 Reactor Fuel 1.1 Coolant Activity

1.1.A.1 Alert

NUMARC IC: Loss or potential loss of fuel clad

APPLICABILITY:

Operating Conditions	1	2	3			

EMERGENCY ACTION LEVEL:

Coolant activity GT 300 μCi/gm dose equivalent iodine

BASES:

This EAL is indicative of the loss of the fuel clad barrier. Fuel Clad barrier damage is indicated by a coolant activity of 300 μCi/gm dose equivalent I-131. This amount of activity is well above that expected for iodine spikes and corresponds to approximately 2-5% fuel clad failure in accordance with assessment performed by the NUMARC EAL task force. This amount of clad failure indicates significant clad heating and, thus, the Fuel Clad barrier is considered lost.

REFERENCE(S):

Columbia Generating Station Technical Specifications

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Fission Product Barrier Basis Information for Table 3

Columbia Generating Station Fission Product Barrier Evaluation

Columbia Generating Station Plant Specific EAL Guideline, FC1.1

1 Reactor Fuel 1.1 Coolant Activity

1.1.S.1 Site Area Emergency

NUMARC IC: Loss or potential loss of any two fission product barriers

APPLICABILITY:

Operating Conditions

1	2	3			

EMERGENCY ACTION LEVEL:

Coolant activity GT 300 μCi/gm dose equivalent iodine

AND ANY of the following:

- Any RCS Loss Indicators, Table 1
- Total RCS leakage GT 30 gpm inside PC or EDR-FRS-623, Pen 2 upscale high
- Any PC Loss Indicators, Table 2
- Cannot maintain plant parameters within SRVTPLL or PSP

BASES:

This EAL represents loss of the fuel clad barrier (coolant activity GT 300 μCi/gm dose equivalent iodine) in combination with indications of a loss of RCS (Table 1) or potential loss of RCS (Total RCS leakage GT 30 gpm inside PC or EDR-FRS-623, Pen 2 upscale high) or loss of PC (Table 2) or potential loss of PC (Cannot maintain plant parameters within SRVTPLL or PSP).

Fuel Clad barrier damage is indicated by a coolant activity of 300 μCi/gm dose equivalent I-131. This amount of activity is well above that expected for iodine spikes and corresponds to approximately 2-5% fuel clad failure in accordance with assessment performed by the NUMARC EAL task force. This amount of clad failure indicates significant clad heating and, thus, the Fuel Clad barrier is considered lost.

Total leakage is considered to be the total of both identified and unidentified leakage as measured on EDR-FRS-623 pen 1 (unidentified - Floor Drain Sump Fill Rate) and pen 2 (identified - Equipment Drain Sump Fill Rate). The maximum measurable identified leak rate (pen 2) in the Control Room at Columbia Generating Station is 30 gpm, therefore 30 gpm is used instead of the 50 gpm limit recommended by NUMARC.

The inability to maintain plant parameters within the SRVTPLL or PSP represents a potential loss of PC.

The SRV Tail Pipe Level Limit (SRVTPLL) is the highest wetwell water level at which the opening of an SRV will not result in exceeding the code allowable stresses in the tailpipe, tailpipe supports, quenchers or quencher supports. This level is a function of RPV pressure and the Limit is utilized to preclude SRV system failure and containment failure. The consequences of operating SRVs when wetwell water level exceeds SRVTPLL may include direct pressurization of the containment from a break in the SRV tailpipe. The resulting primary containment pressurization could cause containment failure.

Entry into the unsafe region of the Pressure Suppression Pressure curve (PPM 5.2.1, "Primary Containment Control", Figure F, PSP) is included as a potential primary containment barrier loss. A rapid depressurization of the RPV (e.g., occurrence of a large break LOCA or initiation of ADS) at wetwell pressures in excess of the PSP may cause either:

- Wetwell pressure responses indicative of failure in the drywell-to-wetwell boundary, or
- Wetwell pressure increases to or beyond the Primary Containment Pressure Limit (PPM 5.2.1, "Primary Containment Control", Figure B, PCPL).

Refer to Attachment 4.2 for the bases of each of the following referenced barrier loss/potential loss indicators.

Table 1 RCS Barrier Loss Indicators
• Containment Radiation Monitor CMS-RIS-27E and CMS-RIS-27F reading GT 70 R/hr • RPV level LT -161 in. (For ATWS conditions, RPV level LT -192 inches) • Drywell pressure GT 1.68 psig with indications of RCS leakage inside drywell

Table 2 PC Barrier Loss Indicators
• Rapid unexplained decrease of PC pressure following an initial increase • Failure of containment isolation valves (LCS Table 1.6.1.3-1) in any one line to close following auto or manual initiation AND downstream pathway outside primary containment exists

REFERENCE(S):

Columbia Generating Station Technical Specifications

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Fission Product Barrier Basis Information for Table 3

Columbia Generating Station Fission Product Barrier Evaluation

Columbia Generating Station Plant Specific EAL Guideline, FC1.1 + RCS3.1, RCS2.1, RCS1.2, PC1.1, PC2.1

1 Reactor Fuel 1.1 Coolant Activity

1.1.G.1 General Emergency

NUMARC IC: A loss of any two fission product barriers and loss or potential loss of the third

APPLICABILITY:

Operating Conditions	1	2	3			

EMERGENCY ACTION LEVEL:

Coolant activity GT 300 μCi/gm dose equivalent iodine
AND
Drywell pressure response not consistent with LOCA conditions

BASES:

This EAL represents loss of the fuel clad barrier (coolant activity GT 300 μCi/gm dose equivalent iodine) in combination with an indication of a loss of both RCS and PC (Drywell pressure response not consistent with LOCA conditions).

Fuel Clad barrier damage is indicated by a coolant activity of 300 μCi/gm dose equivalent I-131. This amount of activity is well above that expected for iodine spikes and corresponds to approximately 2-5% fuel clad failure in accordance with assessment performed by the NUMARC EAL task force. This amount of clad failure indicates significant clad heating and, thus, the Fuel Clad barrier is considered lost.

Containment or drywell pressure responses not consistent with LOCA conditions indicate a loss of the Primary Containment barrier. This may be noticed as a decrease in drywell pressure when no operation action (e.g., starting drywell cooling fans) has been taken. It would also include a failure of the drywell pressure to increase as expected during a LOCA.

REFERENCE(S):

Columbia Generating Station Technical Specifications

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Fission Product Barrier Basis Information for Table 3

Columbia Generating Station Fission Product Barrier Evaluation

Columbia Generating Station Plant Specific EAL Guideline, FC1.1 + PC1.2

1 Reactor Fuel 1.1 Coolant Activity

1.1.G.2 General Emergency

NUMARC IC: A loss of any two fission product barriers and loss or potential loss of the third

APPLICABILITY:

Operating Conditions	1	2	3			

EMERGENCY ACTION LEVEL:

Coolant activity GT 300 μCi/gm dose equivalent iodine
AND
Total RCS leakage GT 30 gpm inside PC or EDR-FRS-623, Pen 2 upscale high
AND
Any PC Loss Indicators, Table 2

BASES:

This EAL represents loss of the fuel clad barrier (coolant activity GT 300 μCi/gm dose equivalent iodine) in combination with an indication of a potential loss of RCS (Total RCS leakage GT 30 gpm inside PC or EDR-FRS-623, Pen 2 upscale high) and loss of PC (Any PC Loss Indicators, Table 2).

Fuel Clad barrier damage is indicated by a coolant activity of 300 μCi/gm dose equivalent I-131. This amount of activity is well above that expected for iodine spikes and corresponds to approximately 2-5% fuel clad failure in accordance with assessment performed by the NUMARC EAL task force. This amount of clad failure indicates significant clad heating and, thus, the Fuel Clad barrier is considered lost.

Total leakage is considered to be the total of both identified and unidentified leakage as measured on EDR-FRS-623 pen 1 (unidentified - Floor Drain Sump Fill Rate) and pen 2 (identified - Equipment Drain Sump Fill Rate). The maximum measurable identified leak rate (pen 2) in the Control Room at Columbia Generating Station is 30 gpm, therefore 30 gpm is used instead of the 50 gpm limit recommended by NUMARC.

Refer to Attachment 4.2 for the bases of each of the following PC barrier loss indicators:

Table 2 PC Barrier Loss Indicators
• Rapid unexplained decrease of PC pressure following an initial increase • Failure of containment isolation valves (LCS Table 1.6.1.3-1) in any one line to close following auto or manual initiation AND downstream pathway outside primary containment exists

REFERENCE(S):

Columbia Generating Station Technical Specifications

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Fission Product Barrier Basis Information for Table 3

Columbia Generating Station Fission Product Barrier Evaluation

Columbia Generating Station Plant Specific EAL Guideline, FC1.1 + RCS1.2 + PC1.1, PC2.1

1 Reactor Fuel 1.1 Coolant Activity

1.1.G.3 General Emergency

NUMARC IC: A loss of any two fission product barriers and loss or potential loss of the third

APPLICABILITY:

Operating Conditions	1	2	3			

EMERGENCY ACTION LEVEL:

Coolant activity GT 300 μCi/gm dose equivalent iodine
AND
Any RCS Loss Indicators, Table 1
AND either of the following:
Any PC Loss Indicators, Table 2
OR
Cannot maintain plant parameters within HCTL, SRVTPLL, or PSP

BASES:

This EAL represents loss of the fuel clad barrier (coolant activity GT 300 μCi/gm dose equivalent iodine) in combination with any of the Table 1 RCS loss indications and either a potential loss of PC (Cannot maintain plant parameters within HCTL, SRVTPLL, or PSP) or any of the Table 2 PC loss indicators.

Fuel Clad barrier damage is indicated by a coolant activity of 300 μCi/gm dose equivalent I-131. This amount of activity is well above that expected for iodine spikes and corresponds to approximately 2-5% fuel clad failure in accordance with assessment performed by the NUMARC EAL task force. This amount of clad failure indicates significant clad heating and, thus, the Fuel Clad barrier is considered lost.

The Heat Capacity Temperature Limit (HCTL) is the highest wetwell temperature at which initiation of RPV depressurization will not result in exceeding the Primary Containment Pressure Limit (PCPL) before the rate of energy transfer to the containment is within the capacity of the containment vent. The HCTL is used to preclude failure of the containment or equipment necessary for safe shutdown by assuring that RPV blowdown does not cause containment pressure to exceed the PCPL. The potential loss occurs when RPV pressure and wetwell temperature cannot be maintained below HCTL.

The SRV Tail Pipe Level Limit (SRVTPLL) is the highest wetwell water level at which opening of an SRV will not result in exceeding the code allowable stresses in the tailpipe, tailpipe supports, quenchers or quencher supports. This level is a function of RPV pressure and the Limit is utilized to preclude SRV system failure and containment failure. The consequences of operating SRVs when wetwell water level exceeds the SRVTPLL may include direct pressurization of the containment from a break in the SRV tail pipe. The resulting primary containment pressurization could cause containment failure.

Entry into the unsafe region of the Pressure Suppression Pressure curve (PPM 5.2.1, "Primary Containment Control", Figure F, PSP) is included as a potential primary containment barrier loss. A rapid depressurization of the RPV (e.g., occurrence of a large break LOCA or initiation of ADS) at wetwell pressures in excess of the PSP may cause either:

- Wetwell pressure responses indicative of a failure in the drywell-to-wetwell boundary, or
- Wetwell pressure increases to or beyond the Primary Containment Pressure Limit (PPM 5.2.1, "Primary Containment Control", Figure B, PCPL).

Refer to Attachment 4.2 for the bases of each of the following referenced barrier loss/potential loss indicators.

Table 1 RCS Barrier Loss Indicators
• Containment Radiation Monitor CMS-RIS-27E and CMS-RIS-27F reading GT 70 R/hr • RPV level LT -161 in. (For ATWS conditions, RPV level LT -192 inches) • Drywell pressure GT 1.68 psig with indications of RCS leakage inside drywell

Table 2 PC Barrier Loss Indicators
• Rapid unexplained decrease of PC pressure following an initial increase • Failure of containment isolation valves (LCS Table 1.6.1.3-1) in any one line to close following auto or manual initiation AND downstream pathway outside primary containment exists

REFERENCE(S):

Columbia Generating Station Technical Specifications

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Fission Product Barrier Basis Information for Table 3

Columbia Generating Station Fission Product Barrier Evaluation

Columbia Generating Station Plant Specific EAL Guideline, FC1.1 + (RCS2.1, 3.1, 4.1) + (PC1.1, PC2.1, PC 1.5, PC5.2)

1 Reactor Fuel 1.2 Radiation Monitors

1.2.U.1 Unusual Event

NUMARC IC: SU4 - Fuel Clad Degradation

APPLICABILITY:

Operating Conditions

1	2	3			

EMERGENCY ACTION LEVEL:

SJAE CONDSR OUTLET RAD HI-HI alarm (P602)

BASES:

This Initiating Condition is considered to be a potential degradation in the level of safety of the plant and a potential precursor of more serious problems.

SJAE CONDSR OUTLET RAD HI HI monitor and alarm reflects the offgas effluent and, therefore, may be one of the first indicators of degrading fuel conditions. The alarm is confirmed by verification of greater than current alarm setpoint on Recorder OG RIS-612 on Panel P604 or high offgas pre treatment air activity [determined by sample results] greater than limits specified in Technical Specification 3.7.5. The Hi alarm setpoint corresponds to a fraction of the Technical Specification limit thereby alerting the plant of the need to sample prior to exceeding limits.

REFERENCE(S):

Columbia Generating Station Technical Specifications

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Unusual Event SU4

Instrument Master Data Sheet(s) for instrument(s) listed in the EAL(s)

Columbia Generating Station Plant Specific EAL Guideline, SU4.2

1 Reactor Fuel 1.2 Radiation Monitors

1.2.A.1 Alert

NUMARC IC: Loss or potential loss of RCS

APPLICABILITY:

Operating Conditions

1	2	3			

EMERGENCY ACTION LEVEL:

Containment Radiation Monitor CMS-RIS-27E and CMS-RIS-27F reading GT 70 R/hr

BASES:

This indicator is considered to be a loss of the RCS barrier.

A 70 R/hr reading on CMS-RIS-27E and CMS-RIS-27F is used to indicate a loss of the Reactor Coolant System barrier. This value assumes a 0.1% clad damage and the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory into the drywell atmosphere. The value of 0.1% clad damage was assumed to be the greatest amount of fuel failure under which power operation could occur.

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Fission Product Barrier Basis Information for Table 3

Columbia Generating Station Fission Product Barrier Evaluation

Engineering Calculation No. NE-02-94-57

Columbia Generating Station Plant Specific EAL Guideline, RCS3.1

1 Reactor Fuel 1.2 Radiation Monitors

1.2.S.1 Site Area Emergency

NUMARC IC: Loss or potential loss of any two fission product barriers

APPLICABILITY:

Operating Conditions	1	2	3			

EMERGENCY ACTION LEVEL:

Containment Radiation Monitor CMS-RIS-27E and CMS-RIS-27F reading GT 3,600 R/hr

BASES:

This indicator is considered to be a loss of both the RCS and fuel clad barriers.

A 3,600 R/hr reading on CMS-RIS-27E and CMS-RIS-27F is used to indicate a release of reactor coolant, with elevated activity indicative of fuel damage, into the drywell. Therefore, this condition represents loss of both the fuel clad and RCS barriers. This value assumes an instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with coolant concentration associated with 5% clad failures into the drywell atmosphere. Columbia Generating Station has elected to provide an example dealing with the top end of the 2-5% range discussed in NESP-007. This value assumes an instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory into the drywell atmosphere.

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Fission Product Barrier Basis Information for Table 3

Columbia Generating Station Fission Product Barrier Evaluation

Engineering Calculation No. NE-02-94-57

Columbia Generating Station Plant Specific EAL Guideline, FC3.1 (RCS3.1)

1 Reactor Fuel 1.2 Radiation Monitors

1.2.G.1 General Emergency

NUMARC IC: A loss of any two fission product barriers and loss or potential loss of the third

APPLICABILITY:

Operating Conditions

1	2	3			

EMERGENCY ACTION LEVEL:

Containment Radiation Monitor CMS-RIS-27E and CMS-RIS-27F reading GT 14,000 R/hr

BASES:

This indicator is considered to be a loss of both the RCS and fuel clad barriers with the potential loss of PC.

A 14,000 R/hr reading on CMS-RIS-27E and CMS-RIS-27F is used to indicate potential failure of the primary containment barrier. It is a value that indicates significant fuel damage well in excess of that associated with the loss of both Fuel Clad and RCS barriers. A major release of radioactivity requiring offsite protective actions is not possible unless a major failure of fuel cladding allows radioactive material to be released from the core into the reactor coolant. Regardless of whether containment is challenged, this amount of activity in containment, if released, could have such severe consequences that it is prudent to treat this as a potential loss of containment, such that a General Emergency declaration is warranted. This level of activity is indicative of approximately 20% clad failure. This value assumes an instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory into the drywell atmosphere.

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Fission Product Barrier Basis Information for Table 3

Columbia Generating Station Fission Product Barrier Evaluation

Engineering Calculation No. NE-02-94-57

Columbia Generating Station Plant Specific EAL Guideline, PC3.1 (FC3.1, RCS3.1)

1 Reactor Fuel 1.2 Radiation Monitors

1.2.G.2 General Emergency

NUMARC IC: A loss of any two fission product barriers and loss or potential loss of the third

APPLICABILITY:

Operating Conditions

1	2	3			

EMERGENCY ACTION LEVEL:

Containment Radiation Monitor CMS-RIS-27E and CMS-RIS-27F reading GT 3,600 R/hr AND ANY of following:

- Cannot maintain plant parameters within HCTL, SRVTPLL, or PSP
- Drywell pressure response not consistent with LOCA conditions
- Any PC Loss Indicators, Table 2

BASES:

This combination of conditions is considered to be a loss of both the RCS and fuel clad barriers with the loss or potential loss of PC.

A 3,600 R/hr reading on CMS-RIS-27E and CMS-RIS-27F is used to indicate a release of reactor coolant, with elevated activity indicative of fuel damage, into the drywell. Therefore, this condition represents loss of both the fuel clad and RCS barriers. This value assumes an instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with coolant concentration associated with 5% clad failures into the drywell atmosphere. Columbia Generating Station has elected to provide an example dealing with the top end of the 2-5% range discussed in NESP-007. This value assumes an instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory into the drywell atmosphere.

The Heat Capacity Temperature Limit (HCTL) is the highest wetwell temperature at which initiation of RPV depressurization will not result in exceeding the Primary Containment Pressure Limit (PCPL) before the rate of energy transfer to the containment is within the capacity of the containment vent. The HCTL is used to preclude failure of the containment or equipment necessary for safe shutdown by assuring that RPV blowdown does not cause containment pressure to exceed the PCPL. The potential loss occurs when RPV pressure and wetwell temperature cannot be maintained below HCTL.

The SRV Tail Pipe Level Limit (SRVTPLL) is the highest wetwell water level at which opening of an SRV will not result in exceeding the code allowable stresses in the tailpipe, tailpipe supports, quenchers or quencher supports. This level is a function of RPV pressure and the Limit is utilized to preclude SRV system failure and containment failure. The consequences of operating SRVs when wetwell water level exceeds the SRVTPLL may include direct pressurization of the containment from a break in the SRV tail pipe. The resulting primary containment pressurization could cause containment failure.

Entry into the unsafe region of the Pressure Suppression Pressure curve (PPM 5.2.1, "Primary Containment Control", Figure F, PSP) is included as a potential primary containment barrier loss. A rapid depressurization of the RPV (e.g., occurrence of a large break LOCA or initiation of ADS) at wetwell pressures in excess of the PSP may cause either:

- Wetwell pressure responses indicative of a failure in the drywell-to-wetwell boundary, or
- Wetwell pressure increases to or beyond the Primary Containment Pressure Limit (PPM 5.2.1, "Primary Containment Control", Figure B, PCPL).

Containment or drywell pressure responses not consistent with LOCA conditions indicate a loss of the Primary Containment barrier. This may be noticed as a decrease in drywell pressure when no operation action (e.g., starting drywell cooling fans) has been taken. It would also include a failure of the drywell pressure to increase as expected during a LOCA.

Refer to Attachment 4.2 for the bases of each of the following referenced PC barrier loss indicators.

Table 2 PC Barrier Loss Indicators
• Rapid unexplained decrease of PC pressure following an initial increase • Failure of containment isolation valves (LCS Table 1.6.1.3-1) in any one line to close following auto or manual initiation AND downstream pathway outside primary containment exists

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Fission Product Barrier Basis Information for Table 3

Columbia Generating Station Fission Product Barrier Evaluation

Engineering Calculation No. NE-02-94-57

Columbia Generating Station Plant Specific EAL Guideline, FC3.1 + PC1.5, PC5.2, PC1.1, PC1.2, PC2.1

1 Reactor Fuel 1.3 Refueling Incidents

1.3.U.1 Unusual Event

NUMARC IC: AU2 - Unexpected decrease in water covering irradiated fuel assemblies

APPLICABILITY:

Operating Conditions

1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

Uncontrolled water level decrease in the reactor cavity or SFP below the level of the weirs with all irradiated fuel assemblies remaining covered by water

BASES:

The SFP cooling system weirs were chosen as a readily identifiable level. In addition to being readily visible from the refueling floor, dropping below the level of the weir, only a few inches, impairs the operability of the SFP cooling system with subsequent alarms.

These events tend to have long lead times relative to potential for radiological release outside the site boundary thus, the impact to public health and safety is very low.

Uncontrolled water level decrease may result in unplanned increases in in plant radiation levels represent a degradation in the control of radioactive material and represent a potential degradation in the level of safety of the plant. This EAL escalates to an ALERT per EAL 1.3.A.2 if the water level drops lower to the level of the gates.

REFERENCE(S):

Columbia Generating Station Technical Specifications

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Unusual Event AU2

Columbia Generating Station Plant Specific EAL Guideline, AU2.1, AU2.2

1 Reactor Fuel 1.3 Refueling Incidents

1.3.A.1 Alert

NUMARC IC: AA2 - Major damage to irradiated fuel or loss of water level that has resulted or will result in the uncovering of irradiated fuel outside the RPV

APPLICABILITY:

Operating Conditions	1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

HIGH alarm on ARM-RIS-1 (Fuel Pool ARM) resulting from an uncontrolled irradiated fuel handling process

BASES:

Due to the decreased amount of decay heat present, there is time available to take corrective actions and little potential for substantial fuel damage. In addition, NUREG/CR-4982, "Severe Accident in Spent Fuel Pools in Support of Generic Safety Issue 82," July 1987, indicates that even if corrective actions are not taken, no prompt fatalities are predicted, and that risk of injury is low.

The setpoint for the listed ARM is given in PPM 4.602.A5 1-1. After review of refuel floor radiation monitoring capability it was determined that three of the four ARMs included in this EAL should not be listed for the following reasons.

ARM-RIS-3 and ARM-RIS-3A detect radiation near the bottom of the new fuel storage pit and, therefore, would not provide adequate indication of radiation levels associated with decreasing water level above irradiated fuel.

ARM-RIS-2, Fuel Pool Area Radiation Monitor, alarm setpoint is arbitrarily adjusted to a level slightly above normal background to monitor operator performance during fuel handling and would, therefore, not be indicative of a potential refueling accident. Its high alarm setpoint is typically 15 mrem/hr.

The high alarm on ARM-RIS-1, Fuel Pool Area Radiation Monitor, is the threshold condition for this EAL. Its setpoint is nominally 300 mrem/hr. The EAL is worded so that the alarm must be the result of an uncontrolled irradiated fuel handling process avoiding an unnecessary declaration if the condition were caused by a spurious alarm signal or condition not related to fuel handling.

REFERENCE(S):

Columbia Generating Station Technical Specifications

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, AA2

NUREG/CR-4982, Severe Accident in Spent Fuel Pools in Support of Generic Safety, Issue 82, July 1987

Columbia Generating Station Plant Specific EAL Guideline, AA2.1

1 Reactor Fuel 1.3 Refueling Incidents

1.3.A.2 Alert

NUMARC IC: AA2 - Loss of water level that has or will result in the uncovering of irradiated fuel outside the reactor vessel.

APPLICABILITY:

Operating Conditions	1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

Water level, when not intentionally lowered, observed to be below the top of the gate sill separating the reactor cavity and the SFP

BASES:

Unintentional water level decrease may result in unplanned increases in in plant radiation levels which represent a degradation in the control of radioactive material and represent a potential degradation in the level of safety of the plant.

No level indication is available in Reactor Cavity or spent fuel pools for monitoring pool level outside of a narrow band around the normal operating range. In lieu of using a visual observation of pool level below top of fuel racks, the bottom of the water gates between the pools were chosen as a reference point for classification purposes. The intent of this EAL is to allow observations from plant personnel to be factored into the declaration and is not intended to direct an entry into an area solely to observe pool level.

REFERENCE(S):

Columbia Generating Station Technical Specifications

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, AA2

NUREG 0818, Emergency Action Levels for Light Water Reactors

NUREG/CR-4982, Severe Accident in Spent Fuel Pools in Support of Generic Safety, Issue 82, July 1987

Columbia Generating Station Plant Specific EAL Guideline, AA2.4

1.3.A.3 Alert

NUMARC IC: AA2 - Loss of water level that has or will result in the uncovering of irradiated fuel outside the reactor vessel.

APPLICABILITY:

Operating Conditions	1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

Report of visual observation of irradiated fuel uncovered or uncovering imminent

BASES:

Unintentional water level decrease may result in unplanned increases in in plant radiation levels which represent a degradation in the control of radioactive material and represent a potential degradation in the level of safety of the plant.

No level indication is available in Reactor Cavity or spent fuel pools for monitoring pool level outside of a narrow band around the normal operating range. The intent of this EAL is to allow observations from plant personnel to be factored into the declaration and is not intended to direct an entry into an area solely to observe pool level. The terminology "uncovering imminent" is intended to address conditions in which water level decrease mitigative actions are not being successful.

REFERENCE(S):

Columbia Generating Station Technical Specifications

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, AA2

NUREG 0818, Emergency Action Levels for Light Water Reactors

NUREG/CR-4982, Severe Accident in Spent Fuel Pools in Support of Generic Safety, Issue 82, July 1987

Columbia Generating Station Plant Specific EAL Guideline, AA2.2

2 REACTOR PRESSURE VESSEL (RPV)

The reactor pressure vessel provides a volume for the coolant which covers the reactor core. The RPV and associated pressure piping (reactor coolant system) together provide a barrier to limit the release of radioactive material should the reactor fuel cladding integrity fail.

There are two RPV parameters which are indicative of conditions which may pose a threat to RPV or fuel cladding integrity:

- RPV Water Level: RPV water level is directly related to the status of adequate core cooling, and therefore fuel cladding integrity. Excessive reactor coolant to drywell leakage indications are utilized to indicate potential pipe cracks which may propagate to an extent threatening fuel clad, RPV and primary containment integrity. Conditions under which all attempts at establishing adequate core cooling have failed require primary containment flooding.

- Reactivity Control: The inability to control reactor power below certain levels can pose a direct threat to reactor fuel, RPV and primary containment integrity.

2 RPV 2.1 RPV Water Level

2.1.U.1 Unusual Event

NUMARC IC: SU5 - RCS Leakage

APPLICABILITY:

Operating Conditions	1	2	3			

EMERGENCY ACTION LEVEL:

Valid unidentified leakage GE 10 gpm or upscale high indicated on recorder EDR-FRS-623, Pen 1 (P632) (non-RCC)
OR
Valid identified leakage GE 25 gpm indicated on recorder EDR-FRS-623, Pen 2 (P632)

BASES:

This Initiating Condition may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation in the level of safety of the plant. The value for identified leakage is set at a higher value due to the lesser significance of identified leakage in comparison to unidentified and pressure boundary leakage. RCC is not considered part of RCS leakage for this EAL.

REFERENCE(S):

Columbia Generating Station Technical Specifications

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Unusual Event SU5

Columbia Generating Station Plant Specific EAL Guideline, SU5.1

2 RPV 2.1 RPV Water Level

2.1.A.1 Alert

NUMARC IC: Loss or potential loss of RCS

APPLICABILITY:

Operating Conditions

1	2	3			

EMERGENCY ACTION LEVEL:

Total RCS leakage GT 30 gpm inside PC or EDR-FRS-623, Pen 2 upscale high

BASES:

Total leakage is considered to be the total of both identified and unidentified leakage as measured on EDR-FRS-623 pen 1 (unidentified - Floor Drain Sump Fill Rate) and pen 2 (identified - Equipment Drain Sump Fill Rate). The maximum measurable identified leak rate (pen 2) in the Control Room at Columbia Generating Station is 30 gpm, therefore 30 gpm is used instead of the 50 gpm limit recommended by NUMARC.

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Fission Product Barrier Basis Information for Table 3

Columbia Generating Station Fission Product Barrier Evaluation

Columbia Generating Station Plant Specific EAL Guideline, RCS1.2

2 RPV 2.1 RPV Water Level

2.1.S.1 Site Area Emergency

NUMARC IC: Loss or potential loss of any two fission product barriers

APPLICABILITY:

Operating Conditions	1	2	3	4	5	

EMERGENCY ACTION LEVEL:

RPV level LT -161 inches (for ATWS conditions, RPV level LT -192") or cannot be determined.

BASES:

This EAL represents a loss of RCS barrier and potential loss of fuel clad. Indicating fuel clad barrier potential loss at -161″ (Top of Active Fuel) or -192" (for ATWS conditions) ensures that an emergency is declared before fuel perforation occurs. A level decrease to below -161″ or -192" (for ATWS conditions) is also indicative of a large RCS break, or a smaller break with loss of high pressure makeup. This also represents conditions where RPV level cannot be determined, such as entry into PPM 5.1.4, RPV Flooding.

During ATWS conditions, level may be intentionally lowered to reduce reactor power. This level decrease is not indicative of an RCS break as long as level is maintained above -192" (minimum steam cooling RPV water level).

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Fission Product Barrier Basis Information for Table 3

Columbia Generating Station Fission Product Barrier Evaluation

Columbia Generating Station Plant Specific EAL Guideline, SS5.1, FC2.2/RCS4.1

2 RPV 2.1 RPV Water Level

2.1.G.1 General Emergency

NUMARC IC: A loss of any two fission product barriers and loss or potential loss of the third

APPLICABILITY:

Operating Conditions

1	2	3			

EMERGENCY ACTION LEVEL:

Entry into Severe Accident Guidelines

BASES:

This EAL represents loss of both fuel clad and RCS in conjunction with a potential loss of containment.

Entry into Severe Accident Guidelines is indicative of both a loss of fuel clad and RCS barriers because:

1) RPV water level cannot be restored and maintained above -161 inches, or

2) For ATWS conditions, RPV water level cannot be maintained above -192″, or

3) If RPV water level cannot be determined, RPV flooding for ATWS or non-ATWS conditions cannot be established or maintained.

During ATWS conditions, level may be intentionally lowered to reduce reactor power. This level decrease is not indicative of an RCS break as long as level is maintained above -192" (minimum steam cooling RPV water level).

Entry into PPM 5.7.1, "RPV & Primary Containment Flooding Severe Accident Guidelines" is indicative of a potential loss of primary containment because actions to flood the containment may jeopardize the pressure suppression capability of the containment or result in the need to vent the RPV or primary containment.

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Fission Product Barrier Basis Information for Table 3

Columbia Generating Station Fission Product Barrier Evaluation

Columbia Generating Station Plant Specific EAL Guideline, FC2.1/PC4.1

2 RPV 2.1 RPV Water Level

2.1.G.2 General Emergency

NUMARC IC: A loss of any two fission product barriers and loss or potential loss of the third

APPLICABILITY:

Operating Conditions

1	2	3			

EMERGENCY ACTION LEVEL:

RPV level LT -161 inches (for ATWS conditions, RPV level LT -192") or cannot be determined AND ANY of following:

- Rapid unexplained decrease of PC pressure following an initial increase
- Drywell pressure response not consistent with LOCA conditions
- Failure of containment isolation valves (LCS Table 1.6.1.3-1) in any one line to close following auto or manual initiation AND downstream pathway outside primary containment exists

BASES:

This EAL represents loss of both RCS and PC in conjunction with a potential loss of fuel clad.

Indicating fuel clad barrier potential Loss at -161″ (Top of Active Fuel) or -192" (for ATWS conditions) ensures that an emergency is declared before fuel perforation occurs. An unintentional level decrease to below -161″ (or -192" during ATWS conditions) is also indicative of a large RCS break, or a smaller break with loss of high pressure makeup.

During ATWS conditions, level may be intentionally lowered to reduce reactor power. This level decrease is not indicative of an RCS break as long as level is maintained above -192" (minimum steam cooling RPV water level).

Rapid unexplained loss of pressure (i.e., not attributable to drywell spray or condensation effects) following initial pressure increase indicates a loss of containment integrity. In interpreting this EAL, an initial increase is any PC pressure increase above 1.68 psig. A rapid decrease should be considered any decrease which occurs faster than the initial increase.

Containment or drywell pressure responses not consistent with LOCA conditions indicate a loss of the Primary Containment barrier. This may be noticed as a decrease in drywell pressure when no operation action (e.g., starting drywell cooling fans) has been taken. It would also include a failure of the drywell pressure to increase as expected during a LOCA.

The failure to isolate condition is intended to cover containment isolation failures allowing a direct flow path to the environment such as a failure of MSIVs to close with open valves downstream to the turbine or condenser. Downstream path outside primary containment does not mean leakage incident to normal system integrity. Only those penetrations required to isolate per Technical Specifications should be considered.

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Fission Product Barrier Basis Information for Table 3

Columbia Generating Station Fission Product Barrier Evaluation

Columbia Generating Station Plant Specific EAL Guideline, FC2.2/RCS4.1 + PC1.1, PC1.2, PC2.1

2 RPV 2.2 Reactivity Control

2.2.U.1 Unusual Event

NUMARC IC: SU8 - Inadvertent Criticality. An extended and unplanned positive period or sustained positive period observed on nuclear instrumentation while not performing a reactor startup.

APPLICABILITY:

Operating Conditions		2	3	4	5	

EMERGENCY ACTION LEVEL:

An extended and unplanned sustained positive period observed on NIs while NOT performing a reactor startup.

BASES:

Inadvertent criticalities indicate a potential degradation of the level of safety of the plant. This EAL excludes inadvertent criticalities that occur during planned reactivity changes associated with reactor startups, such as achieving criticality earlier than estimated.

The term extended is used in order to allow exclusion of expected short term positive periods from planned fuel bundle or control rod movement during core alterations.

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, final draft rev 3, Unusual Event SU8.

2 RPV 2.2 Reactivity Control

2.2.A.1 Alert

NUMARC IC: SA2 - Failure of Reactor Protection System (RPS) instrumentation to complete or initiate a reactor scram AND manual scram was successful.

APPLICABILITY:

Operating Conditions	1	2				

EMERGENCY ACTION LEVEL:

Any RPS setpoint (including manual) has been exceeded per T.S. 3.3.1.1
AND
RPS actuation failed to result in a control rod pattern which alone always assures reactor shutdown under all conditions
AND
Manual actions (mode switch in shutdown, manual push buttons, and ARI) result in reactor power LE 5%.

BASES:

This condition indicates a failure of the automatic protection system to scram the reactor sufficient to achieve shutdown under all conditions without boron but the reactor was successfully manually scrammed sufficient to achieve shutdown under all conditions without boron.

A manual scram is any set of actions by the reactor operator(s) which results in a scram. These actions include placing the reactor mode switch in shutdown, depressing the RPS Manual Scram push buttons and/or placing ARI switches to trip.

Failure of a manual scram to reduce reactor power below APRM downscale levels or resulting in exceeding 110 degrees F in the suppression pool would escalate this event to Site Area Emergency 2.2.S.1.

REFERENCE(S):

Columbia Generating Station Technical Specifications

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Alert SA2

Columbia Generating Station Emergency Operating Procedure Flowchart Training Manual, PPM 5.0.10

Columbia Generating Station Plant Specific EAL Guideline, SA2.1

2 RPV 2.2 Reactivity Control

2.2.S.1 Site Area Emergency

NUMARC IC: SS2 - Failure of RPS instrumentation to complete or initiate an automatic reactor scram once a RPS setpoint has been exceeded AND manual scram was NOT successful.

APPLICABILITY:

Operating Conditions

1	2				

EMERGENCY ACTION LEVEL:

Any RPS setpoint (including manual) has been exceeded per T.S. 3.3.1.1
AND
RPS actuation failed to result in a control rod pattern which alone always assures reactor shutdown under all conditions
AND either:
Reactor power GT 5% or unknown
OR
Wetwell temperature GT 110 °F

BASES:

This condition indicates a failure of both the automatic protection system and manual efforts to scram the reactor with continued power generation.

A manual scram is any set of actions by the reactor operator(s) which results in a scram. These actions include placing the reactor mode switch in shutdown, depressing the RPS Manual Scram push buttons and/or placing ARI switches to trip. Injection of boron is not considered in reducing reactor power below 5%. A concurrent challenge to the ability to cool the core or a significant challenge to decay heat removal capability would escalate this event to General Emergency 2.2.G.1.

As specified in the NUMARC/NESP-007 Questions and Answers, January 1993, ". . . a scram is considered unsuccessful if it does not result in achieving a state in which the reactor will remain shutdown under all conditions without boron injection." For 2.2.A.1, if manual actions result in the reactor being shutdown under all conditions without boron injection, an ALERT is declared. Escalation to a Site Area Emergency (2.2.S.1) is not required. If sufficient control rods are not inserted to reduce reactor power to below the APRM downscale setpoints, an immediate Site Area Emergency (2.2.S.1) is declared. If the APRM downscale setpoint is achieved, but suppression pool temperature is greater than Boron Injection Initiation Temperature (110°F), a precursor exists for a threat to containment and thus a "Site Area Emergency is warranted."

APRM downscale trip setpoint for Columbia Generating Station is 5%, the Boron Injection Initiation Temperature is defined as 110°F. The conditions ". . . control rod pattern which alone always assures reactor shutdown" is defined to mean that the reactor is shutdown under all conditions without boron.

REFERENCE(S):

Columbia Generating Station Emergency Operating Procedure RPV Control-ATWS, PPM 5.1.2

Columbia Generating Station Technical Specifications

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Site General Emergency SS2

Columbia Generating Station Plant Specific EAL Guideline, SS2.1

2 RPV 2.2 Reactivity Control

2.2.G.1 General Emergency

NUMARC IC: SG2 - Failure of the RPS to complete an automatic scram AND manual scram was NOT successful AND there is indication of an extreme challenge to the ability to cool the core.

APPLICABILITY:

Operating Conditions

1	2				

EMERGENCY ACTION LEVEL:

Any RPS setpoint (including manual) has been exceeded per T.S. 3.3.1.1
AND
RPS actuation failed to result in a control rod pattern which alone always assures reactor shutdown under all conditions
AND
Wetwell temperature cannot be maintained LT the HCTL

BASES:

This condition indicates a failure of both the automatic protection system and manual efforts at Control Room Panel P-603 to scram the reactor concurrent with a challenge to the ability to cool the core or a significant challenge to decay heat removal capability.

Heat removal capability is extremely challenged if the wetwell temperature cannot be maintained below the Heat Capacity Temperature Limit curve.

A manual scram is any set of actions by the reactor operator(s) which results in a scram. These actions include placing the reactor mode switch in shutdown, depressing the RPS Manual Scram push buttons and/or placing ARI switches to trip. Injection of boron is not considered in reducing reactor power below 5%.

REFERENCE(S):

Columbia Generating Station Emergency Operating Procedure RPV Control-ATWS, PPM 5.1.2

Columbia Generating Station Emergency Operating Procedure Primary Containment Flooding, PPM 5.1.7

Columbia Generating Station Technical Specifications

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, General Emergency SG2

Columbia Generating Station Plant Specific EAL Guideline, SG2.1

3 PRIMARY CONTAINMENT (PC)

The primary containment structure is a pressure suppression system. It forms a fission product barrier designed to limit the release of radioactive fission products generated from any postulated accident so as to preclude exceeding offsite exposure limits.

The primary containment structure is a low leakage pressure suppression system housing the reactor pressure vessel (RPV), the reactor coolant recirculation piping and other branch connections of the reactor primary system. The primary containment is equipped with isolation valves for most systems which penetrate the containment boundary. These valves automatically actuate to isolate systems under emergency conditions.

There are four primary containment parameters which are indicative of conditions which may pose a threat to primary containment integrity or indicate degradation of RPV or reactor fuel integrity.

- Primary Containment Pressure: Excessive primary containment pressure is also indicative of either primary system leaks into containment or loss of containment cooling function. Primary containment pressures at or above specified limits pose a direct threat to primary containment integrity and the pressure suppression function.

- Wetwell Temperature/Level: Excessive suppression pool water temperatures or abnormally high or low wetwell levels can result in a loss of the pressure suppression capability of containment and thus be indicative of severely degraded RPV and containment conditions.

- Combustible Gas Concentration: The existence of combustible gas concentrations in containment pose a severe threat to containment integrity and are indicative of severely degraded reactor core and RPV conditions.

- Containment Isolation Status: The existence of an unisolable steam line break outside containment constitutes a loss of containment integrity as well as a loss of RCS boundary. Should a loss of fuel cladding integrity occur, the potential for release of large amounts of radioactive materials to the environment exists.

3 Primary Containment 3.1 Primary Containment Pressure

3.1.U.1 Unusual Event

NUMARC IC: Loss or potential loss of primary containment

APPLICABILITY:

Operating Conditions

1	2	3			

EMERGENCY ACTION LEVEL:

Rapid unexplained decrease of PC pressure following an initial increase

BASES:

This indicator is considered to be a loss of the PC barrier.

Rapid unexplained loss of pressure (i.e., not attributable to drywell spray or condensation effects) following initial pressure increase indicates a loss of containment integrity. In interpreting this EAL, an initial increase is any PC pressure increase above 1.68 psig. A rapid decrease should be considered any decrease which occurs faster than the initial increase.

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Fission Product Barrier Basis Information for Table 3

Columbia Generating Station Fission Product Barrier Evaluation

Columbia Generating Station Plant Specific EAL Guideline, PC1.1

3 Primary Containment 3.1 Primary Containment Pressure

3.1.A.1 Alert

NUMARC IC: Loss or potential loss of either fuel clad or RCS

APPLICABILITY:

Operating Conditions

1	2	3			

EMERGENCY ACTION LEVEL:

Drywell pressure GT 1.68 psig with indications of RCS leakage inside drywell

BASES:

This indicator is considered to be a loss of the RCS barrier.

The 1.68 psig drywell pressure for the Reactor Coolant System barrier loss is based on the drywell pressure scram and isolation setpoint and indicates a Loss of Coolant Accident (LOCA). A potential loss of the Reactor Coolant System barrier would not result in an increasing drywell pressure and, therefore, no indicator is provided. The qualifier of "with indications of RCS leak inside drywell" is included as an indicator of RCS boundary degradation and eliminates a drywell pressure increase due to a loss of drywell ventilation.

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Fission Product Barrier Basis Information for Table 3

Columbia Generating Station Fission Product Barrier Evaluation

Columbia Generating Station Plant Specific EAL Guideline, RCS2.1

3 Primary Containment 3.1 Primary Containment Pressure

3.1.S.1 Site Area Emergency

NUMARC IC: Loss or potential loss of any two fission product barriers

APPLICABILITY:

Operating Conditions	1	2	3			

EMERGENCY ACTION LEVEL:

Drywell pressure response not consistent with LOCA conditions

BASES:

This indicator is considered to be a loss of both the RCS and PC barriers.

Containment or drywell pressure responses not consistent with LOCA conditions indicate a loss of the Primary Containment barrier. This may be noticed as a decrease in drywell pressure when no operation action (e.g., starting drywell cooling fans) has been taken. It would also include a failure of the drywell pressure to increase as expected during a LOCA. Also, a loss of suppression function in conjunction with a LOCA would indicate a loss of the Primary Containment barrier. Exceeding PSP is an indication of loss of pressure suppression function.

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Fission Product Barrier Basis Information for Table 3

Columbia Generating Station Fission Product Barrier Evaluation

Columbia Generating Station Plant Specific EAL Guideline, PC2.1

3 Primary Containment 3.1 Primary Containment Pressure

3.1.S.2 Site Area Emergency

NUMARC IC: Loss or potential loss of either fuel clad or RCS barrier in conjunction with a loss of containment

APPLICABILITY:

Operating Conditions	1	2	3			

EMERGENCY ACTION LEVEL:

Rapid unexplained decrease of PC pressure following an initial increase
AND ANY of the following:

- Total RCS leakage GT 30 gpm inside PC or EDR-FRS-623, Pen 2 upscale high
- Containment Radiation Monitor CMS-RIS-27E and CMS-RIS-27F reading GT 70 R/hr
- Drywell pressure GT 1.68 psig with indications of RCS leakage inside drywell

BASES:

This indicator is considered to be a loss of both the RCS and PC barriers.

Rapid unexplained loss of pressure (i.e., not attributable to drywell spray or condensation effects) following initial pressure increase indicates a loss of containment integrity. In interpreting this EAL, an initial increase is any PC pressure increase above 1.68 psig. A rapid decrease should be considered any decrease which occurs faster than the initial increase.

Total leakage is considered to be the total of both identified and unidentified leakage as measured on EDR-FRS-623 pen 1 (unidentified - Floor Drain Sump Fill Rate) and pen 2 (identified - Equipment Drain Sump Fill Rate). The maximum measurable identified leak rate (pen 2) in the Control Room at Columbia Generating Station is 30 gpm, therefore 30 gpm is used instead of the 50 gpm limit recommended by NUMARC.

A 70 R/hr reading on CMS-RIS-27E and CMS-RIS-27F is used to indicate a loss of the Reactor Coolant System barrier. This value assumes a 0.1% clad damage and the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory into the drywell atmosphere. The value of 0.1% clad damage was assumed to be the greatest amount of fuel failure under which power operation could occur.

The 1.68 psig drywell pressure for the Reactor Coolant System barrier loss is based on the drywell pressure scram and isolation setpoint and indicates a Loss of Coolant Accident (LOCA). A potential loss of the Reactor Coolant System barrier would not result in an increasing drywell pressure and, therefore, no indicator is provided. The qualifier of "with indications of RCS leak inside drywell" is included as an indicator of RCS boundary degradation and eliminates a drywell pressure increase due to a loss of drywell ventilation.

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Fission Product Barrier Basis Information for Table 3

Columbia Generating Station Fission Product Barrier Evaluation

Columbia Generating Station Plant Specific EAL Guideline, PC1.1 + RCS1.2, RCS3.1, RCS2.1

3 Primary Containment 3.1 Primary Containment Pressure

3.1.G.1 General Emergency

NUMARC IC: A loss of any two fission product barriers and loss or potential loss of the third

APPLICABILITY:

Operating Conditions

1	2	3			

EMERGENCY ACTION LEVEL:

PC pressure exceeds PCPL

BASES:

This indicator is considered to be a loss of both the Fuel Clad and RCS barriers in conjunction with the potential loss of PC.

Containment pressures that exceeds 34.7 psig, the maximum expected pressure following a LOCA, have the potential to result in a loss of the containment barrier. Preparations to vent containment are required by PPM 5.2.1, "Primary Containment Control" before the Wetwell pressure reaches the Primary Containment Pressure Limit (PCPL). Therefore this condition is considered a potential loss of containment.

With PC pressure GT PCPL and increasing, a loss of the RCS barrier has occurred due to the elevated containment pressure. Continued wetwell pressure increase could result in complete and uncontrolled loss of the primary containment due to containment failure. With no assurance as to where the containment may fail, an attendant loss of the suppression pool should be assumed with a consequent complete and unrecoverable loss of core cooling whereby the degraded core condition and loss of containment integrity releases substantial amounts of radioactivity to the general environment. Therefore, this condition is also considered to be a loss of both fuel clad and RCS.

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Fission Product Barrier Basis Information for Table 3

Columbia Generating Station Fission Product Barrier Evaluation

Columbia Generating Station Plant Specific EAL Guideline, PC1.3

3 Primary Containment 3.2 Wetwell Temperature/Level

3.2.U.1 Unusual Event

NUMARC IC: Loss or potential loss of primary containment

APPLICABILITY:

Operating Conditions	1	2	3			

EMERGENCY ACTION LEVEL:

Cannot maintain plant parameters within SRVTPLL

BASES:

This indicator is considered to be a potential loss of the PC barrier.

The SRV Tail Pipe Level Limit (SRVTPLL) is the highest wetwell water level at which opening of an SRV will not result in exceeding the code allowable stresses in the tailpipe, tailpipe supports, quenchers or quencher supports. This level is a function of RPV pressure and the Limit is utilized to preclude SRV system failure and containment failure. The consequences of operating SRVs when wetwell water level exceeds the SRVTPLL may include direct pressurization of the containment from a break in the SRV tail pipe. The resulting primary containment pressurization could cause containment failure.

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Fission Product Barrier Basis Information for Table 3

Columbia Generating Station Fission Product Barrier Evaluation

Columbia Generating Station Plant Specific EAL Guideline, PC1.5

3 Primary Containment 3.2 Wetwell Temperature/Level

3.2.S.1 Site Area Emergency

NUMARC IC: Loss or potential loss of any two fission product barriers

SS4 - Complete loss of functions needed to achieve and maintain hot shutdown.

APPLICABILITY:

Operating Conditions	1	2	3			

EMERGENCY ACTION LEVEL:

RPV pressure and wetwell temperature cannot be maintained below HCTL

BASES:

This indicator is considered to be a loss of the RCS in conjunction with a potential loss of the PC barrier. This condition also is representative of a loss of ability to maintain the plant in hot shutdown.

This EAL addresses complete loss of functions, including ultimate heat sink and reactivity control, required for hot shutdown with the reactor at pressure and temperature. Under these conditions, there is an actual major failure of a system intended for protection of the public. Thus, declaration of a Site Area Emergency is warranted. Escalation to a General Emergency would occur by Abnormal Rad Levels/Radiological Effluent, Fission Product Barrier Degradation, or Emergency Director Judgment Initiating Conditions.

Functions required for hot shutdown consist of the ability to achieve reactor shutdown and to discharge decay heat energy from the reactor to the ultimate heat sink. Appropriate emergency declarations required by the inability to achieve reactor shutdown are addressed by EALs 2.2.A.1, 2.2.S.1 and 2.2.G.1. Inability to remove decay heat energy is reflected in an increase in suppression pool temperature. Elevated suppression pool temperature is addressed by the Heat Capacity Temperature Limit (HCTL). The HCTL is a function of RPV pressure and suppression pool temperature. If RPV pressure and suppression pool temperature cannot be maintained below the HCTL, the ultimate heat sink is threatened and declaration of a Site Area Emergency is warranted.

The Heat Capacity Temperature Limit (HCTL) is the highest wetwell temperature at which initiation of RPV depressurization will not result in exceeding the Primary Containment Pressure Limit (PCPL) before the rate of energy transfer to the containment is within the capacity of the containment vent. The HCTL is used to preclude failure of the containment or equipment necessary for safe shutdown by assuring that RPV blowdown does not cause containment pressure to exceed the PCPL. The potential loss occurs when RPV pressure and wetwell temperature cannot be maintained below HCTL.

This indicator is also considered to be a loss of RCS barrier in that a wetwell temperature rise sufficient to exceed the HCTL could only be the result of a sustained discharge of a primary system into the wetwell.

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Fission Product Barrier Basis Information for Table 3

Columbia Generating Station Fission Product Barrier Evaluation

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Site Area Emergency, SS4

Columbia Generating Station Plant Specific EAL Guideline, SS4.1

3 Primary Containment 3.2 Wetwell Temperature/Level

3.2.S.2 Site Area Emergency

NUMARC IC: Loss or potential loss of any two fission product barriers

APPLICABILITY:

Operating Conditions	1	2	3			

EMERGENCY ACTION LEVEL:

Cannot maintain plant parameters within SRVTPLL, or PSP
AND ANY of the following:

- Total RCS leakage GT 30 GPM inside PC or EDR-FRS-623, Pen 2 upscale high
- Containment Radiation Monitor CMS-RIS-27E and CMS-RIS-27F reading GT 70 R/hr
- Drywell pressure GT 1.68 psig with indications of RCS leakage inside drywell

BASES:

This indicator is considered to be a potential loss of PC in combination with a loss or potential loss of RCS barrier.

The inability to maintain plant parameters within the SRVTPLL, or PSP represents a potential loss of PC.

The SRV Tail Pipe Level Limit (SRVTPLL) is the highest wetwell water level at which the opening of an SRV will not result in exceeding the code allowable stresses in the tailpipe, tailpipe supports, quenchers or quencher supports. This level is a function of RPV pressure and the Limit is utilized to preclude SRV system failure and containment failure. The consequences of operating SRVs when wetwell water level exceeds SRVTPLL may include direct pressurization of the containment from a break in the SRV tailpipe. The resulting primary containment pressurization could cause containment failure.

Entry into the unsafe region of the Pressure Suppression Pressure curve (PPM 5.2.1, "Primary Containment Control", Figure F, PSP) is included as a potential primary containment barrier loss. A rapid depressurization of the RPV (e.g., occurrence of a large break LOCA or initiation of ADS) at wetwell pressures in excess of the PSP may cause either:

- Wetwell pressure responses indicative of failure in the drywell-to-wetwell boundary, or
- Wetwell pressure increases to or beyond the Primary Containment Pressure Limit (PPM 5.2.1, "Primary Containment Control", Figure B, PCPL).

Total leakage is considered to be the total of both identified and unidentified leakage as measured on EDR-FRS-623, Pen 1 (unidentified Floor Drain Sump Fill Rate) and Pen 2 (identified Equipment Drain Sump Fill Rate). The maximum measurable identified leak rate (Pen 2) in the Control Room at Columbia Generating Station is 30 gpm, therefore, 30 gpm is used instead of the 50 gpm limit recommended by NUMARC.

A 70 R/hr reading on CMS-RIS-27e and CMS-RIS-27F is used to indicate a loss of the Reactor Coolant System barrier. This value assumes a 0.1% clad damage and the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory into the drywell atmosphere. The value of 0.1% clad damage was assumed to be the greatest amount of fuel failure under which power operation could occur.

The 1.68 psig drywell pressure for the Reactor Coolant System barrier loss is based on the drywell pressure scram and isolation setpoint and indicates a Loss of Coolant Accident (LOCA). A potential loss of the Reactor Coolant System barrier would not result in an increasing drywell pressure and, therefore, no indicator is provided. The qualifier of "with indications of RCS leak inside the drywell" is included as an indicator of RCS boundary degradation and eliminates a drywell pressure increase due to loss of drywell ventilation.

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Fission Product Barrier Basis Information for Table 3

Columbia Generating Station Fission Product Barrier Evaluation

Columbia Generating Station Plant Specific EAL Guideline, PC1.5/PC5.2 + RCS1.2, RCS3.1, RCS2.1

3 Primary Containment 3.3 Combustible Gas Concentrations

3.3.G.1 General Emergency

NUMARC IC: A loss of any two fission product barriers and loss or potential loss of the third

APPLICABILITY:

Operating Conditions

1	2	3			

EMERGENCY ACTION LEVEL:

PC H_2 and O_2 concentrations GT 6% H_2 and 5% O_2

BASES:

This indicator represents a loss of both the fuel clad and RCS with a potential loss of the PC barrier.

Hydrogen and Oxygen concentrations at or above PPM 5.2.1, "Primary Containment Control", Table 19, Combustible Limits, in the drywell or wetwell represent a potential for a deflagration with a subsequent containment failure.

When hydrogen and oxygen concentrations approach or exceed combustible limits (6% hydrogen and 5% oxygen, respectively), the fuel clad barrier has been lost because, in order to reach such hydrogen concentration levels, it would have been necessary to overheat the fuel to metal-water reaction temperatures. For these concentrations to be detected in the primary containment would also indicate that, as a minimum, a potential loss of the RCS barrier has occurred.

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Fission Product Barrier Basis Information for Table 3

Columbia Generating Station Fission Product Barrier Evaluation

Columbia Generating Station Plant Specific EAL Guideline, PC1.4

3 Primary Containment 3.4 Containment Isolation Status

3.4.U.1 Unusual Event

NUMARC IC: Loss or potential loss of primary containment

APPLICABILITY:

Operating Conditions

1	2	3			

EMERGENCY ACTION LEVEL:

Failure of containment isolation valves (LCS Table 1.6.1.3-1) in any one line to close following auto or manual initiation

AND

downstream pathway outside primary containment exists

BASES:

This indicator is considered to be a loss of the PC barrier.

The failure to isolate condition is intended to cover containment isolation failures allowing a direct flow path to the environment such as a failure of MSIVs to close with open valves downstream to the turbine or condenser. Downstream path outside primary containment does not mean leakage incidental to normal system integrity. Only those penetrations required to isolate per Technical Specifications should be considered.

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Fission Product Barrier Basis Information for Table 3

Columbia Generating Station Fission Product Barrier Evaluation

Columbia Generating Station Plant Specific EAL Guideline, PC2.1

3 Primary Containment 3.4 Containment Isolation Status

3.4.A.1 Alert

NUMARC IC: Main steam line break outside containment with isolation

APPLICABILITY:

Operating Conditions	1	2	3			

EMERGENCY ACTION LEVEL:

Indications of a MSL break
AND
MSIV closure <u>has isolated</u> the break

BASES:

This EAL is based on design basis accident analyses which show that even if MSIV closure occurs within design limits, offsite (Exclusion Area Boundary) dose consequences from a "puff" release could exceed 50 but less than 100 mrem total effective dose equivalent (TEDE). Specifically FSAR Table 15.6-9 demonstrates 53 mrem TEDE. Only MSIV isolation signals that definitely indicate leakage outside primary RCS boundary with a path to the environment are included.

Examples of indications of Main Steam Line breaks are, but not limited to:

- Main Steam Line Tunnel temperature greater than 156°F on LD-TR-608, P-632 and "LEAK DET MSL TUNNEL TEMP HI-HI" alarm(s) on P-601-A2 &/or A3.
- Main Steam Line Tunnel delta temperature greater than 80°F on LD-TR-611, P-632 and "LEAK DET MSL TUNNEL HI-HI" alarm(s) on P601-A2 &/or A3.
- Main Steam Line flow greater than 140% as indicated by "MSL ISOL MAIN STEAM LINE FLOW HIGH" alarm on P-601-A11 &/or A12.

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Fission Product Barrier Basis Information for Table 3

Columbia Generating Station Plant Specific EAL Guideline, RCS1.1

3.4.S.1 Site Area Emergency

NUMARC IC: Loss or potential loss of either fuel clad or RCS barrier in conjunction with a loss or potential loss of containment

APPLICABILITY:

Operating Conditions

1	2	3			

EMERGENCY ACTION LEVEL:

Failure of containment isolation valves (LCS Table 1.6.3.1-1) in any one line to close following auto or manual initiation AND downstream pathway outside of primary containment exists
AND ANY of following:

- Total RCS leakage GT 30 gpm inside PC or EDR-FRS-623, Pen 2 upscale high
- Containment Radiation Monitor CMS-RIS-27E and CMS-RIS-27F reading GT 70 R/hr
- Drywell pressure GT 1.68 psig with indications of RCS leakage inside drywell

BASES:

This indicator is considered to be a loss of the PC barrier in conjunction with a loss or potential loss of RCS barrier.

The failure to isolate condition is intended to cover containment isolation failures allowing a direct flow path to the environment such as a failure of MSIVs to close with open valves downstream to the turbine or condenser. Downstream path outside primary containment does not mean leakage incidental to normal system integrity. Only those penetrations required to isolate per Technical Specifications should be considered.

Total leakage is considered to be the total of both identified and unidentified leakage as measured on EDR-FRS-623 pen 1 (unidentified - Floor Drain Sump Fill Rate) and pen 2 (identified - Equipment Drain Sump Fill Rate). The maximum measurable identified leak rate (pen 2) in the Control Room at Columbia Generating Station is 30 gpm, therefore, 30 gpm is used instead of the 50 gpm limit recommended by NUMARC.

A 70 R/hr reading on CMS-RIS-27E and CMS-RIS-27F is used to indicate a loss of the Reactor Coolant System barrier. This value assumes a 0.1% clad damage and the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory into the drywell atmosphere. The value of 0.1% clad damage was assumed to be the greatest amount of fuel failure under which power operation could occur.

The 1.68 psig drywell pressure for the Reactor Coolant System barrier loss is based on the drywell pressure scram and isolation setpoint and indicates a Loss of Coolant Accident (LOCA). A potential loss of the Reactor Coolant System barrier would not result in an increasing drywell pressure and, therefore, no indicator is provided. The qualifier of "with indications of RCS leak inside drywell" is included as an indicator of RCS boundary degradation and eliminates a drywell pressure increase due to a loss of drywell ventilation.

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Fission Product Barrier Basis Information for Table 3

Columbia Generating Station Fission Product Barrier Evaluation

Columbia Generating Station Plant Specific EAL Guideline, PC2.1 + RCS1.2, RCS3.1, RCS2.1

3.4.S.2 Site Area Emergency

NUMARC IC: Loss or potential loss of either fuel clad or RCS barrier in conjunction with a loss or potential loss of containment

APPLICABILITY:

Operating Conditions

1	2	3			

EMERGENCY ACTION LEVEL:

Indications of a MSL break
AND
MSIV closure has not isolated the break

BASES:

This indicator is considered to be a loss of the PC barrier in conjunction with a loss or potential loss of RCS barrier.

Examples of indications of Main Steam Line breaks are, but not limited to:

- Main Steam Tunnel temperature greater than 156°F on LD-TR-608, P-632 and "LEAK DET MSL TUNNEL TEMP HI-HI" alarm(s) on P-601-A2 or A3, or both.
- Main Steam Line Tunnel delta temperature greater than 80°F on LD-TR-611, P-632 and "LEAK DET MSL TUNNEL HI-HI" alarm(s) on P-701-A2 or A3, or both.
- Main Steam Line flow greater than 140% as indicated by "MSL ISOL MAIN STEAM LINE FLOW HI" alarm on P-601-A11 or A12, or both.

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Fission Product Barrier Basis Information for Table 3

Columbia Generating Station Plant Specific EAL Guideline, RCS1.1

3 Primary Containment 3.4 Containment Isolation Status

3.4.G.1 General Emergency

NUMARC IC: A loss of any two fission product barriers and loss or potential loss of the third

APPLICABILITY:

Operating Conditions	1	2	3			

EMERGENCY ACTION LEVEL:

Intentional venting of Primary Containment to prevent failure.

BASES:

This indicator represents a loss of both the fuel clad and RCS with a loss of the PC barrier.

Venting, if necessary to prevent failure of primary containment, is included as a loss of primary containment. This is specified in Emergency Operating Procedures when containment hydrogen and oxygen concentrations are in excess of, or cannot be determined to be below, combustible limits or when wetwell pressure approaches PCPL. However, routine venting per PPM 2.3.1, as long as radioactivity release rates are maintained, is not considered a loss of primary containment.

For either condition, a loss of the primary containment barrier exists because the containment is being vented. Additional losses and potential losses of fission product barriers that must also occur when either of these conditions occur are discussed below.

a. When wetwell pressure approaches or exceeds the PCPL, a prior loss of the RCS barrier has occurred due to the elevated containment pressure and the requirement to emergency depressurize the RPV at pressures below 39 psig (Pressure Suppression Pressure). Continued wetwell pressure increase could result in complete and uncontrolled loss of the primary containment due to containment failure. With no assurance as to where the containment may fail, an attendant loss of the suppression pool should be assumed with a consequent complete and unrecoverable loss of core cooling whereby the degraded core condition and loss of containment integrity releases substantial amounts of radioactivity to the general environment.

b. When hydrogen and oxygen concentrations approach or exceed combustible limits (6% hydrogen and 5% oxygen, respectively), the fuel clad barrier has been lost because, in order to reach such hydrogen concentration levels, it would have been necessary to overheat the fuel to metal-water reaction temperatures. For these concentrations to be detected in the primary containment would also indicate that, as a minimum, a potential loss of the RCS barrier has occurred.

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Fission Product Barrier Basis Information for Table 3

Columbia Generating Station Fission Product Barrier Evaluation

Columbia Generating Station Plant Specific EAL Guideline, PC2.1

4 SECONDARY CONTAINMENT (SC)

The secondary containment is comprised of the reactor building and associated ventilation, isolation and effluent systems. The secondary containment serves as an effective fission product barrier and is designed to minimize any ground level release of radioactive materials which might result from a serious accident.

The reactor building provides secondary containment during reactor operation and serves as primary containment when the reactor is shutdown and the drywell is open, as during refueling. Because the secondary containment is an integral part of the complete containment system, conditions which pose a threat to vital equipment located in the secondary containment are classifiable as emergencies.

There are two secondary containment parameters which are indicative of conditions which may pose a threat to secondary containment integrity or equipment located in secondary containment or are indicative of a direct release by a primary system into secondary containment:

- Secondary Containment Temperatures: Abnormally high secondary containment area temperatures can also pose a threat to the operability of vital equipment located inside secondary containment including RPV water level instrumentation. High area temperatures may limit personnel accessibility to vital areas. High area temperatures may also be indicative of either primary system discharges into secondary containment or fires.

- Secondary Containment Area Radiation Levels: Abnormally high area radiation levels in secondary containment, although not necessarily posing a threat to equipment operability, may pose a threat to personnel safety and the ability to operate vital equipment due to a lack of accessibility. Abnormally high area radiation levels may also be the result of a primary system discharging into the secondary containment and be indicative of precursors to significant radioactivity release to the environment.

4 Secondary Containment 4.1 Reactor Building Temperature/Radiation Levels

4.1.S.1 Site Area Emergency

NUMARC IC: Loss or potential loss of either fuel clad or RCS barrier in conjunction with a loss or potential loss of containment

APPLICABILITY:

Operating Conditions

1	2	3			

EMERGENCY ACTION LEVEL:

Unisolable primary system discharging outside PC resulting in any area temperature or radiation level above Maximum Safe Operating Values (as defined in the Emergency Operating Procedures).

BASES:

This indicator is considered to be a loss of the PC in conjunction with a potential loss RCS barrier.

The presence of elevated area temperatures and/or radiation levels in the secondary containment may be indicative of an unisolable primary system leakage outside the primary containment. These conditions represent a loss of the containment barrier and a potential loss of the RCS barrier.

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Fission Product Barrier Basis Information for Table 3

Columbia Generating Station Fission Product Barrier Evaluation

Columbia Generating Station Plant Specific EAL Guideline, RCS1.3/PC2.3

4 Secondary Containment 4.1 Reactor Building Temperature/Radiation Levels

4.1.G.1 General Emergency

NUMARC IC: A loss of any two fission product barriers and loss or potential loss of the third

APPLICABILITY:

Operating Conditions

1	2	3			

EMERGENCY ACTION LEVEL:

Unisolable primary system discharging outside PC resulting in any area temperature or radiation level above Maximum Safe Operating Values (as defined in the Emergency Operating Procedures). AND ANY of the following:

- Containment Radiation Monitor CMS-RIS-27E and CMS-RIS-27F reading GT 3,600 R/hr
- RPV level LT -161 inches (for ATWS conditions, RPV level LT -192 inches) or cannot be determined
- Coolant activity GT 300 μCi/gm dose equivalent iodine

BASES:

This indicator is considered to be a loss of the PC in conjunction with a loss or potential loss RCS barrier and loss or potential loss of the fuel clad barrier.

The presence of elevated area temperatures and/or radiation levels in the secondary containment may be indicative of an unisolable primary system leakage outside the primary containment. These conditions represent a loss of the containment barrier and a potential loss of the RCS barrier.

A 3,600 R/hr reading on CMS-RIS-27E and CMS-RIS-27F is used to indicate a release of reactor coolant, with elevated activity indicative of fuel damage, into the drywell. Therefore, this condition represents loss of both the fuel clad and RCS barriers. This value assumes an instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with coolant concentration associated with 5% clad failures into the drywell atmosphere. Columbia Generating Station has elected to provide an example dealing with the top end of the 2-5% range discussed in NESP-007. This value assumes an instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory into the drywell atmosphere.

Indicating fuel clad barrier potential Loss at -161″ (Top of Active Fuel) or -192 inches (for ATWS conditions) ensures that an emergency is declared before fuel perforation occurs. An unintentional level decrease to below -161″ or -192 inches is also indicative of a large RCS break, or a smaller break with loss of high pressure makeup.

During ATWS conditions, level may be intentionally lowered to reduce reactor power. This level decrease is not indicative of an RCS break as long as level is maintained above -192 inches (minimum steam cooling RPV water level.

Fuel Clad barrier damage is indicated by a coolant activity of 300 μCi/gm dose equivalent I-131. This amount of activity is well above that expected for iodine spikes and corresponds to approximately 2-5% fuel clad failure in accordance with assessment performed by the NUMARC EAL task force. This amount of clad failure indicates significant clad heating and, thus, the Fuel Clad barrier is considered lost.

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Fission Product Barrier Basis Information for Table 3

Columbia Generating Station Fission Product Barrier Evaluation

Engineering Calculation No. NE-02-94-57

Columbia Generating Station Plant Specific EAL Guideline, RCS1.3/PC2.3 + FC1.1, FC3.1, RCS4.1

5 RADIOACTIVITY RELEASE

Many EALs are based on actual or potential degradation of fission product barriers because of the increased potential for offsite radioactivity release. Degradation of fission product barriers though, is not always apparent via non-radiological symptoms. Therefore, direct indication of increased radiological effluents are appropriate symptoms for emergency classification.

At lower levels, abnormal radioactivity releases may be indicative of a failure of containment systems or precursors to more significant releases. At higher release rates, offsite radiological conditions may result which require offsite protective actions.

There are two basic indications of radioactivity release rates which warrant emergency classifications.

- Offsite Release: Direct indication of effluent radiation monitoring systems provides a rapid assessment mechanism to determine releases in excess of classifiable limits. Projected offsite doses (based on effluent monitor readings) or actual offsite field measurements indicating doses or dose rates above classifiable limits.

- Area Radiation: Radiation monitoring systems are specifically designed to provide indication of loss of control of radioactive material in the plant which may impede personnel access to safe shutdown areas.

5 Radioactivity Release 5.1 Offsite Release

5.1.U.1 Unusual Event

NUMARC IC: AU1 - Any unplanned release of gaseous or liquid radioactivity to the environment that exceeds two times ODCM limits for 60 minutes or longer.

APPLICABILITY:

Operating Conditions

1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

A valid reading exists which exceeds or is expected to exceed Table 3 column "UE" for GT 60 min.

BASES:

Unplanned releases that continue for 60 minutes or longer represent an uncontrolled situation and, hence, a potential degradation in the level of safety. The final integrated dose (which is very low in the Unusual Event emergency class) is not the primary concern here; rather, it is the degradation in plant control implied by the fact that the release was not isolated within 60 minutes.

It is not intended that the release be averaged over 60 minutes. Further, the ED should not wait until 60 minutes has elapsed, but should declare the event as soon as it is determined that the release will exceed 60 minutes. It is expected that the offsite dose analysis will be performed on the Emergency Dose Projection System (EDPS) using actual meteorology.

If the monitor reading(s) is sustained for longer than 60 minutes and the required assessments cannot be completed within this period, then the declaration should be made based on the valid reading.

Monitor indications and alarms are based on the methodology of the ODCM which demonstrates compliance with 10 CFR 20 and 10 CFR 50, Appendix I, requirements. The six year average meteorology is also used for basing alarm setpoints.

REFERENCE(S):

Columbia Generating Station Technical Specifications

Columbia Generating Station Offsite Dose Calculation Manual

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Unusual Event AU1

Columbia Generating Station Plant Specific EAL Guideline, AU1.2

5 Radioactivity Release 5.1 Offsite Release

5.1.U.2 Unusual Event

NUMARC IC: AU1 - Any unplanned release of gaseous or liquid radioactivity to the environment that exceeds two times ODCM limits for 60 minutes or longer.

APPLICABILITY:

Operating Conditions	1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

Offsite dose calculations indicate offsite dose rates GT Table 4 column "UE"
OR
Sample analysis indicates release GT 2 times ODCM 6.2.1.1 or 6.2.1.2 limits for GT 60 min.

BASES:

Unplanned releases that continue for 60 minutes or longer represent an uncontrolled situation and, hence, a potential degradation in the level of safety. The final integrated dose (which is very low in the Unusual Event emergency class) is not the primary concern here; rather, it is the degradation in plant control implied by the fact that the release was not isolated within 60 minutes.

It is not intended that the release be averaged over 60 minutes. Further, the ED should not wait until 60 minutes has elapsed, but should declare the event as soon as it is determined that the release will exceed 60 minutes. It is expected that the offsite dose analysis will be performed on the Emergency Dose Projection System (EDPS) using actual meteorology. However, if classification is based upon notification of ODCM limits being exceeded upon discovery that the condition has existed previously, initiate a Transitory Event Notification per EPIP 13.4.1.

REFERENCE(S):

Columbia Generating Station Technical Specifications
Columbia Generating Station Offsite Dose Calculation Manual
NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Unusual Event AU1
Columbia Generating Station Plant Specific EAL Guideline, AU1.1
EPIP 13.4.1, Emergency Notifications

5 Radioactivity Release 5.1 Offsite Release

5.1.A.1 Alert

NUMARC IC: AA1 - Any unplanned release of gaseous or liquid radioactivity to the environment that exceeds 200 times the radiological specifications for 15 minutes or longer.

APPLICABILITY:

Operating Conditions

1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

A valid reading exists which exceeds or <u>is expected</u> to exceed Table 3 column "Alert" for GT 15 min.

BASES:

The required release duration has been reduced to 15 minutes in recognition of the decreased plant safety, and to provide prompt classification. It is expected that the offsite dose analysis will be performed on the Emergency Dose Projection System (EDPS).

If the monitor reading(s) is sustained for longer than 15 minutes and the required assessments cannot be completed within this period, then the declaration should be made based on the valid reading.

REFERENCE(S):

Columbia Generating Station Technical Specifications

Columbia Generating Station Offsite Dose Calculation Manual

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Unusual Event AA1

Columbia Generating Station Plant Specific EAL Guideline, AA1.2

5 Radioactivity Release 5.1 Offsite Release

5.1.A.2 Alert

NUMARC IC: AA1 - Any unplanned release of gaseous or liquid radioactivity to the environment that exceeds 200 times the radiological specifications for 15 minutes or longer.

APPLICABILITY:

Operating Conditions

1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

Offsite dose calculations indicate dose rates GT Table 4 column "Alert"
OR
Sample analysis indicates release GT 200 times ODCM 6.2.1.1 or 6.2.1.2 limits for GT 15 min.

BASES:

The required release duration has been reduced to 15 minutes in recognition of the decreased plant safety, and to provide prompt classification. It is expected that the offsite dose analysis will be performed on the Emergency Dose Projection System (EDPS). If a dose projection cannot be performed and the monitor reading is sustained for longer than the specified time, then the declaration must be made based on the valid reading. However, if classification is based upon notification of ODCM limits being exceeded upon discovery that the condition has existed previously, initiate a Transitory Event Notification per EPIP 13.4.1.

REFERENCE(S):

Columbia Generating Station Technical Specifications
Columbia Generating Station Offsite Dose Calculation Manual
NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Unusual Event AA1
Columbia Generating Station Plant Specific EAL Guideline, AA1.1
EPIP 13.4.1, Emergency Notifications

5 Radioactivity Release 5.1 Offsite Release

5.1.S.1 Site Area Emergency

NUMARC IC: AS1 - Offsite dose resulting from an actual or imminent release of gaseous radioactivity that exceeds 100 mrem TEDE or 500 mrem thyroid CDE for the actual or projected duration of the release.

APPLICABILITY:

Operating Conditions	1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

A valid reading exists which exceeds or <u>is expected</u> to exceed Table 3 column "Site Area" for GT 15 min.

BASES:

Effluent readings may be used for the classification of fast breaking events until actual dose projections can be made. Dose assessment, since it uses current plant values, will be more accurate and should be used.

The 100 mrem integrated dose in this Initiating Condition is based on 10 CFR 20 annual average population exposure. This value also provides a desirable gradient (one order of magnitude) between the Alert, Site Area Emergency and General Emergency classes. It is calculated that exposures less than this time limit are not consistent with the Site Area Emergency class description. The 500 mrem integrated thyroid CDE dose was established in consideration of the 1:5 ratio of the EPA Protection Action Guidelines for TEDE and Thyroid Committed Dose Equivalent.

In establishing the emergency action levels, a release duration of one hour is assumed. If an emergency dose projection cannot be performed and the monitor reading(s) is expected to exceed 15 minutes, then the declaration should be made based on the valid reading.

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Site Area Emergency AS1

10 CFR 20

Columbia Generating Station Offsite Dose Calculation Manual

Environmental Protection Agency 400, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents, October 15, 1991

Columbia Generating Station Plant Specific EAL Guideline, AS1.3, AS1.4

5.1.S.2 Site Area Emergency

NUMARC IC: AS1 - Offsite dose resulting from an actual or imminent release of gaseous radioactivity that exceeds 100 mrem TEDE or 500 mrem thyroid CDE for the actual or projected duration of the release.

APPLICABILITY:

Operating Conditions

1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

Offsite dose calculations indicate doses or dose rates GT Table 4 column "Site Area"
OR
Field survey or survey sample analysis indicates offsite dose rates GT Table 4 column "Site Area"

BASES:

The 100 mrem integrated dose in this Initiating Condition is based on the proposed 10 CFR 20 annual average population exposure. This value also provides a desirable gradient (one order of magnitude) between the Alert, Site Area Emergency and General Emergency classes. It is calculated that exposures less than this time limit are not consistent with the Site Area Emergency class description. The 500 mrem integrated thyroid CDE dose was established in consideration of the 1:5 ratio of the EPA Protection Action Guidelines for TEDE and Thyroid Committed Dose Equivalent.

In establishing the emergency action levels, a release duration of one hour is assumed. If an emergency dose projection cannot be performed and the monitor reading(s) is expected to exceed 15 minutes and the required assessments cannot be completed within this period, then the declaration should be made based on the valid reading.

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Site Area Emergency AS1

10 CFR 20

Columbia Generating Station Offsite Dose Calculation Manual

Environmental Protection Agency 400, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents, October 15, 1991

Columbia Generating Station Plant Specific EAL Guideline, AS1.1

5 Radioactivity Release 5.1 Offsite Release

5.1.G.1 General Emergency

NUMARC IC: AG1 - Offsite dose resulting from an actual or imminent release of gaseous radioactivity that exceeds 1000 mrem TEDE or 5000 mrem thyroid CDE for the actual or projected duration of the release using actual meteorology.

APPLICABILITY:

Operating Conditions	1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

A valid reading exists which exceeds or is expected to exceed Table 3 column "General" for GT 15 min.

BASES:

Effluent readings may be used for the classification of fast breaking events until actual dose projections can be made. Dose assessment, since it uses current plant values, will be more accurate and should be used.

In establishing the emergency action levels, a release duration of one hour is assumed. If an emergency dose projection cannot be performed and the monitor reading(s) is expected to exceed 15 minutes, then the declaration should be made based on the valid reading.

The 1000 mrem Total Effective Dose Equivalent or 5000 mrem thyroid Committed Dose Equivalent integrated dose are based on the EPA protective action guidance which indicates that public protective actions are indicated if the dose exceeds 1 rem Total Effective Dose Equivalent or 5 rem thyroid Committed Dose Equivalent. This logic is consistent with the emergency class description for a General Emergency and constitutes the upper level of the desirable gradient for the Site Area Emergency.

Actual meteorology is specifically identified in the Initiating Condition since it gives the most accurate dose assessment.

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Site Area Emergency AS1

10 CFR 20

Columbia Generating Station Offsite Dose Calculation Manual

Environmental Protection Agency 400, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents, October 15, 1991

Columbia Generating Station Plant Specific EAL Guideline, AG1.3, AG1.4

5.1.G.2 General Emergency

NUMARC IC: AG1 - Boundary dose resulting from an actual or imminent release of gaseous radioactivity that exceeds 1000 mrem TEDE or 5000 mrem thyroid CDE equivalent for the actual or projected duration of the release using actual meteorology.

APPLICABILITY:

Operating Conditions	1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

Offsite dose calculations indicate doses or does rates GT Table 4 column "General"
OR
Field survey or survey sample analysis indicates offsite dose rates GT Table 4 column "General".

BASES:

The 1000 mrem Total Effective Dose Equivalent or 5000 mrem thyroid Committed Dose Equivalent integrated dose are based on the EPA protective action guidance which indicates that public protective actions are indicated if the dose exceeds 1 rem Total Effective Dose Equivalent or 5 rem thyroid Committed Dose Equivalent. This logic is consistent with the emergency class description for a General Emergency and constitutes the upper level of the desirable gradient for the Site Area Emergency.

Actual meteorology is specifically identified in the Initiating Condition since it gives the most accurate dose assessment.

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Site Area Emergency AS1

10 CFR 20

Columbia Generating Station Offsite Dose Calculation Manual

Environmental Protection Agency 400, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents, October 15, 1991

Columbia Generating Station Plant Specific EAL Guideline, AG1.1

Table 3 Effluent Monitor Classification Thresholds				
Monitor	UE	Alert	Site Area	General
NOTE: If a dose projection cannot be performed and the monitor reading is sustained for longer than the specified time, then the declaration must be made based on the valid reading.	60 min.	15 min.	15 min	15 min
PRM-RE-1B Reactor Bldg. Exhaust Inter.	1.35E4 cps	N/A	N/A	N/A
PRM-RE-1C Reactor Bldg. Exhaust Hi	N/A	1.14E3cps	9.65E3 cps	9.35E4 cps
TEA-RIS-13 Turbine Bldg. Exhaust, low	1.7E4 cpm	4.4E4 cpm	4.4E5 cpm	N/A
TEA-RIS-13A Turbine Bldg. Exhaust, Int	N/A	N/A	N/A	11 PMU
WEA-RIS-14 Radwaste Bldg. Exhaust, low	1.2E5 cpm	1.7E5 cpm	1.7E6 cpm	N/A
WEA-RIS-14A Radwaste Bldg. Exhaust, Int	N/A	N/A	N/A	29 PMU
TSW-RIS-5 TSW Effluent	3.9E3 cpm	3.9E5 cpm	N/A	N/A
FDR-RIS-606 Rad. Waste Effluent	2 x Hi-Hi alarm	200 x Hi-Hi alarm	N/A	N/A
SW-RIS-604 SW 'A' Process	2.0E2 cps	2.0E4 cps	N/A	N/A
SW-RIS-605 SW 'B' Process	2.0E2 cps	2.0E4 cps	N/A	N/A

cps=counts per second cpm=counts per minute PMU=panel meter units N/A=not applicable (outside of meter range)

Table 4 Offsite Dose Calculation/Field Survey Sample Analysis Classification Thresholds at 1.2 miles				
	UE	Alert	Site Area	General
TEDE	N/A	N/A	100 mrem	1000 mrem
CDE Thyroid	N/A	N/A	500 mrem	5000 mrem
TEDE rate	0.1 mrem/hr	10 mrem/hr	100 mrem/hr[1]	1000 mrem/hr[1]
CDE Thyroid rate	0.3 mrem/hr	50 mrem/hr	500 mrem/hr[2]	5000 mrem/hr[2]

[1] (Projected GT 60 min)
[2] (For GT 1 HR inhalation)

5 Radioactivity Release 5.2 Area Radiation

5.2.U.1 Unusual Event

NUMARC IC: AU2 - Unexpected increase in plant radiation levels.

APPLICABILITY:

Operating Conditions

1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

Valid reading GT 5E3 mR/hr on ANY of the following ARMs:

- ARM-RIS-4 thru ARM-RIS-18
- ARM-RIS-20 thru ARM-RIS-30
- ARM-RIS-32 thru ARM-RIS-34 (High Range)

BASES:

This Initiating Condition is not applicable for alarms resulting from the controlled movement of radioactive materials in the plant or expected increases in radiation levels due to the backwashing of demineralizer filters or other planned operation.

Unplanned increases in in plant radiation levels represent a degradation in the control of radioactive material and represent a potential degradation in the level of safety of the plant. This EAL escalates to an ALERT per 5.2.A.1 if the radiation level increase impairs safe operation of the plant.

REFERENCE(S):

Columbia Generating Station Technical Specifications

Columbia Generating Station Instrument Master Data Sheets for referenced instruments

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Unusual Event AU2

Columbia Generating Station Plant Specific EAL Guideline, AU2.4

5.2.A.1 Alert

NUMARC IC: AA2 - Release of radioactive material or increases in radiation levels within the facility that impedes operation of systems required to maintain safe operations or to establish or maintain cold shutdown.

APPLICABILITY:

Operating Conditions	1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

Valid reading GT 15 mR/hr on ARM-RIS-19 (CR)
OR
Valid reading GT 1E4 mR/hr on ANY of the following ARMs:

- ARM-RIS-4 thru ARM-RIS-18
- ARM-RIS-23
- ARM-RIS-24
- ARM-RIS-32 thru ARM-RIS-34 (High Range)

BASES:

Areas requiring continuous occupancy include the Control Room. The value of 15 mrem/hr is derived from the Generic Design Criteria (GDC) 19 value of 5 rem in 30 days with adjustment for expected occupancy times. Although Section III.D.3 or NUREG 0737, "Clarification of TMI Action Plan Requirements", provides that the 15 mrem/hr value can be averaged over the 30 days, the value is used here without averaging, as a 30-day duration implies an event potentially more significant than an Alert.

This Initiating Condition addresses increased radiation levels that 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. It is this impaired ability to operate the plant that results in the actual or potential substantial degradation of the level of safety of the plant. The cause and/or magnitude of the increase in radiation levels is not a concern of this Initiating Condition.

This Initiating Condition is not meant to apply to increases in the containment radiation monitors as these events are addressed in the fission product barrier Initiating Conditions, nor is it intended to apply to anticipated temporary increases due to planned events (e.g., incore detector movement, Radwaste container movement, depleted resin transfers, etc.).

The Emergency Director should determine the cause of the increase in radiation levels and review other Initiating Conditions for applicability.

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, AA3

NUREG 0737, Clarification of TMI Action Plan Requirements

Columbia Generating Station Instrument Master Data Sheets for referenced instruments

Columbia Generating Station Plant Specific EAL Guideline, AA3.1, AA3.2

6 ELECTRICAL FAILURES

Loss of vital plant electrical power can compromise plant safety system operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity.

The events of this category have been grouped into the following two loss of electrical power types:

- Loss of AC Power Sources: This category includes losses of onsite and/or offsite AC power sources including station blackout events.

- Loss of DC Power Sources: This category involves total losses of vital plant 125 VDC power sources.

6 Electrical Failures6.1 AC Power Loss

6.1.U.1 Unusual Event

NUMARC IC: SU1 - Loss of all offsite power to critical AC busses for greater than 15 minutes.

APPLICABILITY:

Operating Conditions

1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

Power is unavailable to SM-7 and SM-8 from offsite AC sources GT 15 min.

BASES:

Even though power may still be available from offsite sources, there must be a functional flowpath to the Critical busses. Prolonged loss of offsite AC power reduces the required redundancy and potentially degrades the level of safety by rendering the plant more vulnerable to a complete loss of AC power (Station Blackout).

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

Credit is not taken in this EAL for the Division 3 Standby Diesel Generator because it only supplies power to the High Pressure Core Spray (HPCS) pump and associated loads, but not for any long-term decay heat removal systems and, in particular, wetwell cooling mechanisms that would be essential subsequent to a station blackout.

Failure of either the Division 1 or Division 2 Standby Diesel Generator would escalate this event to Alert 6.1.A.2.

REFERENCE(S):

Columbia Generating Station Technical Specifications

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Unusual Event SU1

Columbia Generating Station Plant Specific EAL Guideline, SU1.6

6 Electrical Failures6.1 AC Power Loss

6.1.A.1 Alert

NUMARC IC: SA1 - Loss of all offsite power and loss of all onsite power to critical AC busses for greater than 15 minutes.

APPLICABILITY:

Operating Conditions

			4	5	def

EMERGENCY ACTION LEVEL:

Complete loss of all AC power to SM-7 and SM-8 GT 15 min.

BASES:

A loss of the minimum required offsite circuits and failure of the diesel generators to restore power to the emergency busses results in a loss of AC power to all plant safety systems requiring AC power including RHR, ECCS, containment cooling systems, spent fuel heat removal systems, and wetwell cooling systems. This significant reduction in decay heat removal is a substantial reduction in the level of safety of the plant due to a potential for temperature and pressure increases.

Credit is not taken in this EAL for the Division 3 Standby Diesel Generator because: 1) it is not required for all situations during Operating Conditions 4 and 5; and 2) although it does supply power to the HPCS pump which is a source of makeup water, it does not supply power to any systems that could be used to remove energy from the reactor and wetwell thereby limiting the long-term decay heat removal effectiveness.

When the plant is in a cold shutdown or refueling condition, RPV temperature and pressure are lower than they would be in other operating conditions. These lower pressures and temperatures increase the margin of safety allowing more time before power must be restored to an emergency bus than would be available during Operating Conditions 1, 2 or 3.

Fifteen minutes was selected as a conservative lower threshold that retains the anticipatory nature of EALs while excluding transient or momentary power losses.

Escalation of this event to a Site Area Emergency would be via the Increased Radiation Release to the Environment (see EAL 5.1.S.1 and 5.1.S.2) or Emergency Director Judgment (see EAL 9.1.S.1 and 9.1.S.2).

If this same set of conditions were to occur in Operating Conditions 1, 2 or 3, they would be classified a Site Area Emergency (see EAL 6.1.S.1).

REFERENCE(S):

Columbia Generating Station Technical Specifications

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Alert SA1

Columbia Generating Station Plant Specific EAL Guideline, SA1.1

6 Electrical Failures6.1 AC Power Loss

6.1.A.2 Alert

NUMARC IC: SA5 - Power capability to critical AC busses reduced to a single power source for greater than 15 minutes such that any additional single failure would result in station blackout.

APPLICABILITY:

Operating Conditions

1	2	3			

EMERGENCY ACTION LEVEL:

Available emergency bus AC power has been reduced to only one of the following sources for GT 15 min.

- TR-N1 (SM-7 and/or SM-8)
- TR-S (SM-7 and/or SM-8)
- TR-B (SM-7 and/or SM-8)
- DG-1 (SM-7)
- DG-2 (SM-8)

BASES:

This Initiating Condition and its associated EAL provide an escalation from EAL 6.1.U.1, "Loss of All Offsite Power to Critical Busses for Greater than 15 Minutes". 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 EAL includes a loss of both offsite power sources with only one diesel generator powering its respective emergency bus as well as a failure of both diesel generators such that they would not be able to power their respective emergency busses with only one offsite power source available.

Credit is not taken in this Initiating Condition for bus SM-4 and the Division 3 Standby Diesel Generator because they only supply power to the HPCS pump and associated loads but not to any decay heat removal systems that would be essential subsequent to a station blackout.

Power to busses SM-7 and SM-8 may come from either its respective standby diesel generator or from the Switch Yard through the Startup or Backup Transformers. Regardless of the source of power, failure of the remaining power source would result, at least temporarily, in a station blackout. The determining factor of whether or not to classify then becomes the amount of time that power is not available.

Escalation for a Site Area Emergency for a station blackout would be via EAL 6.1.S.1.

REFERENCE(S):

Columbia Generating Station Technical Specifications

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Alert SA5

Columbia Generating Station Plant Specific EAL Guideline, SA5.1

6 Electrical Failures6.1 AC Power Loss

6.1.S.1 Site Area Emergency

NUMARC IC: SS1 - Loss of all offsite power and loss of all onsite power to critical AC busses for greater than 15 minutes.

APPLICABILITY:

Operating Conditions

1	2	3			

EMERGENCY ACTION LEVEL:

Complete loss of <u>all</u> AC power to SM-7 and SM-8 GT 15 min.

BASES:

The condition indicated by this EAL is a station blackout and reflects a condition more serious than that described in Alert 6.1.A.2 in that both Division 1 and Division 2 Emergency Busses are deenergized and have been without power for 15 minutes or longer. Station blackouts lasting less than 15 minutes or electrical system faults resulting in only one emergency bus powered from only one source are classified according to Alert 6.1.A.2.

Credit is not taken in this EAL for Bus SM-4 and the Division 3 Standby Diesel Generator because it only supplies power to the HPCS pump and associated loads but not to any decay heat removal systems that would be essential subsequent to a station blackout.

Fifteen minutes was chosen as a conservative time to maintain the anticipatory nature of EALs while excluding transient or momentary power losses.

This event would be upgraded to a General Emergency per 6.1.G.1 if it appears that power cannot be restored to Bus SM-7 or SM-8 within 4 hours or if the Emergency Director determines that a loss or potential loss of a fission product barrier is imminent in accordance with the Fission Product Barrier Degradation table. Imminent in this context means mitigation strategies and actions are not successful in preventing a challenge to Fuel Clad, Reactor Coolant Pressure Boundary or Primary Containment.

REFERENCE(S):

Columbia Generating Station Technical Specifications

Columbia Generating Station FSAR, Section 1.5.2, SBO Coping Study

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Site Area Emergency SS1

Columbia Generating Station Plant Specific EAL Guideline, SS1.1

6 Electrical Failures6.1 AC Power Loss

6.1.G.1 General Emergency

NUMARC IC: SG2 - Prolonged loss of all offsite power and prolonged loss of all onsite power to critical AC busses.

APPLICABILITY:

Operating Conditions

1	2	3			

EMERGENCY ACTION LEVEL:

Complete loss of all AC power to SM-7 and SM-8
AND either of the following:

In the judgment of the Emergency Director, AC power to either SM-7 or SM-8 is not likely to be restored within 4 hrs.
OR
RPV level LT -161 in. (For ATWS conditions, RPV level LT -192 inches)

BASES:

The condition indicated by this Initiating Condition and its associated EAL is a station blackout lasting long enough to degrade or potentially degrade a fission product barrier.

Loss of all AC power compromises all plant safety systems requiring AC power including RHR, ECCS, containment cooling systems, spent fuel heat removal systems and Wetwell cooling systems. Prolonged loss of all AC power may lead to loss of integrity of the fuel clad, reactor coolant system or containment.

Credit is not taken in this EAL for Bus SM-4 and the Division 3 Standby Diesel Generator because it only supplies power to the HPCS pump and associated loads but not to any decay heat removal systems that would be essential subsequent to a station blackout.

Under these conditions, fission product barrier monitoring capability may be degraded. It may be difficult to predict when power can be restored. However, the Emergency Director should determine the need to declare a General Emergency based on two major considerations:

1. Are there any present indications that core cooling is already degraded to the point that loss or potential loss of a fission product barrier is imminent? Imminent in this context means mitigation strategies and actions are not successful in preventing a challenge to Fuel Clad, Reactor Coolant Pressure Boundary or Primary Containment.

2. If there are no present indications of such core cooling degradation, how likely is it that power can be restored in time to assure that a loss of two barriers with a potential loss of the third can be prevented?

The first question is answered by indications of a loss of adequate core cooling, which for the purpose of this EAL is considered to be RPV water level less than -161 in. (or -192 inches for ATWS conditions). The second question should be answered by the Emergency Director by making a realistic assessment of the time required to complete any necessary repairs. This EAL requires the Emergency Director to classify the event as soon as his assessment indicates that necessary repairs will take longer than 4 hours rather than waiting for the 4 hours to expire.

During ATWS conditions, level may be intentionally lowered to reduce reactor power. This level decrease is not indicative of an RCS break as long as level is maintained above -192 inches (minimum steam cooling RPV water level.

REFERENCE(S):

Columbia Generating Station Technical Specifications

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, General Emergency SG1

Columbia Generating Station FSAR, Section 1.5.2, SBO Coping Study

Columbia Generating Station Plant Specific EAL Guideline, SG1.1

6 Electrical Failures6.2 DC Power Loss

6.2.U.1 Unusual Event

NUMARC IC: SU7 - Degradation of all critical DC power for greater than 15 minutes.

APPLICABILITY:

Operating Conditions				4	5	def

EMERGENCY ACTION LEVEL:

Degradation of both Division 1 and Division 2 critical DC voltage as indicated by bus voltage LT 110 VDC on both 125 V Dist. Panels S1-1 and S1-2 voltmeters (Bd. C) for GT 15 min.

BASES:

This Initiating Condition and its EAL recognize a loss of DC power compromising the ability to monitor and control the removal of decay heat during cold shutdown, refueling, or defueled. This EAL is intended to be anticipatory in that the operating crew may not have the necessary indication and control of equipment needed to respond to the loss. This is a less severe condition than that described in Site Area Emergency 6.2.S.1 because initial temperatures and pressures are lower than they would be for Operating Conditions 1, 2 or 3 and, normally, less decay would be present.

Credit is not taken in this EAL for the Division 3 DC bus because it only supplies control power to loads associated with the HPCS pump and not to any decay heat removal systems.

The Columbia Generating Station battery sizing calculations for the 125 VDC batteries, S1-1 and S1-2, reveal that a worst case LOCA analysis would permit a 15 minute margin between 106.3 volts and 105.0 volts. However, a conservative value of one hundred ten (110) volts DC is used as a minimum bus voltage. It is based on providing a 15 minute margin of operation before bus voltage drops below 105 volts DC at which time bus loads cannot be guaranteed to function. One hundred ten volts was also selected based on instrument accuracy of ± 2% full scale or ± 3 volts and scale increments of 2 volts.

The same set of conditions as described in this EAL would be classified Site Area Emergency 6.2.S.1 if they occurred during Operating Conditions 1, 2 or 3.

REFERENCE(S):

Columbia Generating Station Technical Specifications

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Unusual Event SU7

Engineering Calculation 2.05.01 (Battery Sizing Calc)

Columbia Generating Station FSAR Section 8.3.2.1, Batteries

CVI 51A-00,8 Exide Manual

Columbia Generating Station Plant Specific EAL Guideline, SU7.1

6 Electrical Failures6.2 DC Power Loss

6.2.S.1 Site Area Emergency

NUMARC IC: SU7 - Degradation of all critical DC power for greater than 15 minutes.

APPLICABILITY:

Operating Conditions

1	2	3			

EMERGENCY ACTION LEVEL:

Degradation of both Division 1 and Division 2 critical DC voltage as indicated by bus voltage LT 110 VDC on both 125 V Dist. Panels S1-1 and S1-2 voltmeters (Bd. C) for GT 15 min.

BASES:

This Initiating Condition and its EAL recognize a loss of DC power compromising the ability to monitor and control the removal of decay heat during power operations, startup and hot shutdown conditions. It is intended to be anticipatory in that the operating crew may not have the necessary indication and control of equipment needed to respond to the loss. This EAL represents a more serious condition that than described in Unusual Event 6.2.U.1 in that the initial temperatures, pressures and available decay heat may be substantially higher than in Unusual Event 6.2.U.1 resulting in significantly less time available before failure of systems needed to protect the public.

Loss of all DC power compromises the ability to monitor and control plant safety functions. Prolonged loss of all DC power may result in core uncovery and loss of containment integrity when there is significant decay heat and residual heat in the reactor coolant system.

Credit is not taken in this EAL for the Division 3 DC bus because it only supplies control power to loads associated with the HPCS pump and not to any decay heat removal systems.

The Columbia Generating Station battery sizing calculations for the 125 VDC batteries, S1-1 and S1-2, reveal that a worst case LOCA analysis would permit a 15 minute margin between 106.3 volts and 105.0 volts. However, a conservative value of one hundred ten (110) volts DC is used as a minimum bus voltage. It is based on providing a 15 minute margin of operation before bus voltage drops below 105 volts DC at which time bus loads cannot be guaranteed to function. One hundred ten volts was also selected based on instrument accuracy of ± 2% full scale or ± 3 volts and scale increments of 2 volts.

REFERENCE(S):

Columbia Generating Station Technical Specifications

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Site Area Emergency SS3

Engineering Calculation 2.05.01 (Battery Sizing Calc)

Columbia Generating Station FSAR Section 8.3.2.1, Batteries

CVI 51A-00,8 Exide Manual

Columbia Generating Station Plant Specific EAL Guideline, SS3.1

7 EQUIPMENT FAILURES

Numerous plant system related equipment failure events which warrant emergency classification, based upon their potential to pose actual or potential threats to plant safety, have been identified in this category.

The events of this category have been grouped into the following event types:

- System Failures: This subcategory includes conditions related to the failure of the plant to be brought to the required plant operating condition required by technical specifications and events which are indicative of a loss of ability to maintain the plant in cold shutdown.

- Control Room Evacuation: This category addresses losses of Control Room habitability and the ability to establish plant control from remote shutdown panels.

- Loss of Indication/Communications: Certain events which degrade the plant operators ability to effectively assess plant conditions or communicate with essential personnel within or external to the plant warrant emergency classification. Under this event type are losses of annunciators and/or communication equipment.

7 Equipment Failures 7.1 System Failures

7.1.U.1 Unusual Event

NUMARC IC: SU2 - Inability to reach required shutdown within technical specification limits.

APPLICABILITY:

Operating Conditions

1	2	3			

EMERGENCY ACTION LEVEL:

Plant is not brought to required operating mode within T.S. LCO action statement time

BASES:

Limiting Conditions of Operation (LCOs) require the plant to be brought to a required shutdown mode when the Technical Specification required configuration cannot be restored. Depending on the circumstances, this may or may not be an emergency or precursor to a more severe condition. In any case, the initiation of plant shutdown required by the 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 Notification of Unusual Event is required when the plant is not brought to the required operating mode within the allowable action statement time in the Technical Specifications. Declaration of an Unusual Event is based on the time at which the LCO-specified action statement time period elapses under the site Technical Specifications and is not related to how long a condition may have existed. Other required Technical Specification shutdowns that involve precursors to more serious events are addressed by other categories of Initiating Conditions.

When a Limiting Condition is not met, and the associated ACTION requirement is not met within the required time allowed, the plant is not within its safety envelope and not within a T. S. LCO action statement time. Declaration of the Unusual Event should occur when Technical Specification action time elapses.

REFERENCE(S):

Columbia Generating Station Technical Specifications

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Unusual Event SU2

Columbia Generating Station Plant Specific EAL Guideline, SU2.1

7 Equipment Failures 7.1 System Failures

7.1.U.2 Unusual Event

NUMARC IC: HU1 - Natural and destructive phenomena affecting the Protected Area Boundary.

APPLICABILITY:

Operating Conditions

1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

Uncontrolled flooding in SSB (Table 5) that has the potential to affect safety related equipment needed for the current operating mode.

BASES:

Declaration of this EAL identifies the occurrences of an event of sufficient magnitude to be of concern to the operating crew. This EAL addresses the effect of flooding caused by internal events such as component failures, equipment mis-alignment, or outage activity mishaps. The SSBs contain systems required for safe shutdown that are not designed to be wetted or submerged.

For the purposes of this EAL, Safe Shutdown Buildings are considered to be the following locations:

- Vital portions of the Radwaste/Control Building
- Reactor Building
- Turbine Building
- Standby Service Water Pump Houses
- Diesel Generator Building
- Diesel Generator Fuel Oil Storage Area

This list was developed from equipment locations identified in FSAR Table 3.2-1, Seismic Category I. Equipment in Safe Shutdown Buildings is identified in FSAR Table 3.2-1, Seismic Category I. The only equipment in the Turbine Building listed in FSAR Table 3.2-1, Seismic Category I is:

- DEH Pressure Switches
- RPS switches on turbine throttle valves
- Main Steam Line Radiation Monitors
- Turbine Building Ventilation Radiation Monitors
- Main Steam Piping up to MS-V-146 and the first stop valves

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, final draft Rev. 3, Unusual Event HU1.6

7 Equipment Failures 7.1 System Failures

7.1.A.1 Alert

NUMARC IC: SA3 - Inability to maintain plant in cold shutdown.

APPLICABILITY:

Operating Conditions						
				4	5	

EMERGENCY ACTION LEVEL:

Inability to restore and maintain reactor coolant temp LT 200 °F

BASES:

This EAL addresses loss or degradation of functions required for core cooling during refueling and cold shutdown modes such that the technical specification limit cannot be maintained. Determination of "Inability to maintain" includes making an evaluation that considers both current and future system performance in relation to the current values and trends of relevant parameters. A momentary unplanned excursion above 200°F when adequate heat removal function is available is not intended to constitute an Alert.

REFERENCE(S):

Columbia Generating Station Technical Specifications

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Alert SA3

Columbia Generating Station Plant Specific EAL Guideline, SA3.1

7 Equipment Failures 7.1 Natural Events

7.1.A.2 Alert

NUMARC IC: HA1 - Natural and destructive phenomena affecting Safe Shutdown Buildings.

APPLICABILITY:

Operating Conditions	1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

Report by plant personnel confirming the occurrence of uncontrolled plant internal flooding in a safe shutdown building, Table 5
AND
Affected safe shutdown system parameters indicate degraded performance

BASES:

Flooding conditions within the plant affecting safe shutdown areas have the potential to directly impact the safe operation of the plant. The uncontrolled flooding event may pose a direct threat to safety-related equipment. As such, the potential exists for substantial degradation of the level of safety of the plant. Flooding is indicated by ECCS room level alarms on P601 and sump hi-hi alarms on P602.

This EAL is intended to address events that may have resulted in Safe Shutdown Buildings being subjected to forces beyond design limits and, thus, damage may be assumed to have occurred to safe shutdown systems. The initial "report" should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in these EALs to assess the actual magnitude of the damage. Escalation to a higher emergency class, if appropriate, will be based on the specific system malfunctions, fission product barrier degradation, abnormal radiological releases, or Emergency Director judgment ICs.

For the purposes of this EAL, Safe Shutdown Buildings are considered to be the following locations:

- Vital portions of the Radwaste/Control Building
- Reactor Building
- Turbine Building
- Standby Service Water Pump Houses
- Diesel Generator Building
- Diesel Generator Fuel Oil Storage Area

This list was developed from equipment locations identified in FSAR Table 3.2-1, Seismic Category I. Equipment in Safe Shutdown Buildings is identified in FSAR Table 3.2-1, Seismic Category I. The only equipment in the Turbine Building listed in FSAR Table 3.2-1, Seismic Category I is:

- DEH Pressure Switches
- RPS switches on turbine throttle valves
- Main Steam Line Radiation Monitors
- Turbine Building Ventilation Radiation Monitors
- Main Steam Piping up to MS-V-146 and the first stop valves

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Unusual Event HA1

FSAR Table 3.2-1

Columbia Generating Station Plant Specific EAL Guideline, HA1.7

7 Equipment Failures 7.2 Control Room Evacuation

7.2.A.1 Alert

NUMARC IC: HA5 - Control room evacuation has been initiated.

APPLICABILITY:

Operating Conditions

1	2	3	4	5	

EMERGENCY ACTION LEVEL:

The decision to evacuate the Control Room has been made.

BASES:

The Alert condition addresses events which involve a substantial degradation of the level of safety of the plant. Frequently, a distinguishing characteristic of a "substantial degradation" is the need for increased monitoring of or assistance in monitoring plant functions. With the Control Room evacuated, additional support, monitoring and direction through the Technical Support Center and/or Operations Support Center is necessary. Therefore, an Alert should be declared when the Control Room must be evacuated.

An inability to establish plant control from outside the Control Room will escalate this event to a Site Area Emergency.

REFERENCE(S):

Columbia Generating Station Control Room Evacuation and Remote Cooldown, ABN-CR-EVAC

NUMARC/NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Alert HA5

Columbia Generating Station Plant Specific EAL Guideline, HA5.1

7 Equipment Failures 7.2 Control Room Evacuation

7.2.S.1 Site Area Emergency

NUMARC IC: HS2 - Control room evacuation has been initiated, but plant control CANNOT be established.

APPLICABILITY:

Operating Conditions

1	2	3	4	5	

EMERGENCY ACTION LEVEL:

CR evacuation initiated
AND
Control of plant equipment needed to maintain adequate core cooling cannot be established at either the Remote Shutdown Panel or Alternate Remote Shutdown panel within 15 min. of the SRO in charge of the CR physically leaving the CR

BASES:

This Initiating Condition and its associated EAL address a condition where evacuation of the Control Room is necessary but expeditious transfer of safety systems has not occurred. Fission product barrier damage may not yet be indicated. A 15 minute transfer time was chosen for control to be reestablished to ensure that core uncovery with subsequent core damage does not occur and is consistent with NUMARC methodology.

Escalation of this event, if appropriate, would be by Fission Product Barrier Degradation, Abnormal Rad Levels/Radiological Effluent, or Emergency Director Judgment Initiating Conditions.

REFERENCE(S):

NUMARC/NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Site Area Emergency HS2

Columbia Generating Station Control Room Evacuation and Remote Cooldown, ABN-CR-EVAC

Columbia Generating Station Plant Specific EAL Guideline, HS2.1

7 Equipment Failures 7.3 Loss of Indication/Communications

7.3.U.1 Unusual Event

NUMARC IC: SU3 - Unplanned loss of most or all safety system annunciators or indication in the control room for greater than 15 minutes.

APPLICABILITY:

Operating Conditions

1	2	3			

EMERGENCY ACTION LEVEL:

Unplanned loss of most or all annunciators on P601, P602, P603, and Bd. C associated with safety-related equipment GT 15 minutes

BASES:

This Initiating Condition and its associated EAL recognize the difficulty associated with monitoring plant conditions without the use of a major portion of the annunciation equipment.

Quantification of "most" is left to the Emergency Director. It is not intended that plant personnel perform a detailed count of the instrumentation lost but rather make a judgment call with approximately 75% being the threshold. It is estimated that if approximately 75% of the annunciators are lost, there is an increased risk that a degraded plant condition could go undetected.

Control Room panels with annunciators for safety-related equipment required for off normal or emergency plan response include:

- P601
- P602
- P603
- Electrical Distribution on Bd C

Indications are available at other locations including Control Room back panels, using them to safely operate the plant would require increased surveillance.

Plant design provides redundant safety system indication powered from separate uninterruptible power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This will be addressed by the specific Technical Specification. The initiation of a Technical Specification- imposed plant shutdown related to the instrument loss will be reported via 10 CFR 50.72. If the shutdown is not in compliance with the Technical Specification Action Statement, the Unusual Event is based on 7.1.U.1, "Inability to Reach Required Shutdown Within Technical Specification Limits".

Compensatory nonalarming indications include the Process Computers and the Graphic Display System (GDS). It may include other permanently or temporarily installed monitoring systems if they allow the plant operators to compensate for the failed indications.

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

No Initiating Condition is indicated during cold shutdown and refueling due to the limited number of safety systems required for operation.

This event should be escalated to Alert 7.3.A.1 if a transient is in progress or the compensatory indications become unavailable.

REFERENCE(S):

NUMARC/NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Unusual Event SU3

Columbia Generating Station Technical Specifications

Columbia Generating Station Plant Specific EAL Guideline, SU3.1

7.3.U.2 Unusual Event

NUMARC IC: SU6 - Significant loss of onsite or offsite communications capabilities.

APPLICABILITY:

Operating Conditions

1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

Unplanned loss of ALL of the following onsite communications capability:

- Plant Public Address (PA) System
- Plant Telephone System
- Plant Radio System Operations and Security Channels

BASES:

This Initiating Condition and its associated EALs recognize a loss of communications capability that significantly degrades the Plant Operations staff's ability to perform tasks necessary for plant operations or the ability to communicate with offsite authorities.

REFERENCE(S):

NUMARC/NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Unusual Event SU6

Columbia Generating Station Plant Specific EAL Guideline, SU6.1

7 Equipment Failures 7.3 Loss of Indication/Communications

7.3.U.3 Unusual Event

NUMARC IC: SU6 - Significant loss of onsite or offsite communications capabilities.

APPLICABILITY:

Operating Conditions

1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

Unplanned loss of ALL of the following offsite communications capability:

- State/County Notification (Crash) System
- Offsite calling capability from the Control Room via direct telephone and fax lines
- Long distance calling capability on the Plant ("2000") Switch and Kootenai (Plant Support Facility)/Deschutes (Plant Engineering Center) ("8000") Switch

BASES:

This Initiating Condition and its associated EALs recognize a loss of communications capability that significantly degrades the Plant Operations staff's ability to perform tasks necessary for plant operations or the ability to communicate with offsite authorities. The loss of offsite communications capability is more comprehensive than that addressed by 10 CFR 50.72.

Long distance capability may be confirmed by placing a direct long distance call from both the 2000 and 8000 switches.

REFERENCE(S):

NUMARC/NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Unusual Event SU6

Columbia Generating Station Plant Specific EAL Guideline, SU6.1

7 Equipment Failures 7.3 Loss of Indication/Communications

7.3.A.1 Alert

NUMARC IC: SA4 - Unplanned loss of most or all safety system annunciators or indications in the control room with EITHER: 1) a significant transient in progress; OR 2) Compensatory nonalarming indicators are unavailable.

APPLICABILITY:

Operating Conditions	1	2	3			

EMERGENCY ACTION LEVEL:

Unplanned loss of <u>most</u> or <u>all</u> annunciators on P601, P602, P603, and Bd C associated with safety-related equipment GT 15 min.
AND either of the following:
A significant plant transient is in progress
OR
Compensatory non alarming indications are not available (plant computer systems and GDS)

BASES:

This Initiating Condition and its associated EAL recognize the difficulty associated with monitoring plant conditions without the use of a major portion of the annunciation equipment. It represents an increase in severity above that described in Unusual Event 7.3.U.1 in that either compensatory indications are <u>not</u> available or a significant transient is in progress.

Quantification of "most" is left to the Emergency Director. It is not intended that plant personnel perform a detailed count of the annunciation lost but, rather, make a judgment call with approximately 75% being the threshold. It is estimated that if approximately 75% of the annunciators are lost, there is an increased risk that a degraded plant condition could go undetected.

Control Room panels with annunciators and indicators for safety-related equipment for off normal or emergency plan response include:

- P601
- P602
- P603
- Electrical Distribution on Bd C

Indications are available at other locations including Control Room back panels. However, using them to safely operate the plant would require increased surveillance.

Plant design provides redundant safety system indication powered from separate uninterruptible power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This will be addressed by the specific Technical Specification. The initiation of a Technical Specification-imposed plant shutdown related to the instrument loss will be reported via 10 CFR 50.72. If the shutdown is not in compliance with the Technical Specification action statement, the Unusual Event is based on AU1, "Inability to Reach Required Shutdown Within Technical Specification Limits".

Compensatory nonalarming indications include the Process Computer Systems. It may include other permanently or temporarily installed monitoring systems if they allow the plant operators to compensate for the failed indications.

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

A "significant transient" includes response to automatic or manually initiated functions such as scrams, runbacks involving greater than 25% thermal power change, ECCS injection, or thermal power oscillations of 10% or greater.

No Initiating Condition is indicated during cold shutdown or refueling due to the limited number of safety systems required for operation.

This event should be escalated to Site Area Emergency 7.3.S.1 if the operating crew cannot monitor a transient in progress.

REFERENCE(S):

NUMARC/NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Alert SA4

Columbia Generating Station Technical Specifications

Columbia Generating Station Plant Specific EAL Guideline, SA4.1

7.3.S.1 Site Area Emergency

NUMARC IC: SS6 - Inability to monitor a significant transient in progress.

APPLICABILITY:

Operating Conditions	1	2	3			

EMERGENCY ACTION LEVEL:

Loss of most or all annunciators on P601, P602, P603 and Bd. C associated with safety-related equipment
AND
Compensatory non-alarming indications are unavailable (process computer system and GDS)
AND
Significant transient in progress
AND
Loss of indications needed to monitor ANY of the following plant critical safety parameters:

- Reactor power
- RPV level
- RPV pressure
- Drywell pressure
- Drywell temperature
- Wetwell pressure
- Wetwell/Drywell H2/O2 Concentrations
- Wetwell level
- Wetwell temperature
- Radioactive Gaseous Effluents

BASES:

This Initiating Condition and its associated EAL recognize the inability of the Control Room staff to monitor plant response to a transient. A Site Area Emergency is considered to exist if the Control Room staff cannot monitor the critical safety functions needed for protection of the public.

Critical safety functions include those plant parameters and functions that allow the plant operators to verify they have a coolable core geometry, that core cooling is maintained, and that containment is intact. The Columbia Generating Station Safety Analysis Report states that the safety functions include:

1. The accommodation of abnormal operational transients and postulated design basis accidents;

2. The maintenance of containment integrity;

3. The assurance of Emergency Core Cooling; and

4. The continuance of reactor coolant pressure boundary integrity.

Compensatory nonalarming indications include the Process Computer Systems. It may include other permanently or temporarily installed monitoring systems if they allow the plant operators to compensate for the failed indications.

A "significant transient" includes response to automatic or manually initiated functions such as scrams, runbacks involving greater than 25% thermal power change, ECCs injection, or thermal power oscillations of 10% or greater.

REFERENCE(S):

NUMARC/NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Alert SS6

Columbia Generating Station Technical Specifications

Columbia Generating Station Plant Specific EAL Guideline, SS6.1

8.1.U.1 Unusual Event

NEI 99-01 IC: Unexpected increase is ISFSI radiation

APPLICABILITY:

Operating Conditions

1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

Vailid reading for irradiated spent fuel in dry storage GT 2 times the ISFSI Technical Specification limits.

BASES:
This EAL addresses the degradation of irradiated spent fuel stored onsite in dry storage casks and provides a classification threshold for an unplanned or uncontrolled increase in radiation levels of the ISFSI storage casks. The casks are designed to standards identified in 10CFR72. Readings of twice ISFSI Technical Specification values are indicative of degradation of the irradiated spent fuel or storage cask.

REFERENCES(S):

10 CFR 72

ISFSI TS 3.2.2

ISFSI TS 3.2.3

NEI 99-01 Final Draft revision 4, February 2000 IC E-AU1

ISFSI FSAR

8.1.U.2 Unusual Event

NEI 99-01 IC: Damage to a Loaded Cask Confinement Boundary

APPLICABILITY:

Operating Conditions	1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

Any of the following conditions:

1) Natural phenomena events affecting a loaded cask confinement boundary:

Fire, Tornado
Flood, Earthquake
Explosion, Lightning
Complete SFSC air inlet blockage
Burial under debris
Extreme environmental temperature

OR

2) Accident conditions affecting a loaded cask confinement boundary:

Cask handling accident (e.g., drop)
Cask tip-over

OR

3) Any condition, in the opinion of the Emergency Director, that indicates a loss of loaded storage cask confinement boundary

BASES:

This EAL addresses the occurrence of an event of sufficient magnitude that a loaded cask confinement boundary is damaged or violated. This includes classification based on a loaded fuel storage cask confinement boundary loss with a potential to lead to degradation of the fuel during storage or posing an operational safety problem with respect to its removal from storage.

The results of the ISFSI Safety Analysis Report (SAR) were used to develop the list of natural phenomena events and accident conditions. This EAL addresses the response to a dropped cask, a tipped over cask, explosion, fire damage, or natural phenomena affecting a cask (e.g., seismic event, tornado, etc.).

Extreme environmental temperatures are unlikely to result in MPC or overpack equilibrated temperatures in excess of design limits. The limiting condition is taken to be the overpack concrete at 350 degrees Fahrenheit, per ISFSI FSAR Table 2.2.3. Due to the larger thermal inertia associated with the overpack concrete, extreme ambient temperature swings at this site will not cause the overpack concrete to exceed its design limit.

For conditions not explicitly detailed as EAL threshold values, the Emergency Director may use his judgement based on known conditions to classify a potential degradation in the level of ISFSI safety as an Unusual Event.

"Confinement boundary" is defined in the HI-STORM FSAR (Section 7.1) as:

the confinement boundary of the MPC consists of:
MPC shell
bottom baseplate
MPC lid (including the vent and drain port cover plates)
MPC closure ring
associated welds

The above items form a totally seal-welded vessel for the storage of design basis spent fuel assemblies.

ISFSI Technical Specifications allow time to complete required actions if cask confinement boundary integrity is not maintained; therefore, classification should not be made based on a loss of confinement boundary integrity by itself. However, loss of confinement boundary integrity coincident with an accident condition or natural phenomena affecting a cask would justify classification.

REFERENCE(S):

NEI 99-01 Final Draft Revision 4 February 2000 IC E-HU1

ISFSI FSAR Table 2.2.3

ISFSI FSAR 7.1

ISFSI FSAR 11.2.15

8.1.U.3 Unusual Event

NEI 99-01 IC: Confirmed security event with a potential loss of the level of safety of the ISFSI

EMERGENCY ACTION LEVEL:

Security event as identified by the Physical Security Plan and confirmed by on shift security supervision.

BASES:

This EAL is based on the Columbia Generating Station Physical Security Plan. Security events which do not represent a potential degradation in the level of safety of the ISFSI, are reported under 10CFR73.71, or in some cases, 10CFR50.72.

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

REFERENCE(S):

NEI 99-01 Final Draft Revision 4, February 2000 IC E-HU2
ISFSI FSAR

9 Hazards

Hazards are those nonplant system-related events which can directly or indirectly impact plant operation or reactor plant and personnel safety.

The events of this category have been grouped into the following types:

- Security Threats: This category includes unauthorized entry attempts into the Protected Area as well as bomb threats and sabotage attempts. Also addressed are actual security compromises threatening loss of physical control of the plant.

- Fire or Explosions: Fires can pose significant hazards to personnel and reactor safety. Appropriate for classification are fires within the site Protected Area or which may affect operability of vital equipment.

- Man-made Events: Man-made events are those nonnaturally occurring events which can cause damage to plant facilities such as aircraft crashes, missile impacts, toxic or flammable gas leaks or explosions from whatever source.

- Natural Events: Events such as hurricanes, earthquakes or tornadoes which have potential to cause damage to plant structures or equipment significant enough to threaten personnel or plant safety.

9 Hazards 9.1 Security Threats

9.1.U.1 Unusual Event

NUMARC IC: HU4 - Confirmed security event which indicates a potential degradation in the level of safety of the plant.

APPLICABILITY:

Operating Conditions

1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

Bomb device discovered within Plant Protected Area <u>but</u> outside a Safe Shutdown Building, Table 5
OR
Confirmed report of an attempted entry, sabotage or security threat that cannot be properly compensated for within 10 minutes

BASES:

Events which are believed by the Emergency Director to indicate a potential degradation of the level of safety of the plant should be declared an Unusual Event.

Security events which do not represent at least a potential degradation in this level of safety of the plant are reported under 10 CFR 73.71 or, in some cases, 10 CFR 50.72.

The 10 minute criteria to compensate is derived from regulatory guidance on implementation of 10 CFR 73.71, Reporting of Security Events.

For the purposes of this EAL, Safe Shutdown Buildings are considered to be the following locations:

- Vital portions of the Radwaste/Control Building
- Reactor Building
- Vital portions of the Turbine Building
- Standby Service Water Pump Houses
- Diesel Generator Building
- Diesel Generator Fuel Oil Storage Area

This list was developed from equipment locations identified in FSAR Table 3.2-1, Seismic Category I. The only equipment in the Turbine Building listed in FSAR Table 3.2-1, Seismic Category I is:

- DEH Pressure Switches
- RPS switches on turbine throttle valves
- Main Steam Line Radiation Monitors
- Turbine Building Ventilation Radiation Monitors
- Main Steam Piping up to MS-V-146 and the first stop valves

REFERENCE(S):

Columbia Generating Station Safeguards Contingency Plan

NUMARC/NESP-007, Methodology for Development of Emergency Actions Levels, Rev. 2, Unusual Event HU4

FSAR Table 3.2-1

Columbia Generating Station Plant Specific EAL Guideline, HU4.1

9 Hazards 9.1 Security Threats

9.1.U.2 Unusual Event

NUMARC IC: HU4 - Confirmed security event which indicates a potential degradation in the level of safety of the plant.

APPLICABILITY:

Operating Conditions

1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

Security event as defined by the Physical Security Plan
AND
reported by on-shift security supervision

BASES:

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

This EAL is based on Safeguards Contingency Plans contained within the Physical Security Plan. Security events which do not represent a potential degradation in the level of safety of the plant are reported under 10CFR 73.71 or in some cases under 10CFR50.72. Examples of security events that indicate a potential degradation in the level of safety of the plant are provided below for consideration.

Consideration should be given to the following types of security events when evaluating an event against the criteria of Physical Security Plan:

Sabotage
Hostage
Extortion
Civil disturbance
Strike action

Intrusion into the Protected Area by a hostile force would result in EAL escalation to an Alert.

REFERENCES(S):

Columbia Generating Station Safeguards Contingency Plan

NUMARC/NESP-007, Methodology for Development of Emergency Actions Levels, Rev. 2, Unusual Event HU4

Columbia Generating Station Plant Specific EAL Guideline, HU4.1

9 Hazards 9.1 Security Threats

9.1.U.3 Unusual Event

NUMARC IC: HU4 - Confirmed security event which indicates a potential degradation in the level of safety of the plant.

APPLICABILITY:

Operating Conditions	1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

Credible notification of a security threat to Columbia Generating Station

BASES:

The intent of this EAL is to ensure that appropriate notifications for the security threat are made in a timely manner.

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

A higher initial classification could be made based upon the nature and timing of the threat an potential consequences. Consideration should me made for upgrading the emergency response and classification in accordance with the Safeguards Contingency Plans and the Emergency Plan.

REFERENCES(S):

Columbia Generating Station Safeguards Contingency Plan

NUMARC/NESP-007, Methodology for Development of Emergency Actions Levels, Rev. 2, Unusual Event HU4

Columbia Generating Station Plant Specific EAL Guideline, HU4.1

9 Hazards 9.1 Security Threats

9.1.A.1 Alert

NUMARC IC: HA4 - Security event in a Plant Protected Area.

APPLICABILITY:

Operating Conditions

1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

Confirmed report of an intrusion by a hostile force into the Plant Protected Area

BASES:

This class of security events represents an escalated threat to plant safety above that contained in the Unusual event. For the purpose of this Initiating Condition, a civil disturbance which penetrates the Protected Area Boundary as well as an individual or group of individuals with known or suspected malicious intent is considered a hostile force.

Intrusion into a Safe Shutdown Building by a hostile force as defined in Site Area Emergency NS1 will escalate this event to Site Area Emergency.

REFERENCE(S):

Columbia Generating Station Safeguards Contingency Plan

NUMARC/NESP-007, Methodology for Development of Emergency Actions Levels, Rev. 2, Unusual Event HA4

Columbia Generating Station Plant Specific EAL Guideline, HA4.1

9 Hazards 9.1 Security Threats

9.1.S.1 Site Area Emergency

NUMARC IC: HS1 - Security event in a Plant Vital Area.

APPLICABILITY:

Operating Conditions

1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

Bomb device discovered or detonated within a Safe Shutdown Building, Table 5
OR
Confirmed report of intrusion by a hostile force into a Safe Shutdown Building, Table 5

BASES:

This class of security event represents an escalated threat to plant safety above that contained in Alert 8.1.A.1 in that a hostile force has progressed from the Protected Area to a Safe Shutdown Building.

For the purposes of this EAL, Safe Shutdown Buildings are considered to be the following locations:

- Vital portions of the Radwaste/Control Building
- Reactor Building
- Vital portions of the Turbine Building
- Standby Service Water Pump Houses
- Diesel Generator Building
- Diesel Generator Fuel Oil Storage Area

Equipment in Safe Shutdown Buildings is identified in FSAR Table 3.2-1, Seismic Category I. The only equipment in the Turbine Building listed in FSAR Table 3.2-1, Seismic Category I is:

- DEH Pressure Switches
- RPS switches on turbine throttle valves
- Main Steam Line Radiation Monitors
- Turbine Building Ventilation Radiation Monitors
- Main Steam Piping up to MS-V-146 and the first stop valves

For the purposes of this EAL, a civil disturbance which penetrates the Protected Area Boundary as well as an individual or group of individuals with known or suspected malicious intent can be considered a hostile force.

REFERENCE(S):

Columbia Generating Station Physical Security Plan

NUMARC/NESP-007, Methodology for Development of Emergency Actions Levels, Rev. 2, Unusual Event HS1

FSAR Table 3.2-1

Columbia Generating Station Plant Specific EAL Guideline, HS1.1, HS1.2

9 Hazards 9.1 Security Threats

9.1.G.1 General Emergency

NUMARC IC: HG1 - Security event resulting in loss of ability to reach and maintain cold shutdown.

APPLICABILITY:

Operating Conditions

1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

Loss of physical control of the CR due to security event
OR
Loss of physical control of the remote shutdown capability due to security event

BASES:

This Initiating Condition encompasses conditions under which a hostile force has taken physical control of areas required to reach and maintain cold shutdown.

For the purpose of this EAL, a civil disturbance which penetrates the Protected Area Boundary as well as an individual or group of individuals with known or suspected malicious intent is considered a hostile force.

REFERENCE(S):

Columbia Generating Station Safeguards Contingency Plan

NUMARC/NESP-007, Methodology for Development of Emergency Actions Levels, Rev. 2, Unusual Event HG1

Columbia Generating Station Plant Specific EAL Guideline, HG1.1, HG1.2

9.2.U.1 Unusual Event

NUMARC IC: HU2 - Fire within the Protected Area Boundary not extinguished within 15 minutes of detection or an explosion within Protected Area Boundary.

APPLICABILITY:

Operating Conditions	1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

Fire within or adjacent to any Safe Shutdown Building, Table 5, which is not extinguished within 15 minutes of either CR notification by plant personnel or receipt of an alarm
OR
Report by plant personnel of an unplanned explosion within the Protected Area Boundary resulting in visible damage to permanent structures or equipment

BASES:

This Initiating Condition and its associated EAL address fires that are of sufficient magnitude that they may be potentially significant precursors to damage to safety systems. This excludes items such as fires within administrative buildings or other structures not contiguous with a safe shutdown building, and other fires of no safety consequence or threat to a safe shutdown building such as slipping drive belts or overheated bearings, or other equipment not listed in FSAR Table 3.2-1, Seismic Category 1.

A fire alarm can be confirmed by multiple/redundant indications such as additional alarms on FCP-1 or FCP-2, fire pumps starting, fire suppression system discharge, fire water header pressure fluctuations or by notification by plant personnel. If an alarm must be verified by dispatching an individual to the scene, the 15 minute clock starts at the time of the alarm.

If an inspection of the area is completed within 15 minutes with no evidence of a fire (spurious alarm), no declaration need be made.

NEI 99-01 revision 4 defines fire as combustion characterized by heat and light. Sources of heat and smoke, such as slipping drive belts or overheated electrical components do not constitute fires. Observation of flame is preferred but not required if large quantities of heat and smoke are observed.

No attempt is made to assess the magnitude of the damage. The occurrence of the explosion with reports of damage (deformation/scorching) is sufficient for declaration.

Any security aspects of this event should be considered under Event Category 8.1, "Security Threats". If structural or equipment damage occurs within areas housing safe shutdown equipment and functions, the event may be escalated to Alert, 8.2.A.1.

For the purposes of this EAL, Safe Shutdown Buildings are considered to be the following locations:

- Vital portions of the Radwaste/Control Building
- Reactor Building
- Vital portions of the Turbine Building
- Standby Service Water Pump Houses
- Diesel Generator Building
- Diesel Generator Fuel Oil Storage Area

This list was developed from equipment locations identified in FSAR Table 3.2-1, Seismic Category I. Equipment in Safe Shutdown Buildings is identified in FSAR Table 3.2-1, Seismic Category I. The only equipment in the Turbine Building listed in FSAR Table 3.2-1, Seismic Category I is:

- DEH Pressure Switches
- RPS switches on turbine throttle valves
- Main Steam Line Radiation Monitors
- Turbine Building Ventilation Radiation Monitors
- Main Steam Piping up to MS-V-146 and the first stop valves

REFERENCE(S):

Title 10 of the Code of Federal Regulations, Part 50, Domestic Licensing of Production and Utilization Facilities, Appendix R, Fire Protection Program for Nuclear Power Facilities

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Unusual Event HU2

FSAR Table 3.2-1

Columbia Generating Station Plant Specific EAL Guideline, HU2.1

9 Hazards 9.2 Fire/Explosion Caused by Equipment Failure

9.2.A.1 Alert

NUMARC IC: HA2 - Fire or explosion affecting the operability of plant safety systems required to establish or maintain safe shutdown.

APPLICABILITY:

Operating Conditions

1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

Confirmed fire or explosion in a safe shutdown building, Table 5
AND either of the following:
Affected safe shutdown system parameters indicate degraded performance
OR
Report by plant personnel of visible damage to the affected safe shutdown building or equipment contained within the affected safe shutdown building

BASES:

As used here, an explosion is a rapid, violent, unconfined combustion or catastrophic failure of pressurized equipment, that potentially imparts significant energy to near-by structures and materials. The inclusion of a "report of visible damage" should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The occurrence of the explosion with reports of evidence of damage (e.g., deformation, scorching) is sufficient for declaration.

It is important to note that this EAL addresses a fire and not the degradation in performance of affected systems. The reference to damage of systems is used to identify the magnitude of the fire and to discriminate against minor fires, such as fires or explosions not near equipment listed in FSAR Table 3.2-1, Seismic Category I. The reference to Safe Shutdown Buildings is included to discriminate against fires in areas having a low probability of affecting safe operation. The significance here is not that a safety system was degraded but the fact that the fire was large enough to cause damage to these systems.

For the purposes of this EAL, Safe Shutdown Buildings are considered to be the following locations:

- Vital portions of the Radwaste/Control Building
- Reactor Building
- Vital portions of the Turbine Building
- Standby Service Water Pump Houses
- Diesel Generator Building
- Diesel Generator Fuel Oil Storage Area

This list was developed from equipment locations identified in FSAR Table 3.2-1, Seismic Category I. Equipment in Safe Shutdown Buildings is identified in FSAR Table 3.2-1, Seismic Category I. The only equipment in the Turbine Building listed in FSAR Table 3.2-1, Seismic Category I is:

- DEH Pressure Switches
- RPS switches on turbine throttle valves
- Main Steam Line Radiation Monitors
- Turbine Building Ventilation Radiation Monitors
- Main Steam Piping up to MS-V-146 and the first stop valves

REFERENCE(S):

Title 10 of the Code of Federal Regulations, Part 50, Domestic Licensing of Production and Utilization Facilities, Appendix R, Fire Protection Program for Nuclear Power Facilities

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Unusual Event HA2

ABN-CR-EVAC, Control Room Evacuation and Remote Cooldown

FSAR Appendix F, Fire Safety Evaluation

Columbia Generating Station Plant Specific EAL Guideline, HA2.1

9.3.U.1 Unusual Event

NUMARC IC: HU1 - Natural and destructive phenomena affecting the Protected Area Boundary.

APPLICABILITY:

Operating Conditions	1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

Vehicle crash into or projectile which impacts a Safe Shutdown Building, Table 5

BASES:

This EAL addresses such items as plane, helicopter, train, car, truck, or barge crash, or impact of other projectiles that may potentially damage plant structures containing functions and systems required for safe shutdown of the plant. The impact is of such force that damage to structures or equipment inside a Safe Shutdown Building may have occurred. If the crash is confirmed to affect equipment in a Safe Shutdown Building, the event may be escalated to Alert.

For the purposes of this EAL, Safe Shutdown Buildings are considered to be the following locations:

- Vital portions of the Radwaste/Control Building
- Reactor Building
- Vital portions of the Turbine Building
- Standby Service Water Pump Houses
- Diesel Generator Building
- Diesel Generator Fuel Oil Storage Area

Equipment in Safe Shutdown Buildings is identified in FSAR Table 3.2-1, Seismic Category I. The only equipment in the Turbine Building listed in FSAR Table 3.2-1, Seismic Category I is:

- DEH Pressure Switches
- RPS switches on turbine throttle valves
- Main Steam Line Radiation Monitors
- Turbine Building Ventilation Radiation Monitors
- Main Steam Piping up to MS-V-146 and the first stop valves

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Unusual Event HU1
Columbia Generating Station Plant Specific EAL Guideline, HU1.4

9 Hazards 9.3 Man-Made Events

9.3.U.2 Unusual Event

NUMARC IC: HU1 - Natural and destructive phenomena affecting the Protected Area Boundary.

APPLICABILITY:

Operating Conditions

1	2	3			

EMERGENCY ACTION LEVEL:

Turbine failure resulting in casing penetration or damage to turbine or generator seals

BASES:

Turbine failure with casing penetration or seal failure increases the potential for leakage of combustible fluids (oils and gas).

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Unusual Event HU1

Columbia Generating Station Plant Specific EAL Guideline, HU1.6

9 Hazards 9.3 Man-Made Events

9.3.U.3 Unusual Event

NUMARC IC: HU3 - Release of toxic or flammable gases affecting the Protected Area Boundary deemed detrimental to safe operation of the plant.

APPLICABILITY:

Operating Conditions	1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

Report or detection of toxic or flammable gases that could enter or have entered within the Protected Area Boundary in amounts that could affect the health of plant personnel or safe plant operation
OR
Report by local, county or state officials for evacuation or shelter of site personnel based on offsite event

BASES:

This Initiating Condition and its associated EALs are based on releases in concentrations within the Protected Area Boundary that may affect the health of plant personnel or the safe operation of the plant. This includes releases that originate onsite as well as releases that originate offsite but threaten onsite areas.

A toxic gas is considered to be any gas that is dangerous to life or limb by reason of inhalation or skin contact.

A combustible gas, if maintained at a concentration lower than the Lower Explosive Limit (LEL), will not explode due to ignition.

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Unusual Event HU3

Columbia Generating Station Plant Specific EAL Guideline, HU3.1

9 Hazards 9.3 Man-Made Events

9.3.A.1 Alert

NUMARC IC: HA1 - Natural and destructive phenomena affecting Safe Shutdown Buildings.

APPLICABILITY:

Operating Conditions	1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

Vehicle crash or projectile impact which impedes access to or damages equipment in a Safe Shutdown Building, Table 5

BASES:

This EAL addresses such items as plane, helicopter, train, car, truck, or barge crash, or impact of other projectiles that may affect plant structures containing functions and systems required for safe shutdown of the plant. If the crash is confirmed to affect equipment in a Safe Shutdown Building, then the event is an Alert.

This EAL is intended to address events that may have resulted in Safe Shutdown Buildings being subjected to forces beyond design limits and, thus, damage may be assumed to have occurred to safe shutdown systems. The initial "report" should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in these EALs to assess the actual magnitude of the damage. Escalation to a higher emergency class, if appropriate, will be based on the specific system malfunctions, fission product barrier degradation, abnormal radiological releases, or Emergency Director judgment EALs.

For the purposes of this EAL, Safe Shutdown Buildings are considered to be the following locations:

- Vital portions of the Radwaste/Control Building
- Reactor Building
- Vital portions of the Turbine Building
- Standby Service Water Pump Houses
- Diesel Generator Building
- Diesel Generator Fuel Oil Storage Area

This list was developed from equipment locations identified in FSAR Table 3.2-1, Seismic Category I. Equipment in Safe Shutdown Buildings is identified in FSAR Table 3.2-1, Seismic Category I. The only equipment in the Turbine Building listed in FSAR Table 3.2-1, Seismic Category I is:

- DEH Pressure Switches
- RPS switches on turbine throttle valves
- Main Steam Line Radiation Monitors
- Turbine Building Ventilation Radiation Monitors
- Main Steam Piping up to MS-V-146 and the first stop valves

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Unusual Event HA1

FSAR Table 3.2-1

Columbia Generating Station Plant Specific EAL Guideline, HA1.5

9 Hazards 9.3 Man-Made Events

9.3.A.2 Alert

NUMARC IC: HA1 - Natural and destructive phenomena affecting Safe Shutdown Buildings.

APPLICABILITY:

Operating Conditions

1	2	3			

EMERGENCY ACTION LEVEL:

Missiles generated from a turbine failure have resulted in visible structural damage to or penetration of a safe shutdown building, Table 5

BASES:

Turbine causing penetration, caused by failure of turbine rotating components, can result in missiles (blading, pieces of diaphragm, etc.) being hurled through the casing penetrations with such force, they can penetrate the turbine and come to rest a significant distance away. These turbine-generated missiles pose a threat to safety-related equipment if they cause visible structural damage to, or if they penetrate, Safe Shutdown Buildings.

This EAL is intended to address events that may have resulted in Safe Shutdown Buildings being subjected to forces beyond design limits and, thus, damage may be assumed to have occurred to safe shutdown systems. The initial "report" should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in these EALs to assess the actual magnitude of the damage. Escalation to a higher emergency class, if appropriate, will be based on the specific system malfunctions, fission product barrier degradation, abnormal radiological releases, or Emergency Director judgment EALs.

For the purposes of this EAL, Safe Shutdown Buildings are considered to be the following locations:

- Vital portions of the Radwaste/Control Building
- Reactor Building
- Vital portions of the Turbine Building
- Standby Service Water Pump Houses
- Diesel Generator Building
- Diesel Generator Fuel Oil Storage Area

This list was developed from equipment locations identified in FSAR Table 3.2-1, Seismic Category I. Equipment in Safe Shutdown Buildings is identified in FSAR Table 3.2-1, Seismic Category I. The only equipment in the Turbine Building listed in FSAR Table 3.2-1, Seismic Category I is:

- DEH Pressure Switches
- RPS switches on turbine throttle valves
- Main Steam Line Radiation Monitors
- Turbine Building Ventilation Radiation Monitors
- Main Steam Piping up to MS-V-146 and the first stop valves

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Unusual Event HA1

FSAR Table 3.2-1

Columbia Generating Station Plant Specific EAL Guideline, HA1.6

9.3.A.3 Alert

NUMARC IC: HA3 - Release of toxic or flammable gases within a facility structure which jeopardizes operation of systems required to maintain safe operations or to establish or maintain cold shutdown.

APPLICABILITY:

Operating Conditions	1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

Report or detection of toxic or flammable gases within a safe shutdown building, Table 5, in concentrations that will be life threatening to plant personnel or impede access to equipment needed for safe plant operation

BASES:

This EAL is based on gases that have entered a plant structure impeding access to equipment necessary for the safe operation of the plant. This EAL applies to Safe Shutdown Buildings and areas contiguous to plant vital areas or other significant buildings or areas. The intent of this EAL is not to include buildings (i.e., warehouses) or other areas that are not contiguous or immediately adjacent to plant vital areas or Safe Shutdown Buildings. It is appropriate that increased monitoring be done to ascertain whether consequential damage has occurred.

For the purposes of this EAL, Safe Shutdown Buildings are considered to be the following locations:

- Vital portions of the Radwaste/Control Building
- Reactor Building
- Vital portions of the Turbine Building
- Standby Service Water Pump Houses
- Diesel Generator Building
- Diesel Generator Fuel Oil Storage Area

This list was developed from equipment locations identified in FSAR Table 3.2-1, Seismic Category I. Equipment in Safe Shutdown Buildings is identified in FSAR Table 3.2-1, Seismic Category I. The only equipment in the Turbine Building listed in FSAR Table 3.2-1, Seismic Category I is:

- DEH Pressure Switches
- RPS switches on turbine throttle valves
- Main Steam Line Radiation Monitors
- Turbine Building Ventilation Radiation Monitors
- Main Steam Piping up to MS-V-146 and the first stop valves

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Unusual Event HA3

FSAR Table 3.2-1

Columbia Generating Station Plant Specific EAL Guideline, HA3.1, HA3.2

9 Hazards 9.4 Natural Events

9.4.U.1 Unusual Event

NUMARC IC: HU1 - Natural and destructive phenomena affecting the Protected Area Boundary.

APPLICABILITY:

Operating Conditions	1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

MINIMUM SEISMIC EARTHQUAKE alarm (H13-P851-S1-2.5)
AND
CR receives report from plant personnel who have felt an earthquake

BASES:

The method of detection associated with an earthquake of this intensity is based on the condition for a "felt earthquake" as defined in the EPRI-sponsored "Guidelines for, Nuclear Plant Response to an Earthquake". These methods include the activation of seismic monitoring instrumentation as evidenced by a valid alarm on P851-S1-2.5, "MINIMUM SEISMIC EARTHQUAKE EXCEEDED" along with confirmation from plant personnel who have physically felt the ground motion and recognize the event as an earthquake. An earthquake of this magnitude may be sufficient to cause some minor damage to plant structures or equipment within the Protected Area. Damage is considered to be minor since it does not affect physical or structural integrity. The event is not expected to affect the capabilities of plant safety functions. Due to the unpredictable nature of earthquakes, this may be a precursor to a more serious event and, therefore, represents a potential degradation in the level of safety of the plant.

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Unusual Event HU1

EPRI Guidelines for Nuclear Plant Response to an Earthquake

9 Hazards 9.4 Natural Events

9.4.U.2 Unusual Event

NUMARC IC: HU1 - Natural and destructive phenomena affecting the Protected Area Boundary.

APPLICABILITY:

Operating Conditions	1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

Weather Service projected winds GT 80 mph
OR
CR measured winds GT 66 mph (5 minute average at 33 ft)
OR
Report by plant personnel confirming the occurrence of a tornado striking within the Protected Area Boundary

BASES:

This event is a natural and potentially destructive phenomena that may accompany certain events such as a tornado or hurricane. These sustained high winds may also be produced by unstable weather conditions. However this event occurs, it may be a precursor to a more serious event and, therefore, represents a potential degradation in the level of safety of the plant.

A tornado touching down within the Protected Area is an observed event with the potential to cause damage to structures containing systems or functions necessary for the safe shutdown of the plant. As such, the occurrence of a tornado strike represents a potential degradation in the level of safety of the plant. If structural damage is confirmed, this event would be escalated to Alert 8.4.A.2. If it is determined that the occurrence of the tornado strike has either affected or caused the loss of shutdown cooling functions, then the consequences of the event are assessed under event category 7.1, "System Failures". The event may then be escalated via this category if appropriate.

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Unusual Event HU1

Columbia Generating Station Tornado/High Winds, ABN-WIND

Columbia Generating Station Plant Specific EAL Guideline, HU1.1

9 Hazards 9.4 Natural Events

9.4.U.3 Unusual Event

NUMARC IC: HU1 - Natural and destructive phenomena affecting the Protected Area Boundary.

APPLICABILITY:

Operating Conditions	1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

Range fires near the plant which threaten to reduce the level of safety

BASES:

Columbia Generating Station is located on a dry land steppe. Range fires routinely occur in this type of environment. This event has the potential to affect or cause the loss of safe shutdown systems and functions and, therefore, may be a precursor to a more serious event.

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Unusual Event HU1

Columbia Generating Station Plant Specific EAL Guideline, HU1.3

9.4.U.4 Unusual Event

NUMARC IC: HU1 - Natural and destructive phenomena affecting the Protected Area Boundary.

APPLICABILITY:

Operating Conditions

1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

Visible ash fallout from volcanic activity

BASES:

In May of 1980, Mount St. Helens volcano erupted. Prevailing winds spread up to 1/4″ of volcanic ash on the Columbia Generating Station site, with much heavier concentrations of ash several miles north of the site. Ash can clog diesel-generator air intakes and can be highly abrasive to rotating machinery. This event represents a potential degradation in the level of safety of the plant.

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Unusual Event HU1

Columbia Generating Station Design Basis Ash Fallout, PPM 4.12.4.5

Columbia Generating Station Plant Specific EAL Guideline, HU1.3

9 Hazards 9.4 Natural Events

9.4.U.5 Unusual Event

NUMARC IC: HU1 - Natural and destructive phenomena affecting the Protected Area Boundary.

APPLICABILITY:

Operating Conditions

1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

River level increase which threatens to flood the river pumphouse

BASES:

The Columbia Generating Station is located on an elevated plateau, well removed from risk of flooding by the Columbia River. The river pumphouse, located lower and closer to the river, may be prone to flooding. Should the river pumphouse be lost, the Standby Service Water Ultimate Heat Sink spray ponds have a 30 day supply of water. However, loss of the river pumphouse is deemed a potential degradation in the level of safety of the plant. The first Control Room indication of river pumphouse flooding would be TMU-LI-7 off-scale high.

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Unusual Event HU1

ABN-FLOODING, Flooding

Columbia Generating Station Plant Specific EAL Guideline, HU1.7

9.4.A.1 Alert

NUMARC IC: HA1 - Natural and destructive phenomena affecting Safe Shutdown Buildings.

APPLICABILITY:

Operating Conditions

1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

OPERATING BASIS EARTHQUAKE alarm (H13-P851-S1-5.1)
AND
CR receives report from plant personnel who have felt an earthquake

BASES:

An earthquake that exceeds the OBE level is beyond the design basis limits for the plant as specified in the Safety Analysis Report, Section 3.7, Seismic Design. A seismic event of this magnitude can cause damage to safety-related systems and functions. Detection of this event includes activation of seismic monitoring instrumentation as evidenced by a valid alarm on P851-S1-5.1, "OPERATING BASIS EARTHQUAKE EXCEEDED" along with confirmation from plant personnel who have physically felt the associated ground motion. An evaluation along with a thorough inspection of plant areas and systems will be used to determine the extent of plant damage and will provide the necessary information to determine if escalation to a higher emergency classification is required.

This EAL is intended to address events that may have resulted in Safe Shutdown Buildings being subjected to forces beyond design limits and, thus, damage may be assumed to have occurred to safe shutdown systems. The initial "report" should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in these EALs to assess the actual magnitude of the damage. Escalation to a higher emergency class, if appropriate, will be based on the specific system malfunctions, fission product barrier degradation, abnormal radiological releases, or Emergency Director judgment ICs.

For the purposes of this EAL, Safe Shutdown Buildings are considered to be the following locations:

- Vital portions of the Radwaste/Control Building
- Reactor Building
- Vital portions of the Turbine Building
- Standby Service Water Pump Houses
- Diesel Generator Building
- Diesel Generator Fuel Oil Storage Area

This list was developed from equipment locations identified in FSAR Table 3.2-1, Seismic Category I. Equipment in Safe Shutdown Buildings is identified in FSAR Table 3.2-1, Seismic Category I. The only equipment in the Turbine Building listed in FSAR Table 3.2-1, Seismic Category I is:

- DEH Pressure Switches
- RPS switches on turbine throttle valves
- Main Steam Line Radiation Monitors
- Turbine Building Ventilation Radiation Monitors
- Main Steam Piping up to MS-V-146 and the first stop valves

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Unusual Event HA1

FSAR Table 3.2-1

Columbia Generating Station Plant Specific EAL Guideline, HA1.1

9 Hazards 9.4 Natural Events

9.4.A.2 Alert

NUMARC IC: HA1 - Natural and destructive phenomena affecting Safe Shutdown Buildings.

APPLICABILITY:

Operating Conditions

1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

Weather Service projected winds GT 100 mph
OR
CR measured winds GT 76 mph (5 minute average at 33 ft)
OR
Report by plant personnel confirming the occurrence of a tornado striking a plant safe shutdown building, Table 5

BASES:

This event is a natural and potentially destructive phenomena that may accompany certain events such as a tornado or hurricane. These sustained high winds may also be produced by unstable weather conditions. However this event occurs, it may be a precursor to a more serious event and, therefore, represents a potential for substantial degradation in the level of safety of the plant. Sustained high winds at this level are beyond the design basis limits for the plant as described in SAR Section 3.3, Wind Loading. Wind loads of this magnitude have the potential to damage safety-related systems and functions. As such, the potential exists for substantial degradation of the level of the safety of the plant.

This EAL is intended to address events that may have resulted in Safe Shutdown Buildings being subjected to forces beyond design limits and, thus, damage may be assumed to have occurred to safe shutdown systems. The initial "report" should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in these EALs to assess the actual magnitude of the damage. Escalation to a higher emergency class, if appropriate, will be based on the specific system malfunctions, fission product barrier degradation, abnormal radiological releases, or Emergency Director judgment ICs.

For the purposes of this EAL, Safe Shutdown Buildings are considered to be the following locations:

- Vital portions of the Radwaste/Control Building
- Reactor Building
- Vital portions of the Turbine Building
- Standby Service Water Pump Houses
- Diesel Generator Building
- Diesel Generator Fuel Oil Storage Area

This list was developed from equipment locations identified in FSAR Table 3.2-1, Seismic Category I. Equipment in Safe Shutdown Buildings is identified in FSAR Table 3.2-1, Seismic Category I. The only equipment in the Turbine Building listed in FSAR Table 3.2-1, Seismic Category I is:

- DEH Pressure Switches
- RPS switches on turbine throttle valves
- Main Steam Line Radiation Monitors
- Turbine Building Ventilation Radiation Monitors
- Main Steam Piping up to MS-V-146 and the first stop valves

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Unusual Event HA1

ABN-WIND, Columbia Generating Station Tornado/High Winds

FSAR Table 3.2-1

Columbia Generating Station Plant Specific EAL Guideline, HA1.2

9 Hazards 9.4 Natural Events

9.4.A.3 Alert

NUMARC IC: HA1 - Natural and destructive phenomena affecting Safe Shutdown Buildings.

APPLICABILITY:

Operating Conditions

1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

Ash fallout from volcanic activity is severe enough to warrant plant shutdown

BASES:

In May of 1980, Mount St. Helens volcano erupted. Prevailing winds spread up to 1/4″ of volcanic ash on the Columbia Generating Station site, with much heavier concentrations of ash several miles north of the site. Ash can clog diesel-generator air intakes and can be highly abrasive to rotating machinery. Should the Ash fallout be severe enough to warrant plant shutdown, the event additionally represents a potential for substantial degradation in the level of safety of the plant.

This EAL is intended to address events that may have resulted in Safe Shutdown Buildings being subjected to forces beyond design limits and, thus, damage may be assumed to have occurred to safe shutdown systems. The initial "report" should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in these EALs to assess the actual magnitude of the damage. Escalation to a higher emergency class, if appropriate, will be based on the specific system malfunctions, fission product barrier degradation, abnormal radiological releases, or Emergency Director judgment ICs.

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Unusual Event HA1

Columbia Generating Station Design Basis Ash Fallout, PPM 4.12.4.5

Columbia Generating Station Plant Specific EAL Guideline, HA1.4

9 Hazards 9.4 Natural Events

9.4.A.4 Alert

NUMARC IC: HA1 - Natural and destructive phenomena affecting Safe Shutdown Buildings.

APPLICABILITY:

Operating Conditions

1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

Report by plant personnel of an event causing visible structural damage to a safe shutdown building, Table 5

BASES:

Visible structural damage is any observed physically degraded condition that indicates a significant impairment of the structural integrity of the building or area. An example of such a condition would be where a building sustained enough damage that it appears as if the roof could collapse at any time. The damage is based upon a report only. A detailed investigation or engineering evaluation is not required in order to classify the event.

This EAL is intended to address events that may have resulted in Safe Shutdown Buildings being subjected to forces beyond design limits and, thus, damage may be assumed to have occurred to safe shutdown systems. The initial "report" should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in these EALs to assess the actual magnitude of the damage. Escalation to a higher emergency class, if appropriate, will be based on the specific system malfunctions, fission product barrier degradation, abnormal radiological releases, or Emergency Director judgment ICs.

For the purposes of this EAL, Safe Shutdown Buildings are considered to be the following locations:

- Vital portions of the Radwaste/Control Building
- Reactor Building
- Vital portions of the Turbine Building
- Standby Service Water Pump Houses
- Diesel Generator Building
- Diesel Generator Fuel Oil Storage Area

This list was developed from equipment locations identified in FSAR Table 3.2-1, Seismic Category I. Equipment in Safe Shutdown Buildings is identified in FSAR Table 3.2-1, Seismic Category I. The only equipment in the Turbine Building listed in FSAR Table 3.2-1, Seismic Category I is:

- DEH Pressure Switches
- RPS switches on turbine throttle valves
- Main Steam Line Radiation Monitors
- Turbine Building Ventilation Radiation Monitors
- Main Steam Piping up to MS-V-146 and the first stop valves

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Unusual Event HA1

FSAR Table 3.2-1

Columbia Generating Station Plant Specific EAL Guideline, HA1.3

10 OTHER

The EALs defined in categories 1.0 through 9.0 specify the predetermined symptoms or events which are indicative of emergency or potential emergency conditions, and which warrant classification. While these EALs have been developed to address the full spectrum of possible emergency conditions which may warrant classification and subsequent implementation of the Emergency Plan, a provision for classification of emergencies based on operator/management experience and judgment is still necessary. The EALs of this category provide the Shift Manager or Emergency Director (ED) the latitude to classify emergency conditions consistent with the established classification criteria, based upon their judgment.

10 Other

10.1.U.1 Unusual Event

NUMARC IC: HU5 - Other conditions existing which, in the judgment of the Emergency Director, warrant declaration of an Unusual Event.

APPLICABILITY:

Operating Conditions

1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

In the judgment of the Emergency Director, events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant

BASES:

Events which are believed by the Emergency Director to indicate a potential degradation of the level of safety of the plant should be declared an Unusual Event. For those cases where the degradation in the level of the safety of the plant is tied to equipment or system malfunctions, the decision that the component is degraded should be based upon its functionality and not its operability.

A system, subsystem, train, component or device, though degraded in equipment condition or configuration, is functional if it is capable of maintaining respective system parameters within acceptable design limits.

Releases of radioactive material requiring offsite response or monitoring are not expected to occur at the Unusual Event level unless further degradation of safety systems occur. However, if one does occur, it will be classified under Category 5 “Radioactivity Release”.

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Unusual Event HU5

Columbia Generating Station Plant Specific EAL Guideline, HU1.5

10 Other

10.1.U.2 Unusual Event

NUMARC IC: Loss or potential loss of Primary Containment

APPLICABILITY:

Operating Conditions	1	2	3			

EMERGENCY ACTION LEVEL:

Any event, in the judgment of the Emergency Director, that could lead to or has led to a loss or potential loss of primary containment as indicated by Fission Product Barrier Degradation Table, Table 6

BASES:

This EAL addresses any other factors that are to be used by the Emergency Director in determining whether the containment barrier is lost or potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this EAL as a factor in Emergency Director judgment that the barrier may be considered lost or potentially lost.

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Fission Product Barrier Basis Information for Table 3

Columbia Generating Station Fission Product Barrier Evaluation

Columbia Generating Station Plant Specific EAL Guideline, PC6.1

10 Other

10.1.A.1 Alert

NUMARC IC: HA6 - Other conditions existing which, in the judgment of the Emergency Director, warrant declaration of an Alert.

APPLICABILITY:

Operating Conditions

1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

In the judgment of the Emergency Director, events are in progress or have occurred which indicate actual or potential substantial degradation of the level of safety of the plant

BASES:

This Emergency Action Level is intended to address unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the Alert emergency class. This includes a determination by the Emergency Director that additional assistance similar to that provided by the TSC and OSC staffs, including a transfer of the Emergency Director responsibilities to the TSC, is necessary for the event to be effectively mitigated. Transfer of Emergency Director duties for classification, offsite notifications and PAR decisions, is used as an initiator since an event significant enough to warrant transfer of command and control is a substantial reduction in the level of safety of the plant.

Activation of the TSC outside of the Emergency Plan in support of the Control Room staff is permissible. Releases that are expected to be limited to a small fraction of the EPA Protective Action Guideline exposure levels are addressed under Category 5 "Radioactivity Release".

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Unusual Event HA6

Columbia Generating Station Plant Specific EAL Guideline, HA6.1

10 Other

10.1.A.2 Alert

NUMARC IC: Loss or potential loss of fuel clad or RCS

APPLICABILITY:

Operating Conditions

1	2	3			

EMERGENCY ACTION LEVEL:

Any event, in the judgment of the Emergency Director, that could lead or has led to a loss or potential loss of either fuel clad or RCS barrier as indicated by Fission Product Barrier Degradation Table, Table 6

BASES:

This EAL addresses any other factors that are to be used by the Emergency Director in determining whether the fuel clad or RCS barriers are lost or potentially lost. In addition, the inability to monitor the barriers should also be considered in this EAL as a factor in Emergency Director judgment that the barriers may be considered lost or potentially lost.

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Fission Product Barrier Basis Information for Table 3

Columbia Generating Station Fission Product Barrier Evaluation

Columbia Generating Station Plant Specific EAL Guideline, FC5.1, RCS6.1

10 Other

10.1.S.1 Site Area Emergency

NUMARC IC: HS3 - Other conditions existing which, in the judgment of the Emergency Director, warrant declaration of a Site Area Emergency.

APPLICABILITY:

Operating Conditions

1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

In the judgment of the Emergency Director, events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public

BASES:

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

Radioactive releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels except within the site boundary. Radioactive releases to the general public are addressed under Category 5 "Radioactivity Release".

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Unusual Event HS3

Columbia Generating Station Plant Specific EAL Guideline, HS3.1

10 Other

10.1.S.2 Site Area Emergency

NUMARC IC: Loss or potential loss of any two fission product barriers

APPLICABILITY:

Operating Conditions

1	2	3			

EMERGENCY ACTION LEVEL:

Any event, in the judgment of the Emergency Director, that could lead or has led to a loss or potential loss of any two fission product barriers as indicated by Fission Product Barrier Degradation Table 6.

BASES:

This EAL addresses unanticipated conditions affecting fission product barriers which are not addressed explicitly elsewhere. Declaration of an emergency is warranted because conditions exist which are believed by the Emergency Director to fall under the emergency class description for Site Area Emergency.

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Fission Product Barrier Basis Information for Table 3

Columbia Generating Station Fission Product Barrier Evaluation

Columbia Generating Station Plant Specific EAL Guideline

10 Other

10.1.G.1 General Emergency

NUMARC IC: HG2 - Other conditions existing which, in the judgment of the Emergency Director, warrant declaration of a General Emergency.

APPLICABILITY:

Operating Conditions

1	2	3	4	5	def

EMERGENCY ACTION LEVEL:

In the judgment of the Emergency Director, other conditions exist which indicate either of the following:

- Actual or imminent substantial core degradation or melting with the potential for loss of containment integrity

  OR

- Potential for uncontrolled radionuclide releases which can reasonably be expected to exceed EPA PAG plume exposure levels outside the site boundary

BASES:

This Emergency Action Level is intended to address unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the General Emergency class. Imminent in this context means mitigation strategies and actions are not successful in preventing a challenge to Fuel Clad, Reactor Coolant Pressure Boundary or Primary Containment.

Radioactive releases may exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area. Radioactive releases to the general public are addressed under Category 5 "Radioactivity Release".

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Unusual Event HG2

Columbia Generating Station Plant Specific EAL Guideline, HG2.1

10 Other

10.1.G.2 General Emergency

NUMARC IC: A loss of any two fission product barriers and loss or potential loss of the third

APPLICABILITY:

Operating Conditions

1	2	3			

EMERGENCY ACTION LEVEL:

Any event, in the judgment of the Emergency Director, that could lead or has led to a loss of any two fission product barriers and loss or potential loss of the third as indicated by Fission Product Barrier Degradation Table, Table 6

BASES:

This EAL addresses unanticipated conditions affecting fission product barriers which are not addressed explicitly elsewhere. Declaration of an emergency is warranted because conditions exist which are believed by the Emergency Director to fall under the emergency class description for the General Emergency class.

REFERENCE(S):

NUMARC NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, Fission Product Barrier Basis Information for Table 3

Columbia Generating Station Fission Product Barrier Evaluation

Columbia Generating Station Plant Specific EAL Guideline

TABLE 6, FISSION PRODUCT BARRIER DEGRADATION TABLE BASES

Fuel Clad Loss	Fuel Clad Potential Loss	RCS Loss	RCS Potential Loss	PC Loss	PC Potential Loss
Coolant activity GT 300 μCi/gm dose equivalent iodine	RPV level LT -161 inches (for ATWS conditions, RPV level LT -192 inches)	Containment Radiation Monitor CMS-RIS-27E and CMS-RIS-27F reading GT 70 R/hr	Total RCS leakage GT 30 gpm inside PC or EDR-FRS-623, Pen 2 upscale high	Rapid unexplained decrease of PC pressure following an initial increase	Containment Radiation Monitor CMS-RIS-27E and CMS-RIS-27F reading GT 14,000 R/hr
Containment Radiation Monitor CMS-RIS-27E and CMS-RIS-27F reading GT 3,600 R/hr		RPV level LT -161 inches (for ATWS conditions, RPV level LT -192 inches)	Unisolable primary system discharging outside PC resulting in any area temperature or radiation level above Maximum Safe Operating Values (PPM 5.3.1, "Secondary Containment Control")	Drywell pressure response not consistent with LOCA conditions	PC H_2 and O_2 concentrations GT 6% H_2 and 5% O_2
Entry into Severe Accident Guidelines		Drywell pressure GT 1.68 psig with indications of RCS leakage inside drywell		Failure of containment isolation valves (LCS Table 1.6.3.1-1) in any one line to close following auto or manual initiation AND downstream pathway outside primary containment exists OR	Entry into Severe Accident Guidelines Loss of pressure suppression function
				Unisolable primary system discharging outside PC resulting in any area temperature or radiation level above Maximum Safe Operating Values (PPM 5.3.1, "Secondary Containment Control")	Cannot maintain plant parameters within HCTL, SRVTPLL
				Intentional venting per PPM 5.2.1, "Primary Containment Control"	Wetwell pressure exceeds PSP
					PC pressure GT PCPL

<u>Any</u> event, in the judgment of the Emergency Director, that could lead or has led to a loss or potential loss of the fuel clad barrier	<u>Any</u> event, in the judgment of the Emergency Director, that could lead or has led to a loss or potential loss of the RCS barrier	<u>Any</u> event, in the judgment of the Emergency Director, that could lead to or has led to a loss or potential loss of primary containment barrier

Fuel Clad Loss Indicators

Coolant activity GT 300 μCi/gm dose equivalent iodine

Fuel Clad barrier damage is indicated by a coolant activity of 300 μCi/gm dose equivalent I-131. This amount of activity is well above that expected for iodine spikes and corresponds to approximately 2-5% fuel clad failure in accordance with assessment performed by the NUMARC EAL task force. This amount of clad failure indicates significant clad heating and, thus, the Fuel Clad barrier is considered lost.

Containment Radiation Monitor CMS-RIS-27E and CMS-RIS-27F reading GT 3,600 R/hr

A 3,600 R/hr reading on CMS-RIS-27E and CMS-RIS-27F is used to indicate a release of reactor coolant, with elevated activity indicative of fuel damage, into the drywell. Therefore, this condition represents loss of both the fuel clad and RCS barriers. This value assumes an instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with coolant concentration associated with 5% clad failures into the drywell atmosphere. Columbia Generating Station has elected to provide an example dealing with the top end of the 2-5% range discussed in NESP-007. This value assumes an instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory into the drywell atmosphere.

Entry into Severe Accident Guidelines

Entry into Severe Accident Guidelines is indicative of both a loss of fuel clad and RCS barriers because:

1) RPV water level cannot be restored and maintained above -161 inches, or

2) For ATWS conditions, RPV water level cannot be maintained above -192", or

3) If RPV water level cannot be determined, RPV flooding for ATWS or non-ATWS conditions cannot be established or maintained.

Entry into Severe Accident Guidelines is indicative of a potential loss of primary containment because actions to flood the containment may jeopardize the pressure suppression capability of the containment or result in the need to vent the RPV or primary containment.

Fuel Clad Potential Loss Indicators

RPV level LT -161 inches (for ATWS conditions, RPV level LT -192 inches)

Indicating fuel clad barrier potential Loss at -161″ (Top of Active Fuel) or -192 inches for ATWS conditions, ensures that an emergency is declared before fuel perforation occurs. An unintentional level decrease to below -161″ is also indicative of a large RCS break, or a smaller break with loss of high pressure makeup.

During ATWS conditions, level may be intentionally lowered to reduce reactor power. This level decrease is not indicative of an RCS break as long as level is maintained above -192 inches (minimum steam cooling RPV water level.

RCS Loss Indicators

Containment Radiation Monitor CMS-RIS-27E and CMS-RIS-27F reading GT 70 R/hr

A 70 R/hr reading on CMS-RIS-27E and CMS-RIS-27F is used to indicate a loss of the Reactor Coolant System barrier. This value assumes a 0.1% clad damage and the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory into the drywell atmosphere. The value of 0.1% clad damage was assumed to be the greatest amount of fuel failure under which power operation could occur.

RPV level LT -161 inches (for ATWS conditions, RPV level LT -192 inches)

An unintentional level decrease to below -161″ or -192 inches for ATWS conditions is also indicative of a large RCS break, or a smaller break with loss of high pressure makeup.

During ATWS conditions, level may be intentionally lowered to reduce reactor power. This level decrease is not indicative of an RCS break as long as level is maintained above -192 inches (minimum steam cooling RPV water level.

Drywell pressure GT 1.68 psig with indications of RCS leakage inside drywell

The 1.68 psig drywell pressure for the Reactor Coolant System barrier loss is based on the drywell pressure scram and isolation setpoint and indicates a Loss of Coolant Accident (LOCA). A potential loss of the Reactor Coolant System barrier would not result in an increasing drywell pressure and, therefore, no indicator is provided. The qualifier of "with indications of RCS leak inside drywell" is included as an indicator of RCS boundary degradation and eliminates a drywell pressure increase due to a loss of drywell ventilation.

RCS Potential Loss Indicators

Total RCS leakage GT 30 gpm inside PC or EDR-FRS-623, Pen 2 upscale high

Total leakage is considered to be the total of both identified and unidentified leakage as measured on EDR-FRS-623 pen 1 (unidentified - Floor Drain Sump Fill Rate) and pen 2 (identified - Equipment Drain Sump Fill Rate). The maximum measurable identified leak rate (pen 2) in the Control Room at Columbia Generating Station is 30 gpm, therefore 30 gpm is used instead of the 50 gpm limit recommended by NUMARC.

Unisolable primary system discharging outside PC resulting in any area temperature or radiation level above Maximum Safe Operating Values (PPM 5.3.1, "Secondary Containment Control")

The presence of elevated area temperatures and/or radiation levels in the secondary containment may be indicative of an unisolable primary system leakage outside the primary containment. These conditions represent a loss of the containment barrier and a potential loss of the RCS barrier.

PC Loss Indicators

Rapid unexplained decrease of PC pressure following an initial increase

Rapid unexplained loss of pressure (i.e., not attributable to drywell spray or condensation effects) following initial pressure increase indicates a loss of containment integrity. In interpreting this EAL, an initial increase is any PC pressure increase above 1.68 psig. A rapid decrease should be considered any decrease which occurs faster than the initial increase.

Drywell pressure response not consistent with LOCA conditions

Containment or drywell pressure responses not consistent with LOCA conditions indicate a loss of the Primary Containment barrier. This may be noticed as a decrease in drywell pressure when no operation action (e.g., starting drywell cooling fans) has been taken. It would also include a failure of the drywell pressure to increase as expected during a LOCA.

Failure of containment isolation valves (LCS Table 1.6.1.3-1) in any one line to close following auto or manual initiation
AND
downstream pathway outside primary containment exists
OR

Unisolable primary system discharging outside PC resulting in any area temperature or radiation level above Maximum Safe Operating Values (PPM 5.3.1, "Secondary Containment Control")

The failure to isolate condition is intended to cover containment isolation failures allowing a direct flow path to the environment such as a failure of MSIVs to close with open valves downstream to the turbine or condenser. Downstream path outside primary containment does not mean leakage incident to normal system integrity. Only those penetrations required to isolate per Technical Specifications should be considered.

The presence of elevated area temperatures and/or radiation levels in the secondary containment may be indicative of an unisolable primary system leakage outside the primary containment. These conditions represent a loss of the containment barrier and a potential loss of the RCS barrier.

Intentional venting per PPM 5.2.1, "Primary Containment Control"

Venting, if necessary to prevent failure of primary containment, is included as a loss of primary containment. This is specified in PPM 5.2.1, "Primary Containment Control" when containment hydrogen and oxygen concentrations are in excess of or cannot be determined to be below combustible limits or when wetwell pressure approaches PCPL. However, routine venting per PPM 2.3.1, as long as radioactivity release rates are maintained, is not considered a loss of primary containment.

PC Potential Loss Indicators

Containment Radiation Monitor CMS-RIS-27E and CMS-RIS-27F reading GT 14,000 R/hr

An 14,000 R/hr reading on CMS-RIS-27E and CMS-RIS-27F is used to indicate potential failure of the primary containment barrier. It is a value that indicates significant fuel damage well in excess of that associated with the loss of both Fuel Clad and RCS barriers. A major release of radioactivity requiring offsite protective actions is not possible unless a major failure of fuel cladding allows radioactive material to be released from the core into the reactor coolant.

Regardless of whether containment is challenged, this amount of activity in containment, if released, could have such severe consequences that it is prudent to treat this as a potential loss of containment, such that a General Emergency declaration is warranted. This level of activity is indicative of approximately 20% clad failure. This value assumes an instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory into the drywell atmosphere.

PC H_2 and O_2 concentrations GT 6% H_2 and 5% O_2

Hydrogen and Oxygen concentrations at or above PPM 5.2.1, "Primary Containment Control", Table 19, Combustible Limits, in the drywell or wetwell represent a potential for a deflagration with a subsequent containment failure.

Entry into Severe Accident Guidelines

Entry into Severe Accident Guidelines is indicative of a potential loss of primary containment because actions to flood the containment may jeopardize the pressure suppression capability of the containment or result in the need to vent the RPV or primary containment.

Loss of pressure suppression function

Physical degradation of the containment structure as indicated by an equalization between suppression chamber and drywell pressures constitutes a loss of pressure suppression capability and should be considered a potential loss of containment.

Cannot maintain plant parameters within HCTL, or SRVTPLL

The Heat Capacity Temperature Limit (HCTL) is the highest wetwell temperature at which initiation of RPV depressurization will not result in exceeding the Primary Containment Pressure Limit (PCPL) before the rate of energy transfer to the containment is within the capacity of the containment vent. The HCTL is used to preclude failure of the containment or equipment necessary for safe shutdown by assuring that RPV blowdown does not cause containment pressure to exceed the PCPL. The potential loss occurs when RPV pressure and wetwell temperature cannot be maintained below HCTL.

The SRV Tail Pipe Level Limit (SRVTPLL) is the highest wetwell water level at which opening of an SRV will not result in exceeding the code allowable stresses in the tailpipe, tailpipe supports, quenchers or quencher supports. This level is a function of RPV pressure and the Limit is utilized to preclude SRV system failure and containment failure. The consequences of operating SRVs when wetwell water level exceeds the SRVTPLL may include direct pressurization of the containment from a break in the SRV tail pipe. The resulting primary containment pressurization could cause containment failure.

Wetwell pressure exceeds PSP

Entry into the unsafe region of the Pressure Suppression Pressure curve (PPM 5.2.1, "Primary Containment Control", Figure F, PSP) is included as a potential primary containment barrier loss. A rapid depressurization of the RPV (e.g., occurrence of a large break LOCA or initiation of ADS) at wetwell pressures in excess of the PSP may cause either:

- Wetwell pressure responses indicative of a failure in the drywell-to-wetwell boundary, or
- Wetwell pressure increases to or beyond the Primary Containment Pressure Limit (PPM 5.2.1, "Primary Containment Control", Figure B, PCPL).

PC pressure GT PCPL and increasing

Containment pressures that exceeds 34.7 psig, the maximum expected pressure following a LOCA, have the potential to result in a loss of the containment barrier. Preparations to vent containment are required by PPM 5.2.1, "Primary Containment Control" when Drywell pressure exceeds PCPL and before the Wetwell pressure reaches the Primary Containment Pressure Limit (PCPL). Therefore, this condition is considered a potential loss of containment.