WOLF CREEK
NUCLEAR OPERATING CORPORATION

December 15, 1993

Robert C. Hagan
Vice President Nuclear Assurance

NA 93-0236

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-137
Washington, D. C. 20555

Subject: Docket No. 50-482: Submittal of Revised Emergency Action Levels for Review and Approval

Gentlemen:

This letter submits a revision to the Emergency Action Levels (EALs) for Wolf Creek Nuclear Operating Corporation (WCNOC). These changes implement the NUMARC Guidance on EALs as endorsed by Regulatory Guide 1.101, "Emergency Planning and Preparedness for Nuclear Power Reactors," revision 3. Due to the significance of changes and based on guidance provided in the NUMARC workshops, the changes are being submitted for review and approval prior to implementation. A review of these EAL changes has been performed in accordance with the requirements of 10 CFR 50.54(q) and it has been determined that the changes do not decrease the effectiveness of the Wolf Creek Radiological Emergency Response Plan. WCNOC has discussed this submittal with Mr. W. D. Reckley, NRC Project Manager for WCNOC. Based on this discussion and this submittal, WCNOC would like to implement the revised EALs by the end of the first quarter or beginning of the second quarter 1994.

Attachment 1 of this letter provides a comparison matrix reflecting WCNOC's currently approved EALs, the NUMARC proposed EAL and the WCNOC proposed EAL. This matrix is being supplied to aid in this review. Attachment 2 of this letter provides a copy of procedure EPP 01-2.1, "Emergency Classifications," which incorporates the revised EALs. The Radiological Emergency Response Plan will be revised within 30 days of NRC approval of the revised EALs.

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P.O. Box 411 / Burlington, KS 66839 / Phone: (316) 364-8831

If you have any questions concerning this submittal please contact me at (316)-364-8831, extension 4553 or Mr. Kevin J. Moles at extension 4565.

Very truly yours,

Robert C. Hagan

Robert C. Hagan
Vice President Nuclear Assurance

RCH/jra

Attachments (2)

cc: J. L. Milhoan (NRC), w/attachments
G. A. Pick (NRC), w/attachments
W. D. Reckley (NRC), w/attachments
D. B. Spitzberg (NRC), w/attachments
L. D. Yandell (NRC), w/attachments

Attachment 1 to NA 93-0237

EAL MATRIX

ABNORMAL RAD LEVELS

NUMARC	NEW WOLF CREEK	OLD WOLF CREEK
AU 1	RER 6 (NUE)	Radioactive effluent Tech Spec release limit(s) exceeded (NUE)
AU 2	RER 3 (NUE)	Rad levels or airborne which indicates a severe degradation in the control of Rad materials (e.g. increase by factor of 1000 in direct radiation reducing within facility (Alert)
	FHA 1 (NUE)	NONE
AA1	RER 2 (Alert)	A measured or projected dose rate of one mR/Hr @ site boundary under actual meteoroligical conditions (Alert)
AA2	FHA 2 (Alert) Fuel Bldg FHA 3 (Alert) CTMT	An irradiated fuel handling accident which breaches the cladding with release of radioactivity to containment or fuel building (Alert)
AA3	RER 7 (Alert)	Rad levels or airborne which indicates a severe degradation in the control of Rad materials (e.g. increase by factor of 1000 in direct radiation reducing within facility (Alert)
AS1	RER 4 (SAE)	Measured or projected dose rate of >50mR/Hr for 1/2 hour or >500 mR/Hr WB for 2 min. or 5 times these levels to thyroid @ site boundary under actual meteorological conditions (SAE)
AG1	RER 5 (GE)	Measured or projected dose rates of one R/Hr WB or 5 R/Hr thyroid at site boundary under actual meteorological condition (GE)

EAL MATRIX

FISSION PRODUCT BARRIER DEGRADATION

NUMARC	NEW WOLF CREEK	OLD WOLF CREEK
FU 1	ADM 3	Same as ADM 3
FA 1 FS 1 FG 1	RCS LRCB 2 SGTF 2 SGTF 12 MSLB 3 MSLB 6 MSLB 8 FEF 2	See copies of old Wolf Creek EAL for fission product barrier breach or challenge indications
	FUEL LRCB 3 SGTF 1 MSLB 1 FEF 1	
	CONTAINMENT LRCB 4 LRCB 6 LRCB 7 SGTF 6 SGTF 11 SGTF 17 SGTF 19 MSLB 5 MSLB 10 FEF 3 FEF 4 FEF 5	

Attachment 2.0
INDICATIONS OF FUEL CLADDING
BREACH OR CHALLENGE
(Page 1 of 1)

Sub criticality CSFST: Red or Orange →

OR

Core Cooling CSFST: Red or Orange →

OR

Heat Sink CSFST: Red →

OR

CVCS letdown Rad monitor reading ≥ High Alarm

AND

Confirmed analysis indicates an INCREASE of ≥ 63 (uCi/g) gross activity in the RCS in any 30 minute period →

OR

Containment Air Particulate, Iodine and Radiogas monitors, Area Radiation Monitors, OR Containment High Rad Monitors increase significantly with the RCS intact due to coolant which normally leaks to containment →

OR

SAMPLING VERIFIES THE EXISTENCE OF:

≥ 600 uCi/g gross activity →

OR

≥ 5 times the Tech. Spec. 3.4.8 limit of 100/E bar uCi/g of gross radioactivity →

OR

≥ 5 times the dose equivalent I^{131} limits of Tech Spec Figure 3.4.1 →

FUEL CLADDING ASSUMED BREACHED OR CHALLANGED

Attachment 3.0
INDICATIONS OF REACTOR COOLANT SYSTEM (RCS)
BREACH OR CHALLENGE
(Page 1 of 1)

RCS Integrity CSFST: Red or Orange	→	RCS IS ASSUMED BREACHED OR CHALLENGED
OR		
Heat Sink CSFST: Red	→	
OR		
General: The inability to maintain normal liquid inventory within the Reactor Coolant System (RCS) by normal operation of the Chemical and Volume Control System which is considered as one CCP discharging to the charging header.	→	
OR		
Loss to Containment: Both Containment Air Particulate, Iodine or Radiogas monitors increase very rapidly to off scale high	→	
OR		
Loss to Steam Generator: Any narrow range steam generator level increasing in an uncontrolled manner	→	

Attachment 4.0
INDICATIONS OF CONTAINMENT
BREACH OR CHALLENGE

Containment Integrity CSFST: Red or Orange	→	CONTAINMENT ASSUMED BREACHED OR CHALLENGED
OR		
Non-isolable leak of Main Steam Line outside Containment	→	
OR		
Loss of Containment Integrity where applicable Technical Specifications cannot be satisfied	→	
OR		
RCS leak to interfacing system outside containment (e.g. RHR) with indication of release to environment as indicated by: 1) Unit Vent Rad monitor OR 2) Radwaste Vent Rad monitor OR 3) Radiation Surveys	→	

EAL MATRIX

HAZARDS AND OTHER CONDITIONS

NUMARC	NEW WOLF CREEK	OLD WOLF CREEK
HU 1	NP 1 (NUE) Earthquake	Earthquake effects have been seen, heard or felt inside the PAB (NUE)
	NP 6 (NUE) Tornado	Tornado touching down in PAB or Switchyard (NUE)
	OH 3 (NUE) OH 7 (NUE) (Vehicle Crash)	Aircraft crash inside PAB or Switchyard (NUE)
	OH 5 (NUE) (Explosion)	Explosion in a vital area resulting in major equipment damage (NUE)
	OH 6 (NUE) (Turbine failure)	NONE
HU 2	FR 1 (NUE)	Fire in PAB requiring offsite assistance (NUE)
HU 3	OH 1 (NUE)	Toxic or flammable gas release which presents a danger to personnel in the PAB (NUE)
HU 4	LPC/SC1 (NUE)	Confirmed security threat or attempted sabotage (NUE)
HU 5	OH 9 (NUE)	Judgemental authority of the DED/DEM
HA 1	NP 2 (Alert) Earthquake	OBE limits exceeded: Alarm 98C or 98D and earthquake effects have been seen, heard or felt (Alert)
	NP 7 (Alert) NP 5 (Alert) Tornado and highwinds	Continuous winds of $\geq$ 95 mph (Alert). And natural phenomenon which threaten one fission product barrier. (Alert)
	OH 4 (Alert)	Aircraft crash, missiles, fire or explosions causing severe damage to both trains of safe shutdown equipment or causes challenges to two Fission Product Barriers (SAE)

EAL MATRIX

HAZARDS AND OTHER CONDITIONS

NUMARC	NEW WOLF CREEK	OLD WOLF CREEK
HA 2	FR 2 (Alert)	Fire which threatens one Fission Product Barrier (Alert)
HA 3	OH 2 (Alert)	Entry of uncontrollable flammable or toxic gases into a vital area (SAE)
HA 4	LPC/SC 2 (Alert)	Security compromise which threatens one Fission Product Barrier (Alert)
HA 5	LPC/SC 5 (Alert)	Evacuation of Control Room anticipated or required with control of shutdown system established from Aux. Shutdown Panel (Alert)
HA 6	OH 10 (Alert)	Judgemental Authority of the DED/DEM (Alert)
HS 1	LPC/SC 3 (SAE)	Security compromise which threatens two Fission Product Barriers (SAE)
HS 2	LPC/SC 6 (SAE)	Evacuation of Control Room AND control of shutdown systems not established from Aux. Shutdown panel in 15 minutes (SAE)
HS 3	OH 11 (SAE)	Judgemental authority of the DED/DEM (SAE)
HG 1	LPC/SC 4 (GE)	Physical control of the plant is lost (GE)
HG 2	OH 12 (GE)	Judgemental authority of the DED/DEM (GE)

EAL MATRIX

SYSTEM MALFUNCTION

NUMARC	NEW WOLF CREEK	OLD WOLF CREEK
SU 1	LEP/AC 1 (NUE)	Loss of Offsite Power (NUE)
SU 2	ADM 2 (NUE)	Any plant shutdown initiated by Tech Specs (NUE)
SU 3	LEP/AC 6 (NUE)	Unplanned loss of PK02 or more than 75% of Main Control Board annunciators for more than 15 minutes without a major plant transient in progress. (Alert)
SU 4	FEF 5 (NUE)	NONE
SU 5	LRCB 1 (NUE)	NONE
SU 6	LEP/AC 8 (NUE)	Complete loss of all telephones AND the ENS (NUE)
SU 7	LEP/AC 12 (NUE)	NONE
SA 1	LEP/AC 3 (Alert)	NONE
SA 2	SSFM 14 (Alert)	Red or Orange path on subcriticality (Alert)
SA 3	SSFM 8 (Alert)	Mode 5; Mode 6 <23' above the flange. Loss of both trains of RHR > 15 MIN OR Loss of both trains of RHR with RCS Temp exceeding 162°F 14 Mid loop condition with RCS not intact (Alert)
SA 4	LEP/AC 7 (Alert)	Unplaned loss of PK02 or more than 75% of the Main Control Board annunciators with a major plant transient in progress (e.g. any trnasient which causes a major temperature and/or pressure change in the RCS. (SAE)
SA 5	LEP/AC 10 (Alert)	NONE
SS 1	LEP/AC 2 (SAE)	Loss of offsite power AND loss of all onsite AC power (more than 15 min.) (SAE)
SS 2	SSFM 5 (SAE)	Red or Orange path on subcriticality; (used with other loss of barriers for classification)

EAL MATRIX

SYSTEM MALFUNCTION

NUMARC	NEW WOLF CREEK	OLD WOLF CREEK
SS 3	LEP/AC 5 (SAE)	Loss of NK01 and NK04 (more than 15 min.) (SAE)
SS 4	SSFM 7 (SAE)	Mode 4: Loss both trains of RHR and all SG levels <4% narrow range and total available feedwater flow <260,000 lb/hr and unable to dump steam to condenser or release steam with ARVs (SAE)
SS 5	SSFM 9 (SAE)	Loss of both trains of RHR with the core reaching saturation condition wth containment closure not set (SAE) OR Core uncovery with RCS not intact but containment closure maintained (SAE)
SS 6	LEP/AC 11 (SAE)	NONE
SG 1	LEP/AC 4 (GEN)	NONE
SG 2	SSFM 6 (GEN)	NONE

EAL MATRIX

OTHER WOLF CREEK EALs

NUMARC	NEW WOLF CREEK	OLD WOLF CREEK
NONE	NONE	Loss of offsite Power and loss of all onsite AC power (less than 15 min. (Alert)
NONE	NONE	Loss of NK01 and NK04 DC Power (less than 15 min.) (Alert)
NONE	NP3	Safe Shutdown Earthquake limits exceeded; Annunciator Windows 98A "R SPECTRUM SSE EXCEEDED" or 98B SSE in Alarm and Earthquake effects have been see, heard or felt (SAE)
NONE	NONE	Natural phenomonon which threatens two Fission Product Barriers (SAE)
NONE	NONE	Core uncovery with RCS not intact and Containment closure not set (GE)
NONE	NONE	Fire which threatens three Fission Product Barriers (GE)
NONE	NONE	Security compromise which threatens three Fission Product Barriers (GE)
NONE	NONE	Natural phenomenon which threatens three Fission Product Barriers (GE)
Containment Barrier	LRCB 7 SGTF 19 MSLB 13 FEF 5	Containment Hi Rad Monitor confirmed reading 10,000 R/HR (GE)

Attachment 2 to NA 93-0237

WOLF CREEK GENERATING STATION

DRAFT

EMERGENCY CLASSIFICATION

EPP 01-2.1

NUMARC Revision

EMERGENCY PLANNING REVIEW	DATE
PSRC APPROVAL RECOMMENDATION	DATE
PRESIDENT & CHIEF EXECUTIVE OFFICER APPROVAL	DATE
RELEASED	DATE

1.0 PURPOSE

This procedure provides guidance to evaluate plant conditions during an actual or potential emergency situation, assess the Emergency Action Level (EAL) exceeded and classify the emergency according to its severity.

2.0 APPLICABILITY

This procedure applies to the Shift Supervisor as Duty Emergency Director (SS/DED), Duty Emergency Director (DED), and Duty Emergency Manager (DEM). This procedure shall be implemented immediately upon recognition of an emergency or off-normal condition.

3.0 DEFINITIONS

3.1 Alert

Events are in process or have occurred which involve an actual cr potential substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of the Environmental Protection Agency (EPA) Protective Action Guideline exposure levels.

3.2 Critical Safety Function Status Trees (CSFST)

For purposes of emergency classification, the following conditions are conservatively defined:

3.2.1 Red Path

Barriers will be considered breached when the CSFST associated with the barrier is proceeding along a red path.

3.2.2 Orange Path

Barriers will be considered to be subject to a severe challenge and shall be considered breached when listed in an individual initiating condition. Other situations will be handled at the discretion of the DED/DEM for classification purposes.

3.3 Emergency Action Levels (EALs)

Plant or radiological parameters which are the basis for quantifying the initiating condition and classifying the severity of the emergency.

3.4 Emergency Classification

A system used to define the severity of emergencies into one of four categories based upon projected or confirmed initiating conditions/emergency action levels. Classifications listed in order of increasing severity are: Notification of Unusual Event, Alert, Site Area Emergency and General Emergency.

3.5 Emergency Conditions

Situations occurring which cause or may threaten to cause radiological hazards affecting the health and safety of employees or the public, or which may result in damage to property.

3.6 Exclusion Area

That area surrounding the Containment Building to a distance of 1200 meters.

3.7 General Emergency

Events are in process or have occurred which involve actual or imminent substantial core degradation or melting with the potential for loss of containment integrity or the potential loss of reactor coolant system integrity. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

3.8 Imminent

The fact that an event will or may occur very soon or that an unavoidable event will happen even if it is in the future.

3.9 Notification of 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.

3.10 Protected Area

That area around the plant which is encompassed by physical barriers and to which access is controlled for security purposes.

3.11 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.

3.12 Vital Area

Area of the site which contains structures required for safe shutdown of the plant. These structures are:

Reactor Building
Control Building
Fuel Building
Diesel Generator Building
Diesel FOST Access Vaults
Turbine Building (for structural framing integrity only)
Communications Corridor (for structural framing integrity only)
ESW Pump House
ESW Electrical Manholes
ESW Valve House
ESW Access Vaults

4.0 INSTRUCTIONS

4.1 Precautions

4.1.1 Attachment 1, "Initiating Conditions for Emergency Classification", cites specific conditions that denote, beneath thirteen emergency event categories, whether the emergency is to be classified as a Notification of Unusual Event , Alert, Site Area Emergency or General Emergency.

4.1.2 In all cases the decision to declare, upgrade, or proceed to recovery or closeout of an emergency rests with the DED/DEM. The flowcharts are provided as guidance to assist the DED/DEM in making that decision. In many cases a very general statement has been used in a block of the flowchart. This was done intentionally to allow the DED/DEM flexibility to assess any undefinable parameters which may exist at the time.

4.1.3 Plant-specific operator actions required to mitigate the emergency condition are prescribed in the appropriate Emergency Procedures (EMG) or Off-Normal Procedures (OFN) and are independent of any actions required by this procedure.

4.1.4 The DED/DEM should consider the effect that combinations of initiating events have upon the emergency classification level. That is, events if taken individually would constitute a lower emergency classification level. However, collectively they may warrant a higher emergency classification level.

4.2 Use of Attachments

4.2.1 Start in the upper left corner of the chart to be used.

> **CAUTION**: Many charts have blocks that contain multiple initiating conditions separated by "or" and blocks that combine initiating conditions into two distinct sets "or" plus "or" - "and".

4.2.2 Follow the arrows horizontally for yes statements and vertically for no statements.

4.2.3 For purposes of these flowcharts "site" is considered the Exclusion Area Boundary, "plant" is considered the Protected Area.

4.2.4 The designator at the upper right hand corner of the boxes is the reference to the bases document in Attachment 3. Attachment 3, "Explanations/Bases for EALs", gives the reasoning for the box and should be referenced if any clarification is needed.

4.2.5 Full size (approximate 11 x 14 inch), copies of Attachments 1, 2, and 3 are maintained in the Control Room, the Technical Support Center, the Emergency Operations Facility, and the Simulator. Attachment 1 color coding (similar to that used in the EMG's) is as follows:

- No Action In This Category - GREEN
- Notification of Unusual Event - BLUE
- Alert - YELLOW
- Site Area Emergency - ORANGE
- General Emergency - RED

4.2.6 Attachment 1 contained in the procedure is not color coded.

4.3 Initial Actions

> **CAUTION**: Outage/shutdown conditions should be given special consideration as they are likely to create abnormalities such as the loss of RCS pressure boundary (refueling, mid-loop operations, equipment hatch open, etc.). This type of boundary violation combined with a plant transient (loss of AC power, etc.) may create a worse situation than would be expected if the Unit was in power operations.

4.3.1 Upon recognition that an abnormal or emergency condition exists, the on-duty Shift Supervisor shall be immediately notified. Recognition of the event can occur as a result of either Control Room personnel or other plant personnel observing the abnormal or emergency condition.

4.3.2 Control Room personnel shall continue to monitor the appropriate plant parameters and instrument readings or any other symptoms which would be indicative of further systems degradation.

4.3.3 Operators shall refer to the appropriate EMGs and take any actions called for based upon the indicated symptoms.

4.3.4 The on-duty Shift Supervisor shall report to the Control Room, if possible, and evaluate the event to determine the need for classifying the emergency condition into one of the four emergency classification levels.

4.3.5 The on-duty Shift Supervisor shall refer to Attachment 1 of this procedure to ascertain whether or not the event fits the general description for any of the initiating conditions listed. If the event does not fit any of these general descriptions, the on-duty Shift Supervisor should evaluate the implications of the event and, if appropriate, classify the emergency condition based upon professional judgment. If no classification is warranted, no further action is required except to continue monitoring the event.

4.3.6 If the on-duty Shift Supervisor determines that the event does fit one or more of the emergency classifications listed in Attachment 1, the on-duty Shift Supervisor shall assume the role of DED as prescribed in EPP 01-1.0, "Control Room Organization".

4.3.7 The on-duty Shift Supervisor shall declare the appropriate emergency classification and implement the necessary actions using the appropriate checklist referenced in EPP 01-1.0 "Control Room Organization".

4.4 Subsequent Actions

The Shift Supervisor and DED shall continually monitor plant conditions to determine whether a change in emergency classification is warranted.

Whenever the Shift Supervisor or DED identifies a change in an original (initiating) condition, Attachment 1 shall be referenced to determine whether to escalate the emergency classification or proceed to recovery/closeout of the event in accordance with EPP 01-12.1 "Recovery Operations".

5.0 REFERENCES

5.1 PIR 92-0604

5.2 PDR 91-0038

5.3 PIR 92-0731

5.4 Wolf Creek Generating Station Radiological Emergency Response Plan.

5.5 EPP 01-1.0, "Control Room Organization"

5.6 EPP 01-10.1, "Protective Action Recommendations'

5.7 EPP 01-12.1, "Recovery Operations"

5.8 Emergency Procedures (EMG)

5.9 Off-Normal Procedures (OFN)

5.10 "Technical Specifications for Wolf Creek Unit 1", Docket No. 50-482, NUREG 1136

5.11 Reg. Guide 1.101, "Emergency Planning and Preparedness for Nuclear Power Reactors.

5.12 WCGS Off-Site Dose Calculation Manual

5.13 WCGS Updated Safety Analysis Report

6.0 RECORDS

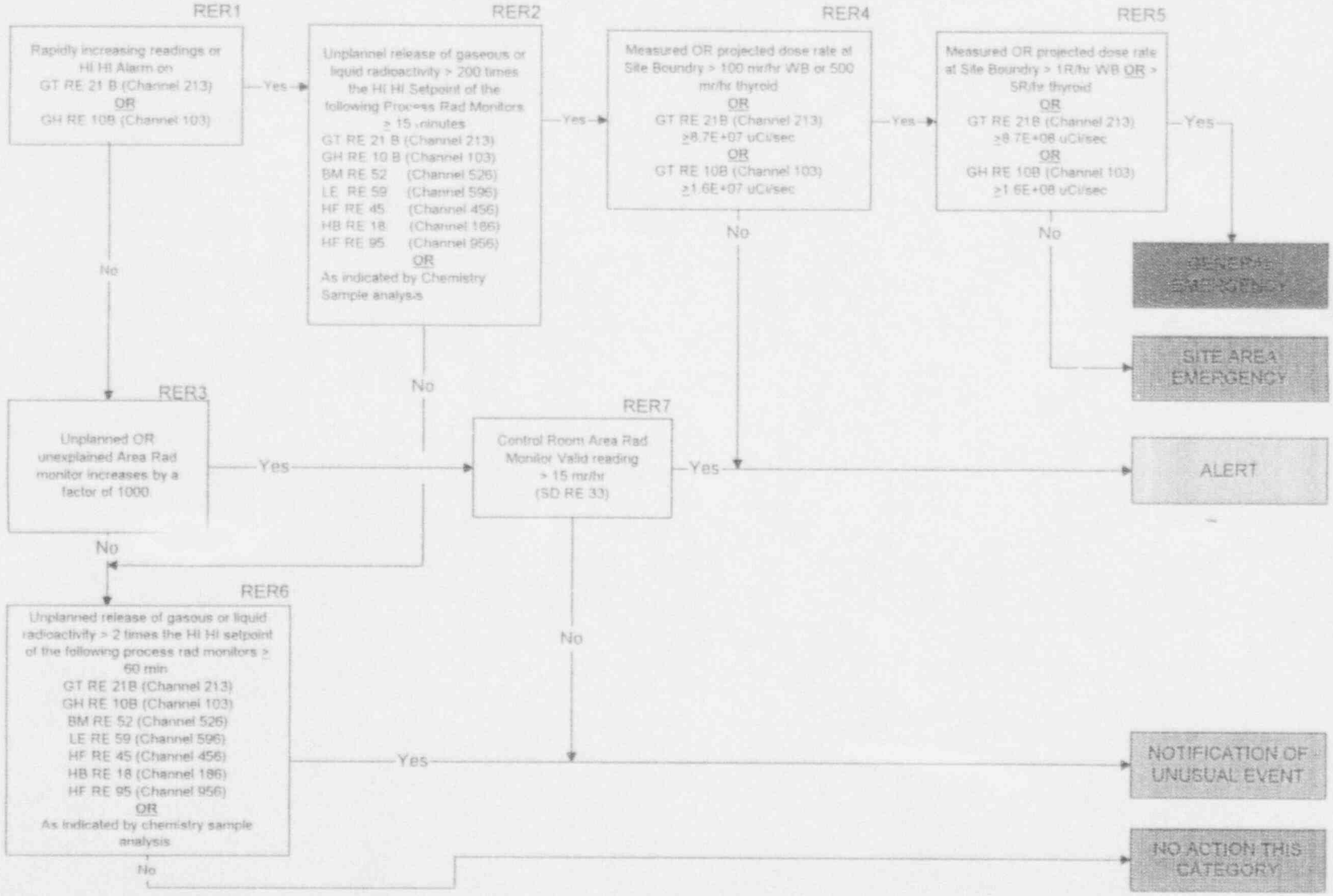
6.1 NONE

7.0 ATTACHMENTS

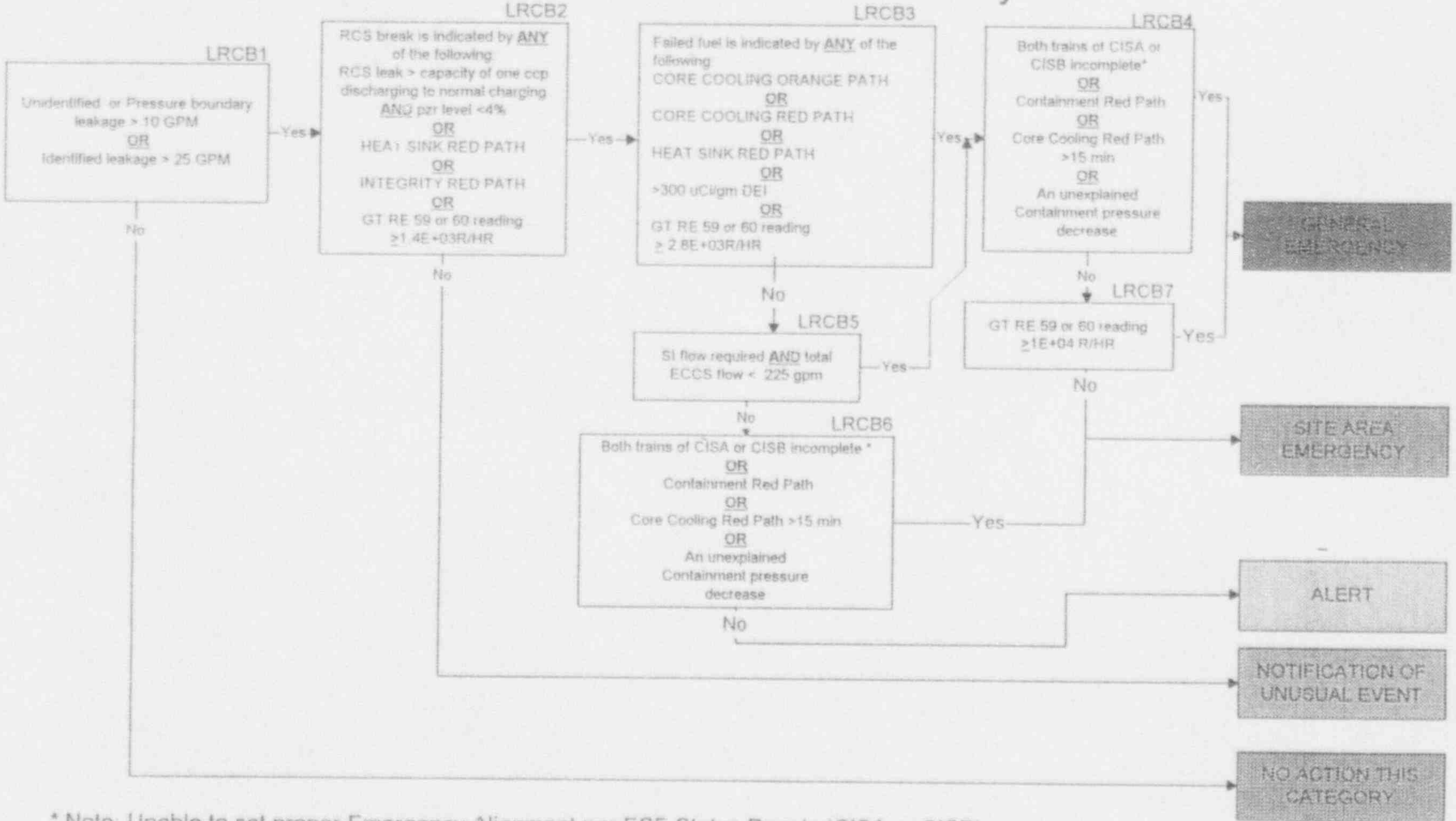
7.0 ATTACHMENTS

7.1	Attachment 1,	"Initiating Conditions for Emergency Classifications"
7.2	Attachment 2,	"Indications of a Loss of Function"
7.3	Attachment 3,	"Explanations/Bases for EALs"

Radioactive Effluent Release



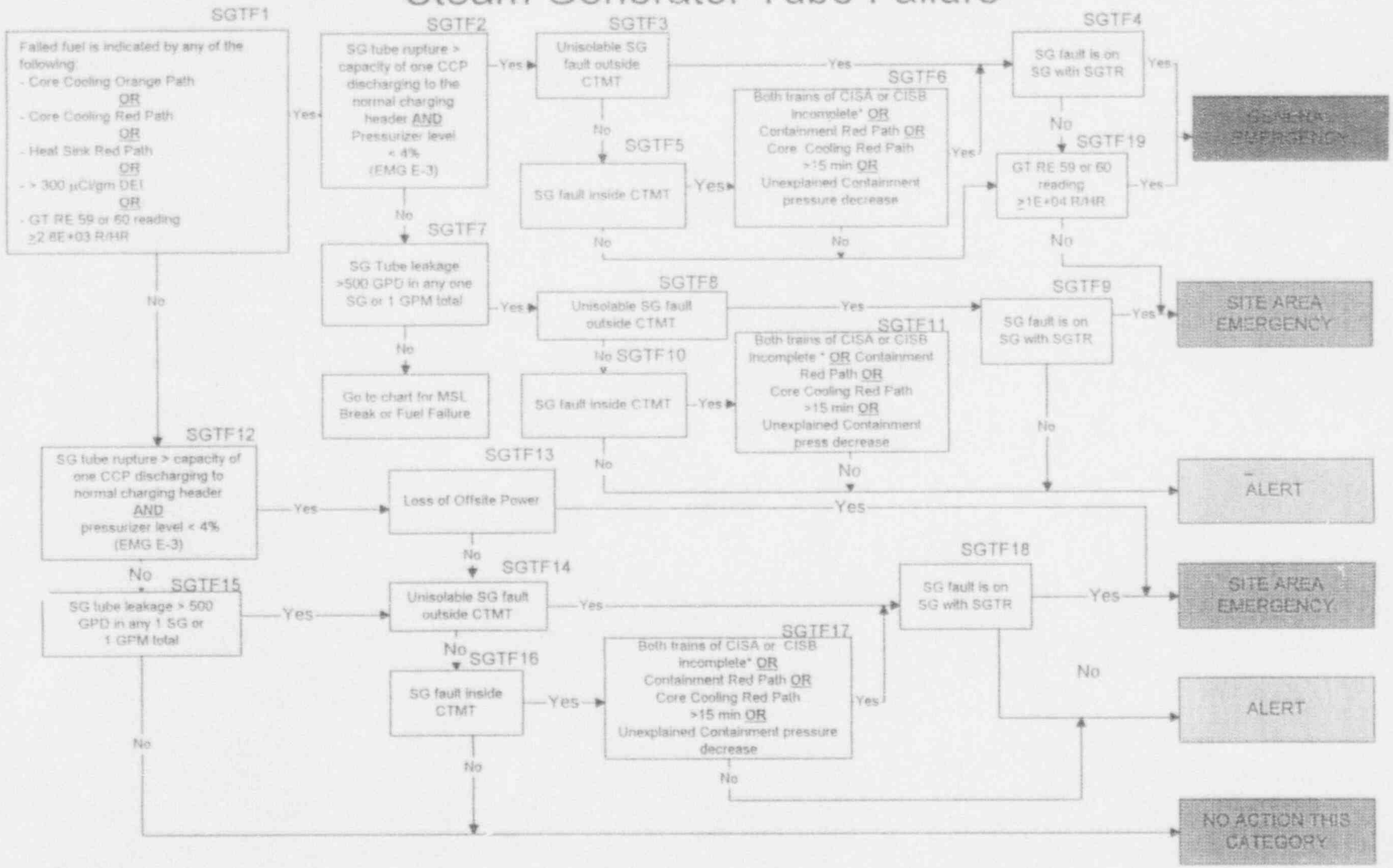
Loss of Reactor Coolant Boundary



* Note: Unable to set proper Emergency Alignment per ESF Status Panels (CISA or CISB)

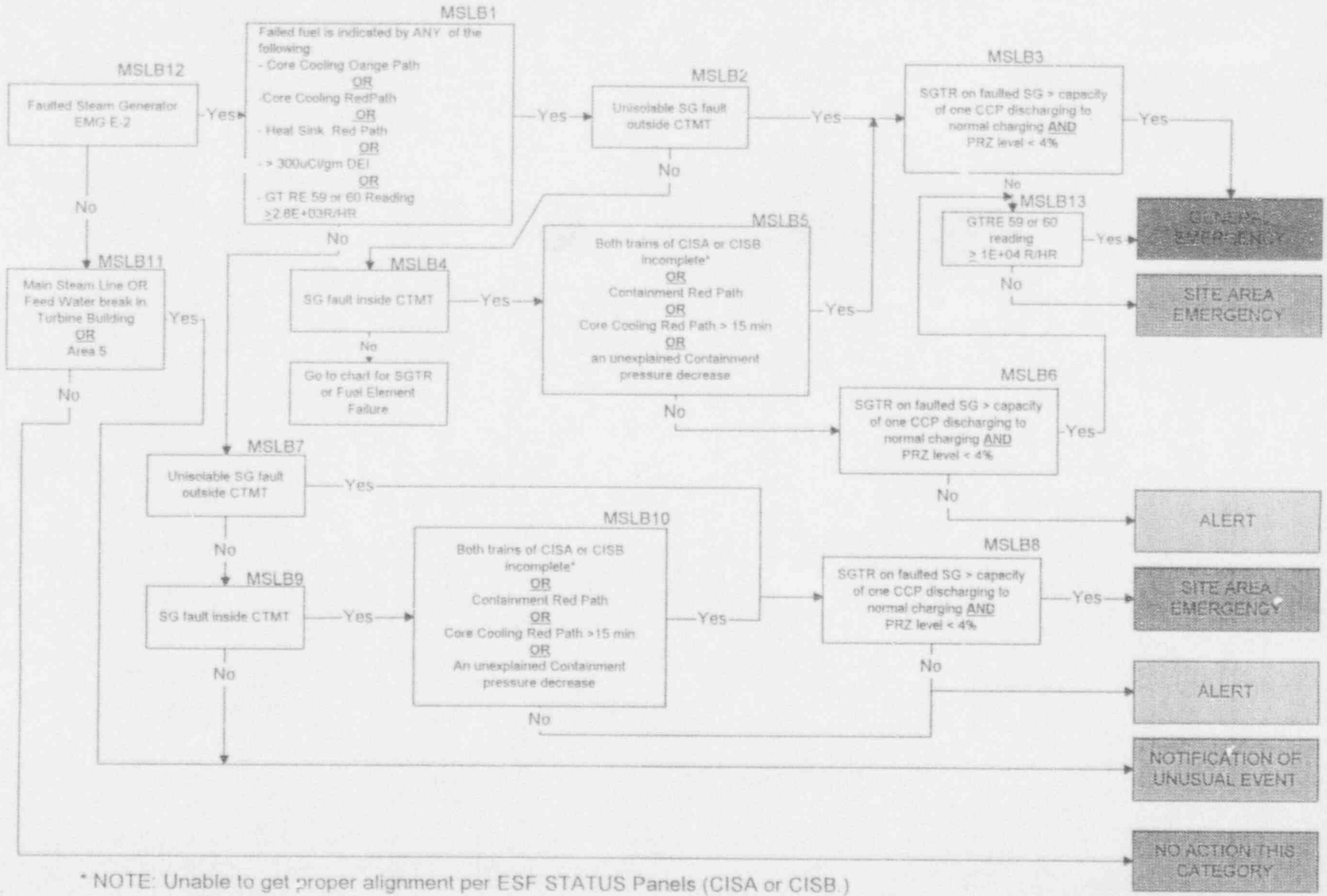
NOTE: This chart shall not be used if Steam Generator Tube Rupture Failure is the only event. Go to "Steam Generator Tube Failure."

Steam Generator Tube Failure



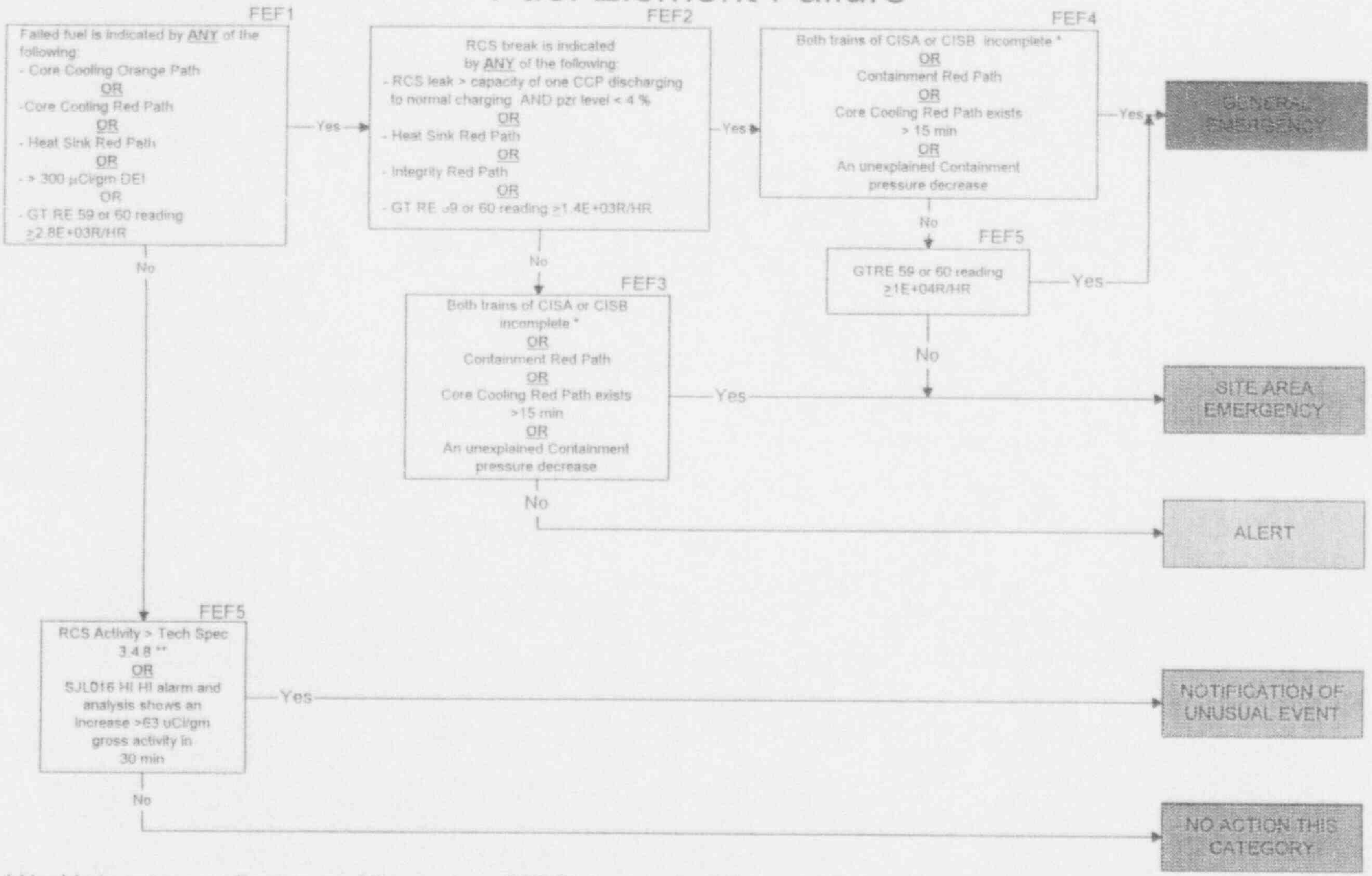
* Note: Unable to set proper Emergency Alignment per ESF Status Panels (CISA or CISB)

Main Steam Line Break



* NOTE: Unable to get proper alignment per ESF STATUS Panels (CISA or CISB.)

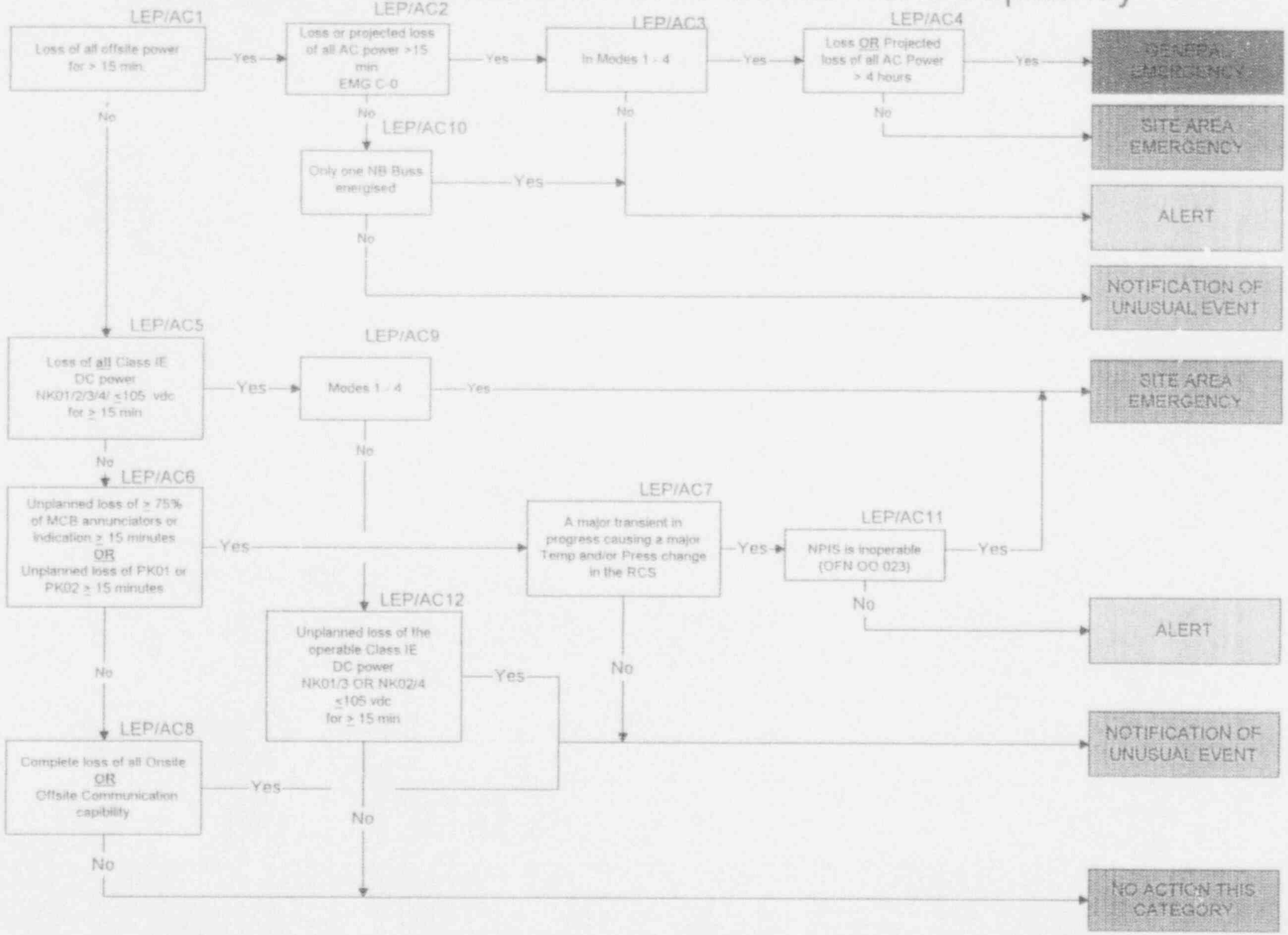
Fuel Element Failure



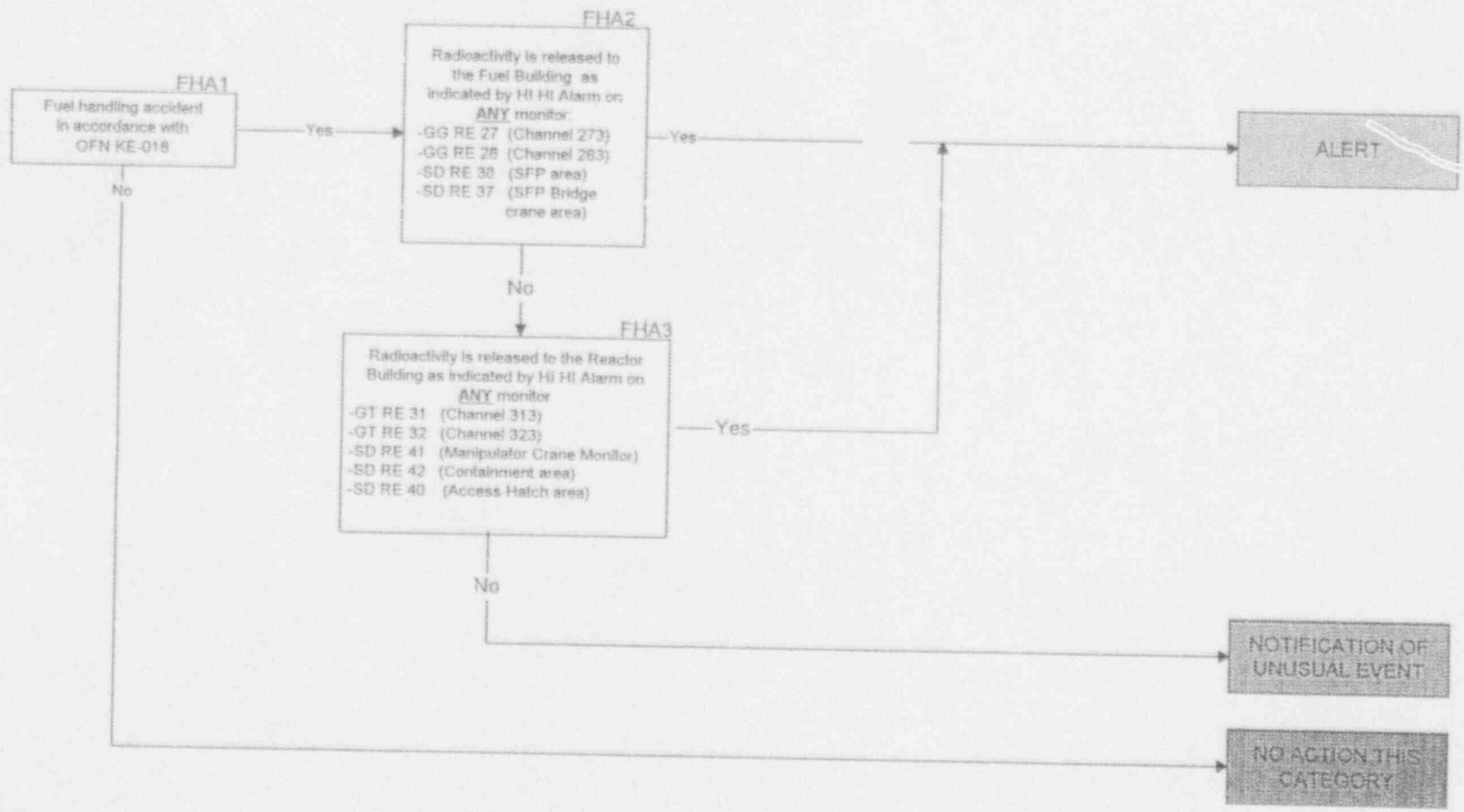
* Unable to get proper Emergency Alignment per ESF Status panels (CISA or CISB sections)

** When DEI exceeds limit for >48 hr during one continous interval OR exceeds the limit line on Figure 3.4-1 of TS 3.4.8 OR gross activity >100/ E BAR

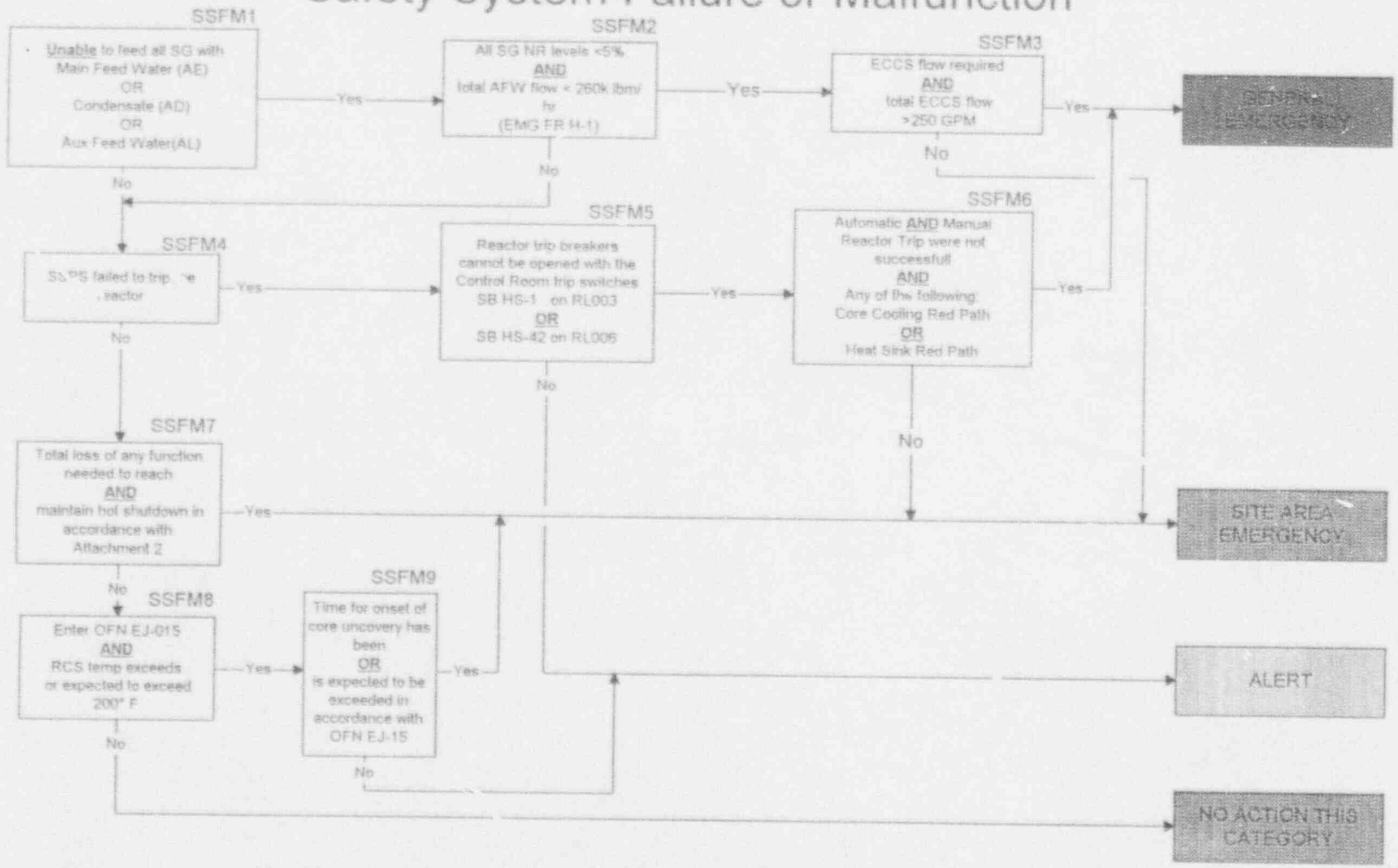
Loss of Electrical Power / Assessment Capabiltiy



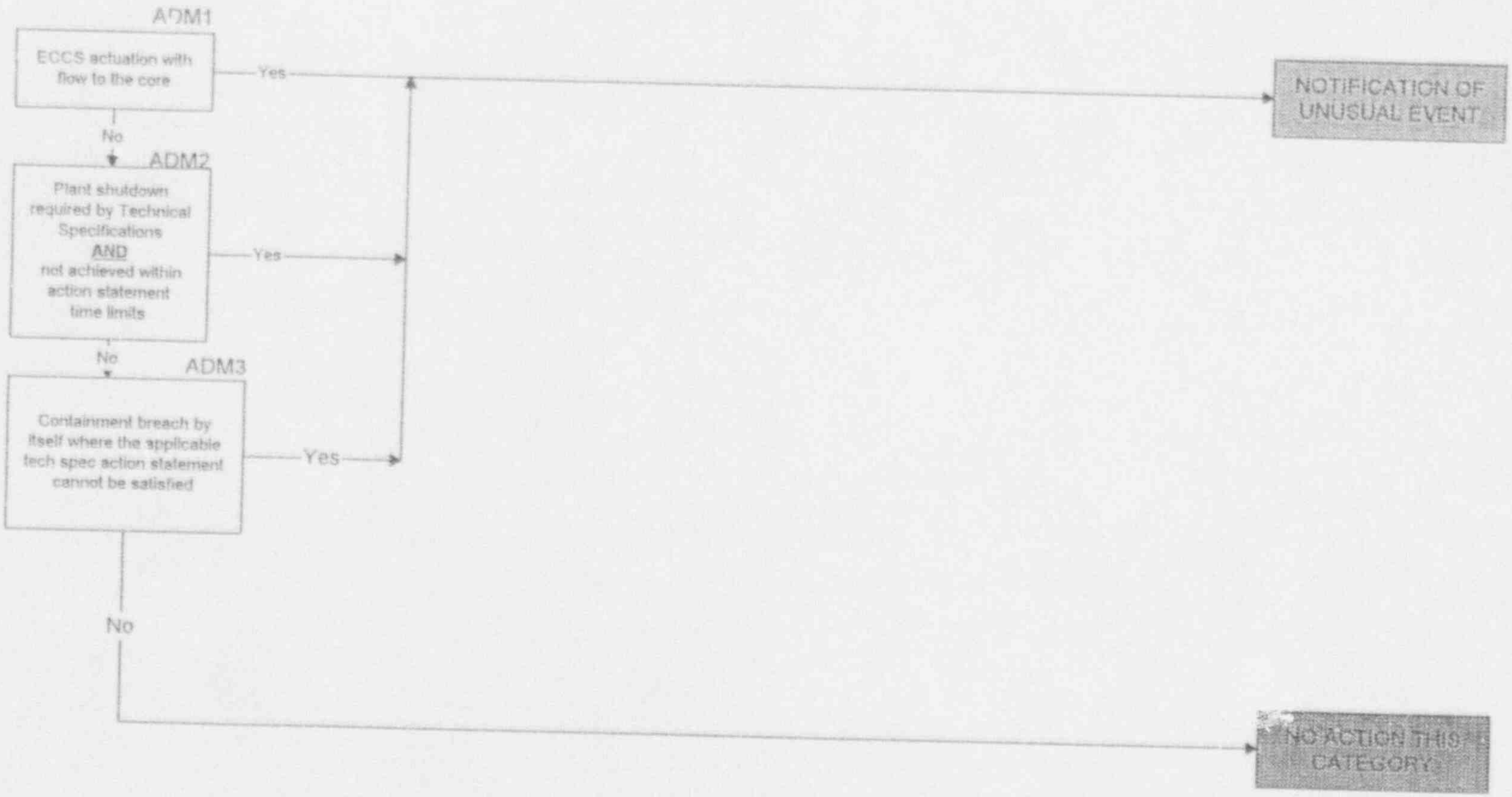
Fuel Handling Accident



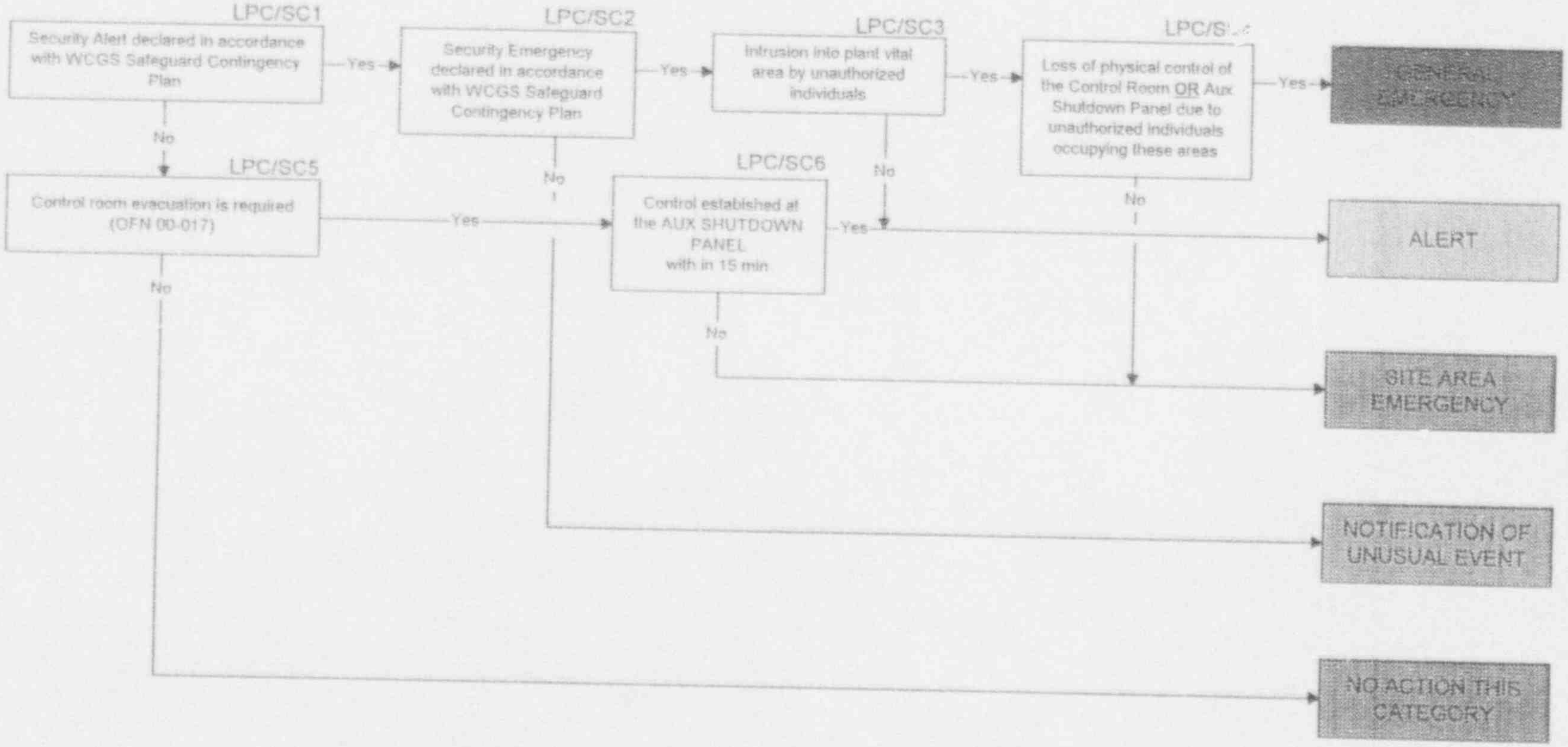
Safety System Failure or Malfunction



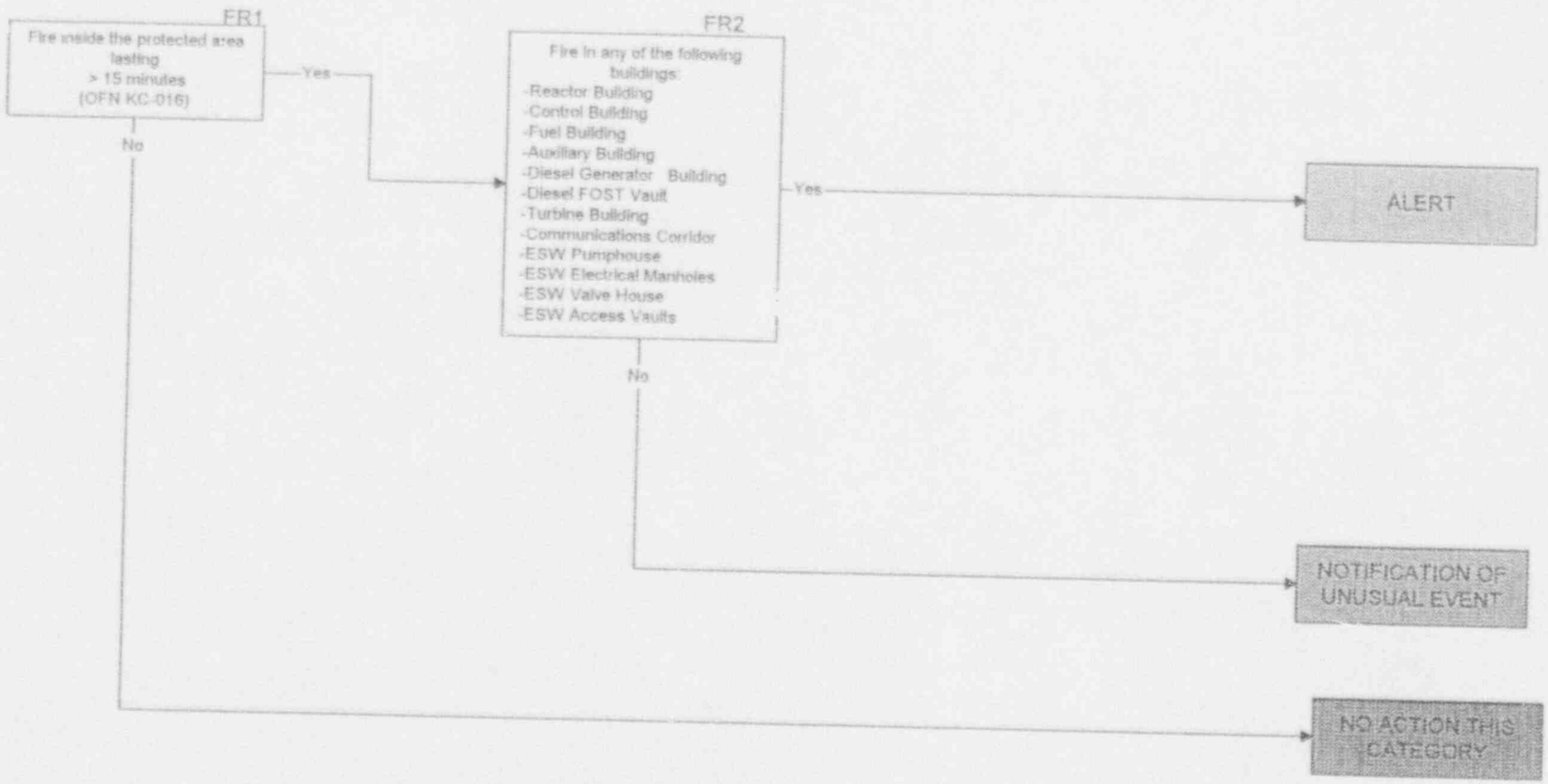
Administrative



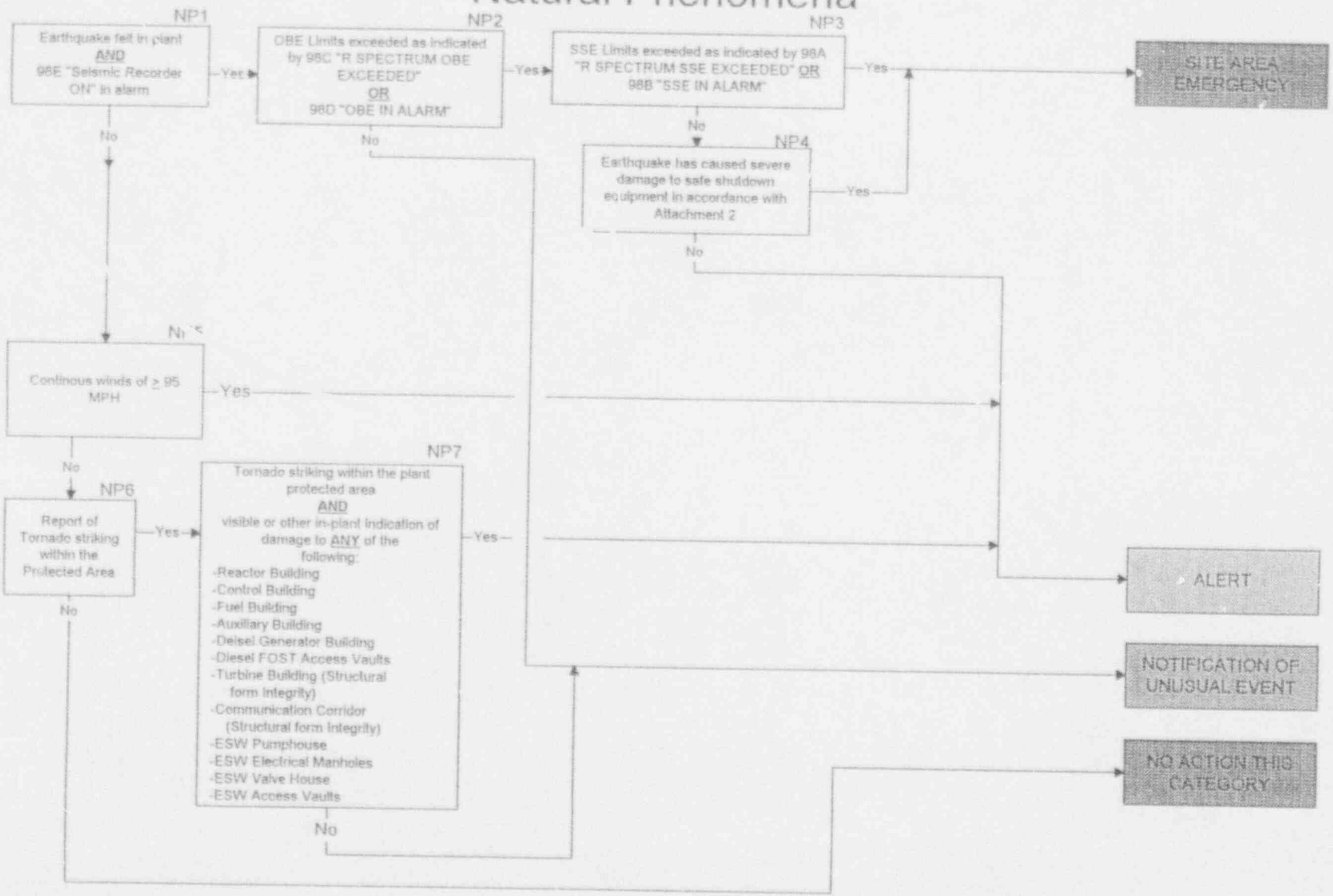
Loss of Plant Control / Security Compromise



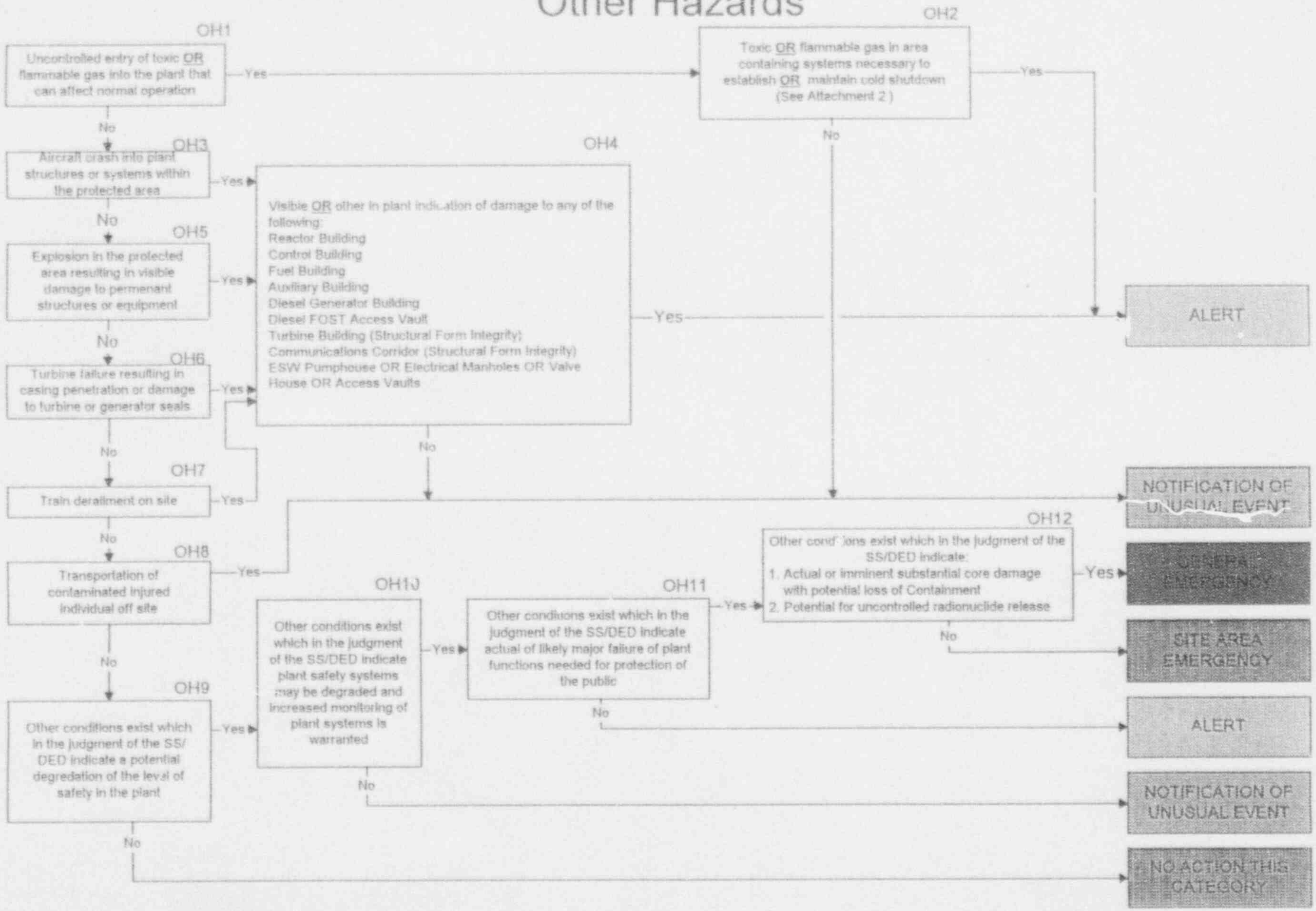
Fire


FR1
Fire inside the protected area lasting > 15 minutes (OFN KC-016)
Yes
No
FR2
Fire in any of the following buildings:
-Reactor Building
-Control Building
-Fuel Building
-Auxiliary Building
-Diesel Generator Building
-Diesel FOST Vault
-Turbine Building
-Communications Corridor
-ESW Pumphouse
-ESW Electrical Manholes
-ESW Valve House
-ESW Access Vaults
Yes
No
ALERT
NOTIFICATION OF UNUSUAL EVENT
NO ACTION THIS CATEGORY

Natural Phenomena



Other Hazards



ATTACHMENT 2
INDICATION OF A LOSS OF FUNCTION
(Page 1 of 1)

A. Indication of a Complete Loss of Function needed to achieve OR maintain Hot Shutdown (Mode 4)

1. **ALL** of the following
   a. Failure to bring the Reactor subcritical with the Control Rods fully inserted.
   b. Complete loss of all Boron Injection Flow Paths.

OR

2. **ALL** of the following
   a. All SG levels <10% wide range
   b. All Steam Dumps to the condenser (AB UV 34,41,45) will not operate.
   c. All SG ARVs will not operate (AB PIC 1A, 2A, 3A, 4A).
   d. Complete loss of both RHR trains. (A complete loss of ESW or CCW constitutes a complete loss of RHR)

OR

3. The Ultimate Heat Sink (UHS) is inoperable because of **ANY** of the following.
   a. UHS level <1070 ft.
   b. UHS temperature >90 degrees

B. Indication of a loss of any function needed to maintain Cold Shutdown (Mode 5)

1. Complete loss of RHR (A complete loss of ESW or CCW constitutes a complete loss of RHR).

AND

2. **Any** of the following
   a. >200 degrees on any valid incore thermocouple.
   b. Uncontrolled temperature rise with no action available that will prevent approaching 200 degrees on any valid incore thermocouple.

EXPLANATIONS/BASES
CHART - RADIOACTIVE EFFLUENT RELEASE

RER 1. - MODES; ALL

This box is used to indicate unexpected vent stack release rates or releases greater than ODCM allowable values.

RER 2. - MODES: ALL

Valid means that a radiation monitor reading has been confirmed by the operators to be correct.

This event escalates from the Unusual Event by escalating the magnitude of the release by a factor of 100. Prorating the 500 mR/year criterion for both time (8766 hr/year), and the 200 multiplier of the associated site boundary dose rate would be 10 mR/hr. The required release duration was reduced to 15 minutes in recognition of the increased severity.

Wolf Creek has eliminated effluent technical specifications as provided in NRC Generic Letter 89-01, the corresponding maximum limit from the site's Offsite Dose Calculation Manual, multiplied by 200, was used as the numeric basis of this EAL.

Monitor indications should be calculated on the basis of the methodology of the site Offsite Dose Calculation Manual (ODCM), or other site procedures that are used to demonstrate compliance with 10 CFR 20 and/or 10 CFR 50 Appendix I requirements -- adjusted upwards by a factor of 200. Annual average meteorology should be used where allowed.

RER 3. - MODES: ALL

Addresses unplanned increases in in-plant radiation levels that represent a degradation in the control of radioactive material, and represent a potential degradation in the level of safety of the plant.

EXPLANATIONS/BASES
CHART - RADIOACTIVE EFFLUENT RELEASE

RER 4.- MODES; ALL

Valid means that a radiation monitor reading has been confirmed by the operators to be correct.

The 100 mR 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 deemed that exposures less than this limit are not consistent with the Site Area Emergency class description. The 500 mR integrated child thyroid dose was established in consideration of the 1:5 ratio of the EPA Protective Action Guidelines for whole body and thyroid.

Integrated doses are generally not monitored in real-time. In establishing the emergency action levels, a duration of one hour was assumed, and that the EAL is based on a site boundary dose of 100 mR/hour whole body or 500 mR/hour child thyroid, whichever is more limiting.

Unit Vent and Radwaste Vent numbers were obtained using the WCGS Emergency Dose Computer Program (EDCP). [These numbers may change after Jan. 1, 1994 due to implementation of the new EPA 400 and 10 CFR 20 guidelines]

RER 5 - MODES: ALL

Valid means that a radiation monitor reading has been confirmed by the operators to be correct.

The 1000 mR whole body and the 5000 mR child thyroid integrated dose are based on the EPA protective action guidance which indicates that public protective actions are indicated if the dose exceeds 1 Rem whole body or 5 Rem child thyroid. This is consistent with the emergency class description for a General Emergency. This level 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. Actual meteorology (including forecasts) should be used whenever possible.

Integrated doses are generally not monitored in real-time. In establishing the emergency action levels, a duration of one hour was assumed, and that the EAL is based on site boundary doses for either whole body or child thyroid, whichever is more limiting.

Unit Vent and Radwaste Vent numbers were obtained using the WCGS Emergency Dose Computer Program (EDCP). [These numbers may change after Jan. 1, 1994 due to implementation of the new EPA 400 and 10 CFR 20 guidelines]

EXPLANATIONS/BASES
CHART - RADIOACTIVE EFFLUENT RELEASE

RER 6. - MODES: ALL

The term "Unplanned", as used in this context, includes any release for which a radioactive discharge permit was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable permit.

Valid means that a radiation monitor reading has been confirmed by the operators to be correct.

Unplanned releases in excess of two times the ODCM specification 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; it is the degradation in plant control implied by the fact that the release was not isolated within 60 minutes. Therefore, it is not intended that the release be averaged over 60 minutes. For example, a release of 4 times T/S for 30 minutes does not exceed this initiating condition. Further, the Duty Emergency Director/Duty Emergency Manager should not wait until 60 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 60 minutes.

RER 7. - MODES: ALL

Valid means that a radiation monitor reading has been confirmed by the operators to be correct.

This IC addresses increased radiation levels that impede necessary access to WCGS, 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 IC. The Duty Emergency Director/Duty Emergency Manager must consider the source or cause of the increased radiation levels and determine if any other IC may be involved. For example, a dose rate of 15 mR/hr in the Control Room may be a problem in itself. However, the increase may also be indicative of high dose rates in the Containment Building due to a LOCA. In this latter case, an SAE or GE may be indicated by the fission product barrier matrix ICs.

Areas requiring continuous occupancy includes the Control Room and the Central Alarm Station. Section III.D.3 of NUREG-0737, "Clarification of TMI Action Plan Requirements", provides that the 15 mR/hr value can be averaged over 30 days, the value is used here without averaging, as a 30 day duration implies an event potentially more significant than an Alert.

EXPLANATIONS/BASES
CHART - LOSS OF REACTOR COOLANT BOUNDARY

NOTE: The note is provided to direct the user to the Steam Generator Tube Failure chart vice Loss of Reactor Coolant Boundary when Reactor Coolant leakage is via the Steam Generator tubes only.

LRCB 1.- MODES: 1 THROUGH 4 This IC is included as an Unusual Event because it may be a precursor of more serious conditions and , as result, is considered to be a potential degradation of the level of safety of the plant. The 10 gpm value for the unidentified and pressure boundary leakage was selected as it is observable with normal Control Room indications. Lesser values must generally be determined through time-consuming surveillance tests. This EAL for identified leakage in comparison to unidentified or pressure boundary leakage.

LRCB 2.- MODES: 1 THROUGH 4 1. Critical Safety Function Status: This EAL is for using Critical Safety Function Status Tree (CSFST) monitoring and functional recovery procedures. RED path indicates an extreme challenge to the safety function derived from appropriate instrument readings, and these CSFs indicate a potential loss of RCS barrier.
2. RCS Leak Rate: The "Potential Loss" EAL is based on the inability to maintain normal liquid inventory within the Reactor Coolant System (RCS) by normal operation of the Chemical and Volume Control System which is considered as one centrifugal charging pump discharging to the charging header. Pressurizer level was added because this is Safety Injection Criteria per WCGS procedure OFN BB-007 "SG/RCS Leakage High".
3. Containment Radiation Monitoring: The 1.4 E +3 R/hr. reading is a value which indicates the release of reactor coolant to the containment. The reading assumes the instantaneous release and dispersal of the reactor coolant concentrations (i.e., within T/S) into the containment atmosphere. Per Tech Spec 3.4.8 we can operate up to 48 hours with DEI of 63 μc/gm. This is equal to 1% failed fuel per WCGS USAR. The readings were obtained from WCGS EPP 01-2.4 "Core Damage Assessment Methodology", Attachment 1.0 for failed fuel.

EXPLANATIONS/BASES
CHART - LOSS OF REACTOR COOLANT BOUNDARY

LRCB 3.- MODES: 1 THROUGH 4 1. Critical Safety Function Status : This EAL is for using Critical Safety Function Status Tree (CSFST) monitoring and functional recovery proced-. RED path indicates an extreme challenge to the safety function. ORANGE path indicates a severe challenge to the safety function.

Core Cooling - ORANGE indicates subcooling has been lost and that some clad damage may occur. Heat Sink - RED indicates the ultimate heat sink function is under extreme challenge and thus these two items indicate potential loss of the Fuel Clad Barrier.

Core Cooling - RED indicates significant superheating and core uncovery and is considered to indicate loss of the Fuel Clad Barrier.

2. Primary Coolant Activity Level: The 300 μCi/cc DEI assessment by the NUMARC EAL Task Force indicates that this amount of coolant activity is well above that expected for iodine spikes and corresponds to about 2% to 5% fuel clad damage. This amount of clad damage indicates significant clad heating and thus the Fuel Clad Barrier is considered lost.

3. Containment Radiation Monitoring: This reading is a value which indicates the release of reactor coolant, with elevated activity indicative of fuel damage, into the containment. The reading should be calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 300 μCi/gpm dose equivalent I-131 into the containment atmosphere. Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations (including iodine spiking) allowed within technical specifications and are therefore indicative of fuel damage (approximately 2-5% clad failure depending on core inventory and RCS volume). To be conservative, 2 % clad failure and 10 hours after shutdown were selected from WCGS EPP 01-2.4, "Core Damage Assessment Methodology", Attachment 1.0.

EXPLANATIONS/BASES
CHART - LOSS OF REACTOR COOLANT BOUNDARY

LRCB 4 - MODES: 1 THROUGH 4 Containment Isolation Valve Status After Containment Isolation : This EAL is for using Critical Function Status Tree (CSFST) monitoring and functional recovery procedures. RED path indicates an extreme challenge to the safety function derived from appropriate instrument readings and/or sampling results, and thus represents a potential loss of containment. Conditions leading to a containment RED path result from RCS barrier and /or Fuel Clad Barrier Loss. Thus, this EAL is primarily a discriminator between Site Area Emergency and General Emergency representing a potential loss of the third barrier.
In this EAL, the function restoration procedures are those emergency operating procedures that address the recovery of the core cooling critical safety functions. The procedure is considered effective if the temperature is decreasing or if the vessel level is increasing.
The conditions in this potential loss EAL represent imminent melt sequence which, if not corrected, could lead to vessel failure and an increased potential for containment failure. In conjunction with the core exit thermocouple EALs in the Fuel and RCS barrier columns, this EAL would result in the declaration of a General Emergency -- loss of two barriers and the potential loss of a third. If the function restoration procedures are ineffective, there is no "success" path.
Severe accident analyses (e.g., NUREG-1150) have concluded that function restoration procedures can arrest core degradation within the reactor vessel in a significant fraction of the core damage scenarios, and that the likelihood of containment failure is very small in these events. Given this, it is appropriate to provide a reasonable period to allow function restoration procedures to arrest the core melt sequence. Whether or not the procedures will be effective should be apparent within 15 minutes. The Duty Emergency Director/Duty Emergency Manager should make the declaration as soon as it is determined that the procedures have been, or will be ineffective. The reactor vessel level chosen should be consistent with the emergency response guides applicable to the facility.

LRCB 5 - MODES: 1 THROUGH 4 This IC used to determine if any ECCS System is capable of delivering sufficient volume of water to the core. 225 gpm was chosen because it is conservatively larger then Tech Spec delta P requirement of ≈210 gpm at 2400 PSID.

LRCB 6 - MODES: 1 THROUGH 4 See **LRCB 4.**

LRCB 7 - MODES: 1 THROUGH 4 This reading is a value which indicates significant fuel damage well in excess of the EALs associated with both loss of Fuel Clad and loss of RCS Barriers. As stated in Section 3.8 of Reference 5.11, a major release of radioactivity requiring offsite protective actic , from core damage 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. NUREG-1228, "Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents," indicates that such conditions do not exist when the amount of clad damage is less than 20%.

EXPLANATIONS/BASES
CHART - STEAM GENERATOR TUBE RUPTURE

SGTF 1 - MODES: 1 THROUGH 4 1. This EAL is for using Critical Safety Function Status Tree (CSFST) monitoring and functional recovery procedures. RED path indicates an extreme challenge to the safety function. ORANGE path indicates a severe challenge to the safety function.
Core Cooling- ORANGE indicates subcooling has been lost and that some clad damage may occur. Heat Sink- RED indicates the ultimate heat sink function is under extreme challenge and thus these two items indicate potential loss of the Fuel Clad Barrier.
Core Cooling- RED indicates significant superheating and core uncovery and is considered to indicate loss of the Fuel Clad Barrier.
2. Primary Coolant Activity Level: The 300 µCi/cc DEI assessment by the NUMARK EAL Task Force indicates that this amount of coolant activity is well above that expected for iodine spikes and corresponds to about 2% to 5% fuel clad damage. This amount of clad damage indicates significant clad heating and thus the Fuel Clad Barrier is considered lost.
3. Containment Radiation Monitoring: This reading is a value which indicates the release of reactor coolant, with elevated activity indicative of fuel damage, into the containment. The reading should be calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 300 µCi/gpm dose equivalent I-131 into the containment atmosphere. Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations (including iodine spiking) allowed within technical specifications and are therefore indicative of fuel damage (approximately 2-5% clad failure depending on core inventory and RCS volume).

SGTF 2.- MODES: 1 THROUGH 4 RCS Leak Rate EAL is based on the inability to maintain normal liquid inventory within the Reactor Coolant System (RCS) by normal operation of the Chemical and Volume Control System which is considered as one centrifugal charging pump discharging to the charging header. Pressurizer level was added because this is Safety Injection Criteria per WCGS procedure OFN BB-007 "SG/RCS Leakage High".

SGTF 3. - MODES: 1 THROUGH 4 A check for S/G fault is made to determine if the next fission product boundary is under challenge or lost. The release path looked for is either a faulted, ruptured S/G or a faulted S/G to a challenged Containment. Unisolable means that the steam release from the faulted S/G cannot be stopped until the S/G has blown dry.

SGTF 4. - MODES: 1 THROUGH 4 Once a faulted S/G has been determined, a release path via a faulted, ruptured S/G is checked.

SGTF 5. - MODES: 1 THROUGH 4 This box checks for an unisolable secondary side steam release to the Containment atmosphere.

SGTF 6. - MODES: 1 THROUGH 4 See LRCB 4

EXPLANATIONS/BASES
CHART - STEAM GENERATOR TUBE RUPTURE

SGTF 7. - **MODES: 1 THROUGH 4**
SG Tube leakage in excess of Tech Spec limits (Tech Spec 3.4.6.2)

SGTF 8. - **MODES: 1 THROUGH 4** See SGTF 3

SGTF 9. - **MODES: 1 THROUGH 4** See SGTF 4

SGTF 10. - **MODES: 1 THROUGH 4** See SGTF 5

SGTF 11. - **MODES: 1 THROUGH 4** See LRCB 4

SGTF 12. - **MODES: 1 THROUGH 4** See SGTF 2

SGTF 13. - **MODES: 1 THROUGH 4**
With a S/G tube rupture of several hundred gpm in progress and no offsite power available, a single failure of a DG would present severe challenges to S/G and fuel integrity. Limited RCS make-up capability and difficulty of RCS pressure and temperature control could lead to core uncovery and/or S/G overfill.

EXPLANATIONS/BASES
CHART - STEAM GENERATOR TUBE RUPTURE

SGTF 14. - **MODES: 1 THROUGH 4** See SGTF 3

SGTF 15. - **MODES: 1 THROUGH 4** See SGTF 7

SGTF 16. - **MODES: 1 THROUGH 4** See SGTF 5

SGTF 17. - **MODES: 1 THROUGH 4** See LRCB 4

SGTF 18. - **MODES: 1 THROUGH 4** See SGTF 4

SGTF 19. - **MODES: 1 THROUGH 4** See LRCB 7

EXPLANATIONS/BASES
CHART - MAIN STEAM LINE BREAK

MSLB 1. - MODES: 1 THROUGH 4 1. Critical Safety Function Status: This EAL is for using Critical Safety Function Status Tree (CSFST) monitoring and functional recovery procedures. RED path indicates an extreme challenge to the safety function. ORANGE path indicates a severe challenge to the safety function.

Core Cooling - ORANGE indicates subcooling has been lost and that some clad damage may occur. Heat Sink - RED indicates the ultimate heat sink function is under extreme challenge and thus these two items indicate potential loss of the Fuel Clad Barrier.

Core Cooling - RED indicates significant superheating and core uncovery and is considered to indicate loss of the Fuel Clad Barrier.

2. Primary Coolant Activity Level: The 300 uCi/cc DEI assessment by the NUMARC EAL Task Force indicates that this amount of coolant activity is well above that expected for iodine spikes and corresponds to about 2% to 5% fuel clad damage. This amount of clad damage indicates significant clad heating and thus the Fuel Clad Barrier is considered lost.

3. Containment Radiation Monitoring: This reading is a value which indicates the release of reactor coolant, with elevated activity indicative of fuel damage, into the Containment. The reading should be calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 300 uCi/gm dose equivalent I-131 into the Containment atmosphere. Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations (including iodine spiking) allowed within technical specifications and are therefore indicative of fuel damage (approximately 2 - 5% clad failure depending on core inventory and RCS volume).

MSLB 2. - MODES: 1 THROUGH 4 A check for S/G fault is made to determine if the next fission product boundary is under challenge or lost. The release path looked for is either a faulted, ruptured S/G or a faulted S/G to a challenged Containment. Unisolable means that the steam release from the faulted S/G cannot be stopped until the S/G has blown dry.

MSLB 3. - MODES: 1 THROUGH 4 Once a faulted S/G has been determined, a release path via a faulted, ruptured S/G is checked. Leakage greater than charging capacity of one CCP to a S/G constitutes failure of the FCS fission product boundary.

MSLB 4. - MODES: 1 THROUGH 4 This box checks for an unisolable secondary side steam release to the Containment atmosphere.

EXPLANATIONS/BASES
CHART - MAIN STEAM LINE BREAK

MSLB 5. - MODES: 1 THROUGH 4 See LRCB 4

MSLB 6. - MODES: 1 THROUGH 4 See MSLB 3

MSLB 7. - MODES: 1 THROUGH 4 See MSLB 2

MSLB 8. - MODES: 1 THROUGH 4 See MSLB 3

MSLB 9. - MODES: 1 THROUGH 4 See MSLB 4

MSLB 10. - MODES: 1 THROUGH 4 See LRCB 4

MSLB 11. - MODES: 1 THROUGH 4 Rapid depressurization of the secondary due to a MSL break which is isolable from the S/G's. (i.e. downstream of the MSIV's) A main steam line or feed water break in the Turbine Building or Area 5 could cause a potential degradation of the level of safety of the plant.

MSLB 12. - MODES: 1 THROUGH 4 This procedure provides actions to identify and isolate a faulted steam generator. A Main Steam Line break inside or outside containment could cause a potential degradation of the level of safety of the plant.

MSLB 13. - MODES: 1 THROUGH 4 See LRCB 7.

EXPLANATIONS/BASES
CHART - FUEL ELEMENT FAILURE

FEF 1. - MODES: ALL 1. Critical Safety Function Status: This EAL is for using Critical Safety Function Status Tree (CSFST) monitoring and functional recovery procedures. RED path indicates an extreme challenge to the safety function. ORANGE path indicates a severe challenge to the safety function.

Core Cooling - ORANGE indicates subcooling has been lost and that some clad damage may occur. Heat Sink - RED indicates the ultimate heat sink function is under extreme challenge and thus these two items indicate potential loss of the Fuel Clad Barrier.

Core Cooling - RED indicates significant superheating and core uncovery and is considered to indicate loss of the Fuel Clad Barrier.

2. Primary Coolant Activity Level: The 300 uCi/cc DEI assessment by the NUMARC EAL Task Force indicates that this amount of coolant activity is well above that expected for iodine spikes and corresponds to about 2% to 5% fuel clad damage. This amount of clad damage indicates significant clad heating and thus the Fuel Clad Barrier is considered lost.

3. Containment Radiation Monitoring: This reading is a value which indicates the release of reactor coolant, with elevated activity indicative of fuel damage, into the containment. The reading should be calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 300 uCi/gm dose equivalent I-131 into the containment atmosphere. Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations (including iodine spiking) allowed within technical specifications and are therefore indicative of fuel damage (approximately 2 - 5% clad failure depending on core inventory and RCS volume).

FEF 2. - MODES: ALL 1. Critical Safety Function Status: This EAL is for using Critical Safety Function Status Tree (CSFST) monitoring and functional recovery procedures. RED path indicates an extreme challenge to the safety function derived from appropriate instrument readings, and these CSFs indicate a potential loss of RCS barrier.

2. RCS Leak Rate: The "Potential Loss" EAL is based on the inability to maintain normal liquid inventory within the Reactor Coolant System (RCS) by normal operation of the Chemical and Volume Control System which is considered as one centrifugal charging pump discharging to the charging header. Pressurizer level added because this is safety injection criteria per WCGS procedure OFN BB-007, "SG/RCS Leakage High".

EXPLANATIONS/BASES
CHART - FUEL ELEMENT FAILURE

FEF 3 & 4. - MODES: ALL See LRCB 4

FEF 5. - MODES: ALL This IC is included as an Unusual Event because it is considered to be a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. This EAL addresses coolant samples exceeding coolant technical specifications for iodine spike. Escalation of this IC to the Alert level is via the Fission Product Barrier Degradation Monitoring ICs.

FEF 6. - MODES: ALL See LRCB 7

EXPLANATIONS/BASES
CHART - LOSS OF ELECTRICAL POWER/ASSESSMENT CAPABILITY

LEP/AC 1 - MODES: ALL Prolonged loss of AC power reduces required redundancy and potentially reduces 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.

LEP/AC 2 - MODES: 5, 6, & E Loss of all AC power compromises all plant safety systems requiring electric power, including RHR, ECCS, Containment Heat Removal, and the Ultimate Heat Sink. Prolonged loss of all AC power will cause core uncovering and loss of containment integrity, thus this event can escalate to a General Emergency. The 15 minute time duration was selected to exclude transient or momentary power loss.

LEP/AC 3 - MODES: 1 THROUGH 4 Loss of all AC power compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal, Spent Fuel Heat Removal, and the Ultimate Heat Sink. When in cold shutdown, refueling, or defueled mode the event can be classified as an Alert, because of the significantly reduced decay heat, lower temperature and pressure, increasing the time to restore one of the emergency busses, relative to that specified for the Site Area Emergency EAL. Escalating to Site Area Emergency, if appropriate, is by Radioactive Effluent Release, or Duty Emergency Director/Duty Emergency Manager Judgment ICs. Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

LEP/AC 4 - MODES: 1 THROUGH 4 Loss of all AC power compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal and the Ultimate Heat Sink. Prolonged loss of all AC power will lead to loss of fuel clad, RCS, and containment. The 4 hours to restore AC power was based on the site blackout coping analysis performed in conformance with 10 CFR 50.63 and Reg. Guide 1.155, "Station Blackout". Although this IC may be viewed as redundant to the Fission Barrier Degradation IC, its inclusion is necessary to help assure timely recognition and emergency response.

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

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

In addition, under these conditions, fission product barrier monitoring capability may be degraded. Although it may be difficult to predict when power can be restored, it is necessary to give the Duty Emergency Director/Duty Emergency Manager a reasonable idea of how quickly they may 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 Fission Product Barriers is IMMINENT? (CSFST shows
   Red or Orange path on Core Cooling <u>OR</u> Red path on Heat Sink)
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 barrier can be prevented ?

Thus, indication of continuing core cooling degradation must be based on Fission Product Barrier monitoring with particular emphasis on Duty Emergency Director/Duty Emergency Manager judgment as it relates to IMMINENT Loss or Potential Loss of fission product barriers and degraded ability to monitor fission product barriers.

EXPLANATIONS/BASES
CHART - LOSS OF ELECTRICAL POWER/ASSESSMENT CAPABILITY

LEP/AC 5 - MODES: 1 THROUGH 4

Loss of all DC power compromises ability to monitor and control plant safety functions. Prolonged loss of all DC power will cause core uncovering and loss of containment integrity when there is significant decay heat and sensible heat in the reactor system Escalation to a General Emergency would occur be Abnormal Rad Levels/ Radiological Effluent, Fission Product Barrier Degradation, or Duty Emergency Director/Duty Emergency Manager Judgment ICs. Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

105 VDC bus voltage was based on the minimum bus voltage necessary for the operation of safety related equipment. This voltage value incorporates a margin of at least 15 minutes of operation before the onset of inability to operate those loads. This voltage is usually near the minimum voltage selected when battery sizing is performed. Typically the value for the entire battery set is approximately 105 VDC. For a 60 cell string of batteries the cell voltage is 1.75 Volts per cell.

LEP/AC 6 - MODES: 1 THROUGH 4

This IC and its associated EAL are intended to recognize the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment.

Quantification of "Most" is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions. This judgment is supported by the specific opinion of the Shift Supervisor that additional operating personnel will be required to provide increased monitoring of system operation to safely operate the unit.

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, the Unusual Event is based on ADM 2 "Inability to Reach Required Shutdown Within Technical Specification Limits."

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

"Unplanned" loss of annunciators or indicator excludes scheduled maintenance and testing activities.

LEP/AC 7 - MODES: 1 THROUGH 4

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

"Significant Transient" includes response to automatic or manually initiated functions such as scrams, runbacks involving greater than 25% thermal power change, ECCS injections, or thermal power oscillations of 10% or greater.

EXPLANATIONS/BASES
CHART - LOSS OF ELECTRICAL POWER/ASSESSMENT CAPABILITY

LEP/AC 8 -DC Power - MODES: ALL

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

The onsite communications loss includes **all** of the following:

1. Complete failure of the plant telephone system.
2. Complete failure of the Gaitronics system.
3. Complete failure of the plant radio system.

The offsite communications loss includes **all** of the following:

1. Complete failure of the ENS line.
2. Complete failure of offsite telephone service (inability to receive or call a location offsite).
3. Complete failure of onsite fax machines.

This EAL is intended to be used only when extraordinary means are being utilized to make communications possible (relaying of information from radio transmissions, individuals being sent to offsite locations, etc.)

LEP/AC 9 - MODES : N/A

LEP/AC 5 only applicable in Modes 1 - 4. Unplanned loss of DC Power in Modes 5 & 6 is covered in LEP/AC 8.

LEP/AC 10 -MODES: 1 THROUGH 4

This IC and the associated EAL are intended to provide an escalation from LEP/AC 1. The condition indicated by this IC is the degradation of the offsite and onsite power systems such that any additional single failure would result in a station blackout. This condition could occur due to a loss of offsite power with a concurrent failure of one emergency generator to supply power to its emergency busses. The subsequent loss of this single power source would escalate the event to a Site Area Emergency in accordance with LEP/AC 2.

LEP/AC 11 -MODES: 1 THROUGH 4

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

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

EXPLANATIONS/BASES
CHART - LOSS OF ELECTRICAL POWER/ASSESSMENT CAPABILITY

LEP/AC 12 -MODES: 1 THROUGH 4

The purpose of this IC and its associated EALs is to recognize a loss of DC power compromising the ability to monitor and control the removal of decay heat during Cold Shutdown or Refueling operations. This EAL is intended to be anticipatory in as much as the operating crew may not have necessary indication and control of equipment needed to respond to the loss.

Unplanned is included in this IC and EAL to preclude the declaration of an emergency as a result of planned maintenance activities. Routinely plants will perform maintenance on a Train related basis during shutdown periods. It is intended that the loss of the operating (operable) train is to be considered.

105 VDC bus voltage was based on the minimum bus voltage necessary for the operation of safety related equipment. This voltage value should incorporate a margin of at least 15 minutes of operation before the onset of inability to operate those loads. This voltage is usually near the minimum voltage selected when battery sizing is performed. Typically the value for the entire battery set is approximately 105 VDC. For a 60 cell string of batteries the cell voltage is 1.75 Volts per cell.

EXPLANATIONS/BASES
CHART - FUEL HANDLING ACCIDENT

FHA 1. - <u>MODES: ALL</u> This procedure provides the necessary instructions to minimize the release of airborne activity following a fuel handling accident which indicates a potential degradation of the level of safety of the plant.

FHA 2 & 3. - <u>MODES: ALL</u> NUREG-0818, "Emergency Action Levels for Light Water Reactors," forms the basis for these EALs. There is time available to take corrective actions, and there is 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.

Thus, an Alert Classification for this event is appropriate. Escalation, if appropriate, would occur via Abnormal Rad Level/Radiological Effluent or Emergency Director Judgment.

EXPLANATIONS/BASES
CHART - SAFETY SYSTEM FAILURE OR MALFUNCTION

SSFM 1. - MODES: 1 THROUGH 3 A complete loss of secondary heat sink is indicated.

SSFM 2. - MODES: 1 THROUGH 3 In combination with a loss of MFW capability (addressed in box SSFM 1) a complete loss of secondary heat sink is indicated. CSFST indicators are used as determining factors.

SSFM 3. - MODES: 1 THROUGH 3 This box is used to determine if any ECCS system is capable of delivering water to the core. 250 GPM is chosen because it is greater than the flow from one CCP at the PRZR PORV lift setpoint. It is anticipated that conditions leading to this box will require operator initiation of RCS bleed and feed in accordance with EMG FRH1. FRG actions should raise ECCS injection flow to that required for adequate heat removal.

SSFM 4 - MODES: 1 This condition indicates failure of the automatic protection system to scram the reactor. This condition is more than a potential degradation of a safety system in that a front line automatic protection system did not function in response to a plant transient and thus the plant safety has been compromised, and design limits of the fuel may have been exceeded. An Alert is indicated because conditions exist that lead to potential loss of fuel clad or RCS. Reactor protection system setpoint being exceeded (rather than limiting safety system setpoint being exceeded) is specified here because failure of the automatic protection system is the issue. A manual scram is any set of actions by the reactor operator(s) at the reactor control console which causes control rods to be rapidly inserted into the core and brings the reactor subcritical (e.g., reactor trip switch). Failure of manual scram would escalate the event to a Site Area Emergency.

SSFM 5.- MODES: 1 Automatic and manual scram are not considered successful if action away from the reactor control console was required to scram the reactor.
Under these conditions, the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed. A Site Area Emergency is indicated because conditions exist that lead to imminent loss or potential loss of both fuel clad and RCS. Although this IC may be viewed as redundant to the Fission Product Barrier Degradation IC, its inclusion is necessary to better assure timely recognition and emergency response. Escalation of this event to a General Emergency would be via Fission Product Barrier Degradation or Duty Emergency Director/Duty Emergency Manager Judgment ICs.

EXPLANATIONS/BASES
CHART - SAFETY SYSTEM FAILURE OR MALFUNCTION

SSFM 6. - MODES: 1 Automatic and manual scram are not considered successful if action away from the reactor control console is required to scram the reactor.
Under the conditions of this IC and its associated EALs, the efforts to bring the reactor subcritical have been unsuccessful and, as a result, the reactor is producing more heat than the maximum decay heat load for which the safety systems were designed. Although there are capabilities away from the reactor control console, such as emergency boration, the continuing temperature rise indicates that these capabilities are not effective. This situation could be a precursor for a core melt sequence.
The extreme challenge to the ability to cool the core is intended to mean that the core exit temperatures are at or approaching 1200°F or that the reactor vessel water level is below the top of active fuel. For CSFSTs, this EAL equates to a Core Cooling RED condition.
Another consideration is the inability to initially remove heat during the early stages of this sequence. If emergency feedwater flow is insufficient to remove the amount of heat required by design from a least one steam generator, an extreme challenge should be considered to exist. This EAL equates to a Heat Sink RED condition on the CSFSTs.
In the event either of these challenges exist at a time that the reactor has not been brought below the power associated with the safety system design (typically 3 to 5% power) a core melt sequence exists. In this situation, core degradation can occur rapidly. for this reason, the General Emergency declaration is intended to be anticipatory of the fission product barrier matrix declaration to permit maximum offsite intervention time.

SSFM 7. - MODES: 1 THROUGH 4 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 be via Abnormal Rad Levels/Radiological Effluent, Duty Emergency Director /Duty Emergency Manager Judgment, or Fission Product Barrier Degradation ICs.

EXPLANATIONS/BASES
CHART - SAFETY SYSTEM FAILURE OR MALFUNCTION

SSFM 8. - MODES: 5 & 6 This EAL addresses complete loss of functions required for core cooling during refueling and cold shutdown modes. Escalation to Site Area Emergency or General Emergency would be via Abnormal Rad Levels/Radiological Effluent or Duty Emergency Director/Duty Emergency Manager Judgment, or Fission Product Barrier Degradation ICs.

This IC and its associated EAL are based on concerns raised by Generic Letter 88-17, "Loss of Decay Heat Removal." A number of phenomena such as pressurization, vortexing, steam generator U-tube draining, RCS level differences when operating at a mid-loop condition, decay heat removal system design, and level instrumentation problems can lead to conditions where decay heat removal is lost and core uncovery can occur. NRC analyses show that sequences that can cause core uncovery in 15 to 20 minutes and severe core damage within an hour after decay heat removal is lost. Under these conditions, RCS integrity is lost and fuel clad integrity is lost or potentially lost, which is consistent with a Site Area Emergency.

"Uncontrolled" means that system temperature increase is not the result of planned actions by the plant staff.

The EAL guidance related to uncontrolled temperature rise is necessary to preserve the anticipatory philosophy of NUREG-0654 for events starting from temperatures much lower than the cold shutdown temperature limit.

SSFM 9.- MODES: 5&6 Under the conditions specified by this IC, severe core damage can occur and reactor coolant system pressure boundary integrity may not be assured. This IC covers sequences such as prolonged boiling following loss of decay heat removal.

Thus, declaration of a Site Area Emergency is warranted under the conditions specified by the IC. Escalation to a General Emergency is via radiological effluent IC.

EXPLANATIONS/BASES
CHART - ADMINISTRATIVE

ADM 1. - MODES 1 THROUGH 2 Required reporting for ECCS actuations and conditions that could lead to a degraded level of safety of the plant.

ADM 2. - MODES: 1 THROUGH 4 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 site Technical Specifications requires a one hour report under 10 CFR 50.72 (b) Non-emergency events. The plant is within its safety envelope when being shut down within the allowable action statement time in the Technical Specifications. An immediate Notification of an 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 System Malfunction, Hazards, or Fission Product Barrier Degradation ICs.

ADM 3. - MODES: 1 THROUGH 4 A Containment Breach, by itself, is classified as a Notification of Unusual Event in accordance with Reg. Guide 1.101.

EXPLANATIONS/BASES
CHART - LOSS OF PLANT CONTROL/SECURITY COMPROMISE

LPC/SC 1. - MODES: ALL This EAL is based on the WCGS Site Security Plan. Security events which do not represent at least a potential degradation in the level of safety of the plant, are reported under 10 CFR 73.71 or in some cases under 10 CFR 50.72. The plant Protected Area Boundary is typically that part within the security isolation zone and is defined in the WCGS security plan. Bomb devices discovered within the plant Vital Area would result in EAL escalation.

LPC/SC 2. - MODES: ALL This class of security events represents an escalated threat to plant safety above that contained in the Unusual Event. For the purposes of this IC, a civil disturbance which penetrates the protected area boundary can be considered a hostile force. Intrusion into a vital area by a hostile force will escalate this event to a Site Area Emergency.

LPC/SC 3. - MODES: ALL This class of security events represents an escalated threat to plant safety above that contained in the Alert IC in that a hostile force has progressed from the Protected Area to the Vital Area.

LPC/SC 4. - MODES: ALL This IC encompasses conditions under which a hostile force has taken physical control of vital area required to reach and maintain safe shutdown.

LPC/SC 5. - MODES: ALL With the Control Room evacuated, additional support, monitoring and direction through the Technical Support Center and/or the Emergency Operations Facility is necessary. Inability to establish plant control from outside the Control Room will escalate this event to a Site Area Emergency.

LPC/SC 6. - MODES: ALL Expeditious transfer of safety systems has not occurred but fission product barrier damage may not yet be indicated. WCGS time for transfer based on analysis or assessments as to how quickly control must be reestablished without core uncovering and/or core damage. This time should not exceed 15 minutes. In cold shutdown and refueling modes, operator concern is directed toward maintaining core cooling such as is discussed in Generic Letter 88-17, "Loss of Decay Heat Removal." In power operation, hot standby, and hot shutdown modes, operator concern is primarily directed toward maintaining critical safety functions and thereby assuring fission product barrier integrity. Escalation of this event, if appropriate, would be by Fission Product Barrier Degradation, Abnormal Rad Levels/Radiological Effluent, or Duty Emergency Director/Duty Emergency Manager Judgment ICs.

EXPLANATIONS/BASES
CHART - FIRE

FR 1. - MODES: ALL The purpose of this IC is to address the magnitude and extent of fires that may be potentially significant precursors to damage to safety systems. This excludes such items as fires within administration buildings, waste-basket fires, and other small fires of no safety consequence. This IC applies to buildings and areas contiguous to plant vital areas or other significant buildings or areas. The intent of this IC is not to include buildings (i.e., warehouses) or areas that are not contiguous or immediately adjacent to plant vital areas. Verification of the alarm in this context means those actions taken in the control room to determine that the control room alarm is not spurious.

FR 2. - MODES: ALL The specified areas contain functions and systems required for the safe shutdown of the plant. This list was obtained from WCGS USAR Table 3.3-1. This will make it easier to determine if the fire or explosion is potential affecting one or more redundant trains of safety systems. Escalation to a higher emergency class, if appropriate, will be based on System Malfunction, Fission Product Barrier Degradation, Abnormal Rad Levels/Radiological Effluent, or Duty Emergency Director/Duty Emergency Manager Judgment ICs.

EXPLANATIONS/BASES
CHART - NATURAL PHENOMENA

NP 1. - MODES: ALL NP 1 was developed on WCGS basis. Damage may be caused to some portions of the site, but should not affect ability of safety functions to operate. Method of detection can be based on instrumentation, validated by a reliable source, or operator assessment. As defined in the EPRI-sponsored "Guidelines for Nuclear Plant Response to an Earthquake", dated October 1989, a "felt earthquake" is: An earthquake of sufficient intensity such that: (a) the vibratory ground motion is felt at the nuclear plant site and recognized as an earthquake based on a consensus of Control Room operators on duty at the time, and (b) for plants with operable seismic instrumentation, the seismic switches of the plant are activated. At Wolf Creek these seismic switches are set at an acceleration of 0.01g.

NP 2. - MODES: ALL NP 2 based on WCGS USAR design basis. Seismic events of this magnitude can cause damage to safety functions.

NP 3. - MODES: ALL An earthquake greater than SSE could place the safety systems in a severely degraded condition. Equipment can be expected to be exposed to forces greater than design limits.

NP 4. - MODES: ALL If visual inspection of plant safety related equipment and structures indicate a loss of a function needed to reach cold shutdown, then emergency escalation is warranted.

NP 5. - MODES: ALL NP 5 is based on WCGS USAR Section 3.3.1.1. Wind loads of this magnitude can cause damage to safety functions.

NP 6. - MODES: ALL NP 6 is based on the assumption that a tornado striking (touching down) within the protected boundary may have potentially damaged plant structures containing functions or systems required for safe shutdown of the plant. If such damage is confirmed visually or by other in-plant indications, the event may be escalated to Alert.

NP 7. - MODES: ALL This EAL specifies structure containing systems and functions required for a safe shutdown of the plant. This list was obtained from WCGS USAR Table 3.3-1.

EXPLANATIONS/BASES
CHART - OTHER HAZARDS

OH 1. - MODES: ALL This IC is based on releases in concentrations within the site boundary that will affect the health of plant personnel or affecting the safe operation of the plant with the plant being within the evacuation area of an offsite event (i.e., tanker truck accident releasing toxic gases, etc.)

OH 2. - MODES: ALL This IC is based on gases that have entered a plant structure affecting the safe operation of the plant. This IC applies to buildings and areas contiguous to plant Vital Areas or other significant buildings or areas (i.e., Service Water Pump house). The intent of this IC is not to include buildings (i.e., warehouses) or other areas that are not contiguous or immediately adjacent to plant Vital Areas. It is appropriate that increased monitoring be done to ascertain whether consequential damage has occurred. Escalation to a higher emergency class, if appropriate, will be based on System Malfunction, Fission Product Barrier Degradation, Abnormal Rad Levels/Radioactive Effluent, or Duty Emergency Director/Duty Emergency Manager Judgment ICs.

OH 3. - MODES: ALL This EAL is intended to address such items as plane or helicopter crash, or on some sites, train crash, or barge crash that may potentially damage plant structures containing functions and systems required for safe shutdown of the plant. If the crash is confirmed to affect a plant vital area, the event may be escalated to Alert.

OH 4. - MODES: ALL This EAL specifies structures containing systems and functions required for safe shutdown of the plant. This list was obtained from WCGS USAR Table 3.3-1.

OH 5. - MODES: ALL For this EAL only those explosions of sufficient force to damage permanent structures or equipment within the protected area should be considered. As used here, an explosion is a rapid, violent, unconfined combustion, or a catastrophic failure or pressurized equipment, that potentially imparts significant energy to near-by structures and materials. 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. The Duty Emergency Director/Duty Emergency Manager also needs to consider any security aspects of the explosion, if applicable.

EXPLANATIONS/BASES
CHART - OTHER HAZARDS

OH 6. - MODES: ALL T - EAL is intended to address main turbine rotating component failures of sufficient magnitude to cause observable damage to the turbine casing or to the seals of the turbine generator. Of major concern is the potential for leakage of combustible fluids (lubricating oils) and gases (hydrogen cooling) to the plant environs. Actual fires and flammable gas build up are appropriately classified under OH 1 or the Fire IC for Emergency Cla sification. This EAL is consistent with the definition of an Unusual Event while maintaining the anticipatory nature desired and recognizing the risk to non-safety related equipment. Escalation of the emergency classification is based on potential damage done by missiles generated by the failure, or in conjunction with a steam generator tube rupture. These latter events would be classified by the radiological ICs or Fission Product Barrier ICs.

OH 7. - MODES: ALL Train derailment that could involve the shipment of radioactive material or euipment and possible new or spent fuel.

OH 8. - MODES: ALL Required notification of transport of a contaminated injured individual to an offsite hospital.

OH 9. - MODES: ALL This EAL is intended to address unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Duty Emergency Director/Duty Emergency Manager to fall under the Unusual Event emergency class.

From the broad perspective, one area that may warrant Duty Emergency Director/Duty Emergency Manager judgment is related to likely or actual breakdown of site specific mitigating actions. Examples to consider include inadequate emergency response procedures, transient response either unexpected or not understood, failure or unavailability of emergency systems during an accident in excess of that assumed in accident analysis, or insufficient availability of equipment and/or support personnel.

Specific example of actual events that may require Duty Emergency Director/Duty Emergency Manager judgment for Unusual Event declaration are listed here for consideration. However, this list is by no means all inclusive and is not intended to limit the discretion of the site to provide further examples.

- Aircraft crash on-site.
- Train derailment on-site.
- Near-site explosion which may adversely affect normal site activities.
- Near-site release of toxic or flammable gas which may adversely affect normal site activities.
- Uncontrolled RCS Cooldown due to Secondary Depressurization.

It is also intended that the Duty Emergency Directors/Duty Emergency Managers judgment not be limited by any list of events as defined here or as augmented by the site. This list is provided solely as examples for consideration and it is recognized that actual events may not always follow a pre-conceived description.

EXPLANATIONS/BASES
CHART - OTHER HAZARDS

OH 10. - MODES: ALL This EAL is intended to address unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Duty Emergency Director/Duty Emergency Manager to fall under the Alert emergency class.

OH 11. - MODES: ALL This EAL is intended to address unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Duty Emergency Director/Duty Emergency Manager to fall under the General Emergency class.

OH 12. - MODES: ALL This EAL is intended to address unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Duty Emergency Director/Duty Emergency Manager to fall under the Site Area Emergency class.