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Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402

CNL-14-039

April 16, 2014

10 CFR 50.4
10 CFR 50, Appendix E
10 CFR 50.54(q)
10 CFR 72.4
10 CFR 72.44(f)

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Browns Ferry Nuclear Plant, Units 1, 2, and 3
Renewed Facility Operating License Nos. DPR-33, DPR-52, and DPR-68
NRC Docket Nos. 50-259, 50-260, and 50-296

Sequoyah Nuclear Plant, Units 1 and 2
Facility Operating License Nos. DPR-77 and DPR-79
NRC Docket Nos. 50-327 and 50-328

Watts Bar Nuclear Plant, Unit 1
Facility Operating License No. NPF-90
NRC Docket No. 50-390

Subject: **Radiological Emergency Plan Revision**

In accordance with the requirements of Title 10 of the Code of Federal Regulations (10CFR) 50.54(q) and 10 CFR 72.44(f), enclosed is Revision 103 of the Tennessee Valley Authority (TVA) Radiological Emergency Plan (REP). Revision 103 became effective on March 17, 2014.

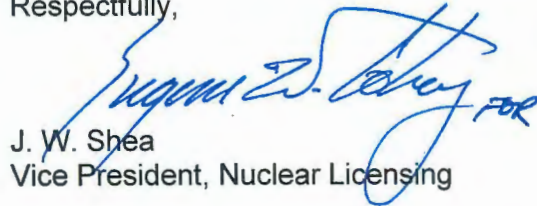
Enclosure 1 provides a summary of the changes and the analysis performed that concluded the changes do not reduce the effectiveness of the plan, and the plan, as revised, continues to meet the requirements in Appendix E to 10 CFR 50 and the planning standards of 10 CFR 50.47(b).

April 16, 2014

Enclosure 2 provides a copy of the TVA Radiological Emergency Plan - Generic. Enclosure 3 provides a copy of REP Appendix A: Browns Ferry Nuclear Plant. Enclosure 4 provides a copy of REP Appendix B: Sequoyah Nuclear Plant. Enclosure 5 provides a copy of REP Appendix C: Watts Bar Nuclear Plant.

There are no new regulatory commitments in this letter. If you have any questions regarding this submittal, please contact Edward D. Schrull at (423) 751-3850.

Respectfully,

A handwritten signature in blue ink, appearing to read "J. W. Shea", is written over the typed name and title.

J. W. Shea
Vice President, Nuclear Licensing

Enclosures:

1. Summary of Changes and Analysis
2. Radiological Emergency Plan, Generic, Revision 103
3. Radiological Emergency Plan, Appendix A: Browns Ferry Nuclear Plant, Revision 103
4. Radiological Emergency Plan, Appendix B: Sequoyah Nuclear Plant, Revision 103
5. Radiological Emergency Plan, Appendix C: Watts Bar Nuclear Plant, Revision 103

cc (Enclosures):

NRC Regional Administrator - Region II
NRC Senior Resident Inspector - Browns Ferry Nuclear Plant
NRC Senior Resident Inspector - Sequoyah Nuclear Plant
NRC Senior Resident Inspector - Watts Bar Nuclear Plant Unit 1
NRC Senior Resident Inspector - Watts Bar Nuclear Plant Unit 2
NRR Project Manager - Browns Ferry Nuclear Plant
NRR Project Manager - Sequoyah Nuclear Plant
NRR Project Manager - Watts Bar Nuclear Plant

Enclosure 1
Summary of the Changes and Analysis for Radiological Emergency Plan (REP)
Revision 103

Affected Document	Description of Changes	Summary of Analysis of Change
REP, Generic	Changed wording to state that if a siren silent test is scheduled on the same date as a full test the silent test is not performed [PER 789854].	The change does not alter or change TVA's commitment to performing timely and accurate accident assessments during an emergency. This change does not reduce or negatively impact any of the functions or elements identified. This revision will continue to meet all capability and timeliness commitments. The activity does continue to comply with the requirements of 10 CFR 50.47(b)(9) and 10 CFR Part 50 Appendix E. The proposed activity does not constitute a Reduction in Effectiveness in the TVA REP.
REP, Generic	Changed requirements for revisions to include entry into the Rev Log.	This change is considered minor and has no effect on any planning standard or timely implementation of the REP. The proposed activity does not constitute a Reduction in Effectiveness in the TVA REP.
REP, Appendix A	Removed last sentence of first paragraph of basis for EAL 1.1-S1 as it was misleading and could lead to misclassification;	This change is considered minor and has no effect on any planning standard or timely implementation of the REP. The proposed activity does not constitute a Reduction in Effectiveness in the TVA REP.
REP, Appendix A	Changed Minimum Alternate RPV Flooding Press (MARFP) to Minimum Steam Cooling Press (MSCP) to align with current EOI verbiage. (PER 810374).	This change is considered minor and has no effect on any planning standard or timely implementation of the REP. The proposed activity does not constitute a Reduction in Effectiveness in the TVA REP.
REP, Appendix B	Added note and its basis to EALs 1.1.5 and 1.3.5;	The change does not alter or change the objective and purpose of the EAL. This change does not reduce or negatively impact any of the functions or elements identified. The activity does continue to comply with the requirements of 10 CFR 50.47(b)(9) and 10 CFR Part 50 Appendix E. The proposed activity does not constitute a Reduction in Effectiveness in the TVA REP.
REP, Appendix B	Removed MSS label from fig B-4	The proposed activity does continue to comply with the requirements. The proposed activity does not constitute a reduction in effectiveness (RIE). This change is considered minor and has no effect on any planning standard or timely implementation of the REP.
REP, Appendix C	Updated EAL values for EALs 1.1.5 and 1.3.5 based on approval of calculation TIRPS162 R14	The change does not alter or change the objective and purpose of the EAL. The proposed activity does continue to comply with the requirements. The proposed activity does not constitute a reduction in effectiveness (RIE). This change has no effect on any planning standard or timely implementation of the REP.
REP, Appendix C	Updated Table 7-1based on approval of calculation WBNTSR115 Rev 7;	The change does not alter or change the objective and purpose of the EAL. The proposed activity does continue to comply with the requirements. The proposed activity does not constitute a reduction in effectiveness (RIE). This change has no effect on any planning standard or timely implementation of the REP.

Enclosure 1
Summary of the Changes and Analysis for Radiological Emergency Plan (REP)
Revision 103

Affected Document	Description of Changes	Summary of Analysis of Change
REP, Appendix C	Revised Table 7.1 Note 3 on RM 90-450	The change does not alter or change the objective and purpose of the EAL. The proposed activity does continue to comply with the requirements. The proposed activity does not constitute a reduction in effectiveness (RIE). This change has no effect on any planning standard or timely implementation of the REP.
REP, Appendix C	Updated Table 7-2 based on approval of calculation WBNTSR115 Rev 7.	The change does not alter or change the objective and purpose of the EAL. The proposed activity does continue to comply with the requirements. The proposed activity does not constitute a reduction in effectiveness (RIE). This change has no effect on any planning standard or timely implementation of the REP.
REP, Appendix C	Updated references to WBNTSR115 Rev. 2 to WBNTSR115 Rev. 7	The change does not alter or change the objective and purpose of the EAL. The proposed activity does continue to comply with the requirements. The proposed activity does not constitute a reduction in effectiveness (RIE). This change is considered minor and has no effect on any planning standard or timely implementation of the REP.

Enclosure 2

Radiological Emergency Plan (REP)

Generic

Revision 103

Tennessee Valley Authority	Title	REP REV. 103
	RADIOLOGICAL EMERGENCY PLAN (GENERIC PART)	
		Effective Date: 03/17/2014

QUALITY RELATED

WRITTEN BY: Jimmy Watson REVIEWED BY: Dennis Jones 11/07/13
Signature *Signature* *Date*

PLAN EFFECTIVENESS DETERMINATION
 REVIEWED BY: John Parshall 11/07/13
Signature *Date*

CONCURRENCES

Concurrence Signature	Date	Concurrence Signature	Date
EP Manager (BFN)		BFN PORC Chairman	
John Kulisek	<u>12/18/13</u>	D. Lang Hughes	<u>12/18/13</u>
John Kulisek	<u>12/18/13</u>	D. Lang Hughes	<u>12/18/13</u>
EP Manager (SQN)		SQN PORC Chairman	
Mike Halter	<u>12/30/13</u>	Tom Marshall	<u>12/30/13</u>
Mike Halter	<u>12/30/13</u>	Tom Marshall	<u>12/30/13</u>
EP Manager (WBN)		WBN PORC Chairman	
Tom Detchemendy	<u>12/03/13</u>	Sean M. Connors	<u>12/04/13</u>
Tom Detchemendy	<u>12/03/13</u>	Sean M. Connors	<u>12/04/13</u>
Manager, EP Program Planning and Implementation		<input type="checkbox"/>	
John Parshall	<u>11/07/13</u>		
John Parshall	<u>11/07/13</u>		
Sr. Manager Emergency Preparedness		<input type="checkbox"/>	
Walter H. Lee	<u>01/13/14</u>		
Walter H Lee	<u>01/13/14</u>		

APPROVAL

APPROVED BY: Renee Milner V.P. Nuclear Support 01/22/14
Signature *Title* *Organization* *Date*

1.0 DEFINITIONS AND ABBREVIATIONS

Activated - Minimum Activation Staff Positions (MASP) are present onsite or in the facility within ~ 60(90 for WBN) minutes following declaration, and ready to assume assigned functions. Turnovers should be done as soon as practical, with consideration to performing them after the facility is activated.

Annual - Any 12 months, plus or minus 3 months.

Exceptions:

1. Exercises, drills, emergency information for residents, media training, and offsite emergency response training are defined as "once per calendar year."
2. TVA annual training is for a 12-month period which includes a grace period extending to the end of the calendar quarter in which training is due.

ANI - American Nuclear Insurers.

AUO - Assistant Unit Operator.

BFN - Browns Ferry Nuclear Plant.

BFN-EIPs (Browns Ferry Nuclear Plant Emergency Plan Implementing Procedures) - The set of BFN emergency response procedures developed to ensure that the capabilities described in the NP-REP are fulfilled at BFN.

CDE - Committed Dose Equivalent as defined by 10 CFR 20.1201.

CECC (Central Emergency Control Center) - The offsite TVA emergency response facility located in Chattanooga with the overall TVA responsibility for response to an emergency. It consists of a director and staff to coordinate and direct TVA's efforts during the emergency.

CECC-EIPs (Central Emergency Control Center Emergency Plan Implementing Procedures) - The set of emergency response procedures developed to ensure that the capabilities described in the NP-REP are fulfilled in the CECC and offsite.

COO - Chief Operating Officer.

COC - TVA Chattanooga Office Complex, Chattanooga, Tennessee.

DAC - Derived Air Concentration.

DDE - Deep Dose Equivalent as defined by 10 CFR 20.1201

DHS - Department of Homeland Security - An agency of the U.S. Government.

DOE - U.S. Department of Energy.

DOT - U.S. Department of Transportation.

Drill - A supervised instruction period aimed at testing, developing, and maintaining skills in a particular operation. A drill is often a component of an exercise.

EAL (Emergency Action Level) - Specific events and criteria used to determine the appropriate emergency classification.

EDO - Emergency Duty Officer.

Emergency Classification (Also Class or Classification) - A scheme derived to categorize a plant accident into one of four classes according to severity so that appropriate actions might be rapidly taken.

EMR (Emergency Medical Responder) - An individual certified under a recognized TVA system to provide emergency and related services to victims of illness or injury.

EMT - Emergency Medical Technician.

ENS (Emergency Notification System) - The phone line used to notify and inform the NRC of Event Status Data.

Environs - The atmospheric, terrestrial, and aquatic areas outside the site boundary.

EOC - Emergency Operations Center.

EOF - Emergency Operations Facility.

EP - Emergency Preparedness.

EP Staff - Emergency Services, Emergency Preparedness Staff.

EPA (Environmental Protection Agency) - An agency of the U.S. Government.

EPZ (Emergency Planning Zone) - The area surrounding the site for which planning is performed to prepare to respond to a nuclear plant accident. The two zones are (1) Plume Exposure EPZ - 10-mile radius; (2) Ingestion Exposure EPZ - 50-mile radius.

Exclusion Area Boundary - The area for which TVA has absolute authority for exclusion of personnel and property within the site boundary. This boundary is used in FSAR dose assessments to define the distance to the first member of the public and is defined in the FSAR.

Exercise - An event that tests the integrated capability and a major portion of the basic elements existing within the emergency plan.

FEMA - An agency of the Federal Government. Formerly agency in charge of emergency preparedness, responsibility being shifted within parent agency DHS.

FRERP - Federal Radiological Emergency Response Plan.

FSAR (Final Safety Analysis Report) - The final safety report that is submitted to the NRC in support of each plant's application for an operating license.

His - The use of "he," "him," "his," or any other similar terminology is not intended to imply or refer exclusively to the masculine gender. Rather, all such terms are to be read as applicable without regard to sex.

Hostile Action - An act toward a Nuclear Site or its personnel that includes the use of violent force to destroy equipment, takes hostages, and/or intimidates the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the Nuclear Site.

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

HPN (Health Physics Network) - The NRC's health physics information line.

IC - (Initiating Condition) - An event or condition that aligns with the definition of one of the four emergency classification levels by virtue of the potential or actual effects or consequences.

INPO - Institute for Nuclear Power Operations.

JIC (Joint Information Center) - A center established near the affected site to assist the news media in providing press coverage during an emergency.

LRC (Local Recovery Center) - A facility located near the affected site used as additional office space, if necessary, for TVA personnel during recovery operations. The facility is also available for NRC use during an incident.

MCR - Main Control Room.

MERT - Medical Emergency Response Team.

Missiles - As used in the EALs, a missile is any hurled object (e.g., debris from explosions, fragments from rotating equipment breaks).

Monthly - Any 30-day period, plus or minus 7 days.

MS - Management Services

NE - Nuclear Engineering.

NOAA - National Oceanic and Atmospheric Administration.

NOUE - Notification of Unusual Event.

NP - Nuclear Power.

NP-REP (Nuclear Power Radiological Emergency Plan) - The plan which provides the policies and the actions to be used to minimize the impact on personnel, public, and the environment from an accident at a TVA nuclear plant.

NRC - Nuclear Regulatory Commission.

NRP - National Response Plan.

NSS - Nuclear Security Services.

NSSS - Nuclear Steam Supply System.

Offsite - The area around a nuclear plant site that is not onsite.

Onsite - Onsite is defined according to the subject ... (1) in relation to FSAR dose assessment, onsite is "within the exclusion area," (2) in relation to accountability and site notifications, onsite is "within the site's outermost secured area," (3) in relation to EP dose assessments is defined as "1000 meter radius," (4) in other contexts onsite is "within the reservation boundary."

ODS (Operations Duty Specialist) - The 24-hour per day emergency contact for the Tennessee Valley Authority.

ORAU (Oak Ridge Associated Universities) - A nonprofit corporation and prime contractor with DOE for operation of the REAC/TS facility.

ORMMC (Oak Ridge Methodist Medical Center) - In conjunction with the REAC/TS facility, provides continuing medical care to radiological accident victims.

OSL (Optically Stimulated Luminescence) - refers to dosimetry technology used for personal monitoring for x, gamma, and beta radiation. OSL dosimeter and primary dosimeter references are synonymous.

OSC (Operations Support Center) - An area set aside within the plant for providing an assembly area for operational support personnel during an emergency situation.

PABX (Private Automatic Branch Exchange) - A communications system, controlled by TVA, employing microwave and land line transmissions.

PAR - Protective Action Recommendation.

PED - Plan Effectiveness Determination.

Plant Duty Manager - Key plant management serving as the shift engineer's supervisory contact during off-hours.

PORC (Plant Operations Review Committee) - A group of plant supervisors whose function is to provide a safety review of procedures and operations for the plant and make recommendations to the plant manager on these matters.

PSO - Power System Operations.

PSS - Public Safety Service.

Quarterly - Any three-month period, plus or minus one month.

RAA - Radiological Assessment Area of CECC.

RAD PROTECTION - Radiological Protection.

R or r - For purposes of this plan and its implementing procedures, radiation exposure as expressed in units of R/hr and subunits, thereof, is equivalent to dose (rad) and dose equivalent (rem).

RCI - Radiological Control Instructions.

RCS - Reactor Coolant System.

REAC/TS (Radiation Emergency Assistance Center/Training Site) - A special facility that is operated by ORAU for DOE, to provide a sophisticated facility to handle radiological accident victims. The REAC/TS facility is located in ORMMC.

Recovery - The post emergency activities in which the plant conditions are assessed and the plant is returned to an operational mode.

REND (Radiological Emergency Notification Directory) - A directory of key personnel for support of the CECC.

REP - Radiological Emergency Plan.

RMCC (Radiological Monitoring Control Center) - An environmental monitoring coordination center.

RPT - Recirculation Pump Trip.

SAE - Site Area Emergency.

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

SED - Site Emergency Director.

Semiannual - Any six-month period, plus or minus 45 days. (The exception to this is for drills for which it is defined as "twice each calendar year.")

SEOC- State Emergency Operations Center

Site Boundary - The appropriate boundary between "onsite" and "offsite."

SM - Shift Manager.

SQN - Sequoyah Nuclear Plant.

SQN-EIPs (Sequoyah Nuclear Plant Emergency Plan Implementing Procedures) - The set of SQN emergency response procedures developed to ensure that the capabilities described in the NP-REP are fulfilled at SQN.

STA - Shift Technical Advisor.

TEDE - Total Effective Dose Equivalent as defined by 10 CFR 20.

TLD - Thermoluminescent Dosimeter.

TRO - Transmission and Reliability Organization.

TSC (Technical Support Center) - An onsite assembly/work area for designated support individuals knowledgeable of and responsible for engineering and management support of reactor operations in the event of an accident.

UE Unusual Event - terminology that is commonly used referring to the term Notification of Unusual Event (NOUE). UE and NOUE are synonymous in the REP.

WARL (Western Area Radiological Laboratory) - TVA laboratory located in Muscle Shoals, Alabama, capable of analyzing environmental samples for radioactive content.

WBN - Watts Bar Nuclear Plant.

WBN-EIPs (Watts Bar Nuclear Plant Emergency Plan Implementing Procedures) - The set of WBN emergency response procedures developed to ensure that the capabilities described in the NP-REP are fulfilled at WBN.

Weekly - Any seven-day period, plus or minus two days.

2.0 INTRODUCTION

The development, implementation, and maintenance of the NP-REP is the responsibility of Nuclear Power Group (NPG). The Vice President of Nuclear Operations Support has delegated the authority for overall program control of the NP-REP to Emergency Services Emergency Preparedness Manager.

2.1 NP Radiological Emergency Plan (NP-REP) Purpose

NP-REP has been developed to provide protective measures for TVA personnel, and to protect the health and safety of the public in the event of a radiological emergency resulting from an accident at a TVA nuclear plant. This plan fulfills the requirements set forth in Part 50, Title 10 of the Code of Federal Regulations, and was developed in accordance with the NRC and FEMA guidance. As specified in NUREG-0654, Criteria for Preparation and Evaluation of Radiological Emergency Response Plans in Support of Nuclear Power Plants and REG Guide 1.101, the NP-REP provides for the following:

1. Adequate measures are taken to protect employees and the public.
2. Individuals having responsibilities during an accident are properly trained.
3. Procedures exist to provide the capability to cope with a spectrum of accidents ranging from those of little consequence to major core melt.
4. Equipment is available to detect, assess, and mitigate the consequences of such occurrences.
5. Emergency action levels and procedures are established to assist in making decisions.

The Radiological Emergency Plan consists of the NP-REP and appendices which are complementary with the State plans referenced in Appendix E.

2.2 Plan

The NP-REP addresses organizational responsibilities, capabilities, actions, and guidelines for TVA during a radiological emergency. It also describes the centralized emergency management concept which was approved by the NRC Commissioners.

2.3 Appendices

Radiological Emergency Plan information specific to each site is included as appendices.

<u>Site</u>	<u>Appendices</u>
Browns Ferry	A, E
Sequoyah	B, E
Watts Bar	C, E

Appendices A through C detail facility features, capabilities, equipment, and responsibilities. The NP-REP together with the appendices, describes the methods TVA will use to:

1. Detect an emergency condition.
2. Evaluate the severity of the problems.
3. Notify Federal, State, and local agencies of the condition.
4. Activate emergency organizations.
5. Evaluate the possible offsite consequences.
6. Recommend protective actions for the public.
7. Mitigate the consequences of the accident.

Since TVA authority is limited to TVA-owned and -controlled property, State and local agencies are responsible for ordering and implementing actions offsite to protect the health and safety of the public. Appendix E is a list of various State plans which supplement the NP-REP.

2.4 Implementing Procedures

Specific procedures are developed to ensure that the plan is implemented as designed. These implementing procedures are designed to ensure that accidents are properly evaluated, rapid notifications made, and assessment and protective actions performed. These procedures are compiled in the EIPs. Site specific procedures for abnormal and emergency operation and control exist but are not included in the EIPs. These plant operating procedures are designed to ensure the implementation of the EIPs.

2.5 State Radiological Emergency Plans

The State Radiological Emergency Plans, as well as the plans for those portions of states within the 50-mile ingestion pathway, are referenced in Appendix E. These plans provide for the coordinated response of the State and affected local governments as well as the States and local governments within the 50-mile ingestion pathway.

The responsibilities of these major organizations are summarized in Figure 2-1.

2.6 National Response Plan (NRP)

The NRP's Nuclear/ Radiological Incident Annex provides for timely, coordinated response by Federal agencies to a fixed nuclear power plant facility incident. This emergency plan is activated by either the affected State notifying the Department of Homeland Security (DHS), or the utility notifying the NRC of a radiological emergency at a nuclear plant site. The NRP is not included as part of the TVA Radiological Emergency Plan. Should additional radiological monitoring support be required the appropriate State agency will make the request through DHS. The persons authorized to request this assistance, the specific resources expected, and resources available to support the Federal response are provided in the respective State plans.

The NRP may be used by Federal agencies in radiological emergencies. It primarily concerns offsite Federal response in support of State and local governments with jurisdiction for the emergency. The NRP provides the Federal Government's concept of operations for responding to radiological emergencies, outlines Federal policies and planning assumptions, and specifies authorities and responsibilities of each Federal agency that may have a significant role in such emergencies. The CECC Director is the TVA person authorized to request Federal assistance. Such a request from TVA will be made to NRC.

Additional information about the NRP and available support is provided in NRC RIS 2005-13 and NUREG-0728 as well as in the TVA Agency Emergency Response Plan.

FIGURE 2-1

PRINCIPAL ORGANIZATIONAL RESPONSIBILITIES

	<u>Local</u>	<u>State</u>	<u>TVA</u>
Command and Control	X	X	X
Warning	X	X	X
Notification Communications	X	X	X
Public Information	X	X	X
Accident Assessment		X	X
Public Health and Sanitation	X	X	
Social Services	X		
Fire and Rescue	X		X
Traffic Control	X		
Emergency Medical Services	X	X	X
Law Enforcement	X	X	
Transportation	X		
Protective Response	X	X	
Radiological Exposure Control	X	X	X

3.0 EMERGENCY MANAGEMENT ORGANIZATION

The TVA emergency organization is divided into two categories: the onsite organization and the offsite organization. A block diagram of the onsite organization is presented in the site specific appendix and the offsite organization is presented in Figure 3-1. All designated emergency response personnel are required to participate in the Fitness for Duty Program.

The onsite organization is comprised of the Site Emergency Director and technical staff located in the Technical Support Center, a Control Room Staff of operations personnel, and additional support personnel located in the Operations Support Center. The onsite organization is responsible for the onsite response to an emergency condition. All activities onsite will be directed by the Site Emergency Director and will include such functions as control room operations, technical assessment, accident mitigation analysis, onsite radiation surveys, and dose tracking for site personnel.

The offsite emergency organization is designated as the Central Emergency Control Center (CECC) Staff. The CECC staff is comprised of a CECC Director, a supporting group of technical assistants, and representatives of other TVA organizations. The CECC Director and supporting technical assistants report to the CECC during an emergency as required. Other TVA organizations will send representatives to the CECC as requested by the CECC Director.

The CECC is responsible for directing and coordinating the overall TVA response to an emergency condition. Functions such as offsite radiological monitoring and dose assessment, public information, State and local government coordination, and additional plant assessment are handled by the CECC relieving the onsite organization of the many peripheral duties necessary for the successful emergency response.

3.1 Onsite Organization

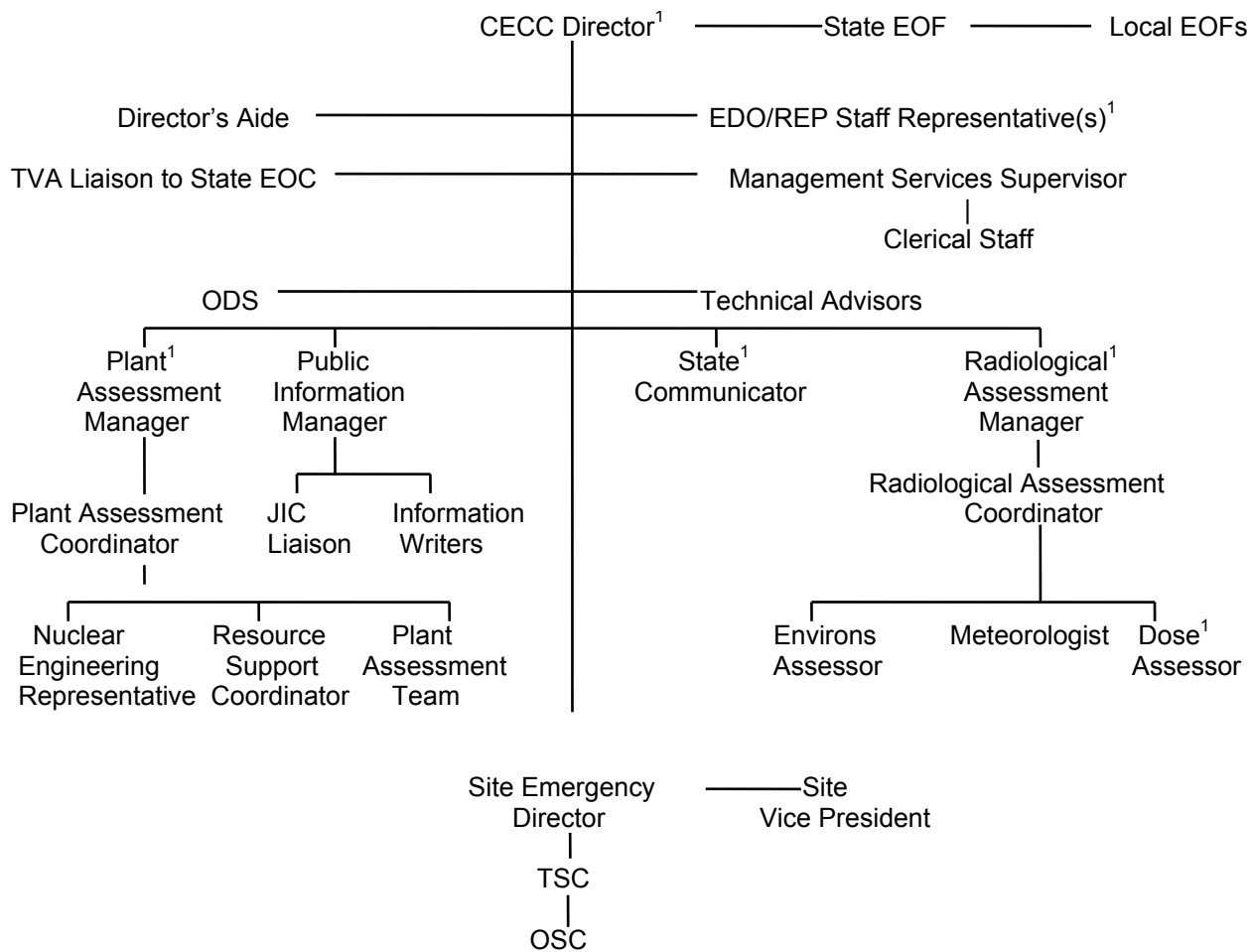
Under normal conditions the Site Vice President is in charge of all activities at the site and the Plant Manager is responsible for the safe efficient operation of the plant. The person primarily responsible for mitigation of an emergency is the Site Emergency Director. Upon declaration of an emergency the SM initially fills the position of Site Emergency Director and directs emergency response from the Control Room. This position is transferred to the TSC when that center is activated. Once the TSC is activated the Site Emergency Director and the TSC can provide technical support to the Control Room as part of their overall response to the emergency.

The minimum staffing requirements for operation are found in the plant Technical Specifications and/or FSAR. The staff responsibilities are as outlined in FSAR, and are unchanged during an emergency. The on-shift composition listed in the site-specific Appendices may be less than the minimum requirements for a period of time in order to accommodate unexpected absences of personnel provided action is taken to restore the composition within the following time requirements:

- Unit Shift Operations Staff (per Technical Specifications)
- Fire Shift Operations Staff (per Fire Protection Report)
- Security Shift Staff (per the Physical Security Plan)
- Rad Protection Shift Staff (one technician, per Technical Specifications)
(remaining staff, notified within two hours to arrive, as soon as possible)
- Chemistry and Rad Chem/Support Shift Staffing (notified within two hours to arrive, as soon as possible)
- Maintenance Shift Staff (notified within two hours to arrive, as soon as possible)

FIGURE 3-1

OFFSITE EMERGENCY ORGANIZATION



¹These offsite positions will be staffed within approximately 60 minutes.

Under emergency conditions, the normal plant staff is supplemented as shown in the site-specific appendix. The responsibilities of the personnel used to augment the normal plant operating organization are described in the site-specific appendix. Support personnel will be notified to report as required by the situation. Staffing time for the augmenting forces is indicated in the site-specific appendix. This time could vary slightly, depending upon the time of day, weather conditions, immediate availability of personnel, and radiological conditions.

The site emergency organization augments the shift operations crew. If members of the site emergency organization are not present when an emergency occurs, the Shift Manager on duty, or a designated Unit Supervisor when acting as the Shift Manager, is designated the Site Emergency Director and acts for him until relieved by the Plant Manager or his alternate.

Upon detection of a known or suspected emergency, the Shift Manager on duty refers to the site-EPIP-1 to determine the classification of the emergency. After determining the classification of the incident, the Shift Manager assumes the responsibilities of Site Emergency Director and initiates the appropriate procedure referenced by site-EPIP-1. Staffing instructions for the site emergency support centers are specified in the site-EIPs.

Site procedures shall designate site personnel who shall staff the ENS and HPN (NRC FTS 2000 System) Communication Systems. Site procedures shall designate the interface during TSC operation.

Each site will at a minimum establish the following positions within its emergency response organization with corresponding responsibilities as outlined below. The site-specific appendix gives detailed staffing and organizational data, including additional positions deemed necessary by the site.

3.1.1 Site Vice President

The Site Vice President serves as a corporate interface for the SED, relieving him from duties which could distract from the SED's primary purpose of plant operations and accident mitigation activities. The Site Vice President provides assistance to the SED by providing TVA policy direction; directing site resources to support the SED in accident mitigation activities; and providing a direct interface on overall site response activities with NRC, DHS, or other Federal organizations responding to the site, CECC Director, or onsite media.

At his discretion, he may provide an interface at the appropriate offsite location on the overall site response activities with State and local agencies, NRC region/corporate, or Joint Information Center. He also provides support to other emergency operation centers as necessary.

3.1.2 Site Emergency Director

The SED is responsible for directing onsite accident mitigation activities; consulting with the CECC Director and Site Vice President on significant events and their related impacts; protective actions; coordinating accident mitigation actions with the NRC; makes final decision on personnel entrance to radiologically-hazardous areas when Rad Protection recommends against the entry; and initiating long-term 24-hour per day accident mitigation operations.

The SED is responsible for determining the emergency classification. The SED makes recommendations for protective actions (if necessary) to the State and local agencies prior to the CECC being staffed (this responsibility can be transferred only to the CECC Director). The SED or MCR staff is responsible for making initial notifications to the appropriate state organization. The SED is also responsible for emergency dose authorizations for personnel under his direction and control (these responsibilities cannot be delegated).

3.1.3 Operations Manager

The Operations Manager is responsible for onsite operational activities; keeps the SED informed on plant status and operational problems; performs damage assessment as necessary; and recommends solutions and mitigating actions for operational problems.

3.1.4 Technical Assessment Manager

The Technical Assessment Manager is responsible for providing information, evaluations, and projections to the SED; coordinating assessment activities with the CECC; keeping the assessment team informed of plant status; assessing effluents; directing the technical assessment team; and projecting future plant status based on present conditions. Pertinent information is provided to appropriate organizations via a continuously used and monitored telephone communications hookup.

3.1.5 OSC Manager

The OSC Manager is responsible for directing the repairs and corrective actions; performing damage assessment; coordinating OSC teams and ensuring proper briefings and accompaniment by Rad Protection.

3.1.6 Radiological Protection (Rad Protection) Manager

The Rad Protection Manager is responsible for assessing inplant and onsite radiological conditions; directing the onsite Rad Protection activities; coordinating additional Rad Protection support with the CECC; recommending protective actions for onsite personnel to the SED; maintaining the offsite radiological conditions status information; coordinating assessment of radiological conditions with the CECC; maintaining the inplant radiological status boards; assisting the Maintenance Superintendent in briefing maintenance teams; assigning appropriate Rad Protection support to maintenance teams; and making final recommendation to the SED for personnel entry to radiologically hazardous environments.

3.1.7 Chemistry Manager

Chemistry Manager is responsible for coordinating assessment of effluents with the CECC; directing post-accident sampling activities; directing radiochemical lab activities; assessing effects on radwaste and effluent treatment systems.

3.2 Offsite Organization

A diagram of the Offsite Organization is provided in Fig. 3.1. Positions that must respond within approximately 60 minutes of an alert or higher declaration are indicated on the Figure.

Activation time for the CECC is approximately 60 minutes following declaration of an alert or higher classification, depending upon time of day, weather conditions, or immediate availability of personnel.

3.2.1 CECC Director

The CECC Director shall have overall responsibility and authority for ensuring adequate TVA response to affected State/Local governments in protecting the health and safety of the public.

The CECC Director shall direct and coordinate TVA emergency response; make protective action recommendations to the State; review and approve TVA press releases (excluding initial report of event); review adequacy of information to news media/public; and act as the primary point of contact for official TVA positions or recommendations.

The CECC Director shall ensure that key individuals are notified of the condition and severity of the events; information relative to the plant status, radiological impacts, and protective measures is available to emergency responders; NRC, DOE, INPO, insurance underwriters, and the appropriate Federal, State, and local agencies have been notified; points of contact for key types of information from the CECC are provided; and 24-hour/day operations are established if required.

3.2.1.1 Assistant CECC Director

An optional position that may be filled at the CECC Director's discretion to assist him in carrying out his duties. This position will be filled by a person qualified as CECC Director.

3.2.2 REP Staff Representative

Advises the CECC Director regarding all aspects of the NP-REP; confirms the CECC is set up and operating properly; assists the CECC Director in operating the CECC by evaluating, compiling, documenting, and posting data concerning the emergency situation.

3.2.3 State Communicator

Acts as TVA's primary communicator to the State. He clarifies information discrepancies and ensures pertinent information related to plant status, onsite response, and TVA dose assessment is provided to the State. He further assists in providing TVA resource assistance, provides the State with technical advice as necessary, and assists the State Liaison (a State government representative) in briefings and coordinating responses to State inquiries.

3.2.4 TVA Operations Duty Specialist (ODS)

The position of ODS is staffed seven days a week, 24 hours a day. After being notified of an emergency from a site, the ODS is responsible for making initial notification to the TVA REP Emergency Response Organization.

3.2.5 Emergency Duty Officer (EDO)

The EDO is responsible for establishing initial operation of the CECC in the event the NP-REP is activated at the Alert or higher classification. He is responsible for ensuring that all appropriate initial notifications of TVA and offsite emergency response organizations have been made for all emergency classifications.

3.2.6 TVA State Liaison

Acts as the CECC representative to the SEOC to interpret technical aspects of the emergency condition. He will inform the CECC on State problems, requests, and actions.

3.2.7 CECC Plant Assessment Manager

Maintains contact with the SED or Technical Assessment Manager and ensures that necessary support is provided. Requests assistance from other TVA organizations or NSSS vendors as needed. Provides technical support for planning and reentry/recovery operations. Ensures the CECC Director is briefed on information pertaining to plant status and any protective actions indicated for the public, based upon an assessment of plant status by the CECC and TSC assessment teams.

Ensures that periodic status reports are received from the site and are provided to the CECC Director and other TVA support organizations. Makes recommendations to the SED on actions to be considered by the site to mitigate the problem based upon the assessment of plant status by the CECC Assessment Team.

3.2.7.1 Plant Assessment Coordinator

Coordinates the plant status assessment activities in the Plant Assessment Area. Directs overall plant assessment function and reports results to the Plant Assessment Manager. The plant information needed by the coordinator and his plant assessment team is provided by a continuous telephone communications hookup with plant emergency staff.

3.2.7.2 CECC Plant Assessment Team

Will provide a periodic evaluation of plant status information for input back to the TSC and the CECC Plant Assessment Manager. Members of the CECC assessment team will draw upon their knowledge of plant information, procedures, core damage assessment, and industry analysis to evaluate the assessments provided by the site in terms of current and long-range plant conditions. They will apply their evaluation and independent assessment to provide needed data for developing any necessary protective action recommendations for the public. The CECC assessment team will serve as an engineering/operations/core damage assessment consultant for the plant and will reply to plant inquiries based on the available information. The leader will also ensure that appropriate safety parameters are selected for trending and the CECC trend boards are maintained. Maintains a detailed log of the sequence of events during the emergency. Assists the CECC with other site-related communication needs, as necessary.

3.2.7.3 Resource Support Coordinator

Will maintain communications with other NPG technical personnel to coordinate support as necessary. Will coordinate support from other TVA organizations such as legal, medical, finance, and procurement, and will coordinate requests for support from other organizations outside TVA such as equipment vendors and INPO. Will coordinate arrangements for special equipment and supplies.

3.2.7.4 Engineering Representative

Will provide a point of contact in the CECC for onsite and offsite Engineering. Will provide necessary engineering support as needed from the Engineering organization.

3.2.8 Public Information Manager

Will coordinate the decision to activate the JIC with the CECC Director, the General Manager, Client Communications, and SEOC. Will ensure the JIC Spokesperson and the JIC Information Staff are provided information to inform the public and news media about an emergency. Will inform the CECC Director of TVA's Public Information activities in response to an emergency.

Will coordinate all news release drafts with the State and Federal agencies participating at the JIC and secure approval of the CECC Director prior to making a release to the media. Will coordinate the decision to establish the JIC with the SEOC.

3.2.8.1 JIC Liaison

Responsible for contacting responding agencies and transmitting information for coordination. Will establish and maintain an information flow from the JIC or Site Communications to the CECC.

3.2.8.2 Information Writers

Gather information from the CECC officers and technical advisor and prepare written statements based on that information. Will develop information releases for the approval of the CECC Director for release to the TVA employees.

3.2.9 Radiological Assessment Manager (RAM)

Ensures that the CECC Director is briefed on matters concerning offsite and onsite radiological conditions. He provides consultation, technical assistance, and obtains additional services as may be required for plant Rad Protection and offsite environmental radiological surveys. He will ensure that radiological monitoring is conducted in the environment for all areas potentially affected by the emergency and evaluates the radiological information to determine the extent of actual or probable hazard to the public or environment. The RAM is responsible for radiation dose management, including emergency dose authorizations, for personnel under his direction and control. He provides technical support to the CECC Director for formulating protective actions for the public based on radiological conditions.

3.2.9.1 Radiological Assessment Coordinator (RAC)

Coordinates dose assessment, environs, and meteorological assessment activities in the Radiological Assessment Area (RAA). Directs the overall RAA function and communicates assessment results to the Radiological Assessment Manager. Provides protective action recommendations based on dose assessments and field measurements to the RAM. Ensures that information is provided to the TSC on dose projections, recommended offsite protective activities, environs measurements, and meteorological conditions. Coordinates requests for additional Rad Protection equipment and personnel.

3.2.9.2 Environmental Assessor

Responsible for the TVA environs monitoring and assessment activities and coordinates the TVA field monitoring effort with the appropriate State agency. Coordinates the analysis of offsite environs samples with WARL. Provides technical support for planning and reentry/recovery operations. Coordinates with Dose Assessor regarding the results of the environmental assessments. Provides environmental monitoring results to the Radiological Assessment Coordinator or RAM for formulation of protective action recommendations to the CECC Director.

3.2.9.3 Dose Assessor

Initiates and performs dose assessment activities during the radiological emergency and recovery and reentry phase. Consults with appropriate State agencies to resolve significant differences in assessments. Coordinates with Environmental Assessor regarding the predicted position, exposure levels, concentrations, and duration of radiological effluents. Provides dose assessment results to the Radiological Assessment Coordinator or RAM for formulation of protective action recommendations to the CECC Director.

3.2.9.4 Meteorologist

Responsible for evaluating meteorological data and developing forecasts which may be used for dose assessment and other emergency preparedness activities. Reviews adequacy of observed data and replaces missing or invalid observations. Makes forecasts of dispersion conditions that affect radiological effluents. Provides dispersion knowledge to dose assessment staff. Prepares other meteorological forecasts needed for emergency preparedness activities.

3.2.10 Technical Advisors

Provides technical assistance and explanation to the State Communicator, Public Information Staff, and Public Information Manager to ensure accurate information is released to the public and state agencies.

3.2.11 Boardwriter(s)

Maintains the CECC Status Boards and EPZ maps with the most current information.

3.2.12 Management Services

Makes arrangements for and provides for clerical support, food, TVA transportation services, lodging, supplies, drawings, and controlled documents. Authorized to issue checks for payment for emergency services of outside firms.

3.3 Local Support

TVA has agreements with police departments, ambulance services, and hospitals near each site to provide appropriate services as requested. (See Subsection 16.5.)

3.4 Federal Agency Support

TVA has developed an agreement (see Subsection 16.5) with DOE Radiation Emergency Assistance Center/Training Site (REAC/TS), Oak Ridge, Tennessee. Other federal support would be requested through the NRP (see Subsection 2.6).

3.5 Vendor Support

The NSSS vendor has an organization set up to provide technical support during emergency situations. Other vendor support may be procured as needed (see Subsection 16.5).

3.6 Institute of Nuclear Power Operations (INPO)

TVA maintains an agreement, (see Subsection 16.5), with INPO, a consortium of nuclear utilities and other nuclear industries, to obtain any necessary support available from the industry during an emergency.

4.0 EMERGENCY CONDITIONS

4.1 Classification System

TVA utilizes the following emergency classifications:

1. Notification of Unusual Event (NOUE)
2. Alert
3. Site Area Emergency
4. General Emergency

This system of classification is consistent with the systems used by State and local emergency organizations. The emergency classifications are graded according to severity, and immediate actions are taken to cope with the situation (see the site-specific appendix). Escalation to a higher class or termination occurs during the course of an emergency if warranted by conditions. Example of plant conditions and their recommended emergency classes are given in the specific site EIPs. These procedures also specify the initial prompt notifications, information, and recommendations to be provided to State and local emergency organizations. Examples of initiating conditions and specific instrument readings, if appropriate for the various classifications, are given in the site-specific appendix.

4.1.1 Notification of Unusual Event

This class provides early and prompt notification of events that are in process or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

The purposes of Notification of Unusual Event are: (1) to ensure that the first steps in activating emergency organizations have been carried out, and (2) provide current information on the unusual event.

The Notification of Unusual Event class is maintained until closeout or escalation to a higher class. The State authorities are notified and in turn notify the local authorities. Following closeout, State authorities are briefed, and no later than the next working day a written summary of significant events which occurred is forwarded to the State.

4.1.2 Alert

An Alert class is indicated when events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of Hostile Action. Any releases are expected to be limited to small fractions of EPA Protective Action Guideline exposure levels.

The purposes of the Alert class are: (1) to ensure that emergency personnel are readily available to respond if the situation becomes more serious or to perform confirmatory radiation monitoring, if required; and (2) provide offsite authorities current status information.

The Alert class is maintained until event termination or escalation to a higher class. The State authorities are notified and in turn notify the local authorities. Following closeout, State authorities are briefed and no later than the next working day a written summary of significant events which occurred is forwarded to the State.

4.1.3 Site Area Emergency

A Site Area Emergency is declared when events are in process or have occurred which involve an actual or likely major failures of plant functions needed for protection of the public or Hostile Action that results in intentional damage or malicious acts; (1) toward site personnel or equipment that could lead to the likely failure of or; (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

The purposes of the Site Area Emergency class are: (1) to ensure that response centers are staffed; (2) assure that monitoring teams are dispatched; (3) assure that personnel required for evacuation of nearsite areas are at duty stations if the situation becomes more serious; and (4) provide current information for, and consultation with, offsite authorities and the public.

The Site Area Emergency class is maintained until event termination or escalation to a higher class. The State authorities are notified and in turn notify the local authorities. Following closeout, State authorities are briefed and no later than the next working day a written summary of significant events which occurred is forwarded to the State.

4.1.4 General Emergency

A General Emergency is declared when events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or Hostile Action that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

The purposes of the General Emergency class are: (1) to initiate predetermined protective actions for the public, (2) provide continuous assessment of information from the site and offsite, and (3) initiate additional measures as indicated by releases or potential releases of radioactivity.

When a General Emergency is declared, TVA recommends that State and local organizations implement protective actions, as specified in the EPIPs.

The General Emergency is maintained until event termination. The State notifies local authorities unless the initial classification is General Emergency in which case TVA initially notifies the local authorities. Following closeout, State authorities are briefed and no later than the next working day a written summary of significant events which occurred is forwarded to the State.

4.2 Identification of Emergency Classes

A variety of methods must be used to identify emergency situations and to categorize them. As indicated in the site-EIPs, emergencies can be caused by natural disasters such as tornadoes or floods, hazards such as aircraft crashes, releases of toxic gases, or breaches of plant security, as well as by conditions involving plant systems directly.

Recognition of the emergency class is primarily a judgment matter for plant personnel. The initiating conditions used for recognizing and declaring the emergency class are based on specific measurable values or observable conditions defined as Emergency Action Levels (EALs). These can be combinations of specific instrument readings (including their rates of change), annunciator warnings, time periods certain conditions exist, etc. The instrument readings and parameters required for determination of these EALs are detailed in the site EIPs. These EALs are used as thresholds for determining the emergency classifications. EALs are presented in the site-specific appendix. The EALs are reviewed annually by the appropriate State.

TVA maintains the capability to assess, classify, and declare an emergency condition within fifteen (15) minutes after the availability of indication to plant operators that an emergency action level has been exceeded and promptly declare the emergency condition upon identification of the appropriate EAL.

5.0 EMERGENCY NOTIFICATION AND ACTIVATION OF PLAN

Emergency measures are developed to aid in the mitigation of emergency conditions. Emergency measures begin with the declaration of an emergency class and activation of associated emergency organizations. These measures, which will include actions for assessment, correction, and protection, are described in general terms for each emergency class in the following parts of this section. Details of these emergency measures are found in the appropriate sections of the EIPs.

When the plan is activated, certain predetermined actions are performed. Notification is carried out as shown in Figure 5-1 to alert emergency staff personnel to handle the emergency situation.

5.1 Onsite

Upon detection of a known or suspected emergency, the Shift Manager on duty will utilize the site-EPIP-1, to determine the classification of the emergency. After determining the classification of the emergency, the SED will initiate the appropriate procedures referenced by the site-EPIP-1. Each procedure referenced by site-EPIP-1, gives specific instructions on staffing the TSC, the OSC, and for notifying the State, ODS and NRC.

5.2 Offsite

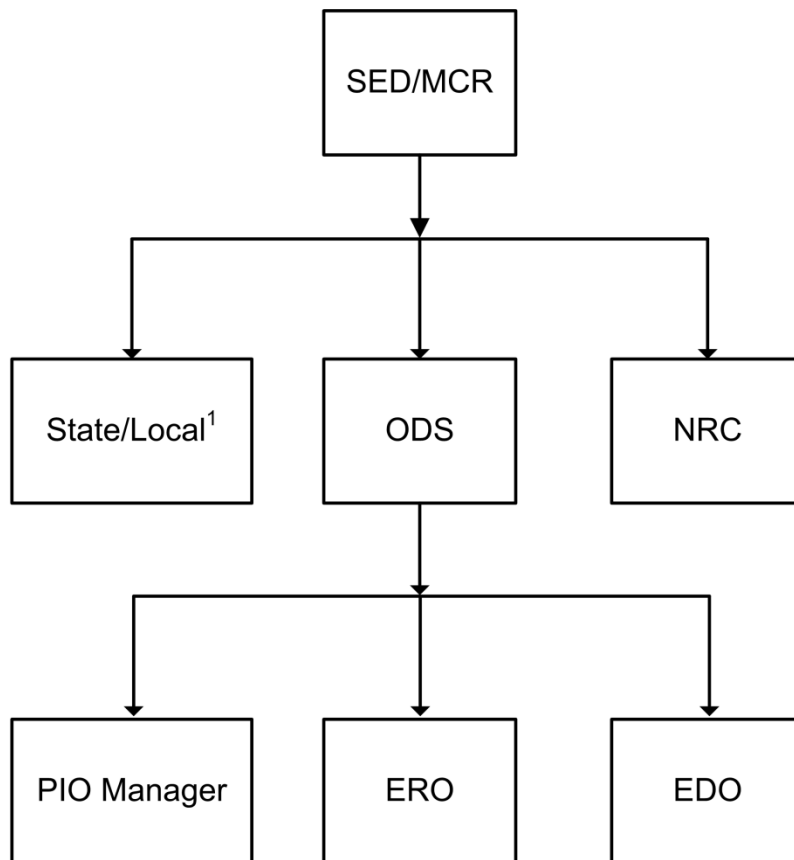
Implementing procedures are provided to activate TVA and State emergency staffs. Essential emergency positions are covered on a 24-hour-a-day basis by duty personnel.. Emergency centers are located to ensure rapid and effective response of personnel needed to assess and evaluate offsite conditions.

5.2.1 Notification of Unusual Event (NOUE)

Upon declaration of this class, the following actions are performed:

1. The ODS in Chattanooga is notified of the event by the MCR. The ODS records the details of the event in accordance with the appropriate EPIP.
2. The MCR notifies and relays the information to the State within 15 minutes of declaration of the event. The ODS notifies and relays the information to the EDO and CECC directors.
3. The EDO keeps the CECC Directors and the PIO Manager informed of the situation as necessary.
4. The PIO Manager notifies the Site Communications Consultant; General Manager, Client Communications; and TVA News Bureau (Knoxville).
5. The SED augments plant shift personnel as necessary to initiate corrective or protective actions.

FIGURE 5-1
CHAINS OF NOTIFICATION



¹ The MCR also notifies the local governments if the initial classification is a General Emergency

5.2.2 Alert

Upon declaration of this class, the following minimum actions are performed:

1. Notifications described in Section 5.2.1 are performed.
2. The CECC is staffed.
3. Environmental sampling teams may be dispatched.
4. The TSC and the OSC are activated.
5. The situation is analyzed and any appropriate corrective or preventive actions initiated.
6. Hourly, or more often as necessary, the State agencies are updated through the CECC, on appropriate plant status and environmental conditions as follows:
 - a. Class of emergency.
 - b. Type of actual or projected release (airborne, waterborne, surface spill) and estimated duration/impact times.
 - c. Estimate of quantity of radioactive material released or being released and the height of release.
 - d. Chemical and physical form of released material, including estimates of the relative quantities and concentration of noble gases, iodines, and particulates.
 - e. Prevailing weather (wind velocity, direction, temperature, atmospheric stability data, form of precipitation, if any).
 - f. Actual or projected doses at site boundary.
 - g. Projected dose rates and integrated dose at about 2, 5, and 10 miles, including sector(s) affected.
 - h. Estimate of any surface spill radioactive contamination.
 - i. Emergency response actions underway.
 - j. Request for any needed onsite support by offsite organizations.
 - k. Prognosis for worsening or termination of event based on plant information.
7. The JIC may be activated.
8. Periodic media releases are provided.
9. The SED augments plant shift personnel, as necessary, to initiate corrective and protective actions.

5.2.3 Site Area Emergency

1. Upon declaration of this class, all the actions performed in section 5.2.2 are performed.
2. Personnel knowledgeable of plant systems are dispatched to the SEOC. Upon notification, these individuals should arrive at the applicable emergency operations center within a timeframe limited only by their commuting time.
3. Any appropriate protective actions for the public are recommended to State agencies by the CECC.
4. The JIC is activated.

5.2.4 General Emergency

1. Upon declaration of this class, all the actions performed in section 5.2.3 are performed. Appropriate protective action recommendations to the State are required upon declaration of General Emergency.
2. If this is the initial classification, the MCR notifies the local government agencies within 15 minutes, and passes along the protective action recommendations.

5.3 Transportation Accidents

5.3.1 Notification by Carrier

In the event of a transportation accident involving a TVA shipment of radioactive materials, the carrier (or other person at the accident site) contacts the ODS. The carrier has procedures outlining the notifications.

5.3.2 Notification by ODS

1. State
2. EDO
3. Shift Manager of the Affected Site
4. CECC Director
5. Radiological Assessment Manager

5.3.3 CECC Director Actions

The CECC Director notifies the NRC, DOT, State authorities, ANI, and DOE (information only). The appropriate State agency, NRC, ANI, and DOE have duty officers available 24 hours a day to facilitate notification of their respective agencies.

5.3.4 Radiological Assessment Manager Actions

The Radiological Assessment Manager will dispatch a radiological monitoring team, if deemed necessary by the CECC Director or requested by the appropriate State agency. A Radioactive Material Specialist may be sent with the team. The TVA Representative at the scene will be the senior TVA person at the site of the incident.

6.0 COMMUNICATIONS

The radiological emergency communications network consists of the Emergency Preparedness (EP) telephone system, the EP paging system, and the EP radio system. These systems are designed to complement each other in the overall plan for REP communications.

The communications facilities described in the following sections are integrated with the requirements for communications to local and State response organizations. Testing is performed in accordance with established procedures.

6.1 EP Telephone System

The EP telephone system includes communications equipment installed at each site and the CECC, a number of leased commercial circuits, and privately owned circuits connecting each nuclear site to the required locations.

6.2 Plant Telephone Switching Equipment

The telephone switching equipment installed at each plant consists of one or more switching centers equipped with fully redundant common logic and redundant power sources. The majority of plant telecommunications services are served from this switching equipment. Principal system features include:

1. Critical areas served by more than one switching center.
2. Dial access to any TVA or offsite location for properly authorized personnel.
3. Dial access to Federal, State, and local emergency response organizations through redundant, diverse pathways for properly authorized personnel.
4. Radio paging access for summoning key employees wearing pagers.
5. Consistent dialing plan with other TVA locations.
6. Plant fire and medical alarm activation through dial access.
7. Executive override privilege for authorized personnel requiring the ability to interrupt conversations in progress.
8. Access to the plant or building loudspeaker paging system.

6.3 Plant or Building Loudspeaker Paging

This system(s) may be accessed from the plant telephone system and is used for normal plant operations (plant paging) and to instruct and notify personnel during an emergency. Also, plant paging executive override is provided at the unit operator's desks and the electrical control desk.

6.4 Offsite Telephone Communications

The offsite communications network is used to communicate with Federal, State, and other supporting agencies. Access to these agencies is provided through several redundant, diverse routes. This diversity provides offsite routing through more than one type of facility. These facilities include, but are not limited to, commercial facilities such as central office trunks, tie-lines and digital services, plus privately owned and maintained microwave and fiber-optic systems. The offsite telecommunications network is designed to facilitate traffic in the most fail-safe manner to the emergency response organizations. Telecommunications services are provided between the following locations in a redundant, diverse manner:

Central Emergency Control Center (CECC) to State Emergency Management Agencies.

CECC to each nuclear site.

State Emergency Management Agencies to County Emergency Management Agencies.

In addition to the above listed emergency response organizations, the following emergency centers are also equipped with public telephone lines:

Joint Information Centers.

Field Coordination Centers.

ENS and HPN (NRC FTS 2000 System) telephones provide communications from each site Technical Support Center, Control Room, and the CECC to the NRC Headquarters and regional offices. These telephones are tested on a monthly basis.

6.5 EP Paging System

The EP paging system is an automated paging system which is used to automatically page key personnel during nuclear emergencies. It is computer-activated via dedicated terminals located in the Control Room at each nuclear site and the Operations Duty Specialist's office in Chattanooga, all of which are manned 24 hours a day.

The EP paging system has provisions to periodically monitor its own performance to detect and report equipment failures.

6.6 TVA Enterprise Emergency Notification System (TEENS)

The TEENS system is a hosted ERO notification system that notifies the ERO by contacting devices such as the ERO's assigned office, mobile, and home telephones, work e-mail, and activates the ERO's assigned pager.

6.7 EP Radio System

The EP radio system is a VHF mobile radio system which provides redundant radio coverage of the 10-mile emergency zone. It provides radiological monitoring vans with mobile communications to other van(s) and to the following locations:

Radiological Control.

Technical Support Center.

Control Room at each plant.

CECC in Chattanooga.

6.8 Other Radio Communications

There is an inplant repeater system utilized by Nuclear Security Service which enables transmission without interruption to various areas of the plant. A separate radio located in the plant Central Alarm Station is a direct link to the local law enforcement officials. The plant ambulance has a radio used for communication with the local hospitals and the plant. Portable two-way radios are available for additional site communications.

7.0 PUBLIC INFORMATION AND EDUCATION

7.1 Purpose

The purpose of TVA emergency public information and education is to ensure timely distribution of accurate information during an emergency. The program also provides education to the public located within the 10-mile EPZ on emergency plans. The program also provides for TVA to coordinate emergency information with non-TVA agencies that have a primary response role prior to its release to the public or news media. A Joint Information Center (JIC) would be established under the program for use during an emergency. The purpose of the JIC is to provide a single location for TVA, local, state and Federal agencies to coordinate public information activities. On an annual, non-emergency basis, the program provides that TVA, in coordination with the state, will disseminate information to the public located within the 10-mile EPZ regarding how they will be notified and what their actions should be in an emergency. In addition, TVA and the state will conduct coordinated annual orientations to acquaint the local area news media with the emergency plans, radiological information, and points of contact for release of information in an emergency.

7.2 Responsibilities

7.2.1 CECC Director

The CECC Director or his delegate is responsible for approving written news statements after the CECC is activated.

7.2.2 JIC Spokesperson

The JIC Spokesperson is responsible for representing TVA during news briefings and coordinating information with other Federal, state, and local spokespersons prior to the briefings.

7.2.3 General Manager, Client Communications

General Manager, Client Communications is responsible for directing emergency public information activities of the agency in accordance with approved procedures. This includes the responsibility for coordinating with the CECC Director and non-TVA agencies, who would participate in JIC activities, in determining when to activate or deactivate the JIC.

7.2.4 Operations Communications

Operations Communications is responsible for the development, implementation, and maintenance of nuclear public information organizations and activities for an emergency, as well as those nuclear public information programs conducted on an annual basis.

7.3 Facilities

Information personnel at three locations: (1) Operations Communications directs the activities of the emergency public news media present at the site; (2) the CECC in the Chattanooga Office Complex where staff will develop news releases and coordinate the releases with offsite agencies; (3) the JIC where staff will coordinate with the offsite agencies in presenting emergency news briefings and respond to public telephone inquiries. The emergency public information organization shall have sufficient staff at all locations to maintain operations on a 24-hour basis.

7.4 Coordination of Information

Prior to activation of the CECC, coordination of public information with non-TVA primary response agencies will be handled through Communications in accordance with emergency public information procedures. Upon activation and staffing of the CECC the responsibility for coordination of public information with non-TVA agencies will shift to the CECC Information Staff. Upon activation and staffing of the JIC, the responsibility for coordination of public information will shift from the CECC to the JIC emergency response staff when and if offsite agencies are also operational at the JIC. The CECC Director will continue to approve written news statements. Non-TVA primary response agencies will be provided a copy of written news statements until they are available to support coordination in the JIC.

7.5 Public Education

Public education materials and programs shall be coordinated with the appropriate State agency. Public information on actions the fixed and transient populations should take in the event of an emergency shall be distributed annually. Mailing lists for the public in the 10-mile EPZ shall be updated annually to assure thorough, accurate distribution of the emergency information.

7.6 Employee Communications

A method of informing TVA employees who do not have emergency response assignments about an emergency shall be TVA Today (a computer data base information system that employees can access for written information).

7.7 Rumor Control/Public Information

Emergency information responsibilities are handled by teams in the JIC. In the JIC, a trained media relations team will respond to news media inquiries by telephone and media briefing and a trained information team will respond to citizen telephone inquiries. Also, in the JIC, a trained media monitoring team will monitor news media coverage. Information activities will be coordinated with offsite agencies at the JIC.

7.8 Training

Emergency public information staff expected to respond to an event shall be adequately trained or retrained on an annual schedule.

8.0 EMERGENCY RESPONSE FACILITIES, EQUIPMENT, AND SUPPLIES

8.1 Nuclear Site Facilities

8.1.1 Technical Support Center (TSC)

Each site will have a TSC. The TSC is an area within the plant near the control room dedicated for use during an emergency. The TSC will be the focal point of onsite activity and will be the primary source of communication from the site with offsite organizations during the event. The TSC will have sufficient staff to provide management control of the site response to the event. Equipment will be available to enable the TSC staff to communicate with onsite and offsite TVA emergency personnel. An area within the TSC will be dedicated for NRC use and will include five telephone sets and the NRC FTS 2000 System telephones. The TSC will have the same habitability as the control room. Sufficient plant parameter information will be available to the TSC to enable the TSC staff to assess the consequences of an event and assist the control room personnel in mitigating the accident. Sufficient information will be transmitted to the CECC to enable the CECC Director to make protective action recommendations to State authorities. Specific plant TSC information is provided in the site-specific appendix. Activation time for the TSC is approximately 60 minutes (90 minutes for WBN) following declaration of an Alert or higher classification depending upon time of day, weather conditions, or immediate availability of personnel.

8.1.2 Operations Support Center (OSC)

Each site will have an OSC. The OSC is a predesignated area for the assembly of personnel to support the control room operations crew during an emergency. The OSC area(s) will be under the control of the SED in the Control Room until the TSC is staffed and will provide damage assessment, maintenance and repair services, and necessary technical services. Communications will be available to the TSC. The OSC will also establish and maintain appropriate communications with any teams that may enter the plant for assessment or repair. Specific plant OSC information is provided in the site-specific appendix. Activation time for the OSC is approximately 60 minutes (90 minutes for WBN) following declaration of an Alert or higher classification, depending upon time of day, weather conditions, or immediate availability of personnel.

8.1.3 Local Recovery Center (LRC)

Each site will have an LRC. The LRC is an area predesignated for use by offsite TVA and NRC personnel that may be assigned to the site for recovery operations. In addition, the LRC may be used by the NRC during the event as an area near the site for assessment and assistance and has the capability to communicate with the TSC and offsite. The LRC will be located near the site so that personnel will have access to necessary drawings and documents. Meteorological information will also be available in the LRC.

Specific site LRC information is provided in the site-specific appendix.

8.1.4 Site Decontamination Facilities

Each site will have facilities for the decontamination of personnel including those with injuries. Information on specific site facilities is provided in the site-specific appendix.

8.1.5 Equipment, Supplies, and Supplemental Data

Each site will have sufficient equipment and supplies for the operation of the site emergency facilities. Additional seismic and hydrological information can be obtained by the CECC from other TVA nuclear plants or the TVA water quality organization.

8.2 Central Emergency Control Center (CECC)

The purpose of the CECC and associated CECC staff is to provide the facilities and manpower for evaluating, coordinating, and directing the overall activities involved in coping with a radiological emergency.

During an emergency, the CECC Director and his staff will review the response to the emergency by TVA and the appropriate State agencies to ensure that an effective and cooperative effort is being made. The CECC Director is responsible for providing TVA's protective action recommendations to the appropriate State officials.

The CECC staff will coordinate with all other TVA emergency centers to ensure an effective TVA effort in response to an accident situation. The CECC staff will also provide an accurate description of the emergency situation for TVA management and public information. In addition, the CECC will coordinate with offsite Federal agencies, such as NRC and DOE, to ensure availability of additional outside resources to TVA.

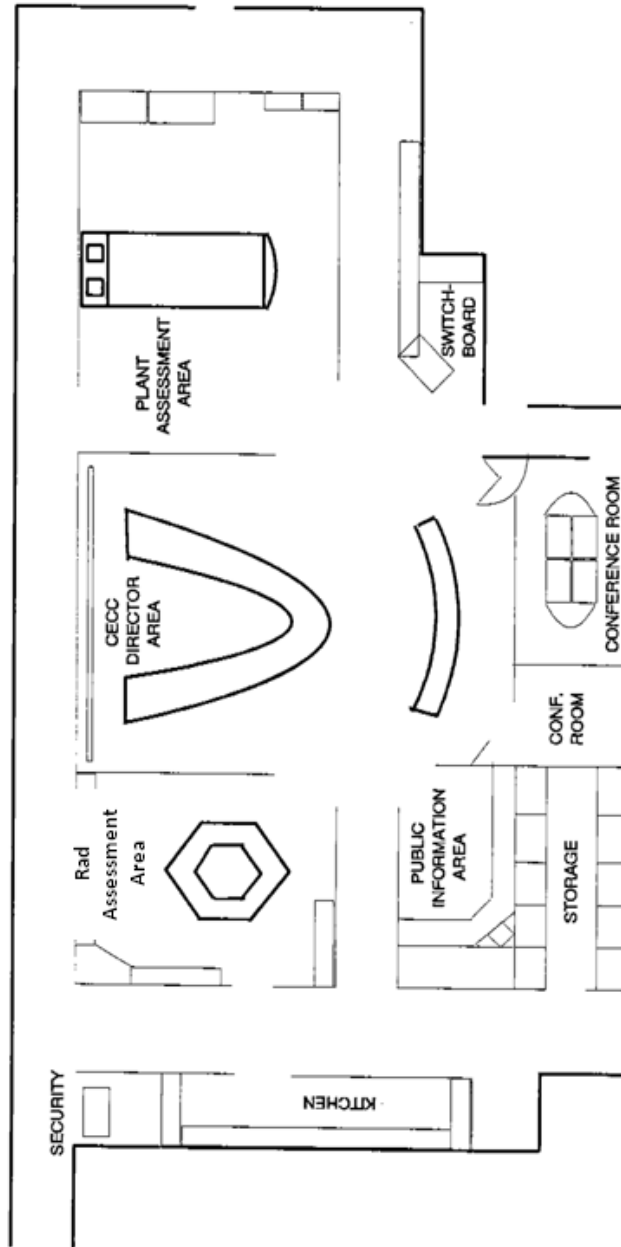
The CECC is located in the Northeast corner of the sixth floor of Lookout Place in the TVA Chattanooga Office Complex (COC) in Chattanooga, Tennessee. It is designed to house the CECC Director and his staff during an emergency situation. Included in the CECC are areas for the Plant Systems Assessment, Radiological Assessment, Information Staff, and the TVA Operations Duty Specialist (ODS). A floor plan for the CECC is provided in figure 8-1. Access control to the CECC is provided by Security personnel.

The CECC is designed to serve as the central point for information collection, assessment, and transfer during an emergency. The CECC is provided with direct communication links with State emergency response centers, other TVA emergency response organizations, the plant sites, the JIC, and offsite Federal and state organizations.

The CECC is activated during radiological emergencies. The degree of activation varies depending upon the emergency class. However, following the declaration of an Alert or higher classification, the CECC Director reports immediately to the CECC and assembles the essential CECC Staff.

Activation time for the CECC is approximately 60 minutes following declaration of an Alert or higher classification, depending upon time of day, weather conditions, or immediate availability of personnel.

FIGURE 8-1
CENTRAL EMERGENCY CONTROL CENTER



8.3 Radiological Monitoring Control Center (RMCC)

The RMCC is staffed by the TVA field Coordinator and personnel from the state. These personnel cooperate in providing direction and control of the monitoring teams.

Monitoring Teams have maps of the area and are directed to selected monitoring points or locations to collect data. This data is transmitted to the RMCC and CECC for analysis.

Facilities at the RMCC include radio and telephone communications, and necessary desks, tables, and chairs. Maps of the 10-mile EPZ and the 50-Mile EPZ are located at the RMCC.

8.4 Joint Information Center (JIC)

Each nuclear facility has a JIC. The JICs are located at:

<u>Site</u>	<u>Location of JIC</u>
Browns Ferry	Calhoun State Community College, Decatur, AL
Sequoyah	TVA-COC-Chattanooga, TN
Watts Bar	TVA-COC-Chattanooga, TN

8.5 Alert and Notification System

The alert and notification system networks consist of fixed sirens and tone-alert radios which are under the control of the state or local Emergency Management Agencies. The systems are designed to provide warning within 15 minutes to the population within 10 miles of the plants.

8.5.1 Fixed Sirens

The fixed siren component consists of electromechanical sirens. The SQN and WBN sirens are activated by the Tennessee Emergency Management Agency (TEMA). Backup activation systems are located in Hamilton County for SQN sirens and Rhea County for WBN sirens. The BFN sirens are activated by local authorities within each 10-mile EPZ county.

The siren systems are activated on a monthly basis by Emergency Management Agencies as a regularly scheduled test. A silent test is conducted every two weeks to test the radio link to the sirens. An electronic feedback system is used to monitor the performance of the sirens during the monthly tests and to ensure continuity of the activation signal path during silent tests. If the silent test is scheduled on the same date as a full test the silent test is not performed. A growl test is conducted as part of annual maintenance.

Preventative maintenance is performed by TVA annually commensurate with the manufacturer's recommendations or technical assessments. Unscheduled maintenance is performed on an as needed basis.

8.5.2 Tone-Alert Radios

The tone-alert radio component consists of radios activated by county frequencies. The radios are placed in institutions where there are concentrations of people. Preventive maintenance is performed by TVA on a annual basis commensurate with the manufacturer's recommendations. Unscheduled maintenance is performed on an as-needed basis.

9.0 ACCIDENT ASSESSMENT

9.1 Onsite

Inplant accident assessment actions are carried out by the plant emergency staff in order to properly characterize and classify the accident, determine the actual or potential radioactivity releases, and determine if there has been any effect on plant personnel or a threat to the public.

Assessment methodology consists of actions carried out through plant operating procedures as well as the site-EIPs. At the onset of an accident, plant operating procedures (normal, abnormal, and emergency) assist the plant operator and SED in identifying the cause of the accident, actions necessary to control the accident, radioactivity release rate, if any, and inplant radiation levels. The site-EIPs assist the SED in: (1) identifying and reassessing accident classification, (2) determining the need for offsite protective actions, (3) determining the need for plant area evacuation, (4) initiating activation of onsite and offsite emergency organizations, (5) directing the utilization of needed medical and/or decontamination facilities, and (6) implementing predetermined security and access control plans.

Each of the above-mentioned activities is described within the plant operating procedures or site-EIPs, as applicable, for a given situation. The distinct breakdown of assessment actions into operating procedures and implementing procedures is necessary since some assessment actions are necessarily carried out prior to identification or classification of an emergency. The procedures to ensure that accidents are properly evaluated, timely notifications are made, and assessment and protective actions are performed, are compiled in the site-EIPs. These procedures are summarized in the site-specific appendix.

Under severe accident conditions, and as required by the plant emergency operating procedures, the onsite emergency response organization is responsible for recognition of severe accident conditions, transition to, and implementation of the Severe Accident Management Guidelines (SAMG).

9.2 Offsite

TVA and State agencies are prepared to assess the consequences of potential or actual releases of radioactivity offsite. State and local agencies implement protective actions for the public. Written messages have been prepared which give the public instructions with regard to specific protective actions to be taken by occupants of affected areas. These messages are included in the State Plans referenced in appendix E.

Implementing procedures have been developed for the CECC to ensure that accidents are properly evaluated, timely notifications are made, and assessment and protective actions are performed. These procedures are compiled in the CECC-EIPs and are summarized below.

CECC-EIP-1 - CENTRAL EMERGENCY CONTROL CENTER OPERATIONS

This procedure is designed to direct the CECC Director and staff to ensure a consistent, accurate, and timely response to the events of an accident. This procedure further serves to identify the necessary information to provide for prompt, accurate public protective action recommendations to appropriate State authorities.

CECC-EPIP-2 - OPERATIONS DUTY SPECIALIST PROCEDURE FOR NOTIFICATION OF UNUSUAL EVENT

This procedure is designed to direct the ODS during a Notification of Unusual Event to ensure a consistent, accurate, and timely response in the event of an emergency.

CECC-EPIP-3 - OPERATIONS DUTY SPECIALIST PROCEDURE FOR ALERT, SITE AREA EMERGENCY, OR GENERAL EMERGENCY

This procedure is designed to direct the ODS during an Alert, a Site Area Emergency, or a General Emergency to ensure a consistent, accurate, and timely response in the event of an emergency.

CECC-EPIP-4 - NOT ACTIVE AT THIS TIME

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CECC-EPIP-5 - NOT ACTIVE AT THIS TIME

CECC-EPIP-6 - CECC PLANT ASSESSMENT STAFF PROCEDURE FOR ALERT, SITE AREA EMERGENCY, AND GENERAL EMERGENCY

This procedure is designed to direct the Plant Assessment Manager and staff to ensure a consistent, accurate, and timely response in the event of an accident. This procedure further serves to identify the necessary information which is provided to the CECC Director to ensure that prompt, accurate public protective action recommendations can be made by the CECC to appropriate State authorities.

CECC-EPIP-7 - CECC RADIOLOGICAL ASSESSMENT STAFF PROCEDURE FOR ALERT, SITE AREA EMERGENCY, AND GENERAL EMERGENCY

This procedure is designed to direct the Radiological Assessment Manager and staff to ensure a consistent, accurate, and timely response in the event of an accident. This procedure further serves to identify the necessary information which is provided to the CECC director to ensure that prompt, accurate public protective action recommendations can be made by the CECC to appropriate State authorities.

CECC-EPIP-8 - DOSE ASSESSMENT STAFF ACTIVITIES DURING NUCLEAR PLANT RADIOLOGICAL EMERGENCIES

This procedure is designed to guide Dose Assessment in obtaining necessary information, calculating doses and dose rates, developing protective action recommendations, and communicating assessment results, used in responding to radiological emergencies at nuclear power plants or arising in shipment of radioactive materials.

CECC-EPIP-9 - EMERGENCY ENVIRONMENTAL RADIOLOGICAL MONITORING PROCEDURES

The objective of this procedure is to provide guidance and instructions to the environs monitoring personnel should a radiological emergency occur at a TVA nuclear plant.

CECC-EPIP-10 - NOT ACTIVE AT THIS TIME

CECC-EPIP-11 - SECURITY OF OFFSITE EMERGENCY FACILITIES

This procedure defines CECC and JIC security requirements and specific instructions for Security personnel when the CECC or JIC is activated.

CECC-EPIP-12 - OPERATIONAL READINESS CHECK OF THE CECC AND THE FIELD COORDINATION CENTERS FOR SQN, BFN, & WBN AND JOINT INFORMATION CENTERS (JIC)

This procedure provides checklists for inventories and equipment checks for the CECC, Field Coordination Centers and Joint Information Centers.

CECC-EPIP-13- NOT ACTIVE AT THIS TIME

CECC-EPIP-14- NUCLEAR EMERGENCY PUBLIC INFORMATION ORGANIZATION AND OPERATIONS

This procedure is designed as guidance for CECC and JIC staff personnel and support personnel during an abnormal event at a TVA nuclear plant to ensure timely and accurate release of information to the public. This procedure also provides information for the activation and deactivation of the JIC and the CECC Information work area.

CECC-EPIP-15- EP FIELD SUPPORT STAFF RADIOLOGICAL EMERGENCY PROCEDURE

This procedure is designed to direct the Field Support staff in providing aquatic monitoring team data for use in protecting the public health.

CECC-EPIP-16- TERMINATION AND RECOVERY

This procedure gives guidance on event termination and transition from the Emergency Response Organization to the Recovery Organization.

CECC-EPIP-17- CENTRAL EMERGENCY CONTROL CENTER METEOROLOGIST PROCEDURES

This procedure is designed to direct the activities of the Meteorologist during a radiological emergency to provide a timely response, consistent and accurate meteorological information, and atmospheric transport and dispersion advice.

CECC-EPIP-18- TRANSPORTATION AND STAFFING UNDER ABNORMAL CONDITIONS

This procedure provides instructions for the transportation of TVA employees under certain limited circumstances. It also includes instructions for lodging and meals as necessary under those circumstances.

CECC-EPIP-19- POST ACCIDENT FUEL DAMAGE ASSESSMENT

This procedure provides a method to assess the degree of reactor core damage from measured fission product concentrations and interpretations of other plant parametric data under accident conditions. The procedure also provides guidance in obtaining necessary information to predict radionuclide releases (source term) from TVA nuclear plants during accident conditions.

CECC-EPIP-20- NOT ACTIVE AT THIS TIME

CECC-EPIP-21- EMERGENCY DUTY OFFICER PROCEDURE FOR NOTIFICATION OF UNUSUAL EVENT, ALERT, SITE AREA EMERGENCY, AND GENERAL EMERGENCY

This procedure is designed to direct the EDO in notifying key TVA organizations and contacts in the event of a Notification of Unusual Event, Alert, Site Area Emergency, or General Emergency.

CECC-EPIP-22- OPERATIONS DUTY SPECIALIST TRANSPORTATION INCIDENTS INVOLVING A SHIPMENT OF RADIOACTIVE MATERIAL

This procedure directs the ODS in obtaining information concerning a transportation accident involving radioactive material.

CECC-EPIP-23- RADIOACTIVE MATERIAL TRANSPORTATION INCIDENTS

The objective of this procedure is to provide guidance and instructions to emergency personnel concerning transportation accidents involving radioactive materials.

9.2.1 Sampling Team

TVA has vans equipped to monitor the environment for radioactivity. Each site van has an air sampler, radiation measurement equipment, a generator, radio, and other assorted equipment. A detailed listing of the minimum required equipment is available in the CECC-EIPs.

These vehicles are dispatched for environmental monitoring for Site Area Emergency and General Emergency classifications. They may be deployed for the Notification of Unusual Event and Alert classifications, if warranted. Van(s) are stationed at each site.

Each team has the capability to:

1. Obtain environmental samples for analysis.
2. Make direct radiation readings.
3. Collect air samples and analyze them for gross beta-gamma radioactivity over a range of energies.
4. Collect air samples and analyze them for radioiodine in the field, to concentrations as low as 10^{-7} microcuries/cc.

Within 30 minutes of an emergency declaration, one sampling team can be deployed from the plant for environmental assessment. Additional teams can be dispatched from other facilities. At least one additional team can be deployed within approximately one hour of notification. Composition and activation of sampling teams are described in the EIPs.

For the Site Area Emergency, and General Emergency classifications, teams are dispatched from the nearest location. They may be deployed for the Notification of Unusual Event or Alert, if warranted. If necessary, teams can be transported in a helicopter or fixed-wing aircraft.

The TSC Rad Protection Manager or CECC Environs Assessor can request assistance from a neighboring plant for environmental monitoring, if deemed necessary.

TVA has aquatic monitoring teams located at Chattanooga, Tennessee and Athens, Alabama. These teams have boats that can be deployed to obtain samples from the river for subsequent analysis for radioactivity in the laboratories.

State agencies have the responsibility to coordinate and evaluate offsite assessment actions. All environmental monitoring activities will be coordinated through the RMCC. State environmental monitoring capabilities and the RMCC operations are referenced in appendix E. TVA will be co-located in the RMCC and coordination of TVA and State monitoring teams will be conducted from that point. Environmental monitoring data will be shared between the State and TVA.

Additional environmental monitoring assistance can be obtained by contacting the DOE offices at Oak Ridge, Tennessee or Aiken, South Carolina. The EPA in Montgomery, Alabama can also provide assistance. Environmental monitoring teams and mobile radioanalytical laboratories can be supplied. The State agencies usually request and coordinate these services.

9.2.2 Analyzing Environmental Samples

Samples obtained by the sampling teams may be sent to the WARL for analysis. The WARL has the capability to perform further quantitative and qualitative analysis. WARL is available at all times and can be operated 24 hours per day. A central point for receipt of samples will be established when needed.

9.2.3 Meteorological Information

9.2.3.1 Primary Meteorological Measurements

The meteorological measurements program is designed to conform to the intent and guidance of Regulatory Guide 1.23. Wind direction, wind speed, and air temperature are measured at three levels. The temperature difference is used to estimate the Pasquill stability class. Precipitation and dew point temperature are also measured. Hourly and 15-minute average meteorological data from the plant Environmental Data Station are available to the CECC, TSC, State, and LRC. More specific information on the meteorological measurements program can be found in the site-specific FSAR.

9.2.3.2 Backup Meteorological Data Estimation Procedures

TVA has prepared objective backup procedures to provide estimates for missing or garbled data needed to perform dose calculations and to determine transport estimates. They incorporate available onsite and offsite data. Each procedure has an accompanying statement of reliability.

9.2.3.3 Real Time and Forecast Meteorological Data

The CECC Meteorologist has the responsibility for providing meteorological information to CECC Staff. The dose assessors use this meteorological information to project offsite doses. Plume positions are plotted on a site area map. The meteorological support actions and projection of doses are discussed in detail in CECC-EIPs. Meteorological support may be provided in the CECC or from a remote location.

9.2.3.4 Remote Access of Meteorological Data

Access of up to the most recent 168 hours of 15-minute and hourly meteorological data is available to authorized users through the CECC computer. The remote access system gathers data from TVA nuclear plants, performs unit conversion, reformats data, and flags questionable values.

9.2.4 Dose Assessment

On-shift dose assessment capability is maintained at the sites that can be implemented (if needed during the initial phase of an accident) until the CECC is activated and assumes the dose assessment function.

Offsite doses from accidental releases of radioactivity are estimated using a combination of calculations, field measurements and laboratory analyses of environmental samples. Data on meteorological conditions are used in determining offsite dispersion factors. Using plant operational data, field measurements, and effluent monitor readings, actual or potential releases of radioactivity are analyzed by the plant staff, the Radiological Assessment Staff, and/or the CECC Plant Assessment Team to generate or modify a source term for use in the dose assessment.

With this information, the CECC dose assessment team can predict offsite doses through the use of several models and/or methods described in the CECC-EIPs. These models provide a means of estimating public exposures throughout the emergency and recovery period. Environs measurements are used, to the extent possible, to confirm doses projected by modeling.

A preliminary dose projection is performed following receipt of measured effluent release data (the source term) and meteorological data. The preliminary dose projection is followed up by a more detailed assessment using computerized dose models. Manual dose assessment methods are available for use in the event that the computer is unavailable. Input to the detailed calculations includes measured source terms, projected future releases, near real-time and forecast meteorological data, field measurements of exposure rates and/or airborne radioactivity in the environs around the plant, or a combination thereof. Field measurements are used to estimate doses, and (especially in the case of an unmonitored release) source terms, and to verify doses projected using models.

After termination of accidental releases to the atmosphere, integrated doses are calculated to assist in recovery/reentry operations. A combination of inputs including results from modeling field exposure rate and air concentration measurements, and laboratory analyses of soil, vegetation, and water samples are used to assess doses. Recommendations are made regarding evacuation sector clearance and reentry based on doses calculated for exposure from ground contamination, inhalation of re-suspended radioactivity, and ingestion of radioactivity in vegetables and milk.

Dilution factors are predicted for radioactive discharges into the river. From this information, concentrations of radioactive material in the river downstream can be predicted and sampling locations identified. Dose calculations are also performed for individuals' drinking water from downstream water supplies.

9.2.5 Transportation Accidents

TVA emergency teams can be dispatched by land vehicle, helicopter, or fixed-wing aircraft to assist in assessing and controlling the situation. The response of emergency teams is decided by the CECC Director.

Appropriate methods described in section 9.2.4 can be applied in assessment of radioactive releases resulting from transportation accidents.

10.0 PROTECTIVE RESPONSE

10.1 Onsite Protective Actions for Radiological Events

In the event of an unplanned significant release of radioactivity or sudden increase in radiation levels, it is the responsibility of the SED to make the decision concerning the necessity for building and area evacuation. In arriving at this decision, the primary consideration is personnel safety. The various radiation and airborne radioactivity monitors placed throughout the plant, with readout in the control room, indicate the extent of the radiological hazards and may be utilized by the SED to determine the extent of evacuation necessary.

The assembly/accountability alarm is used to initiate the assembly of all site personnel. The public address system is used if only specific areas are to be evacuated. Security personnel will patrol the area between the security boundary described in the physical security plan and the site boundary and will evacuate any nonessential personnel.

Upon hearing the emergency siren, all persons in the plant areas will go to their pre-assigned areas to be accounted for and await further instructions from the SED. The pre-assigned areas are designated in approved procedures. Predetermined assembly areas are identified in approved procedures and radiological surveys will be made as required by the TSC. The number of unaccounted individuals should be available within approximately 30 minutes for persons within the security area as defined in the Physical Security Plan.

If only a particular area is cleared, personnel in that area will evacuate to a safe area. An accountability report is made to the SED. Further details of evacuation procedures are described in the site-EIPs.

If radiation levels or airborne radioactivity at an assembly point is significantly higher than alternative assemble areas, or the SED deems it necessary, the SED will order relocation to a safe assembly point. Employees will be released from this assembly point when the SED determines it is suitable.

Procedures require that all potentially contaminated people and vehicles pass through a Rad Protection check-point for survey prior to being released.

In the event of the evacuation of nonessential site personnel, the SED will notify the CECC Director. If the personnel require transportation and sheltering, the CECC Director will coordinate arrangements with the appropriate State agency. If the evacuees require radiological decontamination, they will be informed of transportation, sheltering, and decontamination arrangements prior to leaving the plant site. An alternate decontamination facility is specified in the site-EIPs.

All contaminated personnel will be decontaminated to the limits specified in the site Radiological Control Instructions (RCI's) by methods described in the site instructions before being released by TVA. Additional clothing is available onsite if required.

Procedures also specify the action to be taken by, and the accountability of, personnel having an emergency assignment. Essential plant personnel remaining onsite are protected by plant systems designed to provide a habitable environment even under the most serious accident conditions or by precautionary measures such as the use of respiratory protective equipment and protective clothing. Personnel doses are controlled in accordance with section 11.0.

10.2 Onsite Protective Actions for Hostile Action Events

A range of protective actions to protect onsite personnel during a Hostile Action event have been developed to ensure the continued ability to safely shut down the reactor and perform the functions of the emergency plan. This range of protective actions are contained in site specific abnormal procedures. These site-specific procedures are classified security sensitive.

10.3 Offsite

Should an event be initially classified as a General Emergency, the SED has the responsibility to determine an initial protective action for recommendation to State and local government agencies. A logic diagram is provided in the site-EIPs as a decisional aid to facilitate this recommendation. These diagrams provide the site specific information contained in the CECC logic diagram (Figure 10-1).

After the CECC is staffed, the responsibility for PARs is transferred to the CECC Director. The CECC Plant Assessment Manager will provide an assessment of actual and projected plant conditions. The Radiological Assessment Manager will provide an assessment of actual and/or projected radiological conditions offsite. The Radiological Assessment Manager will provide a recommendation for a specific protective action. The CECC Director will evaluate the recommendation from his staff and make a recommendation to the State. The logic diagram for plume exposure pathway recommendations is provided in Figure 10-1 and in the CECC-EIPs as a decisional aid to facilitate the recommendation. The State and local agencies are responsible for implementing actions to protect the health and safety of the public offsite. Although TVA may recommend protective actions to these agencies, the State and local governments are responsible for deciding if any actions are needed and what they should be. The CECC will discuss and provide ingestion pathway recommendations (i.e., agricultural) and recommendations for liquid releases (i.e., closing of public water supplies) with the state as appropriate.

The decision to implement one or more of the above actions is based upon some or all of the following considerations:

1. Projected offsite integrated doses.
2. Actual measured dose rates.
3. Present and future weather conditions.
4. Projected improvement or deterioration of plant conditions.
5. State protective action guides.
6. Levels of airborne radioactivity.
7. Levels of waterborne radioactivity.
8. Concentrations of radioactivity in items for human consumption.
9. Evacuation time estimates (from Evacuation Time Estimate Manual) or maintained in the appropriate state plan).

FIGURE 10-1
Page 1 of 2

INITIAL PROTECTIVE ACTION RECOMMENDATION FLOWCHART

Note 1: IF flowchart decision block conditions are unknown **THEN** answer NO.

Note 2: State agencies have provided prior knowledge of offsite impediments to evacuation (such as flooding, bridge/road closures, etc.) and recommend that any needed PAR should be shelter(Rec #3)

Note 3: A short term release is defined as "a release that does not exceed a 15 minute duration"

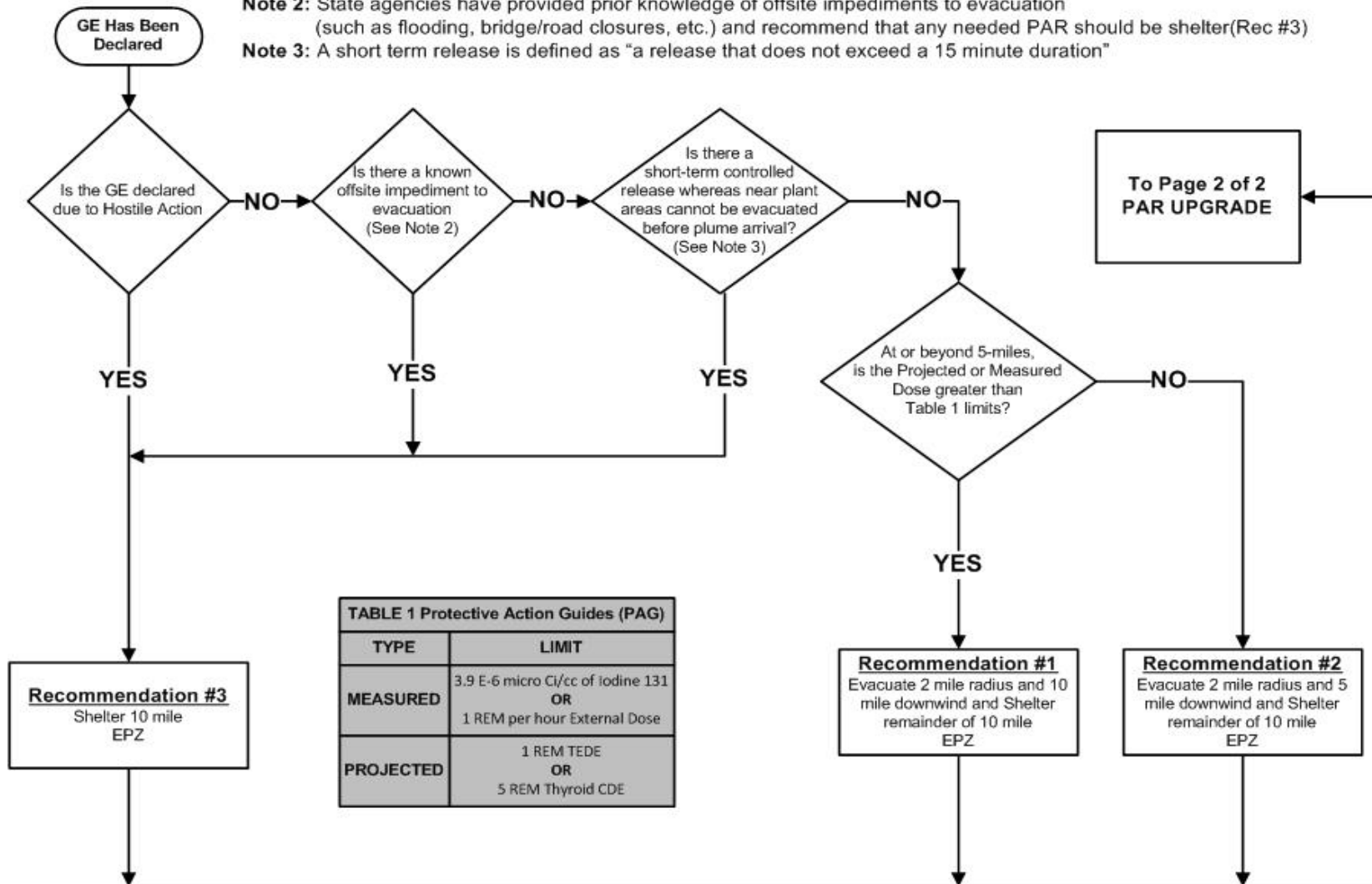


FIGURE 10-1
Page 2 of 2

PROTECTIVE ACTION RECOMMENDATIONS(PAR) UPGRADE

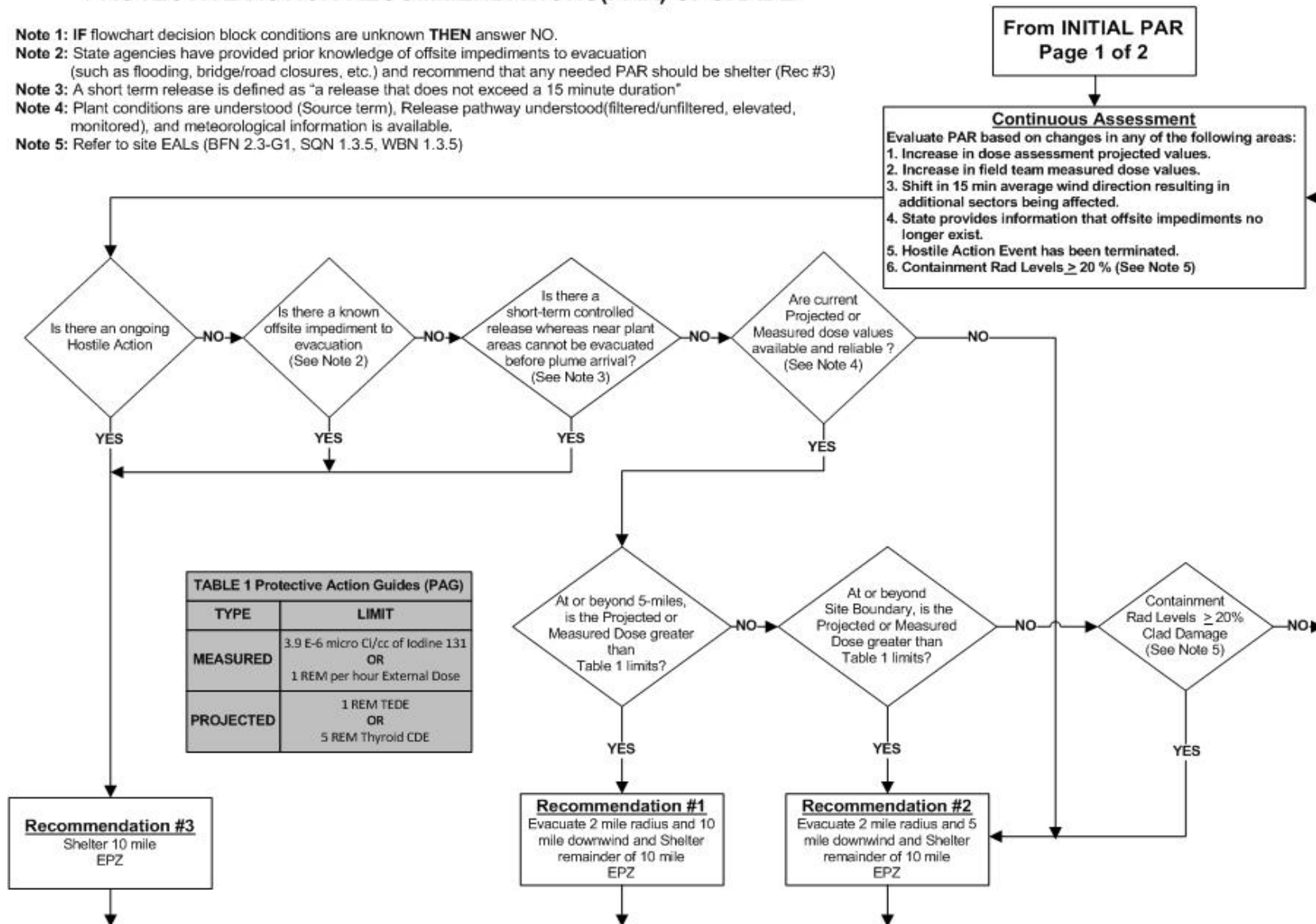
Note 1: IF flowchart decision block conditions are unknown **THEN** answer NO.

Note 2: State agencies have provided prior knowledge of offsite impediments to evacuation (such as flooding, bridge/road closures, etc.) and recommend that any needed PAR should be shelter (Rec #3)

Note 3: A short term release is defined as "a release that does not exceed a 15 minute duration"

Note 4: Plant conditions are understood (Source term), Release pathway understood(filtered/unfiltered, elevated, monitored), and meteorological information is available.

Note 5: Refer to site EALs (BFN 2.3-G1, SQN 1.3.5, WBN 1.3.5)



11.0 RADIOLOGICAL PROTECTION

The Radiological Protection Section at the site is responsible for all Radiological Protection activities onsite. Its function is to develop instructions to implement the requirements of Title 10 Code of Federal Regulations, Part 20, and other required standards as well as the requirements and policies of TVA NPG SPP-5.1, "Radiological Controls." The section provides surveillance during normal operation as well as emergency situations. In addition, the section advises key plant personnel on radiological matters for routine and emergency conditions.

The limiting doses to occupational workers during routine plant operations are found in TVA NPG SPP-5.1, and the site Radiological Control Instructions (RCIs). If possible, these limits will be employed during emergency operations. If these standards cannot be met during emergencies, the dose limits described in figure 11-1 will be used. The site-EIPs describe the methods to use and authorizes the doses outlined in figure 11-1. Figure 11-2 describes the health effects or radiation doses greater than 25 RAD.

For all individuals entering radiation work permit areas, electronic dosimeters and primary dosimeters are issued and read in accordance with the site RCIs. The electronic dosimeters can be read at any time. Primary dosimetry processing and evaluation is performed by an organization currently accredited by the National Voluntary Laboratory Accreditation Program (NVLAP) of the National Institute of Standards and Technology for the type or types of radiation that most closely approximates the type of radiation or radiations for which the individual wearing the dosimeter is monitored. Dose records are maintained on each monitored individual by a computer.

TVA NPG SPP-5.1 contains TVA's criteria used to establish contamination zones and to release personnel, equipment, and clothing. Onsite facilities are available to decontaminate equipment and personnel.

Procedures for using individual respiratory protection and protective clothing are provided in specific plant operating procedures. Procedures for the use of radioprotective drugs are provided in the EIPs. Drinking water and eating controls are established by Radiation Protection.

FIGURE 11-1

EMERGENCY WORKER DOSE GUIDANCE

<u>TEDE Dose</u>	<u>Condition</u>
5 rem	All, maintain dose ALARA
10 rem	Protection of valuable property when lower dose not practicable.
25 rem	Lifesaving or protection of large populations when lower dose not practicable.
Greater than 25 rem	Lifesaving or protection of large populations when lower dose not practicable. Only on a volunteer basis to persons fully aware of the risks involved.

NOTE: Situations may occur in which a dose in excess of regulatory limits (10 CFR 20.1201) would be required for plant and lifesaving operations. It is not possible to prejudge the risk that one person should be allowed to take in these situations. However, persons undertaking an emergency mission in which the dose would exceed regulatory limits should do so only on a voluntary basis and with full awareness of the risks involved (EPA-400).

Guidance for dose to the lens of the eye is three (3) times the listed TEDE value. Dose to any other organ (including skin and body extremities) is ten (10) times the listed TEDE value.

Authorizations for emergency dose limits for onsite personnel will be provided by the SED while authorizations for offsite personnel will be provided by the CECC Radiological Assessment Manager.

In all cases, adequate protective measures shall be provided so that dose, considering both internal and external pathways, will be maintained As Low As Reasonably Achievable (ALARA). Internal dose should be minimized by the use of respiratory protection equipment consistent with maintaining the TEDE ALARA and protective clothing should be used to minimize personnel contamination. If a projected dose to a worker's thyroid is expected to exceed 10 rem during a radiological emergency, Potassium Iodide (KI) should be issued, in accordance with applicable implementing procedures.

Personnel shall not enter any area where dose rates are unknown or unmeasurable with either instruments or available dosimetry.

Receipt of emergency exposures in excess of 10 CFR 20.1201 limits shall be on a voluntary basis. Personnel receiving emergency exposures shall be informed of the risks involved, (EPA-400) including the numerical levels of dose at which acute effects of radiation will be incurred, and numerical estimates of the risk of delayed effects. Figure 11-2 provides information consistent with EPA-400, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," which may be useful for this briefing purpose.

Personnel receiving emergency doses should be restricted from further occupational exposure pending the outcome of exposure evaluations, and if necessary, medical surveillance.

Any personnel dose in excess of five (5) rem TEDE shall be handled in accordance with the TVA Nuclear Radiological Protection Plan.

FIGURE 11-2

HEALTH EFFECTS OF RADIATION DOSES GREATER THAN 25 RAD

I. Health Effects Associated with Whole Body Absorbed Doses Received Within a Few Hours ¹.

Whole Body Absorbed Dose (rad)	Early Fatalities ² (percent)	Whole Body Absorbed Dose (rad)	Prodromal Effects ³ (percent)
140	5	50	2
200	15	100	15
300	50	150	50
400	85	200	85
460	95	250	98

1 Risks will be lower for protracted exposure periods.

2 Supportive medical treatment may increase the dose at which these frequencies occur by approximately 50 percent.

3 Forewarning symptoms of more serious health effects associated with large doses of radiation.

II. Approximate Cancer Risk to Average Individuals from 25 Rem Effective Dose Equivalent Delivered Promptly.

Age at Exposure (years)	Risk of Premature Death (deaths per 1,000 persons exposed)	Average year of life lost if premature death occurs (years)
20 to 30	9.1	24
30 to 40	7.2	19
40 to 50	5.3	15
50 to 60	3.5	11

Note: Tables referenced from the Environmental Protection Agency's "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," (EPA-400), May 1992, page 2-12.

12.0 MEDICAL SUPPORT

Facilities, equipment, medical supplies, and trained personnel are available for first aid/emergency medical treatment of ill or injured persons onsite.

Guidance for medical assistance is found in the site-EIPs. Immediate lifesaving and disability limiting procedures takes precedence over noncritical decontamination and dosimetry assessment measures.

The care, disposition, and reporting of all injuries known or suspected to be associated with excess levels of radiation exposure or contamination are coordinated with the CECC when activated. The purpose of the medical emergency response team (MERT) (team composition specified in the site procedures) is to:

1. Provide first aid/emergency medical treatment for ill or injured persons onsite, including those who may have been exposed to or contaminated with radioactive material.
2. Minimize injury during the rescue, treatment, and transport of injured persons, while minimizing radiological hazards and exposure to the victim.
3. Advise and protect attending personnel from unacceptable and unnecessary radiological hazards and exposures.
4. Identify, document, and control radiation exposure and contamination hazards associated with the emergency.

12.1 Classification and Handling of Medical Emergency Patients

12.1.1 Noncontaminated-Nonirradiated

When it is known that the patient is not contaminated and has not been overexposed to radiation, he is handled according to standard first aid/emergency medical protocol. The patient, ambulance crew, receiving hospital, and attending physician (as applicable) are advised of the absence of radiological complications.

12.1.2 Irradiated-Noncontaminated

The patient is removed from the source of radiation exposure as soon as medical conditions and essential treatments permit. Continued medical care for physical injuries including ambulance transport is provided as indicated. Rad Protection determines and reports radiation exposure levels including affected body areas. Emergency care for the radiation exposure is governed by the dose assessment and the medical status. Involved personnel are advised of the absence of radiological contamination.

12.1.3 Contaminated

Patients known or suspected of being contaminated are provided essential first aid/emergency medical care. Decontamination activities are accomplished as the medical status permits. Involved personnel are advised of the contamination hazard. Continued care and decontamination decisions are made on an individual basis by the responsible medical care provider and Rad Protection.

12.2 Transportation of Injured Personnel

The decision to transport a patient offsite shall be the responsibility of the emergency medical care provider performing patient assessment, i.e., EMT or RN. If conflicting decisions arise, the option which provides the patient with the optimal level of medical care shall be chosen.

When ambulance transportation is indicated, transport may be provided by the site Fire Protection EMTs (using a TVA ambulance) or by an agreement ambulance service. The MERT Team Leader will coordinate any request for offsite ambulance assistance through the SM. The SM will perform initial requests, notifications for assistance.

Arrangements have been made for one or more agreement ambulance services for each nuclear facility, with trained personnel to transport patients, including those who may have been exposed to or contaminated with radioactive material. These services are designated in the site-EPIPs and letters of agreement for response are maintained. (See Section 16.5.)

12.3 Local Hospital Assistance

Arrangements have been made for one or more receiving hospitals for each nuclear facility. These agreement hospitals have adequate equipment and trained personnel to care for ill and injured persons, including those who might have been exposed to or contaminated by radioactive material. Initial notifications are performed by the SM. Hospitals for each site are designated in site EPIPS and letters of agreement are maintained. (See Section 16.5.)

12.4 Interagency Assistance from REAC/TS

Arrangements have been made for assistance from the Radiation Emergency Assistance Center/Training Site (REAC/TS). REAC/TS is a DOE-sponsored facility operated by Oak Ridge Associated Universities Medical and Health Sciences Division in cooperation with the Oak Ridge Methodist Medical Center in Oak Ridge, Tennessee. Specialized facilities and expert personnel are available, after consultation, for backup definitive care for radiation accident victims. A letter of agreement for services is maintained. (See Section 16.5.)

13.0 TERMINATION AND RECOVERY

Most emergencies will not require long-term recovery operations. In those cases where recovery operations are indicated, the following guidelines will be used to establish the recovery phase. Recovery operations will vary greatly depending upon the circumstances of the emergency situation. Criteria and procedures will be developed as required considering maximum protection for plant personnel and the public.

13.1 Termination

The decision to terminate an event for which the onsite and offsite emergency centers have not been activated will be made by the SED/SM.

The decision to terminate and/or enter recovery from an incident for which onsite and offsite emergency centers have been activated will be made by the SED after consultation with the plant technical and operations staffs and will be coordinated with the CECC Director. This decision will be based upon a comprehensive review of plant status and system parameters. These shall include, but not be limited to, the following:

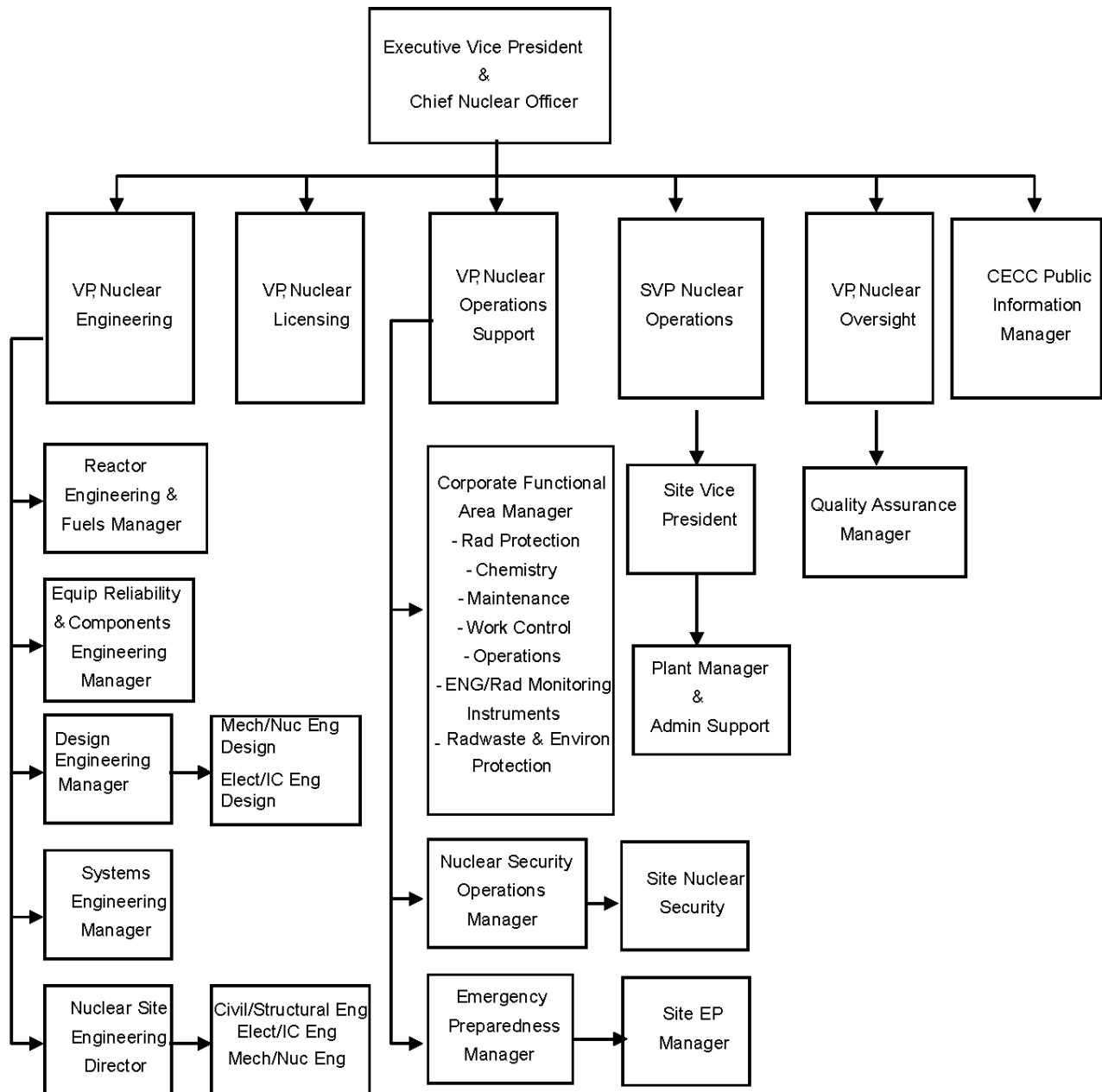
1. Stability of the reactor shutdown condition, i.e., successful progress toward a cold shutdown condition.
2. Integrity of the reactor containment building.
3. Operability of engineered safety systems and decontamination facilities.
4. The availability and operability of a heat sink.
5. The integrity of power supplies and electrical equipment.
6. The operability and integrity of instrumentation including radiation monitoring equipment (also including portable equipment assigned during the emergency).
7. Availability of trained personnel and support services.
8. Control of radiological effluent releases.

Decisions to relax protective actions for the public will be made by the appropriate State representatives. The CECC Director will provide information to the appropriate State agencies to facilitate the decision. The State has the authority and responsibility for offsite recovery efforts. TVA will provide assistance, as requested, through the recovery organization shown in Figure 13-1.

The CECC Director, after consultation with the state, the SED, and NRC (if appropriate) will announce that the emergency has terminated and the recovery phase is to be initiated if appropriate. Procedures and plans shall then be drawn up to implement the most expeditious recovery sequence to return the plant to normal operation.

FIGURE 13-1

TVA RECOVERY ORGANIZATION



13.2 Recovery Organization

13.2.1 Executive Vice President & Chief Nuclear Officer - Will direct the overall recovery effort. If the recovery phase is expected to be a long-term process, he may form a team to be responsible for continuous control of the recovery operation, thus permitting other personnel to return to their normal duties. The organizational structure of such a team would be contingent upon the emergency situation and procedures required for recovery. The LRC is available to provide additional office space near the site for the recovery team at the discretion of the Executive Vice President & Chief Nuclear Officer.

13.2.2 Senior Vice President Nuclear Operations - Responsible for the onsite recovery effort and for developing required recovery procedures. May request any needed offsite support through the Site Vice President and Plant Manager.

13.2.3 Site Vice President (SVP) - Responsible for coordinating the onsite efforts with the overall TVA recovery effort. SVP will be in charge of the LRC should additional office space be needed.

13.2.4 Vice President, Nuclear Operations Support - Corporate Functional Area Managers will provide support in the following areas:

- Radiological Protection
- Chemistry
- Maintenance
- Work Control
- Operations
- Radwaste & Environmental Protection
- Emergency Preparedness
- Nuclear Security
- Eng/Rad Monitoring Instrument
- Technical support and Environs sampling assistance as requested by the State
- Dose Assessment
- Environmental Monitoring

13.2.5 CECC Public Information Manager - Acts as an interface between TVA and the news media. They assist the Chief Nuclear Officer & Executive Vice President in drafting news releases concerning progress of the recovery operation. They coordinate all news releases with TVA management and State and Federal officials as required. They coordinate all press briefings and interviews concerning the incident.

13.2.6 Vice President of Nuclear Licensing - Provides support in various licensing issues including primary interaction with the NRC

13.2.7 Vice President of Nuclear Engineering - Provides support in the area of plant components, Reactor Engineering, fuels and in various engineering disciplines.

13.2.8 Vice President of Nuclear Oversight - Provides Quality Assurance oversight of the recovery process.

13.2.9 Other Resources - All other TVA resources plus other governmental and vendor support will be available through the TVA corporate organization to aid the Site Emergency Director in developing, evaluating, and implementing specific site recovery and reentry operations.

13.3 Onsite Recovery

All major post-incident onsite recovery measures shall be performed in accordance with written procedures. Some procedures which may be developed following an incident include the following activities.

1. The first auxiliary/reactor building entry.
2. The first containment building entry.
3. Damage evaluation.
4. Decontamination.
5. Disassembly.
6. Repair.
7. Disposal.
8. Test and startup of restored facilities.

Appropriate personnel protective measures will be taken on initial entries and throughout assessment and recovery operations to limit exposures to that outlined in section 11.0.

Reentry and recovery individual and population dose estimates may be obtained using dose rate measurements or calculations and population distribution (see section 9.2.4). The CECC-EIPs contain this methodology.

13.4 Local Recovery Center (LRC)

The purpose of the LRC is to provide a facility for TVA recovery management as well as NRC emergency response personnel and other emergency and/or recovery personnel.

The LRC provides adequate space for TVA and others who may locate there to support the site should additional office space near the site become necessary during the recovery phase.

The LRC will provide dedicated space for NRC personnel containing adequate supplies, communications, and data necessary for them to carry out appropriate functions. See the site-specific appendix for the description.

13.5 Offsite Recovery

The State has the authority for actions taken offsite; however, TVA will serve as an important source of technical and analytic assistance for the State in offsite monitoring and sampling needed to determine the extent and methods of offsite recovery. The Chief Nuclear Officer & Executive Vice President, or his designee will serve as the State's contact for coordination of TVA's efforts in offsite monitoring, sampling, and recovery.

14.0 DRILLS AND EXERCISES

14.1 Drills

Drills are conducted to develop and maintain key skills required for emergency response. These drills may be conducted individually or as part of an REP exercise.

The following drills are required:

14.1.1 Medical Emergency Drills

A medical emergency drill involving a simulated contaminated/injured individual, with participation by a TVA or agreement ambulance and each agreement hospital (see Section 16.5), shall be conducted each calendar year for each plant. Scenario development, drill activities, and evaluations are jointly conducted and critiqued by EP and the site.

14.1.2 Radiological Monitoring Drills

Environmental monitoring van drills shall be conducted each calendar year for each plant. These drills include collection and analyses of sample media (i.e., water, air, grass, and/or soil as may be required by the scenario), direct radiation measurements, operation of vehicles, communication equipment, sampling equipment, and recordkeeping. The scenario is developed and the drill conducted and critiqued by the site or EP.

14.1.3 Rad Protection Drills

Rad Protection drills will be conducted twice each calendar year for each plant involving response to, and analysis of, simulated elevated airborne samples and direct radiation readings in the plant. The scenario is developed and the drill conducted and critiqued by the site.

14.1.4 Radiochemistry Drills

Drills shall be conducted each calendar year at each plant to collect and analyze inplant liquid and gaseous samples containing actual or simulated elevated levels, including use or simulated use of the post accident sampling system. The scenario is developed and the drill conducted and critiqued by the site.

14.1.5 Radiological Dose Assessment Drills

Dose assessment drills are conducted at least twice each calendar year to test the procedures, calculation techniques, computer codes, and environmental assessment abilities of the CECC staff and support groups.

These scenarios are developed and the drill conducted and critiqued by EP.

14.1.6 Fire Drills

Fire drills are conducted at each plant in accordance with and as required by specific procedural requirements.

14.1.7 Communications Drills

Communications drills are conducted at least once each calendar year for each site.

14.2 Exercises

14.2.1 Requirements

Exercises shall be scheduled and conducted such that:

1. A biennial exercise shall be conducted for each site, with at least partial participation by the State, to test the REP every 2 calendar years.
2. Each site will ensure that adequate emergency response capabilities are maintained during the interval between biennial exercises by conducting drills, including at least one drill involving a combination of some of the principal functional areas of the onsite emergency response capabilities. The principal functional areas of emergency response include activities such as management and coordination of emergency response, accident assessment, protective action decision making and plant system repair and corrective actions. During these drills, activation of all of the emergency response facilities is not necessary. Sites have the opportunity to consider accident management strategies, supervised instruction is permitted, operating staff have the opportunity to resolve problems (success paths) rather than have controllers intervene, and the drills can focus on onsite training objectives. Sites shall enable the states and local authorities to participate in such drills when requested.
3. An exercise shall be conducted for each site, with full participation by State and local authorities, every two years. Where a State has more than one site it shall participate fully every two years at some site and partially participate at the other sites offsite exercises.
4. An exercise shall be conducted for each site such that the State may exercise emergency plans related to ingestion exposure pathway measures every six years. Where a State has more than one site, this participation should be rotated between sites.
5. All major elements of the emergency plans and organizations shall be tested within a six-year period.
6. Each site will initiate an exercise between 6:00 p.m. and 4:00 am at least once every six years.
7. The exact time of the exercise shall be unannounced.

14.3 Scenario

Drills and exercises shall be conducted in accordance with scenarios that have been properly planned, researched, and developed.

The drill and exercise scenarios shall include, but not be limited to, the following:

1. The basic objectives of each drill or exercise.
2. The date(s), time period, place(s), and participating organizations.
3. The simulated events.
4. A time schedule of real and simulated initiating events.
5. A narrative summary describing the conduct of the exercises or drill, including simulated casualties, offsite fire department assistance, rescue of personnel, use of protective clothing, deployment of radiological monitoring teams, and public information activities.

Drill scenario development and implementation shall be the responsibility of organization responsible for the specific drill.

Exercise scenario development and implementation shall be the responsibility of Emergency Preparedness (EP). Exercise scenario planning and development will be coordinated with representatives of appropriate organizations and State agencies. Scenario specifics shall not be released by those representatives prior to the exercise.

Exercise scenarios will be developed to thoroughly test the REP on a six year cycle. The exact time of an exercise shall not be released; however, a time span within which the exercise is to occur may be supplied to appropriate organizations and the news media so that the exercise is not confused with an actual emergency.

In the event a remedial exercise is required a scenario will be developed to demonstrate corrective measures have been taken regarding the described deficiencies.

14.4 Critiques

Representatives of Quality Assurance, INPO, NRC, DHS, State/local agencies and others may observe the exercise. Additional evaluators may be requested from other organizations as necessary. Evaluators will be provided with sufficient material and a briefing prior to the exercise to become familiar with the emergency plan and exercise scenario.

At the conclusion of each exercise a critique shall be conducted where the exercise and its participants will be evaluated for effectiveness, procedural compliance and good practices. EP shall evaluate critique comments, develop a formal written report, coordinate corrective actions for deficiencies or items needing improvement, and follow up to ensure completion of corrective actions.

Drill critiques, critique reports, coordination of corrective action and follow-up to ensure completion shall be the responsibility of the organization administering the drill.

15.0 TRAINING

Personnel with specific duties and responsibilities in the NP-REP shall receive instruction in the performance of these duties and responsibilities.

15.1 Onsite

Site Nuclear Training/line organizations/site EP will provide training in emergency procedures to all permanent plant personnel and applicable non-plant personnel in accordance with plant training procedures.

For personnel with specific duties involving the NP-REP, this training will consist of initial training classes and annual retraining to maintain familiarity with the features of the REP. Participation in drills, while not a requirement, does augment the training of those personnel who do participate. The site EP group provides training to key site responders in the TSC, OSC, and the SED.

Training for Plant Access is handled in accordance with site specific security procedures.

The Safety and Emergency Response Training Academy (SERTA) provides emergency medical care training to medical personnel, and selected Nuclear Power personnel, stationed at the sites. Successful completion of training, commensurate with their duties, allows personnel to fulfill the role of medical care provider on the site MERT.

15.2 Offsite

CECC personnel will have current fitness for duty training. EP is responsible for ensuring that lesson plans are developed and training is conducted for all CECC personnel. All training provided under this plan is documented on an annual basis. Such documentation includes the date of the training, the names of those trained, and the training administered.

Training and annual retraining is provided to local plant support agencies (security, fire, ambulance, and hospital personnel), who may be involved with direct support of the site during an emergency.

Nuclear Operations Support is responsible for providing agreement hospital and ambulance support training. The sites are responsible for providing fire support training, with assistance from Nuclear Operations Support as needed. The sites are responsible for providing local law enforcement (security) training. Training shall include procedures for notification, basic radiation protection, expected roles, and site access procedures (as applicable).

15.3 Professional Development Training

Full time Emergency Preparedness staff members shall be afforded formal professional development training or activities commensurate with their duties and experience.

16.0 PLAN MAINTENANCE

16.1 NP-REP

16.1.1 Document Identification

Each NP-REP will have a controlled copy number.

Each page of the NP-REP will contain the following information:

NP-REP		NP-REP
Page 1	-or-	Appendix A
Rev. 1		Page A-1
		Rev. 1

Documents referenced in appendix E are issued in accordance with appropriate State procedures.

16.1.2 Periodic Review

The NP-REP and the appendices are reviewed by the sites and EP annually for accuracy, completeness, operational readiness, and compliance with existing regulations and established policy. This review is initiated by EP and results are documented.

TVA has agreements with outside organizations for radiological emergency support to furnish specific services. Copies of the letters documenting these agreements are forwarded to EP and are reviewed annually and updated as necessary by EP.

16.1.3 Changes

Revision to the NP-REP may result from the reviews described in section 16.1.2, drills, exercises, or changes in regulations. Changes are made and distributed according to figure 16-1. Changes identified from these reviews and drills and exercises will be made as expeditiously as possible and will not necessarily be held for submittal with an annual review.

Each line affected by a particular revision will be marked in the margin. All changes in a revision will be stated in the Revision Log; the pages affected and a brief explanation of the change will be included in the Revision Log.

Plant Operations Review Committee (PORC) approval will be obtained on all NP-REP revision to the site-specific appendices prior to their implementation. Changes to the main body of the NP-REP and Appendix E will be coordinated with all sites for PORC review (up to 30 days based on the volume and complexity of the change). If comments cannot be resolved by the Manager, EP, and responsible site management, the comment will be escalated to higher line management up to and including the TVA CNO and Executive Vice President. All changes to the NP-REP will be approved by the Vice President, Nuclear Operations Support, or his designee.

16.1.4 Distribution

Each NP-REP, its additions, and revisions will be authorized by an approval form and distributed by Management Services (MS).

MS issues controlled revisions and ensures all NP-REP holders have received all changes by requiring that copy holders sign a receipt, which is provided, and return it within two weeks.

MS maintains a historical file of all superseded REP material.

To provide REP holders with assurance that the plan is up-to-date, cover pages and revision logs are distributed with each revision or addition. The revision log lists the latest revision number, the date revised, pages revised, and the reason for the revision.

16.2 EIPs

16.2.1 Document Identification

Each EPIP manual bears a copy number. Pages of controlled documents are issued in accordance with approved procedures. Each page contains the following information similar to the following example:

CECC-EPIP-1
Page 5 of 12
Rev. 1

Each procedure in an EPIP will have a cover page listing the revision number and the effective date. Each procedure will also have a revision log or description of the revision. The procedure revision approval form will be signed by the approving authority (or their designee) responsible for that EPIP as listed below:

EIPs

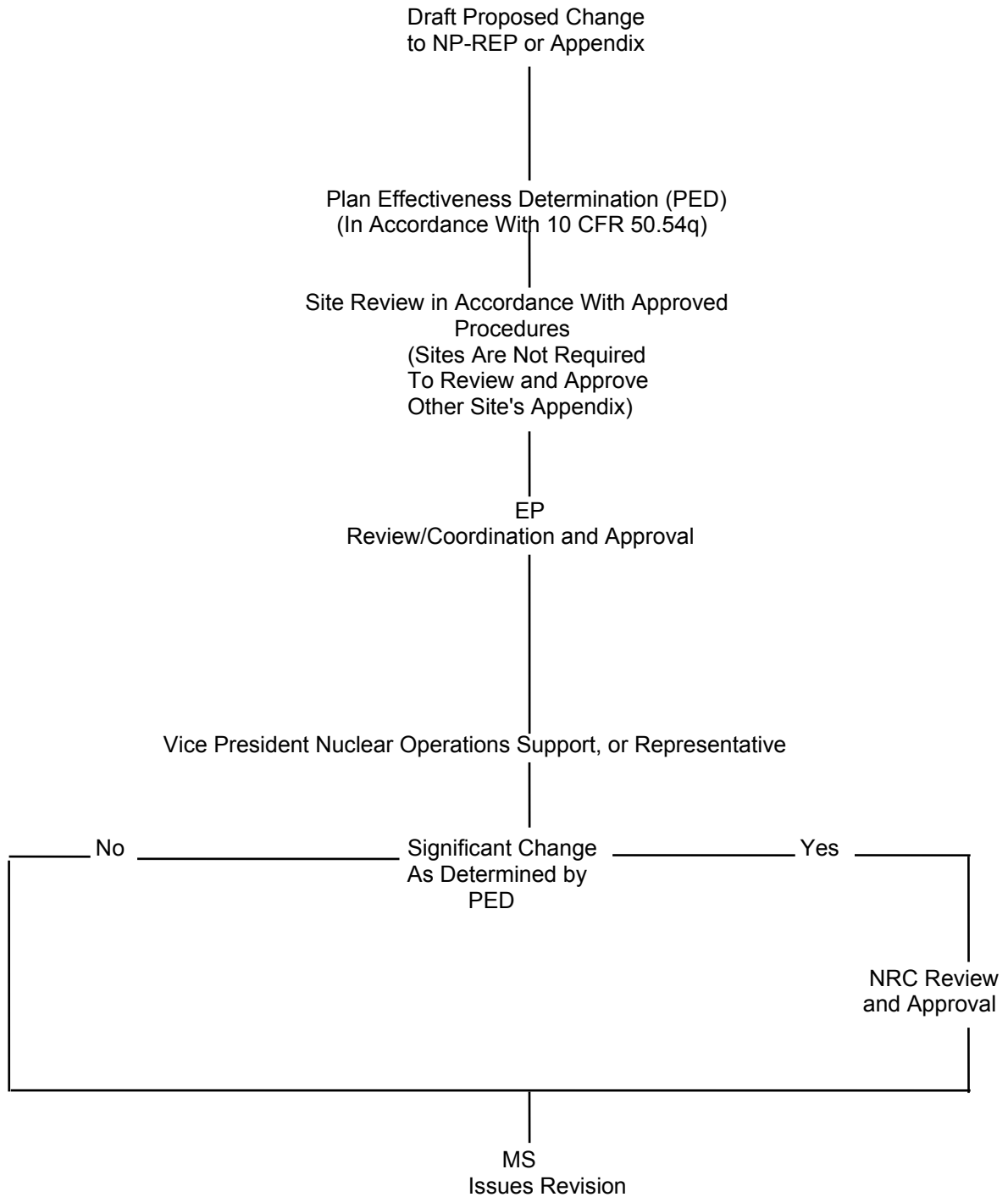
Approving Authority

CECC
BFN
SQN
WBN

Sr Manager, Emergency Preparedness
Plant Manager, BFN
Plant Manager, SQN
Plant Manager, WBN

FIGURE 16-1

UPDATE PROCEDURE FOR NP-REP AND APPENDICES



16.2.2 Periodic Review

The EIPs are reviewed annually for accuracy, completeness, operational readiness, and compliance with existing regulations by the responsible organization listed below. This review is initiated by Nuclear Operations Support and results are documented.

<u>EIPs</u>	<u>Organization</u>
CECC	REP Staff
BFN	Browns Ferry Nuclear Plant
SQN	Sequoyah Nuclear Plant
WBN	Watts Bar Nuclear Plant

EP coordinates a quarterly review of notification lists in the Radiological Emergency Notification Directory (REND). The review covers phone numbers and names and is documented by the REND Revision Log.

16.2.3 EPIP Changes

16.2.3.1 CECC-EPIP Changes

Revision to a CECC-EPIP may result from the reviews described in section 16.2.2, in drills and exercises, or changes to regulations. Changes are made and distributed according to figure 16-2. EIPs receive a review in accordance with 10 CFR 50.54(q) screening process consistent with the REP.

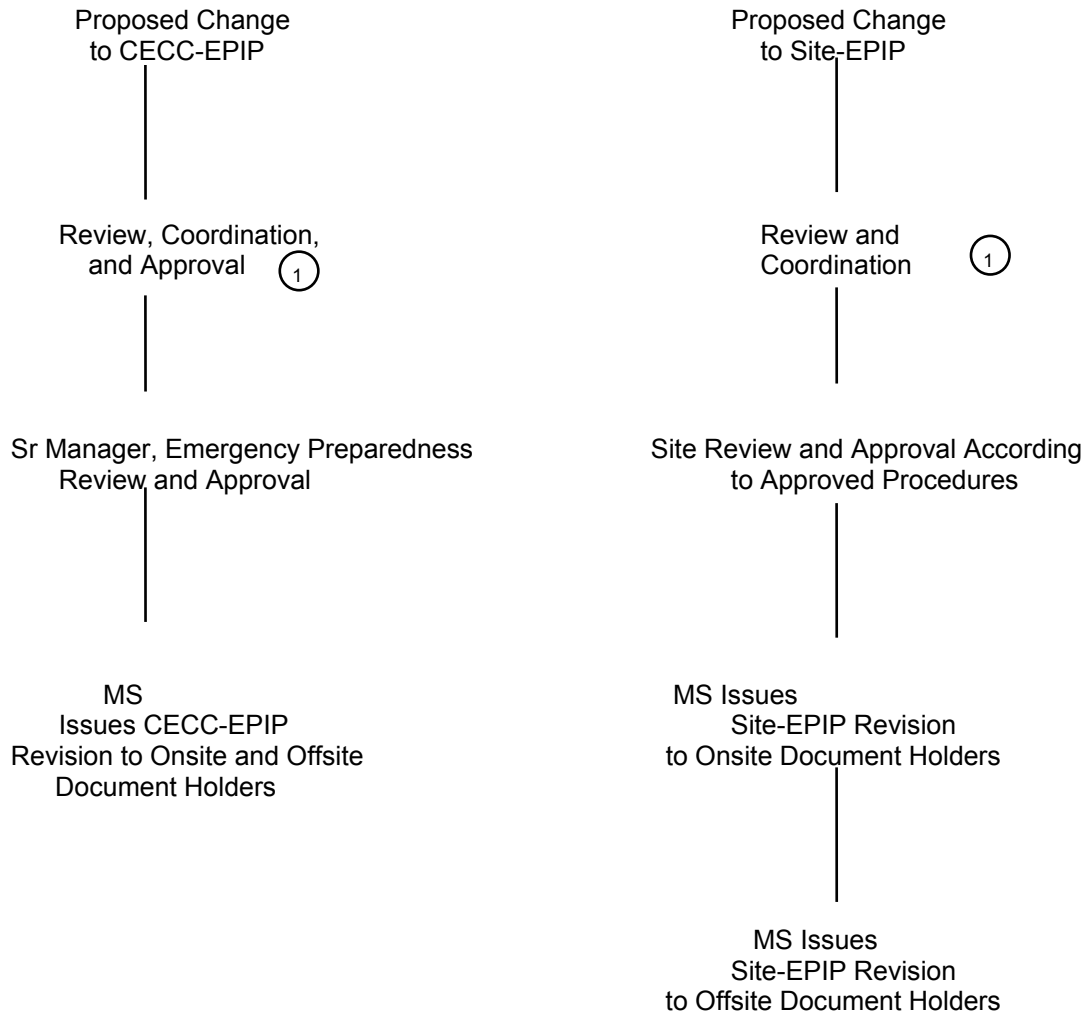
Each line affected by a particular revision will be marked. Whenever an entire page has been added or substantially changed, this is denoted by a statement at the bottom of the page. Whenever an entire procedure is revised, this is denoted on the Revision Log.

16.2.3.2 Site-EPIP Changes

Permanent, temporary, and emergency site-EPIP changes will be issued as controlled documents to plant document holders in accordance with site document control practices. MS will issue the changes to other document holders in accordance with MS document control practices. EIPs receive a review in accordance with 10 CFR 50.54(q) screening process consistent with the REP.

FIGURE 16-2

UPDATE PROCEDURE FOR EIPs



① EIPs receive review in accordance with 10CFR 50.54(q)
Screening process consistent with the REP

16.2.4 Distribution

Each CECC-EPIP or revision will be authorized by an approval form and distributed by MS. Site-EPIP changes will be distributed as discussed in section 16.2.3.2.

Upon receiving revision from EP, those assigned controlled copies of an EPIP sign a receipt, which is provided, and return it within two weeks to MS.

Each revision will be accompanied by a revised cover page for that procedure. MS maintains a historical file on all superseded CECC-EPIP material and the site maintains a historical file on all superseded site-EPIP material.

16.3 Document Relationships

The NP-REP and the associated supporting plans and procedures are issued as separate documents. TVA maintains the following documents:

1. NP-REP
2. CECC-EPIP
3. BFN-EPIP
4. SQN-EPIP
5. WBN-EPIP
6. REND
7. Evacuation Time Estimate Manual
8. On-shift Staffing Analysis Report (BFN, SQN, & WBN)
9. Evaluation and Analysis of the Alert and Notification System for Browns Ferry Nuclear Plant
10. Evaluation and Analysis of the Alert and Notification System for Sequoyah Nuclear Plant
11. Evaluation and Analysis of the Alert and Notification System for Watts Bar Nuclear Plant

These documents, along with the state plans referenced in Appendix E, may be issued separately or in combinations as applicable for the individual document holder.

16.4 Audits

Quality Assurance conducts audits/reviews of the NP-REP program in accordance with 10 CFR 50.54(t) for compliance with existing regulations and its own internal requirements. It is also responsible for offering recommendations on overall plan improvement. The results of the audit/review are documented, reported to appropriate organization management, and retained in the files for a period of five years.

16.5 Agreement Letters

Included in this section is a listing of agreements or contracts maintained for services of outside organizations during an emergency. Agreement letters for offsite law enforcement support are maintained by the site Nuclear Security Services and are updated annually. These agreement letters may be examined upon obtaining approval from the site Nuclear Security Manager. Agreement letters with other offsite organizations are maintained by EP.

- a. Agreements maintained with the following ambulance services for 24-hour availability of EMT-staffed ambulances for the transport of irradiated/contaminated patients:

Hamilton County Emergency Services, Chattanooga, TN
Athens-Limestone Hospital, Athens, AL
Rhea Ambulance Service, Dayton, TN

- b. Agreements maintained with the following medical centers to provide 24-hour availability of medical treatment for patients who may have been exposed to or contaminated with radioactive material:

Erlanger Health System, Chattanooga, TN
Memorial Health Care System (Memorial Northpark), Chattanooga, TN
Huntsville Hospital, Huntsville, AL
Decatur General Hospital, Decatur AL
Athens Regional Medical Center, Athens, TN
Rhea Medical Center, Dayton, TN

- c. Agreements maintained with the following fire departments with 24-hour assistance capabilities:

Rhea County Fire Department, TN
Soddy Daisy Fire Department, TN
Clements Volunteer Fire Department, AL

- d. John C. Calhoun State Community College agrees to provide facilities for use as a Joint Information Center in the event of a major incident at Browns Ferry Nuclear Plant and for drills in preparation for such an event.

- e. DOE Radiation Emergency Assistance Center/Training Site (REAC/TS), Oak Ridge, Tennessee - 24-hour availability of backup assistance to TVA for medical/radiological emergencies which exceed in-house and commercially available capabilities.

- f. INPO will provide assistance in locating and arranging additional emergency manpower, equipment, and the services of various technical experts from industry sources. INPO maintains this utility data in the INPO Emergency Resources Manual.

Enclosure 3

Radiological Emergency Plan (REP)

Appendix A: Browns Ferry Nuclear Plant

Revision 103

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-1 Revision 103
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APPENDIX A

BROWNS FERRY NUCLEAR PLANT

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-2 Revision 103
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REACTOR 1.0

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-3 Revision 103
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WATER LEVEL

1.1-U1

UNUSUAL EVENT

EAL: Uncontrolled water level decrease in Reactor Cavity with irradiated fuel assemblies expected to remain covered by water.

OPERATING CONDITION: Mode 5

BASIS: This event classification only applies during Mode 5 when the Reactor Head is removed. For the purposes of this event classification the Reactor Cavity includes the cavity and the Reactor Vessel.

This event classification is anticipatory to 1.1-A1 and should only be considered if, in the opinion of the Site Emergency Director, the water level decrease is substantial enough to ultimately result in increased dose rates in the area of the Reactor Cavity due to loss of shielding by water covering irradiated fuel. Uncontrolled water level decrease during Mode 5 is indicative of valve manipulation error or failure of equipment in such a manner as to cause uncontrolled drainage of the Reactor Cavity. Uncontrolled water level decrease may be detected by the presence of the low level alarm in the spent fuel storage pool, visual observation, increased radiation levels or various other symptoms that the Site Emergency Director considers valid indicators of the event.

The degraded status of safety systems designed to makeup water to the Reactor Vessel is of particular concern during Mode 5 although plant Technical Specifications require minimum makeup systems be operable except with the spent fuel storage gates removed and water level \geq 22 feet over the top of the reactor pressure vessel flange. These events tend to have long lead times relative to potential for release outside the site boundary, thus impact to public health and safety is very low. Classification as Unusual Event is warranted as a precursor to a more serious event.

Escalation to Alert is by actual uncover of irradiated fuel assemblies.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-AU2 example-1)
Technical Specification 3.5.2

NOTES: **1.1-U1/1.1-A1** - Applicable when the Reactor Head is removed and the Reactor Cavity is flooded.

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-4 Revision 103
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WATER LEVEL

1.1-U2

UNUSUAL EVENT

EAL: Uncontrolled water level decrease in Spent Fuel Storage Pool with irradiated fuel assemblies expected to remain covered by water.

OPERATING CONDITION: ALL

BASIS This event classification is anticipatory to 1.1-A2 and should only be considered if, in the opinion of the Site Emergency Director, the water level decrease is substantial enough to ultimately result in increased dose rates in the area of the Spent Fuel Storage Pool due to loss of shielding by water covering irradiated fuel. Uncontrolled water level decrease may be detected by the presence of the low level alarm in the spent fuel storage pool, visual observation, increased radiation levels or various other symptoms that the Site Emergency Director considers valid indicators of the event.

Uncontrolled water level decrease in Spent Fuel Storage Pools is indicative of failure of equipment in such a manner as to cause uncontrolled drainage. These events tend to have long lead times relative to potential for release outside the site boundary, thus impact to public health and safety is very low. Classification as Unusual Event is warranted as a precursor to a more serious event.

Escalation to Alert is by actual uncover of irradiated fuel assemblies.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-AU2 example-2)

NOTES:

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-5 Revision 103
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WATER LEVEL

1.1-A1

ALERT

EAL: Uncontrolled water level decrease in Reactor Cavity expected to result in irradiated fuel assemblies being uncovered.

OPERATING CONDITION: Mode 5

BASIS: This event classification only applies during Mode 5 when the Reactor Head is removed. For the purposes of this event classification the Reactor Cavity includes the cavity and the Reactor Vessel.

Uncontrolled water level decrease during Mode 5 is indicative of valve manipulation error or failure of equipment in such a manner as to cause uncontrolled drainage of the Reactor Cavity. The degraded status of safety systems designed to makeup water to the Reactor Vessel is of particular concern during Mode 5 although plant Technical Specifications require minimum makeup systems be operable except with the spent fuel storage gates removed and water level ≥ 22 feet over the top of the reactor pressure vessel flange.

Uncontrolled water level decrease may be detected by visual observation, increased radiation levels or various other symptoms that the Site Emergency Director considers valid indicators of the event. Expected fuel uncover may be detected by increased radiation levels, Visual observation, RPV level instrumentation expected to drop below -162 inches, or best judgment of the Site Emergency Director based on present and past events and trends.

Due to the long lead times associated with these events there is time available to take corrective actions, and there is little potential for substantial fuel damage. Significant exposures to onsite personnel are likely during these events and it is probable that additional personnel will be needed onsite; therefore the Alert classification is warranted.

Escalation is by Radiological Release event classifications.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-AA2 example-3)
Technical Specifications 3.5.2

NOTES: **1.1-U1/1.1-A1** - Applicable when the Reactor Head is removed and the Reactor Cavity is flooded.

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-6 Revision 103
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WATER LEVEL

1.1-A2

ALERT

EAL: Uncontrolled water level decrease in Spent Fuel Storage Pool expected to result in irradiated fuel assemblies being uncovered.

OPERATING CONDITION: ALL

BASIS: Uncontrolled water level decrease in Spent Fuel Storage Pools is indicative of failure of equipment in such a manner as to cause uncontrolled drainage. These events tend to have long lead times relative to potential for release outside the site boundary, thus impact to public health and safety is very low.

Uncontrolled water level decrease may be detected by visual observation, increased radiation levels or various other symptoms that the Site Emergency Director considers valid indicators of the event. Expected fuel uncover may be detected by increased radiation levels, Visual observation, or best judgment of the Site Emergency Director based on present and past events and trends.

There is time available to take corrective actions, and there is little potential for substantial fuel damage. Offsite exposures are expected to remain below the Environmental Protection Agency's Protective Action Guidelines; however, exposures to onsite personnel are of particular concern during this event; therefore the Alert classification is warranted.

Escalation is by Radiological Release event classifications.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-AA2 example-4)

NOTES:

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-7 Revision 103
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WATER LEVEL

1.1-S1

SITE AREA EMERGENCY

EAL: Reactor water level can NOT be maintained above -162 inches. (TAF).

OPERATING CONDITION: ALL

BASIS: If reactor water level cannot be maintained above TAF the potential exist for fuel cladding damage. Events most likely to result in coolant inventory loss to this extent are RCS boundary degradation events or station blackout events. For this event to be declared, RPV water level must have decreased or be trending to a value that, in the opinion of the Site Emergency Director, has resulted in or will result in some actual core uncover.

This event classification also applies in Mode 5 when the Reactor Vessel head is installed. Inadvertent draining of the Reactor Vessel is possible under these conditions due to valving errors associated with the RHR system or failures associated with isolation valves during alignment changes of systems connected to the Reactor Vessel below the normal water level.

The fact that the transient was severe enough to result in inability to maintain RPV level coupled with the anticipatory nature of this event classification as a precursor to more serious event warrants the Site Area Emergency event classification.

For events that occur during operation, escalation to General Emergency is based on inability to assure adequate core cooling by restoring and maintaining RPV water level following transients that have resulted in extreme RPV water level decrease. For events that occur during shutdown or Mode 5, escalation is by radioactive release event classifications.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-FS, SS5, SS4, example-1)

NOTES: 1.1-S1 - Applicable in Mode 5 when the Reactor Head is installed.

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-8 Revision 103
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WATER LEVEL	1.1-S2
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SITE AREA EMERGENCY

EAL: Reactor water level can NOT be determined.

OPERATING CONDITION: Mode 1 or 2 or 3

BASIS: Inability to determine Reactor water level during operation may be due to boiling in the reference or variable instrument legs, instrument power failures, or conflicting information from uncontrolled indicator oscillations.

This condition requires Reactor flooding following emergency depressurization. Adequate core cooling is assured by these measures. Due to the severity of these actions and the uncertainty of Reactor status it is appropriate to treat this as a potential loss for Reactor Coolant System and Fuel Cladding integrity; therefore, this event is appropriate for the Site Area Emergency classification.

Escalation to General Emergency is based on inability to assure adequate core cooling in this mode.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-FS)

NOTES:

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-9 Revision 103
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WATER LEVEL

1.1-G1

GENERAL EMERGENCY

EAL: Reactor water level can NOT be restored and maintained above -180 inches.

OPERATING CONDITION: Mode 1 or 2 or 3

BASIS: If Reactor water level cannot be restored and maintained above the Minimum Steam Cooling Reactor Water Level (MSCRWL), core damage is possible due to inadequate steam generation, by the covered portion of the Reactor core, to remove decay heat and prevent cladding heat up to a point that results in clad failure.

For either of the above conditions to be met, the control room operators should have progressed in the execution of the EOIs to the point that all high pressure and all low pressure systems that are available within a reasonable time frame have been attempted and are unsuccessful in reversing the adverse RPV water level trend.

Events most likely to result in coolant inventory loss or loss of makeup capability to this extent are RCS boundary degradation events or events resulting from loss of multiple systems such as station blackout. During such transients or accidents the potential for Primary Containment failure increases substantially; therefore, the General Emergency classification is appropriate.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-FG)

NOTES:

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-10 Revision 103
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WATER LEVEL

1.1-G2

GENERAL EMERGENCY

EAL: Reactor water level can NOT be determined

AND

Either of the following exists:

- The reactor will remain subcritical without boron under all conditions, and
 - Less than 4 MSRVS can be opened, or
 - Reactor pressure can NOT be restored and maintained above Suppression Chamber pressure by at least 70 psi.
- It has NOT been determined that the reactor will remain subcritical without boron under all conditions and unable to restore and maintain MSCP in Table 1.1-G2.

OPERATING CONDITION: Mode 1 or 2 or 3

BASIS: Inability to determine Reactor water level during operation may be due to boiling in the reference or variable instrument legs, instrument power failures, or conflicting information from uncontrolled indicator oscillations. This condition requires Reactor flooding following emergency depressurization. If the reactor will remain subcritical without boron under all conditions, adequate core cooling is assured only if at least 4 MSRVS are opened and Minimum Reactor Flooding Pressure (MRFP) is maintained with Reactor pressure at least 70 PSI above Suppression Chamber pressure. If it has not been determined that the reactor will remain subcritical without boron under all conditions, adequate core cooling can only be assured when the Minimum Steam Cooling Pressure (MSCP) is restored and maintained. If adequate core cooling is not assured core damage is probable under this scenario due to the extreme nature of the plant conditions that resulted in the inability to determine Reactor level (i.e., high containment temperatures, loss of multiple power supplies, etc.). Primary Containment integrity cannot be assured under all these conditions; therefore, the General Emergency classification is appropriate.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-FG)

NOTES: **1.1-G2** - The reactor will remain subcritical under all conditions without boron when:

- Any 19 control rods are inserted to position 02, with all other control rods fully inserted.
- All control rods except one are inserted to or beyond position 00.
- Determined by Reactor Engineering.

WATER LEVEL	1.1-G2
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GENERAL EMERGENCY
CONTINUED

CURVES/TABLES:

TABLE 1.1 - G2 MINIMUM STEAM COOLING PRESS (MSCP)	
NUMBER OF OPEN MSRVs	MSCP (PSIG)
6 or More	190
5	230
4	290

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-12 Revision 103
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SCRAM FAILURE

1.2-A

ALERT

EAL: Failure of RPS automatic scram functions to bring the reactor subcritical

AND

Manual scram or ARI (automatic or manual) was successful.

OPERATING CONDITION: Mode 1 or 2

BASIS: 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.

This event classification indicates failure of the RPS to automatically 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 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 barrier. Any set of actions by the Reactor Operator at Panel 9-5 that cause control rods to rapidly insert into the core and bring the Reactor subcritical is considered a manual scram.

Escalation to Site Area Emergency is based on fuel clad barrier or RCS barrier event classifications.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-SA2)

NOTES: **1.2** - Subcritical is defined as reactor power below the heating range and not trending upward.

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-13 Revision 103
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SCRAM FAILURE

1.2-S

SITE AREA EMERGENCY

EAL: Failure of automatic scram, manual scram, and ARI to bring the reactor subcritical.

OPERATING CONDITION: Mode 1 or 2

BASIS: Manual scram and ARI are not considered successful if action away from the Reactor Control Console (Panel 9-5) was required to scram the Reactor.

A failure of the automatic and manual scram systems may result in the Reactor producing more heat than the maximum decay heat load for which the safety systems are designed. A Site Area Emergency classification is appropriate because conditions exist that lead to potential loss of both fuel clad and Reactor Coolant System (RCS) barriers. Therefore, this event classification ensures timely emergency response to the event before actual barriers loss has taken place.

Escalation to General Emergency is based upon inability to bring Reactor power within decay heat removal capability before Suppression Pool temperature reaches the Heat Capacity Temperature Limit (HCTL).

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-SS2, SS4 example -1)

NOTES: **1.2** - Subcritical is defined as reactor power below the heating range and not trending upward.

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-14 Revision 103
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SCRAM FAILURE

1.2-G

GENERAL EMERGENCY

EAL: Failure of automatic scram, manual scram, and ARI. Reactor power is above 3%

AND

Either of the following conditions exists:

- Suppression Pool temp exceeds HCTL. Refer to Curve 1.2-G.
- Reactor water level can NOT be restored and maintained at or above -180 inches

OPERATING CONDITION: Mode 1 or 2

BASIS: Automatic scram, manual scram, and ARI are not considered successful if action away from the Reactor Control Console was required to scram the Reactor.

Under these conditions all efforts, including boron injection, have been unsuccessful in bringing Reactor power within the decay heat removal capability of the Emergency Core Cooling Systems (ECCS). Additionally, an extreme challenge to the ability to cool the Reactor Core exists if Reactor Pressure Vessel (RPV) water level cannot be maintained sufficient to ensure adequate core cooling.

Another consideration is the inability to remove heat using the Main Condenser or Suppression Pool. In the event that neither heat sink is effective and Reactor power remains above this level, then a core melt sequence exists. In this situation, core degradation can occur rapidly; therefore, a General Emergency classification is appropriate in anticipation of degradation of multiple fission product barriers.

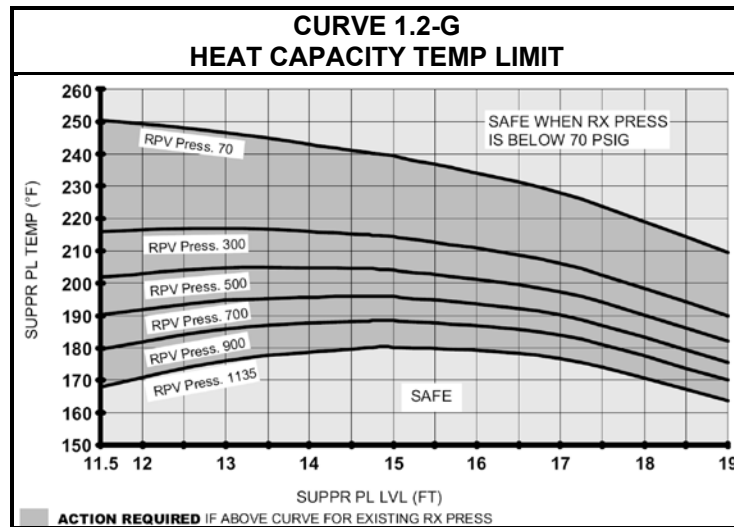
REFERENCES: Reg Guide 1.101 Rev. 3 (NUMARC-SG2)

NOTES:

<h1>SCRAM FAILURE</h1>	<h1>1.2-G</h1>
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GENERAL EMERGENCY
CONTINUED

CURVES/TABLES:



BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-16 Revision 103
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REACTOR COOLANT ACTIVITY

1.3-U

UNUSUAL EVENT

EAL: Reactor coolant activity exceeds 26 $\mu\text{Ci/gm}$ dose equivalent I-131 (Technical Specification Limits) as determined by chemistry sample.

OPERATING CONDITION: ALL

BASIS: Reactor coolant activity samples exceeding Technical Specification limits for Iodine spikes are representative of fuel clad degradation. An Unusual Event is declared because of potential degradation in the level of safety of the plant. Iodine levels exceeding Technical Specification limits are a potential precursor of more serious problems.

Escalation to Alert would be based on higher Reactor coolant activity values indicative of significant fuel cladding failure.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-SU4 example-2)
Technical Specification 3.4.6

NOTES:

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-17 Revision 103
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REACTOR COOLANT ACTIVITY

1.3-A

ALERT

EAL: Reactor coolant activity exceeds 300 $\mu\text{Ci/gm}$ dose equivalent Iodine-131 as determined by chemistry sample.

OPERATING CONDITION: Mode 1 or 2 or 3

BASIS: Reactor coolant activity samples exceeding 300 $\mu\text{Ci/gm}$ dose equivalent Iodine-131 are well above those expected for Iodine spikes and represent a significant loss of the fuel clad barrier. Any loss or potential loss of the fuel clad barrier warrants the declaration of an Alert.

Escalation to Site Area Emergency would be based on the conditions given above coupled with a loss or potential loss of either the Primary Containment or Reactor Coolant System barrier or Radiological Releases.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-FA)
RIMS L36 921201 806

NOTES:

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-18 Revision 103
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MSL/OFFGAS RADIATION

1.4-U

UNUSUAL EVENT

EAL: Valid MAIN STEAM LINE RADIATION HIGH-HIGH alarm, 1, 2, or 3-RA-90-135C

OR

Valid OG PRETREATMENT RADIATION HIGH alarm, 1, 2, or 3-RA-90-157A.

OPERATING CONDITION: Mode 1 or 2 or 3

BASIS: Main Steam Line radiation high high or offgas radiation high is indicative of fuel cladding leakage.

The Main Steam Line radiation high high alarm setpoint is normally set at 3 times normal full power background. 3 times normal full power background is in excess of any spikes expected from operational transients that do not result in cladding failure. This alarm setpoint is substantially above that which would be indicative of fuel cladding damage above Technical Specification allowable limits; however, the presence of a valid alarm warrants declaration of an Unusual Event and consideration of other symptoms and event classifications for possible upgrade of the event based on fission product barrier loss.

The offgas pretreatment radiation high alarm setpoint is set at a value that is indicative of the ODCM allowable limits for radiation release.

Either of these conditions is considered a potential degradation in the level of safety of the plant and a potential precursor of a more serious problem.

Escalation to the Alert is based on either Reactor coolant samples exceeding 300 $\mu\text{Ci/gm}$ or Drywell radiation levels indicative of loss of the fuel cladding barrier.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-SU4 example-1)

NOTES:

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-19 Revision 103
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LOSS OF DECAY HEAT REMOVAL 1.5-A

ALERT

EAL: Reactor moderator temperature can NOT be maintained below 212⁰ F whenever Technical Specifications require Mode 4 conditions or during operations in Mode 5.

OPERATING CONDITION: Mode 4 or 5

BASIS: This event classification addresses loss of decay heat removal functions when Mode 4 is required or during Mode 5. Loss of decay heat removal capability can result in more serious consequences depending upon whether Primary Containment is in tact and Emergency Core Cooling System (ECCS) equipment status. In any condition where Mode 4 is required, loss of decay heat removal capability represents a significant degradation in plant conditions that can lead to fuel cladding damage or RCS degradation. In order to maintain anticipatory philosophy the Alert classification is appropriate for this event.

Escalation to Site Area Emergency or General Emergency is by loss of Reactor water level that has or will uncover the fuel or Radiological Release Event classification.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-SA3)

NOTES:

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-20 Revision 103
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LOSS OF DECAY HEAT REMOVAL 1.5-S

SITE AREA EMERGENCY

EAL: Suppression Pool temperature, level and RPV pressure can NOT be maintained in the safe area of Curve 1.5-S.

OPERATING CONDITION: Mode 1 or 2 or 3

BASIS: Suppression Pool temperature is limited by Curve 1.5-S as a function of suppression pool level and reactor pressure in order to preclude failure of Primary Containment or equipment necessary for the safe shutdown of the plant following emergency depressurization. When Suppression Pool temperature cannot be maintained below the limits of the curve corresponding to existing suppression pool level and reactor pressure, emergency depressurization is required and continued decay heat removal at operating temperature and pressure is no longer permissible.

Suppression Pool level is limited by Curve 1.5-S to the range of 11.5 feet to 19 feet in order to preclude failure of Primary Containment or equipment necessary for the safe shutdown of the plant and preserve the pressure suppression function of the containment for possible future emergency depressurization. When Suppression Pool level cannot be maintained within the limits of the curve, continued decay heat removal at operating pressures and temperatures is no longer permissible and emergency depressurization is required.

Exceeding the limits of Curve 1.5-S represents a loss of heat sink for decay heat removal and inability to maintain Mode 3. Under these conditions there is an actual failure of systems intended for protection of the public; therefore, Site Area Emergency is warranted. Escalation to General Emergency is by Abnormal Rad levels, Radiological Release or Primary Containment failure events.

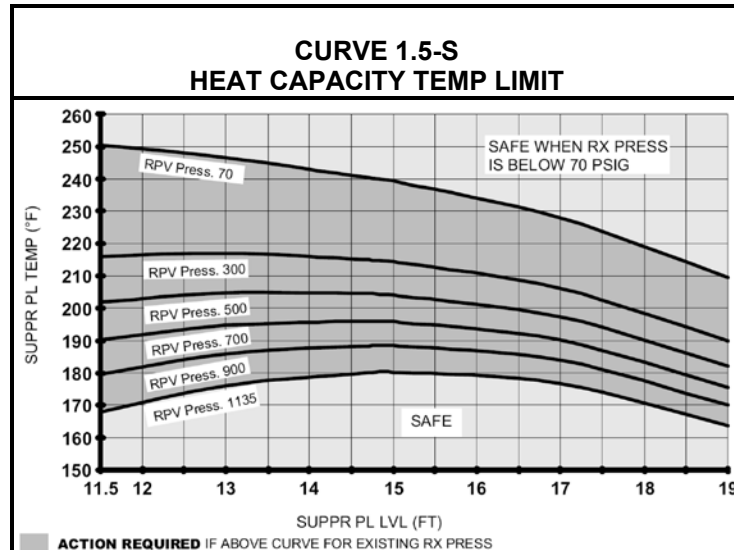
REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-SS4)

NOTES:

CURVES/TABLES:

LOSS OF DECAY HEAT REMOVAL 1.5-S**SITE AREA EMERGENCY**
CONTINUED

CURVES/TABLES:



PRIMARY CONTAINMENT 2.0

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-23 Revision 103
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PRIMARY CONTAINMENT PRESSURE 2.1-A

ALERT

EAL: Drywell pressure at or above 2.45 psig

AND

Indication of Primary System leakage into Primary Containment. Refer to Table 2.1-A.

OPERATING CONDITION: Mode 1 or 2 or 3

BASIS: If Drywell pressure reaches the ECCS initiation and Reactor scram setpoint (2.45 PSIG) there is clear indication that a Primary System leak of sufficient magnitude exists that could result in break propagation leading to significantly larger loss of Reactor coolant inventory.

Efforts to reduce Drywell pressure by additional cooling or Primary Containment venting have been unsuccessful either due to equipment malfunction or the magnitude of the leak. This condition represents a degraded level of safety of the plant due to Reactor Coolant System (RCS) degradation and warrants the Alert classification.

Escalation to Site Area Emergency is by exceeding the Pressure Suppression Pressure Limit or inability to maintain Reactor water level above TAF.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-FA)

NOTES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-24 Revision 103
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PRIMARY CONTAINMENT PRESSURE	2.1-A
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ALERT
CONTINUED

CURVES/TABLES:

TABLE 2.1-A INDICATIONS OF PRIMARY SYSTEM LEAKAGE INTO PRIMARY CONTAINMENT
Primary Containment Pressure High Alarm
Drywell Floor Drain Sump Pump Excessive Operation
Drywell CAM Activity Increasing
Drywell Temperature High Alarm
Chemistry Sample Radionuclide Comparison To Reactor Water

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-25 Revision 103
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PRIMARY CONTAINMENT PRESSURE 2.1-S

SITE AREA EMERGENCY

EAL: Suppression Chamber pressure can NOT be maintained in the safe area of Curve 2.1-S.

OPERATING CONDITION: Mode 1 or 2 or 3

BASIS: The inability to maintain Primary Containment pressure in the safe region of the Pressure Suppression Pressure curve indicates that Drywell and Suppression Chamber sprays cannot be initiated or are not effective in reversing an increasing trend in Primary Containment pressure.

Primary Containment pressure and Suppression Pool water level outside the safe region of Curve 2.1-S represents loss of ultimate heat sink and inability to maintain hot shutdown.

Escalation to General Emergency is based on Primary Containment pressure reaching 55 PSIG or event classifications resulting from loss or potential loss of the fuel clad and RCS Barriers.

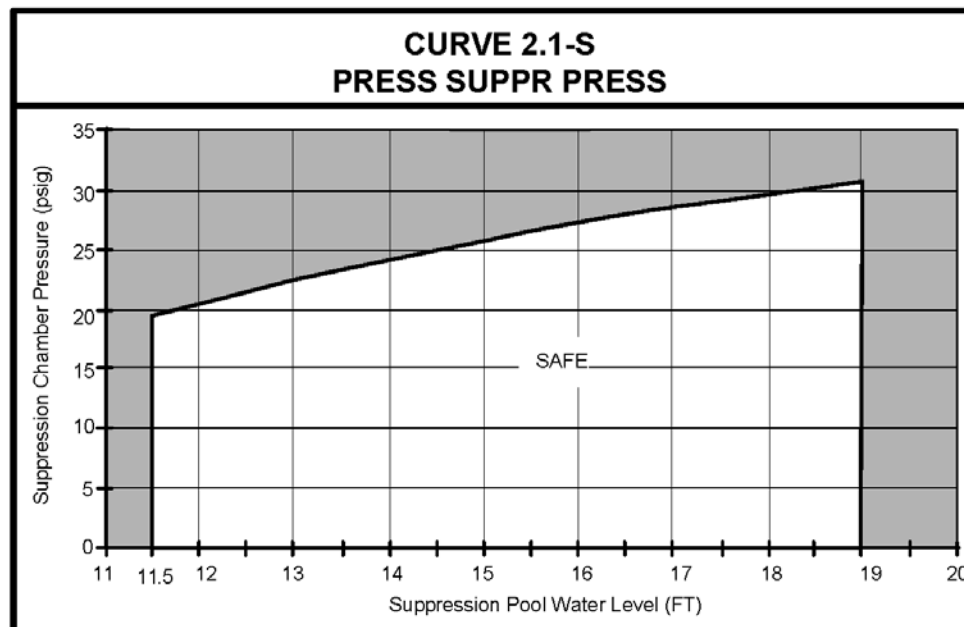
REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-SS4)

NOTES:

PRIMARY CONTAINMENT PRESSURE	2.1-S
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SITE AREA EMERGENCY
(CONTINUED)

CURVES/TABLES:



BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-27 Revision 103
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PRIMARY CONTAINMENT PRESSURE 2.1-G

GENERAL EMERGENCY

EAL: Suppression Chamber pressure can NOT be maintained below 55 psig.

OPERATING CONDITION: Mode 1 or 2 or 3

BASIS: Primary Containment pressures of this magnitude represent loss of the RCS barrier and require emergency venting of Primary Containment irrespective of offsite release rates. Fuel cladding integrity is threatened either directly due to loss of Reactor coolant inventory or potentially due to direction in the Emergency Operating Instructions (EOIs) to spray Primary Containment irrespective of whether adequate core cooling is assured. Under these conditions, potential loss of fuel cladding integrity should be assumed; therefore, the General Emergency classification is appropriate.

REFERENCES: Reg Guide 1.101 Rev. 3 (NUMARC-FG)

NOTES:

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-28 Revision 103
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PRIMARY CONTAINMENT HYDROGEN 2.2-S

SITE AREA EMERGENCY

EAL: Drywell or Suppression Chamber hydrogen concentration at or above 4%

AND

Drywell or Suppression Chamber oxygen concentration at or above 5%

OPERATING CONDITION: Mode 1 or 2 or 3

BASIS: Hydrogen and oxygen concentrations in this range are an indication that an event has occurred more severe than any that have been analyzed in the FSAR or that systems installed for control of hydrogen and oxygen have been unsuccessful in stopping an upward trend in these concentrations. Concentrations of this magnitude indicate severe fuel degradation and are approaching the lower deflagration limits for combustible mixture. Venting through the Standby Gas Treatment System to the elevated release path (Stack) is required. Dilution is accomplished by use of Containment Atmosphere Dilution (CAD) system to control hydrogen and oxygen. Releases can be expected to approach levels associated with the Site Area Emergency Radiological Release event classification.

Escalation to General Emergency is based on higher concentrations of hydrogen which directly threaten Primary Containment integrity and require emergency venting through large unfiltered pathway or through Radioactive Release event classifications.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-FS)

NOTES:

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-29 Revision 103
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PRIMARY CONTAINMENT HYDROGEN 2.2-G

GENERAL EMERGENCY

EAL: Drywell or Suppression Chamber hydrogen concentration at or above 6%

AND

Drywell or Suppression Chamber oxygen concentration at or above 5%.

OPERATING CONDITION: Mode 1 or 2 or 3

BASIS: Hydrogen and oxygen concentrations in this range are an indication that an event has occurred more severe than any that have been analyzed in the FSAR or that systems installed for control of hydrogen and oxygen have been unsuccessful in stopping an upward trend in these concentrations. Concentrations of this magnitude indicate severe fuel degradation in conjunction with RCS barrier failure and have reached the lower deflagration limits for combustible mixture. This event constitutes a potential loss of the Primary Containment barrier because Ignition of the mixture could result in Primary Containment failure. Procedures require venting and purging through the large unfiltered pathway for control of hydrogen and oxygen representing a loss of Primary Containment; therefore the General Emergency classification is appropriate.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-FG)

NOTES:

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-30 Revision 103
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DRYWELL RADIATION 2.3-A

ALERT

EAL: Drywell radiation levels at or above the values listed in Table 2.3-A/2.3-S2, with the RCS barrier intact inside Primary Containment.

OPERATING CONDITION: Mode 1 or 2 or 3

BASIS: Drywell radiation monitor readings as indicated in Table 2.3-A/2.3-S2 are indicative of significant fuel cladding failure. These values are different for Unit 1, 2 & 3 due to detector geometry, and relative shielding. These values have been calculated assuming an Iodine inventory associated with a concentration of 300 $\mu\text{Ci/gm}$ dose equivalent Iodine-131 in Reactor coolant with the RCS barrier intact. Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations (including Iodine spikes) allowed within Technical Specifications. This level indicates a loss of the fuel clad barrier; therefore, the Alert classification is warranted.

Escalation to Site Are Emergency is based upon higher Drywell radiation levels indicative of loss of the Reactor Coolant System pressure boundary in conjunction with significant fuel cladding failure or loss or potential loss of Primary Containment in conjunction with significant fuel cladding failure.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-FA)
Calculation ND-N0090-930055 R12
Technical Specifications 3.4.6

NOTES:

CURVES/TABLES:

TABLE 2.3-A/2.3-S2 DRYWELL RADIATION LEVELS WITH RCS BARRIER INTACT INSIDE PRIMARY CONTAINMENT					
UNIT 1		UNIT 2		UNIT 3	
RAD MONITOR	R/HR	RAD MONITOR	R/HR	RAD MONITOR	R/HR
1-RE-90-272A	196	2-RE-90-272A	642	3-RE-90-272A	196
1-RE-90-273A	297	2-RE-90-273A	297	3-RE-90-273A	297

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-31 Revision 103
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DRYWELL RADIATION

2.3-S1

SITE AREA EMERGENCY

EAL: Drywell radiation levels at or above the values listed in Table 2.3-S1/2.3-G2 with the RCS barrier NOT intact inside Primary Containment.

OPERATING CONDITION: Mode 1 or 2 or 3

BASIS: Drywell radiation monitor readings as indicated in Table 2.3-S1/2.3-G2 are indicative of Reactor Coolant System pressure boundary failure inside Primary Containment in conjunction with significant fuel cladding failures. These values are different for Unit 1, 2 & 3 due to detector geometry, and relative shielding. These values have been calculated assuming instantaneous release and dispersal of Reactor coolant noble gas and Iodine inventory associated with a concentration of 300 $\mu\text{Ci/gm}$ dose equivalent Iodine-131 into the Drywell atmosphere. This value is significantly higher than that specified for fuel clad barrier degradation; therefore, this level indicates either a loss of both the fuel clad barrier and RCS barrier or severe cladding degradation.

Escalation to General Emergency is based upon either loss or potential loss of the Primary Containment barrier or significantly higher radiation levels indicative of gross amounts of fission products in Primary Containment.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-FS)
Calculation ND-N0090-930050 R11

NOTES:

CURVES/TABLES:

TABLE 2.3-S1/2.3-G2 DRYWELL RADIATION LEVELS WITH RCS BARRIER NOT INTACT INSIDE PRIMARY CONTAINMENT					
UNIT 1		UNIT 2		UNIT 3	
RAD MONITOR	R/HR	RAD MONITOR	R/HR	RAD MONITOR	R/HR
1-RE-90-272A	2981	2-RE-90-272A	2263	3-RE-90-272A	2981
1-RE-90-273A	2960	2-RE-90-273A	2960	3-RE-90-273A	2960

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-32 Revision 103
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DRYWELL RADIATION

2.3-S2

SITE AREA EMERGENCY

EAL: Drywell radiation levels at or above the values listed in Table 2.3-A/2.3-S2, with the RCS barrier intact inside Primary Containment,

AND

Either of the following exists:

- Indications of loss of Primary Containment. Refer to Table 2.3/2.5-U.
- Primary Containment integrity can NOT be maintained.

OPERATING CONDITION: Mode 1 or 2 or 3

BASIS: Drywell radiation monitor readings as indicated in Table 2.3-A/2.3-S2 are indicative of significant fuel cladding failure. These values are different for Unit 1, 2 & 3 due to detector geometry, and relative shielding. These values have been calculated assuming an Iodine inventory associated with a concentration of 300 $\mu\text{Ci/gm}$ dose equivalent Iodine-131 in Reactor coolant with the RCS barrier intact inside Primary Containment. Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations (including Iodine spikes) allowed within Technical Specifications. This level indicates a loss of the fuel clad barrier. Table 2.3/2.5U contains indications of loss of primary containment. Primary Containment cannot be maintained encompasses any condition or set of conditions based on current trends or anticipated circumstances that, in the judgment of the Site Emergency Director, will result in inability to maintain the Primary Containment pressure barrier. Additionally, potential loss of Primary Containment should be considered when the Site Emergency Director can determine that a substantial threat exists that may result in Primary Containment failure. Loss or potential loss of the Primary Containment barrier in conjunction with these levels of radiation represents loss of two of the three fission product barriers; therefore, the Site Area Emergency classification is warranted.

Escalation to General Emergency is based upon loss or potential loss of the Reactor Coolant System pressure barrier as indicated by higher levels of Drywell radiation.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-FS)
Calculation ND-N0090-930055 R12

NOTES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-33 Revision 103
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<div> <div>DRYWELL RADIATION</div> <div>2.3-S2</div> </div>

SITE AREA EMERGENCY
CONTINUED

CURVES/TABLES

TABLE 2.3-A/2.3-S2 DRYWELL RADIATION LEVELS WITH RCS BARRIER INTACT INSIDE PRIMARY CONTAINMENT					
UNIT 1		UNIT 2		UNIT 3	
RAD MONITOR	R/HR	RAD MONITOR	R/HR	RAD MONITOR	R/HR
1-RE-90-272A	196	2-RE-90-272A	642	3-RE-90-272A	196
1-RE-90-273A	297	2-RE-90-273A	297	3-RE-90-273A	297

TABLE 2.3/2.5-U INDICATIONS OF LOSS OF PRIMARY CONTAINMENT	
Unexplained Loss Of Containment Pressure	
Exceeding 1, 2, or 3-SI-4.7.A.2.a Limits	
Inability To Isolate Any Line Exiting Containment When Isolation Is Required	
Venting Irrespective Of Offsite Release Rates Per EOIs/SAMGs	

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-34 Revision 103
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DRYWELL RADIATION

2.3-G1

GENERAL EMERGENCY

EAL: Drywell radiation levels at or above the values listed in Table 2.3-G1 with the RCS barrier NOT intact inside Primary Containment.

OPERATING CONDITION: Mode 1 or 2 or 3

BASIS: A Drywell radiation monitor reading as indicated in Table 2.3-G1 corresponds to approximately 20 % fuel clad damage with 100% coolant release into containment. These values are different for Unit 1, 2 & 3 due to detector geometry, and relative shielding.

A major release of radioactivity requiring offsite protection actions 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 Primary Containment, such that 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 cladding damage is less than 20%.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-FG)
Calculation ND-N0090-930050 R11
NUREG-1228 "Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents"

NOTES:

CURVES/TABLES:

TABLE 2.3-G1 DRYWELL RADIATION LEVELS WITH RCS BARRIER NOT INTACT INSIDE PRIMARY CONTAINMENT					
UNIT 1		UNIT 2		UNIT 3	
RAD MONITOR	R/HR	RAD MONITOR	R/HR	RAD MONITOR	R/HR
1-RE-90-272A	90091	2-RE-90-272A	68405	3-RE-90-272A	90091
1-RE-90-273A	89450	2-RE-90-273A	89450	3-RE-90-273A	89450

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DRYWELL RADIATION

2.3-G2

GENERAL EMERGENCY

EAL: Drywell radiation levels at or above the values listed in Table 2.3-S1/2.3-G2 with the RCS barrier NOT intact inside Primary Containment,

AND

Either of the following exists:

- Indications of loss of Primary Containment. Refer to Table 2.3/2.5-U.
- Primary Containment integrity can NOT be maintained.

OPERATING CONDITION: Mode 1 or 2 or 3

BASIS: Drywell radiation monitor readings as indicated in Table 2.3-S1/2.3-G2 are indicative of Reactor Coolant System pressure boundary failure inside Primary Containment in conjunction with significant fuel cladding failures. These values are different for Unit 1, 2 & 3 due to detector geometry, and relative shielding. These values have been calculated assuming instantaneous release and dispersal of Reactor coolant noble gas and Iodine inventory associated with a concentration of 300 $\mu\text{Ci/gm}$ dose equivalent Iodine-131 into the Drywell atmosphere. This value is higher than that specified for fuel clad barrier degradation; therefore, this level indicates a loss of both the fuel clad barrier and RCS barrier. Table 2.3/2.5U contains indications of loss of primary containment. Primary Containment cannot be maintained encompasses any condition or set of conditions based on current trends or anticipated circumstances that, in the judgment of the Site Emergency Director, will result in inability to maintain the Primary Containment pressure barrier. Additionally, potential loss of Primary Containment should be considered when the Site Emergency Director can determine that a substantial threat exists that may result Primary Containment failure. Loss or potential loss of the Primary Containment barrier, in conjunction with these levels of radiation, represents loss of all three fission product barriers; therefore the General Emergency classification is warranted.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-FG)
Calculation ND-N0090-930050 R11

NOTES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-36 Revision 103
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DRYWELL RADIATION

2.3-G2

GENERAL EMERGENCY (CONTINUED)

CURVES/TABLES:

TABLE 2.3-S1/2.3-G2 DRYWELL RADIATION LEVELS WITH RCS BARRIER NOT INTACT INSIDE PRIMARY CONTAINMENT					
UNIT 1		UNIT 2		UNIT 3	
RAD MONITOR	R/HR	RAD MONITOR	R/HR	RAD MONITOR	R/HR
1-RE-90-272A	2981	2-RE-90-272A	2263	3-RE-90-272A	2981
1-RE-90-273A	2960	2-RE-90-273A	2960	3-RE-90-273A	2960

TABLE 2.3/2.5-U INDICATIONS OF LOSS OF PRIMARY CONTAINMENT	
Unexplained Loss Of Containment Pressure	
Exceeding 1, 2, or 3-SI-4.7.A.2.a Limits	
Inability To Isolate Any Line Exiting Containment When Isolation Is Required	
Venting Irrespective Of Offsite Release Rates Per EOIs/SAMGs	

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-37 Revision 103
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DRYWELL INTERNAL LEAKAGE

2.4-U

UNUSUAL EVENT

EAL: Drywell unidentified leakage exceeds 10 gpm

OR

Drywell identified leakage exceeds 40 gpm.

OPERATING CONDITION: Mode 1 or 2 or 3

BASIS: This event classification is included as an Unusual Event because it may be a precursor to more serious conditions (e.g., pipe break or equipment degradation) and is considered to be a potential degradation in the level of safety of the plant. The 10 gpm value for unidentified leakage is two times the Technical Specification limit, indicating an increase beyond the licensed operating value. 10 gpm is also observable using control room instrumentation and Surveillance Instructions.

The 40 gpm value for identified leakage is conservatively below two times the licensed operating value of 30 GPM but within the capacity of the sump pumps if only one pump were operating. 40 GPM is also observable using control room instrumentation and compatible with Surveillance Instructions. Identified leakage above 40 gpm indicates significant equipment degradation and could represent degradation in the level of safety of the plant.

Escalation to Alert is based on 50 gpm unidentified leakage into the Drywell or Drywell pressure at or above 2.45 PSIG.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-SU5)
Technical Specifications 3.4.4

NOTES:

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-38 Revision 103
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DRYWELL INTERNAL LEAKAGE

2.4-A

ALERT

EAL: Drywell unidentified leakage exceeds 50 gpm.

OPERATING CONDITION: Mode 1 or 2 or 3

BASIS: The potential loss of Reactor Coolant System pressure boundary is based on RCS leakage and determined at a level indicative of a small breach of the Reactor Coolant System but well within the makeup capability of the normal or emergency high pressure makeup systems, (i.e., CRD, HPCI, RCIC). Core uncover is not a significant concern for a leak of this magnitude; however, break propagation leading to significantly larger loss of inventory is possible. 50 gpm is within the capacity of the Drywell Floor Drain Sump Pumps and measurable using control room instrumentation. Leakage of this magnitude, if not detected early, will result in isolation of the Primary Containment Isolation Valves on the sump pump discharge and Reactor scram at 2.45 PSIG Drywell pressure.

Escalation to Site Area Emergency is by Suppression Chamber pressure exceeding Pressure Suppression Pressure limits or inability to maintain Reactor water level above TAF.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-FA)
Calculation ND-N0999-930077 R2

NOTES:

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-39 Revision 103
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LOSS OF PRIMARY CONTAINMENT 2.5-U

UNUSUAL EVENT

EAL: Inability to maintain Primary Containment pressure boundary. Refer to Table 2.3/2.5-U.

OPERATING CONDITION: Mode 1 or 2 or 3

BASIS: This event classification is intended to address loss or potential loss of the Primary Containment barrier. Use of the Technical Specification definition for Primary Containment, by itself, is not appropriate for determining if the barrier is lost or threatened because in many cases the Technical Specifications could be exceeded with the barrier still intact. Indications for entry into this event classification are those indications that are consistent with actual breach of the pressure boundary. Some examples of pressure boundary breach are:

- Unexplained loss of pressure in the Drywell or Suppression Chamber.
- Leakage in excess of Technical Requirement Manual 3.6.5.
- Inability to isolate any line exiting containment when isolation is required.
- Intentional venting during EOI execution irrespective of offsite release rates.

Escalation to Site Area Emergency is based upon loss of the fuel clad barrier or RCS barrier in conjunction with loss of the primary containment barrier. There is no Alert classification based on loss of only the primary containment barrier.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-FU)
Technical Requirement Manual 3.6.5

NOTES:

CURVES/TABLES:

TABLE 2.3/2.5-U INDICATIONS OF LOSS OF PRIMARY CONTAINMENT
Unexplained Loss Of Containment Pressure
Exceeding 1, 2, or 3-SI-4.7.A.2.a Limits
Inability To Isolate Any Line Exiting Containment When Isolation Is Required
Venting Irrespective Of Offsite Release Rates Per EOIs/SAMGs

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-40 Revision 103
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SECONDARY CONTAINMENT 3.0

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-41 Revision 103
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SECONDARY CONTAINMENT TEMPERATURE 3.1-S

SITE AREA EMERGENCY

EAL: An unisolable Primary System leak is discharging into Secondary Containment

AND

Any area temperature exceeds the Maximum Safe Operating Temperature limit listed in Table 3.1.

OPERATING CONDITION: Mode 1 or 2 or 3

BASIS: The Maximum Safe Operating Temperatures of table 3.1 are based on the Browns Ferry Environmental Qualification (EQ) program for safety related equipment. EQ program assumptions include single failure criteria for pipe break that isolates as required. Temperatures in excess of those in Table 3.1 are indicative of pipe breaks that fail to isolate as required. Secondary Containment temperatures of this magnitude resulting from primary system leakage are indicative of significant loss of both the RCS pressure boundary and the Primary Containment pressure boundary. The Site Area Emergency classification is appropriate based upon loss of any two barriers.

Escalation to General Emergency is based on loss or potential loss of the fuel cladding barrier or Radioactivity Release event classifications.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-FS)

NOTES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-42 Revision 103
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SECONDARY CONTAINMENT TEMPERATURE 3.1-S

SITE AREA EMERGENCY (CONTINUED)

CURVES/TABLES:

TABLE 3.1 MAXIMUM SAFE OPERATING AREA TEMPERATURE LIMITS				
AREA	APPLICABLE PANEL 9-21 TEMPERATURE ELEMENTS (UNLESS OTHERWISE NOTED)	MAX SAFE OPERATING VALUE °F		
		UNIT 1	UNIT 2	UNIT 3
RHR A/C Pump Room	74-95A	215	150	155
RHR B/D Pump Room	74-95B	150	210	215
HPCI Turbine Area	73-55A	275	270	270
CS A/C Pump and RCIC Turbine Area	71-41A	190	190	190
RCIC Steam Supply Area	71-41B, 41C, 41D	195	200	250
HPCI Steam Supply Area	73-55B, 55C, 55D	245	240	240
RHR A/C Pump Supply Area	74-95H	245	240	240
RHR B/D Pump Supply Area	74-95G	190	240	240
Main Steam Line Leak Detection High	(XA-55-3D-24) Panel 9-3 TIS-1-60A	315	315	315
RHR Valve Room	74-95E	175	170	175
RWCU Isol Logic Channel A/B Temp High	(XA-55-5B-32/33) Panel 9-5 69-835A, B, C, D Aux Inst Room	175	170	175
RWCU Outbd Isol Vlv Area	69-29F	220	220	220
RWCU Hx Area	69-29G	220	220	220
RWCU Hx Exh Duct	69-29H	220	220	220
RWCU Recirc Pump A Area	69-29D	215	215	215
RWCU Recirc Pump B Area	69-29E	215	215	215
RHR A/C Hx Room	74-95C	210	195	200
RHR B/D Hx Room	74-95D	210	195	200
FPC Hx Area	74-95F	160	155	155

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-43 Revision 103
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SECONDARY CONTAINMENT TEMPERATURE 3.1-G

GENERAL EMERGENCY

EAL: An unisolable Primary System leak is discharging into Secondary Containment

AND

Any area temperature exceeds the Maximum Safe Operating Temperature limit listed in Table 3.1

AND

Any indication of potential or significant fuel cladding failure exists. Refer to Table 3.1-G/3.2-G with RCS Barrier intact inside Primary Containment.

OPERATING CONDITION: Mode 1 or 2 or 3

BASIS: The Maximum Safe Operating Temperatures of table 3.1 are based on the Browns Ferry Environmental Qualification (EQ) program for safety related equipment. EQ program assumptions include single failure criteria for pipe break that isolates as required. Temperatures in excess of those in Table 3.1 are indicative of pipe breaks that fail to isolate as required. Secondary Containment temperatures of this magnitude resulting from primary system leakage are indicative of significant loss of both the RCS pressure boundary and the Primary Containment pressure boundary. Table 3.1-G/3.2-G provides guidance for determining if significant fuel failure should be assumed. These values are different for Unit 1, Unit 2, and Unit 3 due to detector geometry and relative shielding.

This event classification represents loss of all three barriers designed to contain fission products during accidents; therefore, the General Emergency classification is appropriate.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-FG)
Calculation ND-N0090-930055 R12

NOTES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-44 Revision 103
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SECONDARY CONTAINMENT TEMPERATURE 3.1-G

GENERAL EMERGENCY (CONTINUED)

CURVES/TABLES:

TABLE 3.1 MAXIMUM SAFE OPERATING AREA TEMPERATURE LIMITS				
AREA	APPLICABLE PANEL 9-21 TEMPERATURE ELEMENTS (UNLESS OTHERWISE NOTED)	MAX SAFE OPERATING VALUE °F		
		UNIT 1	UNIT 2	UNIT 3
RHR A/C Pump Room	74-95A	215	150	155
RHR B/D Pump Room	74-95B	150	210	215
HPCI Turbine Area	73-55A	275	270	270
CS A/C Pump and RCIC Turbine Area	71-41A	190	190	190
RCIC Steam Supply Area	71-41B, 41C, 41D	195	200	250
HPCI Steam Supply Area	73-55B, 55C, 55D	245	240	240
RHR A/C Pump Supply Area	74-95H	245	240	240
RHR B/D Pump Supply Area	74-95G	190	240	240
Main Steam Line Leak Detection High	(XA-55-3D-24) Panel 9-3 TIS-1-60A	315	315	315
RHR Valve Room	74-95E	175	170	175
RWCU Isol Logic Channel A/B Temp High	(XA-55-5B-32/33) Panel 9-5 69-835A, B, C, D Aux Inst Room	175	170	175
RWCU Outbd Isol Vlv Area	69-29F	220	220	220
RWCU Hx Area	69-29G	220	220	220
RWCU Hx Exh Duct	69-29H	220	220	220
RWCU Recirc Pump A Area	69-29D	215	215	215
RWCU Recirc Pump B Area	69-29E	215	215	215
RHR A/C Hx Room	74-95C	210	195	200
RHR B/D Hx Room	74-95D	210	195	200
FPC Hx Area	74-95F	160	155	155

TABLE 3.1-G/3.2-G INDICATIONS OF POTENTIAL OR SIGNIFICANT FUEL CLADDING FAILURE WITH RCS BARRIER INTACT INSIDE PRIMARY CONTAINMENT					
UNIT 1 DRYWELL RADIATION		UNIT 2 DRYWELL RADIATION		UNIT 3 DRYWELL RADIATION	
1-RE-90-272A	> 196 R/HR	2-RE-90-272A	> 642 R/HR	3-RE-90-272A	> 196 R/HR
1-RE-90-273A	> 297 R/HR	2-RE-90-273A	> 297 R/HR	3-RE-90-273A	> 297 R/HR
Reactor Coolant Activity ≥ 300 µCi/gm Dose Equivalent Iodine 131		Reactor Coolant Activity ≥ 300 µCi/gm Dose Equivalent Iodine 131		Reactor Coolant Activity ≥ 300 µCi/gm Dose Equivalent Iodine 131	

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-45 Revision 103
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SECONDARY CONTAINMENT RADIATION 3.2-A

ALERT

EAL:

Any of the following high radiation alarms on Panel 9-3:

- 1, 2, or 3-RA-90-1A, Fuel Pool Floor Alarm
- 1, 2, or 3-RA-90-250A, Reactor, Turbine, Refuel Exhaust
- 1, 2, or 3-RA-90-142A, Reactor Refuel Exhaust
- 1, 2, or 3-RA-90-140A, Refueling Zone Exhaust

AND

Confirmation by Refuel Floor personnel that irradiated fuel damage may have occurred.

OPERATING CONDITION: ALL

BASIS:

This event is indicative of irradiated fuel damage caused by a dropped fuel bundle or other heavy solid objects into the Reactor Cavity or Spent Fuel Storage Pools. The second part of this event classification associates the listed alarms with events that could result in actual irradiated fuel damage versus increased background from other possible sources. Compared to core damage that can occur from full power operating conditions, there is little potential for substantial fuel damage; however, significant exposures to onsite personnel are possible and protective actions for site personnel may be necessary. For these reasons the Alert classification is warranted.

Escalation is by Radiological Release event classifications.

REFERENCES:

Reg Guide 1.101 Rev. 3, (NUMARC-AA2-example-1)

NOTES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-46 Revision 103
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SECONDARY CONTAINMENT RADIATION 3.2-S

SITE AREA EMERGENCY

EAL: An unisolable Primary System leak is discharging into Secondary Containment

AND

Any area radiation level at or above the Maximum Safe Operating Area radiation limit listed in Table 3.2.

OPERATING CONDITION: Mode 1 or 2 or 3

BASIS: Secondary Containment radiation levels of this magnitude are indicative of significant inability to contain or control radioactive materials within the Primary System and Primary Containment. If the Primary System is the source, then Site Area Emergency is warranted based upon loss of any two fission product barriers.

Escalation to General Emergency is based on loss or potential loss of the fuel cladding barrier or Radioactive Release event classifications.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-FS)

NOTES:

CURVES/TABLES:

TABLE 3.2 MAXIMUM SAFE OPERATING AREA RADIATION LIMITS				
AREA	RAD MONITOR	MAX SAFE VALUE MR/HR		
		UNIT 1	UNIT 2	UNIT 3
RHR West Room	90-25A	1000	1000	1000
RHR East Room	90-28A	1000	1000	1000
HPCI Room	90-24A	1000	1000	1000
CS/RCIC Room	90-26A	1000	1000	1000
Core Spray Room	90-27A	1000	1000	1000
Suppr Pool Area	90-29A	1000	1000	1000
CRD-HCU West Area	90-20A	1000	1000	1000
CRD-HCU East Area	90-21A	1000	1000	1000
TIP Drive Area	90-23A	1000	1000	1000
North RWCU System Area	90-13A	1000	1000	1000
South RWCU System Area	90-14A	1000	1000	1000
RWCU System Area	90-9A	1000	1000	1000
MG Set Area	90-4A	1000	1000	1000
Fuel Pool Area	90-1A	1000	1000	1000
Service Flr Area	90-2A	1000	1000	1000
New Fuel Storage	90-3A	1000	N/A	N/A

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SECONDARY CONTAINMENT RADIATION 3.2-G

GENERAL EMERGENCY

EAL: An unisolable Primary System leak is discharging into Secondary Containment

AND

Any area radiation level at or above the Maximum Safe Operating Area radiation limit listed in Table 3.2.

AND

Any indication of potential or significant fuel cladding failure exists. Refer to Table 3.1-G/3.2-G with RCS Barrier intact inside Primary Containment.

OPERATING CONDITION: Mode 1 or 2 or 3

BASIS: Secondary Containment radiation levels of this magnitude are indicative of significant inability to contain or control radioactive materials within the primary system and Primary Containment. If the primary system is the source then these indications represent loss of RCS pressure boundary and Primary Containment pressure boundary. Table 3.1-G/3.2-G provides guidance for determining if significant fuel failure should be assumed.

This event classification represents loss or potential loss of all three barriers designed to contain fission products during accidents; therefore, the General Emergency classification is appropriate.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-FG)
Calculation ND-N0090-930055 R12

NOTES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-48 Revision 103
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SECONDARY CONTAINMENT RADIATION 3.2-G

GENERAL EMERGENCY (CONTINUED)

CURVES/TABLES:

TABLE 3.2 MAXIMUM SAFE OPERATING AREA RADIATION LIMITS				
AREA	RAD MONITOR	MAX SAFE VALUE MR/HR		
		UNIT 1	UNIT 2	UNIT 3
RHR West Room	90-25A	1000	1000	1000
RHR East Room	90-28A	1000	1000	1000
HPCI Room	90-24A	1000	1000	1000
CS/RCIC Room	90-26A	1000	1000	1000
Core Spray Room	90-27A	1000	1000	1000
Suppr Pool Area	90-29A	1000	1000	1000
CRD-HCU West Area	90-20A	1000	1000	1000
CRD-HCU East Area	90-21A	1000	1000	1000
TIP Drive Area	90-23A	1000	1000	1000
North RWCU System Area	90-13A	1000	1000	1000
South RWCU System Area	90-14A	1000	1000	1000
RWCU System Area	90-9A	1000	1000	1000
MG Set Area	90-4A	1000	1000	1000
Fuel Pool Area	90-1A	1000	1000	1000
Service Flr Area	90-2A	1000	1000	1000
New Fuel Storage	90-3A	1000	N/A	N/A

TABLE 3.1-G/3.2-G INDICATIONS OF POTENTIAL OR SIGNIFICANT FUEL CLADDING FAILURE WITH RCS BARRIER INTACT INSIDE PRIMARY CONTAINMENT					
UNIT 1 DRYWELL RADIATION		UNIT 2 DRYWELL RADIATION		UNIT 3 DRYWELL RADIATION	
1-RE-90-272A	≥ 196 R/HR	2-RE-90-272A	≥ 642 R/HR	3-RE-90-272A	≥ 196 R/HR
1-RE-90-273A	≥ 297 R/HR	2-RE-90-273A	≥ 297 R/HR	3-RE-90-273A	≥ 297 R/HR
Reactor Coolant Activity ≥ 300 μCi/gm Dose Equivalent Iodine 131		Reactor Coolant Activity ≥ 300 μCi/gm Dose Equivalent Iodine 131		Reactor Coolant Activity ≥ 300 μCi/gm Dose Equivalent Iodine 131	

RADIOACTIVITY RELEASE 4.0

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GASEOUS EFFLUENT

4.1-U

UNUSUAL EVENT

EAL: Gaseous release exceeds ANY limit and duration in Table 4.1-U.

OPERATING CONDITION: ALL

BASIS: Unplanned radioactivity releases that exceed Table 4.1-U limits and continue for 1 hour or longer represent an uncontrolled situation and potential degradation in the level of safety of the plant. The Offsite Dose Calculation Manual (ODCM) contains the site specific release limits and appropriate surveillance requirements which normally monitor these limits. Table 4.1-U is based on 2 times the ODCM limit. The release should not be averaged over 60 minutes. For example, a release of 4 times ODCM limits for 30 minutes does not meet the requirements for this event classification. The 1 hour time period allows sufficient time to isolate any release after exceeding ODCM limits. Release continuing for more than 1 hour represents inability to isolate or control the release. The Site Emergency Director should declare the event as soon as it is determined that the release duration has or will likely exceed 1 hour. The value of 0.10 mrem/hr at the site boundary is based on a proration of twice the 500 mrem/yr ODCM instantaneous release rate limit.

Utilize Radiation Protection for obtaining site boundary assessments.

Escalation to Alert is based on radiation release rate which exceeds 200 times the ODCM limit.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-AU1 example-1 and 2)
EDMS L63 010206 800
10CFR20

NOTES: **4.1-U** - Prior to making this emergency classification based upon the WRGERMS indication, assess the release by either of the following:

1. Actual field measurements exceed the limits in Table 4.1-U
2. 0-SI 4.8.B.1.a.1 Release Fraction exceeds 2.0

If neither assessment can be conducted within 60 minutes then the declaration must be made on the valid WRGERMS reading.

CURVES/TABLES:

Table 4.1-U RELEASE LIMITS FOR UNUSUAL EVENT			
TYPE	MONITORING METHOD	LIMIT	DURATION
Gaseous Release Rate	Stack Noble Gas (WRGERMS)	$2.88 \times 10^7 \mu\text{Ci/sec}$	1 Hour
Gaseous Release Rate	0-SI 4.8.B.1.a.1	Release Fraction 2.0	1 Hour
Site Boundary Radiation Reading	Field Assessment Team	0.10 MREM/HR Gamma	1 Hour

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GASEOUS EFFLUENT

4.1-A

ALERT

EAL: Gaseous release exceeds ANY limit and duration in Table 4.1-A.

OPERATING CONDITION: ALL

BASIS: This event escalates from Unusual Event by increasing the magnitude of the release by a factor of 100. The release limit is equivalent to 200 times the Offsite Dose Calculation Manual (ODCM) limit. The value of 10 mrem/hr at the site boundary is based on a proration of the 500 mrem/yr criteria for both time (8766 hr/yr) and the 200 multiplier. The required release duration is reduced to 15 minutes in recognition of the increased severity. Table 4.1-A contains the Alert limits and appropriate monitoring points for the releases.

Utilize Radiation Protection for obtaining site boundary assessments.

Escalation to Site Area Emergency is based on radiation release which will yield a dose to a member of the public which exceeds 10CFR20 limits.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-AA1 example-1 and 2)
EDMS L63 010206 800
10CFR20
EPA 400

NOTES: **4.1-A** - Prior to making this emergency classification based upon the WRGERMS indication, assess the release by either of the following:

1. Actual field measurements exceed the limits in Table 4.1-A
2. 0-SI 4.8.B.1.a.1 Release Fraction exceeds 200

If neither assessment can be conducted within 15 minutes then the declaration must be made on the valid WRGERMS reading.

CURVES/TABLES:

Table 4.1-A RELEASE LIMITS FOR ALERT			
TYPE	MONITORING METHOD	LIMIT	DURATION
Gaseous Release Rate	Stack Noble Gas (WRGERMS)	$2.88 \times 10^{-9} \mu\text{Ci/sec}$	15 Minutes
Gaseous Release Rate	0-SI 4.8.B.1.a.1	Release Fraction 200	15 Minutes
Site Boundary Radiation Reading	Field Assessment Team	10 MREM/HR Gamma	15 Minutes

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-52 Revision 103
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GASEOUS EFFLUENT

4.1-S

SITE AREA EMERGENCY

EAL: EITHER of the following conditions exists:

- Gaseous release exceeds or is expected to exceed ANY limit and duration in Table 4.1-S.
- Dose assessment indicates actual or projected dose consequences above 100 mrem TEDE or 500 mrem thyroid CDE.

OPERATING CONDITION: ALL

BASIS: The limits in this event classification are based on 10 percent of the EPA Protective Action Guidelines or the 10CFR20 dose limit for a member of the public. These limits also provide a desirable gradient between Alert, Site Area Emergency, and General Emergency.

Table 4.1-S limits for stack and field surveys measurements are consistent with 10 percent of the EPA Protective Action Guidelines or the 10CFR20 dose limit for a member of the public. Stack Noble Gas Release Rates of 5.9×10^9 $\mu\text{Ci/sec}$ for 15 minutes, site boundary radiation readings of 100 mrem/hr for 1 hour, and Iodine-131 concentration of 3.9×10^{-7} $\mu\text{Ci/cm}^3$ for 1 hour are indicative of dose consequences consistent with the limits described previously. The durations in Table 4.1-S are consistent with NUMARC recommendations and industry standards. If analyses indicated a longer or shorter duration for this period in which the substantial portion of the activity is released these dose rates should be adjusted.

Utilize Radiation Protection for obtaining site boundary. Dose projection assessments should be requested through the CECC by the implementation of CECC EPIP-8, if the CECC is not staffed utilize site Radiation Protection for dose projection assessments through the implementation of BFN EPIP-13.

The 500 mrem thyroid CDE limit was established in consideration of the 1 to 5 ratio of the EPA Protective Action Guidelines for TEDE and thyroid CDE.

Escalation to General Emergency is based on actual or projected dose exceeding 1000 mrem TEDE or 5000 mrem thyroid CDE.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-AS1 example-1, 3 and 4)
EDMS L63 010206 800
10CFR20
EPA 400

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-53 Revision 103
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GASEOUS EFFLUENT

4.1-S

SITE AREA EMERGENCY (CONTINUED)

NOTES:

4.1-S -Prior to making this emergency classification based upon the Gaseous Release Rate indication, assess the release by either of the following methods:

1. Actual field measurements exceed the limits in Table 4.1-S.
2. Projected or Actual Dose Assessments exceed 100 mrem TEDE or 500 mrem CDE.

If neither assessment can be conducted within 15 minutes then the declaration must be made based on the valid WRGERMS reading.

CURVES/TABLES:

Table 4.1-S RELEASE LIMITS FOR SITE AREA EMERGENCY			
TYPE	MONITORING METHOD	LIMIT	DURATION
Gaseous Release Rate	Stack Noble Gas (WRGERMS)	$5.9 \times 10^{-9} \mu\text{Ci/sec}$	15 Minutes
Site Boundary Radiation Reading	Field Assessment Team	100 MREM/HR Gamma	1 Hour
Site Boundary Iodine-131	Field Assessment Team	$3.9 \times 10^{-7} \mu\text{Ci/cm}^3$	1 Hour

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-54 Revision 103
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GASEOUS EFFLUENT

4.1-G

GENERAL EMERGENCY

EAL: EITHER of the following conditions exists:

- Gaseous release exceeds or is expected to exceed ANY limit and duration in Table 4.1-G.
- Dose assessment indicates actual or projected dose consequences above 1000 mrem TEDE or 5000 mrem thyroid CDE.

OPERATING CONDITION: ALL

BASIS: The limits in this event classification are based on the EPA Protective Action Guidelines which require public protective actions if dose consequences of 1000 mrem TEDE or 5000 mrem thyroid CDE are indicated. These limits also provide a desirable gradient between Alert, Site Area Emergency, and General Emergency and represent the upper level of the gradient.

Table 4.1-G limits for stack and field surveys measurements are consistent with the EPA Protective Action Guidelines for dose limits requiring public protective actions. Stack Noble Gas Release Rates of 5.9×10^{10} $\mu\text{Ci/sec}$ for 15 minutes, site boundary radiation readings of 1000 mrem/hr for 1 hour, and Iodine-131 concentration of 3.9×10^{-6} $\mu\text{Ci/cm}^3$ for 1 hour are indicative of dose consequences consistent with the limits described previously. The durations in Table 4.1-G are consistent with NUMARC recommendations and industry standards. If analyses indicated a longer or shorter duration for this period in which the substantial portion of the activity is released these dose rates should be adjusted.

Utilize Radiation Protection for obtaining site boundary assessments. Dose projection assessments should be requested through the CECC by the implementation of CECC EPIP-8, if the CECC is not staffed utilize site Radiation Protection for dose projection assessments through the implementation of BFN EPIP-13.

The 5000 mrem thyroid CDE limit was established in consideration of the 1 to 5 ratio of the EPA Protective Action Guidelines for TEDE and thyroid CDE. Actual meteorology is used in dose assessment calculations to achieve the most accurate dose assessment possible.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-AG1 example-1, 3 and 4)
EDMS L63 010206 800
10CFR20

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-55 Revision 103
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GASEOUS EFFLUENT

4.1-G

GENERAL EMERGENCY (CONTINUED)

NOTES:

4.1-G - Prior to making this emergency classification based upon the Gaseous Release Rate indication, assess the release by either of the following methods:

1. Actual field measurements exceed the limits in Table 4.1-G.
2. Projected or Actual Dose Assessments exceed 1000 mrem TEDE or 5000 mrem CDE.

If neither assessment can be conducted within 15 minutes then the declaration must be made based on the valid WRGERMS reading.

CURVES/TABLES:

Table 4.1-G RELEASE LIMITS FOR GENERAL EMERGENCY			
TYPE	MONITORING METHOD	LIMIT	DURATION
Gaseous Release Rate	Stack Noble Gas (WRGERMS)	$5.9 \times 10^{10} \mu\text{Ci/sec}$	15 Minutes
Site Boundary Radiation Reading	Field Assessment Team	1000 MREM/HR Gamma	1 Hour
Site Boundary Iodine-131	Field Assessment Team	$3.9 \times 10^{-6} \mu\text{Ci} / \text{cm}^3$	1 Hour

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-56 Revision 103
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MAIN STEAM LINE BREAK

4.2-U

UNUSUAL EVENT

EAL: Main Steam Line break outside Primary Containment with isolation.

OPERATING CONDITION: Mode 1 or 2 or 3

BASIS: This event classification is intended to address the puff release associated with a Main Steam Line break outside Primary that isolates by PCIS Logic as required or can be isolated from the Main Control Room. Regardless of whether the break is in the Turbine Building or the Reactor Building a ground level release should be anticipated due to the blowout panels between the two buildings. Design basis analysis shows that even if MSIV closure occurs within design limits, dose consequences from a "puff" release should be expected. Thus this event classification is included due to the possibility of offsite exposures from the "puff" release.

This event is detected by instrumentation which inputs to the PCIS Logic circuitry. Main Steam Line high flow, Reactor low pressure with the mode switch in "Run", And Turbine Building Main Steam Space high temperature are all symptoms of the event and should be evaluated to determine if an actual break has occurred.

Escalation to ALERT is based on radiation release rate event classifications.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-FU)

NOTES:

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-57 Revision 103
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MAIN STEAM LINE BREAK

4.2-S

SITE AREA EMERGENCY

EAL: Unisolable Main Steam Line break outside Primary Containment.

OPERATING CONDITION: Mode 1 or 2 or 3

BASIS: This event classification applies to Main Steam Line Break that cannot be isolated by PCIS Logic or from the Main Control Room. Regardless of whether the break is in the Turbine Building or the Reactor Building a ground level release is expected due to the blowout panels between the two buildings. This event classification represents a loss of two of the three fission product barriers.

Main Steam Line high flow, Reactor low pressure with the mode switch in "Run", and Turbine Building Main Steam Space high temperature are all symptoms of the event. This event is anticipatory to 4.1-S and the threshold for leakage outside Primary and Secondary Containment should be considered to be any continuous discharge of steam through the break that, in the opinion of the Site Emergency Director, could result in exceeding the limits outlined in 4.1-S.

Escalation to General Emergency is based on loss of the Fuel Clad barrier or radioactivity release event classifications.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-FS)

NOTES:

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-58 Revision 103
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LIQUID EFFLUENT

4.3-U

UNUSUAL EVENT

EAL: Liquid release rate exceeds 20 times ECL as determined by chemistry sample

AND

Release duration exceeds or will exceed 60 minutes.

OPERATING CONDITION: ALL

BASIS: Liquid release rates are determined using Surveillance Instructions which utilize liquid samples rather than instrument readings for activity determination. Effluent Concentration Limits (ECL) are those annual concentrations given in 10CFR20 Appendix B, Table 2, Column 2. 10 times ECL is equivalent to the instantaneous ODCM limit. Unplanned radioactivity releases that exceed 20 times ECL (2 times ODCM limit) and continue for 60 minutes or longer represent an uncontrolled situation and potential degradation in the level of safety of the plant. The release should not be averaged over 60 minutes. For example, a release of 40 times ECL for 30 minutes does not meet the requirements of this event classification. The 60 minute time period allows sufficient time to isolate any release after exceeding ECL. Greater than 60 minutes represents inability to isolate or control the release. The Site Emergency Director should declare the event as soon as it is determined that the release duration has or will likely exceed 60 minutes. The Chemistry Department determines the magnitude of the release by sample procedure for any release as required by initiating procedures (i.e., SI, ARP, AOI, EOI). The sample results are reported to the Site Emergency Director as a fraction or multiple of ECL.

Escalation to Alert is based on release in excess of 2000 times ECL for greater than 15 minutes.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-AU1 example-2)
EDMS L63 010206 800
10CFR20

NOTES:

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-59 Revision 103
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LIQUID EFFLUENT

4.3-A

ALERT

EAL: Liquid release rate exceeds 2000 times ECL as determined by chemistry sample

AND

Release duration exceeds or will exceed 15 minutes.

OPERATING CONDITION: ALL

BASIS: This event escalates from Unusual Event by increasing the magnitude of the release by a factor of 100. The required release duration is reduced to 15 minutes in recognition of the increased severity. The Chemistry Department determines the magnitude of the release by sample procedure for any release as required by initiating procedures (e.g., SI, ARP, AOI, EOI). The sample results are reported to the Site Emergency Director as a fraction or multiple of ECL. 10 times ECL is equal to the ODCM limit; therefore, 200 times the ODCM limit is equivalent to 2000 times ECL.

Escalation to Site Area Emergency is based on event classifications indicative of failure of the Reactor Coolant System pressure boundary and Primary Containment barrier.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-AA1 example-2)
EDMS L63 010206 800

NOTES:

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-60 Revision 103
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LOSS OF POWER

5.0

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-61 Revision 103
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LOSS OF AC POWER

5.1-U

UNUSUAL EVENT

EAL: Loss of normal and alternate supply voltage to ALL unit specific 4KV shutdown boards from Table 5.1 for greater than 15 minutes

AND

At least two Diesel Generators supplying power to unit specific 4KV shutdown boards listing in Table 5.1.

OPERATING CONDITION: ALL

BASIS: Prolonged loss normal and alternate AC power (offsite) reduces required redundancy and potentially degrades the level of safety of the plant by rendering the plant more vulnerable to a complete loss of AC power (station blackout). Note 5.1-U specifies that power sources, being taken credit for in this event classification, must be qualified. Qualified in this case means that the power source is capable of supplying the required shutdown loads under accident conditions. A threshold of 15 minutes is provided to exclude transient or momentary power losses. Table 5.1 lists the unit specific shutdown boards applicable to Unit 1, Unit 2, or Unit 3.

Escalation to Alert is based on additional loss of onsite power (Diesel Generators) such that one additional single failure would result in complete loss of voltage to all unit specific shutdown boards from Table 5.1.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-SU1)
TVA Drawings 0-15E500-1, 0-15E500-2, and 3-15E500-3

NOTES: **5.1-U** - Loss of normal and alternate supply voltage implies inability to restore voltage from any qualified source to normal or alternate feeder for at least one of the unit specific boards within 15 minutes. At least two boards must be energized from Diesel power to meet this classification. If only one board can be energized and that board has only one source of power then refer to EAL 5.1-A1 or 5.1-A2.

CURVES/TABLES:

Table 5.1 UNIT 4KV SHUTDOWN BOARD APPLICABILITY	
APPLICABLE UNIT	APPLICABLE 4KV SHUTDOWN BOARDS
UNIT 1	A, B, C, and D
UNIT 2	A, B, C, and D
UNIT 3	3A, 3B, 3C, and 3D

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-62 Revision 103
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LOSS OF AC POWER 5.1-A1

ALERT

EAL: Loss of voltage to ANY THREE unit specific 4KV shutdown boards from Table 5.1 for greater than 15 minutes

AND

Only ONE source of power available to the remaining board.

OPERATING CONDITION: Mode 1 or 2 or 3

BASIS: This event classification provides an escalation from classification 5.1-U. The condition indicated by this event classification is the degradation of the offsite and onsite power systems such that any additional single failure would result in a station blackout. This condition is indicative of loss of voltage to all but one unit specific 4KV shutdown board and only one power supply available to that remaining 4KV shutdown board from either offsite or onsite power and inability to restore any additional source within 15 minutes using any combination of feeders or sources available to provide redundancy. Credit must only be taken for those Power sources that are analyzed as creditable in the plant design analysis (FSAR).

Escalation to Site Area Emergency is based on loss of the remaining single power source such that a station blackout exists for the affected unit.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-SA5)

NOTES: **5.1-A1** - Only one source of power (Diesel or Offsite) is available to any one of the listed unit specific 4KV Shutdown Boards. No power is available to the three remaining boards.

CURVES/TABLES:

Table 5.1 UNIT 4KV SHUTDOWN BOARD APPLICABILITY	
APPLICABLE UNIT	APPLICABLE 4KV SHUTDOWN BOARDS
UNIT 1	A, B, C, and D
UNIT 2	A, B, C, and D
UNIT 3	3A, 3B, 3C, and 3D

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-63 Revision 103
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LOSS OF AC POWER 5.1-A2

ALERT

EAL: Loss of voltage to ALL unit specific 4KV shutdown boards from Table 5.1 for greater than 15 minutes.

OPERATING CONDITION: Mode 4 or 5 or Defueled

BASIS: Loss of all AC power compromises all plant safety systems requiring electric power including Residual Heat Removal (RHR), Emergency Core Cooling Systems (ECCS), spent fuel heat removal, and the ultimate heat sink. When in Mode 4, Mode 5, or defueled the event can be classified as an Alert because of the significantly reduced decay heat and moderator temperature. The time required to restore one of the boards is not as critical in relation to preventing core damage or reducing other significant threats to the public due to degraded plant conditions.

Escalation to Site Area Emergency is based on abnormal radiation level, radioactive release, or reactor water level event classifications. Refer also to basis for 1.5-A loss of decay heat removal capability.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-SA1)

NOTES: **5.1-A2** - Loss of voltage to all unit specific 4KV Shutdown Boards applies to those boards which normally supply emergency AC power to the affected unit only. Determination of the event classification depends on the affected unit operating mode. For units in operation 5.1-S would apply.

CURVES/TABLES:

Table 5.1 UNIT 4KV SHUTDOWN BOARD APPLICABILITY	
APPLICABLE UNIT	APPLICABLE 4KV SHUTDOWN BOARDS
UNIT 1	A, B, C, and D
UNIT 2	A, B, C, and D
UNIT 3	3A, 3B, 3C, and 3D

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-64 Revision 103
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LOSS OF AC POWER

5.1-S

SITE AREA EMERGENCY

EAL: Loss of voltage to ALL unit specific 4KV shutdown boards from Table 5.1 for greater than 15 minutes.

OPERATING CONDITION: Mode 1 or 2 or 3

BASIS: Loss of all AC power compromises all plant safety systems requiring electric power including Residual Heat Removal (RHR), Emergency Core Cooling Systems (ECCS), containment heat removal, and the ultimate heat sink. Prolonged loss of all AC power will cause core uncover and loss of containment integrity; therefore, this event may lead to General Emergency. Fifteen minutes allows restoration following momentary losses and excludes transient or momentary losses.

Escalation to General Emergency is based on prolonged loss of all AC power (Station Blackout) for greater than the time specified in the site specific station blackout coping analysis.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-SS1)

NOTES: **5.1-S** - Loss of voltage to all unit specific 4KV Shutdown Boards applies to those boards which normally supply emergency AC power to the affected unit only. Determination of the event classification depends on the affected unit operating mode. For units in Shutdown or Refuel 5.1-A2 would apply.

CURVES/TABLES:

Table 5.1 UNIT 4KV SHUTDOWN BOARD APPLICABILITY	
APPLICABLE UNIT	APPLICABLE 4KV SHUTDOWN BOARDS
UNIT 1	A, B, C, and D
UNIT 2	A, B, C, and D
UNIT 3	3A, 3B, 3C, and 3D

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-65 Revision 103
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LOSS OF AC POWER

5.1-G

GENERAL EMERGENCY

EAL: Loss of voltage to ALL unit specific 4KV shutdown boards from Table 5.1

AND

Either of the following conditions exists;

- Restoration of at least one 4KV shutdown board is NOT likely within three hours.
- Adequate core cooling can NOT be assured.

OPERATING CONDITION: Mode 1 or 2 or 3

BASIS: Loss of all AC power compromises all plant safety systems requiring electric power including Residual Heat Removal (RHR), Emergency Core Cooling Systems (ECCS), containment heat removal, and the ultimate heat sink. Prolonged loss of all AC power will cause core uncover, loss of fuel clad, and loss of containment integrity. Adequate core cooling can be assumed to exist when the conditions are met in the EOIs that assure adequate core cooling (Refer to EOI Program Manual, Section I-C). The three-hour time limit is based on the Browns Ferry Station Blackout (SBO) Evaluation (RIMS L44-8904118 814) for station blackout (4 hours) minus 1 hour to ensure timely recognition and emergency response. This event classification is redundant to fission product barrier event classification, but is intended to be an earlier indication of imminent release and allow for more timely emergency response.

The likelihood of restoring at least one 4KV shutdown board should be based on a realistic appraisal of conditions and high probability of restoration. Delaying upgrade from Site Area Emergency based on only a chance of restoration could result in a loss of valuable time in preparation and implementation of public protective actions.

In addition, under these conditions, fission product barrier monitoring capability may be degraded. Indications of continuing core cooling degradation must be assessed and Site Emergency Director judgment must be utilized relative to imminent loss or potential loss of fission product barriers and degraded ability to monitor fission product barriers. Refer to event classification 8.4-G.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-SG1)
RIMS B22 91 1216 102
Station Blackout (SBO) Evaluation (RIMS L44 890418 814)

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-66 Revision 103
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<div> <div>LOSS OF AC POWER</div> <div>5.1-G</div> </div>

GENERAL EMERGENCY
(CONTINUED)

NOTES: **5.1-G** - Loss of voltage to all unit specific 4KV Shutdown Boards applies to those boards which normally supply emergency AC power to the affected unit only.

CURVES/TABLES:

Table 5.1 UNIT 4KV SHUTDOWN BOARD APPLICABILITY	
APPLICABLE UNIT	APPLICABLE 4KV SHUTDOWN BOARDS
UNIT 1	A, B, C, and D
UNIT 2	A, B, C, and D
UNIT 3	3A, 3B, 3C, and 3D

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-67 Revision 103
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LOSS OF 250V DC POWER 5.2-U

UNUSUAL EVENT

EAL: Unplanned loss of 250V DC control power to ALL unit specific 4KV shutdown boards from Table 5.2-U for greater than 15 minutes

OR

Unplanned loss of 250V DC control power to unit specific 480V shutdown boards A and B for greater than 15 minutes.

OPERATING CONDITION: Modes 4 or 5

BASIS: The purpose of this event classification is to recognize a loss of 250V DC power compromising the ability to monitor and control the removal of decay heat during Mode 4 or Mode 5 operations. This event classification is anticipatory to other event classifications in as much as the operator may not have necessary control of equipment to respond to the loss. This event classification is a precursor to loss of decay heat removal capability events.

Loss of control power to the applicable boards prevents operation of vital systems (RHR and Core Spray) which are either required by Technical Specifications or required for decay heat removal. The 15 minute time period is provided to exclude transient or momentary power losses.

Note 5.2 specifies the minimum bus voltage necessary for operation of safety related equipment consideration a reserve capacity for at least 30 minutes of operation.

Escalation to Alert is based on loss of decay heat removal capability and inability to maintain Reactor moderator temperature below 212°F.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-SU7)
Technical Specifications 3.8.4
FSAR 8.6.3

NOTES: **5.2** - 250V DC power voltage below 248 volts constitutes a loss of DC power to the affected board. The voltage readings may be obtained at the 250V Shutdown Battery Board (or the 250V Plant Battery Board) that is feeding the affected board.

CURVES/TABLES:

Table 5.2-U UNIT 4KV SHUTDOWN BOARD APPLICABILITY	
APPLICABLE UNIT	APPLICABLE 4KV SHUTDOWN BOARDS
UNIT 1	A, B, C, AND D
UNIT 2	A, B, C, AND D
UNIT 3	3A, 3B, 3C, AND 3D

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-68 Revision 103
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LOSS OF 250V DC POWER

5.2-S

SITE AREA EMERGENCY

EAL: Loss of 250V DC power to ALL combinations (I, II, III, and IV) of essential systems from Table 5.2-S for greater than 15 minutes.

OPERATING CONDITION: Mode 1 or 2 or 3

BASIS: Loss of DC power to all essential systems will lead to core uncover and loss of Primary Containment integrity when there is significant decay heat and sensible heat in the Reactor System. The combinations of systems listed in Table 5.2-S were chosen based on the individual systems ability to supply makeup water to the Reactor Pressure Vessel. At rated temperature and pressure the Main Steam Relief Valves (MSRVs) are utilized lifting at their respective setpoints to remove decay heat from the Reactor and control Reactor pressure. DC power is only required for MSRV operation in the manual blowdown mode. Four MSRVs is the minimum number of MSRVs required for emergency depressurization. DC power supplies listed in Table 5.2-U are only those DC power sources that will render the listed systems unavailable. Other combinations of DC power sources in combination with AC electrical transients could render these systems unavailable. These other transients are covered under other event classifications (e.g., Reactor water level or loss of decay heat removal events) which are caused by loss of ECCS.

DC power supplies to instrumentation (e.g., ECCS inverters) are covered under loss of control room annunciator and instrumentation events. DC control power to Diesel Generators is covered under loss of onsite AC power because loss of Diesel Generator DC power sources renders emergency Diesel Generators inoperable.

Heat removal from Primary Containment is covered under Suppression Pool temperature and Primary Containment pressure events. Loss of 250V DC power is a precursor to Reactor pressure vessel low level, fission product barrier, radiological release, and radiological effluent events.

This event classification provides redundancy to those events but is warranted because it is anticipatory and allows more time for required actions. The 15 minute time period is provided to exclude transient or momentary power losses.

Note 5.2 specifies the minimum bus voltage necessary for operation of safety related equipment consideration a reserve capacity for at least 30 minutes of operation.

Escalation to General Emergency is based on radiological release, radiological effluent, fission product barrier, or Site Emergency Director judgment event classification.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-SS3)
TVA Drawings 45E701-1, 45E702-1, 45E703-1, and associated listed reference drawings
Technical Specifications 3.8.4
FSAR 8.6.3

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-69 Revision 103
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LOSS OF 250V DC POWER

5.2-S

SITE AREA EMERGENCY (CONTINUED)

NOTES:

5.2 - 250V DC power voltage below 248 volts constitutes a loss of DC power to the affected board. The voltage readings may be obtained at the 250V Shutdown Battery Board (or the 250V Plant Battery Board) that is feeding the affected board.

CURVES/TABLES:

Table 5.2-S CRITICAL DC POWER AND ESSENTIAL SYSTEMS		
COMBINATION	LOSS OF CRITICAL 250V DC POWER (Unit Specific Unless Otherwise Noted)	POTENTIALLY RESULTS IN
I	Control Power for 4KV Unit Boards A, B, and C AND Control Power for 480V Unit Boards A and B AND Power for Panel 9-9 Cabinet 1	Loss of Main Condenser AND Loss of Both EHC Pumps AND Loss of All Reactor Feed Pumps
II	Power for 250V DC RMOV Board A	Loss of HPCI
III	Power for 250V DC RMOV Board C	Loss of RCIC
IV	Power for 250V DC RMOV Boards A, B, and C AND Control Power for 4KV Shutdown Boards A, B, C, and D (4KV Shutdown Boards 3A, 3B, 3C, and 3D for Unit 3)	Less than 4 MSRVs AND Loss of All RHR Pumps And Core Spray Pumps

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-70 Revision 103
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HAZARDS

6.0

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-71 Revision 103
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RADIOLOGICAL

6.1-U

UNUSUAL EVENT

EAL: Valid, unexpected increase of ANY in-plant ARM reading to 1000 mrem/hr (except TIP room).

OPERATING CONDITION: ALL

BASIS: This event classification addresses unexpected increases in plant radiation levels that represent a degradation in the control of radioactive material, and constitute a potential degradation in the level of safety of the plant.

The term valid means that the instrument reading can be confirmed by other plant instrumentation indications is consistent with an on-going transient or unplanned event, or that the condition is verified by Radiation Protection. The term unexpected implies an increase not attributable to an anticipated transient such as a radwaste resin transfer, radiography, calibration activity, etc.

The Control Room is not included because it is an area which requires continuous occupancy and is covered under event classification 6.1-A.2. The TIP Room is not included because high ARM readings may not necessarily indicate a degradation in the control of radioactive materials and no personnel actions are required in the TIP area to safely shutdown the plant or maintain safe shutdown.

Escalation to Alert is based on operations required in the affected area(s) to safely shutdown the plant or maintain safe shutdown.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-AU2 EXAMPLE 4)
RIMS R38 940916 970

NOTES:

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-72 Revision 103
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RADIOLOGICAL

6.1-A1

ALERT

EAL: Valid, unexpected increase of ANY in-plant ARM reading to 1000 mrem/hr (except TIP room).

AND

Personnel required in the affected area(s).

OPERATING CONDITION: ALL

BASIS: This event classification addresses increased radiation levels that may impede necessary access to critical operating areas when conditions necessitate equipment operation or maintenance in those areas in order to maintain safe operation or achieve safe shutdown. The impaired ability to perform inplant operations represents an actual or potential degradation in the level of safety of the plant. It is appropriate to ensure that additional personnel are onsite to perform necessary operations and maintenance and provide proper approvals, surveys, and radiation protection; therefore, the alert classification is justified.

The term valid means that the instrument reading can be confirmed by other plant instrumentation indications or is verified by Radiation Protection. The term unexpected implies an increase not attributable to an anticipated transient such as a radwaste resin transfer, radiography, calibration activity, etc.

The Control Room is not included because it is an area which requires continuous occupancy and is covered under event classification 6.1-A.2. The TIP Room is not included because no personnel actions are required in the area to safely shutdown the plant or maintain safe shutdown.

Escalation to Site Area Emergency or General Emergency is based on Secondary Containment radiation, radiological release, or fission product barrier event classifications.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-AA3 EXAMPLE 2)
RIMS R38 940916 970

NOTES:

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-73 Revision 103
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RADIOLOGICAL

6.1-A2

ALERT

EAL: Control Room radiation levels greater than 15 mrem/hr.

OPERATING CONDITION: ALL

BASIS: This event classification applies to areas requiring continuous occupancy to maintain safe operation or safely shutdown the plant. At Browns Ferry the Control Room is the only area within the plant that should require continuous manning under these conditions. Other areas (e.g., TSC, OSC) can be relocated to an area of lower dose. Radwaste operations can be suspended and the Radwaste Control Room can be evacuated. The Central Alarm Station (CAS) is not included because of the location at the plant entrance away from the main building. It would not be possible to achieve these levels at the CAS without already reaching alert classification through radiological release.

This event classification may be redundant to fission product barrier and radiological release event classifications; however the cause of the event is not a concern. This event classification is only intended to address exposures to personnel who must be present at required operating stations for long term operation.

Escalation to Site Area Emergency or General Emergency is based on Secondary Containment radiation, radiological release, or fission product barrier event classifications.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-AA3 EXAMPLE 1)

NOTES:

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-74 Revision 103
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CONTROL ROOM EVACUATION

6.2-A

ALERT

EAL: Control Room Abandonment from entry into 1, 2, or 3-AOI-100-2 or 0-SSI-16 for ANY Unit Control Room.

OPERATING CONDITION: ALL

BASIS: With the control room evacuated, additional support, monitoring, and direction through the Technical Support Center and/or other Emergency Operations Centers may be necessary. The Alert declaration ensures centers are manned to provide the necessary additional support.

Escalation to Site Area Emergency is based on inability to establish plant control from outside the control room within 20 minutes.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-HA5)

NOTES:

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-75 Revision 103
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CONTROL ROOM EVACUATION

6.2-S

SITE AREA EMERGENCY

EAL: Control Room Abandonment from entry into 1, 2, or 3-AOI-100-2 or 0-SSI-16 for ANY Unit Control Room

AND

Control of reactor water level, reactor pressure, and reactor power (for Modes 1, or 2, or 3) or decay heat removal (for Modes 4, or 5) per 1, 2, or 3-AOI-100-2 or 0-SSI-16 as applicable, can NOT be established within 20 minutes after evacuation is initiated.

OPERATING CONDITION: ALL

BASIS: This event classification is intended to recognize loss of control of critical parameters either by failure of equipment designed to automatically initiate for control of the parameter or failure to expeditiously transfer safety system control to the backup controls. Fission product barrier damage may not yet be indicated but should be considered by assessing available parameters versus the status of safety systems and the ability to control critical parameters. In Mode 4 and Mode 5 operator concern should be directed towards maintaining core cooling using decay heat removal systems. In power operation, hot standby, and hot shutdown operator concern is primarily directed toward maintaining critical parameters, (i.e., level, pressure, power, and heat sink) and thereby assuring fission product barrier integrity.

The 20 minute time period is based on time required for personnel to leave the control room, arrive at the appropriate backup control station, and take control of critical parameters before core uncover or core damage has occurred. This timeframe has been projected within the Tennessee Valley Authority, Browns Ferry Nuclear Plant, Fire Protection Report. During execution of procedures and transfer of equipment control, the listed critical parameters may be considered as being controlled if the parameters can be verified as being maintained within safe value ranges by appropriate equipment and automatic initiation functions designed to control the parameter (example: HPCI auto initiated and raised RPV level to a value above the initiation setpoint.).

Escalation to General Emergency is by fission product barrier degradation radioactivity release, or Emergency Director Judgment event classification.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-HS2)
Fire Protection Report

NOTES:

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-76 Revision 103
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TURBINE FAILURE

6.3-U

UNUSUAL EVENT

EAL: Turbine failure resulting in casing penetration

OR

Significant damage to turbine or generator seals during operation.

OPERATING CONDITION: Mode 1, or 2

BASIS: This event classification is intended to address Main Turbine rotating component failures of sufficient magnitude to cause observable damages to the Turbine casing or to the seals of the Turbine or Generator. Of major concern is the potential for leakage of combustible fluids (lubricating oil) and gases (hydrogen) into the plant and environs. Actual fires and flammable gas buildup are classified under other event classification. This event classification is consistent with unusual event while maintaining the anticipatory nature desired and recognizing the risk to nonsafety related equipment.

Escalation to higher event classification is based on potential damage to safety related equipment from missiles generated by the failure or by radioactivity release.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-HU1 EXAMPLE 6)

NOTES:

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-77 Revision 103
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TURBINE FAILURE

6.3-A

ALERT

EAL: Turbine failure resulting in visible structural damage to or visible penetration of ANY of the following structures from missiles:

- ♦Reactor Building
- ♦Intake Structure
- ♦Diesel Generator Building
- ♦Control Bay

OPERATING CONDITION: Mode 1 or 2

BASIS: This event classification is intended to address the threat to safety-related equipment imposed by missiles generated by Main Turbine rotating component failures. Areas included are plant areas containing safety related equipment required to safely operate or safely shutdown the plant. The Alert classification assures adequate personnel are available to perform thorough assessment of damage to structures and equipment in the affected areas.

Escalation to higher event classification is based on system malfunction, fission product barrier degradation, radioactivity release, or Emergency Director Judgment event classifications.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-HA 1 EXAMPLE 6)

NOTES:

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-78 Revision 103
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FIRE / EXPLOSION

6.4-U1

UNUSUAL EVENT

EAL: Confirmed fire in ANY plant area listed in Table 6.4-U1
AND
 NOT extinguished within 15 minutes.

OPERATING CONDITION: ALL

BASIS: The purpose of this event classification 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 with no safety significance. This event classification applies to plant vital areas, buildings and areas contiguous to plant vital areas, or areas which have the potential to cause significant release of radioactive material such as Radwaste. These areas are included in Table 6.4-U1.

Confirmation of fire includes those actions listed in the appropriate Alarm Response Procedure (ARP) to verify the alarm is not spurious or by visual observation by personnel in the field. If confirmation cannot be positively ascertained within 15 minutes and symptoms indicative of a fire persists then confirmation should be assumed and this event classification declared.

Allowance of fifteen minutes for extinguishment is provided to exclude small fires, easily extinguishable, with no significant safety consequence.

Escalation to Alert is based on fire affecting the operability of plant safety systems required for the current operating condition.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-HU2)
 Browns Ferry Nuclear Plant Safe Shutdown Program

NOTES:

CURVES/TABLES:

Table 6.4-U1 APPLICABLE PLANT AREA
Reactor Building
Refuel Floor
4KV Shutdown Board Rooms
4KV Shutdown Battery Board Rooms
480V Shutdown Board Rooms
3A And 3B RMOV Board Rooms
4KV Bus Tie Board Room
Control Bay Elevation 593', 606', And 617'
Diesel Generator Buildings (All Elevations)
Turbine Building (All Elevations)
Intake Pumping Station (All Elevations)
Radwaste Building (All Elevations)
Cable Tunnel (Intake To Turbine Building)
Standby Gas Treatment Building

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-79 Revision 103
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FIRE / EXPLOSION

6.4-U2

UNUSUAL EVENT

EAL: Unanticipated explosion within the protected area resulting in visible damage to ANY permanent structure or equipment.

OPERATING CONDITION: ALL

BASIS: The purpose of this event classification is to recognize only those explosions of sufficient force to damage permanent structures or equipment within the protected area. An explosion is defined as rapid, violent, unconfined combustion, or a catastrophic failure of pressurized equipment that potentially imparts significant energy to nearby structures and materials. This event classification makes no attempt to assess the actual magnitude of the damage. The occurrence of the explosion with reports of evidence of damage (e.g., deformation and scorching) is sufficient for declaration. The Site Emergency Director also needs to consider any security aspects of the explosion if applicable.

Escalation to Alert is based on explosion affecting safety system performance or causing visible damage to structures or equipment required for safe shutdown.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-HU1 EXAMPLE 5)

NOTES:

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-80 Revision 103
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FIRE / EXPLOSION 6.4-A

ALERT

EAL: Fire or explosion in ANY plant area listed in Table 6.4-A affecting safety system performance

OR

Fire or explosion causing visible damage to permanent structure of safety systems in ANY plant area listed in Table 6.4-A.

OPERATING CONDITION: ALL

BASIS: This event classification is intended to address the magnitude and extent of fires that potentially or actually affect one or more redundant trains of safety systems or structures containing safety systems. Areas listed in Table 6.4-A are those plant areas that contain systems or functions required for the safe shutdown of the plant.

With regard to explosions, only those explosions of sufficient force to damage permanent structures or equipment required for safe operation within the identified plant area should be considered. An explosion is defined as a rapid, violent, unconfined combustion, or a catastrophic failure of pressurized equipment that potentially imparts significant energy to nearby structures and materials. The occurrence of an explosion with reports of visible damage is sufficient evidence for determination of this event. No attempt should be made to perform a detailed assessment before declaration is considered. Declaration of Alert with subsequent manning of support personnel will provide adequate personnel to make a detailed assessment. The Site Emergency Director should also consider any security aspects of the explosion.

Escalation to Site Area Emergency is based on system malfunction, fission product barrier degradation, radioactivity release, or Emergency Director Judgment.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-HA2)
Browns Ferry Nuclear Plant Safe Shutdown Program

NOTES:

CURVES/TABLES:

Table 6.4-A APPLICABLE PLANT AREA
Reactor Building
Refuel Floor
4KV Shutdown Board Rooms
4KV Shutdown Battery Board Rooms
480V Shutdown Board Rooms
3A And 3B RMOV Board Rooms
4KV Bus Tie Board Room
Control Bay Elevation 593', 606', And 617'
Diesel Generator Buildings (All Elevations)
Intake Pumping Station (All Elevations)
Cable Tunnel (Intake To Turbine Building)
Standby Gas Treatment Building

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-81 Revision 103
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TOXIC GASES

6.5-U

UNUSUAL EVENT

EAL:

EITHER of the following conditions exists:

- Normal operations impeded due to access restrictions caused by toxic gas concentrations within any building or structure listed in Table 6.5/6.6.
- Confirmed report by local, county, or state officials that a large offsite toxic gas release has occurred within one mile of the site with potential to enter the site boundary in concentrations at or above the Permissible Exposure Limit (PEL) causing an evacuation of any site personnel.

OPERATING CONDITION: ALL

BASIS:

This event classification is based on significant releases of toxic gases that could affect the health and safety of plant personnel or affect the safe operation of the plant with the plant being within the evacuation area of an offsite event (i.e., tanker truck or barge accident releasing toxic or flammable gas, etc.). The evacuation area is determined from the Department of Transportation (DOT) Evacuation Tables for selected hazardous materials, in the DOT Emergency Response Guide for hazardous materials.

Table 6.5/6.6 contains plant vital areas, buildings and areas contiguous to plant vital areas, and other significant plant buildings or structures where operations may be required to assure safe operation of the plant.

Escalation to Alert is based on gases entering plant structures and affecting safe operation of the plant.

REFERENCES:

Reg Guide 1.101 Rev. 3, (NUMARC-HU3)

NOTES:

CURVES/TABLES:

Table 6.5/6.6 APPLICABLE PLANT AREA
Reactor Building
Refuel Floor
Control Bay
Diesel Generator Buildings
Turbine Building
Intake Pumping Station
Radwaste Building
Cable Tunnel (Intake To Turbine Building)
Standby Gas Treatment Building

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-82 Revision 103
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TOXIC GASES

6.5-A

ALERT

EAL:

ALL of the following conditions exist:

- Plant personnel report toxic gas within any building or structure listed in Table 6.5/6.6.
- Plant personnel report severe adverse health reactions due to toxic gas (i.e., burning eyes, throat, or dizziness), or sampling results by Fire Protection or Industrial Safety personnel indicate levels above the Permissible Exposure Limit (PEL).
- Determination by the Site Emergency Director that plant personnel would be unable to perform actions necessary to establish and maintain cold shutdown conditions while utilizing appropriate personnel protective equipment.

OPERATING CONDITION: ALL

BASIS:

This event classification is based on toxic gases that have entered a plant structure affecting safe operation of the plant. This event classification applies to buildings and areas contiguous to plant vital areas or other significant buildings or areas (i.e., intake and Standby Gas Treatment buildings). The intent of this event classification is not to include buildings (i.e., warehouses and administration buildings) or other areas that are not contiguous or immediately adjacent to plant vital areas. It is appropriate that increased monitoring be performed to ascertain whether consequential damage has occurred.

Escalation to higher emergency class is based on system malfunction, fission product barrier degradation, radioactivity release, or Emergency Director Judgment event classification.

REFERENCES:

Reg Guide 1.101 Rev. 3, (NUMARC-HA3)

NOTES:

CURVES/TABLES:

Table 6.5/6.6 APPLICABLE PLANT AREA
Reactor Building
Refuel Floor
Control Bay
Diesel Generator Buildings
Turbine Building
Intake Pumping Station
Radwaste Building
Cable Tunnel (Intake To Turbine Building)
Standby Gas Treatment Building

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-83 Revision 103
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FLAMMABLE GASES

6.6-U

UNUSUAL EVENT

EAL:

EITHER of the following conditions exists:

- Release of flammable gas within the site boundary in concentrations at or above 25% of the Lower Explosive Limit (LEL) for any three readings obtained in a 10 ft. triangular area as indicated by Fire Protection or Industrial Safety personnel using appropriate monitoring instrumentation.
- Confirmed report by local, county, or state officials that a large offsite flammable gas release has occurred within one mile of the site with potential to enter the site boundary in concentrations at or above 25% of the Lower Explosive Limit (LEL).

OPERATING CONDITION: ALL

BASIS:

This event classification is based on significant releases of flammable gases that could affect the safe operation of the plant with the plant being within the evacuation area of an offsite event (i.e., tanker truck or barge accident releasing toxic or flammable gas, etc.). The evacuation area is determined from the Department of Transportation (DOT) Evacuation Tables for selected hazardous materials, in the DOT Emergency Response Guide for hazardous materials.

Escalation to Alert is based on flammable gases entering plant structures and affecting safe operation of the plant.

REFERENCES:

Reg Guide 1.101 Rev. 3, (NUMARC-HU3)

NOTES:

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-84 Revision 103
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FLAMMABLE GASES	6.6-A
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ALERT

EAL: Release of flammable gases within any building or structure listed in Table 6.5/6.6 in concentrations at or above 25% of the Lower Explosive Limit (LEL) for any three readings obtained in a 10 ft. triangular area as indicated by Fire Protection or Industrial Safety personnel using appropriate monitoring instrumentation.

OPERATING CONDITION: ALL

BASIS: This event classification is based on flammable gases that have entered a plant structure with potential to affect safe operation of the plant. This event classification applies to buildings and areas contiguous to plant vital areas or other significant buildings or areas (i.e., intake and Standby Gas Treatment buildings). The intent of this event classification is not to include buildings (i.e., warehouses and administration buildings) or other areas that are not contiguous or immediately adjacent to plant vital areas. It is appropriate that increased monitoring be performed to ascertain whether consequential damage has occurred.

Escalation to higher emergency class is based on system malfunction, fission product barrier degradation, radioactivity release, or Emergency Director Judgment event classification.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-HA3 EXAMPLE 1&2)

NOTES:

CURVES/TABLES:

Table 6.5/6.6 APPLICABLE PLANT AREA
Reactor Building
Refuel Floor
Control Bay
Diesel Generator Buildings
Turbine Building
Intake Pumping Station
Radwaste Building
Cable Tunnel (Intake To Turbine Building)
Standby Gas Treatment Building

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-85 Revision 103
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SECURITY

6.7-U

UNUSUAL EVENT

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

EAL:

1. A SECURITY CONDITION that does NOT involve a HOSTILE ACTION as reported by the Security Shift Supervisor.

OR

2. A credible Browns Ferry threat notification

OR

3. A validated notification from NRC providing information of an aircraft threat.

OPERATING CONDITION: ALL

BASIS:

Security events which do not represent a potential degradation in the level of safety of the plant are reported under 10 CFR 73.71 or in some cases under 10 CFR 50.72. Security events assessed as HOSTILE ACTIONS are classifiable under 6.7-A, 6.7-S, and 6.7-G.

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

EAL #1

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

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

EAL #2

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

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

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-86 Revision 103
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UNUSUAL EVENT
(CONTINUED)

EAL #3

The intent of this EAL is to ensure that notifications for the aircraft threat are made in a timely manner and that OROs and plant personnel are at a state of heightened awareness regarding the credible threat. It is not the intent of this EAL to replace existing non-hostile related EALs involving aircraft.

This EAL is met when a plant receives information regarding an aircraft threat from NRC. Validation is performed by calling the NRC or by other approved methods of authentication. Only the plant to which the specific threat is made need declare the Unusual Event.

The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an airliner (airliner is meant to be a large aircraft with the potential for causing significant damage to the plant). The status and size of the plane may be provided by NORAD through the NRC.

Escalation to Alert emergency classification level would be via 6.7-A would be appropriate if the threat involves an airliner within 30 minutes of the plant.

REFERENCES:

NEI 99-01 R5 HU4
Browns Ferry Physical Security/Contingency Plan

NOTES:

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

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-87 Revision 103
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SECURITY

6.7-A

ALERT

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

EAL:

1. A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervisor.

OR

2. A validated notification from NRC of an airliner attack threat within 30 minutes of the site.

OPERATING CONDITION: ALL

BASIS:

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

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

EAL #1

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

EAL #2

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

The intent of this EAL is to ensure that notifications for the airliner attack threat are made in a timely manner and that OROs and plant personnel are at a state of heightened awareness regarding the credible threat. Airliner is meant to be a large aircraft with the potential for causing significant damage to the plant.

This EAL is met when a plant receives information regarding an airliner attack threat from NRC and the airliner is within 30 minutes of the plant. Only the plant to which the specific threat is made need declare the Alert.

The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an airliner (airliner is meant to be a large aircraft with the potential for causing significant damage to the plant). The status and size of the plane may be provided by NORAD through the NRC.

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-88 Revision 103
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ALERT
CONTINUED

REFERENCES: NEI 99-01 R5 HA4

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

EAL is applicable for any HOSTILE ACTION occurring, or that has occurred, in the OWNER CONTROLLED AREA. This includes ISFSI's that may be outside the PROTECTED AREA but still within the OWNER CONTROLLED AREA.

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-89 Revision 103
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SECURITY

6.7-S

SITE AREA EMERGENCY

HOSTILE ACTION within the PROTECTED AREA.

EAL: A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervisor

OPERATING CONDITION: ALL

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

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

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

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

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

REFERENCES: NEI 99-01 R5 HS4

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

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-90 Revision 103
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SECURITY

6.7-G

GENERAL EMERGENCY

HOSTILE ACTION resulting in loss of physical control of the facility.

EAL:

1. A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain safety functions.

OR

2. A HOSTILE ACTION has caused failure of Spent Fuel Cooling Systems and IMMINENT fuel damage is likely for a freshly off-loaded reactor core in pool.

OPERATING CONDITION: ALL

BASIS:

EAL #1

This EAL encompasses conditions under which a HOSTILE ACTION has resulted in a loss of physical control of VITAL AREAS (containing vital equipment or controls of vital equipment) required to maintain safety functions and control of that equipment cannot be transferred to and operated from another location.

If control of the plant equipment necessary to maintain safety functions can be transferred to another location, then the threshold is not met.

EAL #2

This EAL addresses failure of spent fuel cooling systems as a result of HOSTILE ACTION if IMMINENT fuel damage is likely, such as when a freshly off-loaded reactor core is in the spent fuel pool.

REFERENCES:

NEI 99-01 R5. HG1

NOTES:

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

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-91 Revision 103
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VEHICLE CRASH

6.8-U

UNUSUAL EVENT

EAL: Vehicle crash (for example; aircraft or barge) into plant structures or systems within the protected area boundary.

OPERATING CONDITION: ALL

BASIS: This event classification is intended to address such items as plane, helicopter, or barge crash that may potentially damage plant structures containing functions and systems required for safe shutdown of the plant.

The crash should be of sufficient impact to cause structural damage to a plant system or structure. Visual observation of structural damage is sufficient to trigger this event classification.

Escalation to higher emergency class is based on crash into any plant area affecting equipment required for safe shutdown or by system malfunction event classifications.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-HU1 EXAMPLE 4)

NOTES:

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-92 Revision 103
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VEHICLE CRASH

6.8-A

ALERT

EAL: Vehicle crash (for example; aircraft or barge) into ANY plant vital area.

OPERATING CONDITION: ALL

BASIS: This event classification is intended to address such items as plane, helicopter, or barge crash into a plant vital area. In cases where structural damage has occurred it may be assumed that the area and associated equipment has been subjected to forces beyond design limits and thus damage may be assumed to have occurred to plant safety systems.

It should not be interpreted that a lengthy damage assessment is necessary prior to classification. Declaration of Alert with subsequent manning of support personnel will provide adequate personnel to make a detailed assessment. The Site Emergency Director should also consider any security aspects of the crash.

Escalation to higher emergency classification is based on system malfunction, fission product barrier degradation, radioactivity release, or Emergency Director Judgment event classification.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-HA1 EXAMPLE 5)

NOTES:

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-93 Revision 103
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SPENT FUEL STORAGE

6.9-U

UNUSUAL EVENT

EAL: Damage to a loaded cask CONFINEMENT BOUNDARY from ANY of the following:

- Natural phenomena (e.g., seismic event, tornado, flood, lightning, snow/ice accumulation, etc.)
- Accident (e.g., dropped cask, tipped over cask, explosion, missile damage, fire damage, burial under debris, etc.).
- Judgement of the Site Emergency Director that the CONFINEMENT BOUNDARY damage is a degradation in the level of safety of the ISFSI.

OPERATING CONDITION: ALL

BASIS: This event classification is intended to address those events of sufficient magnitude that a loaded cask CONFINEMENT BOUNDARY is damaged or violated. This includes CONFINEMENT BOUNDARY loss leading to the degradation of the fuel during storage or posing an operational safety problem with respect to removal from storage. Such damage may result from natural phenomena or accident conditions.

This classification is also intended for any condition not otherwise detailed which, in the judgment of the Site Emergency Director, is a potential degradation in the level of safety of the ISFSI.

REFERENCES: NUMARC/NESP-007, January 2003, (E-HU1)

NOTES:

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-94 Revision 103
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NATURAL EVENTS

7.0

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-95 Revision 103
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EARTHQUAKE

7.1-U

UNUSUAL EVENT

EAL: Valid annunciation in Unit 1 Control Room, Panel 1-XA-55-22C, Window 5,
START OF STRONG MOTION ACCELEROGRAPH

AND

Assessment by Unit One and Two Control Room personnel that an earthquake has occurred.

OPERATING CONDITION: ALL

BASIS: The purpose of this event classification is to recognize an earthquake of low intensity that should not affect the ability of safety systems to function. Window 5 on Panel 1-XA-55-22C alarms at $\geq .01g$ Triaxial acceleration to alert the operator of seismic activity.

The assessment by Control Room personnel includes a determination considering the following:

- Apparent ground motion
- Report by other plant personnel
- Verification from the National Earthquake Center (refer to 0-AOI-100-5, Earthquake).

Escalation to Alert is based on the occurrence of an earthquake greater in magnitude than that analyzed in the Final Safety Analysis Report (FSAR) as design basis earthquake [.10g Horizontal Operating Basis Earthquake (OBE)] or system malfunction event classification.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-HU1 EXAMPLE 1)
Browns Ferry Nuclear Plant, FSAR, Volume 1, Section 2.5.5.1
0-AOI-100-5, Earthquake

NOTES:

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-96 Revision 103
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EARTHQUAKE

7.1-A

ALERT

EAL: Valid annunciation in the Unit 1 Control Room, Panel 1-XA-55-22C, Window 6,
1/2 SSE RESPONSE SPECTRUM EXCEEDED

AND

Assessment by Unit One and Two Control Room personnel that an earthquake has occurred.

OPERATING CONDITION: ALL

BASIS: The purpose of this event classification is to recognize an earthquake of greater magnitude (.10g) than the Operating Basis Earthquake (OBE) described in the Final Safety Analysis Report (FSAR). Seismic events of this magnitude can cause damage to plant safety systems or functions. This equipment may be assumed to have been damaged when subjected to forces beyond design limits. Actuation of the emergency centers ensures additional personnel are available to make a subsequent, more thorough, assessment of damage to structures or equipment.

The assessment by Control Room personnel includes a determination considering the following:

- Apparent ground motion
- Report by other plant personnel
- Verification from the National Earthquake Center (refer to 0-AOI-100-5, Earthquake).

Escalation to higher emergency classification is based on system malfunction, fission product barrier degradation, radioactivity release, or Emergency Director Judgment.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC HA1 EXAMPLE 1)
Browns Ferry Nuclear Plant, FSAR Volume 1, Section 2.5.5.1 and Volume 7, Section 12.2
0-AOI-100-5, Earthquake

NOTES:

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-97 Revision 103
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TORNADO / HIGH WINDS	7.2-U
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UNUSUAL EVENT

EAL: Report by plant personnel of tornado striking within the protected area boundary.

OPERATING CONDITION: ALL

BASIS: This event classification is intended to recognize that a tornado touching down within the protected area boundary may have potentially damaged plant structures containing functions or systems required for safe shutdown of the plant.

Escalation to higher emergency class is based on confirmation of damage to plant structures containing functions or systems required for safe shutdown or system malfunction event classifications.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC HU1 EXAMPLE 2)

NOTES:

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-98 Revision 103
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TORNADO / HIGH WINDS

7.2-A

ALERT

EAL: Tornado striking plant vital area

OR

Onsite wind speed above 90 MPH as indicated using the meteorological data screen of the Integrated Computer System (ICS).

OPERATING CONDITION: ALL

BASIS: This event classification addresses wind loads that may have exceeded the design basis wind loads for plant structures containing functions or systems required for safe shutdown.

Environmental data is detected by instrumentation at the meteorological tower, transmitted to the plant, and displayed by the Integrated Computer System. The 90 MPH value was chosen because it is below the 100 MPH design basis value and within the upper range of the instrument scale.

The Alert classification is appropriate because manning of emergency centers will ensure adequate personnel to perform thorough damage assessment of structures and equipment.

Escalation to higher emergency classification is based on system malfunction, fission product barrier degradation, radioactivity release, or Emergency Director Judgment event classifications.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC HA1 EXAMPLE 2)
Browns Ferry Nuclear Plant, FSAR Volume 7, Section 12.2 and Volume 1, Section 2.3

NOTES:

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-99 Revision 103
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<div>FLOOD</div> <div>7.3-U</div>

UNUSUAL EVENT

EAL: Wheeler Lake level exceeds or is predicted to exceed elevation 565 feet.

AND

Water entering permanent plant structures due to flooding.

OPERATING CONDITION: ALL

BASIS: This event classification is intended to address situations that are precursors to more serious events. Water entering any permanent plant structure can cause equipment malfunctions or failures which lead to degraded plant conditions or potential degradation in the level of safety of the plant. Elevation 565 feet corresponds to the elevation of the intake pumping station deck and access passages into most permanent plant structures.

Escalation to Alert is based on lake level exceeding or predicted to exceed elevation 565 feet and breach or failure of any water tight structure or affecting equipment required for safe shutdown.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC HU1 EXAMPLE 7)
Browns Ferry Nuclear Plant, FSAR, Volume 7, Section 12.2

NOTES:

TABLES/CURVES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-100 Revision 103
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FLOOD

7.3-A

ALERT

EAL: Wheeler Lake level exceeds or is predicted to exceed elevation 565 feet.

AND

EITHER of the following conditions exists:

- Breach or failure of any water-tight structure is causing flooding of the structure
- Equipment required for safe shutdown is affected.

OPERATING CONDITION: ALL

BASIS: This event classification is intended to address flooding that may affect equipment or components required for safe shutdown of the plant. Water entering watertight structures will lead to equipment failures if the structure cannot be sealed and places the plant outside the design analysis for flooding representing a potential degradation in the level of safety of the plant. Actual indication of equipment being affected which is required for safe shutdown is an indication of actual degradation in the level of safety of the plant.

Escalation to higher event classification is based on system malfunction, fission product barrier degradation, radioactivity release, or Emergency Director Judgment event classifications.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC HA1 EXAMPLE 7)
Browns Ferry Nuclear Plant, FSAR, Volume 7, Section 12.2

NOTES:

TABLES/CURVES:

EMERGENCY DIRECTOR JUDGMENT 8.0

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-102 Revision 103
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TECHNICAL SPECIFICATIONS

8.1-U

UNUSUAL EVENT

EAL: Inability to reach required shutdown condition (Mode 3 or Mode 4) within Technical Specification Limiting Conditions for Operation (LCO) limits.

OPERATING CONDITION: Mode 1 or 2 or 3

BASIS: Most Technical Specification LCOs require the plant to be brought to a required operating condition when Technical Specification required configuration cannot be restored within a specified time. Depending on the circumstances, this may or may not be an emergency or precursor to a more serious condition. The initiation of plant shutdown required by plant Technical Specifications requires a report under 10 CFR 50.72(b) non-emergency events, but does not of itself meet any EPIP EAL. The plant is within its safety envelope when being shutdown 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 shutdown condition (Mode 3 or Mode 4) 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 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 event classifications.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC SU2)
Technical Specifications

NOTES:

TABLES/CURVES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-103 Revision 103
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LOSS OF COMMUNICATIONS

8.2-U

UNUSUAL EVENT

EAL: Unplanned loss of onsite communication listed in Table 8.2-U that defeats the Plant Operations Staff's ability to perform routine operations

OR

Unplanned loss of ALL off-site communication listed in Table 8.2-U.

OPERATING CONDITION: ALL

BASIS: The purpose of this event classification is to recognize a loss of communications capability that either defeats the Plant Operations Staff's ability to perform routine tasks necessary for plant operations or the ability to communicate with offsite authorities. Table 8.2-U contains all credible means of routine and emergency onsite and offsite communications.

Offsite communications loss encompasses all means of communications with offsite authorities or support organizations. This part of the event classification is intended to exist only when extraordinary means are being utilized to make communication possible (i.e., relaying of messages or sending individuals to offsite locations with messages).

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC SU6)

NOTES:

CURVES/TABLES:

Table 8.2-U LOSS OF COMMUNICATIONS	
Onsite Communications	Offsite Communication
Plant Phone System Node 1	Bell Lines
Two-Way Radio System (NSS 1, NSS 2, OPS F2, and OPS F4)	Digital Microwave
Sound Power Phones	NRC Emergency Telecommunication System
Nextel Communication System	Cellular Phones (If Available)
	Health Physics Radio Network

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-104 Revision 103
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LOSS OF ASSESSMENT CAPABILITY 8.3-U

UNUSUAL EVENT

EAL: Unplanned loss of most or all safety system annunciators or indicators which causes a significant loss of plant assessment capability for greater than 15 minutes

AND

Compensatory non-alarming safety system indications are available (SPDS, ICS)

AND

In the opinion of the Shift Manager, increased surveillance is required to safely operate the plant.

OPERATING CONDITION: MODE 1, or 2, or 3

BASIS: This event classification is 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 and should be based on the Site Emergency Director's Judgment that there is increased risk that a degraded plant condition could go undetected. If in the opinion of the Site Emergency Director additional personnel are required to provide increased monitoring of system or plant operation to safely operate the unit(s) then this event classification should be considered and subsequent declaration of event should be based upon whether a degradation in the level of safety of the plant is represented by the extent of the loss in assessment capability. This judgment should take into consideration those annunciators identified in Alarm Response Procedures (ARPs), Abnormal Operating Instructions (AOIs), Emergency Operating Instructions (EOIs), Emergency Preparedness Implementing Procedures (EPIPs), and the operator's ability to recognize entry conditions based on symptoms that require or support procedure execution.

Individual system annunciators and/or indicators may impact the function of system or component operability. System and component operability are addressed by Technical Specifications. Inoperability of multiple safety systems beyond Technical Specification compliance is addressed at 8.1-U.

Unplanned excludes loss of annunciators due to scheduled maintenance or testing activities. Compensatory non-alarming safety system indications include the Integrated Computer System (ICS) and SPDS.

15 minutes was chosen as a threshold to exclude transient or momentary power losses.

Escalation to Alert is based on a transient or significant power change required or in progress or loss of compensatory non-alarming indications.

REFERENCES:

Reg Guide 1.101 Rev. 3, (NUMARC SU3)

NOTES:

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-105 Revision 103
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LOSS OF ASSESSMENT CAPABILITY 8.3-A

ALERT

EAL: Unplanned loss of most or all safety system annunciators or indicators which causes a significant loss of plant assessment capability for greater than 15 minutes

AND

In the opinion of the Shift Manager, increased surveillance is required to safely operate the plant

AND

EITHER of the following conditions exists:

- Compensatory non-alarming safety system indications are NOT available (SPDS, ICS)
- A significant transient is in progress.

OPERATING CONDITION: MODE 1, or 2, or 3

BASIS: This event classification is intended to recognize the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation and indication equipment during a transient.

Quantification of most is arbitrary and should be based on the Site Emergency Director's Judgment that there is increased risk that a degraded plant condition could go undetected. If in the opinion of the Emergency Director, additional personnel are required to provide increased monitoring of systems or plant operation to safely operate the units and either compensatory monitoring indications are not available or a significant transient is in progress then this event classification should be considered. Subsequent declaration of event should be based upon whether a substantial degradation in the level of safety of the plant has occurred due the extent of the loss of assessment capability.

This judgment should take into consideration those annunciators identified in Alarm Response Procedures (ARPs), Abnormal Operating Instructions (AOIs), Emergency Operating Instructions (EOIs), Emergency Preparedness Implementing Procedures (EPIPs), and the operator's ability to recognize entry conditions based on symptoms that require or support procedure execution.

Loss of individual system annunciators and/or indicators may impact the function of system or component operability. System and component operability are addressed by Technical Specifications. Inoperability of multiple safety systems beyond Technical Specification compliance is addressed by separate event classification at 8.1-U.

Unplanned excludes loss of annunciators due to scheduled maintenance or testing activities. Compensatory non-alarming safety system indications include the Integrated Computer System (ICS) and SPDS.

15 minutes was chosen as a threshold to exclude transient or momentary power losses.

Escalation to Site Area Emergency is based on inability of the operating crew to monitor a significant transient in progress due to loss of annunciators and compensatory non-alarming indications during the transients.

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-106 Revision 103
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LOSS OF ASSESSMENT CAPABILITY

8.3-A

ALERT (CONTINUED)

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC SA4)

NOTES: **8.3** - Significant Transient is an unplanned event involving one or more of the following:

- (1) Automatic turbine runback greater than 25% thermal reactor power, or
- (2) Electrical load reduction greater than 25% full electrical load, or
- (3) Thermal power oscillations greater than 10%, or
- (4) Reactor scram, or
- (5) Valid ECCS initiation.

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-107 Revision 103
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LOSS OF ASSESSMENT CAPABILITY

8.3-S

SITE AREA EMERGENCY

EAL: Loss of most or all annunciators associated with safety systems

AND

Compensatory non-alarming safety system indications are NOT available
(SPDS, ICS)

AND

Indications needed to monitor safety functions are NOT available
(Refer to Table 8.3-S)

AND

A significant transient is in progress.

OPERATING CONDITION: MODE 1, or 2, or 3

BASIS: This event classification is intended to recognize the inability of the Control Room operators 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.

Annunciators that apply to this event classification are those that are required in support of Alarm Response Procedures (ARPs), Abnormal Operating Instructions (AOIs), Emergency Operating Instructions (EOIs), and Emergency Preparedness Implementing Procedures (EPIPs).

Compensatory non-alarming safety system indications include the Integrated Computer System (ICS) and SPDS.

Table 8.3-S includes indications necessary to shutdown the reactor, maintain the core cooled and in a coolable geometry, maintain RCS intact, maintain containment integrity.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC SS6)

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-108 Revision 103
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LOSS OF ASSESSMENT CAPABILITY 8.3-S

SITE AREA EMERGENCY (CONTINUED)

NOTES:

8.3 - Significant Transient is an unplanned event involving one or more of the following:

- (1) Automatic turbine runback greater than 25% thermal reactor power, or
- (2) Electrical load reduction greater than 25% full electrical load, or
- (3) Thermal power oscillations greater than 10%.
- (4) Reactor scram, or
- (5) Valid ECCS initiation.

CURVES/TABLES:

Table 8.3-S APPLICABLE SAFETY FUNCTIONS
Reactor Power
Reactor Pressure
Reactor Level
Subcriticality
Drywell Temperature
Drywell Pressure
Suppression Chamber Pressure
Suppression Pool Temperature
Suppression Pool Level

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-109 Revision 103
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OTHER

8.4-U

UNUSUAL EVENT

EAL: Events are in process or have occurred which indicate a potential degradation in the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs. Refer to Table 8.4-U for examples.

OR

Any loss or any potential loss of containment.

OPERATING CONDITION: ALL

BASIS: This event classification is intended to address conditions not explicitly addressed elsewhere that, in the judgment of the Site Emergency Director, warrant Unusual Event classification. Examples are provided in Table 8.4-U but classification is not restricted to only those events listed in the table. BFN EALs were developed primarily utilizing the symptom based grouping methodology. This approach is consistent with the BFN EOI methodology. It is important to note here that the consideration of fission product barriers has been incorporated within this symptom based approach. Barrier-based EALs refer to the level of challenge to principal barriers used to assure containment of radioactive material. For radioactive materials that are contained within the reactor core, these barriers are: fuel cladding, reactor coolant system pressure boundary, and containment. The level of challenge to these barriers encompasses the extent of damage (loss or potential loss) and the number of barriers currently under challenge. Site Emergency Directors should be continuously aware of all challenges to these barriers and the number of barriers loss or potentially loss. Also Site Emergency Directors should consider that when the loss or potential loss thresholds is imminent (i.e., 1 to 3 hours) use judgment and classify as if the thresholds are exceeded. The threshold for fission product barrier loss is considered to be consistent with the following:

Primary Containment barrier - Refer to 2.5-U.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC HU5, FU)
NRC Bulletin 2005-02, July 18, 2005 - Attachment 2 (Emergency Classification Level changes)
NEI White Paper, "Enhancements to Emergency Preparedness Programs for Hostile Action", May 2005 (Revised November 18, 2005)

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-110 Revision 103
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OTHER	8.4-U
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UNUSUAL EVENT
(CONTINUED)

NOTES:

8.4-U - Table 8.4-U contains only example events that may justify Unusual Event classification. This event classification is intended to address unanticipated conditions not explicitly addressed elsewhere, but warrant declaration of an emergency because conditions exist which the Emergency Director believes to fall under the Unusual Event Classification. Additionally, this EAL should be considered in making emergency classifications regarding challenges to fission product barriers not specifically addressed elsewhere in the EAL matrix.

CURVES/TABLES:

Table 8.4-U OTHER EXAMPLE UNUSUAL EVENTS
Plant Transient Response Unexpected Or Not Understood
Unanalyzed Safety System Configuration Affecting, Threatening Safe Shutdown
Inadequate Personnel To Achieve Or Maintain Safe Shutdown
Degraded Plant Conditions Beyond License Basis Threatening Safe Operation Or Safe Shutdown
Emergency Procedures Not Adequate To Maintain Safe Operation Or Achieve Safe Shutdown

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-111 Revision 103
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OTHER

8.4-A

ALERT

EAL: Events are in process or have occurred which involve an actual or potential substantial degradation in the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

OR

Any loss or potential loss of fuel cladding or RCS pressure boundary.

OPERATING CONDITION: ALL

BASIS: This event classification is intended to address conditions not explicitly addressed elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Site Emergency Director to fall under the Alert classification. BFN EALs were developed primarily utilizing the symptom based grouping methodology. This approach is consistent with the BFN EOI methodology. It is important to note here that the consideration of fission product barriers has been incorporated within this symptom based approach. Barrier-based EALs refer to the level of challenge to principal barriers used to assure containment of radioactive material. For radioactive materials that are contained within the reactor core, these barriers are: fuel cladding, reactor coolant system pressure boundary, and containment. The level of challenge to these barriers encompasses the extent of damage (loss or potential loss) and the number of barriers currently under challenge. Site Emergency Directors should be continuously aware of all challenges to these barriers and the number of barriers loss or potentially loss. Also Site Emergency Directors should consider that when the loss or potential loss thresholds is imminent (i.e., 1 to 3 hours) use judgment and classify as if the thresholds are exceeded.

The threshold for fission product barrier loss is considered to be consistent with the following:

Fuel clad - A Reactor coolant sample that yields a result of 300 µCi/gm Iodine-131 equivalent is indicative of cladding failure (Refer to 1.3-A).

RCS barrier - Reactor coolant leakage of at least 50 GPM from the primary system (Refer to 2.4-A).

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC HA6, FA)
NRC Bulletin 2005-02, July 18, 2005 - Attachment 2 (Emergency Classification Level changes)
NEI White Paper, "Enhancements to Emergency Preparedness Programs for Hostile Action", May 2005 (Revised November 18, 2005)

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-112 Revision 103
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OTHER	8.4-A
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ALERT
(CONTINUED)

NOTES:

8.4-A - This event classification is intended to address unanticipated conditions not explicitly addressed elsewhere, but that warrant declaration of an emergency because conditions exist which the Site Emergency Director believes to fall under the Alert classification. Additionally this EAL should be considered in making emergency classifications regarding challenges to fission product barriers not specifically address elsewhere in the EAL matrix.

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-113 Revision 103
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OTHER	8.4-S
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SITE AREA EMERGENCY

EAL:

Events are in process or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts (1) toward site personnel or equipment that could lead to the likely failure thereof or, (2) prevent effective access to equipment needed for protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

OR

Any loss or potential loss of both fuel cladding and RCS pressure boundary.

OR

Potential loss of either fuel cladding or RCS pressure boundary and loss of any additional barrier.

OPERATING CONDITION: ALL

BASIS:

This event classification is intended to address unanticipated conditions not explicitly addressed elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Site Emergency Director to warrant Site Area Emergency classification. BFN EALs were developed primarily utilizing the symptom based grouping methodology. This approach is consistent with the BFN EOI methodology. It is important to note here that the consideration of fission product barriers has been incorporated within this symptom based approach. Barrier-based EALs refer to the level of challenge to principal barriers used to assure containment of radioactive material. For radioactive materials that are contained within the reactor core, these barriers are: fuel cladding, reactor coolant system pressure boundary, and containment. The level of challenge to these barriers encompasses the extent of damage (loss or potential loss) and the number of barriers currently under challenge. Site Emergency Directors should be continuously aware of all challenges to these barriers and the number of barriers loss or potentially loss. Also Site Emergency Directors should consider that when the loss or potential loss thresholds is imminent (i.e., 1 to 3 hours) use judgment and classify as if the thresholds are exceeded.

Loss or potential loss of any two fission product barriers must be considered along with inability to monitor fission product barriers during extreme conditions. The threshold for fission product barrier loss is considered to be consistent with the following:

Fuel clad - A Reactor coolant sample that yields a result of 300 $\mu\text{Ci/gm}$ Iodine-131 equivalent is indicative of cladding failure (Refer to 1.3-A).

RCS barrier - Reactor coolant leakage of at least 50 GPM from the primary system (Refer to 2.4-A).

Primary Containment barrier - Refer to 2.5-U.

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-114 Revision 103
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OTHER	8.4-S
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SITE AREA EMERGENCY
(CONTINUED)

REFERENCES:

Reg Guide 1.101 Rev. 3, (NUMARC HS3, FS)
NRC Bulletin 2005-02, July 18, 2005 - Attachment 2 (Emergency Classification Level changes)
NEI White Paper, "Enhancements to Emergency Preparedness Programs for Hostile Action", May 2005 (Revised November 18, 2005)

NOTES:

8.4-S - This event classification is intended to address unanticipated conditions not explicitly addressed elsewhere, but that warrant declaration of an emergency because conditions exist which the Site Emergency Director believes to fall under the Site Area Emergency classification. Additionally this EAL should be considered in making emergency classifications regarding challenges to fission product barriers not specifically address elsewhere in the EAL matrix.

CURVES/TABLES:

BFN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX A PAGE A-115 Revision 103
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OTHER

8.4-G

GENERAL EMERGENCY

EAL: Events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

OR

Loss of any two barriers and potential loss of third barrier.

OPERATING CONDITION: ALL

BASIS: This event classification is intended to address unanticipated conditions not explicitly addressed elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Site Emergency Director to fall under the General Emergency classification. BFN EALs were developed primarily utilizing the symptom based grouping methodology. This approach is consistent with the BFN EOI methodology. It is important to note here that the consideration of fission product barriers has been incorporated within this symptom based approach. Barrier-based EALs refer to the level of challenge to principal barriers used to assure containment of radioactive material. For radioactive materials that are contained within the reactor core, these barriers are: fuel cladding, reactor coolant system pressure boundary, and containment. The level of challenge to these barriers encompasses the extent of damage (loss or potential loss) and the number of barriers currently under challenge. Site Emergency Directors should be continuously aware of all challenges to these barriers and the number of barriers loss or potentially loss. Also Site Emergency Directors should consider that when the loss or potential loss thresholds is imminent (i.e., 1 to 3 hours) use judgment and classify as if the thresholds are exceeded.

Loss or potential loss of all fission product barriers must be considered along with inability to monitor fission product barriers during extreme conditions. The threshold for fission product barrier loss is considered to be consistent with the following:

Fuel clad - A Reactor coolant sample that yields a result of 300 µCi/gm Iodine-131 equivalent is indicative of cladding failure (Refer to 1.3-A).

RCS barrier - Reactor coolant leakage of at least 50 GPM from the primary system (Refer to 2.4-A).

Primary Containment barrier - Refer to 2.5-U.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC HG2, FG)
NRC Bulletin 2005-02, July 18, 2005 - Attachment 2 (Emergency Classification Level changes)
NEI White Paper, "Enhancements to Emergency Preparedness Programs for Hostile Action", May 2005 (Revised November 18, 2005)

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OTHER	8.4-G
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GENERAL EMERGENCY
(CONTINUED)

NOTES:

8.4-G - This event classification is intended to address unanticipated conditions not explicitly addressed elsewhere, but that warrant declaration of an emergency because conditions exist which the Site Emergency Director believes to fall under the General Emergency classification. Additionally this EAL should be considered in making emergency classifications regarding challenges to fission product barriers not specifically address elsewhere in the EAL matrix.

CURVES/TABLES:

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A.2 SITE EMERGENCY ORGANIZATION

BFN maintains an organization capable of responding to a radiological emergency. The TSC, OSC, and Control Room staffing for response to emergencies is shown in figure A-1.

A.2.1 Site Vice President

The Site Vice President serves as a corporate interface for the SED, relieving him from duties which could distract from the SED's primary purpose of plant operations and accident mitigation activities. The Site Vice President shall provide assistance in the following areas as needed:

1. Provides TVA policy direction to the Site Emergency Director.
2. Directs the site resources to support the Site Emergency Director in the accident mitigation activities.
3. Provides direct interface on overall site response activities with:
 - a. NRC, FEMA, or other Federal organizations responding to the site.
 - b. CECC Director.
 - c. Onsite media.
4. At his discretion, may provide interface at the appropriate offsite location on the overall site response activities with:
 - a. State and local agencies.
 - b. NRC region/corporate.
 - c. Joint Information Center.
5. Provides support to other emergency operation centers as necessary.

A.2.2 Site Emergency Director

1. Directs onsite emergency accident mitigation activities.
2. Consults with CECC Director and Site Vice President on significant events and their related impacts.
3. Initiates onsite protective actions.
4. Coordinates accident mitigation actions with NRC.
5. Initiates long-term 24-hour accident mitigation operations.
6. Prior to the CECC being staffed, makes recommendations for protective actions (if necessary) to State and local agencies through the Operations Duty Specialist. This responsibility cannot be delegated except to the CECC Director after the CECC is operational.
7. Responsible for determining the emergency classification. This responsibility cannot be delegated.

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8. Makes final decision on personnel entrance to radiological hazardous areas when RP recommends against the entry.
9. Approves emergency exposures when required, this cannot be delegated.

A.2.3 Operations Manager

1. Directs operational activities.
2. Informs Site Emergency Director of plant status and operational problems.
3. Performs damage assessment as necessary.
4. Recommends solutions and mitigating action for operational problems.

A.2.4 Technical Assessment Manager

1. Directs onsite effluent assessment.
2. Directs activities of technical assessment team.
3. Projects future plant status based on present plant conditions.
4. Keeps assessment team informed of plant status.
5. Provides information, evaluations, and projects to Site Emergency Director.
6. Coordinates assessment activities with the CECC plant assessment team.
7. Ensures that Plant Status Boards are maintained.

A.2.5 Maintenance Manager

1. Directs repairs and corrective actions.
2. Performs damage assessment.
3. Directs activities of Operations Support Center.

A.2.6 TSC Clerks

1. Maintain log of events.
2. Answer telephones.
3. Operate facsimile machine.
4. Other duties as assigned by Site Emergency Director.

A.2.7 TSC Communicator

1. Provides information from control room to Technical Assessment team as needed.
2. Completes plant data sheets as needed.

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A.2.8 Nuclear Security Manager

1. Directs activities of Nuclear Security Services personnel.
2. Controls access to site and control rooms.
3. Reports on site accountability/evacuation as defined in BFN-EIPs.

A.2.9 Radiation Protection (RP) Manager

1. Directs and/or performs assessment of inplant and onsite radiological conditions.
2. Directs onsite RP activities.
3. Coordinates additional RP support with CECC Radiological Communicator.
4. Makes recommendations for protective actions for onsite personnel.
5. Maintains status map of offsite radiological conditions.
6. Coordinates assessment of radiological conditions offsite with CECC Radiological Communicator.
7. Maintains inplant radiation status board.
8. Coordinates briefing of maintenance teams with maintenance manager and assigns a RP Technician to accompany them if necessary.
9. Makes final recommendations to the Site Emergency Director for personnel entry to radiological hazardous environment.

A.2.10 OSC Manager

1. Directs repairs and corrective actions.
2. Performs damage assessment.
3. Coordinates OSC Teams, ensuring proper briefings and the accompaniment by RP as applicable.

A.2.11 Chemistry Manager

1. Coordinates assessment of radioactive effluents with CECC Plant Assessment Team.
2. Directs post-accident sampling activities.
3. Directs activities of the radiochemical laboratory.
4. Determines impact of incident on environment, radwaste, various effluent treatment systems.

A.2.12 OSC Mechanical Briefer

1. Directs OSC (Mechanical).
2. Performs damage and repair assessment.

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A.2.13 Technical Assessment Team Leader

1. Performs systems assessment as directed by Technical Assessment Manager.
2. Determines condition of reactor and nuclear fuel.
3. Acts as plant assessment team leader.

A.2.14 OSC I/C Briefer

1. Directs OSC (Instrumentation).
2. Performs damage and repair assessment.

A.2.15 OSC Electrical Briefer

1. Directs OSC (Electrical).
2. Performs damage and repair assessment.

A.2.16 NRC Coordinator

1. Acts as primary liaison with onsite NRC personnel.
2. Updates NRC personnel on plant status.
3. Provides information requests from NRC to TSC personnel.

A.2.17 Operations Communicator

1. Provides operational knowledge for status evaluation of plant systems.
2. Provides advice regarding technical specifications, system response, safety limits, etc.
3. Assists in development of recommended solutions to developing problems.

A.2.18 Emergency Preparedness Manager

1. Advises Site Emergency Director regarding overall radiological emergency plan, use of implementing procedures, emergency equipment availability, and coordination with CECC.
2. Confirms site emergency centers are operating properly.

A.2.19 TAT Mechanical

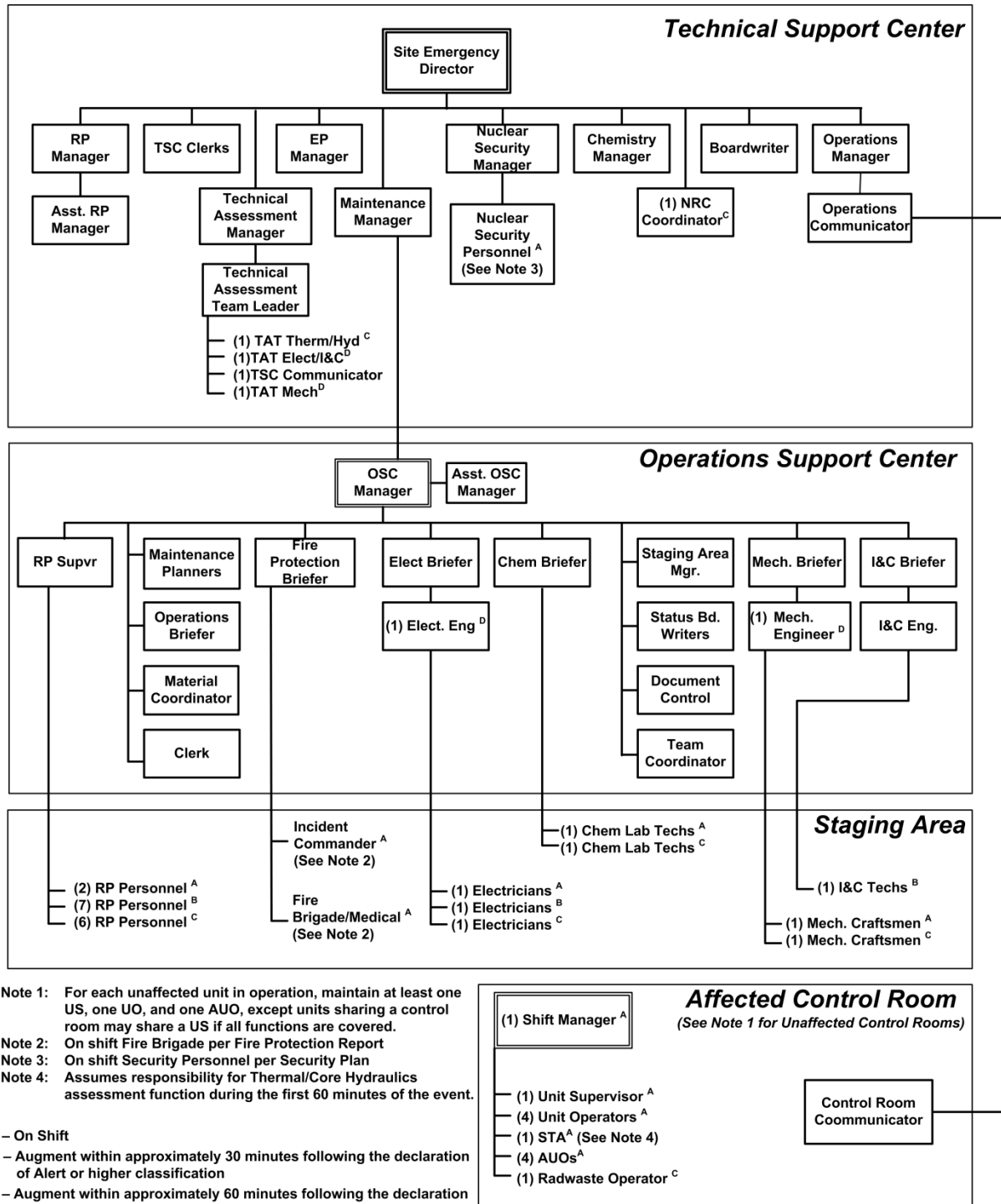
1. Serves as the primary interface with Engineering.
2. Provides for additional engineering support during and/or following a radiological emergency.
3. Coordinates the design and construction of emergency equipment and structures as necessary.

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A.2.20 Technical Assessment Team

1. Prepares and provides periodic current assessments on plant conditions and provides this information to the CECC plant assessment team.
2. Projects future plant status based on present plant conditions.
3. Provides technical support to plant operations on mitigating actions.

**FIGURE A-1
SITE EMERGENCY ORGANIZATION**
 (Including Minimum Staffing and Staff Augmentation)



Note 1: For each unaffected unit in operation, maintain at least one US, one UO, and one AUO, except units sharing a control room may share a US if all functions are covered.
 Note 2: On shift Fire Brigade per Fire Protection Report
 Note 3: On shift Security Personnel per Security Plan
 Note 4: Assumes responsibility for Thermal/Core Hydraulics assessment function during the first 60 minutes of the event.

^A – On Shift

^B – Augment within approximately 30 minutes following the declaration of Alert or higher classification

^C – Augment within approximately 60 minutes following the declaration of an Alert or higher classification

^D – One Mechanical and one Electrical Engineer augmented within approximately 60 minutes. Function can be filled as applicable by either TSC or OSC position.

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A.3 EMERGENCY RESPONSE FACILITIES, EQUIPMENT, AND SUPPLIES

Specific plant areas, facilities, and equipment are selected and provided for use during a radiological emergency. The preselection, allocation, and inclusion of emergency facilities assure that needed services and equipment are available for use during emergency conditions.

A.3.1 Technical Support Center (TSC)

A specific area (between units 1 and 2 control room and unit 3 control room) in the control bay at elevation 617.0 is designated for use as the TSC. The room is provided with communication facilities for plant areas and areas external to the plant. The communication facilities include TVA telephone system (with Bell System access, TVA Microwave System access, and long distance access), a paging-intercom system, and two-way radio capabilities. This room is sufficiently shielded to ensure occupancy during an emergency and is designed to be continuously habitable during all radiological emergencies. All ventilating and air-conditioning facilities have redundant or backup systems. Toilet facilities are available on the same elevation.

The diesel generators will provide emergency power when there is a loss of normal ac power, and cooling water for the air-conditioning equipment can be taken from the emergency equipment cooling water system. Emergency equipment is specially designated and stored near the TSC for use during an emergency. Figure A-2 shows a detailed TSC layout.

Meteorological information is available both in the TSC and main control room and includes wind speed and direction at 10, 46, and 91 meters, and temperature at 10, 46, and 91 meters.

A.3.2 Operations Support Center (OSC)

The role of the OSC is to provide an assembly area for operations support personnel during an emergency situation and is under the supervision of the OSC Director. The restart operations area at elevation 580' in the service building (see figure A-3) is designated for use as the OSC. The OSC is provided with telephone communications. In the event that radiation conditions require evacuation of this area, OSC personnel will report to the office building, elevation 580.

A.3.3 RP Laboratory And Equipment

The RP laboratory is located in the service building adjacent to the personnel corridor at elevation 565.0. The portable radiation monitoring and counting equipment normally used by the plant RP section is kept in this space and is available for use during an emergency. Sufficient resources of instruments/equipment are available to replace those removed from service for calibration or repair. Calibration of equipment is carried out at intervals as specified in applicable site procedures.

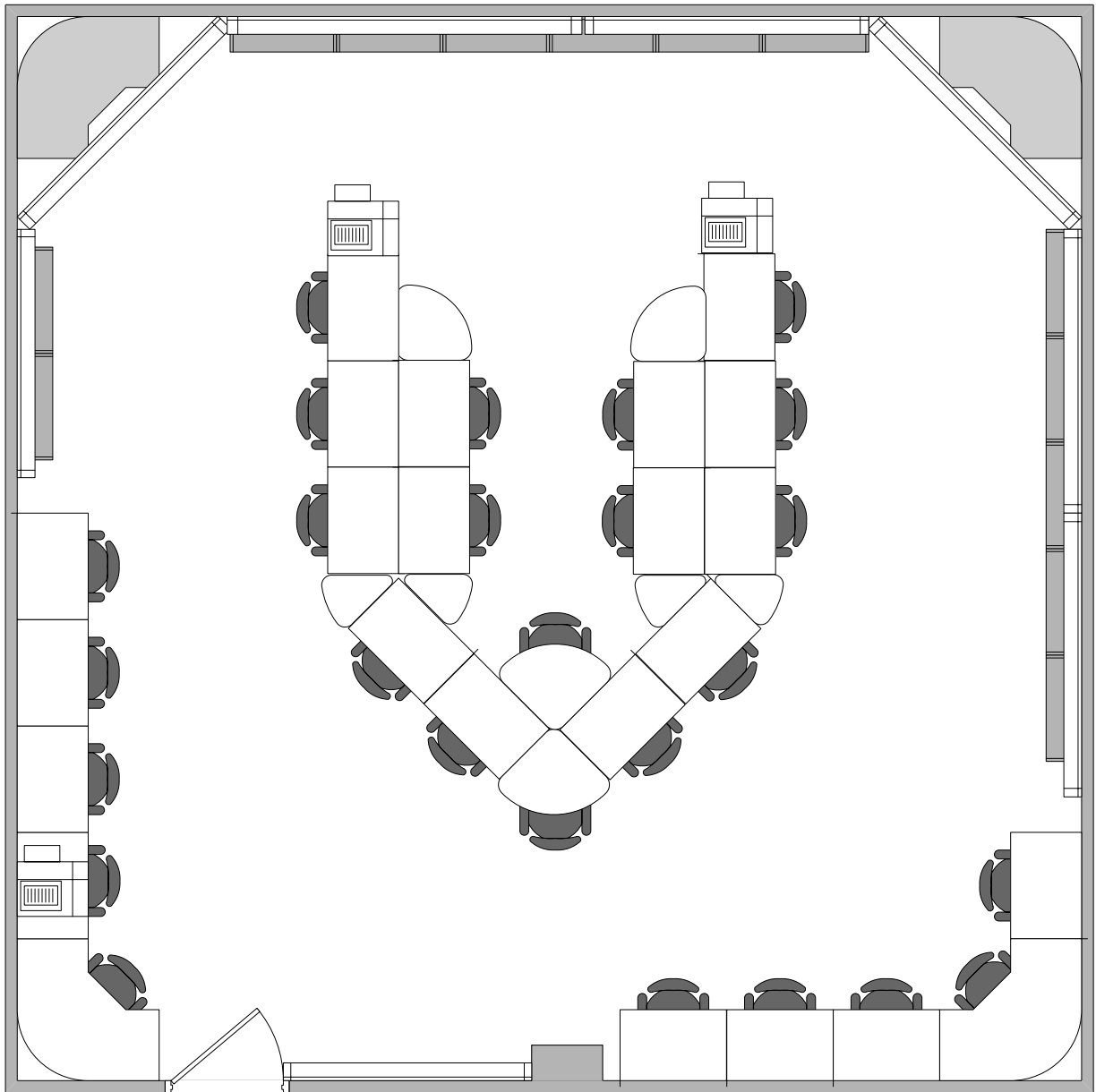
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A.3.4 Alternative Facilities

Following the declaration of Alert or higher, for some conditions, primarily in response to hostile action events when ERO members may not have access to the site, an alternative facility has been identified. This facility serves as a staging area for augmentation staff, which will minimize delays in overall site response and allow for a swift, coordinated augmented response when the site is deemed accessible.

The alternative facility is accessible even if the site is under threat of or experiencing hostile action, to function as a staging area for augmentation of emergency response staff and have the capability for communication with the emergency operations facility, control room, and plant security, for use when onsite emergency facilities cannot be safely accessed during hostile action.

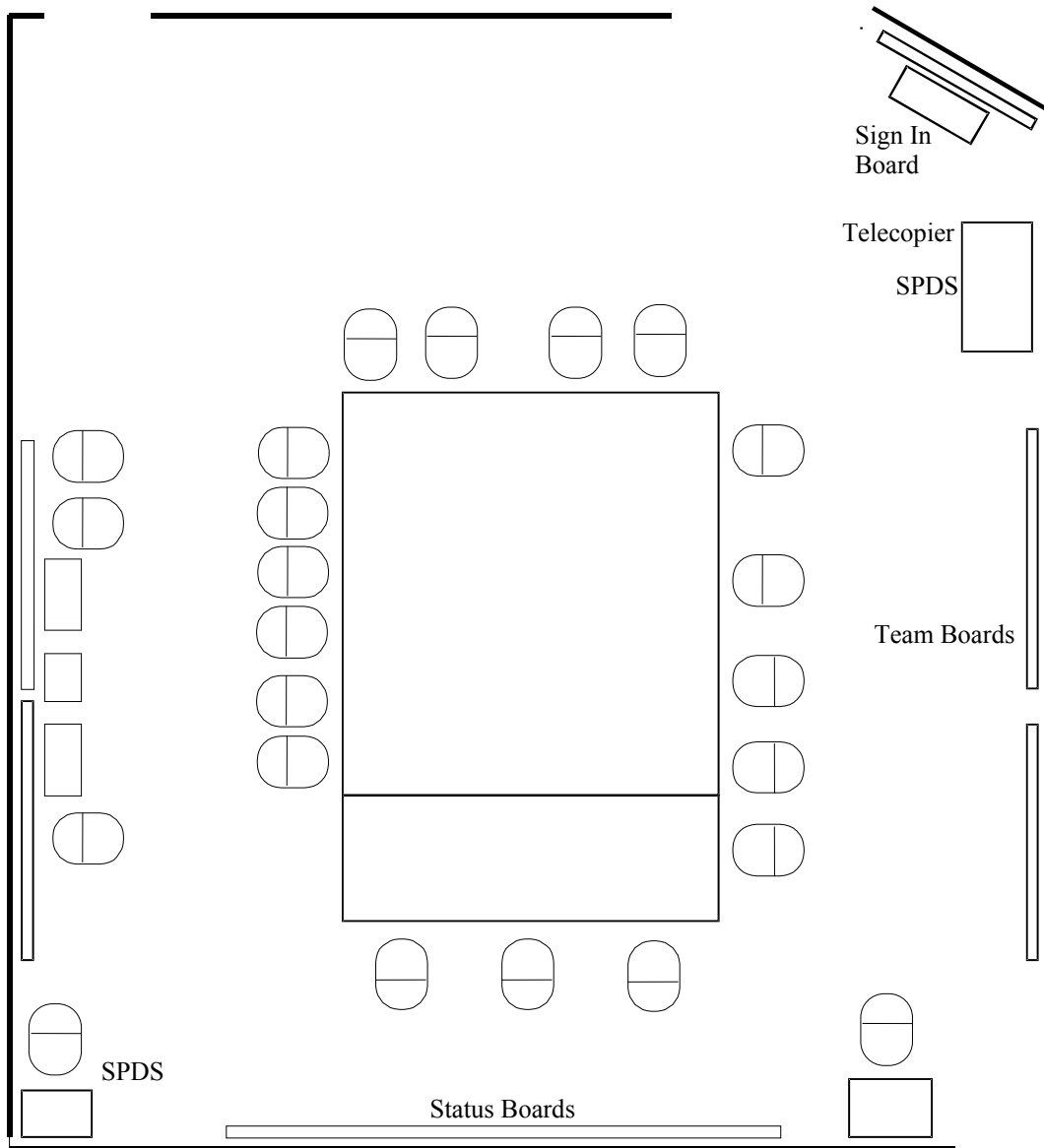
FIGURE A-2
TECHNICAL SUPPORT CENTER



Technical Support Center assignments:

Site Emergency Director	Maintenance Manager	Tech. Assessment Manager	TSC Clerk
Site VP	Chemistry Manager	Nuclear Security Mgr.	
EP Manager	Assistant RP Manager	NRC Coord.	
RP Manager	OPS. Manager	OPS Communicator	

FIGURE A-3
OPERATIONS SUPPORT CENTER



Operations Support Center assignments:

Assistant OSC Mgr.	Document Control	OPS Briefer	OSC Manager	Team Coordr.
Chemistry	Fire Protection Briefer	OSC Engineers	Planners	
Clerk	Material Coordinator	OSC Maintenance Briefers	RP	

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A.3.5 Onsite Monitoring Systems and Equipment

A.3.5.1 Natural Phenomena

In the event an emergency is the result of a natural phenomena, there is instrumentation to monitor its severity. The Environmental Data Station is located onsite and contains instruments capable of measuring wind direction, wind speed, and temperatures. Seismic instrumentation is available in the plant to monitor acceleration levels of ground movement. Hydrological monitoring systems are installed to supply flow and level information for each site. Meteorological and seismic instrumentation have readily accessible readout in the main control room. More specific information on these systems can be found in the Browns Ferry FSAR.

A.3.5.2 Radiological Monitors

The installed Radiation Monitoring System consists of process monitors and area monitors which read out on local panels and in the control room.

A.3.5.2.1 Process Monitors (Radiological)

The process system continuously monitors selected lines containing or possibly containing radioactive effluents. The system's function is to warn personnel of increasing radiation levels, to give early warning of a system malfunction, and to record and control discharges of radioactive liquids and gases to the environment. The system consists of active and redundant channels. Examples of process monitors are:

1. Reactor Building Ventilation Monitoring System
2. Main Steam Line Radiation Monitoring System
3. Main Stack Radiation Monitoring System
4. Plant Ventilation Exhaust Radiation Monitoring System
5. Liquid Radwaste
6. Raw Cooling Water
7. Reactor Building Closed Cooling Water
8. Residual Heat Removal Service Water Discharge

A.3.5.2.2 Area Monitors

Area monitors are placed at specific locations in the plant. Examples of area monitor locations are:

1. Reactor Building
2. New And Spent Fuel Storage Area
3. Turbine Building
4. Main Control Room
5. Radwaste Building
6. Off-Gas Stack

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A.3.5.2.3 Portable Monitors

Portable radiation detection equipment consists of low- and high-range instruments to measure gamma radiation levels from 0.1 mR/hr to 1000 R/hr. Instruments for alpha, beta-gamma, and neutron radiation measurements are available. Sampling equipment is available to take low- or high-volume air samples. Air samplers can be used to collect low-volume samples either onsite or offsite. The counting room has a multichannel analyzer with shielded HPGe detectors, and a liquid scintillation counter.

A.3.5.2.4 Process Monitors (Nonradiological)

Installed in the main control room are the necessary instrumentation readouts to assess plant systems status including reactor coolant system pressure and temperature, containment pressure and temperature, liquid levels, flow rates, fire detection equipment, and meteorological instrumentation. More specific information on control room instrumentation can be found in the Browns Ferry FSAR.

A.3.5.2.5 Fire Protection

The plant's fire protection system is designed to furnish water and other extinguishing agents with the capability of extinguishing any single or probable combination of simultaneous fires that might occur. The use of combustible materials is minimized, and the greatest possible use of fire-retardant materials has been incorporated in plant design. The standards of the National Fire Protection Association and the recommendations of the nuclear insurers are considered in the system design to provide the following:

1. Supply of water for the fire protection system.
2. Automatic fire or smoke detection in the more critical areas.
3. Fire suppression by fixed equipment actuated automatically or manually.
4. Manually operated portable fire extinguishing equipment at strategic locations.
5. Compartmentation to limit the spread of fire.

In addition to the minimum standards prescribed in the technical specifications, Browns Ferry has one fully equipped Class A fire department pumper. Sufficient personnel are available to meet technical specification requirements for a fire brigade and provide the necessary personnel to operate the pumper.

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A.3.5.2.6 Environment

Facilities available for assessing the impact of plant operations on the environment include atmospheric monitoring stations, direct gamma radiation detectors, and automatic water samplers. This equipment is used in the routine environmental radiological monitoring program and is available in the event of a radiological emergency condition.

The atmospheric monitoring network is divided into three subgroups. Local air monitors are located at or adjacent to the boundary in the directions of predominant wind flow. Perimeter monitors are located three to 10 miles from the plant in areas of relatively high population densities and/or in the direction of predominant air flow. Remote monitors (controls) are located at sites greater than 10 miles from the plant.

At each monitor, air is continuously passed through a particulate filter at a regulated flow. In series with, but downstream of, the particulate filter is a charcoal filter used to collect iodine. Each monitor has a collection tray and storage container to collect rainwater on a continuous basis.

Primary dosimeters are placed at approximately 40 sites around the plant. These primary dosimeters are located typically in each of the 16 meteorological sectors at or near the site boundary and at a distance of approximately 4-5 miles. Three dosimeters are usually placed at each site.

Automatic water samplers are located above and below the plant discharge, at the first potable water supply downstream from the plant, and at a ground water well which is down gradient from the plant.

In addition to these facilities, established sampling points for milk, vegetation, soil, fish, and sediment are located in the vicinity of the plant. Samples may be collected from these stations on a nonroutine basis as needed.

All samples are returned to one of TVA's radiological laboratories for processing.

A.3.6 Emergency Equipment

Figure A-4 contains listings of emergency equipment locations throughout the plant. Detailed listings of supplies and locations are included in the BFN-EIPs. Required calibration of equipment is carried out at intervals recommended by the supplier of the equipment or as specified in Browns Ferry FSAR.

A.3.7 First Aid And Medical Facilities

A.3.7.1 Decontamination Facilities

The site is responsible for maintaining supplies and equipment to establish a temporary decontamination area for the purpose of gross radiological decontamination of personnel who may also be injured.

The personnel decontamination room and emergency medical treatment area, complete with sink and shower facilities, is provided in the service building area at elevation 565.0. Equipment and cleaning solutions for the decontamination of personnel are available in this room. Stretcher-bound personnel can also be decontaminated in this facility.

FIGURE A-4
EMERGENCY EQUIPMENT

	<u>Location</u>	<u>Description</u>
1.	RP Laboratory Service Building	Emergency supplies and Radiological survey meters
2.	Control Bay	Emergency SCBAs
3.	Control Building Mechanical Equipment Room	Emergency supplies and radiological survey meters
5.	Emergency Van	General emergency supplies related to environs monitoring
6.	Huntsville Hospital & Decatur General Hospital Emergency Rooms	Supplies specific to radiological injuries
7.	TSC	Emergency supplies
8.	OSC	Emergency supplies

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A.3.7.2 Health Stations and Supplies

Emergency medical equipment is strategically located throughout the plant, with trauma kits and other specialized equipment available for use by the MERT.

A first aid station, staffed by an EMT, is located within the security fence. Medical supplies and treatments for minor injuries are available. First aid treatment is available 24 hours a day.

A medical office, staffed by registered nurses and a physician, is located on the first floor of the BFN Visitor and Training Center. Medical treatment and examinations (employment, routine, occupational) are available during the day shift, Monday - Friday.

Potassium iodide tablets for onsite personnel are controlled and stored by site RP. Specific information including authorization and dispersal of tablets is contained in the site EIPs.

A.3.7.3 Receiving Hospitals and Supplies

Arrangements have been made with a least one hospital to receive patients from BFN. (See Sections 12.3 and 16.5.)

A.3.7.4 Ambulance Service

A TVA ambulance is available at the site and is maintained by Site Fire Protection. Arrangements have been made for offsite ambulance assistance to BFN. (See Sections 12.2 and 16.5.)

A.3.8 Additional Local Support

A.3.7.1 Law Enforcement

Agreements (see section 16.5) are maintained with law enforcement agencies to support TVA when necessary.

A.3.9 Vendor Support

If necessary, the NSSS vendor, General Electric, will be contacted by the CECC Director to provide assistance in the form of manpower, equipment, and technical backup. Other vendors will also be contacted if their assistance is needed.

A.3.10 Emergency Siren

Undulating sirens are provided in strategic plant areas for indicating the assembly of all personnel. Care is exercised in locating the sirens so that they are audible in all plant areas. A three-minute undulating blast on the siren is the signal for assembly.

The sirens are powered by redundant 120V ac supplies. The sirens can be activated from the electrical control desk in the units 1 and 2 control room or the unit 3 control room.

A.3.11 Local Recovery Center

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The LRC for Browns Ferry will be a portion of the second floor of the Administration Building outside the protected area of the site.

A.3.11.1 Communications

The LRC has voice communication capabilities to enable personnel to communicate with the CECC and the Browns Ferry TSC. The communication facilities available in the LRC area is the TVA telephone system (with Bell System access, TVA Microwave System access, and long distance access).

Meteorological information and dose rate calculations are also available to LRC personnel.

Other equipment available for use by LRC personnel include:

1. Facsimile machine
2. Copy machines
3. Hand calculators
4. Plant-specific drawings, manuals, procedures, etc.

A.3.12 REP Implementing Procedures

The following is a listing of the BFN-EIPs.

A.3.12.1 BFN-EPIP-1-Emergency Classification Procedure

This procedure provides guidance to the Shift Manager in determining the classification of an accident to ensure that appropriate predetermined actions are implemented. It details initiating conditions and directs shift personnel to appropriate notification and assessment procedures.

A.3.12.2 BFN-EPIP-2—Notification of Unusual Event

This procedure provides for the timely notification of appropriate individuals when the Shift Manager has determined by EPIP-1 that an incident has occurred which is classified as a Notification of Unusual Event. It details requirements for periodic reassessment and the implementation of appropriate actions.

A.3.12.3 BFN-EPIP-3—Alert

This procedure provides for the timely notification of appropriate individuals when the Shift Manager has determined by EPIP-1 that incident has occurred which is classified as an Alert. It details requirements for periodic reassessment and the implementation of appropriate actions. It also contains information for performing offsite dose assessment.

A.3.12.4 BFN-EPIP-4—Site Area Emergency

This procedure provides for the timely notification of appropriate individuals when the Shift Manager has determined by EPIP-1 that an incident has occurred which is classified as a Site Area Emergency. It details requirements for periodic reassessment and the implementation of appropriate actions. It also contains information for performing offsite dose assessment.

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A.3.12.5 BFN-EPIP-5—General Emergency

This procedure provides for the timely notification of appropriate individuals when the Shift Manager has determined by EPIP-1 that incident has occurred which is classified as a General Emergency.

It details requirements for periodic reassessment and the implementation of appropriate actions. It also contains information for performing offsite dose assessment. It also contains information for determining protective action recommendations for the public.

A.3.12.6 BFN-EPIP-6—Activation and Operation of the TSC

This procedure directs the activation and operation of the TSC during an Alert, Site Area Emergency, or General Emergency. It details notification requirements. Documents issued onsite contain the TSC call-outs lists.

A.3.12.7 BFN-EPIP-7—Activation and Operation of the OSC

This procedure directs the activation and operation of the OSC during an Alert, Site Area Emergency, or General Emergency. It details notification requirements. Documents issued onsite contain the OSC call-out lists.

A.3.12.8 BFN-EPIP-8—Personnel Accountability And Evacuation

This procedure details the requirements for accountability of all personnel and visitors and the orderly evacuation of areas of the plant during a radiological emergency.

A.3.12.9 BFN-EPIP-10—Medical Emergency Procedure

This procedure details actions to be followed during medical emergencies. It provides for the organization and activation of the onsite Medical Emergency Organization. It contains the duties and responsibilities of the onsite Medical Emergency Organization. The procedure provide guidance on the care and handling of patients who may have been exposed to or contaminated with radioactive material, including provisions for the transport of these individuals to offsite medical support facilities. Maps and appropriate instructions are also included.

A.3.12.10 BFN-EPIP-11—Security and Access Control

This procedure details responsibilities and requirements for access control and accountability during a radiological emergency.

A.3.12.11 BFN-EPIP-12— Emergency Equipment Implementing Procedure

This procedure provides the instructions for periodic inspection and maintenance of emergency equipment and supplies.

A.3.12.12 BFN-EPIP-13—Dose Assessment Procedure

The procedure provides instructions to be followed for the conduct of on-site environmental radiological dose assessments.

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A.3.12.13 BFN-EPIP-14—Radiological Control Procedure

This procedure outlines the actions to be followed by RP personnel during a plant emergency. It details responsibilities and RP assessment actions and record keeping requirements. The procedure provides guidance regarding the administration of potassium iodide (KI) to inplant workers.

A.3.12.14 BFN-EPIP-15—Emergency Exposure

This procedure provides guidance on acceptable personnel exposures for various conditions. It specifies absolute exposure and authorizes the Site Emergency Director to permit exposures in excess of 10 CFR 20 limits in order to perform the emergency mission.

A.3.12.15 BFN-EPIP-16—Termination and Recovery Procedure

This procedure outlines responsibilities and provides guidance on termination of an event and recovery after an emergency to ensure adequate planning for efficient utilization of resources and radiation exposure.

A.3.12.16 BFN-EPIP-17—Fire Emergency Procedure

This procedure provides the guidance for the management of the response to fire emergencies.

A.4 ALERT NOTIFICATION SYSTEM

See Generic REP Section

Enclosure 4

Radiological Emergency Plan (REP)

Appendix B: Sequoyah Nuclear Plant

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APPENDIX B

SEQUOYAH NUCLEAR PLANT

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<p align="center">SQN</p>	<p align="center">TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN</p>	<p align="center">NP-REP APPENDIX B Page B-3 Revision 103</p>
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B.1 Introduction

The following information provides a site specific list of Initiating Conditions, specific instrument parameters (when required) and a basis for classifying and declaring emergency events at the Sequoyah Nuclear Plant (SQN).

These conditions apply to each and both Unit-1 and Unit-2. The Site Emergency Director must be aware of the affects of simultaneous events on both units.

Criteria for determining these emergency events was taken from REG GUIDE 1.101, Emergency Planning and Preparedness for Nuclear Power Reactors which allows licenses to use NUMARC/NESP-007, Rev. 2, 1/92, Methodology for Development of Emergency Action Levels.

For the purposes of declaring an emergency SQN used the following emergency classifications: General Emergency, Site Area Emergency, Alert, and Unusual Event.

For a General Emergency to be declared, events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels outside the EXCLUSION AREA BOUNDARY.

For a Site Area Emergency to be declared, events are in process or have occurred which involve an actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; (1) toward site personnel or equipment that could lead to the likely failure of or; (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels outside the EXCLUSION AREA BOUNDARY.

For an Alert to be declared, events are in process or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

For an Unusual Event to be declared, events are in process or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

The goal of these emergency classification levels is to have offsite emergency response authorities prepared to take actions to protect the health and safety of the public in case of a radiological emergency.

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B.2 Emergency Event Methodology

The SQN methodology for event classification and declaration has 36 emergency events broken down into the following seven categories (NUMARC/NESP 007 references are Rev. 2 unless otherwise noted):

FISSION PRODUCT BARRIER MATRIX (Modes 1-4)

SQN Reference	NUMARC/NESP-007 Reference
1.1 Fuel Clad	FC 1,2,3,4,5,7
1.2 RCS	RCS 1,2,3,5,6
1.3 Containment	CNTMT 1,2,3,4,5,8

SYSTEM DEGRADATION

SQN Reference	NUMARC/NESP-007 Reference
2.1 Loss of Instrumentation	SU3, SA4, SS6
2.2 Loss of Communication	SU6
2.3 Failure of Reactor Protection	SA2 (Modified), SS2 (Modified), SG2
2.4 Fuel Clad Degradation	SU4
2.5 RCS Unidentified Leakage	SU5
2.6 RCS Identified Leakage	SU5
2.7 Uncontrolled Cool Down	HU5
2.8 Turbine Failure	HU1, HA1
2.9 Safety Limit	SU2

LOSS OF POWER

SQN Reference	NUMARC/NESP-007 Reference
3.1 Loss of AC (Power Ops)	SU1, SA5, SS1, SG1
3.2 Loss of AC (Shutdown)	SU1, SA1
3.3 Loss of DC	SU7, SS3

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HAZARDS and SED JUDGEMENT

SQN Reference	NUMARC/NESP-007 Reference
4.1 Fire	HU2, HA2
4.2 Explosion	HU1, HA1
4.3 Flammable Gas	HU3, HA3
4.4 Toxic Gas	HU3, HA3
4.5 Control Room Evacuation	HA5, HS2
4.6 Security	HU4, HA4, HS4, HG1(NEI 99-01 R5)
4.7 SED Judgment	HU5, HA6, HS3, HG2

DESTRUCTIVE PHENOMENON

SQN Reference	NUMARC/NESP-007 Reference
5.1 Earthquake	HU1, HA1
5.2 Tornado	HU1, HA1
5.3 Aircraft/Projectile	HU1, HA1
5.4 River Level High	HU1, HA1
5.5 River Level Low	HU1, HA1
5.6 Watercraft Crash	HU1

SHUTDOWN SYSTEM DEGRADATION

SQN Reference	NUMARC/NESP-007 Reference
6.1 Loss of Shutdown Systems	SA3, SS5 (expanded)
6.2 Loss of Shutdown Capability	SU2, SA3, SS4
6.3 Loss of RCS Inventory	SU5

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RADIOLOGICAL

SQN Reference	NUMARC/NESP-007 Reference
7.1 Gaseous Effluent	AU1, AA1, AS1, AG1
7.2 Liquid Effluent	AU1, AA1
7.3 Radiation Levels	AU2, AA3
7.4 Fuel Handling	AU2, AA2
7.5 Spent Fuel Storage	E-HU1 (Rev. 4, 1/03)

In each event there exists a set of Initiating Conditions and associated emergency action levels (where required) which initiate the declaration of the emergency and the required level of onsite and offsite emergency response.

In the SQN Methodology, the following operating modes are used in the declaratory scheme:

- Power operations (1)
- Start up (2)
- Hot Standby (3)
- Hot Shutdown (4)
- Cold Shutdown (5)
- Refueling (6)
- Defueled

B.3 Responsibility

The responsibility of declaring an emergency based on the criteria provided in this section belongs to the Shift Manager/Site Emergency Director (SM/SED) or designated Unit Supervisor when acting as the SM or the SED. These duties cannot be delegated.

B.4 Classification Determination

To determine the classification of the emergency, the SED reviews the Initiating Conditions of the events described in Emergency Plan Implementing Procedure (EPIP 1) One with the known or suspected conditions.

If a Critical Safety Function (CSF) is listed as an Initiating Condition, the respective status tree criteria will be monitored and used to determine the event classification for the modes listed on the classification matrix in EPIP-1.

Declare the highest emergency class based on **events that are in progress** at the time that the classification is made.

If during an ongoing event investigations indicate that a higher classification was met then report that, as information only, to the Operation Duty Specialist (ODS) and the NRC. Do not declare or upgrade to a higher emergency class if the conditions do not exist unless it is a noted exception.

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B.4 Classification Determination (continued)

Following termination of an emergency declaration, if follow-up investigations show that a higher classification was met then report that, as information only, to the ODS and the NRC. Do not declare or upgrade to a higher emergency class if the conditions do not exist.

If plant conditions have returned to a non-emergency state before any emergency can be classified, the highest emergency class that was appropriate shall be reported as information to the ODS and NRC and shall not be declared unless it is a noted exception.

The NRC shall be notified within one hour of all classifications. Once made and reported, a declaration cannot be canceled or rescinded even if it is later determined to be invalid. If there is reason to doubt that a given condition has occurred the SM/SED shall follow indications and proceed with classification, as required, until otherwise proven false.

The State shall be notified within 15 minutes of any declaration. If the State is notified of a declaration that is **invalidated before the NRC is notified**, terminate the classification, if not already done, and report the declaration to the NRC.

References:

10 CFR 50	Domestic Licensing of Production and Utilization Facilities
REG GUIDE-1.101	<i>Emergency Planning and Preparedness For Nuclear Power Reactors endorsing NUMARC NESP-007 Methodology for Development of Emergency Action Levels</i>

Site Technical Specifications (Tech Specs), Abnormal Operating Procedures (AOPs), Emergency Operating Procedures (EOPs), and the Final Safety Analysis Report (FSAR) are also referenced in Appendix B of the Radiological Emergency Plan to support the Emergency Classification Flow Chart.

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BOMB: An explosive device. (See EXPLOSION)

CIVIL DISTURBANCE: A group of twenty (20) or more persons within the EAB violently protesting onsite operations or activities at the site.

CONFINEMENT BOUNDARY: Spent Fuel Storage Cask CONFINEMENT BOUNDARY consists of MPC shell, bottom baseplate, MPC lid (including the vent and drain port cover plates), MPC closure ring, and associated welds.

CRITICAL-SAFETY FUNCTION (CSFs): A plant safety function required to prevent significant release of core radioactivity to the environment. There are six CSFs: Subcriticality, Core Cooling, Heat Sink, Pressurized Thermal Shock, Integrity (Containment) and Inventory (RCS).

EVENT: Assessment of an EVENT commences when recognition is made that one or more of the initiating conditions associated with the event exist. Implicit in this definition is the need for timely assessment within 15 minutes.

EXCLUSION AREA BOUNDARY (EAB): That area surrounding the reactor, in which the reactor licensee has the authority to determine all activities including exclusion or removal of personnel and property from the area. For purposes of Emergency Action Levels, based on radiological field measurements and dose assessments, and for design calculations, the Site Boundary shall be defined as the EAB.

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

EXTORTION: An attempt to cause an action at the site by threat of force.

FAULTED: (Steam Generator) Existence of secondary side leakage (e.g., steam or feed line break) that results in an uncontrolled decrease in steam generator pressure or the steam generator being completely depressurized.

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

FLAMMABLE GAS: Combustible gases of concentrations > the LOWER EXPLOSIVE LIMIT (LEL).

HOSTAGE: A person(s) held as leverage against the site to ensure that demands will be met by the site.

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

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

IMMINENT: Within two hours.

INEFFECTIVE: When the specified restoration action(s) does not result in a reduction in the level of severity of the RED or ORANGE PATH condition within 15 minutes from identification of the CSF Status Tree RED or ORANGE PATH.

INITIATING CONDITIONS: Plant Parameters, radiation monitor readings or personnel observations that identify an Event for purposes of Emergency Plan Classification.

INTRUSION/INTRUDER: Suspected hostile individual present in the protected area without authorization.

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ODCM: Offsite Dose Calculation Manual is a supporting document to the Tech. Specs. that contain Rad Effluent Controls, Environs Monitoring controls, and methodology for calculating routine gaseous and liquid effluent offsite doses and monitor alarm/trip setpoints.

ORANGE PATH: Monitoring of one or more CSFs by FR-0 which indicates that the CSF(s) is under severe challenge; prompt operator action is required.

PROJECTILE: An object ejected, thrown, or launched towards a plant structure resulting in damage sufficient to cause concern regarding the integrity of the affected structure or the operability or reliability of safety equipment contained therein. The source of the projectile may be onsite or offsite.

PROTECTED AREA: The area encompassed by the security fence and to which access is controlled.

RCS: The RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary and secondary isolation valves.

RED PATH: Monitoring of one or more CSFs by FR-0 which indicates that the CSF(s) is under extreme challenge, prompt operator action is required.

RUPTURED: (Steam Generator) Existence of primary to secondary leakage of a magnitude greater than the capacity of one charging pump.

SABOTAGE: Deliberate damage, misalignment, or misoperation of plant equipment with the intent to render the equipment inoperable.

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

SIGNIFICANT TRANSIENT: An UNPLANNED event involving one or more of the following: (1) an automatic turbine runback >15% thermal reactor power; (2) Electrical load rejection >25% full electrical load; (3) Reactor Trip; (4) Safety Injection System Activation; (5) Thermal Power Oscillations $\geq 10\%$.

STRIKE ACTION: A work stoppage within the PROTECTED AREA by a body of workers to enforce compliance with demands made on TVA. The STRIKE ACTION must threaten to interrupt normal plant operations.

TOXIC GAS: A gas that is dangerous to life or limb by reason of inhalation or skin contact (e.g., chlorine, CO₂, etc.).

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

UNPLANNED RELEASE: A release of radioactivity is UNPLANNED if the release has not been authorized by a Discharge Permit (DP) Implicit in the definition are unintentional releases, unmonitored releases, or planned releases that exceed a condition specified in the DP, (e.g., alarm setpoints, minimum dilution flow, minimum release times, maximum release rates, and/or discharge of incorrect tank.

VALID: An indication, report or condition is considered to be VALID when it is conclusively verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) direct observation by plant personnel. Implicit in this definition is the need for timely assessment within 15 minutes.

VISIBLE DAMAGE: Damage to equipment that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example damage includes deformation due to heat or impact, denting, penetration, rupture, cracking, or paint blistering. Surface blemishes (e.g., paint chipping, scratches, etc.) should NOT be included as visible damage.

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

Sequoyah Nuclear Plant

Emergency Classification and Declaration Methodology

BASIS

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<i>Section</i>	<i>1.0</i>	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>	<i>1.1</i>	FUEL CLAD BARRIER
<i>IC</i>	<i>1.1.1</i>	Critical Safety Function Status
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p><u>LOSS:</u></p> <p>Core Cooling Red (FR-C.1).</p> <p><u>Potential LOSS:</u></p> <p>Core Cooling Orange (FR-C.2).</p> <p><u>OR</u></p> <p>Heat Sink Red (FR-H.1) (RHR shut down cooling not in service).</p>
<i>Basis</i>		<p><u>LOSS:</u></p> <p>The “Loss” IC addresses the condition of inadequate core cooling.</p> <p>If the emergency operating procedure status trees indicate a red path the condition must be considered to be an extreme challenge to the safety function needed to ensure protection of the public.</p> <p>Core Cooling - Red indicates significant superheating and core uncover and is considered to indicate a “Loss” of the fuel clad barrier.</p> <p><u>Potential LOSS:</u></p> <p>The “Potential Loss” IC addresses the condition where an inadequate core cooling situation can develop. If the emergency operating procedure status trees indicate an orange path, the conditions must be considered to be a severe challenge to the safety function.</p> <p>Core Cooling Orange indicates subcooling has been lost and that some clad damage may occur. Heat Sink Red indicates the heat sink function is under extreme challenge and thus either of these two items indicate a “Potential Loss” of the fuel clad barrier.</p>
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101 FR-C.1 Inadequate Core Cooling FR-C.2 Degraded Core Cooling FR-H.1 Loss of Heat Sink

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<i>Section</i>	<i>1.0</i>	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>	<i>1.1</i>	FUEL CLAD BARRIER
<i>IC</i>	<i>1.1.2</i>	Primary Coolant Activity Level
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p><u>LOSS:</u></p> <p>RCS sample activity is greater than 300 $\mu\text{Ci/gm}$ dose equivalent Iodine - 131.</p> <p><u>Potential LOSS:</u></p> <p>Not Applicable.</p>
<i>Basis</i>		<p><u>LOSS:</u></p> <p>The "Loss" IC addresses the condition of high RCS activity. If the reading of RCS activity is greater than 300 $\mu\text{Ci/gm}$ it is well above expected iodine spikes and corresponds to about 1% fuel clad damage. This amount of clad damage indicates that significant clad heating has occurred.</p> <p><u>Potential LOSS:</u></p> <p>There is no "Potential Loss" IC associated with this item.</p>
<i>Escalation</i>		Not Applicable.
<i>References</i>		<p>NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101</p> <p>NUREG 1465 Accident Source Terms for Light-Water Nuclear Power Plants</p> <p>SQN, TI-RPS-193 R6 Calibration of High Range Monitors in Containment</p>

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<i>Section</i>	<i>1.0</i>	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>	<i>1.1</i>	FUEL CLAD BARRIER
<i>IC</i>	<i>1.1.3</i>	Incore TCs Hi Quad Average
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p><u>LOSS:</u></p> <p>Greater than 1200°F on XI-94-101 or 102 (EXOSENSOR).</p> <p><u>Potential LOSS:</u></p> <p>Greater than or equal to 700°F on XI-94-101 or 102 (EXOSENSOR).</p>
<i>Basis</i>		<p><u>LOSS:</u></p> <p>The “Loss” IC uses a reading of 1200°F which corresponds to a core cooling red condition on the EOP status trees. A reading of this magnitude corresponds to significant superheating of the reactor coolant and clad heating which results in a “Loss” of fuel clad barrier.</p> <p><u>Potential LOSS:</u></p> <p>The “Potential Loss” IC uses a reading of 700°F which corresponds to a core cooling orange condition on the EOP status trees. A reading of this magnitude corresponds to a loss of RCS subcooling.</p>
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101 FR-C.1 Inadequate Core Cooling FR-C.2 Degraded Core Cooling

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<i>Section</i>	<i>1.0</i>	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>	<i>1.1</i>	FUEL CLAD BARRIER
<i>IC</i>	<i>1.1.4</i>	Reactor Vessel Water Level
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p><u>LOSS:</u></p> <p>Not Applicable.</p> <p><u>Potential LOSS:</u></p> <p>VALID RVLIS Level <42% On LI-68-368 or 371 (No RCP running).</p>
<i>Basis</i>		<p><u>LOSS:</u></p> <p>There is no "Loss" IC corresponding to this item because it is covered by the other fuel clad barrier "Loss".</p> <p><u>Potential LOSS:</u></p> <p>The "Potential Loss" IC is defined by an orange path on the core cooling status tree. The numeric value used is 42% level with no reactor coolant pumps running. This condition indicates that considerable clad heating and loss of RCS subcooling has occurred.</p> <p>A RVLIS reading of 42% is equivalent to a reactor vessel level 3.5' above the bottom of the fuel. This is also equivalent to 8.5' of uncovered fuel.</p>
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101 DCN E-20591A

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Section	1.0	FISSION PRODUCT BARRIER MATRIX									
Event	1.1	FUEL CLAD BARRIER									
IC	1.1.5	Containment Radiation Monitors									
Mode		1,2,3,4									
Description		<p>Note:</p> <p>Containment Radiation Monitors are temperature sensitive and can be affected by temperature-induced currents. These monitors should be used for trending only for 2 minutes after a Steam Line Break or LOCA. Once 2 minutes has expired these monitors can be used for EAL determination.</p> <p><u>LOSS:</u></p> <p>VALID reading on RM-90-271A <u>AND</u> RM-90-272A.greater than:</p> <table><tr><td>RM-90-271A</td><td>2.5E+02 R/hr</td></tr><tr><td>RM-90-272A</td><td>2.5E+02 R/hr</td></tr></table> <p><u>OR</u></p> <p>VALID reading on RM-90-273A <u>AND</u> RM-90-274A greater than:</p> <table><tr><td>RM-90-273A</td><td>1.5E+02 R/hr</td></tr><tr><td>RM-90-274A</td><td>2.1E+02 R/hr</td></tr></table> <p><u>Potential LOSS:</u></p> <p>Not Applicable.</p>		RM-90-271A	2.5E+02 R/hr	RM-90-272A	2.5E+02 R/hr	RM-90-273A	1.5E+02 R/hr	RM-90-274A	2.1E+02 R/hr
RM-90-271A	2.5E+02 R/hr										
RM-90-272A	2.5E+02 R/hr										
RM-90-273A	1.5E+02 R/hr										
RM-90-274A	2.1E+02 R/hr										
Basis		<p><u>LOSS:</u></p> <p>The "Loss" IC is defined by a VALID reading of 2.5E+02 R/hr on the upper containment hi rad monitors or 1.5E+02 R/hr and 2.1E+02 R/hr on the lower containment hi rad monitors. The level of radiation in the containment is indicative of a loss of coolant accident (LOCA) in the containment in conjunction with fuel damage.</p> <p>The reading assumes the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 300 µCi/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 1% clad failure depending on core inventory and RCS volume). Thus, this IC indicates a loss of both the fuel clad barrier and a loss of the RCS barrier.</p> <p><u>Potential LOSS:</u></p> <p>There is no "Potential Loss" IC associated with this item.</p>									

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	<p><i>Note:</i></p> <p><i>PER 30167 documents that SQN's High Range Radiation Monitor (HRRM) cables may be susceptible to Thermally Induced Currents (TICs) during Design Basis Accidents (DBA) for large, rapid thermal transients inside containment. As a result of this PER, Functional Evaluation- High Range Radiation Monitor Thermally Induced Currents, Rev 1 was performed to assess the cables. Below is a summary of the Functional Evaluation, the complete Functional Evaluation is available as an attachment to PER 30167.</i></p> <p><i>During rapid thermal increases at the beginning of a DBA, these currents could be falsely interpreted as indications of the presence of ionizing radiation in upper or lower containment and lead to inappropriate responses by operators and emergency responders to higher than necessary Emergency Classification Levels.</i></p> <p><i>For Radiological Emergency Preparedness, the High Range Radiation Monitors are used to determine the classification of an emergency. Specifically, they are used to declare a Fuel Clad Barrier Loss and a Containment Barrier Potential Loss. The design function of the HRRMs is to detect high radiation conditions during events which result in increased radiation levels. After 120 seconds, when the HRRMs are needed for use, the positive or negative TICs will have little appreciable effect other than the negative TICs may cause the keep alive source to be negated causing a malfunction alarm.</i></p> <p><i>The test report recommends that the radiation monitor output be discounted for approximately 2 minutes following a rapid increase or decrease in containment temperature.</i></p>
<i>Escalation</i>	Not Applicable.
<i>References</i>	NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101 NUREG 1465 Accident Source Terms for Light-Water Nuclear Power Plants SQN, TI-RPS-193 R6 Calibration of High Range Monitors in Containment. SQN Functional Evaluation, High Range Radiation Monitor Thermally Induced Currents, Rev 1

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<i>Section</i>	<i>1.0</i>	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>	<i>1.1</i>	FUEL CLAD BARRIER
<i>IC</i>	<i>1.1.6</i>	Site Emergency Director Judgement
<i>Mode</i>		1,2,3,4
<i>Description</i>		Any condition that, in the judgement of the SM/SED, indicates Loss or Potential Loss of the Fuel Clad Barrier comparable to the conditions listed above.
<i>Basis</i>		<p>This IC gives the SED the latitude to use his judgement in determining if the fuel clad barrier is or will be in a "Loss" or "Potential Loss" condition. This situation is usually considered when plant conditions are present that require the monitoring of CSFs or performance of EOP corrective actions. Specific cases where SED judgement may be required are the loss of instrumentation needed to monitor the CSFs and the loss of all AC power.</p> <p>Although the majority of the ICs provide very specific thresholds, the Site Emergency Director must remain alert to events or conditions that lead to the conclusion that exceeding the IC threshold is imminent. If, in the judgement of the Site Emergency Director, an imminent situation is at hand, the classification should be made as if the thresholds have been exceeded. While this is particularly prudent at the higher emergency classes (as the early classification may provide for more effective implementation of protective measures), it is nonetheless applicable to all emergency classes.</p>
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101

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<i>Section</i>	<i>1.0</i>	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>	<i>1.2</i>	RCS BARRIER
<i>IC</i>	<i>1.2.1</i>	Critical Safety Function Status
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p><u>LOSS:</u> Not Applicable.</p> <p><u>Potential LOSS:</u> Pressurized Thermal Shock Red (FR-P.1).</p> <p style="text-align: center;"><u>OR</u></p> <p>Heat Sink Red (FR-H.1) (RHR shutdown cooling not in service).</p>
<i>Basis</i>		<p><u>LOSS:</u> There is no "Loss" IC associated with this item.</p> <p><u>Potential LOSS:</u> The "Potential Loss" IC is defined by a red path on pressurized thermal shock or a red path on the heat sink CSF status trees. In the case of PTS, consideration is given to a failure of the reactor vessel resulting in a loss of coolant accident (LOCA). In the case of loss of heat sink the eventual method of cooling the reactor core is by safety injection in conjunction with a RCS LOCA.</p>
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101 FR-P.1 Pressurized Thermal Shock FR-H.1 Loss of Heat Sink

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<i>Section</i>	<i>1.0</i>	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>	<i>1.2</i>	RCS BARRIER
<i>IC</i>	<i>1.2.2</i>	RCS Leakage/LOCA
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p><u>LOSS:</u></p> <p>RCS leak results in subcooling <40°F as indicated on XI-94-101 or 102.</p> <p><u>Potential LOSS:</u></p> <p>Non-isolatable RCS leak exceeding the capacity of one charging pump in the normal charging alignment.</p> <p style="text-align: center;"><u>OR</u></p> <p>RCS leakage results in entry into E-1.</p>
<i>Basis</i>		<p><u>LOSS:</u></p> <p>The "Loss" IC addresses conditions where leakage from the RCS is greater than available inventory control capacity such that a loss of subcooling has occurred. The loss of subcooling is the fundamental indication that the inventory control systems are inadequate in maintaining RCS pressure and inventory against the mass loss through the leak.</p> <p><u>Potential LOSS:</u></p> <p>The "Potential Loss" IC is based on the inability to maintain normal liquid inventory within the RCS by normal operation of the Chemical and Volume Control System. Normal operation is considered as one centrifugal charging pump discharging to the charging header and letdown in service. This assures that any event that results in significant RCS inventory shrinkage or loss (e.g., events leading to reactor trip and ECCS actuation) will result in no lower than an "Alert" emergency classification.</p>
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101 E-1 Loss of Reactor or Secondary Coolant

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<i>Section</i>	1.0	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>	1.2	RCS BARRIER
<i>IC</i>	1.2.3	Steam Generator Tube Rupture
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p><u>LOSS:</u></p> <p>SGTR that results in a safety injection actuation.</p> <p style="text-align: center;"><u>OR</u></p> <p>Entry into E-3.</p> <p><u>Potential LOSS:</u></p> <p>Not Applicable.</p>
<i>Basis</i>		<p><u>LOSS:</u></p> <p>The “Loss” IC addresses conditions where the SGTR exists and the RCS flow into the steam generator is such that pressurizer level and pressure cannot be maintained. The inability to maintain level via the normal charging header, requires a safety injection by procedure. If a manual safety injection is not initiated an auto SI will occur due to a low pressurizer pressure.</p> <p>Any event that results in significant RCS inventory shrinkage or loss (e.g., events leading to reactor trip and ECCS actuation) will result in no lower than an “Alert” emergency classification.</p> <p>This IC also addresses the entry into EOP, E-3, Steam Generator Tube Rupture, under any circumstance.</p> <p>This “Loss” IC in conjunction with the containment barrier “Loss” IC 1.3.4 addresses the situation where the S/G that is ruptured is also faulted. This “Loss” of two barriers requires an event classification of Site Area Emergency.</p> <p><u>Potential LOSS:</u></p> <p>There is no “Potential Loss” IC associated with this item.</p>
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101 AOP R.01 Steam Generator Tube Leak E-3 Steam Generator Tube Rupture

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<i>Section</i>	<i>1.0</i>	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>	<i>1.2</i>	RCS BARRIER
<i>IC</i>	<i>1.2.4</i>	Reactor Vessel Water Level
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p><u>LOSS:</u></p> <p>VALID RVLIS level <42% on LI-68-368 or 371 with no RCP running.</p> <p><u>Potential LOSS:</u></p> <p>Not Applicable.</p>
<i>Basis</i>		<p><u>LOSS:</u></p> <p>The "Loss" IC is defined by an orange path on the core cooling status tree (CSF). The numeric value used is 42% level with no reactor coolant pumps running. Inability to maintain reactor vessel water level is the fundamental indication that the RCS barrier has been lost.</p> <p>This "Loss" EAL in conjunction with the fuel clad barrier "Potential Loss" IC 1.1.4 requires an event classification of Site Area Emergency.</p> <p>A RVLIS reading of 42% is equivalent to a reactor vessel level 3.5' above the bottom of the fuel. This is also equivalent to 8.5' of uncovered fuel.</p> <p><u>Potential LOSS:</u></p> <p>There is no "Potential Loss" IC associated with this item.</p>
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101 DCN E-20591A

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<i>Section</i>	<i>1.0</i>	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>	<i>1.2</i>	RCS BARRIER
<i>IC</i>	<i>1.2.5</i>	Site Emergency Director Judgement
<i>Mode</i>		1,2,3,4
<i>Description</i>		Any condition that, in the judgement of the SM/SED, indicates Loss or Potential Loss of the RCS barrier comparable to the conditions listed above.
<i>Basis</i>		<p>This IC gives the SED the latitude to use his judgement in determining if the RCS barrier is or will be in a "Loss or Potential Loss" condition. This situation is usually considered when plant conditions are present that require the monitoring of CSFs or performance of EOP corrective actions. Specific cases where SED judgement may be required are the loss of instrumentation needed to monitor the CSFs and the loss of all AC power.</p> <p>Although the majority of the EALs provide very specific threshold, the SED must remain alert to events or conditions that lead to the conclusion that exceeding the EAL threshold is imminent. If, in the judgement of the SED, an imminent situation is at hand, the classification should be made as if the thresholds have been exceeded. While this is particularly prudent at the higher emergency classes (as the early classification may provide for more effective implementation of protective measures), it is nonetheless applicable to all emergency classes.</p>
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101

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<i>Section</i>	<i>1.0</i>	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>	<i>1.3</i>	CNTMT BARRIER
<i>IC</i>	<i>1.3.1</i>	Critical Safety Function Status
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p><u>LOSS:</u> Not Applicable.</p> <p><u>POTENTIAL LOSS:</u> Containment Red (FR-Z.1).</p> <p><u>OR</u></p> <p>Actions of FR-C.1 (Red Path) are INEFFECTIVE (ie: core TC's trending up).</p>
<i>Basis</i>		<p><u>LOSS:</u> There is no "Loss" IC associated with this item.</p> <p><u>Potential LOSS:</u></p> <p>The first "Potential Loss" IC is defined by a red path on the containment status tree. A red path indicates an extreme challenge to the safety function derived from appropriate instrument readings and/or sampling results, and thus represents a potential loss of containment. Conditions leading to a containment red path result from RCS barrier and/or fuel clad barrier Loss. Thus, this IC is primarily a discriminator between the Site Area Emergency and General Emergency representing a potential loss of the third barrier.</p> <p>The second "Potential Loss" IC is defined by a red path on the core cooling status tree with FR-C.1 ineffective. In this IC, the functional restoration procedures are those emergency operating procedures that address the recovery of the core cooling critical safety functions. The procedure is considered ineffective if the temperature is not decreasing or if the vessel water level is not increasing within 15 minutes.</p> <p>The conditions identified in this potential loss IC represent an imminent melt sequence which, if not corrected, could lead to vessel failure and an increased potential for containment failure. In conjunction with the core exit thermocouple ICs in the fuel clad barrier and RCS barrier columns, this IC would result in the declaration of a General Emergency -- loss of two barriers and the potential loss of a third. If the functional restoration procedures are ineffective, there is no "success" path.</p>
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101 FR-Z.1 High Containment Pressure FR-C.1 Inadequate Core Cooling

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<i>Section</i>	1.0	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>	1.3	CNTMT BARRIER
<i>IC</i>	1.3.2	Containment Pressure/Hydrogen
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p><u>LOSS:</u> Rapid unexplained pressure decrease following initial increase on Pdl 30-44 or 45.</p> <p style="text-align: center;"><u>OR</u></p> <p>Containment pressure or sump level not increasing on LI-63-178 and 179 with a LOCA in progress.</p> <p><u>Potential LOSS:</u> Containment hydrogen increases to >4% by volume on H₂I-43-200 or 210.</p> <p style="text-align: center;"><u>OR</u></p> <p>Pressure >2.8 PSIG (Phase B) with < one full train of containment spray.</p>
<i>Basis</i>		<p><u>LOSS:</u> The first "Loss" IC address a rapid unexplained loss of pressure, (i.e., not attributable to containment spray or condensation effects), following an initial pressure increase, indicating a loss of containment integrity.</p> <p>The second "Loss" IC addresses the situation where containment pressure or sump level is not increasing with a LOCA in progress. This could indicate containment bypass and loss of containment integrity. This IC, in conjunction with RCS barrier IC #2, results in an event classification of Site Area Emergency.</p> <p><u>Potential LOSS:</u> The first "Potential Loss" IC addresses the existence of an explosive mixture of hydrogen and oxygen in the containment, which if ignited, would be a challenge to the containment barrier.</p> <p>The second "Potential Loss" IC represents a potential loss of containment in that the cont. heat removal/depressurization system (e.g., containment sprays, ice condenser, etc.) are either lost or performing in a degraded manner. This is indicated by containment pressure greater than the phase B setpoint of 2.8 psig where the equipment should actuate.</p> <p>The condition of high containment pressure, >12 psig, is addressed by the CSF, containment red, "Potential Loss", IC #1.3.1.</p> <p>These "Potential Loss" ICs are primarily a discrimination between the Site Area Emergency and General Emergency representing a potential loss of the third barrier.</p>
<i>Escalation</i>		Not Applicable.

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<i>Section</i>	<i>1.0</i>	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>	<i>1.3</i>	CNTMT BARRIER
<i>IC</i>	<i>1.3.3</i>	Containment Isolation Status
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p><u>LOSS:</u></p> <p>Containment isolation, when required, is incomplete and a release path to the environment exists.</p> <p><u>Potential LOSS:</u></p> <p>Not Applicable.</p>
<i>Basis</i>		<p><u>LOSS:</u></p> <p>The Loss IC is intended to address incomplete containment isolation that allows a direct release to the environment. It represents a loss of the containment barrier.</p> <p><u>Potential LOSS:</u></p> <p>There is no "Potential Loss" IC associated with this item.</p>
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101

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<i>Section</i>	<i>1.0</i>	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>	<i>1.3</i>	CNTMT BARRIER
<i>IC</i>	<i>1.3.4</i>	Containment Bypass
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p><u>LOSS:</u> RUPTURED S/G that is also faulted outside CNTMT (E-2 and E-3) OR > 4 hours secondary side release outside CNTMT from a S/G with a S/G tube leak > T/S limits (AOP R.01 App A).</p> <p><u>Potential LOSS:</u> Unexpected VALID increase in area or ventilation RAD monitors adjacent to containment (with LOCA in progress).</p>
<i>Basis</i>		<p><u>LOSS:</u></p> <p>The first “Loss” IC addresses a non-isolatable secondary side release from a ruptured steam generator. This allows a direct release of radioactive fission and activation products to the environment. Resultant offsite dose rates are a function of many variables. Examples include: coolant activity, actual leak rate, S/G carry over, iodine partitioning, and meteorology. Therefore, dose assessment in accordance with event Gaseous Effluent (Section 7.1) General Emergency, “Exclusion area boundary dose resulting from an actual or imminent release of gaseous radioactivity that exceeds 1 Rem TEDE or 5 Rem CDE For the actual or projected duration of the release”, is required when there is indication that the fuel clad barrier is potentially lost.</p> <p>This IC would exist in conjunction with the RCS barrier “Loss” IC 1.2.3 and results in an event classification of a Site Area Emergency. Escalation to General Emergency would be based on “Potential Loss” of the fuel clad barrier.</p> <p>The second “Loss” IC addresses a prolonged, greater than four (4) hour, secondary side release outside of the containment from a steam generator having primary to secondary leakage greater than Tech. Spec. limits, (LCO 3.4.6.2). This IC results in an event classification of Unusual Event. This indicator’s intent addresses non-isolatable main steam line breaks outside containment, feedwater line breaks, failed open relief valves, atmospheric dump valves or plant cooldown via atmospheric steam dump due to loss of offsite power or the main condenser. However, it is not the intent of this indicator to address transient events such as (1) MSLB downstream of the MSIV if the MSIV isolates the break, (2) affected S/G isolation in accordance with procedures, or for other similar events. Prolonged steam releases via the main condenser air ejectors or steam driven auxiliary feed pumps exhaust should be classified on the basis of dose assessment rather than the fission product barrier matrix.</p>

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<i>Section</i> 1.0	FISSION PRODUCT BARRIER MATRIX
<i>Event</i> 1.3	CNTMT BARRIER
<i>IC</i> 1.3.4	Containment Bypass (continued)
<i>Mode</i>	1,2,3,4
<i>Basis (continued)</i>	<p><u>Potential LOSS:</u></p> <p>The "Potential Loss" IC addresses an increase in area or ventilation radiation monitors adjacent to containment, with a LOCA in progress, which is indicative of a potential loss of the containment barrier.</p> <p>The SED must take into consideration events in progress to determine if the increase in rad. monitors is expected or explained. Events such as ECCS initiation and recirculation of contaminated water from the containment sump through the RHR, containment spray, and SI systems are expected and may result in an initial increase in area or ventilation rad. monitors.</p> <p>The concern is for potential loss of the containment barrier and not for specific monitor readings. Indications of containment bypass should be derived from unexpected or unexplained trends in rad. monitor readings. Events such as an unexpected increasing trend or lack of an expected decreasing trend on an area or ventilation rad. monitors adjacent to containment would indicate further investigation and validation is warranted. Trends like these may indicate loss, or bypass, of containment that is not readily observable from other indications.</p> <p>This IC in conjunction with the RCS barrier IC 1.2.2 results in an event classification of Site Area Emergency.</p>
<i>Escalation</i>	Not Applicable.
<i>References</i>	NUMARC/NESP-007, Rev. 2, 1/92 per REG GUIDE 1.101 E-2 Faulted Steam Generator Isolation E-3 Steam Generator Tube Rupture AOP-R.01 Steam Generator Tube Leak

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Section	1.0	FISSION PRODUCT BARRIER MATRIX								
Event	1.3	CNTMT BARRIER								
IC	1.3.5	Significant Radioactivity in Containment								
Mode		1,2,3,4								
Description		<p>Note:</p> <p>Containment Radiation Monitors are temperature sensitive and can be affected by temperature-induced currents. These monitors should be used for trending only for 2 minutes after a Steam Line Break or LOCA. Once 2 minutes has expired these monitors can be used for EAL determination.</p> <p><u>LOSS:</u></p> <p>Not Applicable.</p> <p><u>Potential LOSS:</u></p> <p>VALID reading on RM-90-271A <u>AND</u> RM-90-272A.greater than:</p> <table><tr><td>RM-90-271A</td><td>5.8E+03 R/hr</td></tr><tr><td>RM-90-272A</td><td>5.8E+03 R/hr</td></tr></table> <p><u>OR</u></p> <p>VALID reading on RM-90-273A <u>AND</u> RM-90-274A greater than:</p> <table><tr><td>RM-90-273A</td><td>3.4E+03 R/hr</td></tr><tr><td>RM-90-274A</td><td>4.9E+03 R/hr</td></tr></table>	RM-90-271A	5.8E+03 R/hr	RM-90-272A	5.8E+03 R/hr	RM-90-273A	3.4E+03 R/hr	RM-90-274A	4.9E+03 R/hr
RM-90-271A	5.8E+03 R/hr									
RM-90-272A	5.8E+03 R/hr									
RM-90-273A	3.4E+03 R/hr									
RM-90-274A	4.9E+03 R/hr									
Basis		<p><u>LOSS:</u></p> <p>There is no "Loss" IC associated with this item.</p> <p><u>Potential LOSS:</u></p> <p>The "Potential Loss" IC is defined by containment radiation readings of 5.80E+03 R/hr on the upper containment rad monitors or 3.43E+03 R/hr and 4.88E+03 R/hr on the lower containment rad monitors.</p> <p>This reading indicates significant fuel damage well in excess of the EALs associated with both loss of fuel clad and loss of RCS barriers. A major release of radioactivity, requiring offsite protective actions 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%.</p>								

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	<p><i>Note:</i></p> <p><i>PER 30167 documents that SQN's High Range Radiation Monitor (HRRM) cables may be susceptible to Thermally Induced Currents (TICs) during Design Basis Accidents (DBA) for large, rapid thermal transients inside containment. As a result of this PER, Functional Evaluation- High Range Radiation Monitor Thermally Induced Currents, Rev 1 was performed to assess the cables. Below is a summary of the Functional Evaluation, the complete Functional Evaluation is available as an attachment to PER 30167.</i></p> <p><i>During rapid thermal increases at the beginning of a DBA, these currents could be falsely interpreted as indications of the presence of ionizing radiation in upper or lower containment and lead to inappropriate responses by operators and emergency responders to higher than necessary Emergency Classification Levels.</i></p> <p><i>For Radiological Emergency Preparedness, the High Range Radiation Monitors are used to determine the classification of an emergency. Specifically, they are used to declare a Fuel Clad Barrier Loss and a Containment Barrier Potential Loss. The design function of the HRRMs is to detect high radiation conditions during events which result in increased radiation levels. After 120 seconds, when the HRRMs are needed for use, the positive or negative TICs will have little appreciable effect other than the negative TICs may cause the keep alive source to be negated causing a malfunction alarm.</i></p> <p><i>The test report recommends that the radiation monitor output be discounted for approximately 2 minutes following a rapid increase or decrease in containment temperature.</i></p>
<i>Escalation</i>	Not Applicable.
<i>References</i>	<p>NUMARC/NESP-007, Rev. 2, 1/92 per REG GUIDE 1.101</p> <p>NUREG-1228, Source Estimates During Incident Response to Severe Nuclear Power Plant Accidents.</p> <p>NUREG 1465 Accident Source Terms for Light-Water Nuclear Power Plants</p> <p>SQN, TI-RPS-193 R6 Calibration of High Range Monitors in Containment.</p> <p>SQN Functional Evaluation, High Range Radiation Monitor Thermally Induced Currents, Rev 1</p>

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<i>Section</i>	<i>1.0</i>	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>	<i>1.3</i>	CNTMT BARRIER
<i>IC</i>	<i>1.3.6</i>	Site Emergency Director Judgement
<i>Mode</i>		1,2,3,4
<i>Description</i>		Any condition that, in the judgement of the SM/SED, indicates Loss or Potential Loss of the CNTMT Barrier comparable to the conditions listed above.
<i>Basis</i>		<p>This IC gives the SED the latitude to use his/her judgement in determining if the containment barrier is a "Potential Loss" or "Loss". This situation is usually considered when plant conditions are present that require the monitoring of CSFs or performance of EOP corrective actions. Specific cases where SED judgement may be required are the loss of instrumentation needed to monitor the CSFs and the loss of all AC power.</p> <p>Although the majority of the ICs provide very specific thresholds, the SED must remain alert to events or conditions that lead to the conclusion that exceeding the EAL threshold is imminent. If, in the judgement of the SED, an imminent situation is at hand, the classification should be made as if the thresholds have been exceeded. While this is particularly prudent at the higher emergency classes (as the early classification may provide for more effective implementation of protective measures), it is nonetheless applicable to all emergency classes.</p>
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92 per REG GUIDE 1.101

**FISSION PRODUCT
BARRIER UTILIZATION**

in

**EMERGENCY
EVENT
CLASSIFICATION**

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<i>Section</i>	1.0	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>		Not Applicable.
<i>Classification</i>		GENERAL EMERGENCY
<i>IC</i>		FG1
<i>Mode</i>		1,2,3,4
<i>Description</i>		LOSS of any two barriers and Potential LOSS of third barrier.
<i>Basis</i>		<p>Definition: Events are in process or have occurred which involve an actual or imminent substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels outside the EXCLUSION AREA BOUNDARY.</p> <p>The main differentiation between the Site Area and General Emergency classification is whether or not the EPA PAG plume exposure levels are expected to be exceeded outside the exclusion area boundary. This threshold, in addition to dynamic dose assessment considerations, addresses NRC and offsite emergency response agency concerns as to timely declaration of a General Emergency.</p> <p>The main objective of the General Emergency is to determine whether evacuation or sheltering of the general public is indicated based on EPA PAGs, and therefore should be interpreted to include radionuclide release regardless of cause. Consideration must be given to failures of systems and/or structures that provide fission product barrier integrity which is the primary method of preventing uncontrolled radionuclide releases. In terms of fission product barriers, the loss of two barriers with potential loss of the third barrier constitutes a General Emergency.</p> <p>In utilizing the fission product barrier sub-sections (i.e., Fuel Clad Barrier, RCS Barrier and CNTMT Barrier) the Site Emergency Director (SED) will use the instructions in EPIP-1, to determine the General Emergency. These instructions provide clear guidance on the proper use of the classification charts and a correct classification of a General Emergency.</p>
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101

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<i>Section</i>	1.0	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>		Not Applicable
<i>Classification</i>		SITE AREA EMERGENCY
<i>IC</i>		FS1
<i>Mode</i>		1,2,3,4
<i>Description</i>		LOSS or Potential LOSS of any two barriers.
<i>Basis</i>		<p>Definition:</p> <p>Events are in process or have occurred which involve an actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; (1) toward site personnel or equipment that could lead to the likely failure of or; (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels outside the EXCLUSION AREA BOUNDARY.</p> <p>It is considered to be a challenge to plant functions necessary for the protection of the public if the integrity of any two of the three fission product barriers has or has the potential of being degraded. This approach is more conservative than REG GUIDE 1.101 in that the containment barrier is not weighted less significantly than the other two barriers. Thus a "Loss" or "Potential Loss" of any two barriers is a Site Area Emergency.</p> <p>This approach also simplifies the Site Area Emergency classification from the fission product barrier matrix.</p> <p>In utilizing the fission product barrier sub-sections (i.e., Fuel Clad, RCS Barrier and CNTMT Barrier) the Site Emergency Director (SED) will use the instructions in EPIP-1, to determine the Site Area Emergency. These instructions provide clear guidance on the proper use of the classification charts and a correct classification of a Site Area Emergency.</p>
<i>Escalation</i>		Escalation will be based on actual or imminent substantial core degradation.
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101

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<i>Section</i>	1.0	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>		Not Applicable
<i>Classification</i>		ALERT
<i>IC</i>		FA1
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p>Any LOSS or Potential LOSS of Fuel Clad Barrier.</p> <p style="text-align: center;"><u>OR</u></p> <p>Any LOSS or Potential LOSS of RCS barrier.</p>
<i>Basis</i>		<p>Definition:</p> <p>Events are in process or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.</p> <p>The "Loss" or "Potential Loss" of either the fuel clad barrier or RCS barrier is considered to be an actual or potential substantial degradation of the level of safety of the plant. The Alert classification resulting from potential degradation of the fuel clad or RCS integrity also addresses the operation staff's need for help by staffing the Technical Support Center (TSC), independent of whether an actual decrease in plant safety is determined.</p> <p>This increased monitoring can then be used to better determine the actual plant safety state, whether escalation to a higher emergency class is warranted, or whether de-escalation or termination of the emergency class declaration is warranted. Dose consequences from these events are small fractions of the EPA PAG plume exposure levels, i.e., about 10 millirem to 100 millirem.</p> <p>In utilizing the fission product barrier sub-sections (i.e., Fuel Clad Barrier, RCS Barrier and CNTMT Barrier) the Site Emergency Director (SED) will use the instructions in EPIP-1, to determine the Alert. These instructions provide clear guidance on the proper use of the classification charts and a correct classification of an Alert.</p>
<i>Escalation</i>		Escalation will be based on actual or likely major failures of plant functions needed to protect the public.
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101

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<i>Section</i> 1.0	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>	Not Applicable.
<i>Classification</i>	UNUSUAL EVENT
<i>IC</i>	FU1
<i>Mode</i>	1,2,3,4
<i>Description</i>	LOSS or Potential LOSS of Containment Barrier.
<i>Basis</i>	<p>Definition:</p> <p>Events are in process or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.</p> <p>Potential degradation of the level of safety of the plant is indicated primarily by exceeding a plant technical specification Limiting Condition for Operation (LCO) allowable action statement time for achieving required mode change. Precursors of more serious events are also included because precursors do represent a potential degradation in the level of safety of the plant. Minor releases of radioactive materials are included. In this emergency class, however, releases do not require monitoring or offsite response (e.g., dose consequences of less than 10 millirem).</p> <p>The event classification of Unusual Event from the barrier matrix is only from a "Loss" or "Potential Loss" of the containment barrier. This is consistent with the NUMARC/NESP-007 statement, "The fuel clad barrier and the RCS barrier are weighted more heavily than the containment barrier." The "Loss or "Potential Loss" of the containment barrier alone is not considered to be substantial degradation of the level of safety of the plant when the other two fission product barriers are intact. Thus the UE classification is justified.</p> <p>In utilizing the fission product barrier sub-sections (i.e., Fuel Clad Barrier, RCS Barrier and CNTMT Barrier) the Site Emergency Director (SED) will use the instructions in EPIP-1, to determine the Unusual Event. These instructions provide clear guidance on the proper use of the classification charts and a correct classification of an Unusual Event.</p>
<i>Escalation</i>	Escalation will be based on actual or potential substantial degradation of the level of safety of the plant.
<i>References</i>	NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101

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<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.1</i>	LOSS OF INSTRUMENTATION
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" (Section 1) and "Radiological Effluents" (Section 7).
<i>Basis</i>		The basis for a General Emergency in this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected, or Radiological Effluents (Section 7).
<i>Escalation</i>		Not applicable.
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

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Section 2.0	SYSTEM DEGRADATION						
Event 2.1	LOSS OF INSTRUMENTATION						
Classification	SITE AREA EMERGENCY						
Mode	1,2,3,4						
Description	<p>Inability to monitor a SIGNIFICANT TRANSIENT in progress on either unit (1 and 2 and 3 and 4):</p> <p>1. Loss of > 75% of MCR annunciator windows and the annunciator printer and the annunciator CRT in the horseshoe or > 75% of safety system indications.</p> <p>2. Loss of ICS.</p> <p>3. Inability to directly monitor any of the following CSFs:</p> <table><tr><td>Subcriticality</td><td>PTS</td><td>Core Cooling</td></tr><tr><td>Containment</td><td>Heat Sink</td><td>Inventory</td></tr></table> <p>4. SIGNIFICANT TRANSIENT in progress.</p>	Subcriticality	PTS	Core Cooling	Containment	Heat Sink	Inventory
Subcriticality	PTS	Core Cooling					
Containment	Heat Sink	Inventory					
Basis	<p>This IC is intended to recognize the inability of the control room staff to monitor the plant response to a transient.</p> <p>When the loss of safety system annunciators is complicated with an unplanned power change as well as loss of the ICS and control room indications needed to monitor plant critical safety functions, a Site Area Emergency exists. This declaration is prudent because the control room staff cannot monitor safety functions needed for protection of the public.</p> <p>For the purposes of quantification, it is estimated that if 75% of the annunciator windows are lost there is an increased risk that a degraded plant condition could go undetected. It is not intended that a detailed count of the instrumentation be performed but only a rough approximation be used to determine the severity of the condition.</p> <p>SIGNIFICANT TRANSIENT involves an unplanned event involving one or more of the following: (1) an automatic turbine runback > 15% thermal reactor power; (2) electrical load rejection > 25% full electrical load; (3) reactor trip; (4) safety injection system activation; or (5) thermal power oscillations of ≥ 10%.</p> <p>Due to the limited number of safety systems in operation during cold shutdown and refueling modes, no initiating conditions are indicated during these modes of operation.</p>						

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<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.1</i>	LOSS OF INSTRUMENTATION
<i>Classification</i>		SITE AREA EMERGENCY (continued)
<i>Mode</i>		1,2,3,4
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
<i>References</i>		NUMARC/NESP-007, SS6, Rev. 2, 1/92 T.S. 3.3.1 Reactor Trip System Instrumentation T.S. 3.3.2 Engineering Safety Features Actuation System Instrumentation AOP P.08 Loss of Main Control Room Annunciators

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<i>Section</i>	2.0	SYSTEM DEGRADATION
<i>Event</i>	2.1	LOSS OF INSTRUMENTATION
<i>Classification</i>		ALERT
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p>On either unit an UNPLANNED loss of > 75% of the MCR annunciators and annunciator printer or > 75% of safety system indications for > 15 minutes with a SIGNIFICANT TRANSIENT in progress or ICS unavailable (1 and 2 and 3):</p> <ol style="list-style-type: none"> 1. UNPLANNED loss of > 75% of both channels of MCR annunciator windows and the annunciator printer and the annunciator CRT in the horseshoe for > 15 minutes or > 75% of safety system indications for > 15 minutes. 2. SM/SED judgment that increased surveillance is required (>shift compliment) to safely operate the unit. 3. (a or b) <ol style="list-style-type: none"> a. SIGNIFICANT TRANSIENT in progress. OR b. Loss of ICS.
<i>Basis</i>		<p>This IC indicates that when the loss of safety system annunciators and is complicated with the loss of the plant computer, or a plant transient is in progress a deterioration of the level of plant safety has occurred and an Alert should be declared. The loss of annunciators excludes scheduled maintenance and testing activities.</p> <p>Fifteen minutes was selected as a threshold value to exclude momentary transients or power losses.</p> <p>For the purposes of quantification, it is estimated that if 75% of the annunciator windows are lost there is an increased risk that a degraded plant condition could go undetected. It is not intended that a detailed count of the instrumentation be performed but only a rough approximation be used to determine the severity of the condition.</p> <p>The declaration will ensure that adequate resources are available to monitor and control plant systems so that any further degraded condition can be detected and responded to.</p> <p>SIGNIFICANT TRANSIENT involves an unplanned event involving one or more of the following: (1) an automatic turbine runback > 15 % thermal reactor power; (2) electrical load rejection > 25 % full electrical load; (3) reactor trip; or (4) safety injection system activation; or (5) thermal power oscillations \geq 10%.</p>

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<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.1</i>	LOSS OF INSTRUMENTATION
<i>Classification</i>		ALERT (continued)
<i>Mode</i>		1,2,3,4
<i>Basis</i> (continued)		Due to the limited number of safety systems in operation during cold shutdown and refueling modes, no initiating conditions are indicated during these modes of operation.
<i>Escalation</i>		Escalation will be based on the inability of the operating crew to monitor a transient in progress.
<i>References</i>		NUMARC/NESP-007, SA4, Rev. 2, 1/92 T.S. 3.3.1 Reactor Trip System Instrumentation T.S. 3.3.2 Engineering Safety Features Activation System Instrumentation T.S. 3.3.3 Monitoring Instrumentation AOP P.08 Loss of Main Control Room Annunciators

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<i>Section</i>	2.0	SYSTEM DEGRADATION
<i>Event</i>	2.1	LOSS OF INSTRUMENTATION
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p>On either unit UNPLANNED loss of > 75% of the MCR annunciators or > 75% of safety system indications for > 15 minutes and ICS available (1 and 2 and 3):</p> <ol style="list-style-type: none"> 1. UNPLANNED loss of > 75% of both channels of MCR annunciator windows and the annunciator printer and the annunciator CRT in the horseshoe for > 15 minutes or > 75% of safety system indications for > 15 minutes. 2. SM/SED judgement that increased surveillance is required (>shift compliment) to safely operate the unit. 3. The ICS is capable of displaying data requested.
<i>Basis</i>		<p>For this IC, if annunciators are partially or completely lost it is still possible to use other systems to indicate plant conditions (e.g., ICS). However, it is prudent to declare an Unusual Event since there is a greater risk that a degraded condition could go undetected.</p> <p>The loss of annunciators excludes scheduled maintenance and testing activities.</p> <p>Fifteen minutes was selected as a threshold value to exclude momentary power losses or transients.</p> <p>For the purposes of quantification, it is estimated that if 75% of the annunciator windows are lost there is an increased risk that a degraded plant condition could go undetected. It is not intended that a detailed count of the instrumentation be performed but only a rough approximation be used to determine the severity of the condition.</p> <p>The declaration will ensure that adequate resources are available to monitor and control plant systems.</p> <p>Due to the limited number of safety systems in operation during cold shutdown, refueling and defueled modes, no initiating conditions are indicated during these modes of operation.</p>

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<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.1</i>	LOSS OF INSTRUMENTATION
<i>Classification</i>		UNUSUAL EVENT (continued)
<i>Mode</i>		1,2,3,4
<i>Escalation</i>		Escalation will be based on loss of annunciators complicated by the loss of ICS or a transient in progress.
<i>References</i>		NUMARC/NESP-007, SU3, Rev. 2, 1/92 T.S. 3.3.1 Reactor Trip System Instrumentation T.S. 3.3.2 Engineering Safety Features Actuation System Instrumentation AOP P.08 Loss of Main Control Room Annunciators

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<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.2</i>	LOSS OF COMMUNICATION
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Not Applicable.
<i>Basis</i>		The basis for a General Emergency in this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.2</i>	LOSS OF COMMUNICATION
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Not Applicable.
<i>Basis</i>		The basis for a Site Area Emergency in this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

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<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.2</i>	LOSS OF COMMUNICATION
<i>Classification</i>		ALERT
<i>Mode</i>		Not Applicable.
<i>Description</i>		Not Applicable.
<i>Basis</i>		The basis for an Alert in this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92 Expanded

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<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.2</i>	LOSS OF COMMUNICATION
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		All
<i>Description</i>		<p>A. UNPLANNED loss of all in-plant communication capability (1 and 2 and 3):</p> <ol style="list-style-type: none"> 1. UNPLANNED loss of EPABX phones. 2. UNPLANNED loss of all sound powered phones. 3. UNPLANNED loss of all radios. <p style="text-align: center;"><u>OR</u></p> <p>B. UNPLANNED loss of all Offsite Communication capability (1 and 2 and 3 and 4 and 5 and 6):</p> <ol style="list-style-type: none"> 1. UNPLANNED loss of all EPABX phones. 2. UNPLANNED loss of all radio frequencies. 3. UNPLANNED loss of all OPX (Microwave) system. 4. UNPLANNED loss of all 1-FB-Bell lines. 5. UNPLANNED loss of all NRC ENS and HPN phones. 6 UNPLANNED loss of all satellite phones.
<i>Basis</i>		<p>The purpose of this IC is to recognize a loss of communications capability that either defeats the plant operations staff's ability to perform routine tasks necessary for plant operations or the ability to communicate problems with offsite authorities.</p> <p>The loss of offsite communications ability is expected to be significantly more comprehensive than those addressed by 10 CFR 50.72.</p> <p>Onsite communications loss must encompass the loss of all means of routine communications (i.e., phones, page party system and radio/walkie talkies).</p> <p>Offsite communications loss must encompass the loss of all means of communications with offsite authorities. This IC is intended to be used only when extraordinary means are being utilized to make communications possible (i.e., individuals being sent to offsite locations).</p>
<i>Escalation</i>		Escalation of this event will involve the loss of other plant functions.
<i>References</i>		NUMARC/NESP-007, SU6, Rev. 2, 1/92 NP Radiological Emergency Plan (REP) 10 CFR 50.72 NUREG 0654

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<i>Section</i>	2.0	SYSTEM DEGRADATION
<i>Event</i>	2.3	FAILURE OF RX PROTECTION
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		1
<i>Description</i>		<p>Rx power >5% and not decreasing after VALID trip signals and loss of core cooling capability (1 and 2):</p> <ol style="list-style-type: none"> 1. FR-S.1 entered and immediate operator actions did not result in a reactor power of $\leq 5\%$ and decreasing. 2. (a or b) <ol style="list-style-type: none"> a. CSF status tree indicates Core Cooling Red (FR-C.1). <p style="text-align: center;"><u>OR</u></p> <ol style="list-style-type: none"> b. CSF status tree indicates Heat Sink Red (FR-H.1).
<i>Basis</i>		<p>Under the conditions of this IC, the efforts to bring the reactor to five percent or less power have been unsuccessful and, as a result, the reactor is producing more heat than the maximum decay heat load for which the safety systems were designed.</p> <p>Failure of the immediate operator actions listed in FR-S.1 to trip the reactor include actions in the main control room and in other areas of the plant.</p> <p>Although there are additional capabilities (i.e., emergency boration) to bring the plant under control, the indication of a core cooling red indicates these capabilities are not effective and are a precursor for a core melt sequence.</p> <p>In addition, the challenge to the steam generators in the early stages of the event (i.e., Heat Sink Red) indicates insufficient feed water flow to remove heat and is also a precursor for a core melt sequence.</p> <p>In either situation, if these challenges exist at a time that the reactor has not been brought $\leq 5\%$ 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.</p>
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, SG2, Rev. 2, 1/92 T.S. 3.3.1 Reactor Trip System (RTS) Instrumentation FR-S.1 Nuclear Power Generation/ATWS FR-C.1 Inadequate Core Cooling FR-H.1 Loss of Secondary Heat Sink

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<i>Section</i>	2.0	SYSTEM DEGRADATION
<i>Event</i>	2.3	FAILURE OF RX PROTECTION
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		1
<i>Description</i>		<p>Reactor power >5% and not decreasing after VALID auto and manual trip signals:</p> <p>Note: Although a mode change may occur before classification this event will still be classified and declared as a SAE.</p>
<i>Basis</i>		<p>This IC indicates a failure of the automatic and main control room manual signals to scram the reactor.</p> <p>With reactor power greater than 5% there is a challenge to the steam generators in the early stages of the event. Heat sink red indicates insufficient feed water to remove decay heat and is a precursor for a core cooling red condition.</p> <p>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 anticipatory to the fission product barrier degradation IC, its inclusion is necessary to better assure timely recognition and emergency response.</p> <p>An indication or report or condition is considered to be VALID when it is conclusively verified by: (1) an instrument channel check, (2) indications on related or redundant indicators or (3) by direct observation by plant personnel. Implicit in this definition is the need for timely assessment within 15 minutes.</p>
<i>Escalation</i>		Escalation will be based on the inability to trip the reactor and indications of heat sink red or core cooling red.
<i>References</i>		<p>NUMARC/NESP-007, SS2 (Modified), Rev. 2, 1/92</p> <p>T.S.3.3.1 Reactor Trip System (RTS) Instrumentation</p> <p>FR-S.1 Nuclear Power Generation/ATWS</p> <p>FR-H.1 Loss of Secondary Heat Sink</p>

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<i>Section</i>	2.0	SYSTEM DEGRADATION
<i>Event</i>	2.3	FAILURE OF RX PROTECTION
<i>Classification</i>		ALERT
<i>Mode</i>		1,2
<i>Description</i>		<p>Reactor power >5% and not decreasing after VALID auto trip signal but manual trip is successful (1 and 2):</p> <ol style="list-style-type: none"> 1. Reactor power >5% and not decreasing following auto. trip signal. 2. Manual trip from the MCR successfully reduces reactor power to ≤5%. <p><i>NOTE: Although a mode change will occur this event will still be classified as an ALERT.</i></p>
<i>Basis</i>		<p>This IC indicates a failure of the automatic protection system to trip the reactor with a successful manual trip being initiated.</p> <p>The Alert declaration will ensure that adequate resources, through staffing of the technical support center, are available to monitor and control plant systems such that any further degraded condition can be detected and responded to.</p> <p>An indication or report or condition is considered to be VALID when it is conclusively verified by: (1) an instrument channel check, (2) indications on related or redundant indicators or (3) by direct observation by plant personnel. Implicit in this definition is the need for timely assessment, i.e., within 15 minutes.</p>
<i>Escalation</i>		Escalation will be based on the reactor power not being reduced to less than five percent by a successful manual trip.
<i>References</i>		<p>NUMARC/NESP-007, SA2 (Modified), Rev. 2, 1/92</p> <p>T.S.3.3.1 Reactor Trip System (RTS) Instrumentation</p> <p>FR-S.1 Nuclear Power Generation/ATWS</p> <p>WOG Background Document for FR-S.1, Rev. 1B, 2/92</p>

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<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.3</i>	FAILURE OF RX PROTECTION
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" (Section 1).
<i>Basis</i>		The basis for an Unusual Event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation will be based on a successful manual trip the reactor from the main control room.
<i>References</i>		NUMARC/NESP-007, SA2 (Modified), Rev. 2, 1/92

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<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.4</i>	FUEL CLAD DEGRADATION
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" (Section 1).
<i>Basis</i>		The basis for a General Emergency in this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.4</i>	FUEL CLAD DEGRADATION
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" (Section 1).
<i>Basis</i>		The basis for a Site Area Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/924

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<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.4</i>	FUEL CLAD DEGRADATION
<i>Classification</i>		ALERT
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" (Section 1).
<i>Basis</i>		The basis for an Alert for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

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<i>Section</i>	2.0	SYSTEM DEGRADATION
<i>Event</i>	2.4	FUEL CLAD DEGRADATION
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		1,2,3
<i>Description</i>		<p>Reactor coolant system specific activity exceeds Tech. Spec. 3.4.8 LCO:</p> <p>1. Radiochemistry analysis indicates (a or b):</p> <p>a. Dose equivalent iodine (I-131) >0.35 $\mu\text{Ci/gm}$ for >48 hours or in excess of T/S figure 3.4-1 with $T_{\text{ave}} \geq 500^{\circ}\text{F}$.</p> <p style="text-align: center;"><u>OR</u></p> <p>b. Specific activity >100/ \bar{E} $\mu\text{Ci/gm}$ with $T_{\text{ave}} \geq 500^{\circ}\text{F}$.</p>
<i>Basis</i>		<p>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.</p> <p>The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) accident.</p> <p>The LCO contains specific activity limits for both dose equivalent I-131 and gross specific activity. The allowable levels are intended to limit the 2-hour dose at the exclusion area boundary to a small fraction of the 10 CFR 100 dose guideline values.</p> <p>The limits in the LCO are standardized and based on parametric evaluations of offsite radioactivity dose consequences for typical site locations.</p> <p>These parametric evaluations showed the potential offsite dose levels for a SGTR accident were an appropriately small fraction of the 10 CFR 100 guideline dose limits. Each evaluation assumes a broad range of site applicable atmospheric dispersion factors in a parametric evaluation.</p>
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
<i>References</i>		NUMARC/NESP-007, SU4, Rev. 2, 1/92 T.S. 3.4.8 RCS Specific Activity AOP R.06 High RCS Activity

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<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.5</i>	RCS UNIDENTIFIED LEAKAGE
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" (Section 1).
<i>Basis</i>		The basis for a General Emergency in this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.5</i>	RCS UNIDENTIFIED LEAKAGE
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" (Section 1).
<i>Basis</i>		The basis for a Site Area Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.5</i>	RCS UNIDENTIFIED LEAKAGE
<i>Classification</i>		ALERT
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" (Section 1).
<i>Basis</i>		The basis for an Alert for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

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<i>Section</i>	2.0	SYSTEM DEGRADATION
<i>Event</i>	2.5	RCS UNIDENTIFIED LEAKAGE
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>	1,2,3,4	
<i>Description</i>	<p>Unidentified or pressure boundary RCS leakage > 10 GPM:</p> <ol style="list-style-type: none"> 1. Unidentified or pressure boundary leakage (as defined by Tech. Specs.) > 10 GPM as indicated by (a or b): <ol style="list-style-type: none"> a. SI-OPS-068-137.0 results or RCS Flow Balance Calculation (AOP-R.05, Appendix I or J) <p style="text-align: center;"><u>OR</u></p> <ol style="list-style-type: none"> b. With RCS temperature and pressurizer level stable, the VCT level on LI-62-129 or LI-62-130 is dropping at a rate > 10 GPM. <p>NOTE: Refer to "Shutdown System Degradation" (Section 6.3).</p>	
<i>Basis</i>	<p>This IC is included as an Unusual Event because it may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. The 10 gpm value for the unidentified and pressure boundary leakage was selected as it is observable with normal control room indications.</p> <p>If the leak is not reactor coolant pressure boundary leakage, the true RCS, and the leak is isolable from the RCS, the leak should not be classified as RCS leakage. However, if the RHR system is tied to the RCS and is being used as the cooling system for the core, the RHR system is and should be considered RCS.</p> <p>Note: for the leak to be isolable, personnel must be able to operate the isolable valve(s), which isolate the leak (e.g., isolate letdown to stop a leak in the CVCS). If a leak of radioactive fluid is occurring, which is suspected to be RCS, and is not known to be isolable, this leakage should be classified as RCS leakages determined by the shift manager/Site Emergency Director.</p> <p>Only operating modes in which there is fuel in the reactor coolant system and the system is pressurized are specified.</p>	
<i>Escalation</i>	Escalation will be based on "Fission Product Barrier Matrix" (Section 1).	
<i>References</i>	NUMARC/NESP-007, SU5, Rev. 2, 1/92 T.S. 3.4.6.2 RCS Operational Leakage SI-OPS-068-137.0 RCS Water Inventory AOP R.02 Shutdown LOCA (Mode 4 or 5) AOP R.05 RCS Leak and Leak Source Identification	

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<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.6</i>	RCS IDENTIFIED LEAKAGE
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" (Section 1).
<i>Basis</i>		The basis for a General Emergency in this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.6</i>	RCS IDENTIFIED LEAKAGE
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" (Section 1).
<i>Basis</i>		The basis for a Site Area Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.6</i>	RCS IDENTIFIED LEAKAGE
<i>Classification</i>		ALERT
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" (Section 1).
<i>Basis</i>		The basis for an Alert for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

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<i>Section</i>	2.0	SYSTEM DEGRADATION
<i>Event</i>	2.6	RCS IDENTIFIED LEAKAGE
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p>Identified RCS leakage >25 GPM:</p> <p>1. Identified RCS leakage (as defined by Tech. Specs.) >25 GPM as indicated by (a or b or c):</p> <ul style="list-style-type: none"> a. SI-OPS-068-137.0 results or RCS Flow Balance Calculation (AOP-R.05, Appendix I or J). OR b. Level rise in excess of 25 GPM total into PRT, RCDT or CVCS holdup tank (Refer to TI-28). OR c. RCS leakage through a steam generator to the secondary system (primary to secondary leakage) <p>NOTE: Also refer to "Shutdown System Degradation" (Section 6.3).</p>
<i>Basis</i>		<p>This IC is included as an Unusual Event because it may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. The 25 gpm value for the identified leakage was selected as measurable. This IC is set at a higher value than unidentified due to the lesser significance of identified leakage in comparison to unidentified or pressure boundary leakage.</p> <p>If the leak is not reactor coolant pressure boundary leakage, the true RCS, and the leak is isolable from the RCS, the leak should not be classified as RCS leakage. However, if the RHR system is tied to the RCS and is being used as the cooling system for the core, the RHR system is and should be considered RCS.</p> <p>Note: for the leak to be isolable, personnel must be able to operate the isolable valve(s), which isolate the leak (e.g., isolate letdown to stop a leak in the CVCS). If a leak of radioactive fluid is occurring, which is suspected to be RCS, and is not known to be isolable, this leakage should be classified as RCS leakages determined by the shift manager/Site Emergency Director.</p> <p>Only operating modes in which there is fuel in the reactor coolant system and the system is pressurized are specified.</p>
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
<i>References</i>		NUMARC/NESP-007, SU5, Rev. 2, 1/92 T.S. 3.4.6.2 RCS Operational Leakage AOP R.02 Shutdown LOCA (Mode 4 or 5) AOP R.05 RCS Leak and Leak Source Identification

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<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.7</i>	UNCONTROLLED COOLDOWN
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" (Section 1).
<i>Basis</i>		The basis for a General Emergency in this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.7</i>	UNCONTROLLED COOLDOWN
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" (Section 1).
<i>Basis</i>		The basis for a Site Area Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation of this event will be based on "Fission Product Barrier Matrix" (Section 1).
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

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<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.7</i>	UNCONTROLLED COOLDOWN
<i>Classification</i>		ALERT
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" (Section 1).
<i>Basis</i>		The basis for an Alert for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation of this event will be based on "Fission Product Barrier Matrix" (Section 1).
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.7</i>	UNCONTROLLED COOLDOWN
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		1,2,3
<i>Description</i>		<p>UNPLANNED rapid depressurization of the main steam system resulting in a rapid RCS cooldown and safety injection initiation (1 and 2):</p> <ol style="list-style-type: none"> 1. Rapid depressurization of any or all S/Gs or the main steam system to <600 psig on PI-1-2A, 2B, or 9A, 9B, or 20A, 20B, or 27A, 27B. 2. Safety injection has initiated or is required.
<i>Basis</i>		<p>For this IC a rapid depressurization could be caused by a main steam line break or feed line break which results in rapid RCS cool down and safety injection. This EAL is therefore consistent with the definition of an Unusual Event and warrants declaration.</p> <p>Unplanned is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities.</p>
<i>Escalation</i>		Escalation of this event will be based on "Fission Product Barrier Matrix" (Section 1).
<i>References</i>		<p>NUMARC/NESP-007, HU5, Rev. 2, 1/92</p> <p>E-2 Faulted Steam Generator Isolation</p> <p>T.S. 3.3.2 Engineering Safety Features Actuation System Instrumentation (ESFAS)</p>

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<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.8</i>	TURBINE FAILURE
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" (Section 1).
<i>Basis</i>		The basis for a General Emergency in this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.8</i>	TURBINE FAILURE
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" (Section 1).
<i>Basis</i>		The basis for a Site Area Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation of this event will be based on "Fission Product Barrier Matrix" (Section 1).
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

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Section	2.0	SYSTEM DEGRADATION												
Event	2.8	TURBINE FAILURE												
Classification		ALERT												
Mode		1,2,3												
Description	<p>Turbine failure has generated projectiles that cause visible damage to any area containing safety related equipment:</p> <p>Turbine generated projectiles have resulted in visible damage in any of the following areas:</p> <table><tr><td>Control Building</td><td>Diesel Generator Building</td></tr><tr><td>Auxiliary Building</td><td>Refuel Water Storage Tank</td></tr><tr><td>Unit #1 Containment</td><td>Intake Pumping Station</td></tr><tr><td>Unit #2 Containment</td><td>Common Station Service Transformer's</td></tr><tr><td>ERCW Pumping Station</td><td>Condensate Storage Tanks</td></tr><tr><td></td><td>Additional Equipment Buildings</td></tr></table>		Control Building	Diesel Generator Building	Auxiliary Building	Refuel Water Storage Tank	Unit #1 Containment	Intake Pumping Station	Unit #2 Containment	Common Station Service Transformer's	ERCW Pumping Station	Condensate Storage Tanks		Additional Equipment Buildings
Control Building	Diesel Generator Building													
Auxiliary Building	Refuel Water Storage Tank													
Unit #1 Containment	Intake Pumping Station													
Unit #2 Containment	Common Station Service Transformer's													
ERCW Pumping Station	Condensate Storage Tanks													
	Additional Equipment Buildings													
Basis	<p>This IC is intended to address the threat to safety related equipment imposed by projectiles generated by main turbine rotating component failures. The list of areas provided includes all areas containing safety-related equipment, their controls, and their power supplies. This EAL is, therefore, consistent with the definition of an Alert in that if projectiles have damaged or penetrated areas containing safety-related equipment the potential exists for substantial degradation of the level of safety of the plant.</p> <p>PROJECTILE: An object ejected, thrown, or launched towards a plant structure resulting in damage sufficient to cause concern regarding the integrity of the affected structure or the operability or reliability of safety equipment contained therein. The source of the projectile may be onsite or offsite</p> <p>VISIBLE DAMAGE: Damage to equipment that is readily observable without measurements, testing, or analyses. Damage is sufficient enough to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, or paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included as visible damage.</p> <p>It is noted that due to Sequoyah's turbine configuration and the location of the safety related equipment, the probability of turbine projectiles causing damage to these areas is considered remote.</p>													
Escalation	Escalation of this event will be based on "Fission Product Barrier Matrix" (Section 1).													
References	NUMARC/NESP-007, HA1, Rev. 2, 1/92 FSAR 10.2.3 Turbine Missiles													

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<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.8</i>	TURBINE FAILURE
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		1,2,3
<i>Description</i>		<p>Turbine failure results in casing penetration or main generator seal damage:</p> <p>Turbine failure which results in penetration of the turbine casing or damage to main generator seals.</p> <p>NOTE: Refer to "Hazards and SED Judgement" (Section 4.3).</p>
<i>Basis</i>		<p>This IC is intended to address main turbine rotating component failures of sufficient magnitude to cause observable damage to the turbine casing or to the seals of the main turbine generator. Of major concern is the potential for leakage of combustible fluids, lubricating oils and gases (hydrogen cooling) to the plant environs. Actual fires and flammable gas build up are appropriately classified via other events. This EAL is consistent with the definition of an Unusual Event while maintaining the anticipatory nature desired and recognizing the risk to non-safety related equipment.</p>
<i>Escalation</i>		Escalation will be based on potential damage done by turbine projectiles to safety related equipment.
<i>References</i>		NUMARC/NESP-007, HU1, Rev. 2, 1/92 FSAR 10.2.3 Turbine Missiles

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<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.9</i>	SAFETY LIMIT
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Not Applicable.
<i>Basis</i>		Safety limit is not applicable for a General Emergency.
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.9</i>	SAFETY LIMIT
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Not Applicable.
<i>Basis</i>		Safety limit is not applicable for a Site Area Emergency.
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.9</i>	SAFETY LIMIT
<i>Classification</i>		ALERT
<i>Mode</i>		Not Applicable.
<i>Description</i>		Not Applicable.
<i>Basis</i>		Safety limit is not applicable for an Alert.
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

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<i>Section</i>	2.0	SYSTEM DEGRADATION
<i>Event</i>	2.9	SAFETY LIMIT
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		1,2,3,4,5
<i>Description</i>		<p>Safety limit has been exceeded (1 or 2):</p> <p>1. The combination of thermal power, RCS temperature and RCS pressure > safety limit as indicated by SQN TS Fig. 2.1-1 "Reactor Core Safety Limit".</p> <p style="text-align: center;"><u>OR</u></p> <p>2. RCS/ pressurizer pressure exceeds safety limit (>2735 psig).</p>
<i>Basis</i>		<p>This IC requires that specified acceptable fuel design limits must not be exceeded during steady state operation, normal operational transients and anticipated operational transients. This is accomplished with a departure from nucleate boiling design basis that corresponds to 95% probability with a 95% confidence level that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.</p> <p>The restrictions of this safety limit prevent overheating of the fuel and cladding as well as possible cladding perforation resulting in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady-state peak Linear Heat Rate (LHR) below the level at which centerline fuel melting occurs. Overheating of the fuel cladding is prevented by restricting the fuel operation to within the nucleate boiling regime where the heat-transfer coefficient is large and the cladding-surface temperature is slightly above the coolant saturation temperature.</p> <p>Centerline fuel melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the fuel upon centerline melting may cause the pellet to stress the cladding to the point of failure allowing an uncontrolled release of activity to the reactor coolant.</p> <p>Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film high cladding temperatures are reached and a cladding-water (zirconium-water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.</p> <p>This EAL is consistent with the definition of an Unusual Event and warrants declaration.</p>
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92 T.S. 2.1.1 and B.2.1

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<i>Section</i>	<i>3.0</i>	LOSS OF POWER
<i>Event</i>	<i>3.1</i>	LOSS OF AC (Power Ops)
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p>Prolonged loss of all offsite and all onsite AC power to EITHER UNIT (1 and 2):</p> <ol style="list-style-type: none"> 1. Both unit related 6.9KV shutdown boards de-energized for >15 minutes. 2. (a or b) <ol style="list-style-type: none"> a. Core cooling status tree red or orange path. <p style="text-align: center;"><u>OR</u></p> <ol style="list-style-type: none"> b. Restoration of either a 6.9KV shutdown board or a 6.9KV unit board is not likely within 4 hours of the loss.
<i>Basis</i>		<p>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 four hours to restore AC power is based on a site blackout coping analysis performed to conform with 10 CFR 50.63 and Regulatory Guide 1.155, "Station Blackout", as available, with appropriate allowance for offsite emergency response. Although this IC is redundant to the fission product barrier degradation IC, its inclusion is necessary to better assure timely recognition and emergency response.</p> <p>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 is appropriate, based on a reasonable assessment of the event.</p> <p>The 15 minute time duration was selected to exclude transient or momentary power losses.</p> <p>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 Site Emergency Director a reasonable idea of how quickly he may need to declare a General Emergency based on two major considerations:</p> <ol style="list-style-type: none"> 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?

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<i>Section</i>	<i>3.0</i>	LOSS OF POWER
<i>Event</i>	<i>3.1</i>	LOSS OF AC (Power Ops)
<i>Classification</i>		GENERAL EMERGENCY (continued)
<i>Mode</i>		1,2,3,4
<i>Basis (continued)</i>		<p>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?</p> <p>The indication of continuing core cooling degradation is based on fission product barrier monitoring with particular emphasis on Site Emergency Director judgement as it relates to imminent loss or potential loss of fission product barriers and degraded ability to monitor fission product barriers.</p>
<i>Escalation</i>		Not Applicable.
<i>Reference</i>		NUMARC/NESP-007, SG1, Rev 2, 1/92 FSAR 15.2.9 Loss of Offsite Power to the Station Auxiliaries FSAR 15.5.1 Environmental Consequences of a Postulated Loss of AC Power to Plant Auxiliaries T.S. 3.8.1 AC Sources, Operating T.S. 3.8.2 Onsite Power Distribution Systems, Operating AOI P.01 Loss of Offsite Power General Design Criteria 17, App. A, 10 CFR 50 NUREG 1.155 Station Blackout

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<i>Section</i>	3.0	LOSS OF POWER
<i>Event</i>	3.1	LOSS OF AC (Power Ops)
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p>Loss of all offsite and all onsite AC power to EITHER UNIT for > 15 Minutes:</p> <p>Both unit related 6.9KV shutdown boards de-energized for > 15 minutes.</p>
<i>Basis</i>		<p>Loss of 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. This event can escalate to a General Emergency.</p> <p>The 15 minute time duration was selected to exclude transient or momentary power losses.</p>
<i>Escalation</i>		Escalation of this event is based on prolonged loss of all offsite power and prolonged loss of all onsite power when combined with inadequate core cooling, and "Fission Product Barrier Matrix" (Section 1).
<i>References</i>		<p>NUMARC/NESP-007, SS1, Rev. 2, 1/92</p> <p>AOP P.01 Loss of Offsite Power</p> <p>FSAR 15.2.9 Loss of Offsite Power to the Station Auxiliaries</p> <p>General Design Criteria 17, App. A, 10 CFR 50</p> <p>FSAR 15.5.1 Environmental Consequences of a Postulated Loss of AC Power to Plant Auxiliaries</p> <p>T.S. 3.8.1 AC Sources, Operating</p> <p>T.S. 3.8.2 Onsite Power Distribution Systems, Operating</p>

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<i>Section</i>	3.0	LOSS OF POWER
<i>Event</i>	3.1	LOSS OF AC (Power Ops)
<i>Classification</i>		ALERT
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p>Loss of offsite power to EITHER UNIT with degraded onsite AC power for >15 minutes (1a and b or 2):</p> <p>1a. All four (4) 6.9KV unit boards de-energized for >15 minutes.</p> <p>b. One (1) unit related 6.9KV shutdown board de-energized for >15 minutes.</p> <p style="text-align: center;"><u>OR</u></p> <p>2. Any AC power condition lasting >15 minutes where an additional single failure will result in a unit blackout.</p>
<i>Basis</i>		<p>The condition indicated by this IC is the degradation of the offsite and onsite power systems that any additional single failure would result in a station blackout. This condition could occur due to a loss of offsite power with a concurrent failure of one emergency diesel generator to supply power to its emergency busses.</p> <p>The 15 minute time duration was selected to exclude transient or momentary power losses.</p>
<i>Escalation</i>		Escalation will be based on prolonged loss of all offsite power and prolonged loss of all onsite power.
<i>References</i>		<p>NUMARC/NESP-007,SA5, Rev. 2, 1/92</p> <p>AOP P.01 Loss of Offsite Power</p> <p>General Design Criterion 17 of App. A, 10 CFR 50</p> <p>FSAR 15.2.9 Loss of Offsite Power to the Station Auxiliaries</p> <p>FSAR 15.5.1 Environmental Consequences of a Postulated Loss of AC Power to Plant Auxiliaries</p> <p>T.S.3.8.1 AC Sources, Operating</p> <p>T.S.3.8.2 Onsite Power Distribution Systems, Operating</p>

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<i>Section</i>	3.0	LOSS OF POWER
<i>Event</i>	3.1	LOSS OF AC (Power Ops)
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p>Loss of offsite power to EITHER UNIT for > 15 minutes (1 and 2):</p> <ol style="list-style-type: none"> 1. All four (4) 6.9KV unit boards de-energized for > 15 minutes. 2. Both unit related 6.9KV shutdown boards are energized.
<i>Basis</i>		<p>Prolonged loss of AC power reduces required redundancy and potentially degrades the level of safety of the plant by rendering the plant more vulnerable to a complete loss of AC power (station blackout).</p> <p>Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.</p>
<i>Escalation</i>		Loss of one additional power supply to the shutdown board will escalate this event.
<i>References</i>		<p>NUMARC/NESP-007, SU1, Rev. 2, 1/92</p> <p>AOP P.01 Loss of Offsite Power</p> <p>FSAR 15.2.9 Loss of Offsite Power to the Station Auxiliaries</p> <p>FSAR 15.5.1 Environmental Consequences of a Postulated Loss of AC Power to Plant Auxiliaries</p> <p>General Design Criterion 17 of App. A, 10 CFR 50</p> <p>T.S. 3.8.1 AC Sources, Operating</p> <p>T.S. 3.8.2 Onsite Power Distribution Systems, Operating</p>

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<i>Section</i>	<i>3.0</i>	LOSS OF POWER
<i>Event</i>	<i>3.2</i>	LOSS OF AC (Shutdown)
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Not Applicable.
<i>Basis</i>		Loss of AC power in Mode 5 and 6 will not cause a declaration of a General Emergency.
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92

<i>Section</i>	<i>3.0</i>	LOSS OF POWER
<i>Event</i>	<i>3.2</i>	LOSS OF AC (Shutdown)
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Not Applicable.
<i>Basis</i>		Loss of AC power in Mode 5 and 6 will not cause a declaration of a Site Area Emergency.
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92

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<i>Section</i>	3.0	LOSS OF POWER
<i>Event</i>	3.2	LOSS OF AC (Shutdown)
<i>Classification</i>		ALERT
<i>Mode</i>		5,6, or Defuelled
<i>Description</i>		<p>UNPLANNED loss of all offsite and onsite AC power to EITHER UNIT for >15 minutes:</p> <p>Both unit related 6.9KV shutdown boards de-energized >15 minutes.</p> <p>Also refer to "Loss of Shutdown Systems" (Section 6.1).</p>
<i>Basis</i>		<p>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 defuelled mode this event is classified as an Alert, because of the significantly reduced decay heat and lower temperature and pressure, increasing the time to restore one of the emergency busses.</p> <p>Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.</p> <p>Unplanned is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities.</p>
<i>Escalation</i>		Escalation is not applicable from this event.
<i>References</i>		<p>NUMARC/NESP-007 SU1 (expanded), Rev 2, 1/92</p> <p>T.S. 3.8.1 AC Sources Shutdown</p> <p>T.S. 3.8.2 Onsite Power Distribution Systems - Shutdown</p>

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<i>Section</i>	<i>3.0</i>	LOSS OF POWER
<i>Event</i>	<i>3.2</i>	LOSS OF AC (Shutdown)
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		5,6, Defuelled
<i>Description</i>		<p>UNPLANNED loss of all offsite power to EITHER UNIT for >15 minutes (1 and 2):</p> <p>1. All four 6.9KV unit boards de-energized for >15 minutes.</p> <p>2. One unit related 6.9KV shutdown board de-energized for >15 minutes.</p>
<i>Basis</i>		<p>Prolonged loss of AC power reduces required redundancy and potentially degrades the level of safety of the plant by rendering the plant more vulnerable to a complete loss of AC power (station blackout).</p> <p>Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.</p> <p>Unplanned is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities.</p>
<i>Escalation</i>		Escalation will be based on loss of one additional power supply to the shutdown boards.
<i>References</i>		<p>NUMARC/NESP-007, SU1, Rev 2, 1/92</p> <p>T.S. 3.8.1 AC Sources Shutdown</p> <p>T.S. 3.8.2 Onsite Power Distribution Systems Shutdown</p>

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<i>Section</i>	<i>3.0</i>	LOSS OF POWER
<i>Event</i>	<i>3.3</i>	LOSS OF DC
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to "Fission Product Barrier Matrix" (Section 1) and "Loss of Communication" (Section 2.2).
<i>Basis</i>		The basis for a General Emergency in this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92

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<i>Section</i>	3.0	LOSS OF POWER
<i>Event</i>	3.3	LOSS OF DC
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p>Loss of all vital DC power for > 15 minutes:</p> <p>Voltage < 105V DC on 125V DC vital battery board buses I and II and III and IV for > 15 minutes.</p> <p>Also refer to the "Fission Product Barrier Matrix" (Section 1), "Loss of Communication" (Section 2.2), and "Loss of Instrumentation" (Section 2.1).</p>
<i>Basis</i>		<p>Loss of all DC power compromises the ability to monitor and control plant safety functions. Prolonged loss of all DC power will cause core uncovering and loss of containment integrity when there is significant decay heat and sensible heat in the reactor system.</p> <p>Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.</p> <p>The minimum specified independent and redundant DC power sources and distribution systems satisfy the requirements of General Design Criterion 17 of Appendix "A" to 10 CFR 50.</p>
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Matrix" (Section 1) or "Loss of Communication" (Section 2.2).
<i>References</i>		NUMARC/NESP-007, SS3, Rev. 2, 1/92 General Design Criteria 17, App. A, 10 CFR 50 FSAR 8.3.2 DC Power System T.S. 3.8.2 DC Sources - Operating AOP P.02 Loss of 125V DC Battery Board

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<i>Section</i>	<i>3.0</i>	LOSS OF POWER
<i>Event</i>	<i>3.3</i>	LOSS OF DC
<i>Classification</i>		ALERT
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to "Fission Product Barrier Matrix" (Section 1), "Loss of Communication" (Section 2.2), and "Loss of Instrumentation" (Section 2.1).
<i>Basis</i>		There is no Alert classification for this event. Reference should be made to the "Fission Product Barrier Matrix" (Section 1), "Loss of Communication" (Section 2.2), or "Loss of Instrumentation" (Section 2.1) for possible Alert or higher classifications.
<i>Escalation</i>		Escalation will be based on loss of or the inability to monitor a significant transient in progress.
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

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<i>Section</i>	3.0	LOSS OF POWER
<i>Event</i>	3.3	LOSS OF DC
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		5,6
<i>Description</i>		<p>UNPLANNED loss of a required train of DC power for > 15 minutes: (1 or 2)</p> <p>(1) Voltage < 105 VDC on 125V DC vital battery board buses I and III for > 15 Minutes.</p> <p style="text-align: center;">or</p> <p>(2) Voltage < 105 VDC on 125V DC vital battery board buses II and IV for > 15 Minutes.</p>
<i>Basis</i>		<p>The purpose of this IC 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 IC 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.</p> <p>Unplanned is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities.</p> <p>The 105 volt bus voltage is the minimum bus voltage necessary for the operation of safety related equipment. This voltage value should incorporate a margin of at least 15 minutes of operation before the onset of inability to operate those loads. 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. For a 58-string battery set the minimum voltage is typically 1.81 volts per cell.</p> <p>The fifteen minute threshold is utilized to exclude a transient or momentary power losses.</p>
<i>Escalation</i>		The event will escalate if the DC loss results in an inability to maintain cold shutdown.
<i>References</i>		NUMARC/NESP-007, SU7, Rev. 2, 1/92 FSAR 8.3.2 DC Power Sources T.S. 3.8.5 DC Sources, Operating T.S. 3.8.2 Onsite Power Distribution Systems Shutdown AOP P.02 Loss of 125V DC Vital Battery Boards

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<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGEMENT
<i>Event</i>	<i>4.1</i>	FIRE
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to "Fission Product Barrier Matrix" (Section 1).
<i>Basis</i>		The basis for a General Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007 Rev. 2, 1/92

<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGEMENT
<i>Event</i>	<i>4.1</i>	FIRE
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to "Control Room Evacuation," (Section 4.5) and "Fission Product Barrier Matrix" (Section 1).
<i>Basis</i>		<p>The basis for a Site Area Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.</p> <p>In addition the seriousness of a fire in the control room requires reference to the emergency conditions identified in "Control Room Evacuation" (Section 4.5).</p>
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
<i>References</i>		NUMARC/NESP-007 Rev. 2, 1/92

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Section	4.0	HAZARDS AND SED JUDGEMENT												
Event	4.1	FIRE												
Classification		ALERT												
Mode		All												
Description		<p>FIRE in any of the areas listed in Table 4-1 that is affecting safety related equipment required to establish or maintain safe shutdown (1 and 2):</p> <p>1. FIRE in any of the areas listed in Table 4-1.</p> <p>2. (a or b):</p> <p style="padding-left: 40px;">a. Visible damage to permanent structure or safety related equipment in the specified area is observed due to the fire.</p> <p style="text-align: center;"><u>OR</u></p> <p style="padding-left: 40px;">b. Control room indication of degraded safety system or component response due to the fire.</p>												
Basis		<p>Fires that are likely to affect the plant's safety systems represent a degraded plant condition. The fire may have damaged equipment or damage is likely due to the proximity of heat, or flame to the systems required for safe shutdown. The likelihood of damage is subjective but is based on fire location, intensity and duration without performance of a detailed damage assessment prior to classification. The determination of the safety and supporting systems necessary for safe shutdown during the applicable operating mode and the assessment of the impact of the fire on the performance of those systems will be determined by the Site Emergency Director.</p> <p>Table 4-1 Plant structures associated with fire and explosion EALs:</p> <table><tr><td>Control Building</td><td>Refuel Water Storage Tank</td></tr><tr><td>Auxiliary Building</td><td>Intake Pumping Station</td></tr><tr><td>Unit #1 Containment</td><td>Common Station Service Transformer's</td></tr><tr><td>Unit #2 Containment</td><td>Condensate Storage Tanks</td></tr><tr><td>ERCW Pumping Station</td><td>Additional Equipment Buildings</td></tr><tr><td>Diesel Generator Building</td><td></td></tr></table> <p>FIRE is combustion characterized by heat and light. Source of smoke such as slipping drive belts or overheated electrical components do not constitute fires. Observation of flame is preferred but is not required if large quantities of smoke and/or heat are observed.</p>	Control Building	Refuel Water Storage Tank	Auxiliary Building	Intake Pumping Station	Unit #1 Containment	Common Station Service Transformer's	Unit #2 Containment	Condensate Storage Tanks	ERCW Pumping Station	Additional Equipment Buildings	Diesel Generator Building	
Control Building	Refuel Water Storage Tank													
Auxiliary Building	Intake Pumping Station													
Unit #1 Containment	Common Station Service Transformer's													
Unit #2 Containment	Condensate Storage Tanks													
ERCW Pumping Station	Additional Equipment Buildings													
Diesel Generator Building														

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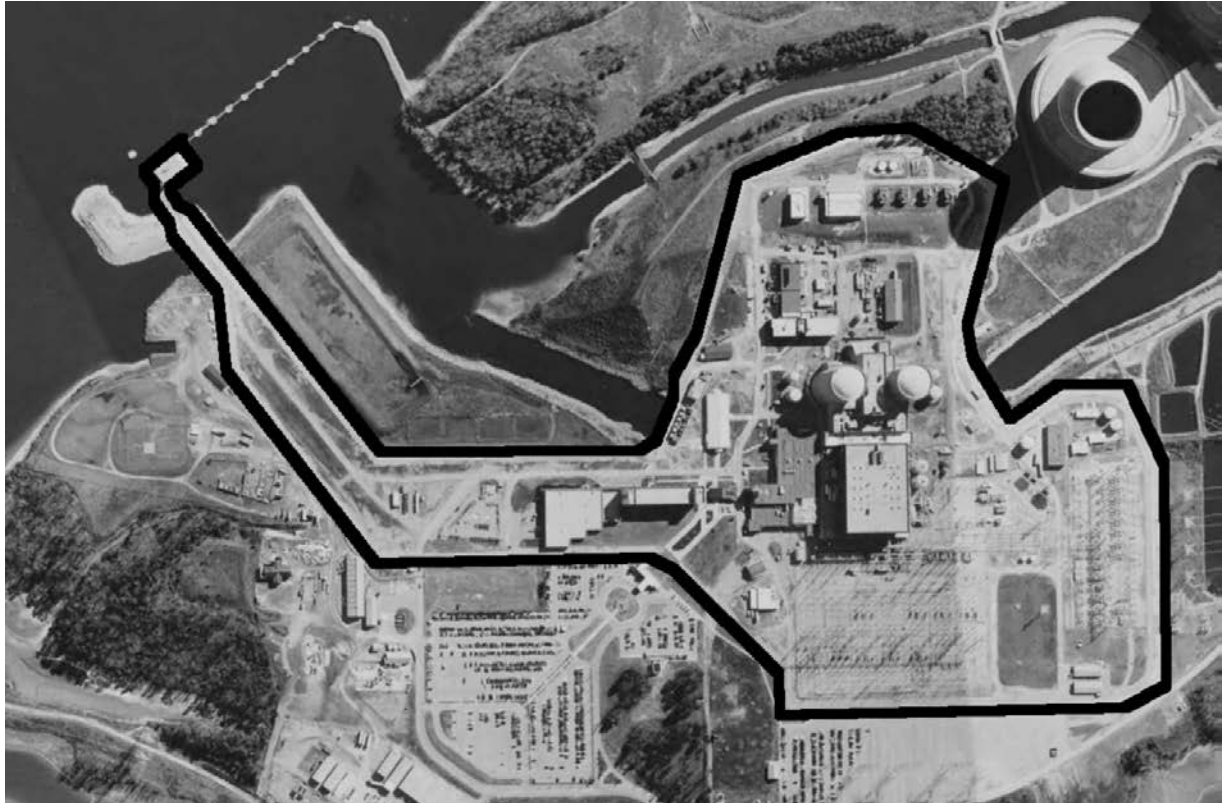
<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGEMENT
<i>Event</i>	<i>4.1</i>	FIRE
<i>Classification</i>		ALERT (continued)
<i>Mode</i>		All
<i>Basis</i> (continued)		VISIBLE DAMAGE is damage to equipment that is readily observable without measurements, testing, or analyses. Damage is sufficient enough to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, or paint blistering. Surface blemishes (e.g., paint chipping, scratches) should NOT be included.
<i>Escalation</i>		Escalation would be based on "Fission Product Barrier Matrix" (Section 1) or "Control Room Evacuation" (Section 4.5).
<i>References</i>		NUMARC/NESP-007, HA2, Rev. 2, 1/92

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Section	4.0	HAZARDS AND SED JUDGEMENT												
Event	4.1	FIRE												
Classification		UNUSUAL EVENT												
Mode		All												
Description		FIRE within the protected area (Figure 4-A) threatening any of the areas listed in Table 4-1 that is not extinguished within 15 minutes from the time of control room notification or verification of control room alarm.												
Basis		<p>This event covers verified fires that occur in selected areas of the plant that house safety systems. It also covers verified fires outside of these areas that may impact structures that contain safety systems due to the proximity of the fire. In either case these fires may be potentially significant precursors to damage of safety systems or may impact structures that contain safety systems. The initiating condition excludes fires that occur outside these key buildings, such as the warehouses, or other small fires that do not potentially affect safety systems.</p> <p>The 15 minute time limit has been established to exclude small fires that can be controlled by plant fire fighting resources.</p> <p>Verification of the fire in this event is either by direct communication with plant personnel confirming that a fire exists or the action taken by the control room personnel to determine that a fire annunciator received in the control room is not due to a spurious signal.</p> <p>Table 4-1 Plant structures associated with fire and explosion EALs:</p> <table><tr><td>Control Building</td><td>Refuel Water Storage Tank</td></tr><tr><td>Auxiliary Building</td><td>Intake Pumping Station</td></tr><tr><td>Unit #1 Containment</td><td>Common Station Service Transformer's</td></tr><tr><td>Unit #2 Containment</td><td>Condensate Storage Tanks</td></tr><tr><td>ERCW Pumping Station</td><td>Additional Equipment Buildings</td></tr><tr><td>Diesel Generator Building</td><td></td></tr></table> <p>These structures are considered to be threatened if a fire is in an area immediately adjacent to or is in actual contact with the listed structure.</p>	Control Building	Refuel Water Storage Tank	Auxiliary Building	Intake Pumping Station	Unit #1 Containment	Common Station Service Transformer's	Unit #2 Containment	Condensate Storage Tanks	ERCW Pumping Station	Additional Equipment Buildings	Diesel Generator Building	
Control Building	Refuel Water Storage Tank													
Auxiliary Building	Intake Pumping Station													
Unit #1 Containment	Common Station Service Transformer's													
Unit #2 Containment	Condensate Storage Tanks													
ERCW Pumping Station	Additional Equipment Buildings													
Diesel Generator Building														

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Figure 4-A
SEQUOYAH PROTECTED AREA



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<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGEMENT
<i>Event</i>	<i>4.1</i>	FIRE
<i>Classification</i>		UNUSUAL EVENT (continued)
<i>Mode</i>		All
<i>Basis</i> (continued)		FIRE is combustion characterized by heat and light. Source of smoke such as slipping drive belts or overheated electrical components do not constitute a fire. Observation of flame is preferred but is not required if large quantities of smoke and/or heat are observed.
<i>Escalation</i>		Escalation will be based on the fire affecting plant safety related equipment required to establish or maintain safe shutdown.
<i>References</i>		NUMARC/NESP-007, HU2, Rev. 2, 1/92 SQN Protected Area Figure 4-A

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<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGEMENT
<i>Event</i>	<i>4.2</i>	EXPLOSIONS
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" (Section 1).
<i>Basis</i>		The basis for a General Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007 Rev. 2, 1/92

<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGEMENT
<i>Event</i>	<i>4.2</i>	EXPLOSIONS
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" (Section 1).
<i>Basis</i>		The basis for a Site Area Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation would be based on "Fission Product Barrier Matrix" (Section 1).
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

SQN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX B Page B-85 Revision 103
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<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGEMENT
<i>Event</i>	<i>4.2</i>	EXPLOSIONS
<i>Classification</i>		ALERT
<i>Mode</i>		All
<i>Description</i>		<p>Explosion in any of the areas listed in Table 4-1 that is affecting safety related equipment required to establish or maintain safe shutdown (1 and 2):</p> <p>1. Explosion in any of the areas listed in Table 4-1.</p> <p>2. (a or b):</p> <p style="padding-left: 40px;">a. Visible damage to permanent structure or safety related equipment in the specified area due to the explosion.</p> <p style="text-align: center;"><u>OR</u></p> <p style="padding-left: 40px;">b. Control room indication of degraded safety system or component response due to the explosion.</p> <p>Refer to Security (Section 4.6).</p>
<i>Basis</i>		<p>EXPLOSIONS include those that are of sufficient magnitude to damage permanent structures or equipment within the identified plant areas. As used here, an explosion is a rapid, violent, unconfined combustion or a catastrophic failure of pressurized or electrical equipment, that imparts energy of sufficient force to potentially damage permanent structures or equipment.</p> <p>VISIBLE DAMAGE is damage to equipment that is readily observable without measurements, testing, or analyses. Damage is sufficient enough to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, or paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included as visible damage. The "report of visible damage" should not be interpreted as requiring a lengthy damage assessment prior to classification.</p> <p>The observation of damage to a structure is sufficient to make a declaration. The declaration of the Alert and the activation of the TSC is warranted and will provide the Site Emergency Director with resources necessary to perform damage assessment.</p>

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Section	4.0	HAZARDS AND SED JUDGEMENT
Event	4.2	EXPLOSIONS
Classification		ALERT (continued)
Mode		All
Basis (continued)		Table 4-1 plant structures associated with fire and explosion EALs: Control Building Diesel Generator Building Auxiliary Building Refuel Water Storage Tank Unit #1 Containment Intake Pumping Station Unit #2 Containment Common Station Service Transformer's ERCW Pumping Station Condensate Storage Tanks Additional Equipment Buildings
Escalation		Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
References		NUMARC/NESP-007, HA2, Rev 2, 1/92

SQN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX B Page B-87 Revision 103
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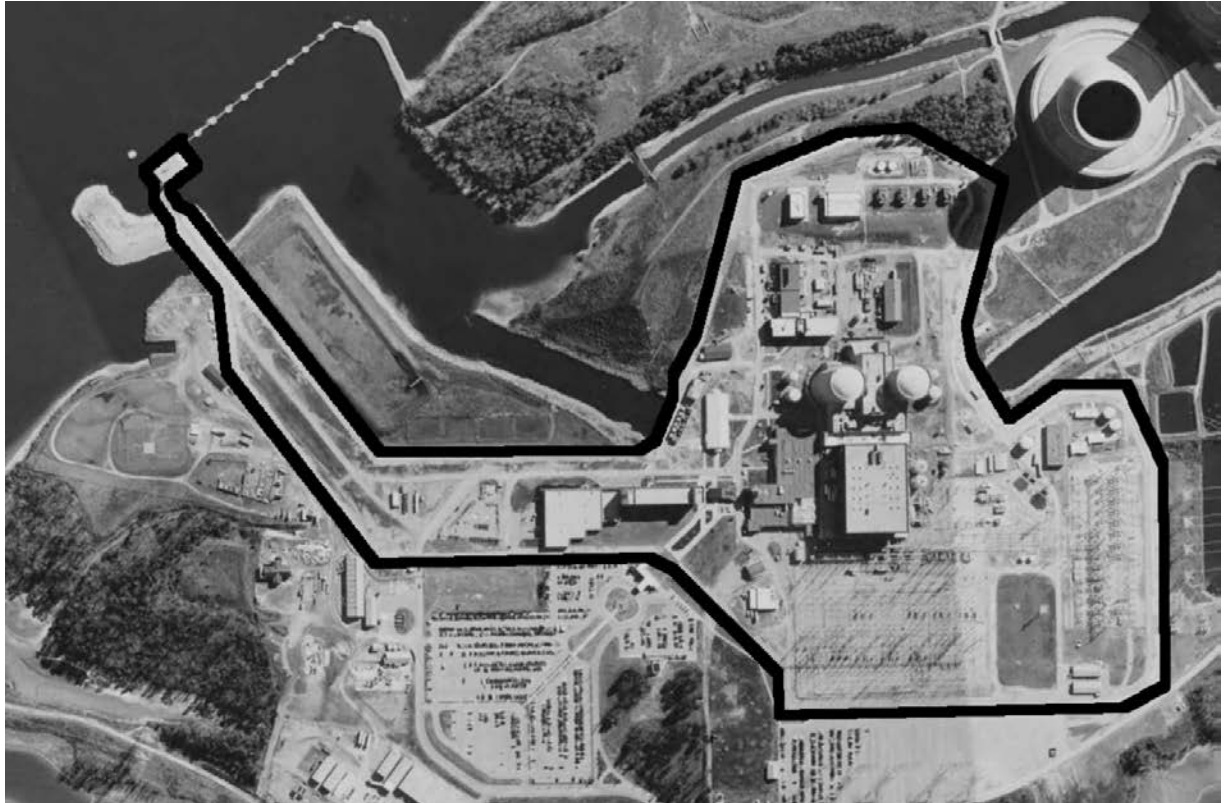
<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGEMENT
<i>Event</i>	<i>4.2</i>	EXPLOSIONS
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		All
<i>Description</i>		<p>UNPLANNED explosion within the protected area (Figure 4-A) resulting in VISIBLE DAMAGE to any permanent structure or equipment.</p> <p>Refer to Security (Section 4.6).</p>
<i>Basis</i>		<p>EXPLOSIONS include those that are of sufficient magnitude to damage permanent structures or equipment within the Protected Area. As used here, an explosion is a rapid, violent, unconfined combustion or a catastrophic failure of pressurized or electrical equipment, that imparts energy of sufficient force to potentially damage permanent structures or equipment. For this event classification, the occurrence of the explosion is sufficient to make the declaration without making a lengthy assessment of the damage.</p> <p>In addition, certain hazardous materials are transported by river barge past the Sequoyah Nuclear Plant site. Explosive materials are also transported over nearby railroad lines. Therefore, these materials were evaluated for their potential to damage the safety related structures of the plant. The materials include TNT, gasoline, liquid natural gas (LNG), and unspecified fertilizers.</p> <p>There is no potential for damage to the Sequoyah plant due to the transport of TNT from or storage of TNT at the VAA plant. The potential for damage to the Sequoyah plant from a gasoline barge explosion is considered to be negligible. It should be noted that barge shipments of LNG past Sequoyah are not likely since natural gas transportation is handled almost entirely by pipeline in this region. Therefore, the potential for an exploding LNG barge near the Intake or ERCW pumping station is a non-credible event.</p> <p>Given the low probability of a barge collision and the low percentage of fertilizer shipments on the Tennessee River, it is concluded that, because of the very low probabilities associated with the event, no hazard exists to the Intake or ERCW pumping station from the transportation of fertilizers by barge on the Tennessee River system.</p> <p>Unplanned is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities.</p>

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<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGEMENT
<i>Event</i>	<i>4.2</i>	EXPLOSIONS
<i>Classification</i>		UNUSUAL EVENT (continued)
<i>Mode</i>		All
<i>Escalation</i>		Escalation will be based on explosion damage to a structure or equipment causing a degradation in the performance of equipment required to shutdown or maintain shutdown.
<i>References</i>		NUMARC/NESP-007, HU1, Rev 2, 1/92 FSAR 2.2 Nearby Industrial, Transportation and Military Facilities

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Figure 4-A
SEQUOYAH PROTECTED AREA



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<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGEMENT
<i>Event</i>	<i>4.3</i>	FLAMMABLE GAS
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" (Section 1).
<i>Basis</i>		The basis for a General Emergency in this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92

<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGEMENT
<i>Event</i>	<i>4.3</i>	FLAMMABLE GAS
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" (Section 1).
<i>Basis</i>		The basis for a Site Area Emergency in this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92

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<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGEMENT
<i>Event</i>	<i>4.3</i>	FLAMMABLE GAS
<i>Classification</i>		ALERT
<i>Mode</i>		All
<i>Description</i>		<p>UNPLANNED release of flammable gas within a facility structure containing safety related equipment or associated with safe operation of the plant:</p> <p>Plant personnel report the average of three (3) readings taken in an approximate 10 ft. triangular area is > 25% Lower Explosive Limit as indicated on the monitoring instrument within any building listed in Table 4-2.</p> <p>Note: Refer to the Material Safety Data Sheet for the LEL.</p>
<i>Basis</i>		<p>Report or detection of flammable gases within plant structures in concentrations that are life threatening to plant personnel, affect the ability to achieve or maintain the plant in a cold shutdown condition, or maintain safe operating conditions is a degradation of the level of safety of the plant and warrants the declaration of an Alert.</p> <p>Table 4-2 Plant structures associated with toxic or flammable gas EALs:</p> <ul style="list-style-type: none"> Unit #1 Containment Unit #2 Containment Auxiliary Building Control Building Diesel Generator Building ERCW Pumping Station Intake Pumping Station CDWE Building Turbine Building Additional Equipment Buildings <p>Unplanned is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities.</p>
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
<i>References</i>		NUMARC/NESP-007, HA3, Rev 2, 1/92

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<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGEMENT
<i>Event</i>	<i>4.3</i>	FLAMMABLE GAS
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		All
<i>Description</i>	<p>A. UNPLANNED release of flammable gas within the EXCLUSION AREA BOUNDARY that may affect normal operations:</p> <p>Plant personnel report the average of three readings taken in an approximate 10 ft. triangular area is > 25% Lower Explosive Limit as indicated on the monitoring instrument within the exclusion area boundary (Figure 4-B).</p> <p style="text-align: center;"><u>OR</u></p> <p>B. Confirmed report by Local, County, or State officials that a large offsite flammable gas release has occurred within one (1) mile of the site (Figure 4-C) with potential to enter the exclusion area boundary (Figure 4-B) in concentrations >25% of Lower Explosive Limit.</p> <p>Note: Refer to the Material Safety Data Sheet for the LEL.</p>	
<i>Basis</i>	<p>Report or detection of flammable gases in concentrations within the exclusion area boundary or within the evacuation area of an offsite event (i.e., tanker truck accident releasing flammable gases, etc.) that will affect the health of plant personnel or affect the safe operation of the plant constitutes an Unusual Event. The evacuation area is as determined from the Department of Transportation (DOT) Evacuation Tables for Selected Hazardous Materials, in the DOT Emergency Response Guide for Hazardous Materials.</p> <p>In addition, it should be noted that there are no industrial or military facilities where large quantities of flammable or toxic chemicals are stored within a five mile radius of the plant. The shipping on the Tennessee River consists mainly of fuel oils, wood products and minerals. Chemicals represent only a minor percentage of the barge shipping by the Sequoyah Nuclear Plant. The release of flammable or toxic materials on the river in the vicinity of the plant will have minimal effect on the plant safety features.</p> <p>The main control room habitability during postulated hazardous chemical releases at or near the plant has been evaluated. This evaluation utilizes the approach outlined in Regulatory Guide 1.78 and concludes that the main control room habitability is not jeopardized by accidental release of chemicals. In addition, plant procedures maintain a list of onsite hazardous materials, their storage facilities, and quantities they are stored in.</p>	

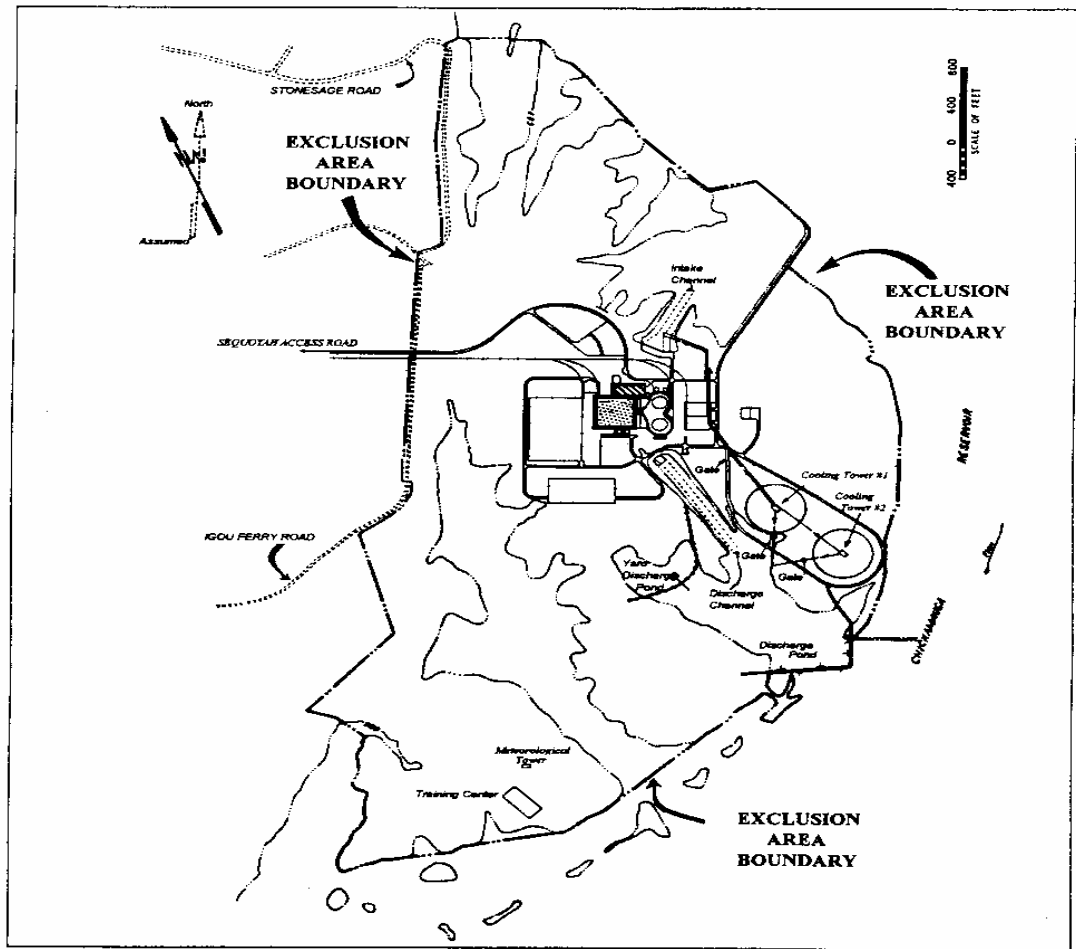
SQN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX B Page B-93 Revision 103
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<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGEMENT
<i>Event</i>	<i>4.3</i>	FLAMMABLE GAS
<i>Classification</i>		UNUSUAL EVENT (continued)
<i>Mode</i>		All
<i>Basis</i> (continued)		<p>Unplanned is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities.</p> <p>EXCLUSION AREA BOUNDARY encompasses all areas in the immediate site environs as shown on Figure 4-B.</p>
<i>Escalation</i>		Escalation of this will be based on flammable gases entering a plant area that jeopardizes life, safe operations or impacts cold shutdown capabilities.
<i>References</i>		NUMARC/NESP-007, HU3, Rev 2, 1/92 FSAR 2.2 Nearby Industrial, Transportation and Military Facilities DOT Emergency Response Guide for Hazardous Material

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Figure 4-B

SEQUOYAH EXCLUSION AREA BOUNDARY



This map illustrates the geographical layout of the Sequoyah Nuclear Plant and its surrounding infrastructure. The Tennessee River flows along the right side of the map, with Skull Island and Blue Springs located further downstream. To the west, Chickamauga Lake is shown, with Hunter Hills and Blue Springs also labeled. The Sequoyah Nuclear Plant is the central focus, featuring a large building complex, a microwave tower, and two prominent cooling towers. A vertical structure labeled 'LOCUST HILL' is situated near the plant. Various communication and support facilities are marked, including the Old Penny Com., Penny Com., Igou Com., and Denny Com. A network of roads, including Shady Grove Shores, Stonesage Road, Igou Ferry Road, Access Road And RR, and Igou Ferry Road, connects the plant to the surrounding area. A 1-mile scale bar is provided for reference. Other labeled locations include Chigger Point, Bluff, Orr Slough, and a Training Center. The map also shows a 'MET TOWER' and a 'Switch Yard' within the plant's vicinity.

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<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGEMENT
<i>Event</i>	<i>4.4</i>	TOXIC GAS OR SMOKE
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" (Section 1).
<i>Basis</i>		The basis for a General Emergency in this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92

<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGEMENT
<i>Event</i>	<i>4.4</i>	TOXIC GAS OR SMOKE
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" (Section 1).
<i>Basis</i>		The basis for a Site Area Emergency in this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92

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Section	4.0	HAZARDS AND SED JUDGEMENT										
Event	4.4	TOXIC GAS OR SMOKE										
Classification		ALERT										
Mode		All										
Description	<p>Release of TOXIC GAS or SMOKE within a facility structure which prohibits safe operation of systems required to establish or maintain cold S/D (1 and 2 and 3):</p> <p>1. Plant personnel report toxic gas or smoke within any building listed in Table 4-2.</p> <p>2. (a or b):</p> <p>a. Plant personnel report severe adverse health reactions due to toxic gas or smoke (i.e., burning eyes, nose, throat, dizziness).</p> <p style="text-align: center;"><u>OR</u></p> <p>b. Sampling indicates > Permissible Exposure Limit (PEL).</p> <p>3. Plant personnel unable to perform actions necessary to establish and maintain cold shutdown while utilizing appropriate personnel protection equipment.</p> <p>Note: Refer to the Material Safety Data Sheet for the PEL.</p>											
Basis	<p>Report or detection of toxic gases or smoke within plant structures in concentrations that are life threatening to plant personnel or affect the ability to achieve or maintain the plant in a cold shutdown condition is a degradation of the level of safety of the plant and warrants the declaration of an Alert.</p> <p>Table 4-2 Plant structures associated with toxic or flammable gas or smoke EALs:</p> <table><tr><td>Control Building</td><td>Diesel Generator Building</td></tr><tr><td>Auxiliary Building</td><td>Intake Pumping Station</td></tr><tr><td>Unit #1 Containment</td><td>CDWE Building</td></tr><tr><td>Unit #2 Containment</td><td>Turbine Building</td></tr><tr><td>ERCW Pumping Station</td><td>Additional Equipment Buildings</td></tr></table> <p>TOXIC GAS or SMOKE is a gas that is dangerous to life or limb by reason of inhalation or skin contact (e.g., chlorine, CO₂, etc.).</p>		Control Building	Diesel Generator Building	Auxiliary Building	Intake Pumping Station	Unit #1 Containment	CDWE Building	Unit #2 Containment	Turbine Building	ERCW Pumping Station	Additional Equipment Buildings
Control Building	Diesel Generator Building											
Auxiliary Building	Intake Pumping Station											
Unit #1 Containment	CDWE Building											
Unit #2 Containment	Turbine Building											
ERCW Pumping Station	Additional Equipment Buildings											
Escalation	Escalation will be based on “Fission Product Barrier Matrix” (Section 1).											
References	NUMARC/NESP-007, HA2, Rev. 2, 1/92											

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<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGEMENT
<i>Event</i>	<i>4.4</i>	TOXIC GAS OR SMOKE
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		All
<i>Description</i>		<p>A. Safe operations impeded due to access restrictions caused by TOXIC GAS or SMOKE concentrations within a facility structure listed in Table 4-2.</p> <p style="text-align: center;"><u>OR</u></p> <p>B. Confirmed report by Local, County, or State officials that an offsite TOXIC GAS release has occurred within one mile of the site (Figure 4-C) with potential to enter the EXCLUSION AREA BOUNDARY (Figure 4-B) in concentrations greater than the Permissible Exposure Limit (PEL) thus causing a site evacuation.</p> <p>Note: Refer to the Material Safety Data Sheet for the PEL.</p>
<i>Basis</i>		<p>Report or detection of a release of toxic gases or smoke in concentrations within the exclusion area boundary or within the evacuation area of an offsite event (i.e., tanker truck accident releasing toxic gases, etc.) that will affect the health of plant personnel or affect the safe operation of the plant constitutes an Unusual Event. The evacuation area is as determined from the DOT evacuation tables for selected hazardous materials, in the DOT Emergency Response Guide for Hazardous Materials.</p> <p>In addition, it should be noted that there are no industrial or military facilities where large quantities of flammable or toxic chemicals are stored within a five mile radius of the plant. The shipping on the Tennessee River consists mainly of fuel oils, wood products, and minerals. Chemicals represent only a minor percentage of the barge shipping by the Sequoyah Nuclear Plant. The release of flammable or toxic materials on the river in the vicinity of the plant will have minimal effect on the plant safety features.</p> <p>The main control room habitability during a postulated hazardous chemical release at or near the plant has been evaluated. This evaluation utilizes the approach outlined in Regulatory Guide 1.78 and concludes that the main control room habitability is not jeopardized by an accidental release of chemicals. In addition, plant procedures maintain a list of onsite hazardous materials, their storage facilities, and quantities they are stored in.</p> <p>TOXIC GAS or SMOKE is a gas that is dangerous to life or limb by reason of inhalation or skin contact (e.g., chlorine, CO₂, etc.).</p> <p>Exclusion Area Boundary encompasses all areas in the immediate site environs as shown on Figure 4-B.</p>

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Section	4.0	HAZARDS AND SED JUDGEMENT	
Event	4.4	TOXIC GAS OR SMOKE	
Classification		UNUSUAL EVENT (continued)	
Mode		All	
Basis (continued)		Table 4-2 Plant Structures associated with Toxic or Flammable Gas or Smoke EALs:	
		Control Building Auxiliary Building Unit #1 Containment Unit #2 Containment ERCW Pumping Station	Diesel Generator Building Intake Pumping Station CDWE Building Turbine Building Additional Equipment Buildings
Escalation		Escalation will be based on toxic gases or smoke entering a plant area that jeopardizes life or impacts cold shutdown capability.	
References		NUMARC/NESP-007, HU3, Rev. 2, 1/92 FSAR 2.2 Nearby Industrial, Transportation and Military Facilities DOT Emergency Response Guide for Hazardous Materials Figure 4-C One Mile Radius	

Figure 4-B

SEQUOYAH EXCLUSION AREA BOUNDARY

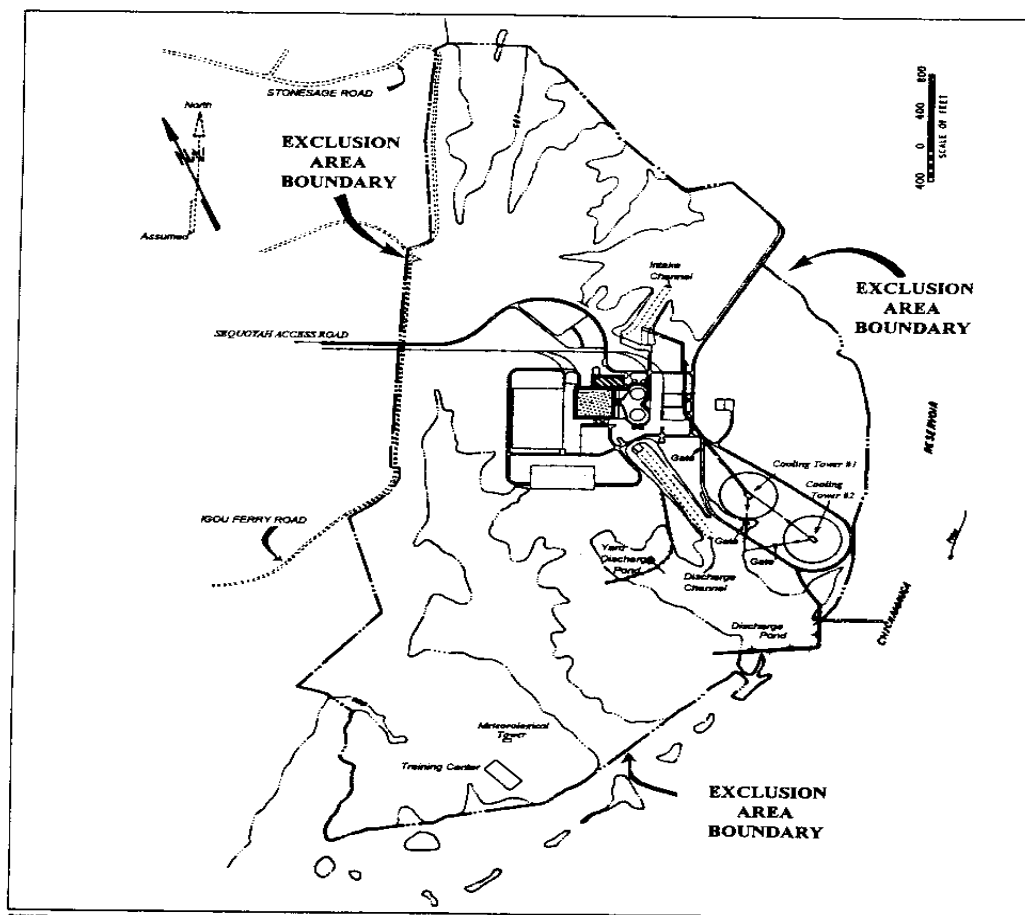
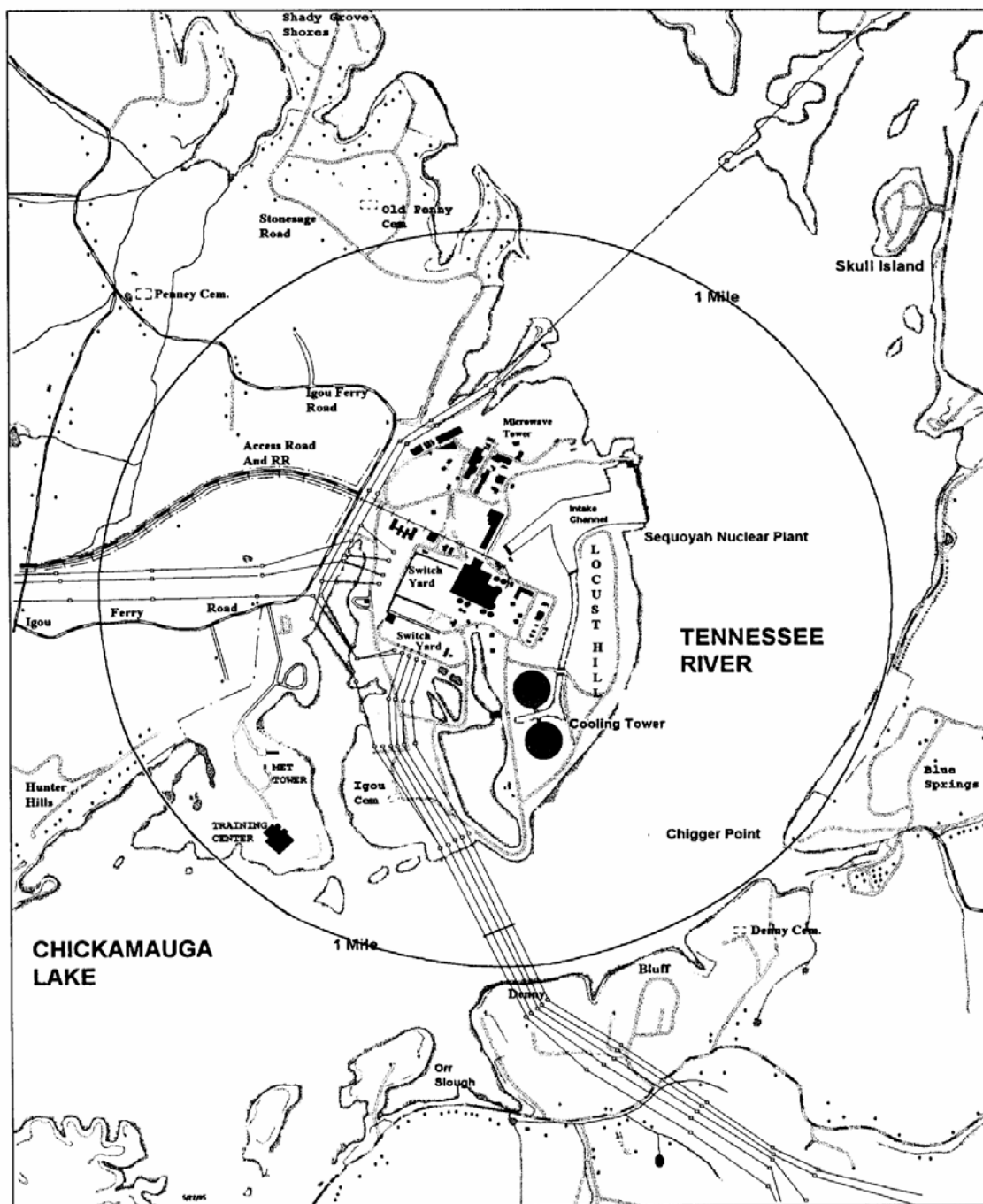


Figure 4-C
SEQUOYAH ONE MILE RADIUS

SQN	TENNESSEE VALLEY AUTHORITY NUCLEAR POWER RADIOLOGICAL EMERGENCY PLAN	NP-REP APPENDIX B Page B-102 Revision 103
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<i>Section</i>	4.0	HAZARDS AND SED JUDGEMENT
<i>Event</i>	4.5	CONTROL ROOM EVACUATION
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" (Section 1).
<i>Basis</i>		The basis for a General Emergency in this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92

<i>Section</i>	4.0	HAZARDS AND SED JUDGEMENT
<i>Event</i>	4.5	CONTROL ROOM EVACUATION
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		All
<i>Description</i>		<p>Evacuation of the main control room has been initiated and control of all necessary equipment has not been established within 15 minutes of staffing the auxiliary control room (1 and 2):</p> <ol style="list-style-type: none"> 1. AOP C.04 "Shutdown From Aux. Control Room" entered. 2. Control has not been established within 15 minutes of staffing the auxiliary control room and completing transfer of switches on Panels L11A and L11B to the Aux. position.
<i>Basis</i>		<p>Transfer of safety system control has not been performed in an expeditious manner and it is unknown if any damage has occurred to the fission product barriers. This condition warrants the declaration of a Site Area Emergency.</p> <p>The 15 minute time limit for transfer of control is based on a reasonable time period for personnel to leave the control room, arrive at the auxiliary control room area, and re-establish plant control to preclude core uncover and/or core damage per (AOP C.04) "Shutdown From Aux. Control Room".</p>
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
<i>References</i>		NUMARC/NESP-007, HS2, Rev 2, 1/92 AOP C.04 "Shutdown From Aux. Control Room"

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<i>Section</i>	4.0	HAZARDS AND SED JUDGEMENT
<i>Event</i>	4.5	CONTROL ROOM EVACUATION
<i>Classification</i>		ALERT
<i>Mode</i>		All
<i>Description</i>		<p>Evacuation of the control room is required:</p> <ol style="list-style-type: none"> 1. AOP C.04 "Shutdown From Aux. Control Room" has been entered.
<i>Basis</i>		Main Control Room evacuation requires establishment of plant control from outside the main control room (auxiliary control room) and support from the Technical Support Center and/or other Emergency Operating Centers and, for this potential substantial degradation, an Alert is warranted. A main control room evacuation represents a serious plant situation since the level of control is not as complete as it would be without the evacuation.
<i>Escalation</i>		Escalation will be based on the inability to establish plant control from outside the Main Control Room within 15 minutes.
<i>References</i>		NUMARC/NESP-007, HA5, Rev. 2, 1/92 AOP C.04 "Shutdown From Aux. Control Room"

<i>Section</i>	4.0	HAZARDS AND SED JUDGEMENT
<i>Event</i>	4.5	CONTROL ROOM EVACUATIONS
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		Not Applicable.
<i>Description</i>		An Unusual Event for this event is "Not Applicable".
<i>Basis</i>		Not Applicable.
<i>Escalation</i>		Escalation will be based on evacuation of the main control room.
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92

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<i>Section</i>	4.0	HAZARDS AND SED JUDGMENT
<i>Event</i>	4.6	SECURITY
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		All
<i>Description</i>		<p>HOSTILE ACTION Resulting in Loss of Physical Control of the Facility: (1 or 2)</p> <ol style="list-style-type: none"> 1. A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain CRITICAL SAFETY FUNCTIONS. 2. A HOSTILE ACTION has caused failure of Spent Fuel Cooling Systems and IMMINENT fuel damage is likely for a freshly off-loaded reactor core in pool.
<i>Basis</i>		<p>Note: Timely and accurate communication between Security Shift Supervisor and the Control Room is crucial for the implementation of effective Security EALs.</p> <p><u>EAL #1</u></p> <p>This EAL encompasses conditions under which a HOSTILE ACTION has resulted in a loss of physical control of VITAL AREAs (containing vital equipment or controls of vital equipment) required to maintain CRITICAL SAFETY FUNCTIONS and control of that equipment cannot be transferred to and operated from another location.</p> <p>If control of the plant equipment necessary to maintain safety functions can be transferred to another location, then the threshold is not met.</p> <p><u>EAL #2</u></p> <p>This EAL addresses failure of spent fuel cooling systems as a result of HOSTILE ACTION if IMMINENT fuel damage is likely, such as when a freshly off-loaded reactor core is in the spent fuel pool.</p>
<i>Escalation</i>		Not Applicable
<i>References</i>		NEI 99-01 R5, HG1

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<i>Section</i>	4.0	HAZARDS AND SED JUDGMENT
<i>Event</i>	4.6	SECURITY
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		All
<i>Description</i>		<p>HOSTILE ACTION within the PROTECTED AREA.</p> <p>A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervisor.</p> <p>Refer to Figure 4-A for a drawing of PROTECTED AREA.</p>
<i>Basis</i>		<p>Note: Timely and accurate communication between Security Shift Supervisor and the Control Room is crucial for the implementation of effective Security EALs.</p> <p>This condition represents an escalated threat to plant safety above that contained in the Alert in that a HOSTILE FORCE has progressed from the OWNER CONTROLLED AREA to the PROTECTED AREA.</p> <p>This EAL addresses the contingency for a very rapid progression of events, such as that experienced on September 11, 2001. It is not premised solely on the potential for a radiological release. Rather the issue includes the need for rapid assistance due to the possibility for significant and indeterminate damage from additional air, land, or water attack elements.</p> <p>The fact that the site is under serious attack with minimal time available for further preparation or additional assistance to arrive requires Offsite Response Organization readiness and preparation for the implementation of protective measures.</p> <p>This EAL addresses the potential for a very rapid progression of events due to a HOSTILE ACTION. It is not intended to address incidents that are accidental events or acts of civil disobedience, such as small aircraft impact, hunters, or physical disputes between employees within the PROTECTED AREA. Those events are adequately addressed by other EALs.</p>
<i>Escalation</i>		Escalation of this emergency classification level, if appropriate, would be based on actual plant status after impact or progression of attack.
<i>References</i>		<p>NEI 99-01 R5, HS4</p> <p>Figure 4-A Protected Area</p>

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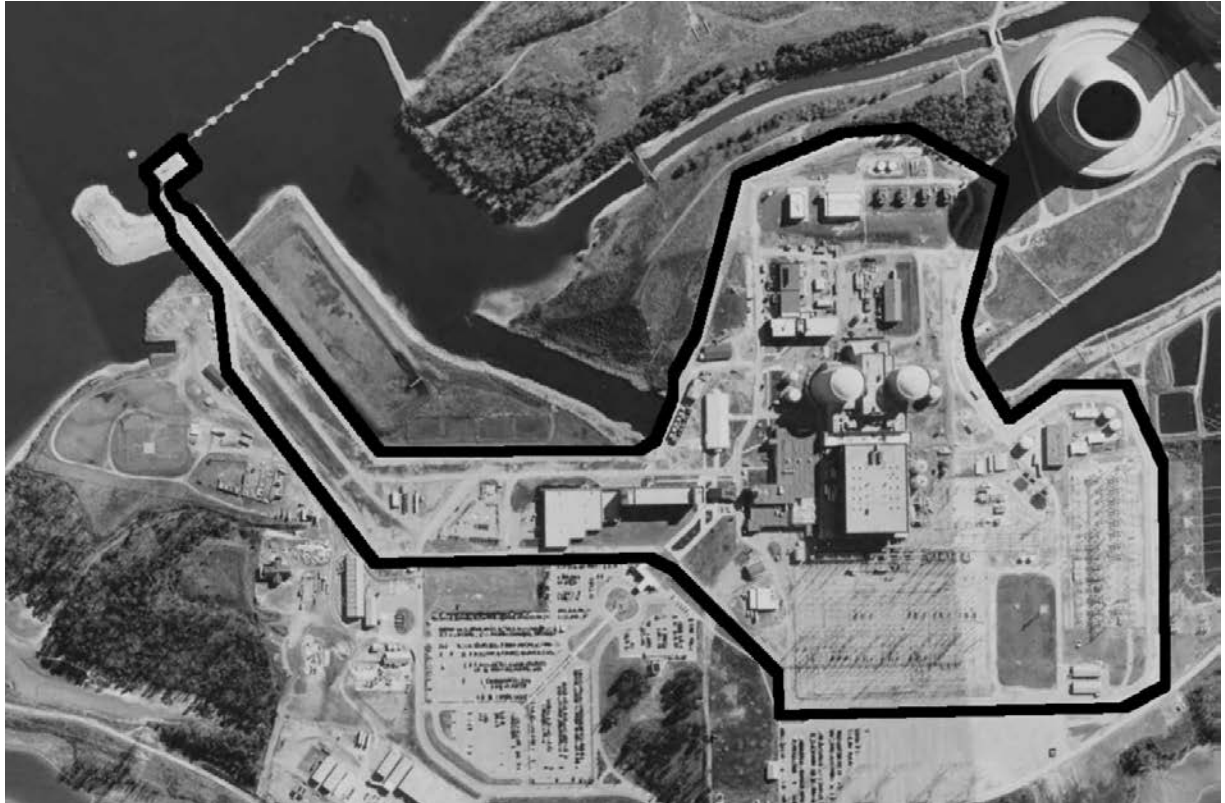
<i>Section</i>	4.0	HAZARDS AND SED JUDGMENT
<i>Event</i>	4.6	SECURITY
<i>Classification</i>		ALERT
<i>Mode</i>		All
<i>Description</i>		<p>HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat: (1 or 2)</p> <ol style="list-style-type: none"> 1. A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervisor. 2. A validated notification from NRC of an airliner attack threat within 30 minutes of the site.
<i>Basis</i>		<p>Note: Timely and accurate communication between Security Shift Supervisor and the Control Room is crucial for the implementation of effective Security EALs.</p> <p>Note: The Owner Controlled Area is defined by the Physical Site Security Plan</p> <p>These EALs address the contingency for a very rapid progression of events, such as that experienced on September 11, 2001. They are not premised solely on the potential for a radiological release. Rather the issue includes the need for rapid assistance due to the possibility for significant and indeterminate damage from additional air, land, or water attack elements.</p> <p>The fact that the site is under serious attack or is an identified attack target with minimal time available for further preparation or additional assistance to arrive requires a heightened state of readiness and implementation of protective measures that can be effective (such as on-site evacuation, dispersal or sheltering).</p> <p><u>EAL #1</u></p> <p>This EAL addresses the potential for a very rapid progression of events due to a HOSTILE ACTION. It is not intended to address incidents that are accidental events or acts of civil disobedience, such as small aircraft impact, hunters, or physical disputes between employees within the OCA. Those events are adequately addressed by other EALs.</p> <p>Note that this EAL is applicable for any HOSTILE ACTION occurring, or that has occurred, in the OWNER CONTROLLED AREA. This includes ISFSI's that may be outside the PROTECTED AREA but still within the OWNER CONTROLLED AREA.</p>

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<i>Section</i>	4.0	HAZARDS AND SED JUDGMENT
<i>Event</i>	4.6	SECURITY
<i>Classification</i>		ALERT (continued)
<i>Mode</i>		All
<i>Basis</i>		<p><u>EAL #2</u></p> <p>This EAL addresses the immediacy of an expected threat arrival or impact on the site within a relatively short time.</p> <p>The intent of this EAL is to ensure that notifications for the airliner attack threat are made in a timely manner and that Offsite Response Organizations and plant personnel are at a state of heightened awareness regarding the credible threat. Airliner is meant to be a large aircraft with the potential for causing significant damage to the plant.</p> <p>This EAL is met when a plant receives information regarding an airliner attack threat from NRC and the airliner is within 30 minutes of the plant. Only the plant to which the specific threat is made need declare the Alert.</p> <p>The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an airliner (airliner is meant to be a large aircraft with the potential for causing significant damage to the plant). The status and size of the plane may be provided by NORAD through the NRC.</p>
<i>Escalation</i>		Escalation of this event would be based on HOSTILE ACTION within the PROTECTED AREA.
<i>References</i>		NEI 99-01 R5, HA4

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**Figure 4-A
SEQUOYAH PROTECTED AREA**



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<i>Section</i>	4.0	HAZARDS AND SED JUDGMENT
<i>Event</i>	4.6	SECURITY
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		All
<i>Description</i>		<p>Confirmed SECURITY CONDITION or threat which indicates a potential degradation in the level of safety of the plant: (1 or 2 or 3)</p> <ol style="list-style-type: none"> 1. A SECURITY CONDITION that does NOT involve a HOSTILE ACTION as reported by the Security Shift Supervisor. 2. A credible SQN security threat notification. 3. A validated notification from NRC providing information of an aircraft threat.
<i>Basis</i>		<p>Note: Timely and accurate communication between Security Shift Supervision and the Control Room is crucial for the implementation of effective Security EALs.</p> <p>Security events which do not represent a potential degradation in the level of safety of the plant are reported under 10 CFR 73.71 or in some cases under 10 CFR 50.72. Security events assessed as HOSTILE ACTIONS are classifiable under EAL 4.6 Alert, SAE, and GE.</p> <p>A higher initial classification could be made based upon the nature and timing of the security threat and potential consequences. The licensee shall consider upgrading the emergency response status and emergency classification level in accordance with the site's Safeguards Contingency Plan and Emergency Plan.</p> <p><u>EAL #1</u></p> <p>Reference is made to site specific security shift supervision because these individuals are the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Safeguards Contingency Plan.</p> <p>This threshold is based on site specific security plans. Site specific Safeguards Contingency Plans are based on guidance provided by NEI 03-12.</p>

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<i>Section</i>	4.0	HAZARDS AND SED JUDGMENT
<i>Event</i>	4.6	SECURITY
<i>Classification</i>		UNUSUAL EVENT (continued)
<i>Basis</i>		<p><u>EAL #2</u></p> <p>This threshold is included to ensure that appropriate notifications for the security threat are made in a timely manner. This includes information of a credible threat. Only the plant to which the specific threat is made need declare the Notification of an Unusual Event.</p> <p>The determination of “credible” is made through use of information found in the site specific Safeguards Contingency Plan.</p> <p><u>EAL #3</u></p> <p>The intent of this EAL is to ensure that notifications for the aircraft threat are made in a timely manner and that Offsite Response Organizations and plant personnel are at a state of heightened awareness regarding the credible threat. It is not the intent of this EAL to replace existing non-hostile related EALs involving aircraft.</p> <p>This EAL is met when a plant receives information regarding an aircraft threat from NRC. Validation is performed by calling the NRC or by other approved methods of authentication. Only the plant to which the specific threat is made need declare the Unusual Event.</p> <p>The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an airliner (airliner is meant to be a large aircraft with the potential for causing significant damage to the plant). The status and size of the plane may be provided by NORAD through the NRC.</p>
<i>Escalation</i>		Escalation to Alert emergency classification level would be via EAL 4.6 would be appropriate if the threat involves an airliner within 30 minutes of the plant.
<i>References</i>		NEI 99-01 R5 HU4

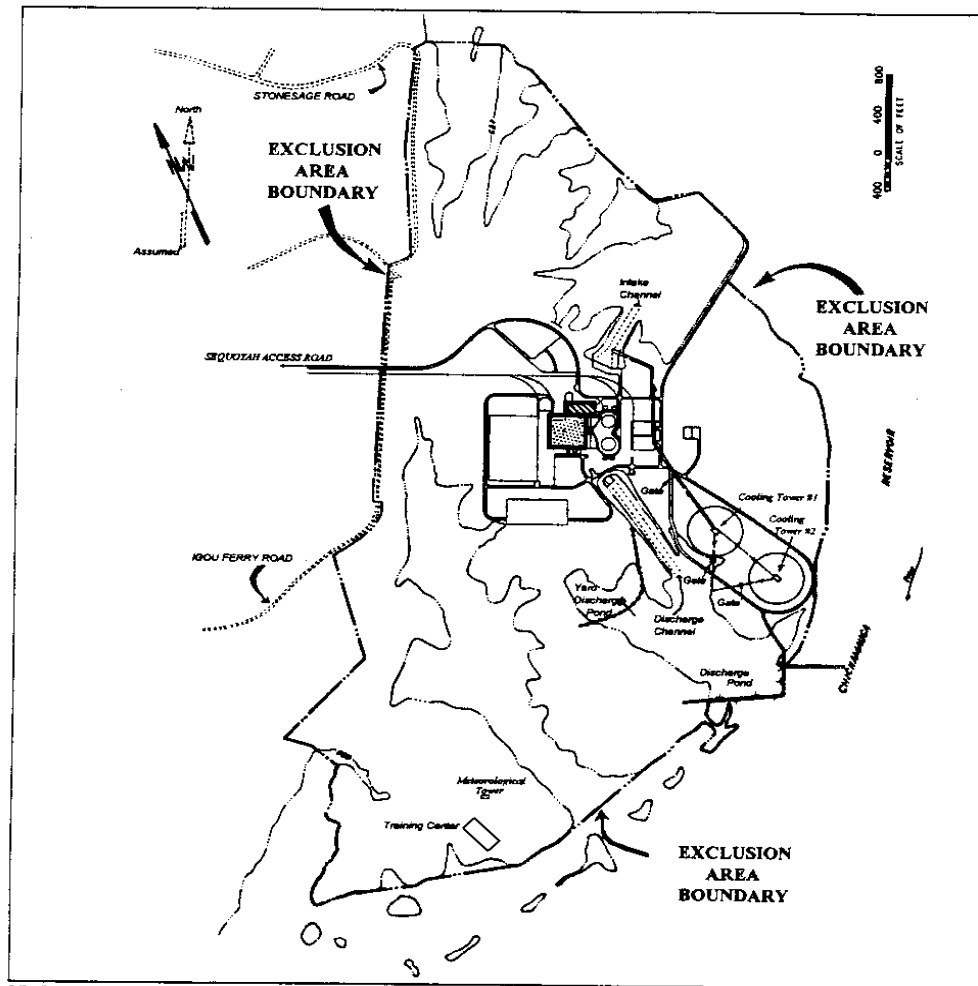
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<i>Section</i>	4.0	HAZARDS AND SED JUDGMENT
<i>Event</i>	4.7	EMERGENCY DIRECTOR JUDGMENT
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		All
<i>Description</i>		Events are in process <u>or</u> have occurred which involve Actual <u>or</u> Imminent Substantial Core Degradation <u>or</u> Melting With Potential for Loss of Containment Integrity <u>or</u> HOSTILE ACTION that result in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline Exposure Levels outside the EXCLUSION AREA BOUNDARY, Refer to Figure 4-B.
<i>Basis</i>		This event classification provides the Shift Supervisor/Site Emergency Director, the flexibility to declare a General Emergency if in their judgment unanticipated conditions not explicitly covered elsewhere warrant declaration of an emergency. The declaration of a General Emergency indicates that there is a very high probability that the fuel has been damaged and the loss of containment integrity is possible or other conditions exist that may result in a release to the environment that may be greater than the EPA Protective Action Guides.
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, HG1, HG2, Rev 2, 1/92; ; HG1, NRC Bulletin 2005-02

<i>Section</i>	4.0	HAZARDS AND SED JUDGMENT
<i>Event</i>	4.7	EMERGENCY DIRECTOR JUDGMENT
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		All
<i>Description</i>		Events are in process <u>or</u> have occurred which involve an Actual <u>or</u> Likely Major Failures of Plant Functions needed for Protection of the Public or HOSTILE ACTION that result in intentional damage or malicious acts; (1) toward site personnel or equipment that could lead to the likely failure of or; (2) that prevent effective access to equipment needed for the Protection of the Public. Any releases are not expected to result in Exposure Levels which Exceed EPA Protective Action Guideline Exposure Levels outside the EXCLUSION AREA BOUNDARY. Refer to Figure 4-B.
<i>Basis</i>		This event classification provides the Shift Supervisor/Site Emergency Director, the flexibility to declare a Site Area Emergency if in their judgment unanticipated conditions not explicitly covered elsewhere warrant declaration. The declaration of a Site Area Emergency indicates high probability of Major failures of plant functions needed to protect the public.
<i>Escalation</i>		Escalation of this event would be based on actual or imminent substantial core degradation.
<i>References</i>		NUMARC/NESP-007, HS2, Rev 2, 1/92; HS1, HS4, NRC Bulletin 2005-02

Figure 4-B

SEQUOYAH EXCLUSION AREA BOUNDARY



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<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGMENT
<i>Event</i>	<i>4.7</i>	EMERGENCY DIRECTOR JUDGMENT
<i>Classification</i>		ALERT
<i>Mode</i>		All
<i>Description</i>		Events are in process <u>or</u> have occurred which involve an Actual <u>or</u> Potential Substantial Degradation of the Level of Safety of the Plant <u>or</u> a Security Event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline Exposure Levels.
<i>Basis</i>		This event classification provides the Shift Supervisor or the Site Emergency Director, the flexibility to declare an Alert if, in their judgment, unanticipated conditions not explicitly covered elsewhere warrant declaration of an emergency.
<i>Escalation</i>		Escalation of this event would be based on actual or likely failures in plant functions needed to protect the public.
<i>References</i>		NUMARC/NESP-007, HA6, Rev 2, 1/92; HA4, HA7, HA8 NRC Bulletin 2005-02

<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGMENT
<i>Event</i>	<i>4.7</i>	EMERGENCY DIRECTOR JUDGMENT
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		All
<i>Description</i>		Events are in process <u>or</u> have occurred which indicate a Potential Degradation of the Level of Safety of the Plant <u>or</u> indicate a Security Threat to facility protection has been initiated. No Releases of Radioactive material requiring Offsite Response <u>or</u> Monitoring are expected unless further degradation of Safety Systems occurs.
<i>Basis</i>		This event classification provides the Shift Supervisor the flexibility to declare an Unusual Event if, in his judgment, unanticipated conditions not explicitly covered elsewhere warrant declaration of an emergency.
<i>Escalation</i>		Escalation of this event would be based on actual degradation of plant safety systems.
<i>References</i>		NUMARC/NESP-007, HU5, Rev 2, 1/92; HU4, NRC Bulletin 2005-02

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<i>Section</i>	<i>5.0</i>	DESTRUCTIVE PHENOMENON
<i>Event</i>	<i>5.1</i>	EARTHQUAKE
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" (Section 1).
<i>Basis</i>		The basis for a General Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	<i>5.0</i>	DESTRUCTIVE PHENOMENON
<i>Event</i>	<i>5.1</i>	EARTHQUAKE
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" (Section 1).
<i>Basis</i>		The basis for a Site Area Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

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<i>Section</i>	<i>5.0</i>	DESTRUCTIVE PHENOMENON
<i>Event</i>	<i>5.1</i>	EARTHQUAKE
<i>Classification</i>		ALERT
<i>Mode</i>		All
<i>Description</i>		<p>Earthquake detected by site seismic instrumentation (1 and 2):</p> <ol style="list-style-type: none"> 1. Panel XA-55-15B alarm window 30 (E-2) plus window 22 (D-1) activated. 2. (a or b) <ol style="list-style-type: none"> a. Ground motion sensed by plant personnel. <p style="text-align: center;"><u>OR</u></p> <ol style="list-style-type: none"> b. National Earthquake Information Center at (303) 273-8500 can confirm the event.
<i>Basis</i>		<p>A seismic event of this level can cause damage to safety related systems.</p> <p>Plant seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix "A" of 10 CFR Part 100. All specified measurement ranges represent the minimum ranges of the instruments.</p>
<i>Escalation</i>		Escalation of this event will be based on "Fission Product Barrier Matrix" (Section 1).
<i>References</i>		<p>NUMARC/NESP-007, HA1, Rev. 2, 1/92</p> <p>TRM 3.3.3.3 Seismic Monitoring Instrumentation</p> <p>NUREG 1.12 "Instrumentation for Earthquakes", April 1974</p> <p>EPRI Report NP-6695 "Guidelines for Nuclear Plant Response to Earthquakes", December 1989</p>

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<i>Section</i>	<i>5.0</i>	DESTRUCTIVE PHENOMENON
<i>Event</i>	<i>5.1</i>	EARTHQUAKE
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		All
<i>Description</i>		<p>Earthquake detected by site seismic instrumentation (1 and 2):</p> <ol style="list-style-type: none"> 1. Panel XA-55-15B alarm window 22 (D-1) activated. 2. (a or b) <ol style="list-style-type: none"> a. Ground motion sensed by plant personnel. <p style="text-align: center;"><u>OR</u></p> <ol style="list-style-type: none"> b. National Earthquake Information Center at (303) 273-8500 can confirm the event.
<i>Basis</i>		<p>A seismic event of this level can cause some minor damage to plant structure or systems but it is not expected to have any impact on overall plant safety functions.</p> <p>Plant seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix "A" of 10 CFR Part 100. All specified measurement ranges represent the minimum ranges of the instruments.</p>
<i>Escalation</i>		Escalation of this event will be based on a safe shutdown earthquake (SSE).
<i>References</i>		NUMARC/NESP-007, HU1, Rev. 2, 1/92 TRM 3.3.3.3 Seismic Monitoring Instrumentation NUREG 1.12 "Instrumentation for Earthquakes", April 1974

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<i>Section</i>	<i>5.0</i>	DESTRUCTIVE PHENOMENON
<i>Event</i>	<i>5.2</i>	TORNADO
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" (Section 1).
<i>Basis</i>		The basis for a General Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	<i>5.0</i>	DESTRUCTIVE PHENOMENON
<i>Event</i>	<i>5.2</i>	TORNADO
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" (Section 1).
<i>Basis</i>		The basis for a Site Area Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

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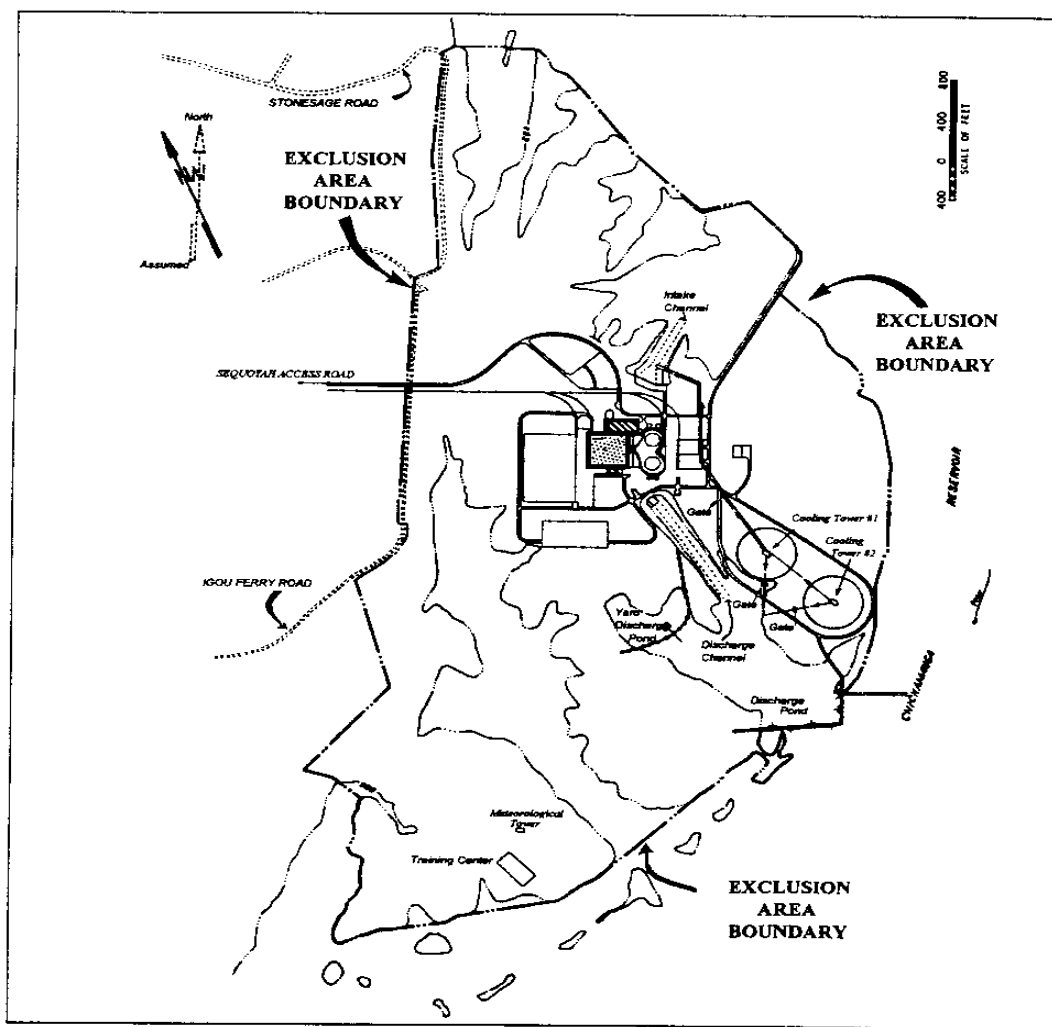
<i>Section</i>	5.0	DESTRUCTIVE PHENOMENON
<i>Event</i>	5.2	TORNADO
<i>Classification</i>		ALERT
<i>Mode</i>		All
<i>Description</i>	<p>Tornado or high winds strikes any structure listed in Table 5-1 and results in VISIBLE DAMAGE (1 and 2):</p> <ol style="list-style-type: none"> 1. Tornado or high winds (sustained > 80 mph > one minute on the plant computer) strikes any structure listed in Table 5-1. 2. (a or b) <ol style="list-style-type: none"> a. Confirmed report of any visible damage. <p style="text-align: center;"><u>OR</u></p> <ol style="list-style-type: none"> b. Control room indications of degraded safety system or component response due to the event. <p>Note: If site MET data is unavailable, the National Weather Service Morristown 1-(423)-586-8400 can provide additional information if needed.</p>	
<i>Basis</i>	<p>Tornadoes or high winds striking the structures listed in Table 5-1 can cause damage to plant structures or systems needed for safe shutdown of the plant. At Sequoyah, tornadoes are a phenomenon whose occurrence cannot be specifically predicted. The FSAR estimates the probability of a tornado occurrence onsite as about one in 6,000 years.</p> <p>Windstorms are relatively infrequent, but may occur several times a year. The records show the highest wind speed recorded in Chattanooga was 82 mph in March 1947. The records show the highest wind speed recorded in Knoxville was 73 mph in July 1961.</p> <p>Table 5-1 Plant Structures Associated With Tornado/Hi Wind EALs</p> <ul style="list-style-type: none"> Control Building Auxiliary Building Unit #1 Containment Unit #2 Containment ERCW Pumping Station Diesel Generator Building Refuel Water Storage Tank Intake Pumping Station Common Station Service Transformer's CDWE Building Turbine Building Condensate Storage Tank Additional Equipment Buildings 	

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<i>Section</i>	<i>5.0</i>	DESTRUCTIVE PHENOMENON
<i>Event</i>	<i>5.2</i>	TORNADO
<i>Classification</i>		ALERT (continued)
<i>Mode</i>		All
<i>Basis</i> (continued)		VISIBLE DAMAGE is intended to be indicative of observed physical degradation. This damage has to affect plant safety systems or equipment required to establish or maintain cold shutdown.
<i>Escalation</i>		Escalation of this event will be based on "Fission Product Barrier Matrix" (Section 1).
<i>References</i>		NUMARC/NESP-007, HAI, Rev. 2, 1/92 FSAR 1.2 General Plant Description FSAR 2.3 Meteorology

<i>Section</i>	<i>5.0</i>	DESTRUCTIVE PHENOMENON
<i>Event</i>	<i>5.2</i>	TORNADO
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		All
<i>Description</i>		Tornado within the Exclusion Area Boundary: Plant personnel report a tornado has been sighted within the exclusion area boundary (Figure 5-A).
<i>Basis</i>		A tornado touchdown near or within the exclusion area boundary may have the potential to damage plant structures containing systems required for safe shutdown of the plant. At Sequoyah, tornadoes are a phenomenon whose occurrence cannot be specifically predicted. The FSAR estimates the probability of a tornado occurrence onsite as about one in 6,000 years. EXCLUSION AREA BOUNDARY is the boundary shown on Figure 5-A.
<i>Escalation</i>		Escalation will be based on the tornado striking plant structures or high sustained winds within the protected area.
<i>References</i>		NUMARC/NESP-007, HA1, Rev. 2, 1/92 FSAR 1.2 General Plant Description

SEQUOYAH EXCLUSION AREA BOUNDARY



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<i>Section</i>	<i>5.0</i>	DESTRUCTIVE PHENOMENON
<i>Event</i>	<i>5.3</i>	AIRCRAFT/PROJECTILE IMPACT
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" (Section 1).
<i>Basis</i>		The basis for a General Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	<i>5.0</i>	DESTRUCTIVE PHENOMENON
<i>Event</i>	<i>5.3</i>	AIRCRAFT/PROJECTILE IMPACT
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" (Section 1).
<i>Basis</i>		The basis for a Site Area Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

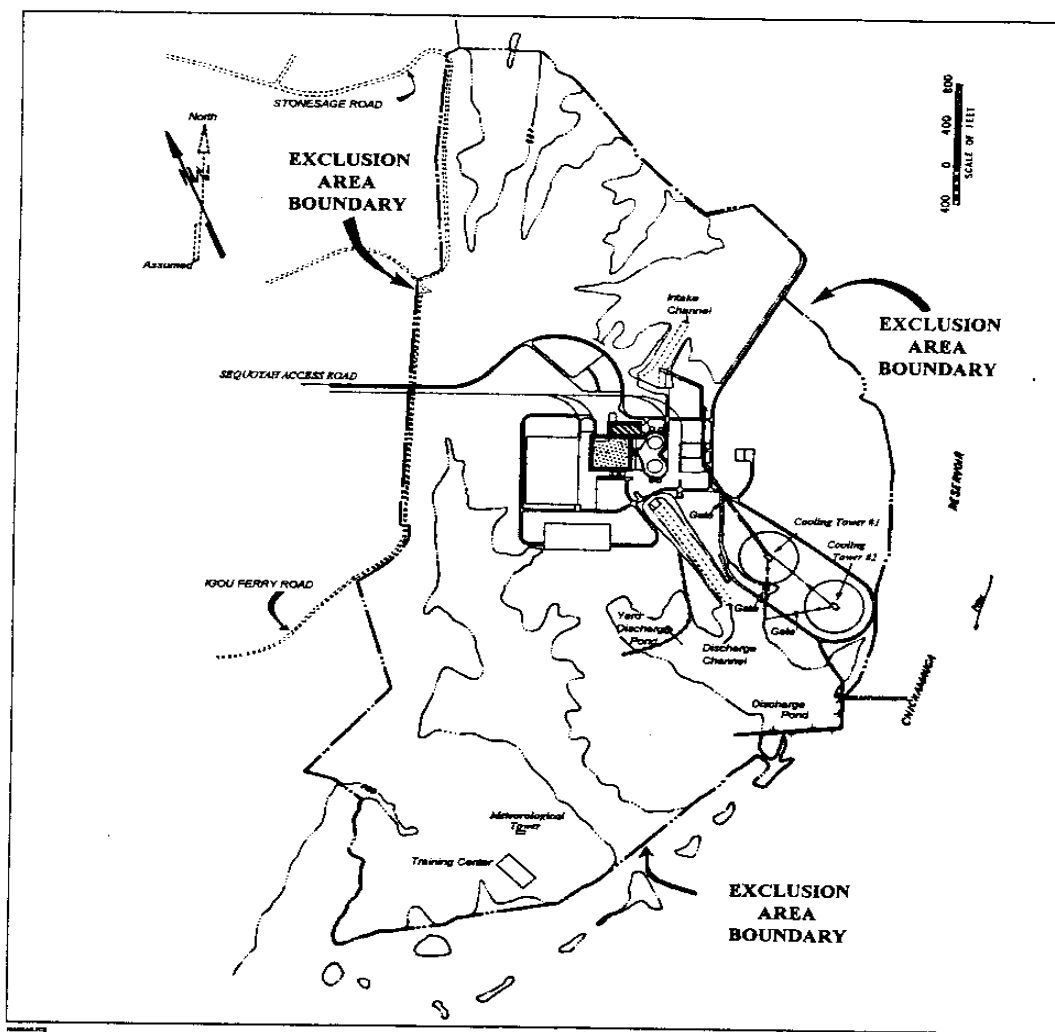
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Section	5.0	DESTRUCTIVE PHENOMENON														
Event	5.3	AIRCRAFT/PROJECTILE IMPACT														
Classification		ALERT														
Mode		All														
Description		<p>Aircraft or PROJECTILE impacts (strikes) any plant structure listed in Table 5-1 resulting in visible damage (1 and 2):</p> <p>1. Plant personnel report aircraft or projectile has impacted any structure listed in Table 5-1.</p> <p>2. (a or b)</p> <p style="padding-left: 40px;">a. Confirmed report of any visible damage.</p> <p style="text-align: center;"><u>OR</u></p> <p style="padding-left: 40px;">b. Control room indications of degraded safety system or component response due to the event within any structure listed in Table 5-1.</p>														
Basis		<p>VISIBLE DAMAGE: Damage to equipment that is readily observable without measurements, testing, or analyses. Damage is sufficient enough to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking or, paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included as visible damage</p> <p>There are no industrial or military facilities within five miles of the Sequoyah Nuclear Plant site which would potentially pose a hazard to the safe operation of the plant.</p> <p>Table 5-1 Plant structures associated with Tornado/Hi Wind and Aircraft/Projectile EALs:</p> <table><tr><td>Control Building</td><td>Intake Pumping Station</td></tr><tr><td>Auxiliary Building</td><td>Common Station Service Transformers</td></tr><tr><td>Unit #1 Containment</td><td>CDWE Building</td></tr><tr><td>Unit #2 Containment</td><td>Turbine Building</td></tr><tr><td>ERCW Pumping Station</td><td>Condensate Storage Tank</td></tr><tr><td>Diesel Generator Building</td><td>Additional Equipment Buildings</td></tr><tr><td>Refuel Water Storage Tank</td><td></td></tr></table>	Control Building	Intake Pumping Station	Auxiliary Building	Common Station Service Transformers	Unit #1 Containment	CDWE Building	Unit #2 Containment	Turbine Building	ERCW Pumping Station	Condensate Storage Tank	Diesel Generator Building	Additional Equipment Buildings	Refuel Water Storage Tank	
Control Building	Intake Pumping Station															
Auxiliary Building	Common Station Service Transformers															
Unit #1 Containment	CDWE Building															
Unit #2 Containment	Turbine Building															
ERCW Pumping Station	Condensate Storage Tank															
Diesel Generator Building	Additional Equipment Buildings															
Refuel Water Storage Tank																
Escalation		Escalation to this event will be based on "Fission Product Barrier Matrix" (Section 1).														
References		NUMARC/NESP-007, HA1, HA2, Rev. 2, 1/92 FSAR 2.2 Nearby Industrial, Transportation And Military Facilities														

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<i>Section</i>	<i>5.0</i>	DESTRUCTIVE PHENOMENON
<i>Event</i>	<i>5.3</i>	AIRCRAFT/PROJECTILE IMPACT
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		All
<i>Description</i>		<p>AIRCRAFT crash or PROJECTILE impacts (strikes) within the Exclusion Area Boundary:</p> <p>Plant personnel report aircraft crash or projectile impact within the exclusion area boundary (Figure 5-A).</p>
<i>Basis</i>		<p>Aircraft or projectile impacts within the Exclusion Area Boundary are off normal events that can indicate a potential degradation of the level of safety of the plant.</p> <p>There are no industrial or military facilities within five miles of the Sequoyah Nuclear Plant site which would potentially pose a hazard to the safe operation of the plant.</p> <p>EXCLUSION AREA BOUNDARY is the boundary shown on Figure 5-A.</p> <p>PROJECTILE includes an object ejected, thrown, or launched towards a plant structure resulting in damage sufficient to cause concern regarding the integrity of the affected structure or the operability or reliability of safety equipment contained therein. The source of the projectile may be onsite or offsite</p>
<i>Escalation</i>		Escalation to this event will be based on an impact on plant structures or barriers.
<i>References</i>		NUMARC/NESP-007, HU1, Rev. 2, 1/92 FSAR 2.2 Nearby Industrial, Transportation And Military Facilities

Figure 5-A
SEQUOYAH EXCLUSION AREA BOUNDARY



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<i>Section</i>	<i>5.0</i>	DESTRUCTIVE PHENOMENON
<i>Event</i>	<i>5.4</i>	RIVER LEVEL HIGH
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" (Section 1).
<i>Basis</i>		The basis for a General Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	<i>5.0</i>	DESTRUCTIVE PHENOMENON
<i>Event</i>	<i>5.4</i>	RIVER LEVEL HIGH
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" (Section 1).
<i>Basis</i>		The basis for a Site Area Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

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<i>Section</i>	5.0	DESTRUCTIVE PHENOMENON
<i>Event</i>	5.4	RIVER LEVEL HIGH
<i>Classification</i>		ALERT
<i>Mode</i>		All
<i>Description</i>		River reservoir level is at Stage II Flood Warning as reported by River Operations.
<i>Basis</i>		<p>The requirements for flood protection ensures that facility protective actions will be taken and operation will be terminated in the event of flood conditions. A Stage 1 flood warning is issued when the water in the forebay is predicted to exceed 697 feet Mean Sea Level USGS datum during October 1 through April 15, or 703 feet Mean Sea Level USGS datum during April 16 through September 30. A Stage II flood warning is issued when the water in the forebay is predicted to exceed 703 feet Mean Sea Level USGS datum. A maximum allowed water level of 703 feet Mean Sea Level USGS datum provides sufficient margin to ensure waves due to high winds cannot disrupt the flood mode preparation. A Stage I or Stage II flood warning requires the implementation of procedures which include plant shutdown. Further, in the event of a loss of communications simultaneous with a critical combination flood, headwaters, and/or seismically induced dam failure the plant will be shutdown and flood protection measures implemented.</p> <p>Chickamauga Lake level during nonflood conditions should be no higher than elevation 685.44 feet, top of gates, and is not likely to exceed elevation 682.5 feet, normal summer level, for any significant time. No conceivable hurricane or cyclonic-type winds could produce the some 20 feet of wave height required to reach plant grade elevation 705 feet.</p>
<i>Escalation</i>		Escalation of this event will be based on "Fission Product Barrier Matrix" (Section 1).
<i>References</i>		NUMARC/NESP-007, HA1, Rev. 2, 1/92 FSAR 2.4 Hydrologic Engineering AOP N.03 Flooding

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<i>Section</i>	5.0	DESTRUCTIVE PHENOMENON
<i>Event</i>	5.4	RIVER LEVEL HIGH
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		All
<i>Description</i>		River reservoir level is at Stage I Flood Warning as reported by River Operations.
<i>Basis</i>		<p>The requirements for flood protection ensures that facility protective actions will be taken and operation will be terminated in the event of flood conditions. A Stage 1 flood warning is issued when the water in the forebay is predicted to exceed 697 feet Mean Sea Level USGS datum during October 1 through April 15, or 703 feet Mean Sea Level USGS datum during April 16 through September 30. A Stage II flood warning is issued when the water in the forebay is predicted to exceed 703 feet Mean Sea Level USGS datum. A maximum allowed water level of 703 feet Mean Sea Level USGS datum provides sufficient margin to ensure waves due to high winds cannot disrupt the flood mode preparation. A Stage I or Stage II flood warning requires the implementation of procedures which include plant shutdown. Further, in the event of a loss of communications simultaneous with a critical combination flood, headwaters, and/or seismically induced dam failure the plant will be shutdown and flood protection measures implemented.</p> <p>Chickamauga Lake level during nonflood conditions should be no higher than elevation 685.44 feet, top of gates, and is not likely to exceed elevation 682.5 feet, normal summer level, for any significant time. No conceivable hurricane or cyclonic-type winds could produce the some 20 feet of wave height required to reach plant grade elevation 705 feet.</p> <p>Because of its inland location, the Sequoyah plant is not endangered by tsunami flooding.</p>
<i>Escalation</i>		Escalation of this event will be based on "Fission Product Barrier Matrix" (Section 1).
<i>References</i>		NUMARC/NESP-007, HU1, Rev. 2, 1/92 FSAR 2.4 Hydrologic Engineering AOP N.03 Flooding

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<i>Section</i>	<i>5.0</i>	DESTRUCTIVE PHENOMENON
<i>Event</i>	<i>5.5</i>	RIVER LEVEL LOW
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" (Section 1).
<i>Basis</i>		The basis for a General Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	<i>5.0</i>	DESTRUCTIVE PHENOMENON
<i>Event</i>	<i>5.5</i>	RIVER LEVEL LOW
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" (Section 1).
<i>Basis</i>		The basis for a Site Area Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

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<i>Section</i>	5.0	DESTRUCTIVE PHENOMENON
<i>Event</i>	5.5	RIVER LEVEL LOW
<i>Classification</i>		ALERT
<i>Mode</i>		All
<i>Description</i>		River reservoir level < 670 feet as reported by River Operations.
<i>Basis</i>		<p>The safety related water supply systems requiring river water are: the essential raw cooling water (ERCW), and that portion of the high-pressure fire-protection system (HPFP) supplying emergency feedwater to the steam generators. The fire/flood mode pumps are submersible pumps located in the intake pumping station. The entrances to the suction pipes for the fire/flood mode pumps are at elevation 651 feet 0 inches which is 32 feet and 24 feet, respectively, below the maximum normal water elevation of 683.0 and the normal minimum elevation of 675.0 for the reservoir. Abnormal reservoir level is 670 feet with a technical specification limit of 674 ft. The ERCW pump sump in this independent station is at elevation 625.0, which is 50.0' below minimum normal water elevation.</p> <p>There is a minor increase to the LBLOCA long-term containment temperature profile when coupled with loss of downstream dam (670' to 639' elevation). The loss of dam event gradually reduces the river head which in turn eventually decreases ERCW flow approximately 7%. For this scenario, long-term containment cooling begins after the reservoir level has decreased below the minimum analysis elevation 670'; this time is more than 2 hours after the peak containment temperature and pressure have already occurred.</p> <p>Since January 1940, water levels at the plant have been controlled by Chickamauga Reservoir. Since then, the minimum level at the dam was 673.3 feet on January 21, 1942.</p> <p>Because of its inland location on a relatively small, narrow lake, low water levels resulting from surges, seiches, or tsunamis are not a potential problem.</p>
<i>Escalation</i>		Escalation to this event will be based on "Fission Product Barrier Matrix" (Section 1).
<i>References</i>		NUMARC/NESP-007, HA1, Rev. 2, 1/92 FSAR 2.4 Hydrologic Engineering FSAR 6.2. Containment Systems T.S. 3.7.5 Ultimate Heat Sink AOP N.04 Break of Downstream Dam

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<i>Section</i>	5.0	DESTRUCTIVE PHENOMENON
<i>Event</i>	5.5	RIVER LEVEL LOW
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		All
<i>Description</i>		River reservoir level < 674 feet as reported by River Operations.
<i>Basis</i>		<p>The ERCW pumping station is located within the plant intake structure, and has direct communication with the main river channel for all reservoir levels including loss of downstream dam. The minimum required reservoir level for normal operation is equal to 674 feet. This level applies for ERCW supply temperature less than or equal to 87°F.</p> <p>Since January 1940, water levels at the plant have been controlled by Chickamauga Reservoir. Since then, the minimum level at the dam was 673.3 feet on January 21, 1942. The targeted minimum water level at SQN is elevation 675', which corresponds to the lower band of the winter operating zone for the Chickamauga Reservoir. Because of its location on the Chickamauga Reservoir, maintaining minimum water levels at the Sequoyah plant does not represent a problem. The high rainfall and runoff of the watershed and the regulation afforded by upstream dams assure minimum flows for plant cooling. Because of its inland location on a relatively small, narrow lake, low water levels resulting from surges, seiches, or tsunamis are not a potential problem.</p>
<i>Escalation</i>		Escalation to this event will be based on reduced river levels.
<i>References</i>		NUMARC/NESP-007, HU1, Rev. 2, 1/92 FSAR 2.4 Hydrologic Engineering FSAR 9.2. Essential Raw Cooling Water T.S. 3.7.5 Ultimate Heat Sink AOP N.04 Break of Downstream Dam

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<i>Section</i>	<i>5.0</i>	DESTRUCTIVE PHENOMENON
<i>Event</i>	<i>5.6</i>	WATERCRAFT CRASH
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" (Section 1).
<i>Basis</i>		The basis for a General Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	<i>5.0</i>	DESTRUCTIVE PHENOMENON
<i>Event</i>	<i>5.6</i>	WATERCRAFT CRASH
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" (Section 1).
<i>Basis</i>		The basis for a Site Area Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

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<i>Section</i>	<i>5.0</i>	DESTRUCTIVE PHENOMENON
<i>Event</i>	<i>5.6</i>	WATERCRAFT CRASH
<i>Classification</i>		ALERT
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" (Section 1).
<i>Basis</i>		The basis for an Alert for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	<i>5.0</i>	DESTRUCTIVE PHENOMENON
<i>Event</i>	<i>5.6</i>	WATERCRAFT CRASH
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		All
<i>Description</i>		<p>Watercraft strikes the ERCW pumping station resulting in a reduction of Essential Raw Cooling Water (ERCW) (1 and 2):</p> <ol style="list-style-type: none"> 1. Plant personnel report a watercraft has struck the ERCW pumping station. 2. (a or b) <ol style="list-style-type: none"> a. ERCW supply header pressure Train A 1(2)-PI-67-493A is <15 psig. <p style="text-align: center;"><u>OR</u></p> <ol style="list-style-type: none"> b. ERCW supply header pressure train B 1(2)-PI-67-488A is <15 psig.
<i>Basis</i>		Based on Sequoyah's river location, the potential for a watercraft accident affecting Essential Raw Cooling Water (ERCW) is remote. In the unlikely event that this accident occurs, the potential exist for possible damage to plant safety systems needed for safe shutdown. With this potential an Unusual Event is warranted.
<i>Escalation</i>		Escalation would be based on "Fission Product Barrier Matrix" (Section 1).
<i>References</i>		NUMARC/NESP-007, HU1, Rev. 2, 1/92 FSAR 2.4.8 Cooling Water Canals and Reservoirs FSAR 7.4 Systems Required for Safe Shutdown FSAR 9.2. Essential Raw Cooling Water T.S. 3.7.9 Ultimate Heat Sink

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<i>Section</i>	<i>6.0</i>	SHUTDOWN DEGRADATION
<i>Event</i>	<i>6.1</i>	LOSS OF SHUTDOWN SYSTEMS
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to "Gaseous Effluents" (Section 7.1).
<i>Basis</i>		The basis for a General Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

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<i>Section</i>	6.0	SHUTDOWN DEGRADATION
<i>Event</i>	6.1	LOSS OF SHUTDOWN SYSTEMS
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		5,6
<i>Description</i>		<p>Loss of water level in the reactor vessel that has or will uncover active fuel in the reactor vessel (1 and 2 and 3):</p> <ol style="list-style-type: none"> 1. Loss of RHR capability. 2. VALID indication that reactor vessel water level < 695'. 3. Incore TCs (if available) indicate RCS temperature > 200°F. <p>Note: If containment is open, also refer to "Gaseous Effluents" (Section 7.1).</p>
<i>Basis</i>		<p>For Sequoyah, this IC is based on concerns raised by Generic Letter 88-17, "Loss of Decay Heat Removal", SECY-91-283, "Evaluation of Shutdown and Low Power Risk Issues." A number of variables such as initial vessel level (e.g., mid-loop, reduced level/flange level, normal, or cavity filled), RCS venting strategy, decay heat removal system design, vortexing pre-disposition, steam generator U-tube draining, and level instrumentation problems can have a significant impact in causing or degrading a loss of decay heat removal. NRC analyses show that specific sequences can result in core uncover in 15 to 20 minutes and severe core damage within an hour after decay heat removal is lost. This EAL is intended to establish the escalation threshold for the declaration of a Site Area Emergency. This Site Area Emergency declaration is consistent with the need to rapidly correct the problem through the augmentation of onsite personnel and the need to inform offsite authorities. Continued degradation can rapidly result in fuel uncover and severe damage with resultant releases of a significant fraction of the gap activity. In the situation where the RCS is vented/opened to containment, the potential exists (if reactor vessel water level is not reestablished) to release radioactivity to the environment.</p> <p>The reactor vessel level indication of elevation 695' represents the water level at the hot leg center line.</p>
<i>Escalation</i>		Escalation to this event will be based on "Gaseous Effluent" (Section 7.1).
<i>References</i>		NUMARC/NESP-007, SS5 (expanded), Rev. 2, 1/92 AOP R.03 RHR System Malfunctions T.S. 3.9.8.2 Low Water Level T.S. 3.9.4 Containment Penetrations

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<i>Section</i>	<i>6.0</i>	SHUTDOWN DEGRADATION
<i>Event</i>	<i>6.1</i>	LOSS OF SHUTDOWN SYSTEMS
<i>Classification</i>		ALERT
<i>Mode</i>		5,6
<i>Description</i>		<p>Inability to maintain unit in cold shutdown when required (1 and 2):</p> <ol style="list-style-type: none"> 1. Cold shutdown required by technical specifications. 2. Incore TCs (if available) indicate core exit temperature is >200° F. <p>NOTE: If containment is open, also refer to "Gaseous Effluents" (Section 7.1).</p>
<i>Basis</i>		<p>Inability to maintain cold shutdown refers to unplanned actions resulting from either equipment malfunctions or operator error that results in an increasing trend in reactor coolant temperature and possible entry into Mode 4.</p> <p>This condition could result from the loss of cooling water to the RHR heat exchanger or equipment failures within the RHR system or AC/DC power loss to the RHR and/or service water components (i.e., CCS, ERCW). Should this condition occur, the first line of defense is to maintain heat sink capability and remove heat via the steam generators.</p> <p>For Sequoyah, 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 which can lead to conditions where decay heat removal is lost and core uncover can occur. NRC analyses show that these sequences can cause core uncover in 15 to 20 minutes and severe core damage within an hour after decay heat removal is lost. This IC and associated EALs are intended to establish the escalation threshold for an Alert. This threshold is intentionally anticipatory in that offsite doses are not expected to be affected by reaching 200° F or at the point of boiling provided the containment barrier is in place. This Alert declaration is consistent with the need to rapidly correct the problem through augmentation of onsite personnel and the need to inform offsite authorities</p>

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<i>Section</i>	6.0	SHUTDOWN DEGRADATION
<i>Event</i>	6.1	LOSS OF SHUTDOWN SYSTEMS
<i>Classification</i>		ALERT (continued)
<i>Mode</i>		5,6
<i>Escalation</i>		Loss of water level in the reactor vessel that has or will uncover fuel in the vessel will escalate this event.
<i>References</i>		NUMARC/NESP-007, SA3, Rev. 2, 1/92 AOP R.03 RHR System Malfunctions T.S. 3.1.1.2 Shutdown Margin - Tavg. ≤ 200°F

<i>Section</i>	6.0	SHUTDOWN DEGRADATION
<i>Event</i>	6.1	LOSS OF SHUTDOWN SYSTEMS
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		Not Applicable.
<i>Description</i>		An Unusual Event for this event is "Not Applicable".
<i>Basis</i>		Not Applicable.
<i>Escalation</i>		Escalation will be based on inability to maintain cold shutdown.
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

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<i>Section</i>	6.0	SHUTDOWN SYSTEM DEGRADATION
<i>Event</i>	6.2	LOSS OF SHUTDOWN CAPABILITY
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" (Section 1).
<i>Basis</i>		The basis for a General Emergency in this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	6.0	SHUTDOWN SYSTEM DEGRADATION
<i>Event</i>	6.2	LOSS OF SHUTDOWN CAPABILITY
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p>Complete loss of function needed to achieve or maintain hot shutdown (1 and 2)</p> <ol style="list-style-type: none"> 1. Hot shutdown required. 2. (a or b) <ol style="list-style-type: none"> a. CSF status tree indicates Core Cooling Red (FR-C.1). <p style="text-align: center;"><u>OR</u></p> <ol style="list-style-type: none"> b. CSF status tree indicates Heat Sink Red (FR-H.1) (RHR shutdown cooling not in service). <p>Note: Also refer to "Failure of Rx Protection" (Section 2.3).</p>
<i>Basis</i>		This IC 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. If RHR cooling is in service then the CSF status tree for Heat Sink Red is not applicable. Therefore, this comment has been added to the IC.
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
<i>References</i>		NUMARC/NESP-007, SS4, Rev. 2, 1/92 T.S. 3.4 RCS FR-C.1 Inadequate Core Cooling, FR-H.1 Loss of Heat Sink

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<i>Section</i>	6.0	SHUTDOWN SYSTEM DEGRADATION
<i>Event</i>	6.2	LOSS OF SHUTDOWN CAPABILITY
<i>Classification</i>		ALERT
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p>Complete loss of function needed to achieve cold shutdown when shutdown required by Tech. Specs.(1 and 2 and 3):</p> <ol style="list-style-type: none"> 1. Shutdown is required by technical specifications. 2. Loss of RHR capability 3. Loss of secondary heat sink and condenser.
<i>Basis</i>		<p>For this IC the inability to achieve cold shutdown when it is required refers to unplanned actions resulting in either equipment malfunctions or operator error that prevents achievement of cold shutdown.</p> <p>This condition could result from a loss of RHR capability, service water to the RHR heat exchanger, equipment failure within the RHR system, or AC/DC loss power to the RHR equipment or service water components (i.e., CCS, ERCW). The combination of this and the loss of the secondary heat sink for cooldown indicates a degradation of the level of safety and warrants the declaration of an Alert.</p>
<i>Escalation</i>		Escalation will be based on complete loss of functions needed to achieve or maintain hot shutdown.
<i>References</i>		NUMARC/NESP-007, SA3, Rev. 2,1/92 T.S. 3.4 RCS FR-C.1 Inadequate Core Cooling FR-H.1 Loss of Heat Sink

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<i>Section</i>	<i>6.0</i>	SHUTDOWN SYSTEM DEGRADATION
<i>Event</i>	<i>6.2</i>	LOSS OF SHUTDOWN CAPABILITY
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p>Inability to reach required shutdown within Tech. Spec. limits:</p> <p>The unit has not been placed in the required mode within the time prescribed by the LCO action statement.</p>
<i>Basis</i>		<p>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.</p>
<i>Escalation</i>		Escalation will be based on complete loss of functions needed to achieve cold shutdown.
<i>References</i>		NUMARC/NESP-007, SU2, Rev 2, 1/92

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<i>Section</i>	<i>6.0</i>	SHUTDOWN SYSTEM DEGRADATION
<i>Event</i>	<i>6.3</i>	LOSS OF RCS INVENTORY
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to "Gaseous Effluents" (Section 7.1).
<i>Basis</i>		The basis for a General Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	<i>6.0</i>	SHUTDOWN SYSTEM DEGRADATION
<i>Event</i>	<i>6.3</i>	LOSS OF RCS INVENTORY
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to "Gaseous Effluents" (Section 7.1).
<i>Basis</i>		The basis for a Site Area Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

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<i>Section</i>	<i>6.0</i>	SHUTDOWN SYSTEM DEGRADATION
<i>Event</i>	<i>6.3</i>	LOSS OF RCS INVENTORY
<i>Classification</i>		ALERT
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to "Gaseous Effluents" (Section 7.1).
<i>Basis</i>		The basis for an Alert for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
<i>References</i>		NUMARC/NESP-007 Rev. 2, 1/92

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<i>Section</i>	6.0	SHUTDOWN SYSTEM DEGRADATION
<i>Event</i>	6.3	LOSS OF RCS INVENTORY
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		5,6
<i>Description</i>		<p>Loss of reactor coolant system inventory with inadequate makeup (1 and 2 and 3):</p> <ol style="list-style-type: none"> 1. RCS is pressurized above atmospheric pressure. 2. Unplanned decrease in RCS or pressurizer level requiring initiation of makeup to the RCS. 3. With RCS temperature stable, the pressurizer level continues to decrease following initiation of RCS makeup.
<i>Basis</i>		<p>The purpose of this IC is to recognize a loss of RCS inventory compromising the ability to monitor and control the removal of decay heat during cold shutdown or refueling operations. The RCS level continuing to decrease after initiation of inventory makeup will eventually lead to loss of decay heat removal due to pump suction vortexing, or manual pump shutdown. Accumulation of leaking RCS water can cause water-induced damage to required equipment, can increase in-plant radiation levels, and challenge the capacity of the waste processing systems. Such a condition is a potential precursor to the loss of decay heat removal and warrants a declaration of an Unusual Event. This IC inherently addresses concerns regarding interfacing systems LOCAs, and freeze seal failures on RCS piping. This IC is intended to be anticipatory in as much as the operating crew may not have necessary equipment needed to respond to the loss.</p> <p>Unplanned is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities.</p>
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
<i>References</i>		NUMARC/NESP-007, SU7, Rev. 2, 1/92

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<i>Section</i>	7.0	RADIOLOGICAL EFFLUENTS
<i>Event</i>	7.1	GASEOUS EFFLUENTS
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		All
<i>Description</i>		<p>Exclusion Area Boundary 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 (1 or 2 or 3):</p> <ol style="list-style-type: none"> 1. A VALID rad monitor reading exceeds the values under General Emergency in Table 7-1 for > 15 minutes, unless assessment within that 15 minutes confirms that the criterion is not exceeded. <u>OR</u> 2. Field surveys indicate > 1000 mrem/hr gamma or an I-131 concentration of $3.9\text{E-}06 \mu\text{Ci}/\text{cm}^3$ at exclusion area boundary (Figure 7-A). <u>OR</u> 3. Dose assessment results indicate exclusion area boundary dose > 1000 mrem TEDE or > 5000 mrem thyroid CDE for the actual or projected duration of the release (Figure 7-A). <p>NOTE: TEDE = Total Effective Dose Equivalent and CDE = Committed Dose Equivalent</p>
<i>Basis</i>		<p>The release rates for the determination of General Emergency from monitor readings are calculated in the same manner as for the Site Area Emergency. The General Emergency site release rate below is equal to 10 times the release rate used for the Site Area Emergency.</p> <p>SQN Engineering Calculation SQS20247 results are included in Table 7.1.</p>
<i>Escalation</i>		Not Applicable.
<i>References</i>		<p>NUMARC/NESP-007, AGI, Rev. 2, 1/92 SQN Calculation SQS20247, Rev. 000 TI-CEM-030-030.0 Manual Calc. Of Plant Gas, Iodine and Particulate Release Rates For ODCM Compliance (ODCM) Offsite Dose Calculation Manual</p>

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<i>Section</i>	7.0	RADIOLOGICAL EFFLUENTS
<i>Event</i>	7.1	GASEOUS EFFLUENTS
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		All
<i>Description</i>		<p>Exclusion Area Boundary gamma 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 (1 or 2 or 3):</p> <ol style="list-style-type: none"> 1. A VALID rad monitor reading exceeds the values under Site Area in Table 7-1 for > 15 minutes unless assessment within that 15 minutes confirms that the criterion is not exceeded. <u>OR</u> 2. Field surveys indicate > 100 mrem/hr gamma or an I-131 concentration of $3.9\text{E-}07 \mu\text{Ci}/\text{cm}^3$ at exclusion area boundary (Figure 7-A). <u>OR</u> 4. Dose assessment results indicate exclusion area boundary dose > 100 mrem TEDE or > 500 mrem thyroid CDE for the actual or projected duration of the release (Figure 7-A).
<i>Basis</i>		<p>The SAE gaseous effluent EAL release rate values are those required to deliver the EAL dose in one hour. EAL release rates are back-calculated from both the 100 mrem TEDE and 500 mrem CDE (thyroid) criteria, separately. The most conservative of these release rates will be used in the determination of the EAL.</p> <p>The meteorology, calculation methodology were used in SQN Engineering Calculation SQS20247. The results of this calculation are included in Table 7.1.</p>
<i>Escalation</i>		Escalation will be based on increased release rates by a factor of 10.
<i>References</i>		<p>NUMARC/NESP-007, AGI, Rev. 2, 1/92 SQN Calculation SQS20247, Rev. 000 TI-CEM-030-030.0 Manual Calc. Of Plant Gas, Iodine and Particulate Release Rates For ODCM Compliance (ODCM) Offsite Dose Calculation Manual</p>

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<i>Section</i>	7.0	RADIOLOGICAL EFFLUENTS
<i>Event</i>	7.1	GASEOUS EFFLUENTS
<i>Classification</i>		ALERT
<i>Mode</i>		All
<i>Description</i>		<p>Any UNPLANNED release of gaseous radioactivity that exceeds 200 times the ODCM section 1.2.2.1 limit for >15 minutes (1 or 2 or 3 or 4):</p> <ol style="list-style-type: none"> 1. A VALID rad monitor reading exceeds the values under Alert in Table 7-1 for > 15 minutes, unless assessment within that 15 minutes confirms that the criterion is not exceeded. <u>OR</u> 2. Field surveys indicate > 10 mrem/hr gamma at the exclusion area boundary (Figure 7-A) > 15 minutes. <u>OR</u> 3. Dose assessment results indicate exclusion area boundary (Figure 7-A) dose >10 mrem TEDE for the duration of the release. <u>OR</u> 4. Sample results exceed 200 times the ODCM limit value for an unmonitored release of gaseous radioactivity > 15 minutes in duration. <p>NOTE: TEDE = Total Effective Dose Equivalent</p>
<i>Basis</i>		For gaseous releases, the EAL value can be determined by multiplying the Unusual Event noble gas monitor readings by a factor of 100.
<i>Escalation</i>		Escalation will be based on dose rates greater than 100 mrem TEDE or 500 mrem thyroid CDE.
<i>References</i>		NUMARC/NESP-007, AAI, Rev. 2, 1/92 SQN Calculation SQS20247, Rev. 000 TI-CEM-030-030.0 Manual Calc. Of Plant Gas, Iodine and Particulate Release Rates For ODCM Compliance (ODCM) Offsite Dose Calculation Manual

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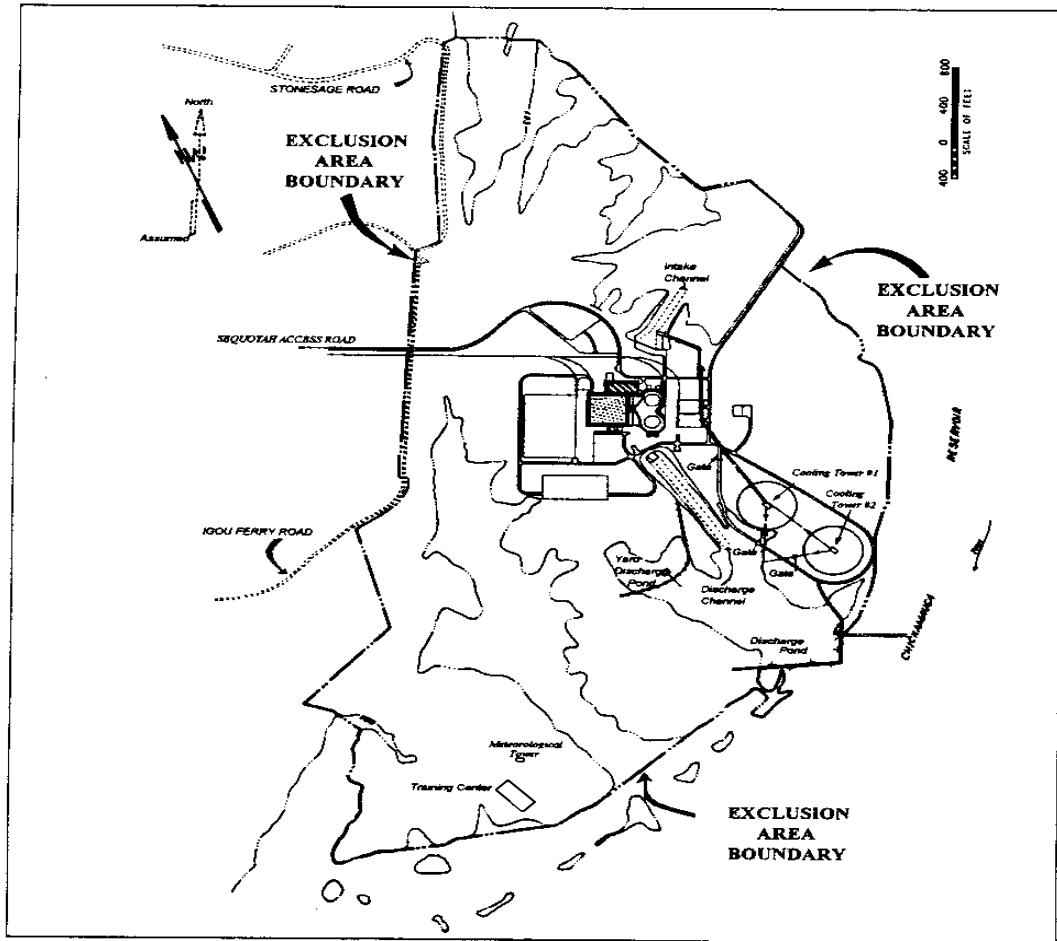
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<i>Event</i>	7.1	GASEOUS EFFLUENTS
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		All
<i>Description</i>		<p>Any UNPLANNED release of Gaseous Radioactivity that exceeds 2 times the ODCM Section 1.2.2.1 limit for >60 minutes (1 or 2 or 3 or 4):</p> <ol style="list-style-type: none"> 1. A VALID rad monitor reading exceeds the values under Unusual Event in Table 7-1 for >60 minutes, unless assessment within that 60 minutes confirms that the criterion is not exceeded. <u>OR</u> 2. Field surveys indicate >0.1 mrem/hr gamma at the exclusion area boundary (Figure 7-A) > 60 minutes. <u>OR</u> 3. Dose assessment results indicate exclusion area boundary (Figure 7-A) dose >0.1 mrem TEDE for the duration of the release. 4. Sample results exceed 2 times the ODCM limit value for an unmonitored release of gaseous radioactivity > 60 minutes in duration. <p>NOTE: TEDE = Total Effective Dose Equivalent</p>
<i>Basis</i>		The ODCM dose factors, meteorology, calculation methodology and the design annual releases (noble gas only) from the SQN FSAR were used in calculation SQN SQS20247, Rev. 000. The results of this calculation are included in Table 7.1.
<i>Escalation</i>		Escalation would be based on increasing the magnitude of the release by a factor of 100.
<i>References</i>		NUMARC/NESP-007, AUI, Rev. 2, 1/92 SQN Calculation SQS20247, Rev. 000 TI-CEM-030.030.0 Manual Calculation of Plant Gas, Iodine, and Particulate Release Rates for ODCM Compliance ODCM - Offsite Dose Calculation Manual

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Figure 7-A

SEQUOYAH EXCLUSION AREA BOUNDARY



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<i>Event</i>	<i>7.2</i>	LIQUID EFFLUENTS
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Not Applicable.
<i>Basis</i>		Not Applicable.
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	<i>7.0</i>	RADIOLOGICAL EFFLUENTS
<i>Event</i>	<i>7.2</i>	LIQUID EFFLUENTS
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Not Applicable.
<i>Basis</i>		Not Applicable.
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

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<i>Section</i>	<i>7.0</i>	RADIOLOGICAL EFFLUENTS
<i>Event</i>	<i>7.2</i>	LIQUID EFFLUENTS
<i>Classification</i>		ALERT
<i>Mode</i>		All
<i>Description</i>		<p>Any UNPLANNED release of liquid radioactivity that exceeds 200 times the ODCM Section 1.2.1.1 limit for >15 minutes (1 or 2):</p> <p>1. A valid rad monitor reading exceeds the values under Alert in Table 7-1 for >15 minutes, unless assessment within this time period confirms that the criterion is not exceeded.</p> <p style="text-align: center;"><u>OR</u></p> <p>2. Sample results indicate an ECL (I-131) exceed 200 times the ODCM limit value for an unmonitored release of liquid radioactivity >15 minutes in duration.</p>
<i>Basis</i>		<p>For liquid releases, the EAL values are determined by multiplying the Unusual Event liquid monitor reading by a factor of 100.</p> <p>Calculation SQS20247 results are included in Table 7.1.</p>
<i>Escalation</i>		Not Applicable.
<i>References</i>		<p>NUMARC/NESP-007, AA2, Rev. 2, 1/92 SQN Calculation SQS20247, Rev. 000 (ODCM) Offsite Dose Calculation Manual 10 CFR 20</p>

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<i>Section</i>	<i>7.0</i>	RADIOLOGICAL EFFLUENTS
<i>Event</i>	<i>7.2</i>	LIQUID EFFLUENTS
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		All
<i>Description</i>		<p>Any UNPLANNED release of liquid radioactivity to the environment that exceeds 2 times the ODCM Section 1.2.1.1 limit for >60 minutes (1 or 2):</p> <p>1. A valid rad monitor reading exceeds the values under UE in Table 7-1 for >60 minutes, unless assessment within this time period confirms that the criterion is not exceeded.</p> <p style="text-align: center;"><u>OR</u></p> <p>2. Sample results indicate an ECL (I-131) exceed 2 times the ODCM limit value for an unmonitored release of liquid radioactivity >60 minutes in duration.</p>
<i>Basis</i>		<p>For liquid releases, the ODCM limit is equal to 10 times the Effluent Concentration Limits (ECL) listed in 10 CFR Part 20 Appendix B, Table 2, Column 2.</p> <p>The calculation methodology is found in SQN Calculation SQS20247, Rev. 000. The results are included in Table 7.1.</p>
<i>Escalation</i>		Escalation will be based on an unplanned release exceeding 200 times the ODCM limit for greater than 15 minutes.
<i>References</i>		<p>NUMARC/NESP-007, AU2, Rev. 2, 1/92</p> <p>SQN Calculation SQS20247, Rev. 000</p> <p>(ODCM) Offsite Dose Calculation Manual</p> <p>10 CFR 20</p>

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**Table 7-1
EFFLUENT RADIATION MONITOR EAL's**

The monitor values below, if met or exceeded, indicate the need to perform the required assessment. If the assessment cannot be completed within 15 minutes (60 minutes for UE), the appropriate emergency classification shall be made based on the **VALID** reading.

	Units	NOUE	Alert	Site Area Emergency	General Emergency
<i>Total Site (Gas)</i>	$\mu\text{Ci/s}$	4.90E+05	4.90E+07	1.34E+08	1.34E+09
<i>Shield Building</i>					
1-RI-90-400 (EFF Level)	$\mu\text{Ci/s}$	4.90E+05	4.90E+07	1.34E+08	1.34E+09
2-RI-90-400 (EFF Level)	$\mu\text{Ci/s}$	4.90E+05	4.90E+07	1.34E+08	1.34E+09
<i>Auxiliary Building</i>					
0-RM-90-101B	cpm	1.03E+05	Offscale ⁽¹⁾	Offscale ⁽¹⁾	Offscale ⁽¹⁾
<i>Service Building</i>					
0-RM-90-132B	cpm	2.62E+06	Offscale ⁽¹⁾	Offscale ⁽¹⁾	Offscale ⁽¹⁾
<i>Steam Generator Discharge⁽²⁾ (Main Steam Line Monitors)</i>					
1-RI-90-421 thru -424	$\mu\text{Ci/cc}$	1.71E-01	1.71E+01	4.68E+01	4.68E+02
2-RI-90-421 thru -424	$\mu\text{Ci/cc}$	1.71E-01	1.71E+01	4.68E+01	4.68E+02
<i>Condenser Vacuum Exhaust</i>					
1-RM-90-255, -256A	mR/h	4.10E+02	4.10E+04	1.12E+05	1.12E+06
2-RM-90-255, -256A	mR/h	4.10E+02	4.10E+04	1.12E+05	1.12E+06
<i>Total Site (Liquid)</i>	$\mu\text{Ci/ml}$	8.20E-03	8.20E-01	N/A	N/A
<i>Radwaste Monitor</i>					
0-RM-90-122	cpm	1.74E+06	Offscale ⁽¹⁾	N/A	N/A
<i>SGBD</i>					
1-RM-90-120,121	cpm	1.27E+06	Offscale ⁽¹⁾	N/A	N/A
<i>SGBD</i>					
2-RM-90-120,121	cpm	1.27E+06	Offscale ⁽¹⁾	N/A	N/A
<i>Condensate Demin</i>					
0-RM-90-225	cpm	1.65E+06	Offscale ⁽¹⁾	N/A	N/A
<i>Turbine Bldg Sump</i>					
0-RM-90-212	cpm	2.92E+03	2.92E+05	N/A	N/A
Release Duration	Minutes	60	15	15	15

(1) The calculated value is outside of the upper range for this detector. The maximum output which can be read is 1E+07 cpm.

(2) These unit values are based on flow rates through one PORV of 890,000 lb/hr at 1078.7 psia with 0.25% carry over (0.9975 quality). Before using these values, ensure a release to the environment is ongoing, (e.g., PORV).

NOTE 1: These EALs are based on the assumption that an emergency release is restricted to one pathway from the plant. In all cases, the total site EAL is the limiting value. Therefore, in the case where there are multiple release paths from the plant, it is the total release EAL (obtained from ICS and/or SQN plant approved procedures) that will determine whether an emergency classification is warranted.

NOTE 2: In the case when there is no CECC dose assessment available, the length and relative magnitude of the release is the key in determining the classification. For example, in the case of the NOUE EAL of 2 times the Tech Spec limit, the classification is based more on the fact that a release above the limit has continued unabated for more than 60 minutes, than on the projected offsite dose.

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<i>Section</i>	<i>7.0</i>	RADIOLOGICAL EFFLUENTS
<i>Event</i>	<i>7.3</i>	RADIATION LEVELS
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" (Section 1) or "Gaseous Effluents"(Section 7.1).
<i>Basis</i>		Not Applicable.
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	<i>7.0</i>	RADIOLOGICAL EFFLUENTS
<i>Event</i>	<i>7.3</i>	RADIATION LEVELS
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" (Section 1) or "Gaseous Effluents"(Section 7.1).
<i>Basis</i>		Not Applicable.
<i>Escalation</i>		Escalation may be based on "Fission Product Barrier Matrix" (Section 1) or "Gaseous Effluent Levels" (Section 7).
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

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<i>Section</i>	7.0	RADIOLOGICAL EFFLUENTS
<i>Event</i>	7.3	RADIATION LEVELS
<i>Classification</i>		ALERT
<i>Mode</i>		All
<i>Description</i>		<p>UNPLANNED increases in radiation levels within the facility that impedes safe operations or establishment or maintenance of cold shutdown (1 or 2):</p> <ol style="list-style-type: none"> Valid area radiation monitor readings or survey results exceed 15 mrem/hr in the control room or CAS. <p style="text-align: center;"><u>OR</u></p> <ol style="list-style-type: none"> (a and b) <ol style="list-style-type: none"> VALID area radiation monitor readings exceed values listed in Table 7-2. Access restrictions impede operation of systems necessary for safe operation or the ability to establish cold shutdown. <p>NOTE: The SED must determine the cause of the increase in radiation levels and review other initiating conditions for applicability (e.g., dose rates of 15 mrem/hr in the control room could be caused by a release associated with a DBA).</p>
<i>Basis</i>		<p>This IC 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.</p> <p>EAL #1 applies to areas that are manned continuously. The value of 15 mrem/hr has been determined to be representative of the criterion. This value was obtained from section III.D.3 of NUREG-0737, "Clarification of TMI Action Plan Requirements", which specified a criterion of 15 mrem/hr averaged over the assumed 30 day duration of the accident. The value was based on the GDC 19 criterion of 5 rem for the duration of the accident, with adjustment for occupancy factors. The value is used here without averaging, as a 30 day duration implies an event potentially more significant than an Alert.</p> <p>The control room, and the central alarm station (CAS) should be continuously manned. Thus, the 15 mrem/hr value applies to these facilities.</p> <p>EAL #2 applies to areas that require infrequent access. Table 7-2 tabulates the areas identified for SQN and the associated radiation level above which access is considered impeded. The areas were selected on the basis of the relative need for access. The specified radiation levels are such that normal radiation exposure control measures intended to maintain doses within normal 10 CFR 20 occupational exposure guidelines would impede necessary access.</p>

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<i>Section</i>	<i>7.0</i>	RADIOLOGICAL EFFLUENTS
<i>Event</i>	<i>7.3</i>	RADIATION LEVELS
<i>Classification</i>		ALERT (continued)
<i>Mode</i>		All
<i>Basis</i> (continued)		<p>This IC is not meant to apply to increases in the containment dome radiation monitors as these are events which are addressed in the fission product barrier matrix ICs. 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.).</p> <p>An indication or report or condition is considered to be VALID when it is conclusively verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel. Implicit in this definition is the need for timely assessment within 15 minutes.</p>
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Matrix" (Section 1) or "Gaseous Effluent" (Section 7.1).
<i>References</i>		NUMARC/NESP-007, AA3, Rev. 2, 1/92 (ODCM) Offsite Dose Calculation Manual

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<i>Section</i>	<i>7.0</i>	RADIOLOGICAL EFFLUENTS
<i>Event</i>	<i>7.3</i>	RADIATION LEVELS
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		All
<i>Description</i>		<p>UNPLANNED increases in radiation levels within the facility:</p> <p>1. Valid area radiation monitor readings increase by a factor of 1000 mrem/hr over the highest reading in the past 24 hours excluding the current peak values.</p> <p>Note: The SED must determine the cause of increase in radiation levels and review other initiating conditions for applicability (e.g., a dose rate of 15 mrem/hr in the control room could be caused by a release associated with a DBA).</p>
<i>Basis</i>		<p>This IC addresses unplanned increases of inplant 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.</p> <p>An indication or report or condition is considered to be VALID when it is conclusively verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel. Implicit in this definition is the need for timely assessment within 15 minutes.</p>
<i>Escalation</i>		Escalation will be based on the inability to access certain operating stations or equipment needed to establish or maintain cold shutdown.
<i>References</i>		NUMARC/NESP-007, AU2, Rev. 2, 1/92

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<i>Section</i>	<i>7.0</i>	RADIOLOGICAL EFFLUENTS
<i>Event</i>	<i>7.4</i>	FUEL HANDLING
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to "Gaseous Effluents" (Section 7.1).
<i>Basis</i>		The basis for a General Emergency is primarily the extent and severity of "Gaseous Effluents" (Section 7.1).
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	<i>7.0</i>	RADIOLOGICAL EFFLUENTS
<i>Event</i>	<i>7.4</i>	FUEL HANDLING
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		"Refer to Gaseous" Effluents (Section 7.1).
<i>Basis</i>		The basis for a Site Area Emergency is primarily the extent and severity of "Gaseous Effluents" (Section 7.1).
<i>Escalation</i>		Not Applicable.
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

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<i>Section</i>	7.0	RADIOLOGICAL EFFLUENTS
<i>Event</i>	7.4	FUEL HANDLING
<i>Classification</i>		ALERT
<i>Mode</i>		All
<i>Description</i>		<p>Major damage to irradiated fuel or loss of water level that has or will uncover irradiated fuel outside the reactor vessel (1 and 2):</p> <ol style="list-style-type: none"> 1. Valid alarm on 0-RM-90-101B or 0-RM-90-102 or 0-RM-90-103 or 1-,2-RM-90-130/131 or 1-,2-RM-90-112. 2. (a or b) <ol style="list-style-type: none"> a. Plant personnel report damage to irradiated fuel sufficient to rupture fuel rods. <p style="text-align: center;"><u>OR</u></p> <ol style="list-style-type: none"> b. Plant personnel report water level drop has or will exceed makeup capacity such that irradiated fuel will be uncovered in the spent fuel pool or fuel transfer canal.
<i>Basis</i>		<p>The major concern of the EAL is a fuel handling accident or loss of water covering spent fuel. Events of this type could cause an increase in radioactivity readings and potentially a release to the environment. Offsite doses during these accidents would be below the EPA Protective Action Guidelines and the classification of an Alert is therefore appropriate.</p> <p>Monitoring radiation on the refueling floor and containment is by particulate, iodine, gas monitors and area monitors. Values for these monitors are set to not exceed safety limits and to ensure that the design basis does not exceed limits referenced in 10 CFR 20.</p> <p>An indication or report or condition is considered to be VALID when it is conclusively verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel. Implicit in this definition is the need for timely assessment within 15 minutes.</p>
<i>Escalation</i>		Escalation will occur by offsite dose rates. Refer to "Gaseous Effluents" (Section 7.1).
<i>References</i>		NUMARC/NESP-007, AA2, Rev. 2, 1/92 AOP M.04 Refueling Malfunctions NRC IEN 90-08 Kr-85 Hazards from Decayed Fuel EPA-520/1-75-001 Manual of Protective Action Guides and Protective Actions for Nuclear Incidents, September 1975 FSAR 15.5.6 Environmental Consequences of a Postulated Fuel Handling Accident T.S. 3.9.4 Containment Penetrations T.S. 3.7.12 Auxiliary Building Gas Treatment System

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<i>Section</i>	<i>7.0</i>	RADIOLOGICAL EFFLUENTS
<i>Event</i>	<i>7.5</i>	INDEPENDENT SPENT FUEL STORAGE INSTALLATION
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		Refer to "Gaseous Effluents" (Section 7.1).
<i>Basis</i>		The basis for a General Emergency is primarily the extent and severity of "Gaseous Effluents" (Section 7.1).
<i>Escalation</i>		Not Applicable.
<i>References</i>		NEI-99-01 Rev 4 (1/03)

<i>Section</i>	<i>7.0</i>	RADIOLOGICAL EFFLUENTS
<i>Event</i>	<i>7.5</i>	INDEPENDENT SPENT FUEL STORAGE INSTALLATION
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable.
<i>Description</i>		"Refer to Gaseous" Effluents (Section 7.1).
<i>Basis</i>		The basis for a Site Area Emergency is primarily the extent and severity of "Gaseous Effluents" (Section 7.1).
<i>Escalation</i>		Not Applicable.
<i>References</i>		NEI-99-01 Rev 4 (1/03)

<i>Section</i>	<i>7.0</i>	RADIOLOGICAL EFFLUENTS
<i>Event</i>	<i>7.5</i>	INDEPENDENT SPENT FUEL STORAGE INSTALLATION
<i>Classification</i>		ALERT
<i>Mode</i>		Not Applicable.
<i>Description</i>		"Refer to Gaseous" Effluents (Section 7.1).
<i>Basis</i>		The basis for a Alert is primarily the extent and severity of "Gaseous Effluents" (Section 7.1).
<i>Escalation</i>		Not Applicable.
<i>References</i>		NEI-99-01 Rev 4 (1/03)

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<i>Section</i>	7.0	RADIOLOGICAL EFFLUENTS
<i>Event</i>	7.5	INDEPENDENT SPENT FUEL STORAGE INSTALLATION
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		This event is applicable after establishment of a CONFINEMENT BOUNDARY.
<i>Description</i>		<p>Damage to a loaded cask CONFINEMENT BOUNDARY from: (1 or 2 or 3)</p> <ol style="list-style-type: none"> 1. Natural phenomena (e.g., seismic event, tornado, flood, lightning, snow/ice accumulation, etc.). 2. Accident (e.g: dropped cask, tipped over cask, explosion, missile damage, fire damage, burial under debris, etc). 3. Judgement of the Site Emergency Director that the CONFINEMENT BOUNDARY damage is a degradation in the level of safety of the ISFSI.
<i>Basis</i>		<p>Spent Fuel Storage Cask CONFINEMENT BOUNDARY consists of MPC shell, bottom baseplate, MPC lid (including the vent and drain port cover plates), MPC closure ring, and associated welds. A NOUE in this IC is based on a loaded fuel storage cask CONFINEMENT BOUNDARY being violated leading to the degradation of the fuel during storage or posing an operational safety problem with respect to its removal from storage. Additionally, the ISFSI is evaluated for damage as part of the appropriate site natural disaster response procedures (Fires, Tornado, Flooding, or Earthquake).</p> <p>The results of the ISFSI Safety Analysis Report (SAR) per NUREG 1536, SAR referenced in the cask's Certificate of Compliance, and the related NRC Safety Evaluation Report were used to develop the list of natural phenomena events and accident conditions. These EALs address responses to a dropped cask, a tipped over cask, explosion, missile damage, fire damage or natural phenomena affecting a cask (e.g., seismic event, tornado, flood, etc.).</p> <p>Any condition or event not otherwise explicitly detailed as an EAL, which, in the judgment of the Site Emergency Director, is a degradation in the level of safety of the ISFSI is an NOUE. Emergency Director judgment is to be based on known conditions and the expected response to mitigating activities within a short time period.</p>
<i>Escalation</i>		Escalation will be occur by offsite dose rates. Refer to "Gaseous Effluents" (Section 7.1).
<i>References</i>		NEI-99-01 Rev 4 E-HU1 (1/03), NUREG 1536, Certificate of Compliance, ISFSI SER, AOPs N.01 Fires, N.02 Tornado Watch/Warning, N.03 Flooding, and N.05 Earthquake

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Table 7-2

ALERT - RADIATION LEVELS

For purposes of comparing the meter/monitor reading values to this table, it can be assumed that mR is equivalent to mrem

Monitor Number	Location Area and Elevation	ALERT Reading Value
1.2-RM-90-1	Spent Fuel Pit ARM E1. 734.0	1.5E + 03 mrem/hr
0-RM-90-5	SFP Pumps ARM El. 714.0	1.5E + 03 mrem/hr
1.2-RM-90-6	CCS HXS ARM El. 714.0	1.5E + 03 mrem/hr
1.2-RM-90-7	Sample Rm ARM El. 690.0	1.5E + 03 mrem/hr
1.2-RM-90-8	AFW Pumps El. 690.0	1.5E + 03 mrem/hr
0-RM-90-9	Waste CndsTks ARM El. 669.0	1.5E + 03 mrem/hr
1.2-RM-90-10	CVCS Bd ARM El. 669.0	1.5E + 03 mrem/hr
0-RM-90-11	Cntmt Spray and RHR Pumps Radmon El. 653.0	1.5E + 03 mrem/hr
0-RM-90-102	Spent Fuel Pit Radmon El. 734.0	1.5E + 03 mrem/hr
0-RM-90-103	Spent Fuel Pit Radmon El. 734.0	1.5E + 03 mrem/hr
0-RM-90-230	CNDS Demineralizer ARM El. 685.0	1.5E + 03 mrem/hr
0-RM-90-231	CNDS Demineralizer ARM El. 706.0	1.5E + 03 mrem/hr

NOTE: All of the above monitors have a range of 0.1 to 1E + 4 mrem/hr.

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B.5 SITE EMERGENCY ORGANIZATION

SQN maintains an organization capable of responding to a radiological emergency. The TSC, OSC, and Control Room staffing for response to emergencies is shown on Figure B-1. The minimum on shift emergency response staffing is shown in Figure B-2.

B.5.1 Emergency Response Positions

TSC and OSC emergency response positions are described in SQN EPIP-6, "Activation and Operation of the Technical Support Center" and SQN EPIP-7, "Activation and Operation of the Operations Support Center."

B.5.1.1 Site Vice President

The Site Vice President serves as a corporate interface for the SED, relieving him from duties which could distract from the SED's primary purpose of plant operations and accident mitigation activities. The Site Vice President shall provide assistance in the following areas:

1. Provides TVA policy direction to the Site Emergency Director.
2. Directs the site resources to support the Site Emergency Director in the accident mitigation activities.
3. Provides direct interface on overall site response activities with:
 - a. NRC, FEMA, or other Federal organizations responding to the site.
 - b. CECC Director.
 - c. Onsite media.
4. At his discretion, may provide interface at the appropriate offsite location on the overall site response activities with:
 - a. State and local agencies.
 - b. NRC region/corporate.
 - c. Joint Information Center.
5. Provides support to other emergency operation centers as necessary.

B.5.1.2 Site Emergency Director

1. Directs onsite emergency accident mitigation activities.
2. Consults with CECC Director and Site Vice President on significant events and their related impacts.
3. Initiates onsite protective actions.
4. Coordinates accident mitigation actions with NRC.

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5. Initiates long-term 24-hour accident mitigation operations.
6. Prior to the CECC being staffed, makes recommendations for protective actions (if necessary) to State and local agencies through the Operations Duty Specialist. This responsibility cannot be delegated except to the CECC Director after the CECC is operational.
7. Responsible for determining the emergency classification. This responsibility cannot be delegated.
8. Approves or authorizes emergency doses. This responsibility cannot be delegated.

B.5.1.3 Operations Manager

1. Directs operational activities.
2. Informs Site Emergency Director of plant status and operational problems.
3. Assures the control room is aware of the accident assessment and response.
4. Recommends solutions and mitigating action for operational problems.

B.5.1.4 Technical Assessment Manager

1. Directs onsite effluent assessment.
2. Directs activities of technical assessment team.
3. Projects future plant status based on present plant conditions.
4. Keeps assessment team informed of plant status.
5. Provides information, evaluations, and projects to Site Emergency Director.
6. Coordinates assessment activities with the CECC plant assessment team.
7. Establishes and maintains a status of significant plant problems.

B.5.1.5 OSC Manager

1. Directs repairs and corrective actions in coordination with the TSC.
2. Performs damage assessment.
3. Directs activities of Operations Support Center.
4. Coordinates maintenance teams and ensures they have received proper briefings and are accompanied by a Rad Protection technician, as necessary.

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B.5.1.6 TSC Clerks

1. Answer telephones.
2. Distribute plant parameter data sheets.
3. Maintain TSC organization board.
4. Operate facsimile machine.
5. Other duties as assigned by Site Emergency Director.

B.5.1.7 Nuclear Security Manager

1. Directs activities of Nuclear Security Services personnel.
2. Controls access to site and control rooms.
3. Reports on site accountability/evacuation as defined in SQN-EIPs.

B.5.1.8 Radiological Protection Manager

1. Directs and/or performs assessment of inplant and onsite radiological conditions.
2. Directs onsite Rad Protection activities.
3. Coordinates additional Rad Protection support with CECC Radiological Assessment Manager.
4. Makes recommendations for protective actions for onsite personnel.
5. Maintains status map of offsite radiological conditions.
6. Coordinates assessment of radiological conditions offsite with CECC Radiological Assessment Manager.
7. Maintains inplant radiation status board.
8. Authorizes issue of KI to onsite personnel.
9. Makes recommendations to the Site Emergency Director for personnel entry to radiological hazardous environment.

B.5.1.9 Chemistry Manager

1. Coordinates assessment of radioactive effluents with CECC Rad Assessment Coordinator.
2. Coordinates post-accident sampling activities.
3. Performs release rate calculations.
4. Determines impact of incident on radwaste and various effluent treatment systems.

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B.5.1.10 NRC Coordinator

1. Acts as primary liaison with onsite NRC personnel.
2. Updates NRC personnel on plant status.
3. Provides information requests from NRC to TSC personnel.

B.5.1.11 Control Room Communicator

1. Provides operational knowledge for status evaluation of plant systems.
2. Provides advice regarding technical specifications, system response, safety limits, etc.
3. Assists in development of recommended solutions to developing problems.
4. Serves as the control room - TSC - OSC link.

B.5.1.12 Emergency Preparedness Manager

1. Advises Site Emergency Director regarding overall radiological emergency plan, use of implementing procedures, emergency equipment availability, and coordination with CECC.
2. Confirms TSC is operating properly.

B.5.1.13 Technical Assessment Team

1. Prepares and provides periodic current assessments on plant conditions and provides this information to the CECC plant assessment team.
2. Projects future plant status based on present plant conditions.
3. Provides technical support to plant operations on mitigating actions.

B.5.1.14 OSC Assistant Manager

1. Oversees the operations of OSC teams.
2. Maintain continuous communications with the TSC.
3. Maintains team tracking boards.
4. Assigns TSC tasks to team briefers.

B.5.1.15 OSC Rad Protection Supervisor

1. Directs activities of the Rad Protection lab.
2. Ensure Rad Protection coverage of damage repair teams.
3. Verify habitability of the TSC, OSC, and Control Room.
4. Briefs the OSC Manager and TSC on radiological status.

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B.5.1.16 Briefing Teams

1. Provide mechanical, electrical, and instrumentation technical expertise.
2. Evaluate task conditions and provide methods best suited to safely perform an assignment.
3. Brief OSC teams.
4. Track OSC teams in the field.
5. Debrief OSC teams after task completion.

B.6 EMERGENCY RESPONSE FACILITIES, EQUIPMENT, AND SUPPLIES

Specific plant areas, facilities, and equipment are selected and provided for use during a radiological emergency. The preselection, allocation, and inclusion of emergency facilities assure that needed services and equipment are available for use during emergency conditions.

B.6.1 Technical Support Center (TSC)

A specific area (adjacent to the relay room) in the control building at elevation 732' is designated for use as the TSC. The room is provided with communication capabilities to plant areas and areas external to the plant. The communication facilities include TVA System telephones, NRC Emergency Notification System and Health Physics Network, access to the plant paging system, and a two-way radio for communications with environmental monitoring vans. This room is sufficiently shielded to ensure occupancy during an emergency and is designed to be continuously habitable during all radiological emergencies. All ventilating and air-conditioning facilities have redundant or backup systems. Toilet facilities are available on the same elevation.

The diesel generators will provide emergency power when there is a loss of normal ac power, and cooling water for the air-conditioning equipment is taken from the essential raw cooling water system. Figure B-3 shows a general TSC layout.

Meteorological information is available both in the TSC and in the main control room and includes wind speed and direction and temperature difference between 10-meter, 46-meter, and 91-meter tower elevations.

B.6.2 Operations Support Center (OSC)

The role of the OSC is to provide assembly areas for operations support personnel during an emergency situation which are under the supervision of the OSC Manager or a designated alternate. The OSC is located on elevation 706' in the southwest corner of the cafeteria in the Plant Office Building. It contains emergency team briefing areas and additional space in the adjacent main dining rooms for staging, briefing and dispatching maintenance teams. The Alternate OSC is located in the Office and Power Stores Building first floor. The OSC is provided with telephone and radio communications. Figure B-4 shows the OSC areas. Respiratory protective devices are located in the elevation 690' Rad Protection Lab. Protective clothing, flashlights, other equipment and tools are available, as needed.

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B.6.3 Rad Protection Laboratory and Equipment

The Rad Protection laboratory is located in the service building, elevation 690'. The portable radiation monitoring and counting equipment normally used by the plant Rad Protection section is kept in this space and is available for use during an emergency. Sufficient reserves of instruments/equipment are available to replace those removed from service for calibration or repair. Calibration of equipment is carried out at intervals as specified in Rad Protection procedures.

B.6.4 Alternative Facilities

Following the declaration of Alert or higher, for some conditions, primarily in response to hostile action events when ERO members may not have access to the site, an alternative facility has been identified. This facility serves as a staging area for augmentation staff, which will minimize delays in overall site response and allow for a swift, coordinated augmented response when the site is deemed accessible.

The alternative facility is accessible even if the site is under threat of or experiencing hostile action, to function as a staging area for augmentation of emergency response staff and have the capability for communication with the emergency operations facility, control room, and plant security, for use when onsite emergency facilities cannot be safely accessed during hostile action.

B.6.5 Onsite Monitoring Systems and Equipment

B.6.5.1 Natural Phenomena

In the event an emergency is the result of a natural phenomena, there is instrumentation to monitor its severity. The Environmental Data Station is located onsite and contains instruments capable of measuring wind direction, wind speed, and temperatures. Seismic instrumentation is available in the plant to monitor acceleration levels of ground movement. Hydrological monitoring systems are installed to supply flow and level information for each site. Meteorological and seismic instrumentation have readily accessible readout in the main control room. More specific information on these systems can be found in the Sequoyah FSAR.

B.6.5.2 Radiological Monitors

The installed Radiation Monitoring System consists of process monitors and area monitors which read out on local panels and in the control room.

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B.6.5.2.1 Process Monitors (Radiological)

The process system continuously monitors selected lines containing or possibly containing radioactive effluents. The system's function is to warn personnel of increasing radiation levels, to give early warning of a system malfunction, and to record and control discharges of radioactive liquids and gases to the environment. The system consists of active and redundant channels.

Examples of process monitors are:

1. Ventilation Gas and Particulate
2. Process Gas and Particulate
3. Containment Gas and Particulate
4. Condenser Vacuum Exhaust
5. Steam Generator Blowdown
6. Liquid Waste
7. Service Water
8. Component Cooling Water
9. Component Cooling Water Heat Exchangers
10. Reactor Coolant System

B.6.5.2.2 Area Radiation Monitors

Area monitors are placed at specific locations in the plant. Examples of area monitor locations are:

1. Containment
2. New and Spent Fuel Storage Area
3. Main Control Room
4. Incore Instrument Room

B.6.5.2.3 Portable Monitors

Portable radiation detection equipment consists of low-range and high-range instruments to measure gamma radiation levels from 0.1mR/hr to 1000 mR/hr. Instruments for alpha, beta-gamma, and neutron radiation measurements are available. Sampling equipment is available to take low- or high-volume air samples. Air samplers can be used to collect low-volume samples either onsite or offsite. The counting room has appropriate equipment for isotopic analysis.

B.6.5.2.4 Process Monitors (Nonradiological)

Installed in the main control room are the necessary instrumentation readouts to assess plant systems status, including reactor coolant system pressure and temperature, containment pressure and temperature, liquid levels, flow rates, fire detection equipment, and meteorological instrumentation. More specific information on control room instrumentation can be found in the Sequoyah FSAR.

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B.6.5.3 Fire Protection

The plant's fire protection system is designed to furnish water and other extinguishing agents with the capability of extinguishing any single or probable combination of simultaneous fires that might occur. The use of combustible materials is minimized, and the greatest possible use of fire-retardant materials has been incorporated in plant design.

The standards of the National Fire Protection Association and the recommendations of the nuclear insurers are considered in the system design to provide the following:

1. Supply of water for the fire protection system.
2. Automatic fire or smoke detection in the more critical areas.
3. Fire suppression by fixed equipment actuated automatically or manually.
4. Manually-operated portable fire extinguishing equipment at strategic locations.
5. Compartmentation to limit the spread of fire.

B.6.5.4 Environment

Facilities available for assessing the impact of plant operations on the environment include atmospheric monitoring stations, direct gamma radiation detectors, and automatic water samplers. This equipment is used in the routine environmental radiological monitoring program and is available in the event of a radiological emergency condition.

The atmospheric monitoring network is divided into three subgroups. Local air monitors are located at or adjacent to the Exclusion Area Boundary in the directions of predominant wind flow. Perimeter monitors are located three to ten miles from the plant in areas of relatively high population densities and/or in the direction of predominant air flow. Remote monitors (controls) are located at sites greater than 10 miles from the plant.

At each monitor, air is continuously passed through a particulate filter at a regulated flow. In series with, but downstream of, the particulate filter is a charcoal filter used to collect iodine. Each monitor has a collection tray and storage container to collect rainwater on a continuous basis.

Primary dosimeters are placed at approximately 40 sites around the plant. These Primary dosimeters are located typically in each of the 16 meteorological sectors at or near the Exclusion Area Boundary and at a distance of approximately four to five miles. Three dosimeters are usually placed at each site.

Automatic water samplers are located above and below the plant discharge and at the first potable water supply downstream from the plant.

In addition to these facilities, established sampling points for milk, vegetation, soil, fish, and sediment are located in the vicinity of the plant. Samples may be collected from these stations on a nonroutine basis as needed.

All samples are returned to TVA's radiological laboratory for processing.

B.6.6 Emergency Equipment

Figure B-5 contains listings of emergency equipment and storage locations throughout the plant.

Required calibration of equipment is carried out at intervals recommended by the supplier of the equipment or as specified in the Sequoyah FSAR.

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B.6.7 First Aid and Medical Facilities

B.6.7.1 Decontamination Facilities

The site is responsible for maintaining supplies and equipment to establish a temporary decontamination area for the purpose of gross radiological decontamination and injured person evaluation and stabilization. This area, complete with supplies, is located in the service building, elevation 690' adjacent to the Rad Protection lab. Equipment and materials for decontamination and first aid are available.

B.6.7.2 First Aid Stations and Supplies

Emergency medical equipment is strategically located throughout the plant, with trauma kits and other specified equipment available for use by the MERT.

First aid is provided by EMTs. Medical supplies and treatment for minor injuries are available. A minimum of one ambulance is also available. First aid treatment is available 24 hours a day.

A medical office, staffed by registered nurses, is located on site. Medical treatment and examinations (employment, routine, occupational) are available during the day shift, Monday-Friday.

Potassium Iodide tablets for onsite personnel are controlled and stored by site Rad Protection. Specific information including authorization and dispersal of tablets is contained in the site EIPs.

B.6.7.3 Receiving Hospitals and Supplies

Arrangements have been made at least two hospitals to receive patients from SQN. (See REP Sections 12.3 and 16.5)

B.6.7.4 Ambulance Service

A TVA ambulance is available at the site and is maintained and staffed in conjunction with the MERT. Arrangements have been made for offsite ambulance assistance to SQN. (See REP Sections 12.2 and 16.5)

B.6.8 Additional Local Support

B.6.8.1 Fire

Arrangements have been made for local fire support upon request. The senior fireman responding will work with and for the TVA Fire Brigade Leader directing the activities of the firemen. Sequoyah will be responsible for providing radiological protection and proper safety clearance in all fire areas. (See REP Section 16.5)

B.6.8.2 Law Enforcement

Agreements are maintained with local law enforcement agencies to support TVA when necessary. (See REP Section 16.5)

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B.6.9 Vendor Support

If necessary, the NSSS vendor, Westinghouse, will be contacted by the TSC to provide assistance in the form of manpower, equipment, and technical backup. Other vendors will also be contacted if their assistance is needed.

B.6.10 Assembly/Accountability Alarm

Undulating sirens are provided in strategic areas for indicating the assembly of plant personnel. A three-minute undulating siren is the signal for assembly.

The sirens are powered by redundant 120V ac supplies. The sirens are activated in the main control room or the auxiliary control room diesel panel.

B.6.11 Local Recovery Center (LRC)

The LRC is a designated space located in the Sequoyah Training Center (STC) outside the protected area of the site approximately 0.75 miles from the plant. Portions of the training center offices, classrooms, etc., (about 88,000 square feet total) would be made available as a nearsite work area for TVA as well as NRC and other response personnel necessary to carry out required recovery efforts.

The LRC has telephone communications capabilities to enable personnel to communicate with the CECC and the Sequoyah TSC.

Meteorological information and dose rate calculations are also available to LRC personnel.

Other equipment in the STC available for use by LRC personnel include:

1. Facsimile machine
2. Copy machine
3. Hand-held calculators
4. Plant-specific drawings, manuals, procedures, etc. (drawings located in nearby SQN Operations Training Area)

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B.7 SQN EMERGENCY PLAN IMPLEMENTING PROCEDURES

The following is a listing of the SQN-EIPs:

B.7.1 SQN EPIP-1 Emergency Plan Classification Matrix

This procedure provides guidance to the Shift Manager (SM)/Site Emergency Director (SED) or TSC SED in determining the classification of an accident to ensure that appropriate predetermined actions are implemented. It details initiating conditions and directs shift personnel to appropriate notification and assessment procedures.

B.7.2 SQN EPIP-2 Notification of Unusual Event

This procedure provides for the timely notification of appropriate individuals when the SM/SED has determined by SQN EPIP-1 that an incident has occurred which is classified as a Notification of Unusual Event. It details requirements for periodic reassessment and the implementation of appropriate actions.

B.7.3 SQN EPIP-3 Alert

This procedure provides for the timely notification of appropriate individuals when the SM/SED has determined by SQN EPIP-1 that an incident has occurred which is classified as an Alert. It details requirements for periodic reassessment and the implementation of appropriate actions.

B.7.4 SQN EPIP-4 Site Area Emergency

This procedure provides for the timely notification of appropriate individuals when the SM/SED has determined by SQN EPIP-1 that an incident has occurred which is classified as a Site Area Emergency. It details requirements for periodic reassessment and the implementation of appropriate actions.

B.7.5 SQN EPIP-5 General Emergency

This procedure provides for the timely notification of appropriate individuals when the SM/SED has determined by SQN EPIP-1 that an incident has occurred which is classified as a General Emergency. It details requirements for periodic reassessment and the implementation of appropriate actions. It also provides for determination of an initial protective action recommendation to State and local agencies.

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B.7.6 SQL EPIP-6 Activation and Operation of the TSC

This procedure directs the activation and operation of the TSC during an Alert, Site Area Emergency, or General Emergency or at the discretion of the SED. It details notification requirements and responsibility for supervision of the TSC.

B.7.7 SQL EPIP-7 Activation and Operation of the OSC

This procedure directs the activation and operation of the OSC during an Alert, Site Area Emergency, or General Emergency or at the discretion of the SED.

B.7.8 SQL EPIP-8 Personnel Accountability and Evacuation

This procedure details the requirements for accountability of all personnel and visitors and the orderly evacuation of areas of the plant during a radiological emergency.

B.7.9 SQL EPIP-9 Accountability and Evacuation of the Sequoyah Training Center

This procedure has been cancelled.

B.7.10 SQL-EPIP-10 Medical Emergency Response

This procedure details actions to be followed during medical emergencies. It provides for the organization and activation of the onsite Medical Emergency Response Team. It contains the duties and responsibilities of the onsite Medical Emergency Response Team. The procedure provides guidance on the care and handling of patients who may have been exposed to or contaminated with radioactive material, including provision for the transport of these individuals to offsite medical support facilities. Maps and appropriate instructions are included.

B.7.11 SQL EPIP-11 Security and Access Control

This procedure details responsibilities and requirements for access control and accountability during a radiological emergency.

B.7.12 SQL EPIP-12 Emergency Equipment and Supplies

This procedure details requirements for periodic inspection and maintenance of emergency equipment and supplies. It assigns responsibility and specifies the inspection frequency and documentation requirements.

B.7.13 SQL EPIP-13 Dose Assessment

This procedure provides a manual method to calculate doses from offsite radiological releases during a radiological emergency.

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B.7.14 SQN EQIP-14 Radiological Protection Response

This procedure outlines the actions to be followed by health physics personnel during a plant emergency. It details responsibilities, Rad Protection assessment actions and record keeping requirements. The procedure provides guidance regarding the administration of potassium iodide (KI).

B.7.15 SQN EPIP-15 Emergency Exposure Guidelines

This procedure provides guidance on acceptable personnel exposures for various conditions. It specifies absolute exposure and authorizes the Site Emergency Director to permit exposures in excess of 10 CFR 20 limits in order to perform an emergency mission.

B.7.16 SQN EPIP-16 Termination and Recovery

This procedure outlines responsibilities and provides guidance on recovery after an emergency to assure adequate planning or efficient utilization of resources and radiation exposure.

B.7.17 SQN EPIP-17 Fire Emergency Procedure

This procedure provides guidance for the management of the response to fire emergencies.

B.7.18 EPIP-18

This procedure has not been issued.

B.7.19 EPIP-19

This procedure has been cancelled.

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B.8 ALERT NOTIFICATION SYSTEM

See Generic REP

B.9 TRAINING AND DRILLS

B.9.1 Training Personnel

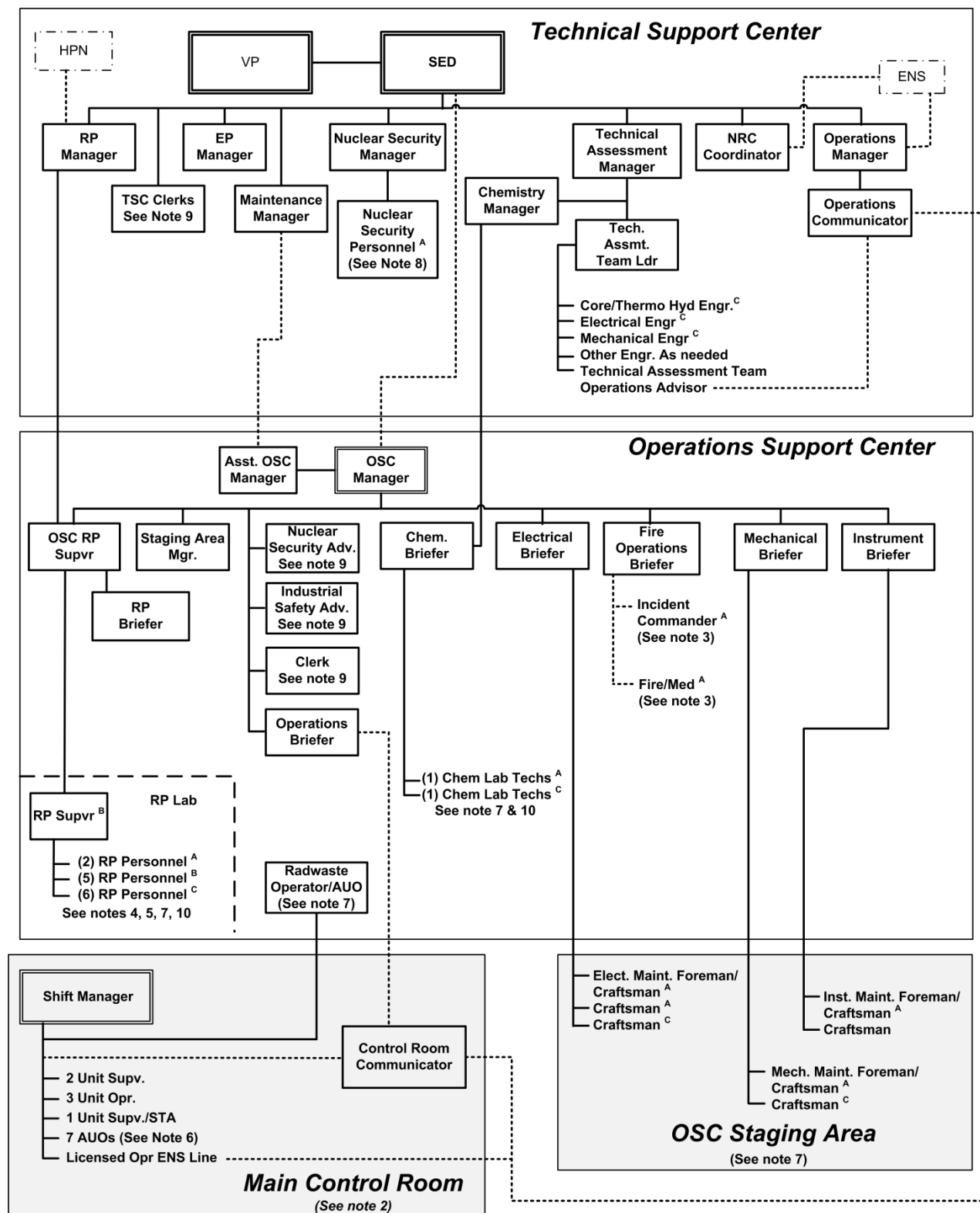
Personnel with specific duties and responsibilities in the SQN REP program receive instruction in the performance of their duties and responsibilities per the Nuclear Power Training Manual, Section TRN-30 (Radiological Emergency Preparedness Training), and as required in REP Section 15.0, (Training).

B.9.2 Drills and Exercises

Drills and exercises are conducted regularly to develop and maintain the key skills that are required for emergency response. The drills identified in REP Section 14.0 (Drills and Exercises) may be conducted individually or as part of a REP exercise.

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Figure B-1
SITE EMERGENCY ORGANIZATION
(Including Minimum Staffing and Augmentation) (See Note 1)



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Figure B-1 (Continued)

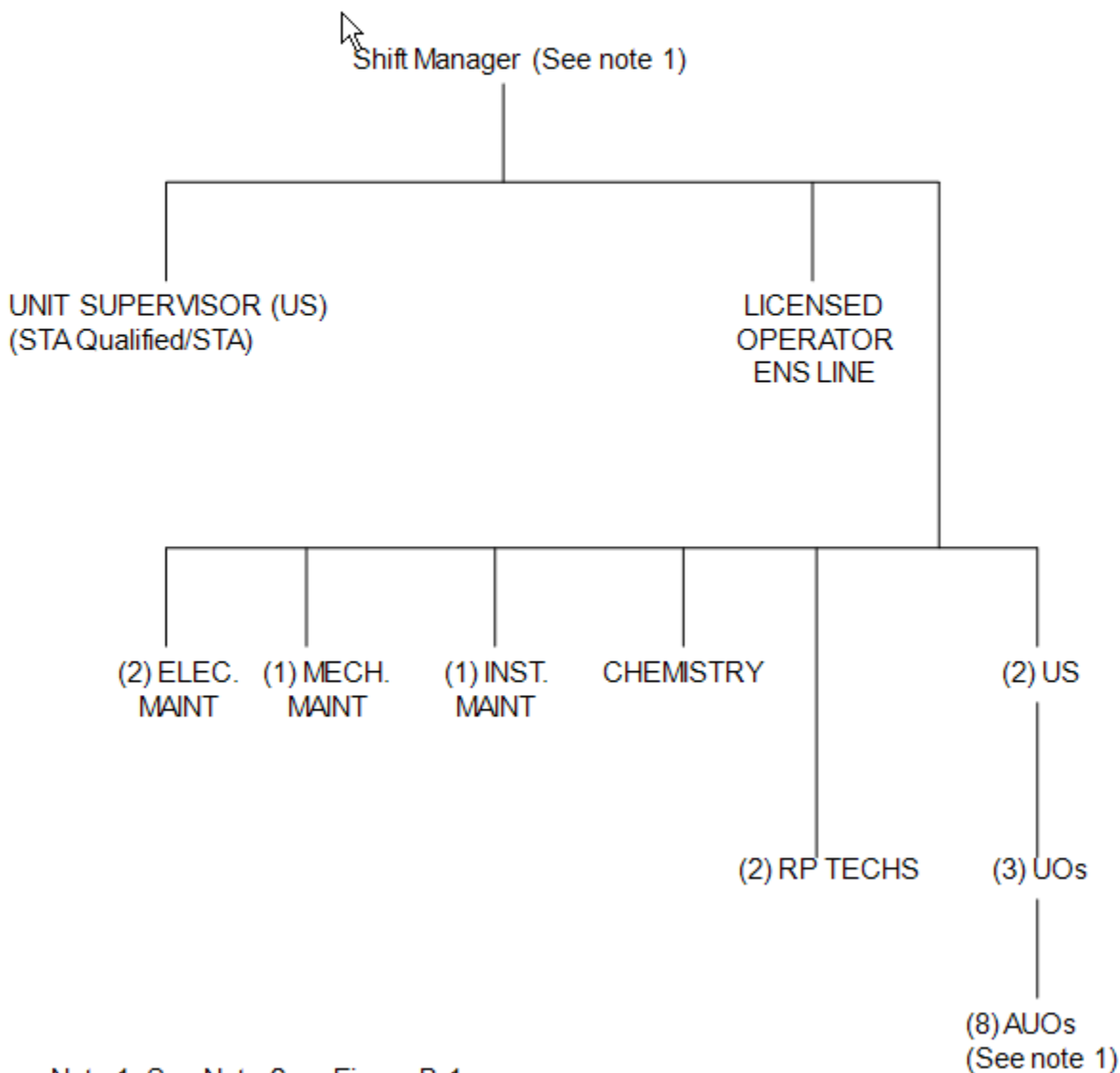
Notes

- Note 1- ERO members shown report to assigned facilities within approximately 60 minutes except as provided in these notes and the following which is used for clarification in NUREG 0654 Table B-1;
- A. on-shift
 - B. 30 minutes
 - C. 60 minutes
- Note 2- Main Control Room on-shift staffing (assuming both units in mode 4 or above) except the Control Room Communicator who is a 60 minute ERO member of the TSC and as provided in the following:
- As allowed by 10CFR50.54(m) Table (2)(i), 2 SROs and 3 Licensed Operators are required as a minimum for two unit plant with a common control room. Temporary deviations shall be in accordance with Tech Specs.
- Additionally, Table B-1 requires that each unaffected nuclear unit in operation maintain at least 1 Unit Supervisor, Unit Operator, and 1 AUO except units sharing a control room may share a Unit Supervisor if all functions are covered. 1 Licensed Operator has been added to the above requirements in order to address communications.
- Note 3- Fire Brigade personnel on-shift will be in accordance with the Fire Protection Report. This group also provides medical and rescue functions.
- Note 4- Offsite Dose Assessment task as shown in NUREG 0654 Table B-1 is provided by a qualified Radiation Protection individual.
- Note 5- Two Radiation Protection Technicians are required on-shift per NUREG 0654 Table B-1. One is required for in-plant surveys, the other may be provided by shift personnel assigned other functions. Six additional techs are required in 30 minutes and six more in 60 minutes. The Radiation Protection Supervisor may fill one of the six 30 minute responder positions.
- Note 6- Depicts reporting to the Shift Manager of non-control room on-shift personnel prior to staffing of the OSC at which time they report to the Staging Area.
- Note 7- Personnel at 30 minutes and/or 60 minutes may be on shift.
- Note 8- On-shift security personnel per the Security Plan.
- Note 9- Call in as needed
- Note 10- Chemistry Supervisor/Radiation Protection Supervisor, if holding proper qualifications, may fill the position of technician.

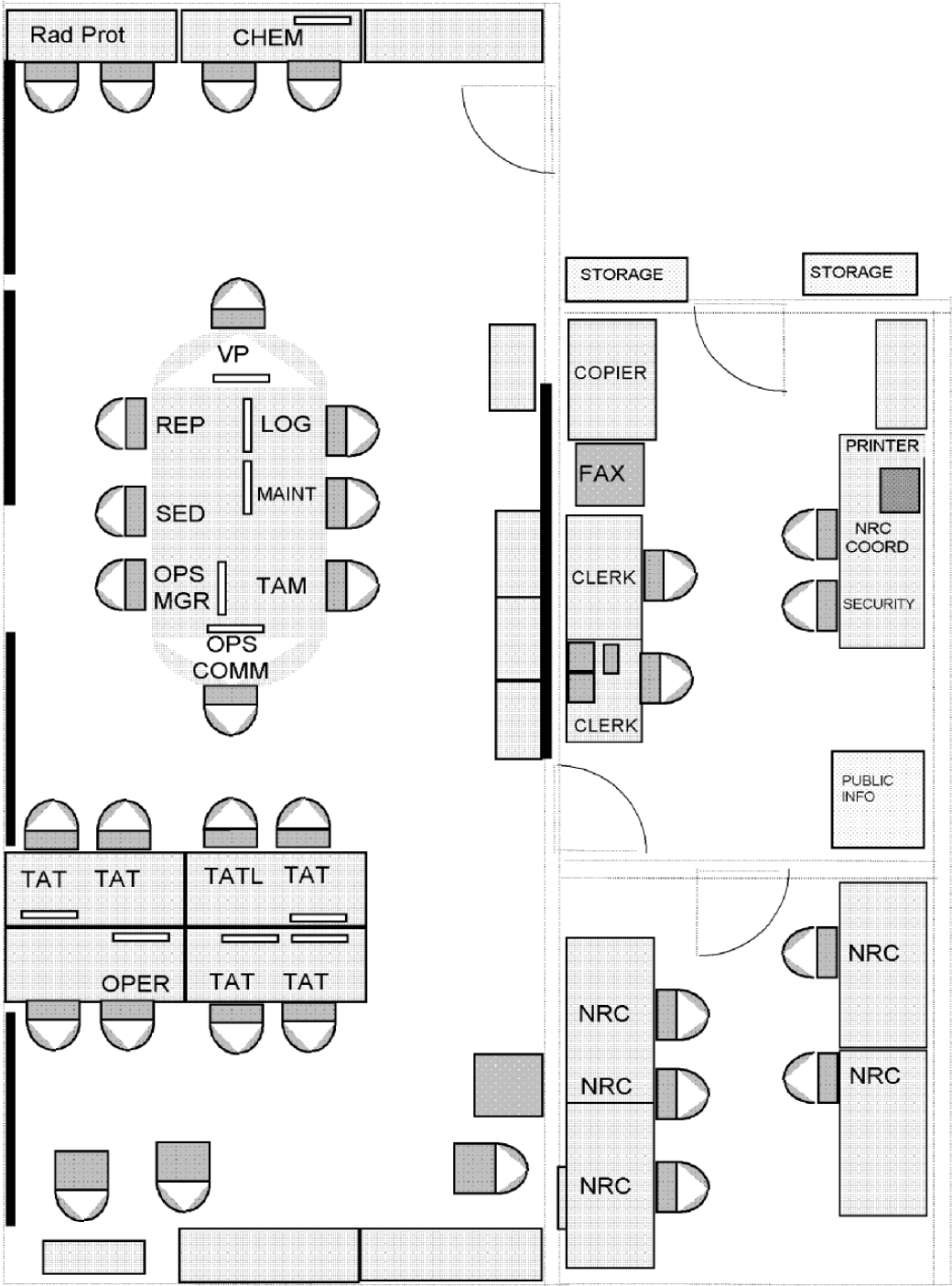
..... **Shows communication networks.**

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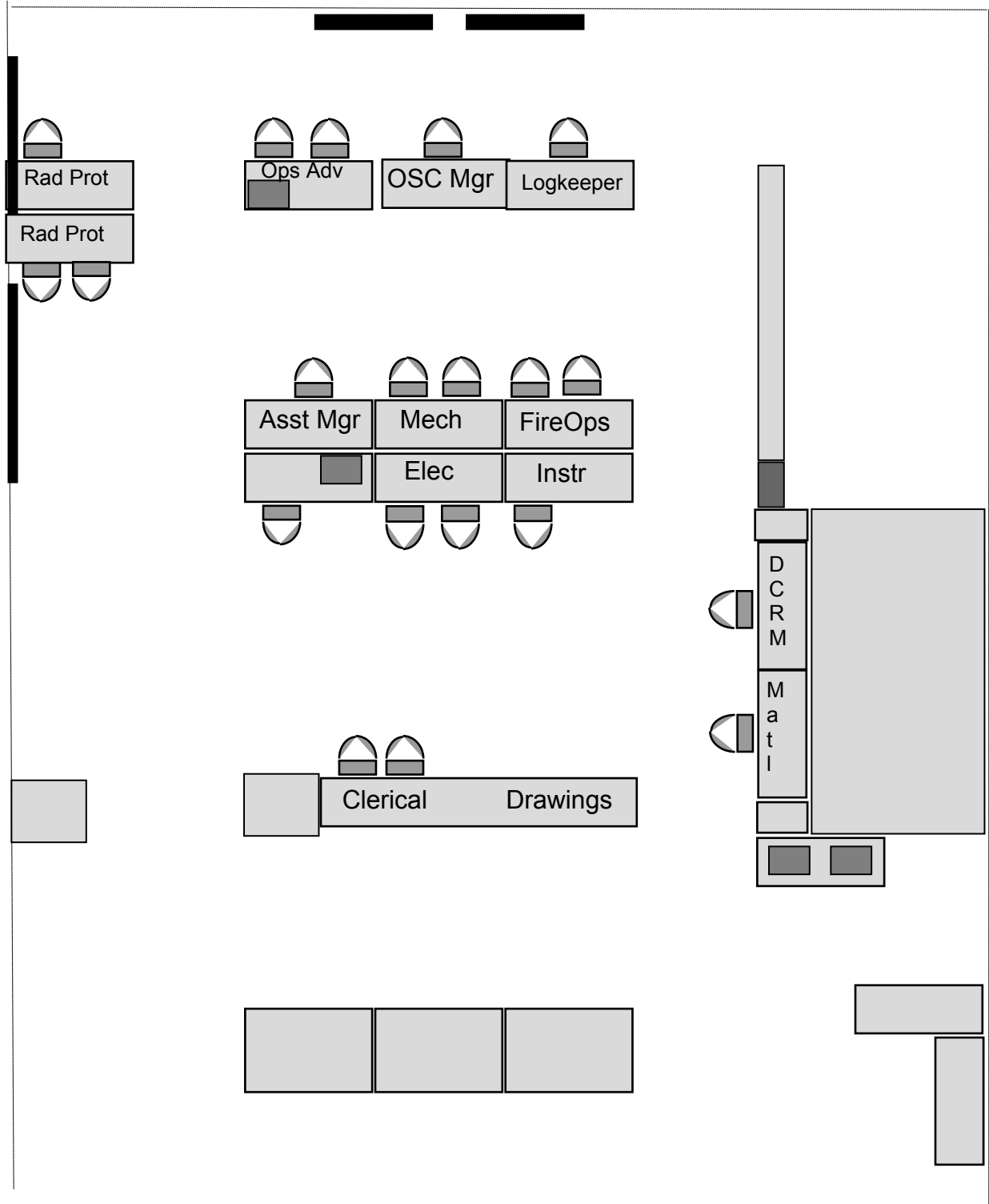
FIGURE B-2
MINIMUM ONSHIFT RESPONSE PERSONNEL



**FIGURE B-3
TECHNICAL SUPPORT CENTER
Control Building Elevation 732**



**FIGURE B-4
OPERATIONS SUPPORT CENTER
Plant Office Building Elevation 706**



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Figure B-5

EMERGENCY EQUIPMENT

<u>Location</u>	<u>Description</u>
1. Rad Protection Lab (Service Bldg., Elv. 690')	Radiological Survey Meters
2. Medical Emergency Supplies	Misc. use medical supplies
3. Various Locations	Emergency SCBA's
4. Decon Facility (Service Bldg. Elv. 690')	Decon supplies
5. Emergency Van (Rad Protection, Environs Monitoring)	Misc. emergency supplies specific to environs monitoring
6. Agreement Hospital	Misc. specific to Emergency Rooms
7. 480V Rx Mov Bd Room	Misc. emergency supplies
8. Communications Room	Misc. emergency supplies
9. Main Control Room Corridor (Elv. 732')	Misc. emergency supplies
10. Technical Support Center	Misc. emergency supplies
11. Operations Support Center	Misc. emergency supplies

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Enclosure 5

Radiological Emergency Plan (REP)

Appendix C: Watts Bar Nuclear Plant

Revision 103

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APPENDIX C

WATTS BAR NUCLEAR PLANT

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C.1 Introduction

The following information provides a site specific list of Initiating Conditions (IC), site specific instrument parameters (when required) and a basis for classifying and declaring Emergency Events at the Watts Bar Nuclear Plant (WBN).

Guidance for determining these Emergency Events was taken from REG GUIDE 1.101, Emergency Planning and Preparedness for Nuclear Power Reactors which allows Licensees to use NUMARC/NESP-007, Rev. 2, 1/92, Methodology for Development of Emergency Action Levels.

For the purposes of declaring an emergency WBN utilized the following Emergency classifications: General Emergency, Site Area Emergency, Alert, and Unusual Event.

For a General Emergency to be declared, events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels outside the EXCLUSION AREA BOUNDARY.

For a Site Area Emergency to be declared, events are in process or have occurred which involve an actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; (1) toward site personnel or equipment that could lead to the likely failure of or; (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels outside the EXCLUSION AREA BOUNDARY.

For an Alert to be declared, events are in process or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

For an Unusual Event to be declared, events are in process or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

The goal of these Emergency classification levels is to have offsite emergency response authorities prepared to take actions to protect the health and safety of the public in the event of a radiological emergency.

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C.2 Emergency Event Methodology

The WBN methodology for event classification and declaration has 36 emergency events broken down into the following seven categories.

FISSION PRODUCT BARRIER MATRIX (Modes 1 - 4)

WBN Reference	NUMARC/NESP-007 Reference
1.1 Fuel Clad	FC 1,2,3,4,5,7
1.2 RCS	RCS 1,2,3,5,6
1.3 Containment	CNTMT 1,2,3,4,5,8

SYSTEM DEGRADATION

WBN Reference	NUMARC/NESP-007 Reference
2.1 Loss of Instrumentation	SU3, SA4, SS6
2.2 Loss of Function/Communication	SU6, SA3 (expanded), SS4
2.3 Failure of Reactor Protection	SA2, SS2, SG2
2.4 Fuel Clad Degradation	SU4
2.5 RCS Unidentified Leakage	SU5
2.6 RCS Identified Leakage	SU5
2.7 Uncontrolled Cool Down	HU5
2.8 Turbine Failure	HU1, HA1
2.9 Technical Specification	SU2
2.10 Safety Limit	SU2

LOSS OF POWER

WBN Reference	NUMARC/NESP-007 Reference
3.1 Loss of AC (Power Ops)	SU1, SA5, SS1, SG1
3.2 Loss of AC (Shutdown)	SU1, SA1
3.3 Loss of DC	SU7, SS3

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HAZARDS and SED JUDGEMENT

WBN Reference	NUMARC/NESP-007 Reference
4.1 Fire	HU2, HA2
4.2 Explosion	HU1, HA1
4.3 Flammable Gas	HU3, HA3
4.4 Toxic Gas	HU3, HA3
4.5 Control Room Evacuation	HA5, HS-2
4.6 Security	HU4, HA4, HS4, HG1 (NEI 99-01 R5)
4.7 SED Judgment	HU5, HA6, HS3, HG2

DESTRUCTIVE PHENOMENON

WBN Reference	NUMARC/NESP-007 Reference
5.1 Earthquake	HU1, HA1
5.2 Tornado	HU1, HA1
5.3 Aircraft Crash	HU1, HA1
5.4 River Level High	HU1, HA1
5.5 River Level Low	HU1, HA1
5.6 Watercraft Crash	HU1

SHUTDOWN SYSTEM DEGRADATION

WBN Reference	NUMARC/NESP-007 Reference
6.1 Loss of Shutdown Systems	SA3, SS5 (expanded)
6.2 Loss of AC (Shutdown)	SU1, SA1
6.3 Loss of DC (Shutdown)	SU7

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RADIOLOGICAL

WBN Reference	NUMARC/NESP-007 Reference
7.1 Gaseous Effluent	AU1, AA1, AS1, AG1
7.2 Liquid Effluent	AU1, AA1
7.3 Radiation Levels	AU2, AA3
7.4 Fuel Handling	AU2, AA2

In each event there exists a set of Initiating Conditions and associated emergency action levels (where required) which trigger the declaration of the emergency and the level of onsite and offsite emergency response.

In the WBN Methodology, the following operating modes were utilized in the declaratory scheme:

- Power operations (1)
- Start up (2)
- Hot Standby (3)
- Hot Shutdown (4)
- Cold Shutdown (5)
- Refueling (6)
- Defueled

C.3 Responsibility

The responsibility of declaring an Emergency based on the guidance provided in this section belongs to the Shift Manager/Site Emergency Director (SM/SED) or designated Unit Supervisor (US) when acting as the SM or the Site Emergency Director (SED). These duties can not be delegated.

C.4 Classification Determination

To determine the classification of the emergency, the SED reviews the Initiating Conditions of the Events described in WBN Emergency Plan Implementing Procedure (EPIP 1) with the known or suspected conditions.

If a Critical Safety Function (CSF) is listed as an Initiating Condition, the respective status tree criteria will be monitored and used to determine the EVENT classification for the Modes listed on the classification flowchart in the procedure.

The highest classification for which an emergency action level (EAL) currently exists shall be declared.

After an event classification, if the followup investigation shows that initiating conditions were met that dictate a higher event classification, the new event classification shall be declared at the clock time of the determination.

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C.4 Classification Determination (continued)

If an EAL for a higher classification was exceeded but the present situation indicates a lower classification, the fact that the higher classification occurred shall be reported to the NRC and Central Emergency Control Center (CECC), but should not be declared.

If the parameter is indeterminate due to instrument malfunction and the existence of the condition can not be reasonably discounted (i.e., spurious or false alarm that can be substantiated within 15 minutes) the condition is considered met and the SM/SED shall follow the indications provided until such time as the alarm is verified to be false.

For monitors that read out in mr/hr, it is assumed that this is equivalent to mrem/hr. For monitors that read out in R/hr, it is assumed that this is equivalent to rem/hr.

If an EAL was exceeded but the emergency has been totally resolved (prior to declaration), the emergency class that was appropriate shall Not be declared but reported to the NRC and Operations Duty Specialist (ODS) at the same clock time.

References

10 CFR 50 Domestic Licensing of Production and Utilization Facilities

REG GUIDE-1.101, Rev. 2 *Emergency Planning and Preparedness For Nuclear Power Reactors endorsing NUMARC NESP-007 Methodology for Development of Emergency Action Levels*

DOT Emergency Response Guide for Hazardous Materials

ANSI Standard N.18.7-1976

Site Technical Specifications (Tech Specs), Abnormal Operating Instructions (AOIs), Emergency Operating Procedures (EOPs) and the Final Safety Analysis Report (FSAR) are also referenced in Appendix C of the Radiological Emergency Plan To Support the Emergency Classification Flow Chart.

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**Watts Bar
Nuclear
Plant**

**Emergency Classification
and
Declaration
Methodology**

BASIS

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<i>Section</i>	<i>1.0</i>	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>	<i>1.1</i>	FUEL CLAD BARRIER
<i>IC</i>	<i>1.1.1</i>	Critical Safety Function Status
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p><u>LOSS:</u> Core Cooling Red</p> <p><u>Potential LOSS</u> Core Cooling Orange OR Heat Sink Red (RHR <u>Not</u> in Service)</p>
<i>Basis</i>		<p><u>LOSS:</u> The "Loss" IC addresses the condition of inadequate Core Cooling.</p> <p>If the Emergency Operating Procedure status trees indicate a red path the condition must be considered to be an extreme challenge to the safety function needed to ensure protection of the public.</p> <p>Core Cooling - Red indicates significant superheating and core uncover and is considered to indicate a "Loss" of the Fuel Clad Barrier.</p> <p><u>Potential LOSS:</u> The "Potential Loss" IC addresses the condition where an inadequate Core Cooling situation can develop. If the Emergency Operating Procedure status trees indicate an orange path, the conditions must be considered to be a severe challenge to the safety function.</p> <p>Core Cooling - Orange indicates subcooling has been lost and that some clad damage may occur.</p> <p>Heat Sink - Red indicates the heat sink function is under extreme challenge. It should be noted that this EAL for "Potential Loss" is not applicable if actions of FR-H.1 are not implemented due to Operator ability to control Aux Feedwater >410 gpm.</p> <p>Either of these two items indicate a "Potential Loss" of the Fuel Clad Barrier.</p>
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101 FR-C.1 Inadequate Core Cooling FR-C.2 Degraded Core Cooling FR-H.1 Loss of Heat Sink

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<i>Section</i>	<i>1.0</i>	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>	<i>1.1</i>	FUEL CLAD BARRIER
<i>IC</i>	<i>1.1.2</i>	Primary Coolant Activity Level
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p><u>LOSS:</u> RCS sample activity is Greater Than 300 $\mu\text{Ci/gm}$ dose equivalent Iodine -131</p> <p><u>Potential LOSS</u> Not Applicable</p>
<i>Basis</i>		<p><u>LOSS:</u> The "Loss" IC addresses the Condition of high RCS activity. If the reading of RCS activity is $\geq 300 \mu\text{Ci/gm}$ it is well above expected iodine spikes and corresponds to about 1% fuel clad damage. This amount of clad damage indicates that significant clad heating has occurred.</p> <p><u>Potential LOSS:</u> There is no "Potential Loss" IC associated with this item.</p>
<i>Escalation</i>		Not Applicable
<i>References</i>		<p>NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101</p> <p>NUREG 1465 Accident Source Terms for Light-Water Nuclear Power Plants</p> <p>WBN, Radiation Monitor Readings for REP Response of the</p> <p>Primary Containment High Range Radiation Monitors, TI-RPS-162 R14.</p>

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<i>Section</i>	<i>1.0</i>	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>	<i>1.1</i>	FUEL CLAD BARRIER
<i>IC</i>	<i>1.1.3</i>	Incore Tcs Hi Quad Average
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p><u>LOSS:</u> Greater Than 1200°F</p> <p><u>Potential LOSS</u> Greater Than 727°F</p>
<i>Basis</i>		<p><u>LOSS:</u> The "Loss" IC uses a reading of 1200°F which corresponds to a Core Cooling Red condition on the EOP status trees. A reading of this magnitude corresponds to significant superheating of the reactor coolant and clad heating which results in a "Loss" of Fuel Clad Barrier (TCs is in reference to Incore Thermocouples.)</p> <p><u>Potential LOSS:</u> The "Potential Loss" IC uses a reading of 727°F which corresponds to a Core cooling Orange Condition on the EOP status trees. A reading of this magnitude corresponds to a loss of RCS subcooling.</p>
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101 FR-C.1 Inadequate Core Cooling FR-C.2 Degraded Core Cooling WBN-OSG4-188, (I-01)

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<i>Section</i>	<i>1.0</i>	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>	<i>1.1</i>	FUEL CLAD BARRIER
<i>IC</i>	<i>1.1.4</i>	Reactor Vessel Water Level
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p><u>LOSS:</u> Not Applicable</p> <p><u>Potential LOSS</u> VALID RVLIS Level <33% (No RCP running)</p>
<i>Basis</i>		<p><u>LOSS:</u> There is no "Loss" IC corresponding to this item because it is covered by the other Fuel Clad Barrier "Loss".</p> <p><u>Potential LOSS:</u> The "Potential Loss" IC is defined by an Orange Path on the Core Cooling status tree. The numeric value used is 33% level with no reactor coolant pumps running. This condition indicates that considerable Clad heating and loss of RCS subcooling has occurred.</p>
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101 FR-C.2 Degraded Core Cooling WBN-OSG4-188, (K-01)

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<i>Section</i>	<i>1.0</i>	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>	<i>1.1</i>	FUEL CLAD BARRIER
<i>IC</i>	<i>1.1.5</i>	Containment Radiation Monitors
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p><u>LOSS:</u> VALID reading increase of Greater Than :</p> <p>332 R/hr On RM-90-271 and 272</p> <p>or</p> <p>265 R/hr On RM-90-273 and 274</p> <p><u>Potential LOSS</u> Not Applicable</p>
<i>Basis</i>		<p><u>LOSS:</u> The "Loss" IC is defined by a VALID reading of 332 R/hr on the upper Containment Hi Rad monitors or 265 R/hr on the lower containment Hi Rad monitors. The level of radiation in the Containment is indicative of a loss of Coolant accident (LOCA) in the Containment in conjunction with fuel damage.</p> <p>The reading assumes the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 300 µCi/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 1% clad failure depending on core inventory and RCS volume). TI-RPS-162 values are based on a tritium core. Thus, this IC indicates a loss of both the fuel clad barrier and a loss of the RCS barrier.</p> <p><u>Potential LOSS:</u> There is no "Potential Loss" IC associated with this item.</p>
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92 NUREG 1465 Accident Source Terms for Light-Water Nuclear Power Plants WBN, Radiation Monitor Readings for REP Response of the Primary Containment High Range Radiation Monitors, TI-RPS-162 R14.

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<i>Section</i>	<i>1.0</i>	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>	<i>1.1</i>	FUEL CLAD BARRIER
<i>IC</i>	<i>1.1.6</i>	Site Emergency Director Judgment
<i>Mode</i>		1,2,3,4
<i>Description</i>		Any condition that, in the judgment of the SM/SED, indicates Loss or Potential Loss of the Fuel Clad Barrier comparable to the conditions listed above.
<i>Basis</i>		<p>This IC gives the SED the latitude to use his judgment in determining if the fuel clad barrier is or will be in a "Loss" or "Potential Loss" condition. This situation is usually considered when plant conditions are present that require the monitoring of CSFs or performance of EOP corrective actions. Specific cases where SED judgment may be required are the loss of instrumentation needed to monitor the CSFs and the loss of all AC power.</p> <p>Although the majority of the ICs provide very specific thresholds, the Site Emergency Director must remain alert to events or conditions that lead to the conclusion that exceeding the IC threshold is imminent. If, in the judgment of the Site Emergency Director, an imminent situation is at hand, the classification should be made as if the thresholds have been exceeded. While this is particularly prudent at the higher emergency classes (as the early classification may provide for more effective implementation of protective measures), it is nonetheless applicable to all emergency classes.</p>
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101

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<i>Section 1.0</i>	FISSION PRODUCT BARRIER MATRIX
<i>Event</i> <i>1.2</i>	RCS BARRIER
<i>IC</i> <i>1.2.1</i>	Critical Safety Function Status
<i>Mode</i>	1,2,3,4
<i>Description</i>	<p><u>LOSS:</u> Not Applicable</p> <p><u>Potential LOSS</u> Pressurized Thermal Shock Red OR Heat Sink Red (RHR <u>Not</u> in Service)</p>
<i>Basis</i>	<p><u>LOSS:</u> There is no "Loss" IC associated with this item.</p> <p><u>Potential LOSS:</u> The "Potential Loss" IC is defined by a Red path on Pressurized thermal Shock or a Red path on the Heat Sink CSF status trees. In the case of PTS, consideration is given to a failure of the reactor vessel resulting in a loss of coolant accident (LOCA).</p> <p>Heat Sink Red is identified since an inability to remove core heat could lead to a vessel or RCS failure. Also, in the case of loss of heat sink, it may become necessary to cool the core by bleed and feed with safety injection. Although this is a deliberate action, the open PORV is a breach of the RCS Barrier that would allow fission products to be released to containment. It should be noted that this (Heat Sink) EAL for "Potential Loss" is not applicable if actions of FR-H.1 are not implemented due to Operator ability to control Aux Feedwater >410 gpm.</p>
<i>Escalation</i>	Not Applicable
<i>References</i>	NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101 FR-P.1 Pressurized Thermal Shock FR-H.1 Loss of Heat Sink

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<i>Section</i>	<i>1.0</i>	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>	<i>1.2</i>	RCS BARRIER
<i>IC</i>	<i>1.2.2</i>	RCS Leakage/LOCA
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p><u>LOSS:</u> RCS Leak results in Loss of subcooling (<65°F Indicated), [85°F ADV]</p> <p><u>Potential LOSS</u> Non Isolatable RCS Leak Exceeding The Capacity Of <u>One</u> Charging Pump in the Normal Charging Alignment</p> <p style="text-align: center;"><u>OR</u></p> <p>RCS Leakage Results in Entry into E-1</p>
<i>Basis</i>		<p><u>LOSS:</u> The "Loss" IC addresses conditions where leakage from the RCS is greater than available inventory control capacity such that a loss of subcooling has occurred. The loss of subcooling is the fundamental indication that the inventory control systems are inadequate in maintaining RCS pressure and inventory against the mass loss through the leak.</p> <p><u>Potential LOSS:</u> The "Potential Loss" IC 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 and letdown in service. This assures that any event that results in significant RCS inventory shrinkage or loss (e.g., events leading to reactor scram and ECCS actuation) will result in no lower than an "Alert" emergency classification.</p>
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101 E-1 Loss of Reactor or Secondary Coolant WBN-OSG4-188, (H-13)

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<i>Section</i>	<i>1.0</i>	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>	<i>1.2</i>	RCS BARRIER
<i>IC</i>	<i>1.2.3</i>	Steam Generator Tube Rupture
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p><u>LOSS:</u> SGTR that results in a safety injection actuation</p> <p style="text-align: center;"><u>OR</u></p> <p>Entry into E-3</p> <p><u>Potential LOSS</u> Not Applicable</p>
<i>Basis</i>		<p><u>LOSS:</u> The "Loss" IC addresses conditions where the steam generator tube rupture (SGTR) exists and the RCS flow into the steam generator is such that pressurizer level and pressure cannot be maintained. The inability to maintain level via the normal charging header, with CVCS letdown in service requires a safety injection by procedure. If a manual safety injection is not initiated an auto SI will occur due to a low pressurizer pressure.</p> <p>Any event that results in significant RCS inventory shrinkage or loss (e.g., events leading to reactor scram and ECCS actuation) will result in no lower than an "Alert" emergency classification.</p> <p>This IC also addresses the entry into EOP, E-3, Steam Generator Tube Rupture, under any circumstance.</p> <p>This "Loss" IC in conjunction with the Containment Barrier "Loss" IC #4 addresses the situation where the S/G that is ruptured and also Faulted. This "Loss" of two barriers requires an Event classification of Site Area Emergency.</p> <p><u>Potential LOSS:</u> There is no "Potential Loss" IC associated with this item.</p>
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101 AOI-33 Steam Generator Tube Leak E-3 Steam Generator Tube Rupture

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<i>Section</i>	<i>1.0</i>	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>	<i>1.2</i>	RCS BARRIER
<i>IC</i>	<i>1.2.4</i>	Reactor Vessel Water Level
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p><u>LOSS:</u> VALID RVLIS level < 33% (No RCP Running)</p> <p><u>Potential LOSS</u> Not Applicable</p>
<i>Basis</i>		<p><u>LOSS:</u> The "Loss" IC is defined by an Orange path on the Core cooling status tree (CSF). The numeric value used is 33% level with no reactor coolant pumps running. Inability to maintain reactor vessel water level is the fundamental indication that the RCS barrier has been lost.</p> <p>This "Loss" EAL in conjunction with the Fuel Clad Barrier "Potential Loss" IC #4 requires an event classification of Site Area Emergency.</p> <p><u>Potential LOSS:</u> There is no "Potential Loss" IC associated with this item.</p>
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101 FR-C.2 Degraded Core Cooling WBN-OSG4-188, (K-01)

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<i>Section</i>	<i>1.0</i>	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>	<i>1.2</i>	RCS BARRIER
<i>IC</i>	<i>1.2.5</i>	Site Emergency Director Judgment
<i>Mode</i>		1,2,3,4
<i>Description</i>		Any Condition that, in the Judgment of the SM/SED, indicates Loss or Potential Loss of the RCS Barrier comparable to the conditions Listed Above.
<i>Basis</i>		<p>This IC gives the SED the latitude to use his judgment in determining if the RCS barrier is or will be in a "Loss or Potential Loss" condition. This situation is usually considered when plant conditions are present that require the monitoring of CSFs or performance of EOP corrective actions. Specific cases where SED judgment may be required are the loss of instrumentation needed to monitor the CSFs and the loss of all AC power.</p> <p>Although the majority of the EALs provide very specific thresholds, the SED must remain alert to events or conditions that lead to the conclusion that exceeding the EAL threshold is imminent. If, in the judgment of the SED, an imminent situation is at hand, the classification should be made as if the thresholds have been exceeded. While this is particularly prudent at the higher emergency classes (as the early classification may provide for more effective implementation of protective measures), it is nonetheless applicable to all emergency classes.</p>
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101

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<i>Section</i>	<i>1.0</i>	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>	<i>1.3</i>	CNTMT BARRIER
<i>IC</i>	<i>1.3.1</i>	Critical Safety Function Status
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p><u>LOSS:</u> Not Applicable</p> <p><u>POTENTIAL LOSS:</u> Containment (FR-Z.1) <u>Red</u> OR Actions of FR-C.1 (Red Path) are INEFFECTIVE</p>
<i>Basis</i>		<p><u>LOSS:</u> There is no "Loss" IC associated with this item.</p> <p><u>Potential LOSS:</u> The first "Potential Loss" IC is defined by a Red Path on the Containment status tree. A Red Path indicates an extreme challenge to the safety function derived from appropriate instrument readings and/or sampling results, and thus represents a potential loss of containment. Conditions leading to a containment Red Path result from RCS barrier and/or Fuel Clad Barrier Loss. Thus, this IC is primarily a discriminator between the Site Area Emergency and General Emergency representing a potential loss of the third barrier.</p> <p>The second "Potential Loss" IC is defined by a Red Path on the core cooling status tree with FR-C.1 ineffective. In this IC, the functional restoration procedures are those emergency operating procedures that address the recovery of the core cooling critical safety functions. The procedure is considered ineffective if the temperature is not decreasing or if the vessel water level is not increasing.</p> <p>The conditions identified in this potential loss IC represent an imminent melt sequence which, if not corrected, could lead to vessel failure and an increased potential for containment failure. In conjunction with the core exit thermocouple ICs in the Fuel and RCS barrier columns, this IC would result in the declaration of a General Emergency -- loss of two barriers and the potential loss of a third. If the functional restoration procedures are ineffective, there is no "success" path.</p>
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101 FR-Z.1 High Containment Pressure FR-C.1 Inadequate Core Cooling

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<i>Section</i>	<i>1.0</i>	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>	<i>1.3</i>	CNTMT BARRIER
<i>IC</i>	<i>1.3.2</i>	Containment Pressure/Hydrogen
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p><u>LOSS:</u> Rapid unexplained pressure decrease following initial pressure increase</p> <p style="text-align: center;"><u>OR</u></p> <p>Containment pressure or Sump Level <u>Not</u> increasing (with LOCA in progress)</p> <p><u>Potential LOSS</u> Containment Hydrogen increases to >4% by volume</p> <p style="text-align: center;"><u>OR</u></p> <p>Pressure >2.8 PSIG (Phase B) with < One full train of Containment spray</p>
<i>Basis</i>		<p><u>LOSS:</u> This first "Loss" IC address a rapid unexplained loss of pressure (i.e., not attributable to containment spray or condensation effects) following an initial pressure increase indicating a loss of containment integrity.</p> <p>The second "Loss" IC addresses the situation where the containment pressure or sump level are not increasing with a LOCA in progress. This could indicate containment, bypass and loss of containment integrity. This IC, in conjunction with RCS barrier IC #2, results in an Event Classification of Site Area Emergency.</p> <p><u>Potential LOSS:</u> The condition of high containment pressure, greater than 13.5 PSIG, is addressed by the CSF, Containment Red, "Potential Loss", IC #1.3.1</p> <p>The first "Potential Loss" IC addresses the existence of an explosive mixture of hydrogen and oxygen in the containment, which if ignited, would be a challenge to the Containment Barrier.</p> <p>The second "Potential Loss" IC represents a potential loss of containment in that the containment heat removal/depressurization system (e.g., containment sprays, ice condenser, etc.) are either lost or performing in a degraded manner, as indicated by containment pressure greater than the setpoint, ØB (2.8 PSIG), at which the equipment was supposed to have actuated.</p> <p>These "Potential Loss" ICs are primarily a discrimination between the Site Area Emergency and General Emergency representing a potential loss of the third barrier.</p>

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<i>Section</i>	<i>1.0</i>	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>	<i>1.3</i>	CNTMT BARRIER
<i>IC</i>	<i>1.3.2</i>	Containment Pressure/Hydrogen (continued)
<i>Mode</i>		1,2,3,4
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101 FR-Z.1 High Containment Pressure

<i>Section</i>	<i>1.0</i>	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>	<i>1.3</i>	CNTMT BARRIER
<i>IC</i>	<i>1.3.3</i>	Containment Isolation Status
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p><u>LOSS:</u> Containment Isolation (when required) is Incomplete <u>and</u> a Release Path to the Environment Exists</p> <p><u>Potential LOSS</u> Not Applicable</p>
<i>Basis</i>		<p><u>LOSS:</u> The Loss IC is intended to address incomplete containment isolation that allows a direct release to the environment. It represents a loss of the Containment Barrier.</p> <p><u>Potential LOSS:</u> There is no "Potential Loss" IC associated with this item.</p>
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101

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<i>Section</i>	<i>1.0</i>	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>	<i>1.3</i>	CNTMT BARRIER
<i>IC</i>	<i>1.3.4</i>	Containment Bypass
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p><u>LOSS:</u> RUPTURED S/G is also FAULTED outside of CNTMT</p> <p style="text-align: center;"><u>OR</u></p> <p>Prolonged (> 4 Hours) Secondary Side release outside CNTMT from a S/G with a SGTL > T/S Limits</p> <p><u>POTENTIAL LOSS:</u> Unexplained VALID increase in area or Ventilation RAD monitors in areas adjacent to CNTMT (with LOCA in progress)</p>
<i>Basis</i>		<p><u>LOSS:</u> The first "Loss" IC addresses a non-isolatable secondary side release from a ruptured steam generator that is also faulted outside containment. This allows a direct release of radioactive fission and activation products to the environment. Resultant offsite dose rates are a function of many variables. Examples include: Coolant Activity, Actual Leak Rate, SG Carry Over, Iodine Partitioning, and Meteorology. Therefore, dose assessment in accordance with event Gaseous Effluent (7.1) General Emergency, "Site Boundary 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", is required when there is indication that the Fuel Clad Barrier is potentially lost.</p> <p>This IC would exist in conjunction with the RCS barrier "Loss" IC #3 and results in an Event classification of a Site Area Emergency. Escalation to General Emergency would be based on "Potential Loss" of the Fuel Clad Barrier.</p> <p>The second "Loss" IC addresses a prolonged, greater than four (4) hour, secondary side release outside of the Containment from a steam generator having primary to secondary leakage greater than Tech. Spec. limits, (LCO 3.4.13). This IC results in an Event classification of Unusual Event. This indicator's intent addresses nonisolable main stream line breaks (MSLB) outside containment, feedwater line breaks, failed open relief valves or atmospheric dump valves or plant cooldown via atmospheric steam dump due to loss of offsite power or main condenser. However, it is not the intent of this indicator to address transient events such as (1) MSLB downstream of the MSIV if the MSIV isolate the break,</p>

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<i>Section</i>	<i>1.0</i>	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>	<i>1.3</i>	CNTMT BARRIER
<i>IC</i>	<i>1.3.4</i>	Containment Bypass (continued)
<i>Mode</i>		1,2,3,4
<i>Basis</i> (continued)		<p><u>LOSS:</u> (continued)</p> <p>or (2) affected S/G isolation occurs in accordance with plant procedures, or for other similar events. Prolonged steam releases via the main condenser air ejectors, or steam-driven auxiliary feed pump exhaust should be classified on the basis of dose assessments rather than the Fission Product Barrier Matrix.</p> <p>RUPTURED: (Steam Generator) Existence of primary to secondary leakage of a magnitude greater than charging pump capacity.</p> <p>FAULTED: (Steam Generator) Existence of secondary side leakage (i.e., steam or feed line break) that results in an uncontrolled decrease in steam generator pressure or the steam generator being completely depressurized.</p> <p><u>Potential LOSS:</u></p> <p>The "Potential Loss" IC addresses an increase in area or ventilation radiation monitors, with a LOCA in progress, which is indicative of a potential loss of the Containment Barrier. This IC in conjunction with the RCS barrier IC #2 results in an Event classification of Site Area Emergency.</p>
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101 E-2 Faulted Steam Generator Isolation

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<i>Section</i>	<i>1.0</i>	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>	<i>1.3</i>	CNTMT BARRIER
<i>IC</i>	<i>1.3.5</i>	Significant Radioactivity in Containment
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p><u>LOSS:</u> Not Applicable</p> <p><u>Potential LOSS</u> VALID reading increase of Greater Than :</p> <p>5600 R/hr On RM-90-271 and RM-90-272</p> <p>or</p> <p>4470 R/hr On RM-90-273 and RM-90-274</p>
<i>Basis</i>		<p><u>LOSS:</u> There is no "Loss: IC associated with this item.</p> <p><u>Potential LOSS:</u> The "Potential Loss" IC is defined by a VALID reading of 5600 R/hr on the upper Containment Hi Rad monitors or 4470 R/hr on the lower containment Hi Rad monitors.</p> <p>This reading indicates significant fuel damage well in excess of the EALs associated with both loss of Fuel Clad and loss of RCS barriers. A major release of radioactivity requiring offsite protective actions 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%. TI RPS-162 values are based on a tritium core.</p>
<i>Escalation</i>		Not Applicable
<i>References</i>		<p>NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101</p> <p>NUREG 1465 Accident Source Terms for Light-Water Nuclear Power Plants</p> <p>NUREG-1228, Source Estimates During Incident Response to Severe Nuclear Power Plant Accidents</p> <p>WBN, Radiation Monitor Readings for REP Response of the Primary Containment High Range Radiation Monitors, TI-RPS-162 R14.</p>

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<i>Section</i>	<i>1.0</i>	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>	<i>1.3</i>	CNTMT BARRIER
<i>IC</i>	<i>1.3.6</i>	Site Emergency Director Judgment
<i>Mode</i>		1,2,3,4
<i>Description</i>		Any condition that, in the judgment of the SM/SED, indicates Loss or Potential Loss of the CNTMT Barrier comparable to the conditions listed above.
<i>Basis</i>		<p>This IC gives the SED the latitude to use his/her judgment in determining if the Containment Barrier is a "Potential Loss" or "Loss". This situation is usually considered when plant conditions are present that require the monitoring of CSFs or performance of EOP corrective actions. Specific cases where SED judgment may be required are the loss of instrumentation needed to monitor the CSFs and the loss of all AC power.</p> <p>Although the majority of the ICs provide very specific thresholds, the SED must remain alert to events or conditions that lead to the conclusion that exceeding the EAL threshold is imminent. If, in the judgment of the SED, an imminent situation is at hand, the classification should be made as if the thresholds have been exceeded. While this is particularly prudent at the higher emergency classes (as the early classification may provide for more effective implementation of protective measures), it is nonetheless applicable to all emergency classes.</p>
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101

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**FISSION PRODUCT
BARRIER UTILIZATION**

in

**EMERGENCY
EVENT
CLASSIFICATION**

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<i>Section</i>	1.0	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>		Not Applicable
<i>Classification</i>		GENERAL EMERGENCY
<i>IC</i>		FG1
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p>LOSS of any two barriers</p> <p style="text-align: center;"><u>and</u></p> <p>Potential LOSS of third barrier</p>
<i>Basis</i>		<p>Definition: Events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels outside the EXCLUSION AREA BOUNDARY.</p> <p>The main differentiation between the Site Area and General Emergency classification is whether or not the EPA PAG plume exposure levels are expected to be exceeded outside the site boundary. This threshold, in addition to dynamic dose assessment considerations, addresses NRC and offsite emergency response agency concerns as to timely declaration of a General Emergency.</p> <p>The main objective of the General Emergency is to determine whether evacuation or sheltering of the general public is indicated based on EPA PAGs, and therefore should be interpreted to include radionuclide release regardless of cause. Consideration must be given to failures of systems and or structures that provide fission product barrier integrity which is the primary method of preventing uncontrolled radionuclide releases. In terms of fission product barriers, the loss of two barriers with potential loss of the third barrier constitutes a General Emergency.</p> <p>In utilizing the Fission Product Barrier sub-sections (i.e., Fuel Clad, RCS Barrier and CNTMT Barrier) the Site Emergency Director (SED) will use the instructions in EPIP 1, to determine the General Emergency. These instructions provide clear guidance on the proper use of the classification charts and a correct classification of a General Emergency.</p>
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101

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<i>Section</i>	1.0	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>		Not Applicable
<i>Classification</i>		SITE AREA EMERGENCY
<i>IC</i>		FS1
<i>Mode</i>		1,2,3,4
<i>Description</i>		LOSS or Potential LOSS of any two barriers
<i>Basis</i>		<p>Definition:</p> <p>Events are in process or have occurred which involve an actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; (1) toward site personnel or equipment that could lead to the like failure of or; (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels outside the EXCLUSION AREA BOUNDARY.</p> <p>It is considered to be a challenge to plant functions necessary for the protection of the public if the integrity of any two of the three fission product barriers has or has the potential of being degraded. This approach is more conservative than REG GUIDE 1.101 in that the containment barrier is not weighted less significant than the other two barriers. Thus a "Loss" or "Potential Loss" of any two barriers is a Site Area Emergency.</p> <p>This approach also simplifies the Site Area Emergency classification from the fission product barrier matrix.</p> <p>In utilizing the Fission Product Barrier sub-sections (i.e., Fuel Clad, RCS Barrier and CNTMT Barrier) the Site Emergency Director (SED) will use the instructions in EPIP 1, to determine the Site Area Emergency. These instructions provide clear guidance on the proper use of the classification charts and a correct classification of a Site Area Emergency.</p>
<i>Escalation</i>		Escalation would be based on Actual or Imminent Substantial Core Degradation
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101

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<i>Section</i>	1.0	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>		Not Applicable
<i>Classification</i>		ALERT
<i>IC</i>		FA1
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p>Any LOSS <u>or</u> Potential LOSS of Fuel Clad Barrier</p> <p style="text-align: center;"><u>OR</u></p> <p>Any LOSS <u>or</u> Potential LOSS of RCS barrier</p>
<i>Basis</i>		<p>Definition:</p> <p>Events are in process or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.</p> <p>The "Loss" or "Potential Loss" of either the Fuel Clad Barrier or RCS barrier is considered to be an actual or potential substantial degradation of the level of safety of the plant. The Alert classification resulting from potential degradation of the fuel clad or RCS integrity also addresses the operation staff's need for help by staffing the Technical Support Center (TSC), independent of whether an actual decrease in plant safety is determined.</p> <p>This increased monitoring can then be used to better determine the actual plant safety state, whether escalation to a higher emergency class is warranted, or termination of the emergency class declaration is warranted. Dose consequences from these events are small fractions of the EPA PAG plume exposure levels, i.e., about 10 millirem (mR) to 100 millirem (mR).</p> <p>In utilizing the Fission Product Barrier sub-sections (i.e., Fuel Clad, RCS Barrier and CNTMT Barrier) the Site Emergency Director (SED) will use the instructions in EPIP 1, to determine the Alert. These instructions provide clear guidance on the proper use of the classification charts and a correct classification of an Alert.</p>
<i>Escalation</i>		Escalation would be based on Actual or Likely Major Failures of Plant Functions needed to Protect the Public.
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101

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<i>Section</i>	<i>1.0</i>	FISSION PRODUCT BARRIER MATRIX
<i>Event</i>		Not Applicable
<i>Classification</i>		UNUSUAL EVENT
<i>IC</i>		FU1
<i>Mode</i>		1,2,3,4
<i>Description</i>		LOSS or Potential LOSS of Containment Barrier
<i>Basis</i>		<p>Definition: Events are in process or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.</p> <p>Potential degradation of the level of safety of the plant is indicated primarily by exceeding a plant technical specification Limiting Condition of Operation (LCO) allowable action statement time for achieving required mode change. Precursors of more serious events are also included because precursors do represent a potential degradation in the level of safety of the plant. Minor releases of radioactive materials are included. In this emergency class, however, releases do not require monitoring or offsite response (e.g., dose consequences of less than 10 millirem) (mR).</p> <p>The event classification of Unusual Event from the barrier matrix is only from a "Loss" or "Potential Loss" of the containment barrier. This is consistent with the NUMARC/NESP-007 statement, "The fuel clad barrier and the RCS barrier are weighted more heavily than the containment barrier." The "Loss or "Potential Loss" of the containment barrier alone is not considered to be substantial degradation of the level of safety of the plant when the other two fission product barriers are intact. Thus the (UE) classification is justified.</p> <p>In utilizing the Fission Product Barrier sub-sections (i.e., Fuel Clad, RCS Barrier and CNTMT Barrier) the Site Emergency Director (SED) will use the instructions in EPIP 1, to determine the Unusual Event. These instructions provide clear guidance on the proper use of the classification charts and a correct classification of an Unusual Event.</p>
<i>Escalation</i>		Escalation would be based on Actual or Potential Substantial Degradation of the Level of Safety of the Plant.
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101

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<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.1</i>	LOSS OF INSTRUMENTATION
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		All
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" and "Radiological Effluents" (Section 7)
<i>Basis</i>		The basis for a General Emergency in this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected, or Radiological Effluents.
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

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<i>Section</i>	2.0	SYSTEM DEGRADATION
<i>Event</i>	2.1	LOSS OF INSTRUMENTATION
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		1,2,3,4
<i>Description</i>	<p>Inability to monitor a SIGNIFICANT TRANSIENT in progress (1 and 2 and 3 and 4)</p> <ol style="list-style-type: none"> 1. Loss of most (>75%) of MCR annunciators (<u>and</u> Annunciator Monitor) <u>or</u> indications. 2. SIGNIFICANT TRANSIENT in progress 3. Loss of Integrated Computer System (ICS) <u>and</u> SPDS 4. Inability to directly monitor any of the following CSFs: <div style="display: flex; justify-content: space-around; margin-top: 10px;"> <div style="text-align: center;">Subcriticality Core Cooling Heat Sink</div> <div style="text-align: center;">PTS Containment Inventory</div> </div> 	
<i>Basis</i>	<p>This IC is intended to recognize the inability of the control room staff to monitor the plant response to a transient.</p> <p>When the loss of safety system annunciators is complicated with an unplanned power change as well as loss of SPDS, ICS and Control Room indications needed to monitor Plant Critical Safety Functions, a Site Area Emergency exists. This declaration is prudent because the control room staff cannot monitor safety functions needed for protection of the public.</p> <p>For the purposes of quantification of MOST it is estimated that if 75% of the annunciators are lost there is an increased risk that a degraded plant condition could go undetected. It is not intended that a detailed count of the instrumentation be performed but only a rough approximation be used to determine the severity of the condition.</p> <p>SIGNIFICANT TRANSIENT involves an UNPLANNED event involving one or more of the following: (1) An automatic turbine runback > 15% thermal reactor power; (2) Electrical load rejection >25% full electrical load; (3) Reactor Trip; or (4) Safety Injection System Activation.</p> <p>Due to the limited number of safety systems in operation during cold shutdown and refueling modes, no initiating conditions are indicated during these modes of operation.</p>	

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<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.1</i>	LOSS OF INSTRUMENTATION
<i>Classification</i>		SITE AREA EMERGENCY (continued)
<i>Mode</i>		1,2,3,4
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Challenges".
<i>References</i>		NUMARC/NESP-007, SS6, Rev. 2, 1/92 T.S. 3.3.1 Reactor Trip System Instrumentation T.S. 3.3.2 Engineering Safety Features Activation System Instrumentation (ESFAS) T.S. 3.3.3 Post Accident Monitoring Instrumentation AOI-26 Loss of Main Control Room Annunciators

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<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.1</i>	LOSS OF INSTRUMENTATION
<i>Classification</i>		ALERT
<i>Mode</i>		1,2,3,4
<i>Description</i>	<p>UNPLANNED loss of most (>75%) MCR annunciators (<u>and</u> Annunciator Monitor) <u>or</u> indications for >15 Minutes <u>with either</u> a SIGNIFICANT TRANSIENT in progress <u>or</u> Integrated Computer System (ICS) and SPDS Unavailable (1 and 2 and 3)</p> <ol style="list-style-type: none"> 1. UNPLANNED loss of most (>75%) of MCR annunciators (<u>and</u> Annunciator Printer) <u>or</u> indications for >15 Minutes 2. SM/SED Judgement that increased surveillance is required to Safely operate the unit (beyond Shift complement) 3. (a or b) <ol style="list-style-type: none"> a. SIGNIFICANT TRANSIENT in progress b. Loss of ICS <u>and</u> SPDS 	
<i>Basis</i>	<p>This IC indicates that when the loss of safety system annunciators is complicated with the loss of SPDS, and ICS or a plant transient a deterioration of the level of plant safety has occurred and an Alert should be declared.</p> <p>Fifteen minutes was selected as a threshold value to exclude momentary power losses or transients.</p> <p>The declaration will ensure that adequate resources are available to monitor and control plant systems so that any further degraded condition can be detected and responded to.</p> <p>SIGNIFICANT TRANSIENT involves an UNPLANNED event involving one or more of the following: (1) An automatic turbine runback > 15% thermal reactor power; (2) Electrical load rejection >25% full electrical load; (3) Reactor Trip; or (4) Safety Injection System Activation.</p> <p>Unplanned loss of annunciators excludes scheduled maintenance and testing activities.</p>	

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<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.1</i>	LOSS OF INSTRUMENTATION
<i>Classification</i>		ALERT (continued)
<i>Mode</i>		1,2,3,4
<i>Basis</i> (continued)		<p>For the purposes of quantification of MOST it is estimated that if 75% of the annunciators are lost there is an increased risk that a degraded plant condition could go undetected. It is not intended that a detailed count of the instrumentation be performed but only a rough approximation be used to determine the severity of the condition.</p> <p>Due to the limited number of safety systems in operation during cold shutdown and refueling modes, no initiating conditions are indicated during these modes of operation.</p>
<i>Escalation</i>		Escalation of this event will be based on the inability of the operating crew to monitor a transient in progress.
<i>References</i>		NUMARC/NESP-007, SA4, Rev. 2, 1/92 T.S. 3.3.1 Reactor Trip System Instrumentation T.S. 3.3.2 Engineering Safety Features Activation System Instrumentation (ESFAS) T.S. 3.3.3 Post Accident Monitoring Instrumentation AOI-26 Loss of Main Control Room Annunciators

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<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.1</i>	LOSS OF INSTRUMENTATION
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		1,2,3,4
<i>Description</i>	<p>UNPLANNED loss of most <u>or</u> All Safety System annunciators <u>or</u> indications in the Control Room for >15 Minutes (1 and 2 and 3)</p> <ol style="list-style-type: none"> 1. Unplanned loss of most (>75%) of MCR annunciators (<u>and</u> Annunciator Monitor) <u>or</u> indications for >15 Minutes 2. SM/SED Judgement that increased surveillance is required to safely operate the unit (beyond Shift complement) 3. Integrated Computer System (ICS) <u>or</u> SPDS is in service and capable of displaying data requested 	
<i>Basis</i>	<p>For this IC, if annunciators are partially or completely lost it is still possible to use other systems to indicate plant conditions (e.g., SPDS or ICS). However, it is prudent to declare an Unusual Event since there is a greater risk that a degraded condition could go undetected.</p> <p>Fifteen minutes was selected as a threshold value to exclude momentary power losses or transients.</p> <p>For the purposes of quantification of MOST it is estimated that if 75% of the annunciators are lost there is an increased risk that a degraded plant condition could go undetected. It is not intended that a detailed count of the instrumentation be performed but only a rough approximation be used to determine the severity of the condition.</p> <p>Unplanned loss of annunciators excludes scheduled maintenance and testing activities.</p> <p>The declaration will ensure that adequate resources are available to monitor and control plant systems.</p> <p>Due to the limited number of safety systems in operation during cold shutdown, refueling and defueling modes, no initiating conditions are indicated during these modes of operation.</p>	

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<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.1</i>	LOSS OF INSTRUMENTATION
<i>Classification</i>		UNUSUAL EVENT (continued)
<i>Mode</i>		1,2,3,4
<i>Escalation</i>		Escalation of this event would be based on loss of annunciators complicated by the loss of SPDS and plant computer or a transient in progress.
<i>References</i>		NUMARC/NESP-007, SU3, Rev. 2, 1/92 T.S. 3.3.1 Reactor Trip System Instrumentation T.S. 3.3.2 Engineering Safety Features Activation System Instrumentation (ESFAS) T.S. 3.3.3 Post Accident Monitoring Instrumentation AOI-26 Loss of Main Control Room Annunciators

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<i>Section 2.0</i>	SYSTEM DEGRADATION
<i>Event</i> 2.2	LOSS OF FUNCTION
<i>Classification</i>	GENERAL EMERGENCY
<i>Mode</i>	Not Applicable
<i>Description</i>	Refer to the "Fission Product Barrier Matrix"
<i>Basis</i>	The basis for a General Emergency in this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>	Not Applicable
<i>References</i>	NUMARC/NESP-007, Rev. 2, 1/92

<i>Section 2.0</i>	SYSTEM DEGRADATION
<i>Event</i> 2.2	LOSS OF FUNCTION
<i>Classification</i>	SITE AREA EMERGENCY
<i>Mode</i>	1,2,3,4
<i>Description</i>	<p>Complete loss of function needed to achieve <u>or</u> maintain Hot Shutdown (1 or 2)</p> <ol style="list-style-type: none"> CSF status tree indicates Core Cooling Red CSF status tree indicates Heat Sink Red (RHR <u>not</u> in service) <p>Note: Also refer to "Failure of Rx Protection" (2.3)</p>
<i>Basis</i>	<p>This IC 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.</p> <p>Heat Sink - Red indicates the heat sink function is under extreme challenge. It should be noted that this EAL is not applicable if actions of FR-H.1 are not implemented due to Operator ability to control Aux Feedwater >410 gpm.</p> <p>If RHR cooling is in service then the CSF status tree for Heat Sink Red is not applicable. Therefore, this comment has been added to the EAL.</p>
<i>Escalation</i>	Escalation will be based on "Fission Product Barrier Challenges".
<i>References</i>	NUMARC/NESP-007, SS4, Rev. 2, 1/92 T.S. 3.4 RCS Loops Mode 1-4 FR-C.1 Inadequate Core Cooling FR-H.1 Loss of Heat Sink

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<i>Section</i>	2.0	SYSTEM DEGRADATION
<i>Event</i>	2.2	LOSS OF FUNCTION
<i>Classification</i>		ALERT
<i>Mode</i>		4
<i>Description</i>	<p>Complete loss of function needed to achieve Cold Shutdown when Shutdown required by Tech Specs (1 and 2 and 3)</p> <ol style="list-style-type: none"> 1. Shutdown is required 2. Loss of RHR capability 3. Loss of secondary heat sink and condenser 	
<i>Basis</i>	<p>For this IC the inability to achieve Cold Shutdown when it is required refers to unplanned actions resulting in either equipment malfunctions or operator error that prevents achievement of Cold Shutdown</p> <p>This condition could result from a loss of RHR capability service water to the RHR, heat exchange or equipment failure with the RHR system or AC/DC power loss to the RHR and or service water components (i.e., CCS, ERCW)</p> <p>The combination of this and the loss of the secondary heat sink for cooldown indicates a degradation of the level of plant safety and warrants the declaration of an Alert.</p>	
<i>Escalation</i>	Escalation of this event would be based on complete loss of functions needed to achieve <u>or</u> maintain Hot Shutdown.	
<i>References</i>	NUMARC/NESP-007, SA3 (expanded) T.S. 3.4 RCS Loops Mode 1-4 FR-C.1 Inadequate Core Cooling FR-H.1 Loss of Heat Sink	

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<i>Section</i>	2.0	SYSTEM DEGRADATION
<i>Event</i>	2.2	LOSS OF FUNCTION
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		All
<i>Description</i>	<p>A. UNPLANNED loss of all In-Plant Communication capability (1 and 2 and 3)</p> <ol style="list-style-type: none"> UNPLANNED loss of EPABX (PAX) phones UNPLANNED loss of all sound powered phones UNPLANNED loss of all radios <p style="text-align: center;"><u>or</u></p> <p>B. UNPLANNED loss of all Offsite Communication capability (1 and 2 and 3 and 4 and 5)</p> <ol style="list-style-type: none"> UNPLANNED loss of all EPABX (PAX) phones UNPLANNED loss of all Radio frequencies UNPLANNED loss of all OPX (Microwave) system UNPLANNED loss of all 1-FB-Bell lines UNPLANNED loss of all FTS 2000 (NRC) system 	
<i>Basis</i>	<p>The purpose of this IC is to recognize a loss of communications capability that either defeats the plant operations staff's ability to perform routine tasks necessary for plant operations or the ability to communicate problems with offsite authorities.</p> <p>The loss of offsite communications ability is expected to be significantly more comprehensive than those addressed by 10 CFR 50.72.</p> <p>Onsite communications loss must encompass the loss of all means of routine communications (i.e., phones, page party system and radio/walkie talkies).</p> <p>Offsite communications loss must encompass the loss of all means of communications with offsite authorities. This IC is intended to be used only when extraordinary means are being utilized to make communications possible (i.e., individuals being sent to offsite locations).</p>	
<i>Escalation</i>	Escalation of this event will involve the loss of other plant functions.	
<i>References</i>	NUMARC/NESP-007, SU6, Rev. 2, 1/92 10 CFR 50.72 NUREG 0654	

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<i>Section 2.0</i>	SYSTEM DEGRADATION
<i>Event</i> 2.3	FAILURE OF RX PROTECTION
<i>Classification</i>	GENERAL EMERGENCY
<i>Mode</i>	1,2
<i>Description</i>	<p><u>Loss of Core Cooling capability and VALID Trip Signals did not result in a reduction of Rx power to <5% and decreasing</u> (1 and 2)</p> <ol style="list-style-type: none"> 1. (a or b) <ol style="list-style-type: none"> a. CSF status tree indicates Core Cooling Red b. CSF status tree indicates Heat Sink Red 2. FR-S.1 entered <u>and</u> subsequent actions <u>Did Not</u> result in a RX Power of <5% and decreasing
<i>Basis</i>	<p>Under the conditions of this IC, the efforts to bring the reactor less than five percent power have been unsuccessful and, as a result, the reactor is producing more heat than the maximum decay heat load for which the safety systems were designed.</p> <p>Failure of the actions listed in FR-S.1 to trip the Reactor include actions in the Main Control Room and in other areas of the plant.</p> <p>Although there are additional capabilities (i.e., emergency boration) to bring the plant under control, the indication of a Core Cooling Red indicates these capabilities are not effective and are a precursor for a core melt sequence.</p> <p>In addition, the challenge to the Steam Generators in the early stages of the event (i.e., Heat Sink Red) indicates insufficient feed water flow to remove heat and is also a precursor for a core melt sequence. It should be noted that this EAL is not applicable if actions of FR-H.1 are not implemented due to Operator ability to control Aux Feedwater >410 gpm.</p> <p>In either situation, if these challenges exist at a time that the reactor has not been brought below 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.</p>
<i>Escalation</i>	Not Applicable
<i>References</i>	NUMARC/NESP-007, SG2, Rev. 2, 1/92 T.S. 3.3.1 Reactor Trip System (RTS) Instrumentation FR-S.1 Nuclear Power Generation/ATWS FR-C.1 Inadequate Core Cooling FR-H.1 Loss of Secondary Heat Sink

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<i>Section</i>	2.0	SYSTEM DEGRADATION
<i>Event</i>	2.3	FAILURE OF RX PROTECTION
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		1,2
<i>Description</i>		<p>Reactor power <u>Not</u> <5% and decreasing after VALID Auto and Manual Trip signals (1 and 2 and 3)</p> <ol style="list-style-type: none"> VALID RX Auto Trip signal received or required. Manual RX Trip from the MCR was <u>Not</u> successful. FR-S.1 has been entered.
<i>Basis</i>		<p>This IC indicates a failure of the automatic and main control room manual signals to scram the reactor.</p> <p>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 anticipatory to the Fission Product Barrier Degradation IC, its inclusion is necessary to better assure timely recognition and emergency response.</p> <p>FR-S.1 lists actions intended to shutdown the reactor. This includes actions in the main control room and in other areas of the plant. FR-S.1 is utilized within the EAL to discriminate between those situations in which immediate manual reactor trip was not possible from the control room. The Unit 1 control room has two trip control locations on the main control board. Both are within immediate access for the reactor operator. If both fail to result in a reactor trip EOP E-0 directs the operator to FR-S.1.</p> <p>An indication or report or condition is considered to be VALID when it is conclusively verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel.. Implicit in this definition is the need for timely assessment, i.e., within 15 minutes.</p>
<i>Escalation</i>		Escalation of this event would be based on the inability to trip the Rx and indications of Heat Sink Red or Core Cooling Red.
<i>References</i>		NUMARC/NESP-007, SS2, Rev. 2, 1/92 T.S. 3.3.1 Reactor Trip System (RTS) Instrumentation FR-S.1 Nuclear Power Generation/ATWS FR-H.1, Loss of Secondary Heat Sink

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<i>Section</i>	2.0	SYSTEM DEGRADATION
<i>Event</i>	2.3	FAILURE OF RX PROTECTION
<i>Classification</i>		ALERT
<i>Mode</i>		1,2
<i>Description</i>		<p>Automatic RX trip did <u>NOT</u> occur after VALID Trip signal and manual trip from MCR was successful. (1 and 2)</p> <ol style="list-style-type: none"> VALID Rx Trip signal received or required Manual RX Trip from the MCR <u>was</u> successful and power is <5% and decreasing.
<i>Basis</i>		<p>This EAL indicates failure of the Reactor Protection System (RPS) to automatically trip the reactor. This condition is a potential degradation of a safety system in that a primary front line automatic protection system did not function in response to a plant transient or condition requiring system actuation. There are analyzed transients (e.g., MSLB) for which the timing of the reactor trip is essential to the safe response of the plant. The importance of this timing of the reactor trip is essential to the safe response of the plant. The importance of this timing is evidenced by the technical specifications governing protective system response.</p> <p>If an automatic reactor trip and the manual trip from the MCR is successful, the event would be classified as an Alert, with further escalation based on the higher events in this tab, or on the basis of fission product barriers.</p> <p>As a result of the manual trip, the reactor is producing less heat than the maximum decay heat load for which the safety systems are designed. On the long-term the plant can be brought to a safe shutdown. However, on the short-term, the power excursion may have caused localized fuel damage. In addition, the extent of the RPS failure and the impact on other plant controls and indication is not known. The Alert declaration will ensure that adequate resources, through staffing of the technical support center, are available to monitor and control plant systems such that any further degraded condition can be detected and responded to.</p> <p>FR-S.1 is not used in this EAL since reactor power is below 5% normally associated with the transition to FR-S.1.</p> <p>An indication or report or condition is considered to be VALID when it is conclusively verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel. Implicit in this definition is the need for timely assessment, i.e., within 15 minutes.</p>
<i>Escalation</i>		Escalation of this event would be based on the reactor power not being reduced to less than five percent by actions of FR-S.1.
<i>References</i>		NUMARC/NESP-007, SA2, Rev. 2, 1/92 T.S. 3.3.1 Reactor Trip System (RTS) Instrumentation FR-S.1 Nuclear Power Generation/ATWS WOG Background Document for FR-S.1, Rev. 1B, 2/92

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<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.3</i>	FAILURE OF RX PROTECTION
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		Not Applicable
<i>Description</i>		Not Applicable
<i>Basis</i>		Not Applicable
<i>Escalation</i>		Escalation of this event is based on a successful manual scram of the Rx from the main control room
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92, T.S. 3.3.1 Reactor Trip System Instrumentation FR-S.1 Nuclear Power Generation/ATWS PAI-2.04 Reactor/Turbine Trip Report

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<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.4</i>	FUEL CLAD DEGRADATION
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to the "Fission Product Barrier Matrix"
<i>Basis</i>		The basis for a General Emergency in this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.4</i>	FUEL CLAD DEGRADATION
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to the "Fission Product Barrier Matrix"
<i>Basis</i>		The basis for a Site Area Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Challenges".
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

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<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.4</i>	FUEL CLAD DEGRADATION
<i>Classification</i>		ALERT
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to the "Fission Product Barrier Matrix"
<i>Basis</i>		The basis for an Alert for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Challenges".
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

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<i>Section</i>	2.0	SYSTEM DEGRADATION
<i>Event</i>	2.4	FUEL CLAD DEGRADATION
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		1,2,3,4,5
<i>Description</i>		<p>Reactor Coolant System specific activity exceeds LCO (Refer to WBN Tech. Spec. 3.4.16) (1)</p> <p>1. Radiochemistry analysis indicates (a or b)</p> <p>a. Dose equivalent Iodine (I-131) >0.265 µCi/gm for >48 Hours <u>or</u> >14 µCi/gm</p> <p>b. Specific activity >100/ ĒµCi/gm</p>
<i>Basis</i>		<p>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. One (1) addresses coolant samples exceeding coolant Tech. Specs. for an Iodine Spike.</p> <p>The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) accident.</p> <p>The LCO contains specific activity limits for both Dose Equivalent I-131 and gross specific activity. The allowable levels are intended to limit the 2-hour dose at the site boundary to a small fraction of the 10 CFR 100 dose guideline values.</p> <p>The limits in the LCO are standardized and based on parametric evaluations of offsite radioactivity dose consequences for typical site locations.</p> <p>These parametric evaluations showed the potential offsite dose levels for a SGTR accident were an appropriately small fraction of the 10 CFR 100 guideline dose limits. Each evaluation assumes a broad range of site applicable atmospheric dispersion factors in a parametric evaluation.</p>
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Challenges".
<i>References</i>		NUMARC/NESP-007, SU4, Rev. 2, 1/92 T.S. 3.4.16 RCS Specific Activity, Bases

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<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.5</i>	RCS UNIDENTIFIED LEAKAGE
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to the "Fission Product Barrier Matrix"
<i>Basis</i>		The basis for a General Emergency in this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.5</i>	RCS UNIDENTIFIED LEAKAGE
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to the "Fission Product Barrier Matrix"
<i>Basis</i>		The basis for a Site Area Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Challenges".
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.5</i>	RCS UNIDENTIFIED LEAKAGE
<i>Classification</i>		ALERT
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to the "Fission Product Barrier Matrix"
<i>Basis</i>		The basis for an Alert for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Challenges".
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

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<i>Section</i>	2.0	SYSTEM DEGRADATION
<i>Event</i>	2.5	RCS UNIDENTIFIED LEAKAGE
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		1,2,3,4,5
<i>Description</i>		<p>Unidentified <u>or</u> pressure boundary RCS leakage >10 GPM</p> <p>1. Unidentified or pressure boundary leakage (as defined by Tech. Specs.) >10 GPM as indicated below (a or b)</p> <p>a. 1-SI-68-32 results</p> <p>b. With RCS Temperature <u>and</u> PZR Level Stable, VCT level Dropping at a Rate >10 GPM</p> <p>NOTE: Applies to Mode 5 if RCS Pressurized</p>
<i>Basis</i>		<p>This IC is included as an Unusual Event because it may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. The 10 gpm value for the unidentified and pressure boundary leakage was selected as it is observable with normal control room indications.</p> <p>If the leak is not reactor coolant pressure boundary leakage, the true RCS, and the leak is isolable from the RCS, the leak should not be classified as RCS leakage. However, if the RHR system is tied to the RCS and is being used as the cooling system for the core, the RHR system is and should be considered RCS.</p> <p>Note: for the leak to be isolable, personnel must be able to operate the isolable valve(s), which isolate the leak (e.g., isolate letdown to stop a leak in the CVCS). If a leak of radioactive fluid is occurring, which is suspected to be RCS, and is not known to be isolable, this leakage should be classified as RCS leakages determined by the shift manager/Site Emergency Director.</p> <p>Only operating modes in which there is fuel in the reactor coolant system and the system is pressurized are specified.</p>
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Challenges".
<i>References</i>		NUMARC/NES-007, SU5, Rev. 2, 1/92 T.S. 3.4.13 RCS Operational Leakage AOI-6 Small Reactor System Leak 1-SI-68-32 Reactor Coolant System Water Inventory Balance

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<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.6</i>	RCS IDENTIFIED LEAKAGE
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to the "Fission Product Barrier Matrix"
<i>Basis</i>		The basis for a General Emergency in this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.6</i>	RCS IDENTIFIED LEAKAGE
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to the "Fission Product Barrier Matrix"
<i>Basis</i>		The basis for a Site Area Emergency in this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Challengers".
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.6</i>	RCS IDENTIFIED LEAKAGE
<i>Classification</i>		ALERT
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to the "Fission Product Barrier Matrix"
<i>Basis</i>		The basis for a Alert for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Challenges".
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

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<i>Section</i>	2.0	SYSTEM DEGRADATION
<i>Event</i>	2.6	RCS IDENTIFIED LEAKAGE
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		1,2,3,4,5
<i>Description</i>		<p>Identified RCS leakage >25 GPM</p> <p>1. Identified RCS leakage (as defined by Tech. Specs.) >25 GPM (a or b)</p> <p>a. I-SI-68-32 results</p> <p>b. Level rise in excess of 25 GPM total into PRT, RCDT or CVCS Holdup Tank</p> <p><i>NOTE: Applies to Mode 5 if RCS Pressurized</i></p>
<i>Basis</i>		<p>This IC is included as an Unusual Event because it may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. The <u>25</u> gpm value for the identified and pressure boundary leakage was selected as it is observable with normal control room indications. This IC is set at a higher value than unidentified due to the lesser significance of identified leakage in comparison to unidentified or pressure boundary leakage.</p> <p>If the leak is not reactor coolant pressure boundary leakage, the true RCS, and the leak is isolable from the RCS, the leak should not be classified as RCS leakage. However, if the RHR system is tied to the RCS and is being used as the cooling system for the core, the RHR system is and should be considered RCS.</p> <p>Note: for the leak to be isolable, personnel must be able to operate the isolable valve(s), which isolate the leak (e.g., isolate letdown to stop a leak in the CVCS). If a leak of radioactive fluid is occurring, which is suspected to be RCS, and is not known to be isolable, this leakage should be classified as RCS leakages determined by the shift manager/Site Emergency Director.</p> <p>Only operating modes in which there is fuel in the reactor coolant system and the system is pressurized are specified.</p>
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Challenges".
<i>References</i>		NUMARC/NESP-007, SU5, Rev. 2, 1/92 T.S. 3.4.13 RCS Operational Leakage AOI-6 Small Reactor System Leak TI.4,Part II, Plant Curve Book 1-SI-68-32 Reactor Coolant System Water Inventory Balance

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<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.7</i>	UNCONTROLLED COOLDOWN
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to the "Fission Product Barrier Matrix"
<i>Basis</i>		The basis for a General Emergency in this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.7</i>	UNCONTROLLED COOLDOWN
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to the "Fission Product Barrier Matrix"
<i>Basis</i>		The basis for a Site Area Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation of this event will be based on "Fission Product Barriers Challenges".
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

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<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.7</i>	UNCONTROLLED COOLDOWN
<i>Classification</i>		ALERT
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to the "Fission Product Barrier Matrix"
<i>Basis</i>		The basis for an Alert for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation of this event will be based on "Fission Product Barriers Challenges".
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.7</i>	UNCONTROLLED COOLDOWN
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		1,2,3
<i>Description</i>		<p>UNPLANNED rapid depressurization of the Main Steam System resulting in a rapid RCS cooldown <u>and</u> Safety Injection initiation (1 and 2)</p> <ol style="list-style-type: none"> 1. Rapid depressurization of Main Steam System (<675 psig) 2. Safety injection has initiated <u>or is</u> required
<i>Basis</i>		<p>For this IC a rapid depressurization could be caused by a Main Steam line break or feed Line break which results in rapid RCS cool down and safety injection.</p> <p>This EAL is therefore consistent with the definition of an Unusual Event and warrants declaration.</p> <p>Unplanned is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities.</p>
<i>Escalation</i>		Escalation of this event will be based on "Fission Product Barrier Challenges".
<i>References</i>		<p>NUMARC/NESP-007, HU5, Rev. 2, 1/92</p> <p>E-2 Faulted Steam Generator Isolation</p> <p>T.S. 3.3.2 Engineering Safety Features Activation System Instrumentation (ESFAS)</p> <p>WBN-OSG4-188, (O-05)</p>

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<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.8</i>	TURBINE FAILURE
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to the "Fission Product Barrier Matrix"
<i>Basis</i>		The basis for a General Emergency in this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.8</i>	TURBINE FAILURE
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to the "Fission Product Barrier Matrix"
<i>Basis</i>		The basis for a Site Area Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation of this event will be based on "Fission Product Barrier Challenges."
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

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Section	2.0	SYSTEM DEGRADATION									
Event	2.8	TURBINE FAILURE									
Classification		ALERT									
Mode		1,2,3									
Description		<p>Turbine Failure has generated PROJECTILES that cause VISIBLE DAMAGE to any area containing Safety Related equipment</p> <p>1. Turbine PROJECTILES have resulted in VISIBLE DAMAGE in any of the following areas:</p> <table><tr><td>Control Building</td><td>Diesel Generator Bldg</td></tr><tr><td>Auxiliary Building</td><td>RWST</td></tr><tr><td>Unit #1 Containment</td><td>Intake Pumping Station</td></tr><tr><td></td><td>CST</td></tr></table>		Control Building	Diesel Generator Bldg	Auxiliary Building	RWST	Unit #1 Containment	Intake Pumping Station		CST
Control Building	Diesel Generator Bldg										
Auxiliary Building	RWST										
Unit #1 Containment	Intake Pumping Station										
	CST										
Basis		<p>This IC is intended to address the threat to safety related equipment imposed by PROJECTILES generated by main turbine rotating component failures. The list of areas provided includes all areas containing safety-related equipment, their controls, and their power supplies. This EAL is, therefore, consistent with the definition of an ALERT in that if PROJECTILES have damaged or penetrated areas containing safety-related equipment the potential exists for substantial degradation of the level of safety of the plant.</p> <p>PROJECTILE: An object ejected, thrown, or launched towards a plant structure. The source of the projectile may be onsite or offsite. Damage is sufficient to cause concern regarding the integrity of the affected structure or the operability or reliability of safety equipment contained therein.</p> <p>VISIBLE DAMAGE: Damage to equipment that is readily observable without measurements, testing, or analyses. Damage is sufficient enough to cause concern regarding the continued operability or reliability of the affected safety structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included).</p> <p>It is noted that due to Watts Bar's Turbine configuration and the location of the safety related equipment, the probability of Turbine Projectiles causing damage to these areas is considered remote.</p> <p>In addition it is recognized that the Condensate Storage Tank (CST) is not considered to be safety related equipment at WBN but, it is added due to its support of other site safety systems.</p>									
Escalation		Escalation of this event will be based on "Fission Product Barrier Challenges".									
References		NUMARC/NESP-007, HA1, Rev. 2, 1/92 FSAR 3.5.1.3 Turbine Missiles									

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<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.8</i>	TURBINE FAILURE
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		1,2,3
<i>Description</i>		Turbine Failure results in Casing penetration 1. Turbine Failure which results in penetration of the Turbine Casing <u>or</u> Damage to Main Generator Seals
<i>Basis</i>		This IC is intended to address main turbine rotating component failures of sufficient magnitude to cause observable damage to the turbine casing or to the seals of the main turbine generator. Of major concern is the potential for leakage of combustible fluids, lubricating oils and gases (hydrogen cooling) to the plant environs. Actual fires and flammable gas build up are appropriately classified via other events. This EAL is consistent with the definition of an Unusual Event while maintaining the anticipatory nature desired and recognizing the risk to non-safety related equipment.
<i>Escalation</i>		Escalation of this event would be based on potential damage done by turbine PROJECTILES to safety related equipment.
<i>References</i>		NUMARC/NESP-007, HU1, Rev. 2, 1/92 FSAR 3.5.1.3 Turbine Missiles

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<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.9</i>	TECHNICAL SPECIFICATION
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Not Applicable
<i>Basis</i>		Technical Specifications is not applicable for a General Emergency
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.9</i>	TECHNICAL SPECIFICATION
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Not Applicable
<i>Basis</i>		Technical Specifications is not applicable for a Site Area Emergency
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

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<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.9</i>	TECHNICAL SPECIFICATION
<i>Classification</i>		ALERT
<i>Mode</i>		Not Applicable
<i>Description</i>		Not Applicable
<i>Basis</i>		Technical Specifications is not applicable for an Alert
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	<i>2.0</i>	SYSTEM DEGRADATION
<i>Event</i>	<i>2.9</i>	TECHNICAL SPECIFICATION
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p>Inability to reach required Shutdown within Tech. Spec. limits (1 and 2)</p> <ol style="list-style-type: none"> Any Tech. Spec. LCO Statement, requiring a Mode reduction, has been entered The Unit has not been placed in the required Mode within the time prescribed by the LCO Action Statement
<i>Basis</i>		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.
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, SU2, Rev. 2, 1/92 T.R. 3.0 Technical Requirements (TR) Applicability (T.R. 3.03)

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<i>Section</i> 2.0	SYSTEM DEGRADATION
<i>Event</i> 2.10	SAFETY LIMIT
<i>Classification</i>	GENERAL EMERGENCY
<i>Mode</i>	Not Applicable
<i>Description</i>	Not Applicable
<i>Basis</i>	Safety limit is not applicable for a General Emergency
<i>Escalation</i>	Not Applicable
<i>References</i>	NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i> 2.0	SYSTEM DEGRADATION
<i>Event</i> 2.10	SAFETY LIMIT
<i>Classification</i>	SITE AREA EMERGENCY
<i>Mode</i>	Not Applicable
<i>Description</i>	Not Applicable
<i>Basis</i>	Safety limit is not applicable for a Site Area Emergency
<i>Escalation</i>	Not Applicable
<i>References</i>	NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i> 2.0	SYSTEM DEGRADATION
<i>Event</i> 2.10	SAFETY LIMIT
<i>Classification</i>	ALERT
<i>Mode</i>	Not Applicable
<i>Description</i>	Not Applicable
<i>Basis</i>	Safety limit is not applicable for an Alert
<i>Escalation</i>	Not Applicable
<i>References</i>	NUMARC/NESP-007, Rev. 2, 1/92

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<i>Section</i> 2.0	SYSTEM DEGRADATION
<i>Event</i> 2.10	SAFETY LIMIT
<i>Classification</i>	UNUSUAL EVENT
<i>Mode</i>	1,2,3,4,5
<i>Description</i>	<p>Safety Limits have been Exceeded (1 or 2)</p> <ol style="list-style-type: none"> 1. The combination of thermal power, RCS temperature, and RCS pressure > safety limits as indicated by WBN Tech. Spec. Figure 2.1.1-1 "Reactor Core Safety Limits" 2. RCS/Pressurizer pressure exceeds safety limit (>2735 psig)
<i>Basis</i>	<p>This IC requires that specified acceptable fuel design limits must not be exceeded during steady-state operation, normal operational transients, and anticipated operational occurrences (AOOs). This is accomplished with a departure from nucleate boiling (DNB) design basis that corresponds to a 95% probability, at a 95% confidence level that DNB, will not occur and by requiring that fuel-centerline temperature stays below the melting temperature.</p> <p>The restrictions of this SL prevent overheating of the fuel and cladding, as well as possible cladding perforation that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady-state peak linear heat rate (LHR) below the level at which centerline fuel melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat-transfer coefficient is large and the cladding-surface temperature is slightly above the coolant-saturation temperature.</p> <p>Centerline fuel melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel-centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.</p> <p>Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat-transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding-water (zirconium-water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.</p> <p>This EAL is consistent with the definition of an Unusual Event and warrants declaration.</p>

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<i>Section</i> 2.0	SYSTEM DEGRADATION
<i>Event</i> 2.10	SAFETY LIMIT
<i>Classification</i>	UNUSUAL EVENT (continued)
<i>Mode</i>	1,2,3,4,5
<i>Escalation</i>	Not Applicable
<i>References</i>	NUMARC/NESP-007, SU2, Rev. 2, 1/92 B.2.1.1 Reactor Core Safety Limits (SLs)

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<i>Section</i>	<i>3.0</i>	LOSS OF POWER
<i>Event</i>	<i>3.1</i>	LOSS OF AC (Power Ops)
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p>Prolonged loss of Offsite <u>and</u> Onsite AC Power (1 and 2)</p> <ol style="list-style-type: none"> 1. 1A <u>and</u> 1B 6.9KV Shutdown Bds de-energized for >15 minutes 2. (a or b) <ol style="list-style-type: none"> a. Core Cooling Red <u>or</u> Orange b. Restoration of Either 1A <u>or</u> 1B 6.9KV Shutdown Bds is not likely within 4 Hours of Loss
<i>Basis</i>		<p>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 four hours to restore AC power was based on a site blackout coping analysis performed in conformance with 10 CFR 50.63 and Regulatory Guide 1.155, "Station Blackout", as available, with appropriate allowance for offsite emergency response. Although this IC is redundant to the Fission Product Barrier Degradation IC, its inclusion is necessary to better assure timely recognition and emergency response.</p> <p>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 is appropriate, based on a reasonable assessment of the event trajectory.</p> <p>The (15 minute) time duration was selected to exclude transient or momentary power losses.</p> <p>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 Site Emergency Director a reasonable idea of how quickly (s)he may need to declare a General Emergency based on two major considerations:</p> <ol style="list-style-type: none"> 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?

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<i>Section</i>	<i>3.0</i>	LOSS OF POWER
<i>Event</i>	<i>3.1</i>	LOSS OF AC (Power Ops)
<i>Classification</i>		GENERAL EMERGENCY (continued)
<i>Mode</i>		1,2,3,4
<i>Basis (continued)</i>	2.	<p>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?</p> <p>The indication of continuing core cooling degradation is based on Fission Product Barrier monitoring with particular emphasis on Emergency Director judgement as it relates to Imminent Loss or Potential Loss of fission product barriers and degraded ability to monitor fission product barriers.</p>
<i>Escalation</i>		Not Applicable
<i>Reference</i>		NUMARC/NESP-007, SG1, Rev 2, 1/92 FSAR 15.2.9 Loss of Offsite Power to the Station Auxiliaries FSAR 15.5.1 Environmental Consequences of a Postulated Loss of AC Power to Plant Auxiliaries T.S. 3.8.1 AC Sources, Operating T.S. 3.8.3 Diesel Fuel, and Lubrication Oil (Diesels) T.S. 3.8.9 Distribution Systems, Operating AOI-35 Loss of Offsite Power General Design Criteria 17, App. ,A 10 CFR 50 NUREG 1.155 Station Blackout

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<i>Section</i>	3.0	LOSS OF POWER
<i>Event</i>	3.1	LOSS OF AC (Power Ops)
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p>Loss of Offsite <u>and</u> Onsite AC power for >15 Minutes</p> <p>1. 1A <u>and</u> 1B 6.9KV Shutdown Bds de-energized for >15 minutes</p>
<i>Basis</i>		<p>The 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.</p> <p>The (15 minute) time duration was selected to exclude transient or momentary power Losses.</p>
<i>Escalation</i>		Prolonged loss of all offsite power and prolonged loss of all onsite power will, when combined with inadequate core cooling, result in an escalation of this event.
<i>References</i>		<p>NUMARC/NESP-007 SS1, Rev. 2, 1/92</p> <p>AOI-35 Loss of Offsite Power</p> <p>FSAR 15.2.9 Loss of Offsite Power to the Station Auxiliaries</p> <p>General Design Criteria 17, App. ,A 10 CFR 50</p> <p>FSAR 15.5.1 Environmental Consequences of a Postulated Loss of AC Power to Plant Auxiliaries</p> <p>T.S. 3.8.1 AC Sources, Operating</p> <p>T.S. 3.8.3 Diesel Fuel, and Lubrication Oil (Diesels)</p> <p>T.S. 3.8.9 Distribution Systems, Operating</p>

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<i>Section</i>	3.0	LOSS OF POWER
<i>Event</i>	3.1	LOSS OF AC (Power Ops)
<i>Classification</i>		ALERT
<i>Mode</i>		1,2,3,4
<i>Description</i>	<p>Loss of Offsite Power and 1A <u>or</u> 1B Diesel Generator (1 and 2)</p> <ol style="list-style-type: none"> 1. C <u>and</u> D CSSTs not available for >15 minutes 2. 1A <u>or</u> 1B Diesel Generator not available 	
<i>Basis</i>	<p>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 diesel generator to supply power to its emergency busses.</p> <p>The (15 minute) time duration was selected to exclude transient or momentary power losses.</p>	
<i>Escalation</i>	Prolonged Loss of all offsite power and prolonged Loss of all onsite power will escalate this event.	
<i>References</i>	<p>NUMARC/NESP-007,SA5, Rev. 2, 1/92 AOI-35 Loss of Offsite Power General Design Criterion 17 of App. A, 10 CFR 50 FSAR 15.2.9 Loss of Offsite Power to the Station Auxiliaries FSAR 15.5.1 Environmental Consequences of a Postulated Loss of AC Power to Plant Auxiliaries T.S. 3.8.1 AC Sources, Operating T.S. 3.8.3 Diesel Fuel, and Lubrication Oil (Diesels) T.S. 3.8.9 Distribution Systems, Operating</p>	

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<i>Section</i>	3.0	LOSS OF POWER
<i>Event</i>	3.1	LOSS OF AC (Power Ops)
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p>Loss of Offsite Power for >15 Minutes (1 and 2)</p> <ol style="list-style-type: none"> 1. C and D CSSTs not available for > 15 minutes 2. Each Diesel Generator is supplying power to its respective Shutdown Bd
<i>Basis</i>		<p>Prolonged loss of AC power reduces required redundancy and potentially degrades the level of safety of the plant by rendering the plant more vulnerable to a complete Loss of AC Power (Station Blackout).</p> <p>Fifteen (15) minutes was selected as a threshold to exclude transient or momentary power losses.</p>
<i>Escalation</i>		Loss of one additional power supply to the shutdown boards will escalate this event.
<i>References</i>		<p>NUMARC/NESP-007 SU1, Rev. 2, 1/92</p> <p>AOI-35 Loss of Offsite Power</p> <p>FSAR 15.2.9 Loss of Offsite Power to the Station Auxiliaries</p> <p>FSAR 15.5.1 Environmental Consequences of a Postulated Loss of AC Power to Plant Auxiliaries</p> <p>General Design Criterion 17 of App. A, 10 CFR 50</p> <p>T.S. 3.8.1 AC Sources, Operating</p> <p>T.S. 3.8.3 Diesel Fuel, and Lubrication Oil (Diesels)</p> <p>T.S. 3.8.9 Distribution Systems, Operating</p>

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<i>Section</i>	<i>3.0</i>	LOSS OF POWER
<i>Event</i>	<i>3.2</i>	LOSS OF AC (Shutdown)
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Not Applicable
<i>Basis</i>		Loss of AC Power in Mode 5 and 6 will Not cause a declaration of a General Emergency
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92

<i>Section</i>	<i>3.0</i>	LOSS OF POWER
<i>Event</i>	<i>3.2</i>	LOSS OF AC (Shutdown)
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Not Applicable
<i>Basis</i>		Loss of AC Power in Mode 5 and 6 will Not cause a declaration of a Site Area Emergency.
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92

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<i>Section</i>	3.0	LOSS OF POWER
<i>Event</i>	3.2	LOSS OF AC (Shutdown)
<i>Classification</i>		ALERT
<i>Mode</i>		5,6, or Defuel
<i>Description</i>		<p>UNPLANNED Loss of Offsite <u>And</u> Onsite AC power for >15 minutes</p> <p>1. 1A <u>and</u> 1B 6.9KV Shutdown Bds de-energized for >15 minutes</p> <p>Also Refer to "Loss of Shutdown Systems" (6.1)</p>
<i>Basis</i>		<p>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 this event is classified as an Alert, because of the significantly reduced decay heat and lower temperature and pressure, increasing the time to restore one of the emergency busses. Fifteen (15) minutes was selected as a threshold to exclude transient or momentary power losses.</p> <p>Unplanned is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities.</p>
<i>Escalation</i>		Escalation is not applicable from this event.
<i>References</i>		<p>NUMARC/NESP-007 SU1 (expanded), Rev 2, 1/92</p> <p>T.S. 3.8.2 AC Sources - Shutdown</p> <p>T.S. 3.8.10 Distribution Systems - Shutdown</p>

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<i>Section</i>	3.0	LOSS OF POWER
<i>Event</i>	3.2	LOSS OF AC (Shutdown)
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		5,6, Defuel
<i>Description</i>		<p>UNPLANNED Loss of Offsite Power for >15 minutes (1 and 2)</p> <ol style="list-style-type: none"> 1. C and D CSSTs not available for >15 minutes 2. Either Diesel Generator is supplying power to its respective Shutdown Board
<i>Basis</i>		<p>Prolonged loss of AC power reduces required redundancy and potentially degrades the level of safety of the plant by rendering the plant more vulnerable to a complete Loss of AC Power (Station Blackout). Fifteen (15) minutes was selected as a threshold to exclude transient or momentary power losses.</p> <p>Unplanned is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities.</p>
<i>Escalation</i>		Loss of one additional power supply to the shutdown boards will escalate this event.
<i>References</i>		<p>NUMARC/NESP-007, SU1, Rev 2, 1/92</p> <p>T.S. 3.8.2 AC Sources - Shutdown</p> <p>T.S. 3.8.10 Distribution Systems - Shutdown</p>

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<i>Section</i>	<i>3.0</i>	LOSS OF POWER
<i>Event</i>	<i>3.3</i>	LOSS OF DC
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to "Fission Product Barrier Matrix" and "Loss of Function" (2.2)
<i>Basis</i>		The basis for a General Emergency in this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Refer to the "Fission Product Barrier Matrix" or " Loss of Function" (2.2) for escalation considerations.
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92

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<i>Section</i>	3.0	LOSS OF POWER
<i>Event</i>	3.3	LOSS OF DC
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		1,2,3,4
<i>Description</i>		<p>Loss of All Vital DC Power for >15 minutes</p> <p>1. Voltage <105 VDC on 125V DC Vital Battery Buses 1-I <u>and</u> 1-II <u>and</u> 1-III <u>and</u> 1-IV for >15 minutes</p> <p>Also Refer to the "Fission Product Barrier Matrix", "Loss of Function" (2.2), and "Loss of Instrumentation" (2.1)</p>
<i>Basis</i>		<p>Loss of all DC power compromises the ability to monitor and control plant safety functions. Prolonged loss of all DC power will cause core uncovering and loss of containment integrity when there is significant decay heat and sensible heat in the reactor system.</p> <p>Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.</p> <p>The minimum specified independent and redundant DC power sources and distribution systems satisfy the requirements of General Design Criterion 17 of Appendix "A" to 10 CFR 50.</p>
<i>Escalation</i>		Escalation would occur through the Fission Product Barrier Matrix Degradation or Loss of Function (2.2)
<i>References</i>		<p>NUMARC/NESP-007, SS3, Rev. 2, 1/92</p> <p>General Design Criteria 17, App. A 10 CFR 50</p> <p>FSAR 8.3.2 DC Power System</p> <p>T.S. 3.8.4 DC Sources - Operating</p> <p>AOI - 21.01 - 21.04 Loss of 125V DC Vital Battery Boards</p>

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<i>Section</i>	3.0	LOSS OF POWER
<i>Event</i>	3.3	LOSS OF DC
<i>Classification</i>		ALERT
<i>Mode</i>		1,2,3,4
<i>Description</i>		Refer to "Fission Product Barrier Matrix", "Loss of Function" (2.2), and "Loss of Instrumentation" (2.1)
<i>Basis</i>		There is <u>NO</u> Alert classification for this event. Reference should be made to the "Fission Product Barrier Matrix", "Loss of Function" (2.2), or "Loss of Instrumentation" (2.1) for possible Alert or higher classifications.
<i>Escalation</i>		Loss of All vital DC Power for greater than 15 minutes or the Inability to monitor a Significant Transient in Progress or Loss of Function needed to Achieve or Maintain Hot Shutdown may cause an escalation in a loss of DC event.
<i>References</i>		NUMARC/NESP-007, SU7, Rev. 2, 1/92 General Design Criteria 17, App. A, 10 CFR 50 T.S. 3.8.4 DC Sources - Operating AOI-21.1-21.4 Loss of 125V DC Vital Battery Boards FSAR 8.3.2 DC Power Systems

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<i>Section</i>	3.0	LOSS OF POWER
<i>Event</i>	3.3	LOSS OF DC
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		5,6 and Defuel
<i>Description</i>		<p>UNPLANNED Loss of the Required Train of DC power for >15 Minutes (1 or 2)</p> <ol style="list-style-type: none"> 1. Voltage <105 VDC on 125V Vital Battery Buses 1-I <u>and</u> 1-III for >15 Minutes 2. Voltage <105 VDC on 125V DC Vital Battery Buses 1-II <u>and</u> 1-IV for >15 Minutes
<i>Basis</i>		<p>The purpose of this IC 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 IC 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.</p> <p>Unplanned is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities.</p> <p>The 105 volt Bus Voltage is the minimum bus voltage necessary for the operation of safety related equipment. This voltage value should incorporate a margin of at least 15 minutes of operation before the onset of inability to operate those loads. 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. For a 58 string battery set the minimum voltage is typically 1.81 Volts per cell.</p> <p>The fifteen minute threshold is utilized to exclude a transient or momentary power losses.</p>
<i>Escalation</i>		The event will escalate if the DC loss results in an inability to maintain cold shutdown.
<i>References</i>		NUMARC/NESP-007, SU7, Rev. 2, 1/92 FSAR 8.3.2 DC Power Sources T.S. 3.8.5 DC Sources Shutdown T.S. 3.8.10 Distribution Systems-Shutdown AOI-21.1-21.4 Loss of 125V DC Vital Battery Boards

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<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGEMENT
<i>Event</i>	<i>4.1</i>	FIRE
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to "Fission Product Barrier Matrix"
<i>Basis</i>		The basis for a General Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007 Rev. 2, 1/92

<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGEMENT
<i>Event</i>	<i>4.1</i>	FIRE
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to "Control Room Evacuation," (4.5) or "Fission Product Barrier Matrix"
<i>Basis</i>		<p>The basis for a Site Area Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.</p> <p>In addition the seriousness of a Fire in the Control Room requires reference to the emergency conditions identified in Section (4.5) "Control Room Evacuation"</p>
<i>Escalation</i>		Escalation would be based on "Fission Product Barrier Challenges"
<i>References</i>		NUMARC/NESP-007 Rev. 2, 1/92

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<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGEMENT
<i>Event</i>	<i>4.1</i>	FIRE
<i>Classification</i>		ALERT
<i>Mode</i>		All
<i>Description</i>		<p>FIRE in any of the areas listed in Table 4-1 That is Affecting Safety Related Equipment (1 and 2)</p> <ol style="list-style-type: none"> 1. FIRE in any of the areas listed in Table 4-1 2. (a or b) <ol style="list-style-type: none"> a. VISIBLE DAMAGE to permanent structure <u>or</u> Safety Related equipment in the specified area is observed due to the FIRE b. Control Room indication of degraded Safety System <u>or</u> component response due to the FIRE
<i>Basis</i>		<p>Fires that are likely to affect the plant's safety systems represent a degraded plant condition. The fire may have damaged equipment or damage is likely due to the proximity of heat, or flame to the systems required for safe shutdown. The likelihood of damage is subjective but is based on fire location, intensity and duration without performance of a detailed damage assessment prior to classification. The determination of the safety and supporting systems necessary for safe shutdown during the applicable operating mode and the assessment of the impact of the fire on the performance of those systems will be determined by the Site Emergency Director.</p> <p>Table 4-1 Plant Structures Associated with Fire and Explosion EALs</p> <ul style="list-style-type: none"> Unit #1 Reactor Building Auxiliary Building Control Building Diesel Generator Building CST Additional Diesel Generator Building Intake Pumping Station Additional Equipment Bldgs (Unit 1 & 2) RWST Turbine Building <p>FIRE is combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical components do not constitute fires. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.</p>

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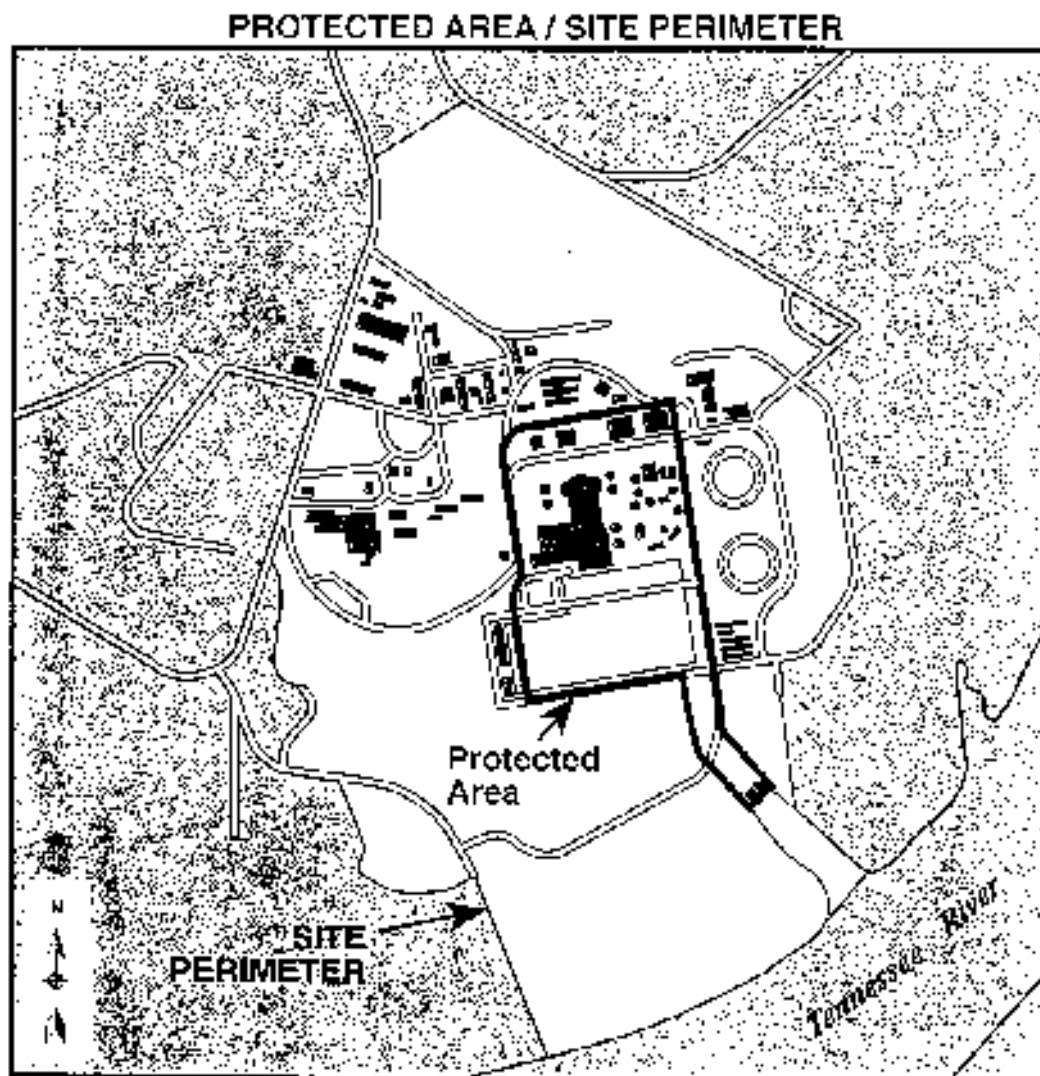
<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGEMENT
<i>Event</i>	<i>4.1</i>	FIRE
<i>Classification</i>		ALERT (continued)
<i>Mode</i>		All
<i>Basis</i> (continued)		VISIBLE DAMAGE is damage to equipment that is readily observable without measurements, testing, or analyses. Damage is sufficient enough to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering. Surface blemishes (e.g., paint chipping, scratches) should NOT be included.
<i>Escalation</i>		Escalation would be based on Fission Product Barrier challenges or Control Room Evacuation (4.5)
<i>References</i>		NUMARC/NESP-007, HA2, Rev. 2, 1/92 Figure 4-A Protected Area and Site Perimeter

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<i>Section</i>	4.0	HAZARDS AND SED JUDGEMENT
<i>Event</i>	4.1	FIRE
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		All
<i>Description</i>		FIRE in the PROTECTED AREA Threatening any of the Areas Listed in Table 4-1 that is <u>Not</u> Extinguished within the 15 Minutes From the Time of Control Room Notification <u>or</u> Verification of Control Room Alarm
<i>Basis</i>		<p>This event covers verified fires that occur in selected areas of the plant that house safety systems. It also covers verified fires outside of these areas that may impact structures that contain safety systems due to the proximity of the fire. In either case these fires may be potentially significant precursors to damage of safety systems or may impact structures that contain safety systems. The initiating condition excludes fires that occur outside these key buildings, such as the warehouses, or other small fires that do not potentially affect safety systems.</p> <p>The 15 minute time limit has been established to exclude small fires that can be controlled by Plant Fire Fighting resources.</p> <p>Verification of the fire in this event is either by direct communication with plant personnel confirming that a fire exists or the action taken by the Control Room personnel to determine that a fire annunciator received in the Control Room is not due to a spurious signal.</p> <p>Table 4-1 Plant Structures Associated with Fire and Explosion EALs</p> <ul style="list-style-type: none"> Unit #1 Reactor Building Auxiliary Building Control Building Diesel Generator Building CST Additional Diesel Generator Building Intake Pumping Station Additional Equipment Bldgs (Unit 1 & 2) RWST Turbine Building

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Figure 4-A



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<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGEMENT
<i>Event</i>	<i>4.1</i>	FIRE
<i>Classification</i>		UNUSUAL EVENT (continued)
<i>Mode</i>		All
<i>Basis</i> (continued)		<p>FIRE is combustion characterized by heat and light. Source of smoke such as slipping drive belts or overheated electrical components do not constitute fires. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.</p> <p>PROTECTED AREA encompasses all owner controlled areas within the security protected area fence as shown on Figure 4-A.</p>
<i>Escalation</i>		Escalation of this event is based on the Fire affecting plant safety related equipment required to establish or maintain safe shutdown.
<i>References</i>		NUMARC/NESP-007, HU2, Rev. 2, 1/92 Figure 4-A Protected Area/Site Perimeter

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<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGEMENT
<i>Event</i>	<i>4.2</i>	EXPLOSIONS
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to the "Fission Product Barrier Matrix"
<i>Basis</i>		The basis for a General Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007 Rev. 2, 1/92

<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGEMENT
<i>Event</i>	<i>4.2</i>	EXPLOSIONS
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to the "Fission Product Barrier Matrix"
<i>Basis</i>		The basis for a Site Area Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation would be based on "Fission Product Barrier Challenges".
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

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<i>Section</i>	4.0	HAZARDS AND SED JUDGEMENT
<i>Event</i>	4.2	EXPLOSIONS
<i>Classification</i>		ALERT
<i>Mode</i>		All
<i>Description</i>		<p>EXPLOSION in any of the areas listed in Table 4-1 that is affecting Safety Related equipment (1 and 2)</p> <ol style="list-style-type: none"> 1. EXPLOSION in any of the areas listed in Table 4-1 2. (a or b) <ol style="list-style-type: none"> a. An EXPLOSION has caused VISIBLE DAMAGE to Safety Related equipment b. Control Room indication of degraded Safety System <u>or</u> component response due to the EXPLOSION <p>Refer to Security (4.6)</p>
<i>Basis</i>		<p>EXPLOSIONS include those that are of sufficient magnitude to damage permanent structures or equipment within the plant vital area. As used here, an EXPLOSION is a rapid, violent, unconfined combustion, or a catastrophic failure of pressurized equipment, that potentially imparts significant energy to near-by structures and material.</p> <p>VISIBLE DAMAGE is damage to equipment that is readily observable without measurements, testing, or analyses. Damage is sufficient enough to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering. Surface blemishes (e.g., paint chipping, scratches) should NOT be included.</p> <p>The "Report of VISIBLE DAMAGE" should not be interpreted as requiring a lengthy damage assessment prior to classification.</p> <p>The observation of damage to a structure is sufficient to make a declaration. The declaration of the Alert and the activation of the TSC is warranted and will provide the Site Emergency Director (SED) with resources necessary to perform damage assessment.</p>

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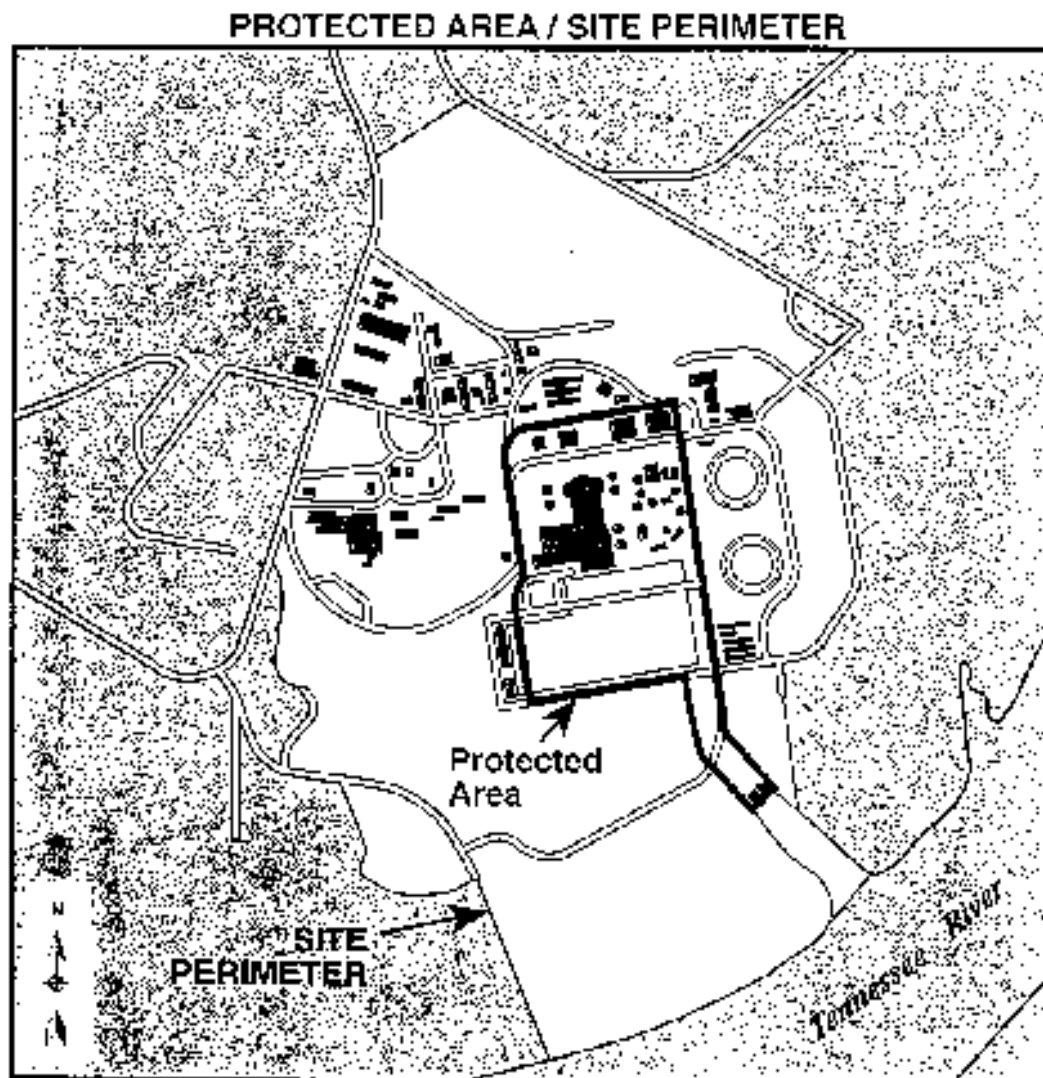
<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGEMENT
<i>Event</i>	<i>4.2</i>	EXPLOSIONS
<i>Classification</i>		ALERT (continued)
<i>Mode</i>		All
<i>Basis</i> (continued)		<p>Table 4-1 Plant Structures Associated with Fire and Explosion EALs</p> <p>Unit #1 Reactor Building Auxiliary Building Control Building Diesel Generator Building CST Additional Diesel Generator Building Intake Pumping Station Additional Equipment Bldgs (Unit 1 & 2) RWST Turbine Building</p>
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Challenges".
<i>References</i>		NUMARC/NESP-007, HA2, Rev 2, 1/92

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<i>Section</i>	4.0	HAZARDS AND SED JUDGEMENT
<i>Event</i>	4.2	EXPLOSIONS
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		All
<i>Description</i>		<p>UNPLANNED EXPLOSION Within the PROTECTED AREA Resulting in VISIBLE DAMAGE to Any Permanent Structure <u>or</u> Equipment (Figure 4-A)</p> <p>Refer to Security (4.6)</p>
<i>Basis</i>		<p>As used here, an EXPLOSION is a rapid, violent, unconfined combustion, or a catastrophic failure of pressurized equipment, that potentially imparts significant energy to near-by structures and material. For this event classification, the occurrence of the EXPLOSION is sufficient to make the declaration without making a lengthy assessment of the damage.</p> <p>In addition, certain hazardous materials are transported by river barge past the Watts Bar Nuclear Plant site. Explosive materials are also transported over nearby railroad lines. Therefore, these materials were evaluated for their potential to damage the safety related structures of the plant. The materials include TNT, gasoline, liquid natural gas (LNG) and unspecified fertilizers.</p> <p>There is no potential for damage to the Watts Bar plant due to the transport of TNT from or storage of TNT at the TVA plant. The potential for damage to the Watts Bar plant from a gasoline barge explosion is considered to be negligible. It should be noted that barge shipments of LNG past Watts Bar are rare since natural gas transportation is handled almost entirely by pipeline in this region. Therefore, the potential for an exploding LNG barge near the Intake Pumping Station is a non-credible event.</p> <p>Given the low probability of a barge collision and the low percentage of fertilizer shipments on the Tennessee River, it is concluded that, because of the very low probabilities associated with the event, no hazard exists to the Intake Pumping Station from the transportation of fertilizers by barge on the Tennessee River system.</p> <p>Unplanned is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities.</p> <p>PROTECTED AREA encompasses all owner controlled areas within the security protected area fence as shown on Figure 4-A.</p>
<i>Escalation</i>		Escalation of this event would be based on EXPLOSION damage to a structure or equipment causing a degradation in the performance of equipment required to shutdown the plant.
<i>References</i>		<p>NUMARC/NESP-007, HU2, Rev 2, 1/92</p> <p>FSAR 2.2 Nearby Industrial, Transportation and Military Facilities</p>

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Figure 4-A



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<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGEMENT
<i>Event</i>	<i>4.3</i>	FLAMMABLE GAS
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to the "Fission Product Barrier Matrix"
<i>Basis</i>		The basis for a General Emergency in this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92

<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGEMENT
<i>Event</i>	<i>4.3</i>	FLAMMABLE GAS
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to the "Fission Product Barrier Matrix"
<i>Basis</i>		The basis for a Site Area Emergency in this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Challenges".
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92

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<i>Section</i>	4.0	HAZARDS AND SED JUDGEMENT
<i>Event</i>	4.3	FLAMMABLE GAS
<i>Classification</i>		ALERT
<i>Mode</i>		All
<i>Description</i>		<p>UNPLANNED release of Flammable Gas within a facility structure containing Safety Related equipment <u>or</u> associated with Power production.</p> <p>1. Plant personnel report the average of three readings taken in a ~10ft Triangular Area is > 25% (LEL) Lower Explosive Limit as indicated on the monitoring instrument within any building listed in Table 4-2</p>
<i>Basis</i>		<p>Report or detection of flammable gases within plant vital structures in concentrations that are life threatening to plant personnel or affect the ability to achieve or maintain the plant in a cold shutdown condition is a degradation of the level of safety of the plant and warrants the declaration of an Alert.</p> <p>Table 4-2 Plant Structures Associated with Toxic or Flammable Gas EALs</p> <ul style="list-style-type: none"> Unit #1 & 2 Reactor Buildings Auxiliary Building Control Building Diesel Generator Building Additional Diesel Generator Building Intake Pumping Station Additional Equipment Bldgs (Unit 1 & 2) CDWE Building Turbine Building <p>Unplanned is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities.</p>
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Challenges".
<i>References</i>		NUMARC/NESP-007, HA3, Rev 2, 1/92 Figure 4-B One Mile Radius/Site Perimeter

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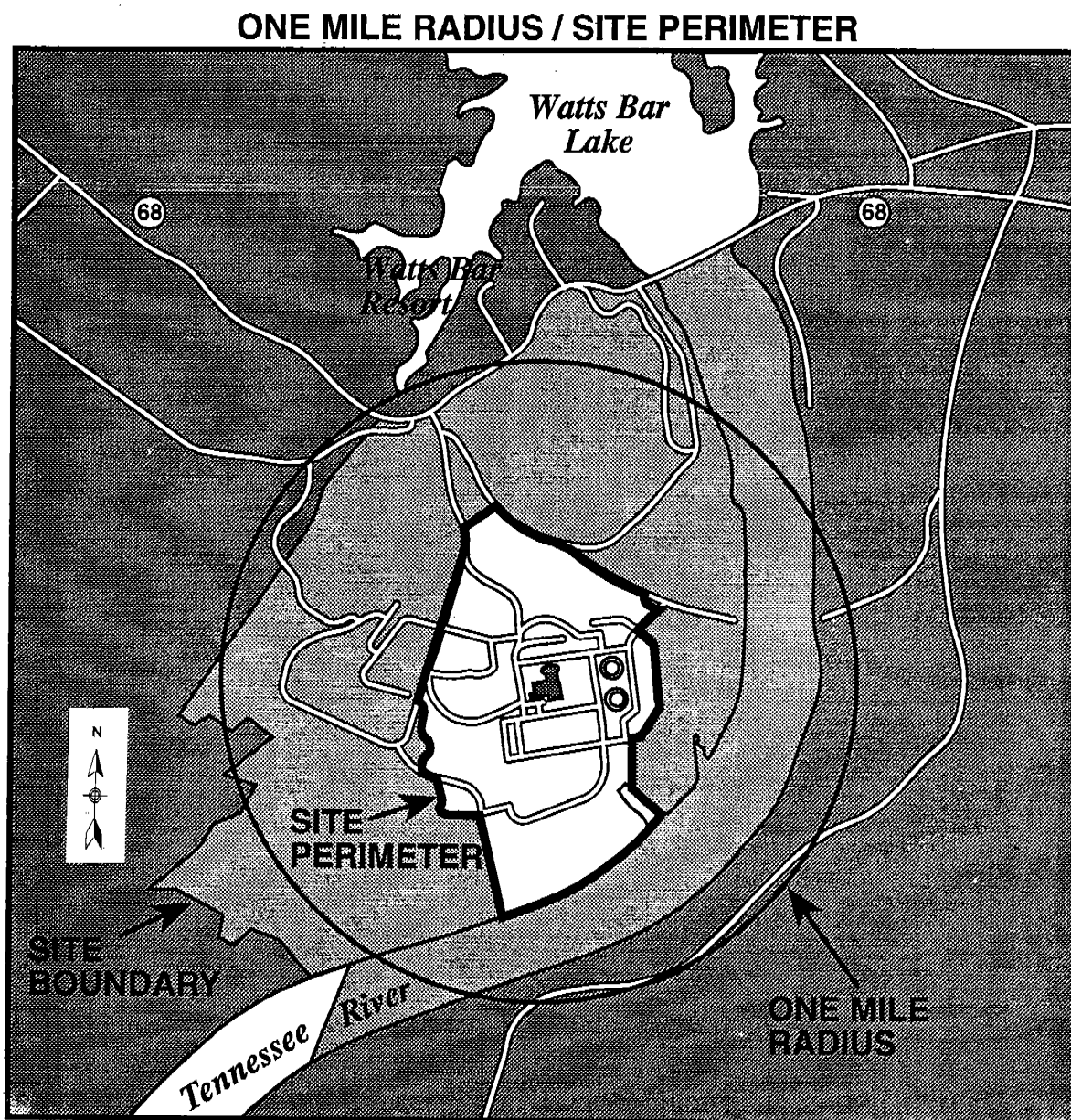
<i>Section</i>	4.0	HAZARDS AND SED JUDGEMENT
<i>Event</i>	4.3	FLAMMABLE GAS
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		All
<i>Description</i>		<p>A. UNPLANNED release of Flammable Gas within the SITE PERIMETER.</p> <p>1. Plant personnel report the average of three readings taken in a ~10ft Triangular Area is > 25% (LEL) Lower Explosive Limit as indicated on the monitoring instrument within the SITE PERIMETER (Refer to Figure 4-B)</p> <p style="text-align: center;"><u>OR</u></p> <p>B. Confirmed report by Local, County, or State Officials That a Large Offsite Flammable Gas release has occurred within One Mile of the Site with potential to enter the SITE PERIMETER in concentrations >25% of LEL (Lower Explosive Limit). (Refer to Figure 4-B)</p>
<i>Basis</i>		<p>Report or detection of flammable gases in concentrations within the site or near the site that will affect the health of plant personnel or affect the safe operation of the plant with the plant being within the evacuation area of an offsite event (i.e., tanker truck accident releasing flammable gases, etc.) constitutes an Unusual Event. The evacuation area is as determined from the DOT Evacuation Tables for Selected Hazardous Materials, in the DOT Emergency Response Guide for Hazardous Materials.</p> <p>In addition, it should be noted that there are no industrial or military facilities where large quantities of flammable or toxic chemicals are stored within a five mile radius of the plant. The shipping on the Tennessee River consists mainly of fuel oils, wood products and minerals. Chemicals represent only a minor percentage of the barge shipping by the Watts Bar Nuclear Plant. The release of flammable or toxic materials on the river in the vicinity of the plant will have minimal effect on the plant safety features.</p> <p>The main control room habitability during postulated hazardous chemical releases at or near the plant has been evaluated. This evaluation utilizes the approach outlined in Regulatory Guide 1.78 and concludes that the main control room habitability is not jeopardized by accidental release of chemicals. In addition, plant procedures maintain a list of onsite hazardous materials, their storage facilities, and quantities they are stored in.</p>

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<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGEMENT
<i>Event</i>	<i>4.3</i>	FLAMMABLE GAS
<i>Classification</i>		UNUSUAL EVENT (continued)
<i>Mode</i>		All
<i>Basis</i> (continued)		<p>Table 4-2 Plant Structures Associated with Toxic or Flammable Gas EALs</p> <ul style="list-style-type: none"> Unit #1 & 2 Reactor Buildings Auxiliary Building Control Building Diesel Generator Building Additional Diesel Generator Building Intake Pumping Station Additional Equipment Bldgs (Unit 1 & 2) CDWE Building Turbine Building <p>Unplanned is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities.</p> <p>SITE PERIMETER encompasses all owner controlled areas in the immediate site environs as shown on Figure 4-B and 7-A.</p>
<i>Escalation</i>		Escalation of this event would be based on flammable gases entering a plant area that jeopardizes life or impacts cold shutdown capabilities.
<i>References</i>		NUMARC/NESP-007, HU3, Rev 2, 1/92 FSAR 2.2 Nearby Industrial, Transportation and Military Facilities DOT Emergency Response Guide for Hazardous Materials

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Figure 4-B



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<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGEMENT
<i>Event</i>	<i>4.4</i>	TOXIC GAS
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to the "Fission Product Barrier Matrix"
<i>Basis</i>		The basis for a General Emergency in this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92

<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGEMENT
<i>Event</i>	<i>4.4</i>	TOXIC GAS
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to the "Fission Product Barrier Matrix"
<i>Basis</i>		The basis for a Site Area Emergency in this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92

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<i>Section</i>	4.0	HAZARDS AND SED JUDGEMENT
<i>Event</i>	4.4	TOXIC GAS
<i>Classification</i>		ALERT
<i>Mode</i>		All
<i>Description</i>	<p>Release of TOXIC GAS within a facility structure which Prohibits Safe Operation of systems required to establish <u>or</u> maintain Cold S/D (1 and 2 and 3)</p> <ol style="list-style-type: none"> 1. Plant personnel report TOXIC GAS within any building listed in Table 4-2 2. (a or b) <ol style="list-style-type: none"> a. Plant personnel report Severe Adverse Health Reactions due to TOXIC GAS (i.e., burning eyes, nose, throat, dizziness) b. Sampling indications > Permissible Exposure Limit (PEL) 3. Plant personnel would be unable to perform actions necessary to establish and maintain Cold Shutdown while utilizing appropriate personnel protection equipment 	
<i>Basis</i>	<p>Report or detection of toxic gases within plant vital structures in concentrations that are life threatening to plant personnel or affect the ability to achieve or maintain the plant in a cold shutdown condition is a degradation of the level of safety of the plant and warrants the declaration of an Alert.</p> <p>Table 4-2 Plant Structures Associated with Toxic or Flammable Gas EALs</p> <ul style="list-style-type: none"> Unit #1 & 2 Reactor Buildings Auxiliary Building Control Building Diesel Generator Building Additional Diesel Generator Building Intake Pumping Station Additional Equipment Bldgs (Unit 1 & 2) CDWE Building Turbine Building <p>TOXIC GAS is a gas that is dangerous to life or limb by reason of inhalation or skin contact (e.g., chlorine).</p>	
<i>Escalation</i>	Escalation will be based on "Fission Product Barrier Challenges".	
<i>References</i>	NUMARC/NESP-007, HA3, Rev 2, 1/92 Figure 4-B One Mile Radius/Site Perimeter	

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<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGEMENT
<i>Event</i>	<i>4.4</i>	TOXIC GAS
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		All
<i>Description</i>	<p>A. Normal Operations impeded due to access restrictions caused by TOXIC GAS concentrations within a Facility Structure Listed in Table 4-2</p> <p style="text-align: center;"><u>OR</u></p> <p>B. Confirmed report by Local, County, <u>or</u> State Officials that a Large Offsite TOXIC GAS release has occurred within One Mile of the Site with potential to enter the SITE PERIMETER in Concentrations > than the Permissible Exposure Limit (PEL) thus causing an Evacuation (Figure 4-B)</p>	
<i>Basis</i>	<p>Report or detection of a release of toxic gases in concentrations within the site or near the site perimeter that will affect the health of plant personnel or affect 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.) constitutes an Unusual Event. The evacuation area is as determined from the DOT Evacuation Tables for Selected Hazardous Materials, in the DOT Emergency Response Guide for Hazardous Materials.</p> <p>In addition, it should be noted that there are no industrial or military facilities where large quantities of flammable or toxic chemicals are stored within a five mile radius of the plant. The shipping on the Tennessee River consists mainly of fuel oils, wood products and minerals. Chemicals represent only a minor percentage of the barge shipping by the Watts Bar Nuclear Plant. The release of flammable or toxic materials on the river in the vicinity of the plant will have minimal effect on the plant safety features.</p> <p>The main control room habitability during a postulated hazardous chemical releases at or near the plant has been evaluated. This evaluation utilizes the approach outlined in Regulatory Guide 1.78 and concludes that the main control room habitability is not jeopardized by an accidental release of chemicals. In addition, plant procedures maintain a list of onsite hazardous materials, their storage facilities, and quantities they are stored in.</p> <p>TOXIC GAS is a gas that is dangerous to life or limb by reason of inhalation or skin contact (e.g., chlorine).</p> <p>SITE PERIMETER encompasses all owner controlled areas in the immediate site environs as shown on Figure 4-A.</p>	

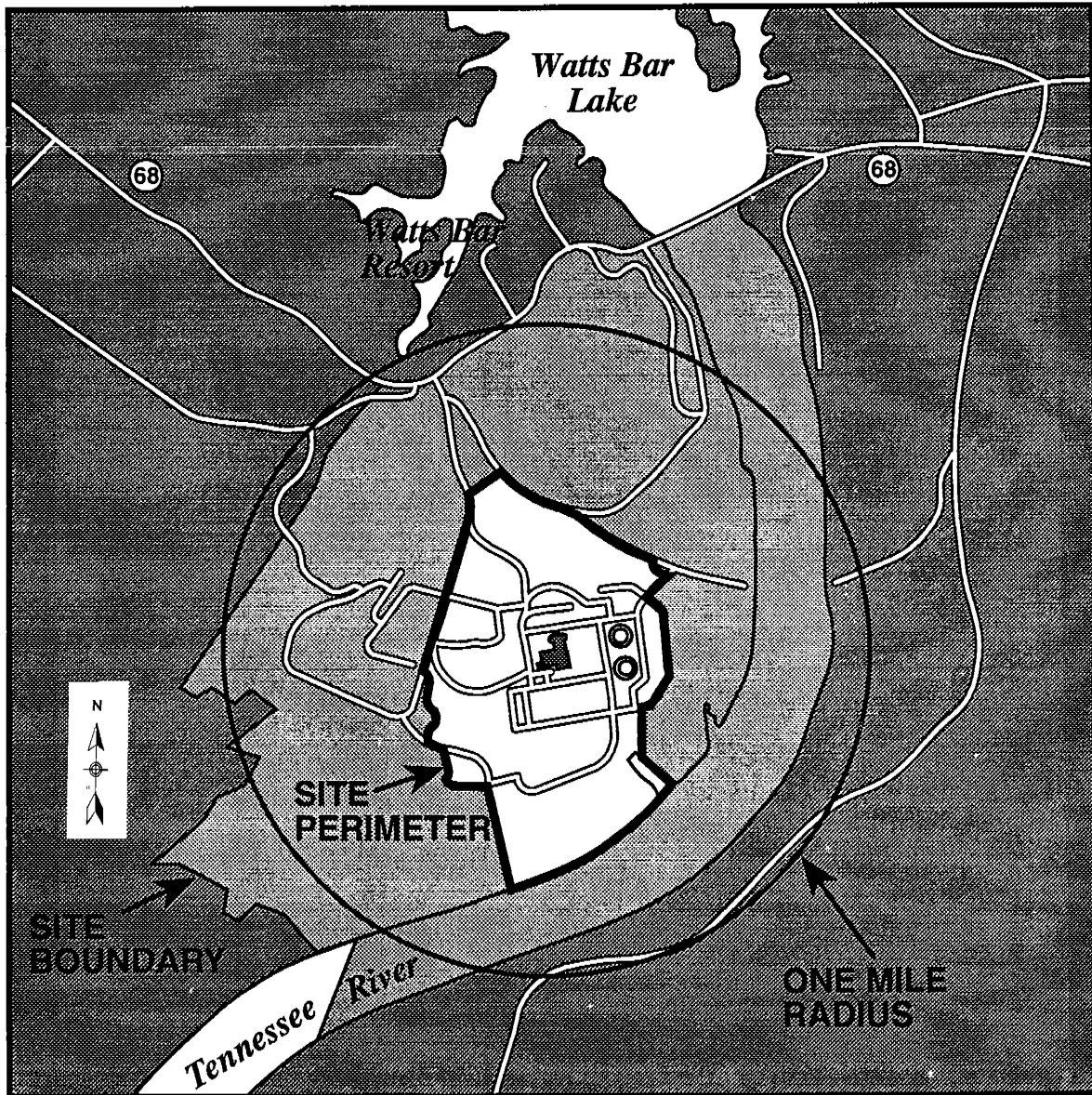
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<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGEMENT
<i>Event</i>	<i>4.4</i>	TOXIC GAS
<i>Classification</i>		UNUSUAL EVENT (continued)
<i>Mode</i>		All
<i>Basis</i> (continued)		Table 4-2 Plant Structures Associated with Toxic or Flammable Gas EALs Unit #1 & 2 Reactor Buildings Auxiliary Building Control Building Diesel Generator Building Additional Diesel Generator Building Intake Pumping Station Additional Equipment Bldgs (Unit 1 & 2) CDWE Building Turbine Building
<i>Escalation</i>		Escalation to this event will be based on toxic gases entering a plant area that jeopardizes life or impacts cold shutdown capability
<i>References</i>		NUMARC/NESP-007, HU3, Rev 2, 1/92 FSAR 2.2 Nearby Industrial, Transportation and Military Facilities DOT Emergency Response Guide for Hazardous Materials Figure 4-B One Mile Radius/Site Perimeter

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Figure 4-B

ONE MILE RADIUS / SITE PERIMETER



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<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGEMENT
<i>Event</i>	<i>4.5</i>	CONTROL ROOM EVACUATION
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to the "Fission Product Barrier Matrix"
<i>Basis</i>		The basis for a General Emergency in this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Challenges".
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92
<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGEMENT
<i>Event</i>	<i>4.5</i>	CONTROL ROOM EVACUATION
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		All
<i>Description</i>		<p>Evacuation of the Control Room has been initiated <u>and</u> Control of all necessary equipment <u>Has Not</u> been established within 15 minutes of manning the Auxiliary Control Room (1 and 2 and 3)</p> <ol style="list-style-type: none"> 1. (a or b) <ol style="list-style-type: none"> a. AOI-30.2 "Fire Safe Shutdown" entered b. AOI-27 "Main Control Room Inaccessibility" entered 2. SM/SED Orders Control Room evacuation 3. Control has <u>Not</u> been established at the Remote Shutdown Panel within 15 minutes of manning the Auxiliary Control Room and transfer of switches on Panels L11A and L11B.
<i>Basis</i>		<p>Transfer of safety system control has not been performed in an expeditious manner and it is unknown if any damage has occurred to the fission product barriers. This condition warrants the declaration of a Site Area Emergency.</p> <p>The 15 minute time limit for transfer of control is based on a reasonable time period for personnel to leave the control room, arrive at the Auxiliary Control Room area, and reestablish plant control to preclude core uncover and/or core damage per (AOI-30.2) Fire Safe Shutdown Inaccessibility. The determination of whether or not control is established at the Remote Shutdown Panel is in the judgement of the SED, who will take all event specific factors into consideration.</p>
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Challenges".
<i>References</i>		NUMARC/NESP-007, HS2, Rev 2, 1/92 AOI-30.2 Fire Safe Shutdown AOI-27 "Main Control Room Inaccessibility"

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<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGEMENT
<i>Event</i>	<i>4.5</i>	CONTROL ROOM EVACUATION
<i>Classification</i>		ALERT
<i>Mode</i>		All
<i>Description</i>		Evacuation of the Control Room is Required (1 and 2) 1. (a or b) a. AOI-30.2 "Fire Safe Shutdown" entered b. AOI-27 "Main Control Room Inaccessibility" entered 2. SM/SED Orders Control Room evacuation
<i>Basis</i>		Main Control Room evacuation requires establishment of plant control from outside the Main Control Room (Auxiliary Control Room) and support from the Technical Support Center and/or other Emergency Operating Centers and, for this potential substantial degradation, an Alert is warranted. A Main Control Room evacuation represents a serious plant situation since the level of control is not as complete as it would be without the evacuation.
<i>Escalation</i>		Escalation of this event would be based on the inability to establish plant control from outside the Main Control Room within 15 minutes.
<i>References</i>		NUMARC/NESP-007, HA5, Rev 2, 1/92 AOI-30.2 Fire Safe Shutdown AOI-27 "Main Control Room Inaccessibility"

<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGEMENT
<i>Event</i>	<i>4.5</i>	CONTROL ROOM EVACUATIONS
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		Not Applicable
<i>Description</i>		An Unusual Event for this event is "Not Applicable"
<i>Basis</i>		Not Applicable
<i>Escalation</i>		Escalation of this event would be based on Evacuation of the Main Control Room.
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92

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<i>Section</i>	4.0	HAZARDS AND SED JUDGEMENT
<i>Event</i>	4.6	SECURITY
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		All
<i>Description</i>		<p>HOSTILE ACTION resulting in loss of physical control of the facility: (1 or 2)</p> <ol style="list-style-type: none"> 1. A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain CRITICAL SAFETY FUNCTIONS. 2. A HOSTILE ACTION has caused failure of Spent Fuel Cooling Systems and IMMINENT fuel damage is likely for a freshly off-loaded reactor core in pool.
<i>Basis</i>		<p>Note: Timely and accurate communication between Security Shift Supervision and the Control Room is crucial for the implementation of effective Security EALs.</p> <p><u>EAL #1</u></p> <p>This EAL encompasses conditions under which a HOSTILE ACTION has resulted in a loss of physical control of VITAL AREAS (containing vital equipment or controls of vital equipment) required to maintain safety functions and control of that equipment cannot be transferred to and operated from another location.</p> <p>If control of the plant equipment necessary to maintain CRITICAL SAFETY FUNCTIONS can be transferred to another location, then the above initiating condition is not met.</p> <p><u>EAL #2</u></p> <p>This EAL addresses failure of spent fuel cooling systems as a result of HOSTILE ACTION if IMMINENT fuel damage is likely, such as when a freshly off-loaded reactor core is in the spent fuel pool.</p>
<i>Escalation</i>		Not Applicable
<i>References</i>		NEI 99-01 R5, HG1.

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<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGMENT
<i>Event</i>	<i>4.6</i>	SECURITY
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		All
<i>Description</i>		<p>HOSTILE ACTION within the PROTECTED AREA:</p> <p>A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervisor.</p> <p>Refer to Figure 4-A for a drawing of PROTECTED AREA and SITE PERIMETER.</p>
<i>Basis</i>		<p>Note: Timely and accurate communication between Security Shift Supervision and the Control Room is crucial for the implementation of effective Security EALs.</p> <p>This condition represents an escalated threat to plant safety above that contained in the Alert in that a HOSTILE FORCE has progressed from the OWNER CONTROLLED AREA to the PROTECTED AREA.</p> <p>This EAL addresses the contingency for a very rapid progression of events, such as that experienced on September 11, 2001. It is not premised solely on the potential for a radiological release. Rather the issue includes the need for rapid assistance due to the possibility for significant and indeterminate damage from additional air, land, or water attack elements.</p> <p>The fact that the site is under serious attack with minimal time available for further preparation or additional assistance to arrive requires ORO readiness and preparation for the implementation of protective measures.</p> <p>This EAL addresses the potential for a very rapid progression of events due to a HOSTILE ACTION. It is not intended to address incidents that are accidental events or acts of civil disobedience, such as small aircraft impact, hunters, or physical disputes between employees within the PROTECTED AREA. Those events are adequately addressed by other EALs.</p>
<i>Escalation</i>		Escalation of this emergency classification level, if appropriate, would be based on actual plant status after impact or progression of attack.
<i>References</i>		<p>NEI 99-01 R5, HS4.</p> <p>Figure 4-A Protected Area and Site Perimeter</p>

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<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGMENT
<i>Event</i>	<i>4.6</i>	SECURITY
<i>Classification</i>		ALERT
<i>Mode</i>		All
<i>Description</i>		<p>HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat:(1 or 2)</p> <ol style="list-style-type: none"> 1. A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervisor. 2. A validated notification from NRC of an airliner attack threat within 30 minutes of the site.
<i>Basis</i>		<p>Note: Timely and accurate communication between Security Shift Supervision and the Control Room is crucial for the implementation of effective Security EALs.</p> <p>These EALs address the contingency for a very rapid progression of events, such as that experienced on September 11, 2001. They are not premised solely on the potential for a radiological release. Rather the issue includes the need for rapid assistance due to the possibility for significant and indeterminate damage from additional air, land or water attack elements.</p> <p>The fact that the site is under serious attack or is an identified attack target with minimal time available for further preparation or additional assistance to arrive requires a heightened state of readiness and implementation of protective measures that can be effective (such as on-site evacuation, dispersal or sheltering).</p> <p><u>EAL #1</u></p> <p>This EAL addresses the potential for a very rapid progression of events due to a HOSTILE ACTION. It is not intended to address incidents that are accidental events or acts of civil disobedience, such as small aircraft impact, hunters, or physical disputes between employees within the OCA. Those events are adequately addressed by other EALs.</p> <p>Note that this EAL is applicable for any HOSTILE ACTION occurring, or that has occurred, in the OWNER CONTROLLED AREA. This includes ISFSI's that may be outside the PROTECTED AREA but still within the OWNER CONTROLLED AREA.</p>

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<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGMENT
<i>Event</i>	<i>4.6</i>	SECURITY
<i>Classification</i>		ALERT (continued)
<i>Mode</i>		All
		<p><u>EAL #2</u></p> <p>This EAL addresses the immediacy of an expected threat arrival or impact on the site within a relatively short time.</p> <p>The intent of this EAL is to ensure that notifications for the airliner attack threat are made in a timely manner and that Offsite Response Organizations and plant personnel are at a state of heightened awareness regarding the credible threat. Airliner is meant to be a large aircraft with the potential for causing significant damage to the plant.</p> <p>This EAL is met when a plant receives information regarding an airliner attack threat from NRC and the airliner is within 30 minutes of the plant. Only the plant to which the specific threat is made need declare the Alert.</p> <p>The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an airliner (airliner is meant to be a large aircraft with the potential for causing significant damage to the plant). The status and size of the plane may be provided by NORAD through the NRC.</p> <p>The Owner Controlled Area is defined by the Site Security Plan</p>
<i>Escalation</i>		Escalation of this event would be based on HOSTILE ACTION within the PROTECTED AREA .
<i>References</i>		NEI 99-01R5, HA4 Figure 4-A Protected Area and Site Perimeter

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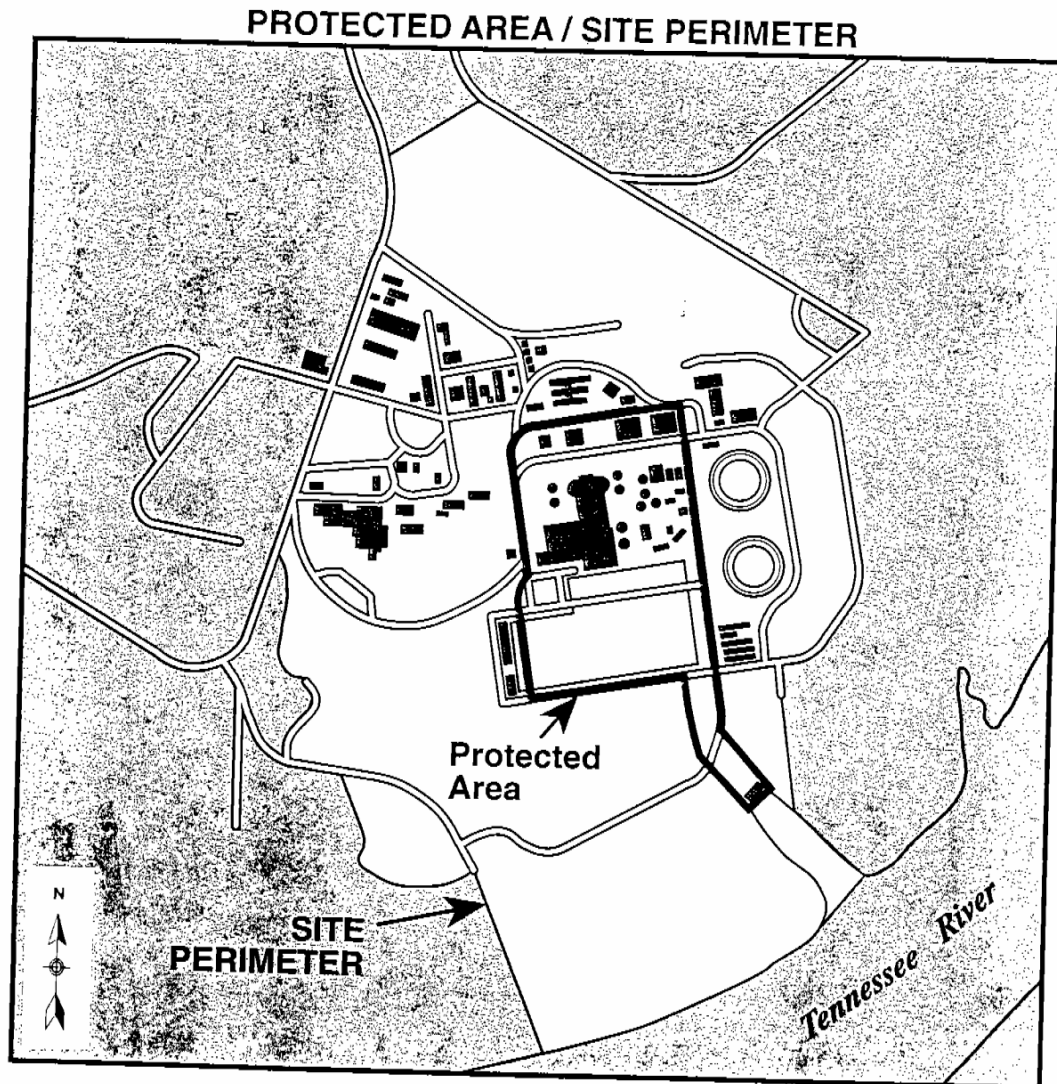
<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGMENT
<i>Event</i>	<i>4.6</i>	SECURITY
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		All
<i>Description</i>		<p>Confirmed SECURITY CONDITION or threat Which Indicates a Potential Degradation in the Level of Safety of the plant: (1 or 2 or 3)</p> <ol style="list-style-type: none"> 1. A SECURITY CONDITION that does NOT involve a HOSTILE ACTION as reported by the Security Shift Supervisor. 2. A site specific credible threat notification. 3. A validated notification from NRC providing information of an aircraft threat.
<i>Basis</i>		<p>Note: Timely and accurate communication between Security Shift Supervision and the Control Room is crucial for the implementation of effective Security EALs.</p> <p>Security events which do not represent a potential degradation in the level of safety of the plant are reported under 10 CFR 73.71 or in some cases under 10 CFR 50.72. Security events assessed as HOSTILE ACTIONS are classifiable under EAL 4.6 Alert, SAE, and GE.</p> <p>A higher initial classification could be made based upon the nature and timing of the security threat and potential consequences. The licensee shall consider upgrading the emergency response status and emergency classification level in accordance with the site's Safeguards Contingency Plan and Emergency Plan.</p> <p><u>EAL #1</u></p> <p>Reference is made to site specific security shift supervision because these individuals are the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Safeguards Contingency Plan.</p> <p>This threshold is based on site specific security plans. Site specific Safeguards Contingency Plans are based on guidance provided by NEI 03-12.</p> <p><u>EAL #2</u></p> <p>This threshold is included to ensure that appropriate notifications for the security threat are made in a timely manner. This includes information of a credible threat. Only the plant to which the specific threat is made need declare the Notification of an Unusual Event.</p> <p>The determination of "credible" is made through use of information found in the site specific Safeguards Contingency Plan.</p>

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<i>Section</i>	<i>4.0</i>	HAZARDS AND SED JUDGMENT
<i>Event</i>	<i>4.6</i>	SECURITY
<i>Classification</i>		UNUSUAL EVENT (continued)
<i>Mode</i>		All
		<p><u>EAL #3</u></p> <p>The intent of this EAL is to ensure that notifications for the aircraft threat are made in a timely manner and that Offsite Response Organizations and plant personnel are at a state of heightened awareness regarding the credible threat. It is not the intent of this EAL to replace existing non-hostile related EALs involving aircraft.</p> <p>This EAL is met when a plant receives information regarding an aircraft threat from NRC. Validation is performed by calling the NRC or by other approved methods of authentication. Only the plant to which the specific threat is made need declare the Unusual Event.</p> <p>The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an airliner (airliner is meant to be a large aircraft with the potential for causing significant damage to the plant). The status and size of the plane may be provided by NORAD through the NRC.</p>
<i>Escalation</i>		Escalation to Alert emergency classification level would be via EAL 4.6 Alert would be appropriate if the threat involves an airliner within 30 minutes of the plant.
<i>References</i>		NEI 99-01R5, HU4 Figure 4-A Protected Area and Site Perimeter

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Figure 4-A



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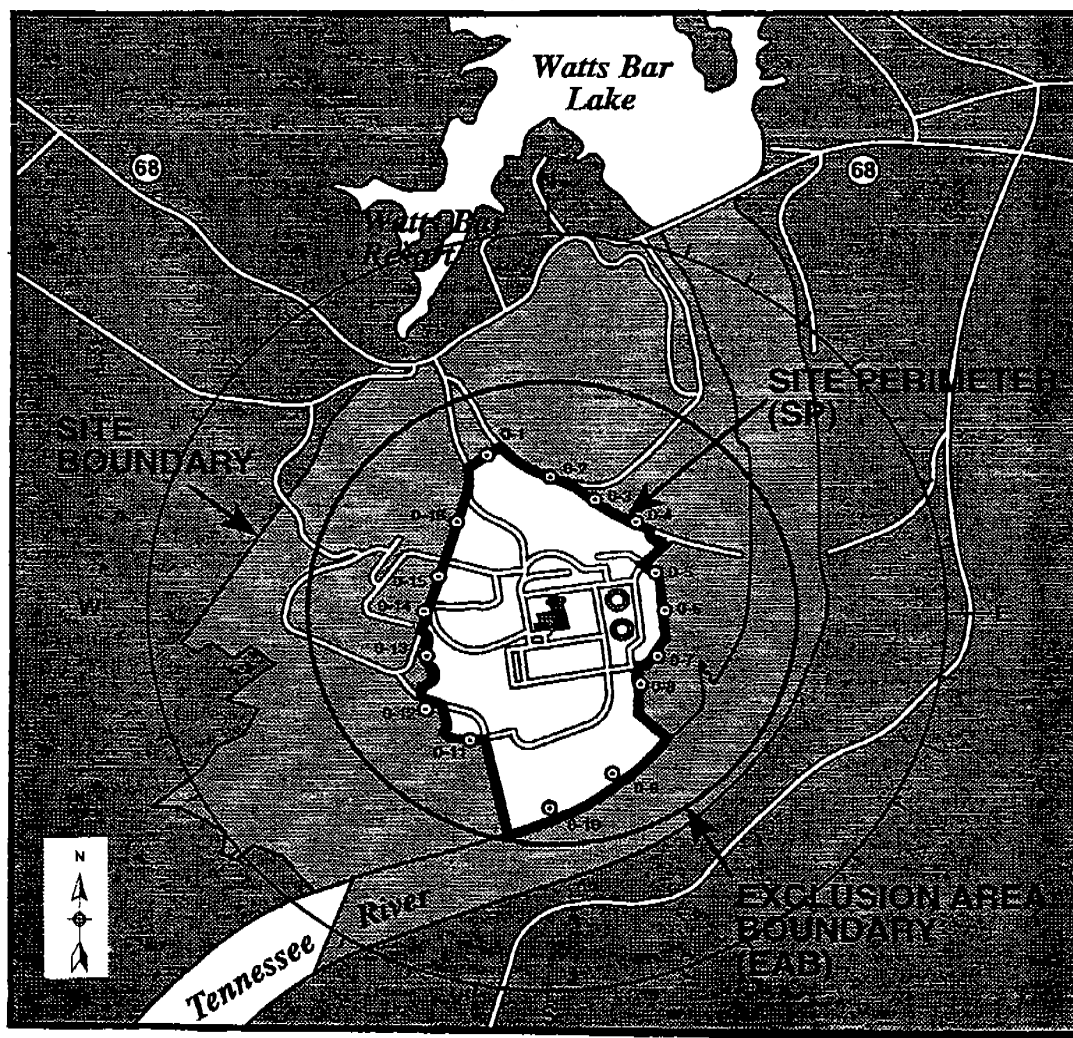
<i>Section</i>	4.0	HAZARDS AND SED JUDGMENT
<i>Event</i>	4.7	EMERGENCY DIRECTOR JUDGMENT
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		All
<i>Description</i>		Events are in process <u>or</u> have occurred which involve Actual <u>or</u> Imminent Substantial Core Degradation <u>or</u> Melting With Potential for Loss of Containment Integrity <u>or</u> HOSTILE ACTION that results, in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline Exposure Levels outside the EXCLUSION AREA BOUNDARY, Refer to Figure 7-A.
<i>Basis</i>		<p>This event classification provides the Shift Supervisor/Site Emergency Director, the flexibility to declare a General Emergency if in their judgment unanticipated conditions not explicitly covered elsewhere warrant declaration of an emergency.</p> <p>The declaration of a General Emergency indicates that there is a very high probability that the fuel has been damaged and the loss of containment integrity is possible or other conditions exist that may result in a release to the environment that may be greater than the EPA Protective Action Guides.</p>
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, HG1, HG2, Rev 2, 1/92; HG1, NRC Bulletin 2005-02

<i>Section</i>	4.0	HAZARDS AND SED JUDGMENT
<i>Event</i>	4.7	EMERGENCY DIRECTOR JUDGMENT
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		All
<i>Description</i>		Events are in process <u>or</u> have occurred which involve an Actual <u>or</u> Likely Major Failures of Plant Functions needed for Protection of the Public <u>or</u> HOSTILE ACTION that results in intentional damage or malicious acts; (1) toward site personnel or equipment that could lead to the likely failure of or; (2) prevent effective access to equipment needed for the Protection of the Public. Any releases are not expected to result in Exposure Levels which Exceed EPA Protective Action Guideline Exposure Levels outside the EXCLUSION AREA BOUNDARY. Refer to Figure 7-A.
<i>Basis</i>		<p>This event classification provides the Shift Supervisor/Site Emergency Director, the flexibility to declare a Site Area Emergency if in their judgment unanticipated conditions not explicitly covered elsewhere warrant declaration.</p> <p>The declaration of a Site Area Emergency indicates high probability of Major failures of plant functions needed to protect the public.</p>
<i>Escalation</i>		Escalation of this event would be based on actual or imminent substantial core degradation.
<i>References</i>		NUMARC/NESP-007, HS2, Rev 2, 1/92; HS1, HS4, NRC Bulletin 2005-02

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Figure 7-A

EXCLUSION AREA/SITE BOUNDARY



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<i>Section</i>	4.0	HAZARDS AND SED JUDGMENT
<i>Event</i>	4.7	EMERGENCY DIRECTOR JUDGMENT
<i>Classification</i>		ALERT
<i>Mode</i>		All
<i>Description</i>		Events are in process <u>or</u> have occurred which involve an Actual <u>or</u> Potential Substantial Degradation of the Level of Safety of the Plant <u>or</u> a Security Event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline Exposure Levels.
<i>Basis</i>		This event classification provides the Shift Supervisor or the Site Emergency Director, the flexibility to declare an Alert if, in their judgment, unanticipated conditions not explicitly covered elsewhere warrant declaration of an emergency.
<i>Escalation</i>		Escalation of this event would be based on actual or likely failures in plant functions needed to protect the public.
<i>References</i>		NUMARC/NESP-007, HA6, Rev 2, 1/92; HA4, HA7, HA8 NRC Bulletin 2005-02

<i>Section</i>	4.0	HAZARDS AND SED JUDGMENT
<i>Event</i>	4.7	EMERGENCY DIRECTOR JUDGMENT
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		All
<i>Description</i>		Events are in process <u>or</u> have occurred which indicate a Potential Degradation of the Level of Safety of the Plant <u>or</u> indicate a Security Threat to facility protection has been initiated. No Releases of Radioactive Material requiring Offsite Response <u>or</u> Monitoring are expected unless further degradation of Safety Systems occurs.
<i>Basis</i>		This event classification provides the Shift Supervisor the flexibility to declare an Unusual Event if, in his judgment, unanticipated conditions not explicitly covered elsewhere warrant declaration of an emergency.
<i>Escalation</i>		Escalation of this event would be based on actual degradation of plant safety systems.
<i>References</i>		NUMARC/NESP-007, HU5, Rev 2, 1/92; HU4, NRC Bulletin 2005-02

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<i>Section</i>	<i>5.0</i>	DESTRUCTIVE PHENOMENON
<i>Event</i>	<i>5.1</i>	EARTHQUAKE
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to the "Fission Product Barrier Matrix"
<i>Basis</i>		The basis for a General Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	<i>5.0</i>	DESTRUCTIVE PHENOMENON
<i>Event</i>	<i>5.1</i>	EARTHQUAKE
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to the "Fission Product Barrier Matrix"
<i>Basis</i>		The basis for a Site Area Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Challenges".
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

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<i>Section</i>	<i>5.0</i>	DESTRUCTIVE PHENOMENON
<i>Event</i>	<i>5.1</i>	EARTHQUAKE
<i>Classification</i>		ALERT
<i>Mode</i>		All
<i>Description</i>		<p>Earthquake detected by site seismic instrumentation (1 and 2)</p> <ol style="list-style-type: none"> 1. (a and b) <ol style="list-style-type: none"> a. Ann.166 D indicates "OBE Spectra Exceeded" b. Ann.166 E indicates "Seismic Recording Initiated" 2. (a or b) <ol style="list-style-type: none"> a. Ground motion sensed by Plant personnel b. National Earthquake Information Center at 1-(303) 273-8500 can confirm the event
<i>Basis</i>		<p>A seismic event of this level can cause damage to safety related systems.</p> <p>Plant seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix "A" of 10 CFR Part 100. All specified measurement ranges represent the minimum ranges of the instruments.</p>
<i>Escalation</i>		Escalation of this event will be based on " Fission Product Barrier Challenges".
<i>References</i>		NUMARC/NESP-007, HA1, Rev. 2, 1/92 FSAR 1.2 General Plant Description FSAR 2.5 Geology, Seismology and Geotechnical Engineering Summary of Foundation Conditions T. R. 3.3.4 (Seismic Monitoring Instrumentation) NUREG 1.12, "Instrumentation for Earthquakes", April 1974 ARI-166-172, Rev. 1 EPRI Report NP-6693 "Guidelines for Nuclear Plant Response to an Earthquake", December 1989

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<i>Section</i>	<i>5.0</i>	DESTRUCTIVE PHENOMENON
<i>Event</i>	<i>5.1</i>	EARTHQUAKE
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		All
<i>Description</i>		<p>Earthquake detected by site seismic instrumentation (1 and 2)</p> <ol style="list-style-type: none"> 1. Ann. 166E indicator "Seismic Recording initiated" 2. (a or b) <ol style="list-style-type: none"> a. Ground motion sensed by Plant personnel b. National Earthquake Information Center at 1-(303) 273-8500 can confirm the event
<i>Basis</i>		<p>A seismic event of this level can cause some minor damage to plant structure or systems but it is not expected to have any impact on overall plant safety functions.</p> <p>Plant seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix "A" of 10 CFR Part 100. All specified measurement ranges represent the minimum ranges of the instruments.</p>
<i>Escalation</i>		Escalation of this event will be based on a Safe Shutdown Earthquake. (SSE)
<i>References</i>		NUMARC/NESP-007, HU1, Rev. 2, 1/92 FSAR 1.2, General Plant Description FSAR 2.5 Geology, Seismology and Geotechnical Engineering Summary of Foundation Conditions T. R. 3.3.4 (Seismic Monitoring Instrumentation) NUREG 1.12, "Instrumentation for Earthquakes", April 1974 ARI-166-172, Rev. 1

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<i>Section</i>	<i>5.0</i>	DESTRUCTIVE PHENOMENON
<i>Event</i>	<i>5.2</i>	TORNADO
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to the "Fission Product Barrier Matrix"
<i>Basis</i>		The basis for a General Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	<i>5.0</i>	DESTRUCTIVE PHENOMENON
<i>Event</i>	<i>5.2</i>	TORNADO
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to the "Fission Product Barrier Matrix"
<i>Basis</i>		The basis for a Site Area Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Challenges".
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

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<i>Section</i>	<i>5.0</i>	DESTRUCTIVE PHENOMENON
<i>Event</i>	<i>5.2</i>	TORNADO
<i>Classification</i>		ALERT
<i>Mode</i>		All
<i>Description</i>	<p>Tornado <u>or</u> High Winds strikes any structure listed in Table 5-1 and results in VISIBLE DAMAGE (1 and 2)</p> <ol style="list-style-type: none"> 1. Tornado <u>or</u> High Winds (Sustained >80 mph > one minute) strikes any structure listed in Table 5-1. 2. (a or b) <ol style="list-style-type: none"> a. Confirmed report of any VISIBLE DAMAGE b. Control Room indications of degraded Safety System <u>or</u> component response due to event <p><i>Note: Site Met Data Instrumentation fails to 0 at >100 mph. National Weather Service Morristown 1-(423)-586-8400 can provide additional information if needed.</i></p>	
<i>Basis</i>	<p>Tornadoes or high winds striking the structures listed in Table 5-1 can cause damage to plant structures or systems needed for Safe Shutdown of the Plant. At Watts Bar, tornadoes are a phenomenon whose occurrence cannot be specifically predicted. The FSAR estimates the probability of a tornado occurrence onsite as one in 6,700 years.</p> <p>Windstorms are relatively infrequent, but may occur several times a year. The records show the highest wind speed recorded in Chattanooga was 82 mph in March 1947. The records show the highest wind speed recorded in Knoxville was 73 mph in July 1961.</p> <p>Table 5-1 Plant Structures Associated With Tornado/Hi Wind and Aircraft EALs</p> <ul style="list-style-type: none"> Unit #1 and 2 Reactor Buildings Auxiliary Building Control Building Diesel Generator Building Additional Diesel Generator Building Intake Pumping Station Additional Equipment Buildings (Units 1 & 2) CDWE Building Turbine Building RWST CST 	

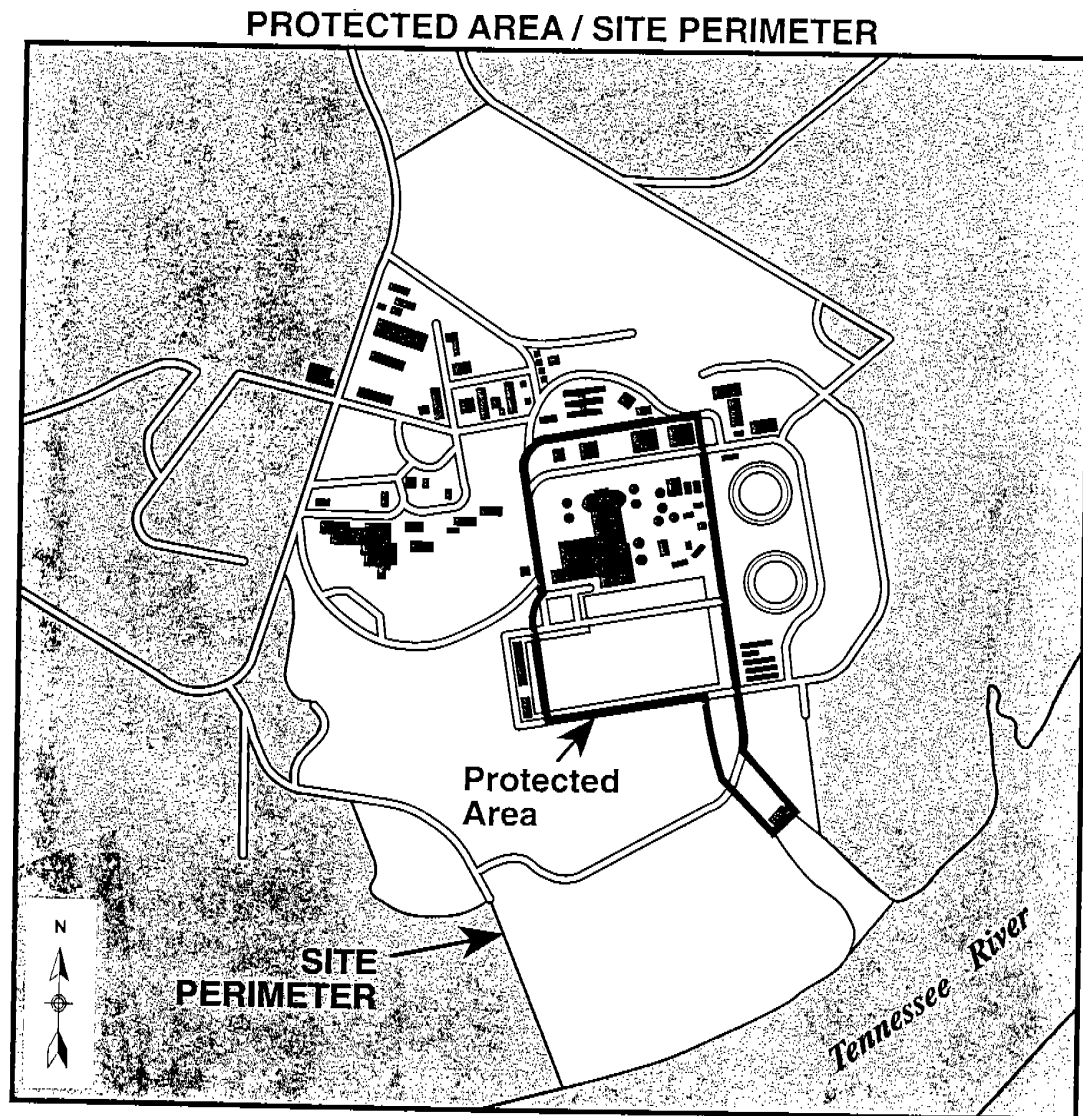
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<i>Section</i>	5.0	DESTRUCTIVE PHENOMENON
<i>Event</i>	5.2	TORNADO
<i>Classification</i>		ALERT (continued)
<i>Mode</i>		All
<i>Basis</i> (continued)		VISIBLE DAMAGE: Damage to equipment that is readily observable without measurements, testing, or analyses. Damage is sufficient enough to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering. Surface blemishes (e.g., paint chipping, scratches should not be included).
<i>Escalation</i>		Escalation of this event will be based on Fission Product Barriers Challenges.
<i>References</i>		NUMARC/NESP-007, HAI, Rev. 2, 1/92 FSAR 1.2 General Plant Description FSAR 2.3 Meteorology

<i>Section</i>	5.0	DESTRUCTIVE PHENOMENON
<i>Event</i>	5.2	TORNADO
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		All
<i>Description</i>		Tornado Within The SITE PERIMETER 1. Plant personnel report a Tornado has been Sighted within the SITE PERIMETER (Refer to Figure 5-A)
<i>Basis</i>		A tornado touchdown near or within the Site Protected Area may have the potential to damage plant structures containing systems required for Safe Shutdown of the plant. At Watts Bar, tornadoes are a phenomenon whose occurrence cannot be specifically predicted. The FSAR estimates the probability of a tornado occurrence onsite as one in 6,700 years. SITE PERIMETER encompasses all owner controlled areas in the immediate site environs as shown on Figure 5-A.
<i>Escalation</i>		Escalation of this event will be based on the tornado striking plant structures or high sustained winds within the protected area.
<i>References</i>		NUMARC/NESP-007, HUI, Rev. 2, 1/92 FSAR 1.2 General Plant Description FSAR 2.3 Meteorology Figure 5-A

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Figure 5-A



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<i>Section</i>	<i>5.0</i>	DESTRUCTIVE PHENOMENON
<i>Event</i>	<i>5.3</i>	AIRCRAFT/PROJECTILE CRASH
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to the "Fission Product Barrier Matrix"
<i>Basis</i>		The basis for a General Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	<i>5.0</i>	DESTRUCTIVE PHENOMENON
<i>Event</i>	<i>5.3</i>	AIRCRAFT/PROJECTILE CRASH
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to the "Fission Product Barrier Matrix"
<i>Basis</i>		The basis for a Site Area Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Challenges".
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

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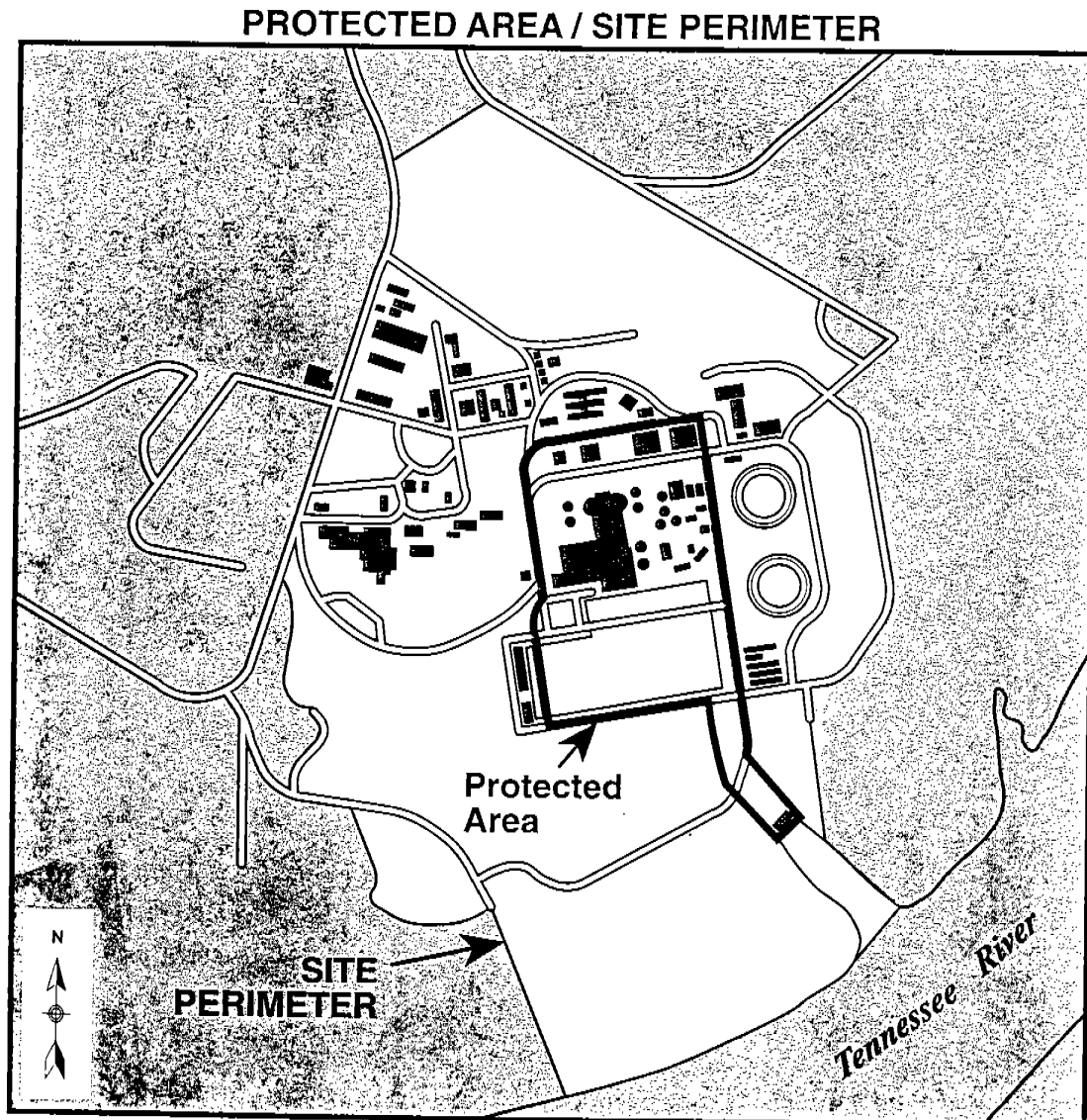
<i>Section</i>	5.0	DESTRUCTIVE PHENOMENON
<i>Event</i>	5.3	AIRCRAFT/PROJECTILE CRASH
<i>Classification</i>		ALERT
<i>Mode</i>		All
<i>Description</i>	<p>Aircraft <u>or</u> PROJECTILE impacts (Strikes) any Plant structure Listed in Table 5-1 resulting in VISIBLE DAMAGE (1 and 2)</p> <ol style="list-style-type: none"> 1. Plant personnel report aircraft <u>or</u> PROJECTILE has impacted any structure listed in Table 5-1 2. (a or b) <ol style="list-style-type: none"> a. Confirmed report of any VISIBLE DAMAGE b. Control Room indications of degraded Safety System <u>or</u> component response due to the event within the specified areas. 	
<i>Basis</i>	<p>VISIBLE DAMAGE: Damage to equipment that is readily observable without measurements, testing, or analyses. Damage is sufficient enough to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering. Surface blemishes (e.g., paint chipping, scratches should not be included).</p> <p>There are no industrial or military facilities within five miles of the Watts Bar Nuclear Plant site which would potentially pose a hazard to the safe operation of the plant.</p> <p>Table 5-1 Plant Structures Associated With Tornado/Hi Wind and Aircraft EALs</p> <ul style="list-style-type: none"> Unit #1 and 2 Reactor Buildings Auxiliary Building Control Building Diesel Generator Building Additional Diesel Generator Building Intake Pumping Station Additional Equipment Buildings (Units 1 & 2) CDWE Building Turbine Building RWST CST 	
<i>Escalation</i>	Escalation to this event will be based on "Fission Product Barriers Challenges".	
<i>References</i>	NUMARC/NESP-007, HA1, HA2, Rev. 2, 1/92 FSAR 2.2 Nearby Industrial, Transportation And Military Facilities Table 5-1	

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<i>Section</i>	<i>5.0</i>	DESTRUCTIVE PHENOMENON
<i>Event</i>	<i>5.3</i>	AIRCRAFT/PROJECTILE CRASH
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		All
<i>Description</i>		Aircraft crash <u>or</u> PROJECTILE impacts within the SITE PERIMETER 1. Plant personnel report Aircraft crash <u>or</u> PROJECTILE impact within the SITE PERIMETER (Refer to Figure 5-A)
<i>Basis</i>		<p>Aircraft or PROJECTILE Impacts within the SITE PERIMETER are off normal events that can indicate a potential degradation of the level of safety of the plant.</p> <p>There are no industrial or military facilities within five miles of the Watts Bar Nuclear Plant site which would potentially pose a hazard to the safe operation of the plant.</p> <p>SITE PERIMETER encompasses all owner controlled areas in the immediate site environs as shown on Figure 5-A and 7-A.</p> <p>PROJECTILE includes an object ejected, thrown, or launched towards a plant structure. The source of the projectile may be onsite or offsite. Damage is sufficient to cause concern regarding the integrity of the affected structure or the operability or reliability of safety equipment contained therein.</p>
<i>Escalation</i>		Escalation to this event will be based on an Impact on plant structures or barriers.
<i>References</i>		NUMARC/NESP-007, HU1, Rev. 2, 1/92 FSAR 2.2 Nearby Industrial, Transportation And Military Facilities

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Figure 5-A



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<i>Section</i>	<i>5.0</i>	DESTRUCTIVE PHENOMENON
<i>Event</i>	<i>5.4</i>	RIVER LEVEL HIGH
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to the "Fission Product Barrier Matrix"
<i>Basis</i>		The basis for a General Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	<i>5.0</i>	DESTRUCTIVE PHENOMENON
<i>Event</i>	<i>5.4</i>	RIVER LEVEL HIGH
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to the "Fission Product Barrier Matrix"
<i>Basis</i>		The basis for a Site Area Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Challenges".
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

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<i>Section</i>	<i>5.0</i>	DESTRUCTIVE PHENOMENON
<i>Event</i>	<i>5.4</i>	RIVER LEVEL HIGH
<i>Classification</i>		ALERT
<i>Mode</i>		All
<i>Description</i>		<p>River Reservoir level is at Stage II Flood Warning (1 or 2)</p> <ol style="list-style-type: none"> 1. River Reservoir level >727 Ft 2. Stage II Flood Warning (AOI-7) has been issued by River Operations
<i>Basis</i>		<p>The requirements for flood protection ensures that facility protective actions will be taken and operation will be terminated in the event of flood conditions. A Stage 1 flood warning is issued when the water in the forebay is predicted to exceed 714.5 feet Mean Sea Level USGS datum during October 1 through April 15, or 726.5 feet Mean Sea Level USGS datum during April 15 through September 30. A Stage II flood warning is issued when the water in the forebay is predicted to exceed 727 feet Mean Sea Level USGS datum. A maximum allowed water level of 727 feet Mean Sea Level USGS datum provides sufficient margin to ensure waves due to high winds cannot disrupt the flood mode preparation. A Stage I or Stage II flood warning requires the implementation of procedures which include plant shutdown. Further, in the event of a loss of communications simultaneous with a critical combination flood, headwaters, and/or seismically induced dam failure the plant will be shutdown and flood protection measures implemented.</p> <p>Chickamauga Lake level during nonflood conditions should be no higher than elevation 685.44, top of gates, and is not likely to exceed elevation 682.5, normal summer level, for any significant time. No conceivable hurricane or cyclonic-type winds could produce the some 20 feet of wave height required to reach plant grade elevation 728.</p>
<i>Escalation</i>		Escalation of this event will be based on "Fission Product Barriers Challenges".
<i>References</i>		NUMARC/NESP-007, HA1, Rev. 2, 1/92 FSAR 2.4 Hydrologic Engineering T.R. 3.7.2 Flood Protection Plan

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<i>Section</i>	<i>5.0</i>	DESTRUCTIVE PHENOMENON
<i>Event</i>	<i>5.4</i>	RIVER LEVEL HIGH
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		All
<i>Description</i>		<p>River Reservoir level is at Stage I Flood Warning (1 or 2 or 3)</p> <ol style="list-style-type: none"> 1. River Reservoir level >726.5 Ft from April 16 thru September 30 2. River Reservoir level >714.5 Ft from October 1 thru April 15 3. Stage I Flood Warning (AOI-7) has been Issued by River Operations
<i>Basis</i>		<p>The requirements for flood protection ensures that facility protective actions will be taken and operation will be terminated in the event of flood conditions. A Stage 1 flood warning is issued when the water in the forebay is predicted to exceed 714.5 feet Mean Sea Level USGS datum during October 1 through April 15, or 726.5 feet Mean Sea Level USGS datum during April 15 through September 30.</p> <p>A Stage I flood warning requires the implementation of procedures which include plant shutdown. Further, in the event of a loss of communications simultaneous with a critical combination flood, headwaters, and/or seismically induced dam failure the plant will be shutdown and flood protection measures implemented.</p> <p>Chickamauga Lake level during nonflood conditions should be no higher than elevation 685.44, top of gates, and is not likely to exceed elevation 682.5, normal summer level, for any significant time. No conceivable hurricane or cyclonic-type winds could produce the some 20 feet of wave height required to reach plant grade elevation 728.</p> <p>Because of its inland location, the Watts Bar plant is not endangered by tsunami flooding.</p>
<i>Escalation</i>		Escalation of this event will be based on river level being at Stage II Flood Warning.
<i>References</i>		NUMARC/NESP-007, HU1, Rev. 2, 1/92 FSAR 2.4 Hydrologic Engineering T.R. 3.7.2 Flood Protection Plan

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<i>Section</i>	<i>5.0</i>	DESTRUCTIVE PHENOMENON
<i>Event</i>	<i>5.5</i>	RIVER LEVEL LOW
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to the "Fission Product Barrier Matrix"
<i>Basis</i>		The basis for a General Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	<i>5.0</i>	DESTRUCTIVE PHENOMENON
<i>Event</i>	<i>5.5</i>	RIVER LEVEL LOW
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to the "Fission Product Barrier Matrix"
<i>Basis</i>		The basis for a Site Area Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Challenges".
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

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<i>Section</i>	<i>5.0</i>	DESTRUCTIVE PHENOMENON
<i>Event</i>	<i>5.5</i>	RIVER LEVEL LOW
<i>Classification</i>		ALERT
<i>Mode</i>		All
<i>Description</i>		River Reservoir level is <668 Ft (AOI-22) as reported by River Operations
<i>Basis</i>		<p>The ERCW pumping station is located within the plant intake structure, and has direct communication with the main river channel for all reservoir levels including loss of downstream dam. The minimum required reservoir level for normal operation is 668 feet. This level applies for ERCW supply temperature less than or equal to 83°F.</p> <p>Since January 1940, water levels at the plant have been controlled by Chickamauga Reservoir. Since then, the minimum level at the dam was 673.3 on January 21, 1942.</p> <p>Because of its inland location on a relatively small, narrow lake, low water levels resulting from surges, seiches, or tsunamis are not a potential problem.</p>
<i>Escalation</i>		Escalation to this event will be based on "Fission Product Barrier Challenges."
<i>References</i>		NUMARC/NESP-007, HA1, Rev. 2, 1/92 FSAR 2.4 Hydrologic Engineering FSAR 9.2.1 Essential Raw Cooling Water T.S. 3.7.9 Ultimate Heat Sink AOI-22 Break of Down Stream Dam

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<i>Section</i>	<i>5.0</i>	DESTRUCTIVE PHENOMENON
<i>Event</i>	<i>5.5</i>	RIVER LEVEL LOW
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		All
<i>Description</i>		River Reservoir level is \leq673 Ft (AOI-22) as reported by River Operations
<i>Basis</i>		<p>The ERCW pumping station is located within the plant intake structure, and has direct communication with the main river channel for all reservoir levels including loss of downstream dam. The minimum required reservoir level for normal operation is 668 feet. This level applies for ERCW supply temperature less than or equal to 83° F.</p> <p>Since January 1940, water levels at the plant have been controlled by Chickamauga Reservoir. Since then, the minimum level at the dam was 673.3 on January 21, 1942. Because of its location on Chickamauga Reservoir, maintaining minimum water levels at the Watts Bar plant does not represent a problem. The high rainfall and runoff of the watershed and the regulation afforded by upstream dams assure minimum flows for plant cooling. Because of its inland location on a relatively small, narrow lake, low water levels resulting from surges, seiches, or tsunamis are not a potential problem.</p>
<i>Escalation</i>		Escalation to this event will be based on reduced river levels.
<i>References</i>		NUMARC/NESP-007, HU1, Rev. 2, 1/92 FSAR 2.4 Hydrologic Engineering FSAR 9.2.1 Essential Raw Cooling Water T.S. 3.7.9 Ultimate Heat Sink AOI-22 Break of Down Stream Dam

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<i>Section</i>	<i>5.0</i>	DESTRUCTIVE PHENOMENON
<i>Event</i>	<i>5.6</i>	WATERCRAFT CRASH
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to the "Fission Product Barrier Matrix"
<i>Basis</i>		The basis for a General Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	<i>5.0</i>	DESTRUCTIVE PHENOMENON
<i>Event</i>	<i>5.6</i>	WATERCRAFT CRASH
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to the "Fission Product Barrier Matrix"
<i>Basis</i>		The basis for a Site Area Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Challenges".
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

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<i>Section</i>	5.0	DESTRUCTIVE PHENOMENON
<i>Event</i>	5.6	WATERCRAFT CRASH
<i>Classification</i>		ALERT
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to the "Fission Product Barrier Matrix"
<i>Basis</i>		The basis for an Alert for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
<i>Escalation</i>		Escalation will be based on "Fission Product Barrier Challenges".
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	5.0	DESTRUCTIVE PHENOMENON
<i>Event</i>	5.6	WATERCRAFT CRASH
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		All
<i>Description</i>		<p>Watercraft Strikes the Intake Pumping Station resulting in a reduction of Essential Raw Cooling Water (ERCW) or Raw Cooling Water (RCW) (1 and 2)</p> <ol style="list-style-type: none"> 1. Plant personnel report a Watercraft has struck the Intake Pumping Station 2. (a or b or c) <ol style="list-style-type: none"> a. ERCW Supply Header Pressure Train A 0-PI-67-18A is <15 psig b. ERCW Supply Header Pressure Train B 0-PI-67-17A is <15 psig c. RCW Supply Header Pressure 0-PI-24-22 is <15 psig
<i>Basis</i>		Based on Watts Bar's river location, the potential for a watercraft accident affecting Essential Raw Cooling Water (ERCW) or Fire Support Water is remote. In the unlikely event that this accident occurs, the potential exist for possible damage to plant safety systems needed for safe shutdown. With this potential an Unusual Event is warranted.
<i>Escalation</i>		Escalation would be based on "Fission Product Barrier Challenges".
<i>References</i>		NUMARC/NESP-007, HU1, Rev. 2, 1/92 FSAR 2.0 Geography and Demography FSAR 2.4.8 Cooling Water Canals and Reservoirs FSAR 7.4 Systems Required for Safe Shutdown FSAR 9.2.1 Essential Raw Cooling Water T.S. 3.7.9 Ultimate Heat Sink

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<i>Section</i>	<i>6.0</i>	SHUTDOWN DEGRADATION
<i>Event</i>	<i>6.1</i>	LOSS OF SHUTDOWN SYSTEMS
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		5,6
<i>Description</i>		Note: Additional Information will be provided later pending NRC/NUMARC guidance on Shutdown EALs. Refer to Gaseous Effluents (7.1)
<i>Basis</i>		Pending
<i>Escalation</i>		Not Applicable
<i>References</i>		NRC/NUMARC, future guidance

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<i>Section</i>	6.0	SHUTDOWN DEGRADATION
<i>Event</i>	6.1	LOSS OF SHUTDOWN SYSTEMS
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		5,6
<i>Description</i>		<p>Loss of water level in the Rx vessel that has <u>or</u> will uncover fuel in the Rx vessel (1 and 2 and 3 and 4)</p> <ol style="list-style-type: none"> 1. Loss of RHR capability 2. Rx vessel water level < el. 718' 3. Incore TCs (if available) indicate RCS Temp. > 200°F 4. RCS is vented/open to CNTMT <p>Note: If CNTMT open, refer to "Gaseous Effluents" (7.1)</p>
<i>Basis</i>		<p>For WBN, this IC is based on concerns raised by Generic Letter 88-17, "Loss of Decay Heat Removal ", SECY-91-283, "Evaluation of Shutdown and Low Power Risk Issues." A number of variables such as initial vessel level (e.g., mid-loop, reduced level/flange level, normal, or cavity filled), RCS venting strategy, decay heat removal system design, vortexing pre-disposition, steam generator U-tube draining, and level instrumentation problems can have a significant impact in causing or degrading a loss of decay heat removal. NRC analyses show that specific sequences can result in core uncover in 15 to 20 minutes and severe core damage within an hour after decay heat removal is lost.</p> <p>This EAL is intended to establish the escalation threshold for the declaration of a Site Area Emergency. This Site Area Emergency declaration is consistent with the need to rapidly correct the problem through the augmentation of onsite personnel and the need to inform offsite authorities. Continued degradation can rapidly result in fuel uncover and severe damage with resultant releases of a significant fraction of the gap activity. In the situation where the RCS is vented/opened to Containment, the potential exists (if reactor vessel water level is not reestablished) to release radioactivity to the environment.</p> <p>The Rx vessel level indication of el 718' represents the water level at the hot leg center line.</p>
<i>Escalation</i>		Escalation to this event will be based on Gaseous Effluent (7.2)
<i>References</i>		NUMARC/NESP-007, SS5 (expanded), Rev. 2, 1/92 AOI-14 Loss of RHR T.S. 3.4.7 & 8 RCS Loops Filled and Not Filled T.S. 3.9.4 Containment Penetrations

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<i>Section</i>	6.0	SHUTDOWN DEGRADATION
<i>Event</i>	6.1	LOSS OF SHUTDOWN SYSTEMS
<i>Classification</i>		ALERT
<i>Mode</i>		5,6
<i>Description</i>	<p>Inability to maintain unit in Cold Shutdown (1 and 2)</p> <ol style="list-style-type: none"> 1. RHR capability is not available for RCS cooling 2. Incore TCs (if available) indicate RCS temp > 200° F. <p>Note: If CNTMT open, refer to "Gaseous Effluents" (7.1)</p>	
<i>Basis</i>	<p>Inability to maintain Cold Shutdown refers to unplanned actions resulting from either equipment malfunctions or operator error that results in an increasing trend in Reactor Coolant Temperature.</p> <p>This condition could result from the loss of Cooling Water to the RHR Heat Exchanger or equipment failures within the RHR System or AC/DC power loss to the RHR and/or Service Water Components (i.e., CCS, ERCW). Should this condition occur, the first line of defense is to maintain Heat Sink Capability and remove heat via the Steam Generators.</p> <p>For WBN, 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 which can lead to conditions where decay heat removal is lost and core uncover can occur. NRC analyses show that these sequences can cause core uncover 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 an Alert. The indicators for these EALs are those methods used by the plant in response to Generic Letter 88-17 which include core exit temperature monitoring and RCS water level monitoring.</p> <p>The inability to achieve this condition warrants declaration of an Alert.</p>	
<i>Escalation</i>	Loss of water level in the reactor vessel that has or will uncover fuel in the vessel will escalate this event.	
<i>References</i>	NUMARC/NESP-007, SA3, Rev. 2, 1/92 AOI-14 Loss of RHR T.S. 3.1.2 Shutdown Margin Tav _g ≤ 200 F. Generic Letter 88-17 "Loss of Decay Heat Removal"	

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<i>Section</i>	6.0	SHUTDOWN DEGRADATION
<i>Event</i>	6.1	LOSS OF SHUTDOWN SYSTEMS
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		5,6
<i>Description</i>		Note: Additional information will be provided later pending NRC/NUMARC Guidance on Shutdown EALs
<i>Basis</i>		Pending
<i>Escalation</i>		Not Applicable
<i>References</i>		NRC/NUMARC, Future guidance

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<i>Section</i>	<i>6.0</i>	SHUTDOWN SYSTEM DEGRADATION
<i>Event</i>	<i>6.2</i>	LOSS OF AC (Shutdown)
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Not Applicable
<i>Basis</i>		Loss of AC Power in Mode 5 and 6 will Not cause a declaration of a General Emergency
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92

<i>Section</i>	<i>6.0</i>	SHUTDOWN SYSTEM DEGRADATION
<i>Event</i>	<i>6.2</i>	LOSS OF AC (Shutdown)
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Not Applicable
<i>Basis</i>		Loss of AC Power in Mode 5 and 6 will Not cause a declaration of a Site Area Emergency.
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92

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<i>Section</i>	6.0	SHUTDOWN SYSTEM DEGRADATION
<i>Event</i>	6.2	LOSS OF AC (Shutdown)
<i>Classification</i>		ALERT
<i>Mode</i>		5,6, or Defuel
<i>Description</i>		<p>UNPLANNED loss of Offsite <u>And</u> Onsite AC Power for >15 Minutes</p> <p>1. 1A <u>and</u> 1B 6.9KV Shutdowns Bds de-energized for >15 minutes</p>
<i>Basis</i>		<p>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 is 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.</p> <p>Unplanned is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities.</p> <p>Fifteen (15) minutes was selected as threshold to exclude transient or momentary power losses.</p>
<i>Escalation</i>		Escalation is not applicable from this event.
<i>References</i>		<p>NUMARC/NESP-007, SU1 (expanded), Rev 2, 1/92</p> <p>T.S. 3.8.2 AC Sources - Shutdown</p> <p>T.S. 3.8.10 Distribution Systems - Shutdown</p>

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<i>Section</i>	6.0	SHUTDOWN SYSTEM DEGRADATION
<i>Event</i>	6.2	LOSS OF AC (Shutdown)
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		5,6 or Defuel
<i>Description</i>		<p>UNPLANNED loss of Offsite Power for >15 minutes (1 and 2)</p> <ol style="list-style-type: none"> 1. C and D CSSTs not available for >15 minutes 2. Either Diesel Generator is supplying power to its respective Shutdown board.
<i>Basis</i>		<p>Prolonged loss of AC power reduces required redundancy and potentially degrades the level of safety of the plant by rendering the plant more vulnerable to a complete Loss of AC Power (Station Blackout).</p> <p>Unplanned is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities.</p> <p>Fifteen (15) minutes was selected as a threshold to exclude transient or momentary power losses.</p>
<i>Escalation</i>		Loss of one additional power supply to the shutdown boards will escalate this event.
<i>References</i>		<p>NUMARC/NESP-007, SU1, Rev 2, 1/92</p> <p>T.S. 3.8.2 AC Sources - Shutdown</p> <p>T.S. 3.8.10 Distribution Systems - Shutdown</p>

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<i>Section</i>	<i>6.0</i>	SHUTDOWN SYSTEM DEGRADATION
<i>Event</i>	<i>6.3</i>	LOSS OF DC (SHUTDOWN)
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Not Applicable
<i>Basis</i>		Loss of DC Power (Shutdown) in Modes 5 and 6 will not cause a General Emergency.
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92

<i>Section</i>	<i>6.0</i>	SHUTDOWN SYSTEM DEGRADATION
<i>Event</i>	<i>6.3</i>	LOSS OF DC (SHUTDOWN)
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Not Applicable
<i>Basis</i>		Loss of DC power (Shutdown) in Modes 5-6 will not cause a Site Area Emergency.
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

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<i>Section</i>	<i>6.0</i>	SHUTDOWN SYSTEM DEGRADATION
<i>Event</i>	<i>6.3</i>	LOSS OF DC (SHUTDOWN)
<i>Classification</i>		ALERT
<i>Mode</i>		Not Applicable
<i>Description</i>		Not Applicable
<i>Basis</i>		Loss of DC power (Shutdown) in Modes 5-6 will not cause an Alert.
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007 Rev. 2, 1/92

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<i>Section 6.0</i>	SHUTDOWN SYSTEM DEGRADATION
<i>Event</i> 6.3	LOSS OF DC (SHUTDOWN)
<i>Classification</i>	UNUSUAL EVENT
<i>Mode</i>	5,6, defueled
<i>Description</i>	<p>UNPLANNED loss of the required train of DC power for >15 minutes (1 or 2)</p> <ol style="list-style-type: none"> 1. Voltage <105 VDC on 125V DC vital battery buses 1-I <u>and</u> 1-III for >15 minutes 2. Voltage <105 VDC on 125V DC vital battery buses 1-II <u>and</u> 1-IV for >15 minutes
<i>Basis</i>	<p>The purpose of this IC 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 IC 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.</p> <p>Unplanned is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities.</p> <p>The 105 volt Bus Voltage is the minimum bus voltage necessary for the operation of safety related equipment. This voltage value should incorporate a margin of at least 15 minutes of operation before the onset of inability to operate those loads. 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. For a 58 string battery set the minimum voltage is typically 1.81 Volts per cell.</p> <p>The fifteen minute threshold is utilized to exclude transient or momentary power losses.</p>
<i>Escalation</i>	Not Applicable
<i>References</i>	NUMARC/NESP-007, SU7, Rev. 2, 1/92 FSAR 8.3.2 DC Power Sources T.S. 3.8.5 DC Sources Shutdown T.S. 3.8.10 Distribution Systems-Shutdown AOI-21.1-21.4 Loss of 125V DC Vital Battery Boards

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<i>Section</i>	7.0	RADIOLOGICAL
<i>Event</i>	7.1	GASEOUS EFFLUENTS
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		All
<i>Description</i>		<p>EAB Dose resulting from an actual <u>or</u> imminent release of Gaseous Radioactivity that exceeds 1000 mrem TEDE <u>or</u> 5000 mrem Thyroid CDE for the actual <u>or</u> projected duration of the release (1 or 2 or 3)</p> <ol style="list-style-type: none"> 1. A VALID rad monitor reading exceeds the values under General in Table 7-1 for >15 minutes, unless assessment within this time period confirms that the criterion is <u>Not</u> exceeded 2. Field survey results indicate >1000 mrem/hr gamma <u>or</u> an I-131 concentration of 3.9E-6 μCi/cc at SP 3. EP dose assessment results indicate EAB dose >1000 mrem TEDE <u>or</u> >5000 mrem Thyroid CDE for the actual or projected duration of the release (Figure 7-A)
<i>Basis</i>		<p>Calculation WBNTSR115 results are included in Table 7.1</p> <p>The release rates for the determination of GE from monitor readings are calculated in the same manner as for the Site Area Emergency. The GE site release rate is equal to 10 times the release rate used for the SAE.</p>
<i>Escalation</i>		Not Applicable
<i>References</i>		<p>NUMARC/NESP-007, AGI, Rev 2, 1/92</p> <p>WBNTSR115 Rev. 7</p> <p>Main Steam System Description, (N3-1-4002)</p> <p>Calibration Factors for the Main Steam Line Radiation Monitors, (WBNAPS3-047)</p> <p>(ODCM) Offsite Dose Calculation Manual</p> <p>TI-18 Calculation Methods for Effluent Radiation Monitors</p> <p>(CTD) Chemistry Technical Document, Gaseous Effluent Radiation Monitor</p> <p>Default Set Point Determination, CR-060795-01-01</p> <p>FSAR 11.3 Gaseous Waste Systems</p>

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<i>Section</i>	7.0	RADIOLOGICAL
<i>Event</i>	7.1	GASEOUS EFFLUENTS
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		All
<i>Description</i>	<p>EAB Dose resulting from an actual <u>or</u> imminent release of Gaseous Radioactivity that exceeds 100 mrem TEDE <u>or</u> 500 mrem Thyroid CDE for the actual <u>or</u> projected duration of the release (1 or 2 or 3)</p> <ol style="list-style-type: none"> 1. A VALID rad monitor reading exceeds values under Site in Table 7-1 for >15 minutes, unless assessment within this time period confirms that the Criterion is <u>Not</u> exceeded 2. Field survey results indicate >100 mrem/hr gamma <u>or</u> an I-131 concentration of 3.9E-7 µCi/cc at the SP (Figure 7-A) 3. EP dose assessment results indicate EAB dose >100 mrem TEDE <u>or</u> >500 mrem Thyroid CDE for the actual <u>or</u> projected duration of the release. Refer to Figure 7-A. 	
<i>Basis</i>	<p>The SAE gaseous effluent EAL release rate values are those required to deliver the EAL dose in one hour. EAL release rates are back calculated from both the 100 mrem TEDE and 500 mrem CDE (thyroid) criteria, separately. The most conservative of these release rates will be used in the determination of the EAL. The meteorology, calculation methodology were used in Engineering Calculation WBNTSR115, The results of this calculation are included in Table 7.1</p>	
<i>Escalation</i>	Escalation would be based on increased release rates by a factor of 10.	
<i>References</i>	<p>NUMARC/NESP-007, AGI, Rev 2, 1/92</p> <p>WBNTSR115 Rev. 7 Main Seam System Description, (N3-1-4002) Calibration Factors for the Main Steam Line Radiation Monitors, (WBNAPS3-047) (ODCM) Offsite Dose Calculation Manual TI-18 Calculation Methods for Effluent Radiation Monitors (CTD) Chemistry Technical Document, Gaseous Effluent Radiation Monitor Default Set Point Determination, CR-060795-01-01 FSAR 11.3 Gaseous Waste Systems</p>	

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<i>Section</i>	7.0	RADIOLOGICAL
<i>Event</i>	7.1	GASEOUS EFFLUENTS
<i>Classification</i>		ALERT
<i>Mode</i>		All
<i>Description</i>	<p>Any UNPLANNED release of Gaseous Radioactivity that exceeds 200 times the ODCM Limit for >15 minutes (1 or 2 or 3)</p> <ol style="list-style-type: none"> 1. A VALID rad monitor reading exceeds values under Alert in Table 7-1 for >15 minutes, unless assessment within this time period confirms that the Criterion is <u>Not</u> exceeded 2. Field survey results indicate >10 mrem/hr gamma at the SP >15 minutes 3. EP dose assessment results indicate EAB dose >10 mrem TEDE for the duration of the release (Figure 7-A) 	
<i>Basis</i>	<p>Calculation WBNTSR115 results are included in Table 7.1 For gaseous releases, the EAL for the Alert can be determined by multiplying the NOUE noble gas monitor readings by a factor of 100.</p>	
<i>Escalation</i>	<p>Escalation would be based on dose rates greater than 100 mrem Total Body or 500 mrem child thyroid.</p>	
<i>References</i>	<p>NUMARC/NESP-007, AAI, Rev 2, 1/92 WBNTSR115 Rev. 7 Main Steam System Description, (N3-1-4002) Calibration Factors for the Main Steam Line Radiation Monitors, (WBNAPS3-047) (ODCM) Offsite Dose Calculation Manual TI-18 Calculation Methods for Effluent Radiation Monitors (CTD) Chemistry Technical Document, Gaseous Effluent Radiation Monitor Default Set Point Determination, CR060795-01-01 FSAR 11.3 Gaseous Waste Systems</p>	

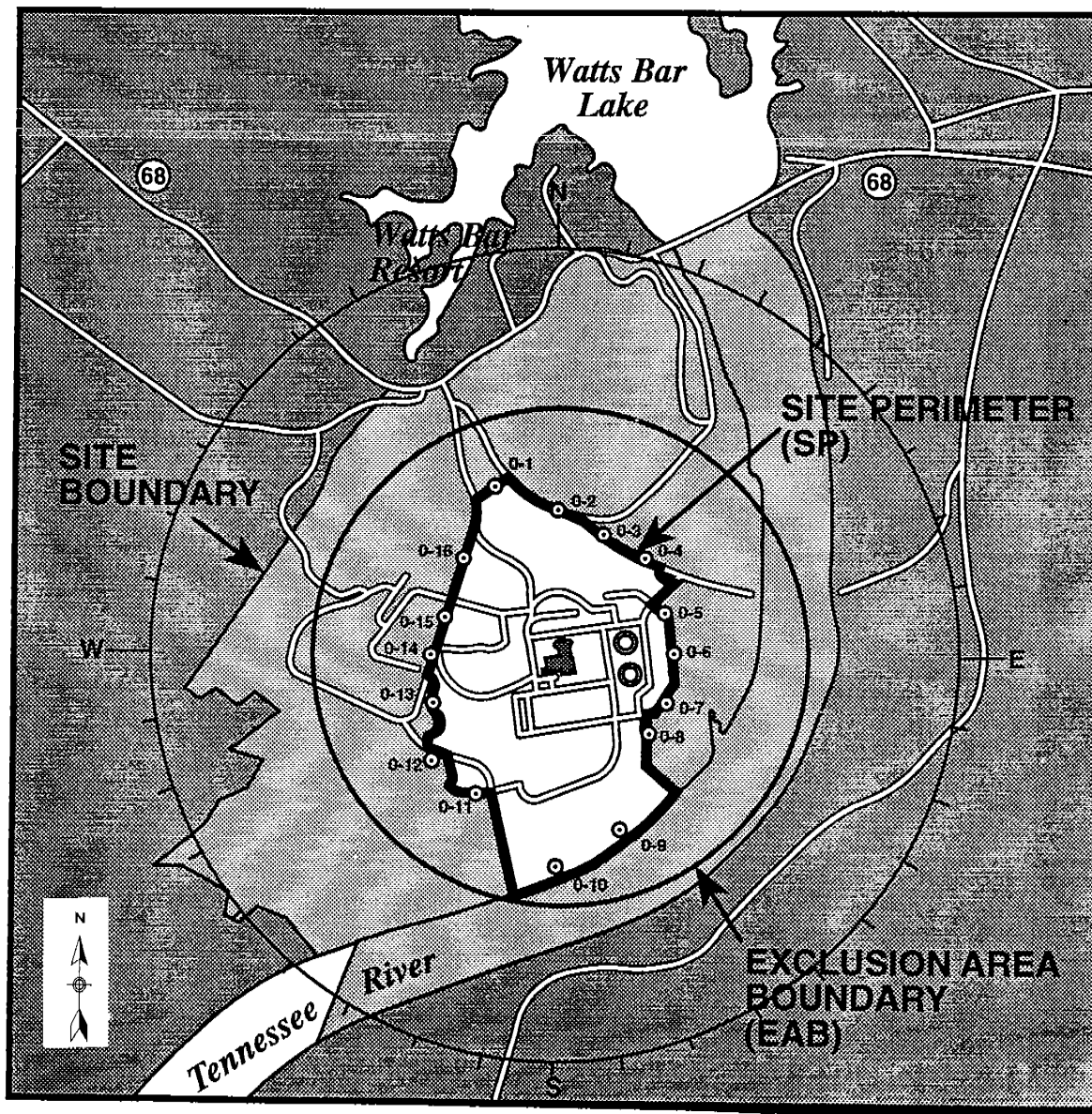
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<i>Section</i>	7.0	RADIOLOGICAL
<i>Event</i>	7.1	GASEOUS EFFLUENTS
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		All
<i>Description</i>		<p>Any UNPLANNED release of Gaseous Radioactivity that exceeds 2 times the ODCM Limit for >60 minutes (1 or 2 or 3)</p> <ol style="list-style-type: none"> 1. A VALID rad monitor reading exceeds values under Unusual Event in Table 7-1 for >60 minutes, unless assessment within this time period confirms that the Criterion is <u>Not</u> exceeded 2. Field survey results indicate >0.1 mrem/hr gamma at the SP >60 minutes 3. EP dose assessment results indicate EAB dose >0.1 mrem TEDE for the duration of the release (Figure 7-A)
<i>Basis</i>		The ODCM dose factors, meteorology, calculation methodology and the design annual releases (noble gas only) from the WBN FSAR were used in Engineering Calculation WBNTSR115, The results of this calculation are included in Table 7.1
<i>Escalation</i>		Escalation would be based on increasing the magnitude of the release by a factor of 100.
<i>References</i>		NUMARC/NESP-007, AUI, Rev 2, 1/92 Main Steam System Description, (N3-1-4002) Calibration Factors for the Main Steam Line Radiation Monitors, (WBNAPS3-047) (ODCM) Offsite Dose Calculation Manual WBNTSR115 Rev. 7 TI-18 Calculation Methods for Effluent Radiation Monitors (CTD) Chemistry Technical Document, Gaseous Effluent Radiation Monitor Default Set Point Determination, CR-060795-01-01 FSAR 11.3 Gaseous Waste Systems

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Figure 7-A

EXCLUSION AREA/SITE BOUNDARY



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<i>Section</i>	<i>7.0</i>	RADIOLOGICAL
<i>Event</i>	<i>7.2</i>	LIQUID EFFLUENTS
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Not Applicable
<i>Basis</i>		Not Applicable
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92

<i>Section</i>	<i>7.0</i>	RADIOLOGICAL
<i>Event</i>	<i>7.2</i>	LIQUID EFFLUENTS
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Not Applicable
<i>Basis</i>		Not Applicable
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92

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<i>Section</i>	7.0	RADIOLOGICAL
<i>Event</i>	7.2	LIQUID EFFLUENTS
<i>Classification</i>		ALERT
<i>Mode</i>		All
<i>Description</i>	<p>Any UNPLANNED release of Liquid Radioactivity that exceeds 200 times the ODCM Limit for >15 minutes (1 or 2)</p> <ol style="list-style-type: none"> 1. A VALID rad monitor reading exceeds the values under Alert in Table 7-1 for >15 minutes, unless assessment within this time period confirms that the Criterion is <u>Not</u> exceeded 2. Sample results exceed 200 times the ODCM limit value for an unmonitored release of liquid radioactivity >15 minutes in duration 	
<i>Basis</i>	<p>Calculation WBNTSR115 results are included in Table 7.1 For liquid release, the monitor values for the Alert classification values, but will be a factor of 100 higher than the NOUE values.</p>	
<i>Escalation</i>	Not Applicable	
<i>References</i>	<p>NUMARC/NESP-007, AA2, Rev 2, 1/92 WBNTSR115 Rev. 7 (ODCM) Offsite Dose Calculation Manual 10 CFR 20 TI-18 Calculation Methods for Effluent Radiation Monitors</p>	

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<i>Section</i>	<i>7.0</i>	RADIOLOGICAL
<i>Event</i>	<i>7.2</i>	LIQUID EFFLUENTS
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		All
<i>Description</i>		<p>Any UNPLANNED release of Liquid Radioactivity to the Environment that exceeds 2 times the ODCM Limit for >60 minutes (1 or 2)</p> <ol style="list-style-type: none"> 1. A VALID rad monitor reading exceeds the values under UE in Table 7-1 for >60 minutes, unless assessment within this time period confirms that the Criterion is <u>Not</u> exceeded 2. Sample results exceed 2 times the ODCM limit value for an unmonitored release of liquid radioactivity >60 minutes in duration
<i>Basis</i>		For liquid releases, the ODCM limit is equal to 10 times the Effluent Concentration Limits (ECL) listed in 10 CFR Part 20 Appendix B, Table 2, Column 2. The calculation methodology is in Engineering Calculation WBNTSR115, The results of this calculation are included in Table 7.1.
<i>Escalation</i>		Escalation would be based on an UNPLANNED release exceeding 200 times the ODCM limit for greater than 15 minutes.
<i>References</i>		NUMARC/NESP-007, AU2, Rev 2, 1/92 (ODCM) Offsite Dose Calculation Manual 10 CFR 20 WBNTSR115 Rev. 7 TI-18 Calculation Methods for Effluent Radiation Monitors

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Condensate Demineralizer not in use, Direct Steam Generator Blowdown release with dilution- Liquid, Case 2						
	ICS Screen	Units	NOUE	Alert	Site Area Emergency	General Emergency
Total Site (Liquid)		$\mu\text{Ci/ml}^{(2)}$	6.05E-03	6.05E-01	N/A	N/A
<i>Radwaste Monitor</i> 0-RE-90-122	4RM2	cpm	3.46E+06	Offscale ⁽¹⁾	N/A	N/A
<i>SGBD</i> 1-RE-90-120,121	4RM2	cpm	1.98E+06	Offscale ⁽¹⁾	N/A	N/A
<i>Condensate Demin</i> 0-RE-90-225	4RM2	cpm	3.46E+06	Offscale ⁽¹⁾	N/A	N/A
<i>Turbine Bldg Sump</i> 0-RE-90-212	4RM2	cpm	3.35E+04	3.35E+06	N/A	N/A
Release Duration		Minutes	60	15	15	15
Assessment Method: ICS or radiation monitor (RM) readings in the control room or local indication as necessary						
Steam Generator Blowdown only, no treatment or dilution- Liquid, Case 3						
	ICS Screen	Units	NOUE	Alert	Site Area Emergency	General Emergency
Total Site (Liquid)		$\mu\text{Ci/ml}^{(2)}$	1.16E-04	1.16E-02	N/A	N/A
<i>Radwaste Monitor</i> 0-RE-90-122	4RM2	cpm	5.80E+04	5.80E+06	N/A	N/A
<i>SGBD</i> 1-RE-90-120,121	4RM2	cpm	5.80E+04	5.80E+06	N/A	N/A
<i>Condensate Demin</i> 0-RE-90-225	4RM2	cpm	5.80E+04	5.80E+06	N/A	N/A
<i>Turbine Bldg Sump</i> 0-RE-90-212	4RM2	cpm	7.44E+04	7.44E+06	N/A	N/A
Release Duration		Minutes	60	15	15	15
Assessment Method: ICS or radiation monitor (RM) readings in the control room or local indication as necessary						

- (1) The calculated value is offscale of the upper range for this detector and other confirmatory data is required for event classification. The maximum output which can be read is 1E+07 cpm.
- (2) These EALs are based on the assumption that an emergency release is restricted to one pathway from the plant. In all cases, the total site EAL is the limiting value. Therefore, in the case where there are multiple release paths from the plant, it is the total release EAL (obtained from ICS) that will determine whether an emergency classification is warranted.
- (3) The Eberline channel (1-RM-90-450) reads out in cpm in the MCR and is used to calculate and process the output of RM-90-404A and RM-90-404B to provide the indication (in $\mu\text{Ci/cc}$) to ICS that would be used to determine the threshold value for a radioactivity release via this pathway.
- (4) These unit values are based on flow rates through one PORV of 970,000 lb/hr at 1185 psig 600 degrees F. Before using these values, ensure a release to the environment is ongoing, (e.g., PORV).

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<i>Section</i>	<i>7.0</i>	RADIOLOGICAL
<i>Event</i>	<i>7.3</i>	RADIATION LEVELS
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" or "Gaseous Effluents" (7.1)
<i>Basis</i>		Not Applicable
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev 2, 1/92

<i>Section</i>	<i>7.0</i>	RADIOLOGICAL
<i>Event</i>	<i>7.3</i>	RADIATION LEVELS
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to the "Fission Product Barrier Matrix" or "Gaseous Effluents" (7.1)
<i>Basis</i>		Not Applicable
<i>Escalation</i>		Escalation may be based on "Fission Product Barrier Challenges" or Gaseous Effluent levels.
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

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<i>Section</i>	7.0	RADIOLOGICAL
<i>Event</i>	7.3	RADIATION LEVELS
<i>Classification</i>		ALERT
<i>Mode</i>		All
<i>Description</i>	<p>UNPLANNED increases in Radiation levels within the Facility that impedes Safe Operations <u>or</u> establishment <u>or</u> maintenance of Cold Shutdown (1 or 2)</p> <ol style="list-style-type: none"> 1. VALID area Radiation Monitor readings <u>or</u> survey results exceed 15 mrem/hr in the Control Room <u>or</u> CAS. 2. (a and b) <ol style="list-style-type: none"> a. VALID area radiation monitor readings exceed values listed in Table 7-2. b. Access restrictions impede operation of systems necessary for Safe Operation <u>or</u> the ability to establish Cold Shutdown <p>See UNUSUAL EVENT Note</p>	
<i>Basis</i>	<p>This IC 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.</p> <p>EAL #1 applies to areas that are manned continuously. The value of 15 mrem/hr has been determined to be representative of the CRITERION. This value was obtained from Section III.D.3 of NUREG-0737, "<u>Clarification of TMI Action Plan Requirements</u>", which specified a criterion of 15 mR/hr averaged over the assumed 30 day duration of the accident. The value was based on the GDC 19 criterion of 5 rem for the duration of the accident, with adjustment for occupancy factors. The value is used here without averaging, as a 30 day duration implies an event potentially more significant than an ALERT.</p> <p>The Control Room, and the Central Alarm Station (CAS) should be continuously manned. Thus, the 15 mR/hr value applies to these facilities.</p> <p>EAL #2 applies to areas that require infrequent access. Table 7-2 tabulates the areas identified for WBN and the associated radiation level, above which access is considered impeded. The areas were selected on the basis of the relative need for access. The specified radiation levels are such that normal radiation exposure control measures intended to maintain doses within normal 10 CFR 20 occupational exposure guidelines would impede necessary access.</p>	

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<i>Section</i>	<i>7.0</i>	RADIOLOGICAL
<i>Event</i>	<i>7.3</i>	RADIATION LEVELS
<i>Classification</i>		ALERT (continued)
<i>Mode</i>		All
<i>Basis</i> (continued)		<p>This IC is not meant to apply to increases in the containment dome radiation monitors as these are events which are addressed in the fission product barrier matrix ICs. 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, component venting, etc.).</p> <p>An indication or report or condition is considered to be VALID when it is conclusively verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel.. Implicit in this definition is the need for timely assessment, i.e., within 15 minutes.</p> <p>A release of radioactivity is UNPLANNED if the release has not been authorized by a Discharge Permit (DP). Implicit in this definition are unintentional releases, unmonitored releases, or planned releases that exceed a condition specified on the DP, e.g. alarm setpoints, minimum dilution flow, minimum release times, maximum release rates, and/or discharge of incorrect tank.</p>
<i>Escalation</i>		Escalation may be based on "Fission Product Barrier Challenges" or Gaseous Effluent levels.
<i>References</i>		NUMARC/NESP-007, AA3, Rev 2, 1/92 WBN QDCN 20764 B - Radiation Monitor Readings for the REP, WBN TSR-044, R0 (B18891227 254) Required Range and Accuracy of the WBN Area Radiation Monitors, WBN TSR-077, R2 (B18 920727 317) Radiation Zones.

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<i>Section</i>	7.0	RADIOLOGICAL
<i>Event</i>	7.3	RADIATION LEVELS
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		All
<i>Description</i>	<p>UNPLANNED increases in Radiation levels within the Facility</p> <p>1. VALID area Radiation Monitor readings increase by a factor of 1000 over normal levels.</p> <p>Note: In Either The UE or ALERT EAL, the SED must determine the cause of Increase in Radiation Levels and Review Other INITIATING/CONDITIONS for Applicability (e.g., a dose rate of 15 mrem/hr in the Control Room could be caused by a release associated with a DBA).</p>	
<i>Basis</i>	<p>This IC 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.</p> <p>An indication or report or condition is considered to be VALID when it is conclusively verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel.. Implicit in this definition is the need for timely assessment, i.e., within 15 minutes.</p> <p>A release of radioactivity is UNPLANNED if the release has not been authorized by a Discharge Permit (DP). Implicit in this definition are unintentional releases, unmonitored releases, or planned releases that exceed a condition specified on the DP, e.g. alarm setpoints, minimum dilution flow, minimum release times, maximum release rates, and/or discharge of incorrect tank.</p>	
<i>Escalation</i>	Escalation will be based on the inability to access certain operating stations or equipment needed to establish or maintain Cold Shutdown.	
<i>References</i>	<p>NUMARC/NESP-007, AU2, Rev 2, 1/92</p> <p>WBN QDCN 20764 B - Radiation Monitor Readings for the REP.</p> <p>WBN TSR-044, R0 (B18 891227 254) Required Range and Accuracy for the WBN Area Radiation Monitors.</p> <p>WBN TSR-077, R2 (B18 920727 317) Radiation Zones.</p>	

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Table 7-2

ALERT - RADIATION LEVELS

Monitor No.	Location Building and Elevation		Monitor Reading 1
1&2-RE-90-1	Auxiliary	El.757.0 (spent fuel pool)	2.5×10^3 mR/hr
1 &2-RE-90-2	Auxiliary	El.757.0 (personnel air lock)	2.5×10^3 mR/hr
0-RE-90-3	Auxiliary	El.729.0 (waste pac. area)	2.5×10^3 mR/hr
0-RE-90-4	Auxiliary	El.713.0 (decon room)	2.5×10^3 mR/hr
0-RE-90-5	Auxiliary	El.737.0 (spt. fuel pool pmp. ar.)	2.5×10^3 mR/hr
1&2-RE-90-6	Auxiliary	El.737.0 (comp. cl. wtr. ht. ex. ar.)	2.5×10^3 mR/hr
1&2-RE-90-7	Auxiliary	El.713.0 (sample room)	2.5×10^3 mR/hr
1&2-RE-90-8	Auxiliary	El.713.0 (aux. feed pump area)	2.5×10^3 mR/hr
0-RE-90-9	Auxiliary	El.692.0 (wst. cond. evap. tk. ar.)	2.5×10^3 mR/hr
1&2-RE-90-10	Auxiliary	El.692.0 (cvcs area)	2.5×10^3 mR/hr
0-RE-90-11	Auxiliary	El.676.0 (ctmt.spry. & rhr pmp ar.)	2.5×10^3 mR/hr
1-RE-90-61	Reactor	El.713.0 (RB low. cmpt .inst. m.)	2.5×10^3 mR/hr
0-RE-90-230	Turbine	El.685.0 (conden. demin.)	2.5×10^3 mR/hr
0-RE-90-231	Turbine	El.685.0 (conden. demin.)	2.5×10^3 mR/hr

NOTE: (1) These monitors read out in mR/hr. It is assumed that this is equivalent to mrem/hr.

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<i>Section</i>	<i>7.0</i>	RADIOLOGICAL
<i>Event</i>	<i>7.4</i>	FUEL HANDLING
<i>Classification</i>		GENERAL EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to Gaseous Effluents (event 7.1)
<i>Basis</i>		The basis for a General Emergency is primarily the extent and severity of Gaseous Effluents (event 7.1)
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

<i>Section</i>	<i>7.0</i>	RADIOLOGICAL
<i>Event</i>	<i>7.4</i>	FUEL HANDLING
<i>Classification</i>		SITE AREA EMERGENCY
<i>Mode</i>		Not Applicable
<i>Description</i>		Refer to Gaseous Effluents (event 7.1)
<i>Basis</i>		The basis for a Site Area Emergency is primarily the extent and severity of Gaseous Effluents (event 7.1)
<i>Escalation</i>		Not Applicable
<i>References</i>		NUMARC/NESP-007, Rev. 2, 1/92

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<i>Section</i>	7.0	RADIOLOGICAL
<i>Event</i>	7.4	FUEL HANDLING
<i>Classification</i>		ALERT
<i>Mode</i>		All
<i>Description</i>	<p>Major damage to irradiated fuel; <u>or</u> loss of Water Level that has or will uncover irradiated fuel outside the Reactor Vessel (1 and 2)</p> <ol style="list-style-type: none"> 1. VALID Alarm on 0-RE-90-101B <u>or</u> 0-RE-90-102 <u>or</u> 0-RE-90-103 <u>or</u> 1-RE-90-130/131 <u>or</u> 1-RE-90-112 <u>or</u> 1-RE-90-400 <u>or</u> 2-RE-90-400 2. (a or b) <ol style="list-style-type: none"> a. Plant personnel report damage of irradiated fuel sufficient to rupture fuel rods b. Plant personnel report water Level drop has <u>or</u> will exceed makeup capacity such that irradiated fuel will be uncovered 	
<i>Basis</i>	<p>The major concern of the EAL is a fuel handling accident or loss of water covering spent fuel. Events of this type could cause an increase in radioactivity readings and potentially a release to the environment. Offsite doses during these accidents would be below the EPA Protective Action Guidelines and the classification of an Alert is therefore appropriate.</p> <p>Monitoring radiation on the refueling floor and containment is by Particulant Iodine Gas Monitors and Area Monitors. Values for these monitors are set so as to not exceed safety limits and to ensure that the Design Basis does not exceed limits referenced in 10 CFR 20.</p> <p>An indication or report or condition is considered to be VALID when it is conclusively verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel. Implicit in this definition is the need for timely assessment, i.e., within 15 minutes.</p>	
<i>Escalation</i>	Escalation would occur by offsite dose rates. See Gaseous Effluents (7.1)	
<i>References</i>	NUMARC/NESP-007, AA2, Rev. 2, 1/92 AOI-29 Dropped or Damaged Fuel or Refueling Cavity Seal Failure NRC Information Notice No. 90-08, Kr-85 Hazards from Decayed Fuel EPA 400-R-92-001, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents FSAR 15.5.6 Environmental Consequences of a Postulated Fuel Handling Accident T.S. 3.9.4 Containment Penetrations T.S. 3.7.12 Auxiliary Building Gas Treatment System	

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<i>Section</i>	<i>7.0</i>	RADIOLOGICAL
<i>Event</i>	<i>7.4</i>	FUEL HANDLING
<i>Classification</i>		UNUSUAL EVENT
<i>Mode</i>		All
<i>Description</i>		<p>UNPLANNED loss of water level in Spent Fuel Pool <u>or</u> Reactor Cavity <u>or</u> Transfer Canal with fuel remaining covered (1 and 2 and 3)</p> <p>1. Plant personnel report water level drop in Spent Fuel Pool <u>or</u> Reactor Cavity, <u>or</u> Transfer Canal</p> <p>2. VALID alarm on 0-RE-90-102 <u>or</u> 0-RE-90-103 <u>or</u> 1-RE-90-59 <u>or</u> 1-RE-90-60</p> <p>3. Fuel remains covered with water</p>
<i>Basis</i>		<p>The term UNPLANNED refers to unplanned actions resulting from either equipment malfunctions or operator error that results in a decreasing water level in the Spent Fuel Pool, Reactor Cavity or Transfer Canal.</p> <p>Unplanned is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities.</p> <p>The main concern of this EAL is the loss of water covering spent fuel and the potential of increased doses to plant staff. This event has a long lead time relative to the potential for a radiological release outside the site boundary, thus the impact to public health and safety is very low. Classifications of an Unusual Event is warranted as a precursor to a more serious event.</p> <p>An indication or report or condition is considered to be VALID when it is conclusively verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel.. Implicit in this definition is the need for timely assessment, i.e., within 15 minutes.</p>
<i>Escalation</i>		Escalation of this event would be based on uncovering an irradiated fuel assembly or indications of high radiation levels on the refueling floor.
<i>References</i>		<p>NUMARC/NESP-007, AU2, Rev. 2, 1/92</p> <p>AOI-29 Dropped or Damaged Fuel or Refueling Cavity Seal Failure</p> <p>NRC Information Notice No. 90-08, Kr-85 Hazards from Decayed Fuel</p> <p>EPA 400-R-92-001, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents</p> <p>FSAR 15.5.6 Environmental Consequences of a Postulated Fuel Handling Accident</p> <p>T.S. 3.9.4 Containment Penetrations</p> <p>T.S. 3.7.12 Auxiliary Building Gas Treatment System</p>

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C.5 SITE EMERGENCY ORGANIZATION

WBN maintains an organization capable of responding to a radiological emergency. The on-shift staffing for response to emergencies is shown on Figure 1-C.

The 90 minute shift augmentation positions are shown in Figure 2C and 3C.

Within the on-shift team, Advanced Radiation Workers (ARWs) are utilized to accomplish some radiological assessment needs.

- C.5.1 The Unit Shift Operations Staff is manned by qualified Operations personnel that meet the requirements established in site Technical Specifications.

Concerning Assistant Units Operators, (AUOs) a normal shift compliment is typically (7) AUOs. A minimum of (5) AUOs are on shift at all times.

- C.5.2 The Fire Shift Operations Staff is manned by qualified personnel that meet the requirements established in the site Fire Protection Report.

- C.5.3 The RP/Chemistry Staff is manned by qualified personnel that meet the requirements established in the site Technical Specifications.

- C.5.4 The Security Shift Staff is manned by qualified personnel that meet the requirements established in the Physical Security Plan.

- C.5.5 The Maintenance Shift Staff is manned by a multi-discipline staffing of personnel who are available to respond to postulated events that could involve one or more of the three commonly recognized areas of : mechanical, electrical, or instrumentation.

Each maintenance team member has a background in one or more discipline(s).

Team members may perform cross-disciplinary work if they have a background to perform the respective task/activity. Team members may include representatives from the following plant work groups: maintenance management, maintenance craft, operations, and maintenance planning.

- C.5.6 The Maintenance team shift managers and team foremen are considered qualified to provide a first response capability within the bounds of the actions required during the initial phase (in the first 90 minutes) of a radiological emergency for their related background(s). Therefore, staffing credit is taken for the foreman or shift manager, when necessary, to satisfy minimum staffing requirements.

The electrical skills necessary during a radiological emergency are considered to be satisfied by an individual with instrumentation skills as indicated by table B-1 in NUREG-0654.

- C.5.7 Upon activation of the OSC (Figure 5-C), the Maintenance onshift staffing falls under the control of the OSC.

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C.5.8 The on-shift composition listed in Figure 1-C may be less than the minimum requirements for a period of time in order to accommodate unexpected absences of personnel provided action is taken to restore the composition to within the following time requirements:

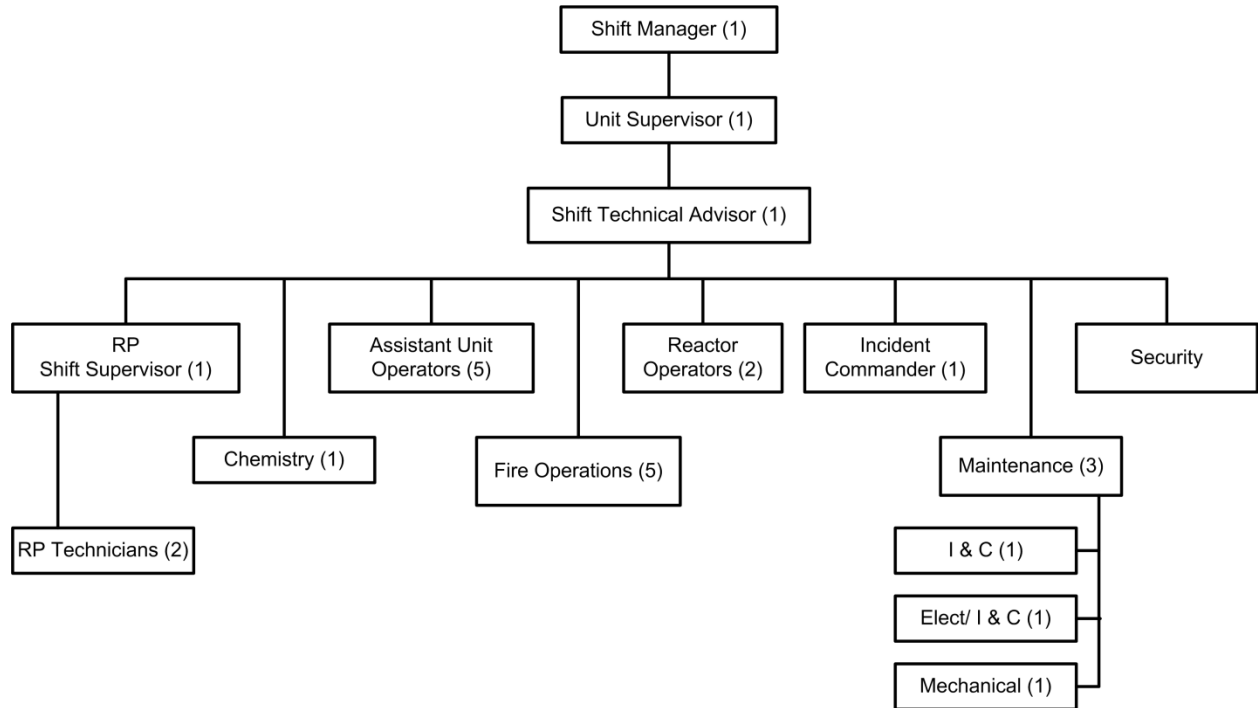
- Unit Shift Operations Staff (per Technical Specifications)
- Fire Shift Operations Staff (per Fire Protection Report)
- Security Shift Staff (per the Physical Security Plan)

- RP Shift Staff (one technician, per Technical Specifications)
(remaining staff, notified within two hour to arrive, as soon as possible)

- Chemistry and Rad Chem Support Shift Staffing (notified within two hour to arrive, as soon as possible)

- Maintenance Shift Staff (notified within two hour to arrive, as soon as possible)

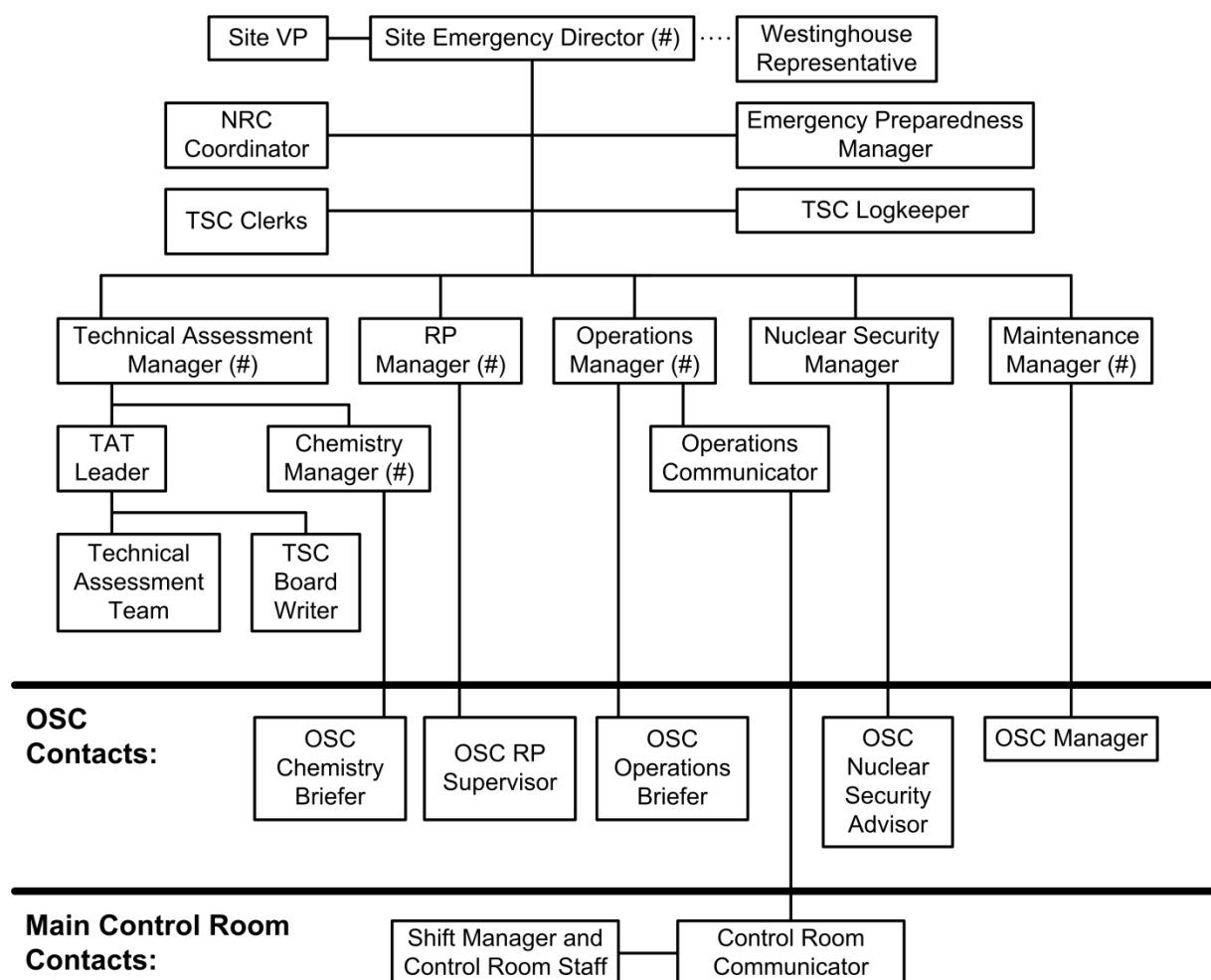
**FIGURE 1-C
ON-SHIFT STAFFING**



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C.5.9 The TSC emergency response positions are described in WBN EPIP-6, "Activation and Operation of the Technical Support Center. Figure 2-C provides the typical staffing of the TSC. (#) denotes 90 minute minimum staffing position(s).

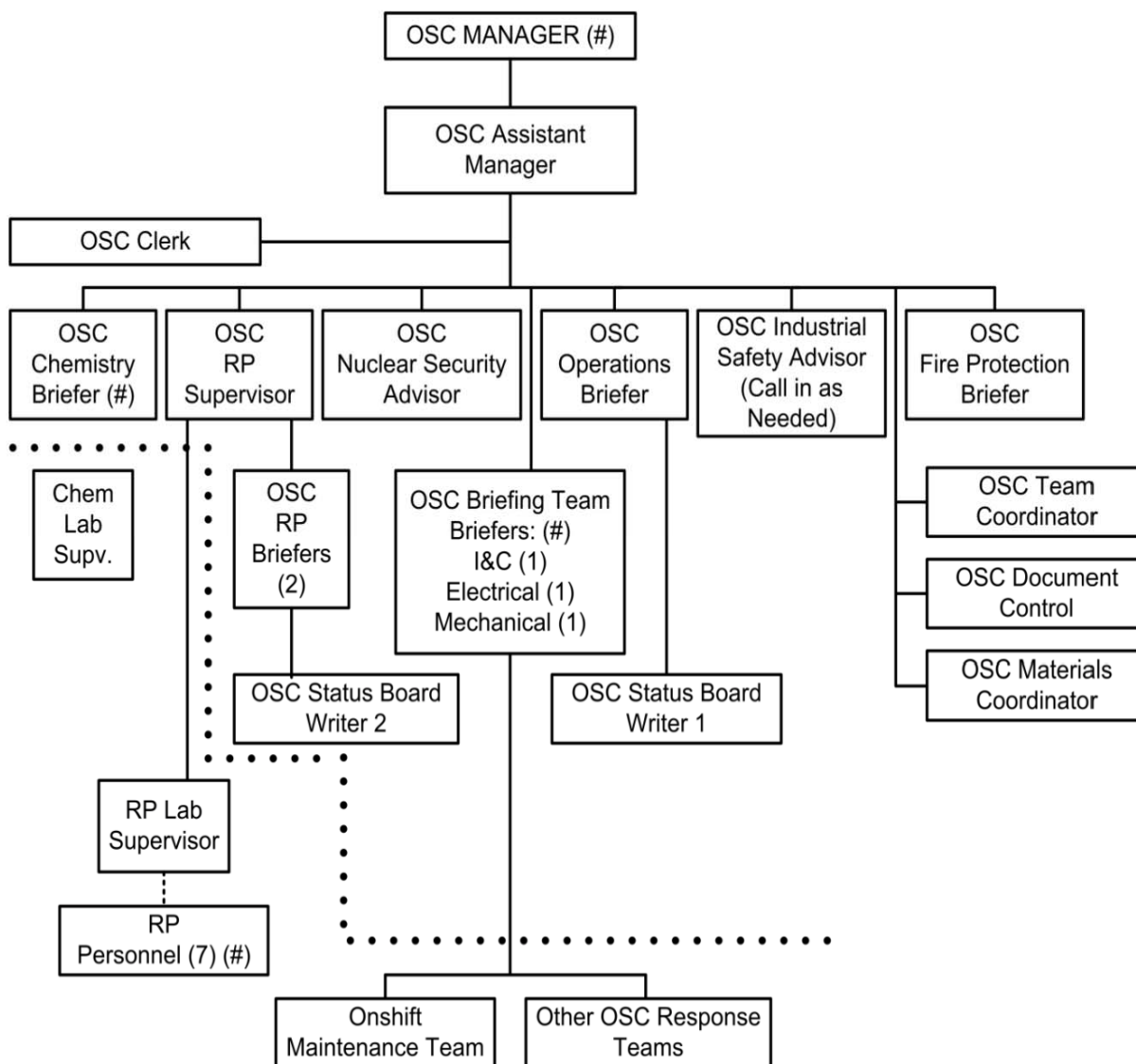
**Figure 2-C
Technical Support Center (TSC)
WBN EMERGENCY RESPONSE ORGANIZATION**



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C.5.10 The Operations Support Center positions are described in WBN EPIP-7 "Activation and Operation of the Operations Support Center" Figure 3-C provides the typical staffing of the OSC.
(#) denotes 90 minute minimum staffing position(s).

**Figure 3-C
OPERATIONS SUPPORT CENTER ORGANIZATION**



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C.5.11 Site Vice President

The Site Vice President serves as a corporate interface for the SED, relieving him from duties which could distract from the SED's primary purpose of plant operations and accident mitigation activities. The Site Vice President shall provide assistance in the following areas:

1. Provides TVA policy direction to the Site Emergency Director.
2. Directs the site resources to support the Site Emergency Director in the accident mitigation activities.
3. Provides direct interface on overall site response activities with:
 - a. NRC, FEMA, or other Federal organizations responding to the site.
 - b. CECC Director.
 - c. Onsite media.
4. At his discretion, may provide interface at the appropriate offsite location on the overall site response activities with:
 - a. State and local agencies.
 - b. NRC region/corporate.
 - c. Joint Information Center.
5. Provides support to other emergency operation centers as necessary.

C.5.12 Site Emergency Director

1. Directs onsite emergency accident mitigation activities.
2. Consults with CECC Director and Site Vice President on significant events and their related impacts.
3. Initiates onsite protective actions.
4. Coordinates accident mitigation actions with NRC.
5. Initiates long-term 24-hour accident mitigation operations.
6. Prior to the CECC being staffed, makes recommendations for protective actions (if necessary) to State and local agencies through the Operations Duty Specialist. This responsibility cannot be delegated except to the CECC Director after the CECC is operational.

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7. Responsible for determining the emergency classification. This responsibility cannot be delegated.
8. Approves or authorizes emergency doses. This responsibility cannot be delegated.

C.5.13 Operations Manager

1. Directs operational activities.
2. Informs Site Emergency Director of plant status and operational problems.
3. Assures the control room is aware of the accident assessment and response.
4. Recommends solutions and mitigating action for operational problems.

C.5.14 Technical Assessment Manager

1. Directs onsite effluent assessment.
2. Directs activities of technical assessment team.
3. Projects future plant status based on present plant conditions.
4. Keeps assessment team informed of plant status.
5. Provides information, evaluations, and projections to Site Emergency Director.
6. Coordinates assessment activities with the CECC plant assessment team.
7. Establishes and maintains a status of significant plant problems.

C.5.15 TSC Clerks

1. Answer telephones.
2. Distribute plant parameter data sheets.
3. Maintain TSC organization board.
4. Operate facsimile machine.
5. Other duties as assigned by Site Emergency Director.

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C.5.16 Nuclear Security Manager

1. Directs activities of Nuclear Security Services personnel.
2. Controls access to site and control rooms.
3. Reports on site accountability/evacuation as defined in WBN-EIPs.

C.5.17 Rad Protection Manager

1. Directs and/or performs assessment of inplant and onsite radiological conditions.
2. Directs onsite RP activities.
3. Coordinates additional RP support with CECC Radiological Assessment Manager.
4. Makes recommendations for protective actions for onsite personnel.
5. Maintains status map of offsite radiological conditions.
6. Coordinates assessment of radiological conditions offsite with CECC Radiological Assessment Coordinator.
7. Maintains in-plant radiation status board.
8. Authorizes issue of KI to onsite personnel.
9. Makes recommendations to the Site Emergency Director for personnel entry to radiological hazardous environment.

C.5.18 Chemistry Manager

1. Coordinates assessment of radioactive effluents with CECC Rad Assessment Coordinator.
2. Coordinates post-accident sampling activities.
3. Performs release rate calculations as needed.
4. Determines impact of incident on radwaste and various effluent treatment systems.

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C.5.19 NRC Coordinator

1. Acts as primary liaison with onsite NRC personnel.
2. Updates NRC personnel on plant status.
3. Provides information requests from NRC to TSC personnel.

C.5.20 Operations Communicator

1. Provides operational knowledge for status evaluation of plant systems.
2. Provides advice regarding technical specifications, system response, safety limits, etc.
3. Assists in development of recommended solutions to developing problems.
4. Serves as the control room - TSC - OSC link.

C.5.21 Emergency Preparedness Manager

1. Advises Site Emergency Director regarding overall radiological emergency plan, use of implementing procedures, emergency equipment availability, and coordination with CECC.
2. Confirms TSC is operating properly.

C.5.22 Technical Assessment Team

1. Prepares and provides periodic current assessments on plant conditions and provides this information to the CECC plant assessment team.
2. Projects future plant status based on present plant conditions.
3. Provides technical support to plant operations on mitigating actions.

C.5.23 OSC Manager

1. Directs repairs and corrective actions in coordination with the TSC.
2. Performs damage assessment.
3. Directs activities of Operations Support Center.
4. Coordinates maintenance teams and ensures they have received proper briefings and are accompanied by a RP technician, as necessary.

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C.5.24 OSC Assistant Manager

1. Oversees the operations of OSC teams.
2. Maintain communications with the TSC.
3. Maintains team tracking boards.
4. Assigns TSC tasks to team briefers.

C.5.25 OSC RP Supervisor

1. Directs activities of the RP lab.
2. Ensure RP coverage of damage repair teams.
3. Verify habitability of the TSC, OSC, and Control Room.
4. Briefs the OSC Manager and TSC on RP status.

C.5.26 OSC Briefing Teams

1. Provide mechanical, electrical, and instrumentation technical expertise.
2. Evaluate task conditions and provide methods best suited to safely perform an assignment.
3. Track OSC teams in the field.
4. Debrief OSC teams after task completion.

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C.6 EMERGENCY RESPONSE FACILITIES, EQUIPMENT, AND SUPPLIES

Specific plant areas, facilities, and equipment are selected and provided for use during a radiological emergency. The preselection, allocation, and inclusion of emergency facilities assure that needed services and equipment are available for use during emergency conditions.

C.6.1 Technical Support Center (TSC)

A specific area (adjacent to the relay room) in the Control Building at elevation 755' is designated for use as the TSC. The room is provided with communication capabilities to plant areas and areas external to the plant. The communication facilities include TVA and Bell System telephones, NRC Emergency Notification System and Health Physics Network, access to a paging-intercom system, and two-way radio. This room is sufficiently shielded to ensure occupancy during an emergency and is designed to be continuously habitable during all radiological emergencies. All ventilating and air-conditioning facilities have redundant or backup systems. Toilet facilities are available on the same elevation.

The diesel generators will provide emergency power when there is a loss of normal ac power, and cooling water for the air-conditioning equipment is taken from the essential raw cooling water system. Figure 4-C shows a detailed TSC layout.

Meteorological information is available in the TSC, OSC and in the Main Control Room and includes wind speed, direction and temperature differences between the 10-meter, 46-meter and 91-meter tower elevations. This information is utilized in the sites initial dose projection procedure, WBN EPIP-13. Should the Met tower be unavailable, WBN EPIP-13 provides backup methods for acquiring the data. Also available in the TSC, OSC and Main Control Room is information from the onsite radiation monitors and radio capabilities to relay information from the WBN Radiological Monitoring Van at any of the 16 site radiological monitoring survey points on the site perimeter.

C.6.2 Operations Support Center (OSC)

The role of the OSC is to provide assembly areas for operations support personnel during an emergency situation which are under the supervision of the OSC Manager or a designated alternate. The OSC is located on Elevation 713' adjacent to the RP Lab. It contains emergency team briefing areas and additional space provided in the adjacent hallway and adjoining rooms for staging, briefing rooms for staging, briefing and dispatching maintenance teams. The Alternate OSC is located in the Outage Control Center (OCC), with additional space provided in the nearby Plant Assembly Room for staging, briefing and dispatching maintenance teams. The OSC is provided with telephone and radio communications. Figures 5-C and 6-C show the OSC areas. Respiratory protective devices, protective clothing, portable lighting, other protective equipment and tools are available, as needed.

Figure 4-C
TECHNICAL SUPPORT CENTER
CONTROL BLDG EI 755'
FACILITY DIAGRAM

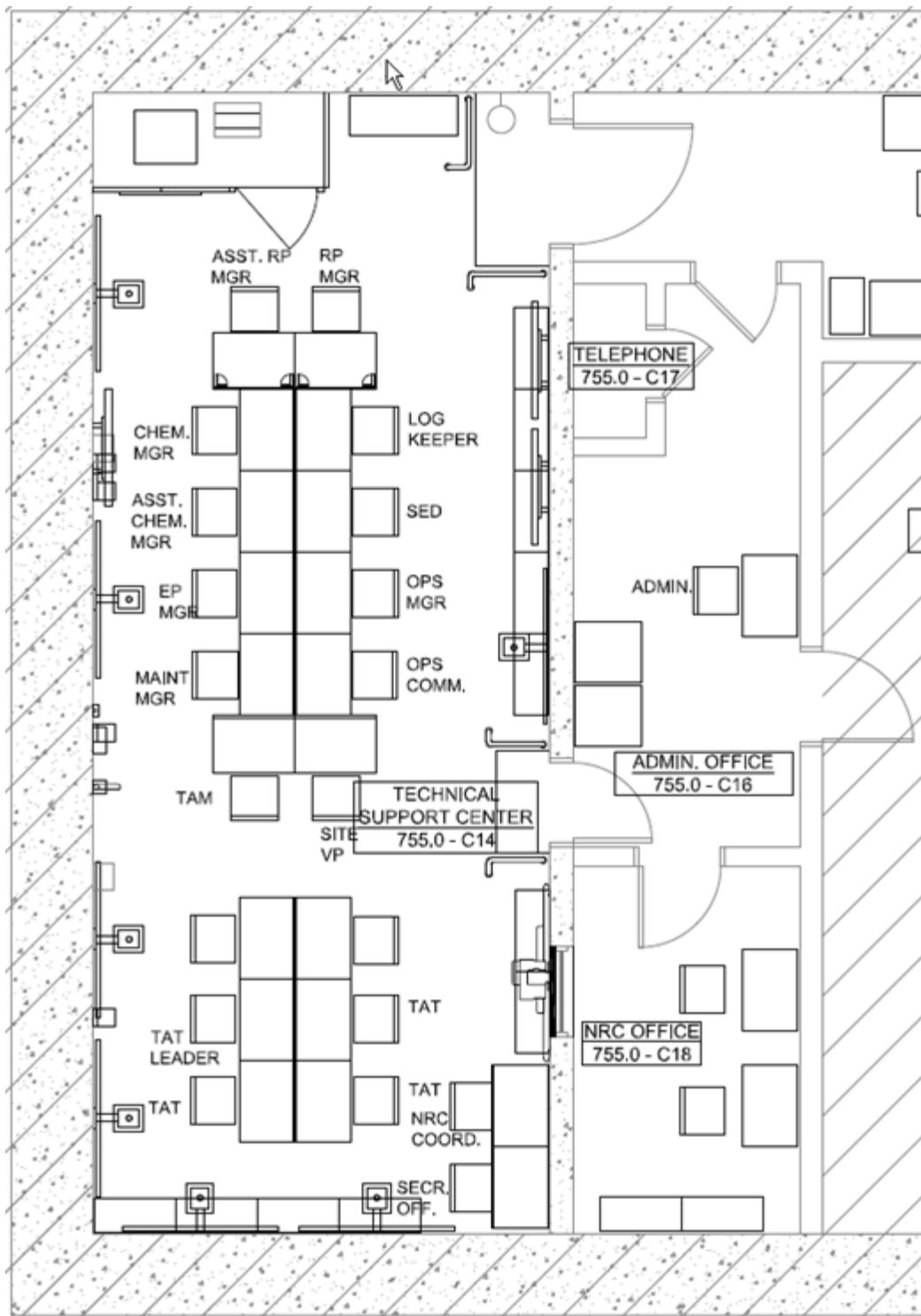


FIGURE 5-C
OPERATIONS SUPPORT CENTER
SERVICE BLDG EI 713'
FACILITY DIAGRAM

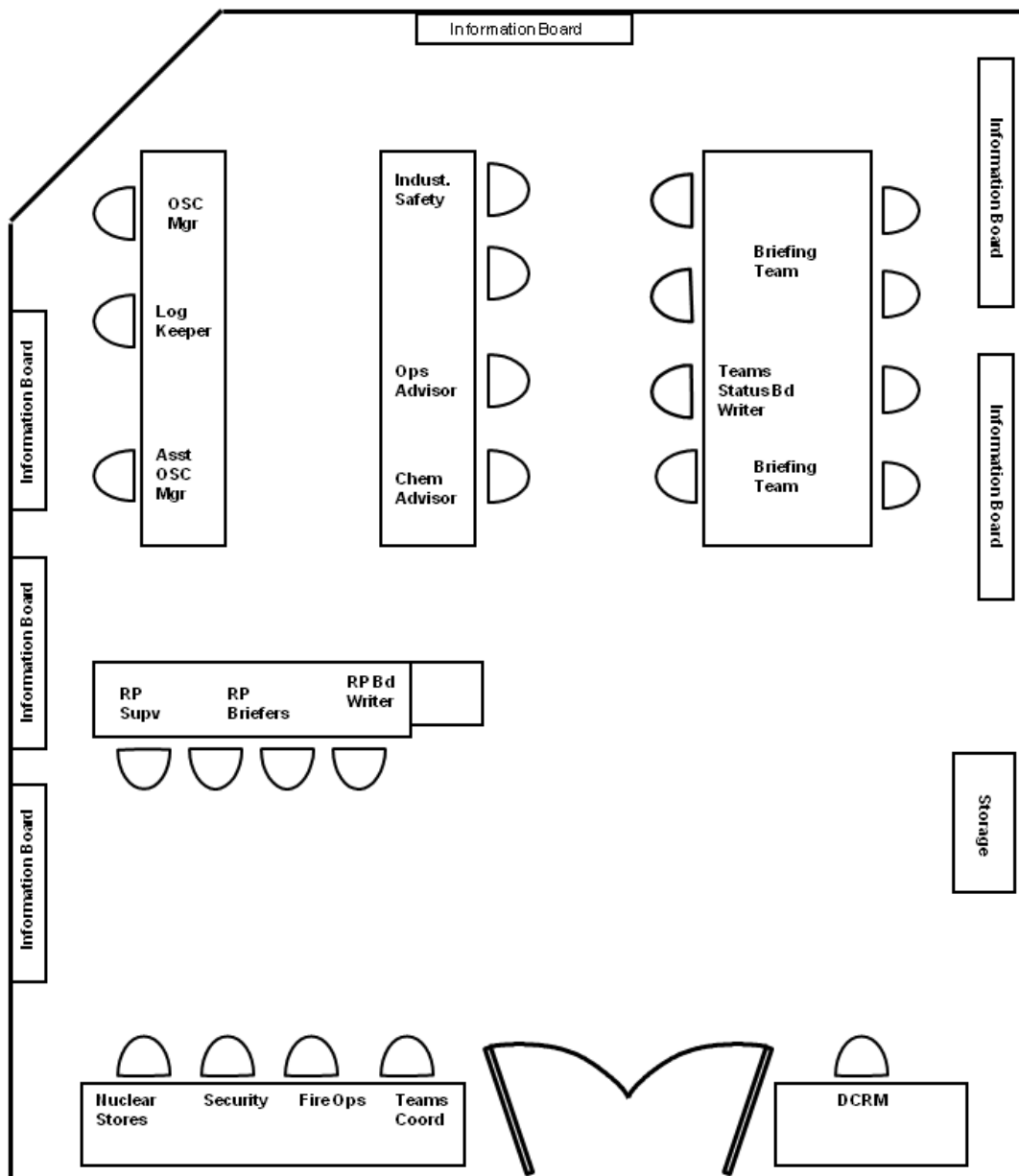
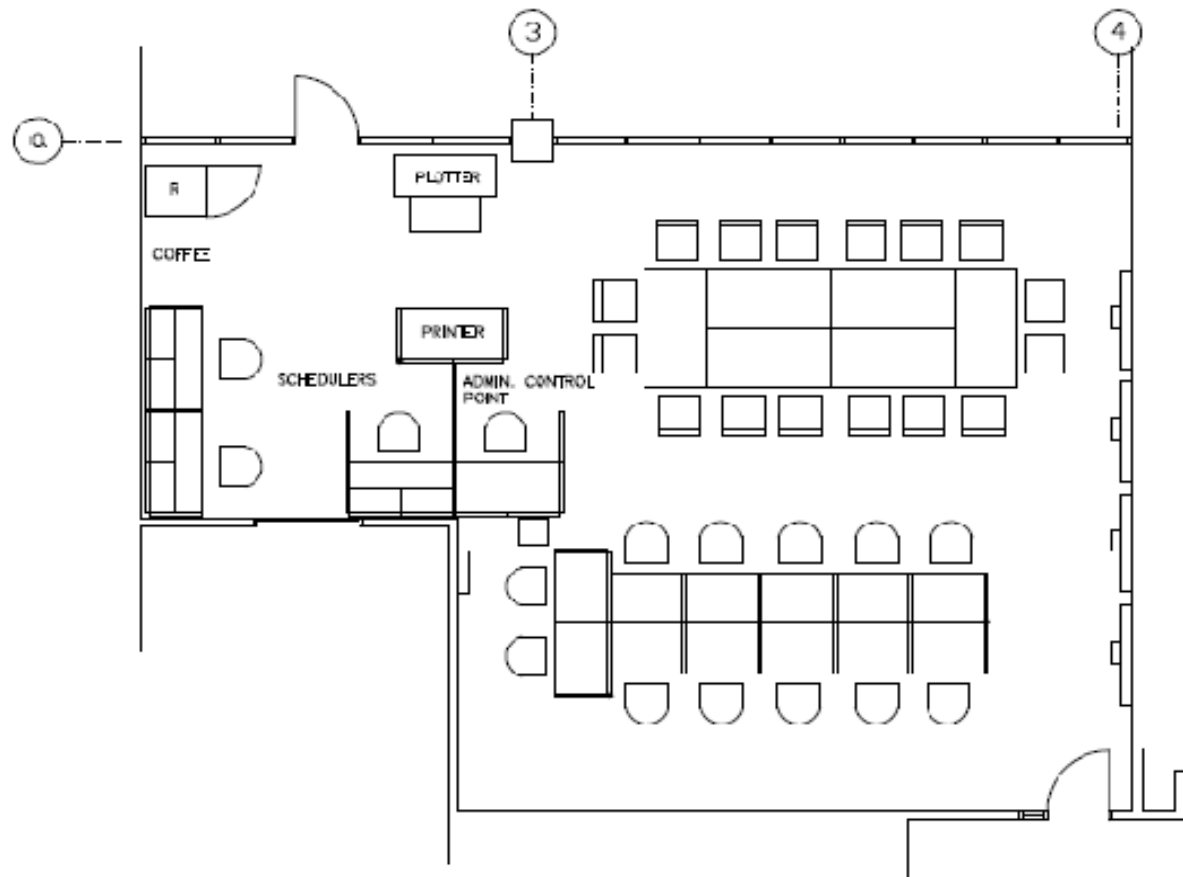


FIGURE 6-C



OUTAGE CONTROL CENTER
PLANT OFFICE BLDG.
WATTS BAR N.P.

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C.6.3 RP Laboratory and Equipment

The RP laboratory is located in the Service Building, Elevation 713. The portable radiation monitoring and counting equipment normally used by the plant RP section is kept in this space and is available for use during an emergency. Sufficient reserves of instruments/equipment are available to replace those removed from service for calibration or repair. Calibration of equipment is carried out at intervals as specified in the Radiation Protection Plan (RPP).

C.6.4 Alternative Facilities

Following the declaration of Alert or higher, for some conditions, primarily in response to hostile action events when ERO members may not have access to the site, an alternative facility has been identified. This facility serves as a staging area for augmentation staff, which will minimize delays in overall site response and allow for a swift, coordinated augmented response when the site is deemed accessible.

The alternative facility is accessible even if the site is under threat of or experiencing hostile action, to function as a staging area for augmentation of emergency response staff and have the capability for communication with the emergency operations facility, control room, and plant security, for use when onsite emergency facilities cannot be safely accessed during hostile action.

C.6.5 Onsite Monitoring Systems and Equipment

C.6.5.1 Natural Phenomena

In the event an emergency is the result of a natural phenomena, there is instrumentation to monitor its severity. The Environmental Data Station is located onsite and contains instruments capable of measuring wind direction, wind speed, and temperatures. Seismic instrumentation is available in the plant to monitor acceleration levels of ground movement. Hydrological monitoring systems are installed to supply flow and level information for each site. Meteorological and seismic instrumentation have readily accessible readout in the main control room. More specific information on these systems can be found in the Watts Bar FSAR.

C.6.5.2 Radiological Monitors

The installed Radiation Monitoring System consists of process monitors and area monitors which read out on local panels and in the control room.

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C.6.5.2.1 Process Monitors (Radiological)

The process system continuously monitors selected lines containing or possibly containing radioactive effluents. The system's function is to warn personnel of increasing radiation levels, to give early warning of a system malfunction, and to record and control discharges of radioactive liquids and gases to the environment. The system consists of active and redundant channels.

Examples of process monitors are:

1. Ventilation Gas and Particulate
2. Process Gas and Particulate
3. Containment Gas and Particulate
4. Condenser Vacuum Exhaust
5. Steam Generator Blowdown
6. Liquid Waste
7. Service Water
8. Component Cooling Water
9. Component Cooling Water Heat Exchangers
10. Reactor Coolant System

C.6.5.2.2 Area Radiation Monitors

Area monitors are placed at specific locations in the plant. Examples of area monitor locations are:

1. Containment
2. New and Spent Fuel Storage Area
3. Main Control Room
4. Incore Instrument Area

C.6.5.2.3 Portable Monitors

Portable radiation detection equipment consists of low-range and high-range instruments to measure gamma dose rates. Instruments for alpha, beta-gamma, and neutron radiation measurements are available. Sampling equipment is available to take low- or high-volume air samples. Air samplers can be used to collect low-volume samples either onsite or offsite. The counting room has appropriate equipment for isotopic analysis.

C.6.5.2.4 Process Monitors (Nonradiological)

Installed in the main control room are the necessary instrumentation readouts to assess plant systems status, including reactor coolant system pressure and temperature, containment pressure and temperature, liquid levels, flow rates, fire detection equipment, and meteorological instrumentation. More specific information on control room instrumentation can be found in the Watts Bar FSAR.

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C.6.5.2.5 Fire Protection

The plant's fire protection system is designed to furnish water and other extinguishing agents with the capability of extinguishing any single or probable combination of simultaneous fires that might occur. The use of combustible materials is minimized, and the greatest possible use of fire-retardant materials has been incorporated in plant design.

The standards of the National Fire Protection Association and the recommendations of the nuclear insurers are considered in the system design to provide the following:

1. Supply of water for the fire protection system.
2. Automatic fire or smoke detection in the more critical areas.
3. Fire suppression by fixed equipment actuated automatically or manually.
4. Manually-operated portable fire extinguishing equipment at strategic locations.
5. Compartmentalization to limit the spread of fire.

C.6.5.2.6 Environment

Facilities available for assessing the impact of plant operations on the environment include atmospheric monitoring stations, direct gamma radiation detectors, and automatic water samplers. This equipment is used in the routine environmental radiological monitoring program and is available in the event of a radiological emergency condition.

The atmospheric monitoring network is divided into three subgroups. Local air monitors are located at or adjacent to the site boundary in the directions of predominant wind flow. Perimeter monitors are located three to ten miles from the plant in areas of relatively high population densities and/or in the direction of predominant air flow. Remote monitors (controls) are located at sites greater than 10 miles from the plant.

At each monitor, air is continuously passed through a particulate filter at a regulated flow. In series with, but downstream of, the particulate filter is a charcoal filter used to collect iodine.

Each monitor has a collection tray and storage container to collect rainwater on a continuous basis.

Primary dosimeters are placed at approximately 40 sites around the plant. These Primary dosimeters are located typically in each of the 16 meteorological sectors at or near the Site Boundary and at a distance of approximately four to five miles. Three dosimeters are usually placed at each site.

Automatic water samplers are located above and below the plant discharge and at the first potable water supply downstream from the plant.

In addition to these facilities, established sampling points for milk, vegetation, soil, fish, and sediment are located in the vicinity of the plant. Samples may be collected from these stations on a nonroutine basis as needed.

All samples are returned to TVA's radiological laboratory for processing.

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C.6.6 Emergency Equipment

Figure 7-C contains listings of emergency equipment and storage locations throughout the plant.

Required calibration of equipment is carried out at intervals recommended by the supplier of the equipment or as specified in the Watts Bar FSAR.

C.6.7 First Aid and Medical Facilities

C.6.7.1 Decontamination Facilities

The site is responsible for maintaining supplies and equipment to establish a temporary decontamination area for the purpose of gross radiological decontamination and injured person evaluation and stabilization. This area, complete with shower and sink, is located in the Service Building, Elevation 713'. Equipment and materials for decontamination and first aid, including a stretcher, are available.

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FIGURE 7-C

EMERGENCY EQUIPMENT

	<u>Location</u>	<u>Description</u>
1.	RP Laboratory (Service Bldg., El. 713')	Radiological survey meters and SCBAs
2.	Site medical station and ambulance	General use emergency medical supplies
3.	Service Bldg. El. 713' (near breathing air compressor)	Emergency SCBA's with additional cylinders
4.	Decon Facility (Service Bldg. El. 713')	Decon supplies
5.	Emergency Van (RP, environmental monitoring) supplies related to	General emergency
6.	Rhea County Medical Center	Supplies specific to Emergency Room radiological injuries
7.	Athens Regional Medical Center	Supplies specific to Emergency Room radiological injuries

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C.6.7.2 First Aid Stations and Supplies

Emergency medical equipment is strategically located throughout the plant, with trauma kits and other specified equipment available for use by the Medical Emergency Response Team (MERT).

First aid is provided by EMTs. Medical supplies and treatment for minor injuries are available. A minimum of one ambulance is also available. First aid treatment is available 24 hours a day.

A medical office, staffed by registered nurses, is located at the west end of the Watts Bar Training Center. Medical treatment is available during the day and evening shifts. Examinations (employment, routine, occupational) are available during the day shift, Monday-Friday.

Potassium Iodide tablets for onsite personnel are controlled and stored by site RP. Specific information including authorization and dispersal of tablets is contained in the site EIPs.

C.6.7.3 Receiving Hospitals and Supplies

Arrangements have been made with the Rhea County Medical Center and Athens Regional Medical Center to receive patients from WBN.

C.6.7.4 Ambulance Service

A TVA ambulance is available at the site and is maintained and staffed in conjunction with the MERT. Arrangements have been made for offsite ambulance assistance to WBN.

C.6.8 Additional Local Support

C.6.8.1 Fire

Arrangements have been made for local fire support upon request. The senior fireman responding will work with and for the TVA Incident Commander in directing the activities of the firemen. Watts Bar will be responsible for providing radiological protection and proper safety clearance in all fire areas.

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C.6.8 Additional Local Support (continued)

C.6.8.2 Law Enforcement

Agreements are maintained with local law enforcement agencies to support TVA when necessary.

C.6.9 Vendor Support

If necessary, the NSSS vendor, Westinghouse, will be contacted by the TSC to provide assistance in the form of manpower, equipment, and technical backup. Other vendors will also be contacted if their assistance is needed.

C.6.10 Assembly/Accountability Alarm

Undulating sirens are provided in strategic areas for indicating the assembly of plant personnel. A three-minute undulating tone of the alarm is the signal for assembly.

The sirens are powered by redundant 120V ac supplies. The sirens are activated in the main control room or the auxiliary control room diesel panel.

C.6.11 Local Recovery Center (LRC)

The LRC is a designated, non-dedicated space located in Classroom 6 of the Watts Bar Training Center (WBN) outside the protected area of the site. Figure 8-C shows the location of the WBN LRC in the WBN Training Center.

The LRC has telephone communications capabilities to enable personnel to communicate with the CECC and the Watts Bar TSC.

Meteorological information and dose rate calculations are also available to LRC personnel.

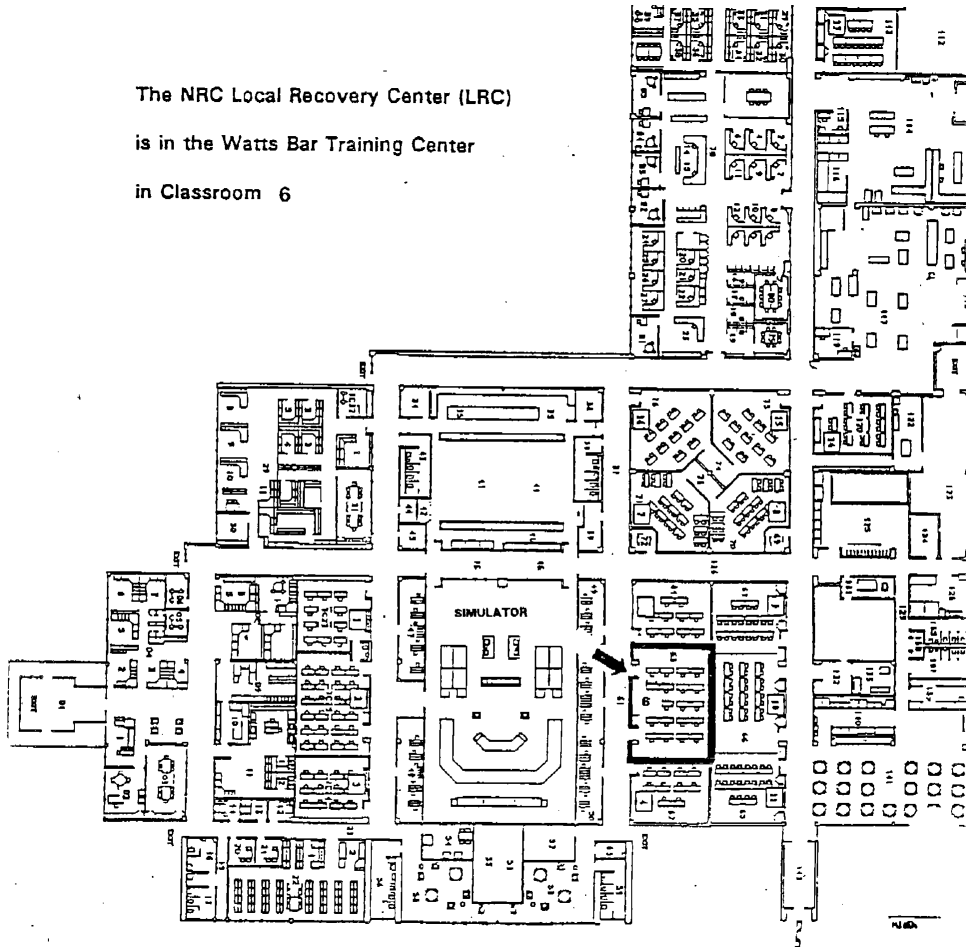
Other equipment in the WTC available for use by LRC personnel include:

1. Facsimile machine
2. Copy machine
3. Hand-held calculators
4. Plant-specific drawings, manuals, procedures, etc. (drawings located in nearby WBN Operations Training Area)

FIGURE 8-C

LOCAL RECOVERY CENTER

The NRC Local Recovery Center (LRC)
is in the Watts Bar Training Center
in Classroom 6



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C.7 WBN Emergency Plan Implementing Procedures

The following is a listing of the WBN-EIPs:

C.7.1 WBN EIP-1 Emergency Plan Classification Logic

This procedure provides guidance to the Shift Manager Site Emergency Director or TSC Site Emergency Director in determining the classification of an accident to ensure that appropriate predetermined actions are implemented. It details initiating conditions and directs shift personnel to appropriate notification and assessment procedures.

C.7.2 WBN EIP-2 Notification of Unusual Event

This procedure provides for the timely notification of appropriate individuals when the Shift Manager SED or TSC SED has determined by WBN EIP-1 that an incident has occurred which is classified as a Notification of Unusual Event. It details requirements for periodic reassessment and the implementation of appropriate actions.

C.7.3 WBN EIP-3 Alert

This procedure provides for the timely notification of appropriate individuals when the Shift Manager SED or TSC SED has determined by WBN EIP-1 that an incident has occurred which is classified as an Alert. It details requirements for periodic reassessment and the implementation of appropriate actions.

C.7.4 WBN EIP-4 Site Area Emergency

This procedure provides for the timely notification of appropriate individuals when the Shift Manager SED or TSC SED has determined by WBN EIP-1 that an incident has occurred which is classified as a Site Area Emergency. It details requirements for periodic reassessment and the implementation of appropriate actions.

C.7.5 WBN EIP-5 General Emergency

This procedure provides for the timely notification of appropriate individuals when the Shift Manager SED or TSC SED has determined by WBN EIP-1 that an incident has occurred which is classified as a General Emergency. It details requirements for periodic reassessment and the implementation of appropriate actions. It also provides for determination of an initial protective action recommendation to State and local agencies.

C.7.6 WBN EIP-6 Activation and Operation of the TSC

This procedure directs the activation and operation of the TSC during an Alert, Site Area Emergency, or General Emergency or at the discretion of the SED. It details notification requirements and responsibility for supervision of the TSC.

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C.7.7 WBN EPIP-7 Activation and Operation of the OSC

This procedure directs the activation and operation of the OSC during an Alert, Site Area Emergency, or General Emergency or at the discretion of the SED.

C.7.8 WBN EPIP-8 Personnel Accountability and Evacuation

This procedure details the requirements for accountability of all personnel and visitors and the orderly evacuation of areas of the plant during a radiological emergency. This procedure also details the requirements for accountability of personnel and visitors at auxiliary facilities and their orderly evacuation during a radiological emergency.

C.7.9 WBN EPIP-9 Loss of the Meteorological Data

Terminated see EPIP-13

C.7.10 WBN EPIP-10 Medical Emergency Response

This procedure details actions to be followed during medical emergencies. It provides for the organization and activation of the onsite Medical Emergency Response Team. It contains the duties and responsibilities of the onsite Medical Emergency Response Team. The procedure provides guidance on the care and handling of patients who may have been exposed to or contaminated with radioactive material, including provision for the transport of these individuals to offsite medical support facilities. Maps and appropriate instructions are included.

C.7.11 WBN EPIP-11 Security and Access Control

This procedure details responsibilities and requirements for access control and accountability during a radiological emergency.

C.7.12 WBN EPIP-12 Emergency Equipment and Supplies

This procedure details requirements for periodic inspection and maintenance of emergency equipment and supplies. It assigns responsibility and specifies the inspection frequency and documentation requirements.

C.7.13 WBN EPIP-13 Initial Dose Assessment for Radiological Emergencies

This procedure provides initial guidance to support site activities concerning dose assessment for an actual or exercise airborne release situation.

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C.7.14 WBN EPIP-14 Radiological Control Response

This procedure outlines the actions to be followed by health physics personnel during a plant emergency. It details responsibilities, RP assessment actions and recordkeeping requirements. The procedure provides guidance regarding the administration of potassium iodide (KI).

C.7.15 WBN EPIP-15 Emergency Exposure Guidelines

This procedure provides guidance on acceptable personnel exposures for various conditions. It specifies absolute dose rates and authorizes the Site Emergency Director to permit dose rates in excess of 10 CFR 20 limits in order to perform an emergency mission.

C.7.16 WBN EPIP-16 Termination of the Emergency and Recovery

This procedure outlines responsibilities and provides guidance to terminate the emergency condition and recovery after, an emergency to assure adequate planning for efficient utilization of resources and radiation exposure.

C.7.17 WBN EPIP-17 Fire Emergency Procedure

This procedure provides guidance on timely responses to fire emergencies at WBN.

C.8 Alert Notification System

See Generic REP Section.

C.9 Training and Drills

C.9.1 Training Personnel

Personnel with specific duties and responsibilities in the WBN REP program receive instruction in the performance of their duties and responsibilities per the Nuclear Power Training Manual, Section TRN-30 (Radiological Emergency Preparedness Training), and as required in REP Section 15.0, (Training).

C.9.2 Drills and Exercises

Drills and exercises are conducted regularly to develop and maintain the key skills that are required for emergency response. The drills identified in Section 14.0 (Drills and Exercises) may be conducted individually or as part of a REP exercise.

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