Browns Ferry Nuclear Plant

Individual Plant Examination For External Events (IPEEE)

Internal Fires
High Winds
Floods
Transportation and Nearby Facility Accidents

JULY 1995

TENNESSEE VALLEY AUTHORITY

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ENCLOSURE

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3

GL 88-20, SUPPLEMENT 4
INDIVIDUAL PLANT EXAMINATION FOR EXTERNAL EVENTS (IPEEE)
FOR SEVERE ACCIDENT VULNERABILITIES

(SEE ATTACHED REPORT)

1. EXECUTIVE SUMMARY

1.1 BACKGROUND AND OBJECTIVES

This report documents the work performed by the Tennessee Valley Authority (TVA) in accordance with the U.S. Nuclear Regulatory Commission (NRC) Generic Letter No. 88-20, Supplement 4 (Reference 1.1). The letter requested each utility to perform an individual plant examination of external events (IPEEE) "(1) to develop an appreciation of severe accident behavior, (2) to understand the most likely severe accident sequences that could occur at its plant under full-power operating conditions, (3) to gain a qualitative understanding of the overall likelihood of core damage and fission product releases, and (4) if necessary, to reduce the overall likelihood of core damage and fission product releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents."

Based on NRC's study, they concluded that five external events need to be included specifically in the IPEEE. They are the following: seismic events, internal fires, high winds, floods, and transportation and nearby facility accidents. Additionally, licensees should confirm that no other plant-unique external events with potential severe accident vulnerability are being excluded from the IPEEE.

TVA's overall objectives of the IPEEE program were to

- Meet the NRC requirements for IPEEE as set forth in Generic Letter No. 88-20, Supplement 4 and in NUREG-1407 (Reference 1.2).
- Identify and implement opportunities for safety enhancement.

1.2 PLANT FAMILIARIZATION

The Browns Ferry Nuclear Plant (BFNP) is located on the north shore of Wheeler Lake at Tennessee River mile 294 in Limestone County, Alabama. The site is approximately 10 miles southwest of Athens, Alabama, and 10 miles northwest of Decatur, Alabama. The plant consists of three units, each with a rated power level of 3,293 MWt. Unit 2 is the only unit currently operating and Unit 3 is currently in a recovery mode and expected to return to commercial operation in December, 1995.

All three units are single-cycle forced-recirculation boiling water reactor (BWR) with nuclear steam supply system supplied by General Electric Corporation. Major structures at Browns Ferry include a reactor building with Mark I drywell containment, a turbine building, a control bay, two diesel generator buildings, a standby gas treatment building, and an intake pumping station.

A detailed description of the plant site, facilities, and safety criteria is documented in the

Browns Ferry Final Safety Analysis Report (Reference 1.3).

1.3 OVERALL METHODOLOGY

The methodologies used in performance of the IPEEE are those as presented in GL 88-20, Supplement 4. The methodology used to perform fire hazards evaluation at Browns Ferry Nuclear Plant Unit 2 is based on the Fire Induced Vulnerability Evaluation (FIVE) methodology that was developed by the Electric Power Research Institute (EPRI), (Reference 1.4). The fire analysis also meets the informational requirements of NUREG-1407 (Appendix C, Section C.3). Section 8.3 of the fire analysis report provides cross references to aid in locating the informational requirements of NUREG-1407. A screening approach as described in the GL 88-20 supplement is used for evaluations of risk from high winds, external floods, and nearby facility/transportation events. No other external events (e.g. volcanic activity) are applicable to the Browns Ferry site.

The results of this IPEEE are based on a significant amount of analysis and plant walkdowns. However, the results are also based on the judgement and experience of both the analyst performing the IPEEE and the technical reviewers of the work. The use of judgement is in keeping with the spirit of the IPEEE generic letter supplement which recognizes the large uncertainties associated with severe accident behavior, especially when the initiating events involved are extremely infrequent and spatially distributed. The explicit recognition of this uncertainty, and the expectation that most risk-significant vulnerabilities would be identified via experience based judgement coupled with plant walkdowns, is a great strength of the GL supplement.

The IPEEE is performed as a "best estimate". There is no attempt to combine uncertainties to create bounding, worst case but highly improbable scenarios, or to artificially assume that the physical condition of BFNP is degraded at the time of the initiating event. This IPEEE makes every attempt to reflect the true condition of BFNP so that the results are realistic. The analysis methods and results were subjected to extensive reviews before acceptance.

1.4 SUMMARY OF MAJOR FINDINGS

The major findings of the IPEEE are presented here for each of the two analyses:

1. Internal Fires
2. High winds, external floods, nearby facilities/transportation

While no vulnerabilities were identified in the course of this evaluation for Internal Fires, several items of interest were noted:

- In general, essential switchgear rooms were noted to have low conditional

core damage frequencies. This is due to the large amount of partitioning between divisions and trains at the Browns Ferry plant. For example, RHR pumps A, B, C and D are each supplied from a different 4kV shutdown board, each of which is located in a different fire area. The four core spray pumps are supplied in a similar fashion. This design prevents the failure of a single shutdown board, whether due to fire or due to independent hardware failure, from failing an entire division of a given ECCS system. Also, the unit battery boards are set up to allow the maximum level of flexibility and redundancy between the three units.

- The walkdowns performed in conjunction with the fire hazard evaluation identified several plant locations where a potential fire could damage risk significant components and cables. However, due to the redundant nature of the plant design (i.e. numerous and diverse methods of maintaining reactor makeup, etc.), these situations do not affect overall plant risk significantly.

- Several potential plant fire hazards, such as oil filled transformers and reactor recirculation MG sets, have the potential to generate significant amounts of heat. These components have been previously identified and evaluated by the fire protection program. In part due to this previous evaluation effort and subsequent detail evaluation in conjunction with this fire evaluation, these hazards were not found to impact plant risk significantly.

The screening approach used in analysis of external floods, and nearby facilities/transportation accidents demonstrates that they meet U.S. NRC Standard Review Plan (SRP) 1975 criteria and has adequate defense against these threats. Since Browns Ferry does not meet the SRP 1975 criteria for high winds, a bounding analysis was performed. This analysis showed the contribution to core damage frequency due to high winds to be less than the IPEEE screening criteria of 10^{-6}. No plant modifications were identified from any of the analysis.

1.5 REFERENCES FOR EXECUTIVE SUMMARY

1.1 "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities", Generic Letter 88-20, Supplement 4, June 28, 1991.

1.2 "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities", U.S. NRC NUREG-1407.

1.3 Browns Ferry Nuclear Plant Final Safety Analysis Report Amendment 11.

1.4 Electrical Power Research Institute, "Fire Induced Vulnerability Evaluation (FIVE)", EPRI TR-100370, April 1992.

2. EXAMINATION DESCRIPTION

2.1 INTRODUCTION

BFNP began work on the IPEEE subsequent to the conclusion of the IPE effort. The IPEEE is a request for information contained in Supplement 4 to Generic Letter 88-20. As stated in the generic letter supplement, the purpose of the IPEEE is similar to that of the IPE in that each licensee is expected to:

1. Develop an appreciation of severe accident behavior.

2. Understand the most likely severe accident sequences that could occur at its plant under full power operating conditions.

3. To gain a qualitative understanding of the overall likelihood of core damage and radioactive material release.

4. If necessary, to reduce the overall likelihood of core damage and radioactive material release by modifying hardware and procedures that help prevent or mitigate severe accidents.

Thus, the IPEEE is a methodical search for vulnerabilities, that is weakness in core, vessel and containment defense, given an external event/threat to BFNP. This search is part of an overall process of risk characterization and reduction begun with the issuance of the Severe Accident Policy Statement by the NRC in 1985. Because of the large uncertainties associated with both the frequency of initiating event occurrence, and the impact of these external events on the plant, the IPEEE emphasizes qualitative rather than quantitative estimates of core damage and radioactivity release. In keeping with the state of knowledge about external event frequency and impact, significant judgements are expected to be utilized in both the IPEEE scope and analysis.

2.2 CONFORMANCE WITH GENERIC LETTER AND SUPPORTING MATERIAL

Generic Letter 88-20, Supplement 4 requests a response from each licensee outlining the IPEEE approach and schedule. The supplement also includes guidance regarding types of external threats to be evaluated, the methods of examination, determination of vulnerabilities and documentation. NUREG-1407 provides further details of acceptable approaches to the IPEEE response.

TVA Browns Ferry transmitted the proposed IPEEE approach and schedule via letter dated December 20, 1991(Reference 2.1). BFNP identified the five(5) suggested external events to be utilized for BFN's evaluation and will be included with BFNP's IPEEE report.

The IPEEE is performed using the methodologies suggested in the Supplement and NUREG 1407. Fire risk is determined using the FIVE methodology and current BFNP Appendix R Safe Shutdown Analysis program. Screening against the 1975 SRP criteria is performed to study risks due to high winds, external flooding and nearby facility/transportation accidents. The report and supporting calculations had an extensive review by the IPEEE participants.

BFNP's documentation of the IPEEE conforms to the "two tier" approach recommended in the Supplement. The IPEEE report represents the first tier and the numerous calculation packages retained by BFNP make up the second tier of documentation.

2.3 GENERAL METHODOLOGY

The IPEEE is performed using the methodologies suggested in the GL Supplement and NUREG 1407. This section provides a general discussion of these methodologies. GL Supplement recommends different methodologies be used to analyze the different external events. Fire risk has been determined using FIVE methodology and screening against the 1975 SRP criteria has been performed to study risks due to high winds, external flooding and nearby facility/transportation accidents.

2.3.1 INTERNAL FIRES

The EPRI FIVE documentation (Reference 2.2) is used as a basis for evaluation of fire hazards and for screening fires from further consideration, based on a screening criteria of less than 10^{-6} core damage frequency due to fire related initiating events.

The FIVE documentation describes the fire evaluation process in three phases. The steps involved in each of these phases are shown in Figure 1-1 in Section 4 of this submittal and are briefly described as follows:

Phase I - Qualitative screening and fire compartment interaction analysis. During this phase, plant areas can be removed from further consideration based on the absence of safe shutdown equipment and no identified need for plant trip. Also, fire boundaries are reviewed to ensure that a fire could not develop and then spread to other areas that may contain safe shutdown equipment.

Phase II - Quantitative evaluation of plant areas. This phase accounts for the largest portion of effort for the fire hazard evaluation process. This portion of the fire hazard evaluation consisted of the following three steps:

Step1 - identified individual and generic plant fire hazards and their associated fire ignition frequencies for the unscreened plant fire areas and zones. These values were used to generate initiating event frequencies for the various plant areas. Within the EPRI FIVE documentation, this

value is identified as "F1." If this value is less than 10^{-6}, the area can be screened from further consideration.

Step 2 - evaluated the plant model impacts caused by the fires of concern. When taken with the fire damage analysis from Step 3 (described below), this generates a conditional core damage frequency," or "P2" value, as it is identified in the EPRI FIVE documentation.

Step 3 - supported the development of a PRA model to refine the results of Step 2 by identifying the potential plant impacts due to fires in the various areas through identification of plant component locations and cable routing information.

From a quantitative standpoint, if the fire related core damage frequency, or F2 value (= F1 x P2), is less than 10^{-6}, the area can be screened from further consideration. If the area can not be screened from further consideration, the assumptions used during the screening evaluation are reviewed to evaluate the area for relaxation of overly conservative assumptions. The various parts of Phase II are then repeated as necessary to complete the quantitative screening process.

For some plant areas, such as the Control Building and Turbine Building, probabilistic models of fire behavior were used in lieu of deterministic fire hazards techniques due to the difficulty in establishing specific fire source/target scenarios.

Phase III - Results and issues. The final phase of the fire evaluation process consists of documentation of results and identification of any new or remaining issues, including those addressed by the Sandia Fire Risk Scoping Study (NUREG/CR 5088) and the evaluation of containment performance.

2.3.2 HIGH WINDS, EXTERNAL FLOODING, NEARBY FACILITY AND TRANSPORTATION ACCIDENTS

A screening approach is used for evaluation of risk from high winds, external floods and nearby facility/transportation events. The flowchart shown in the GL Supplement (Figure 1 of Reference 2.3) is used as basic foundation of the screening approach. Basically, the method consists of reviewing the analyses previously completed in supporting of licensing, reviewing changes to plant environs since Operating License (OL) issuance and verifying that the plant design conforms with the 1975 SRP criteria.

2.4 INFORMATION ASSEMBLY

The IPEEE information consists of this report, the supporting calculations, studies and the references. The IPEEE report organization follows the standard table of contents provided in Reference 2.4. Supporting calculations are retained by BFNP. To eliminate duplication of paper, much information is incorporated by use of the references. These references are either publicly available or retained by BFNP.

2.5 REFERENCES FOR EXAMINATION DESCRIPTION

2.1 TVA letter dated December 20, 1991, from O.J. Zeringue to U.S. NRC (RIMS R08911220969).

2.2 Electrical Power Research Institute, "Fire Induced Vulnerability Evaluation (FIVE)", EPRI TR-100370, April 1992.

2.3 "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities", Generic Letter 88-20, Supplement 4, June 28, 1991.

2.4 "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities", U.S NRC NUREG-1407.

3. SEISMIC ANALYSIS

The seismic portion of the IPEEE for BFN Units 1, 2, and 3 will be completed in conjunction with the Generic Letter 87-02, Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors and the Unresolved Safety Issue A-46 program.

4. INTERNAL FIRE ANALYSIS

The Internal Fire analysis (attached) was prepared as a "stand alone" report based on the Fire Induced Vulnerability Evaluation (FIVE) methodology developed by the Electrical Power Research Institute. The paragraph numbering convention for the Internal Fires Analysis differs from the guidance provided in NUREG 1407, Appendix C.

RESPONSE TO GENERIC LETTER 88-20, SUPPLEMENT 4

INDIVIDUAL PLANT EXAMINATION FOR EXTERNAL EVENTS (IPEEE - FIRE)

BROWNS FERRY NUCLEAR PLANT UNIT 2

FIRE INDUCED VULNERABILITY EVALUATION (FIVE) METHODOLOGY

TENNESSEE VALLEY AUTHORITY

APRIL, 1995

Browns Ferry Nuclear Plant Unit 2
FIVE

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Bill Parkinson (SAIC)	Peer Review

Also, portions of the information presented in this report were developed as a result of Browns Ferry participation in the EPRI Tailored Collaboration Project for the development of the FIVE software.

Tennessee Valley Authority

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EXECUTIVE SUMMARY

This report documents the evaluation performed for Browns Ferry Unit 2 in response to Supplement 4 to Generic Letter 88-20 to determine the plant vulnerability to internal fire events. This evaluation is based on the Fire Induced Vulnerability Evaluation (FIVE) methodology developed by the Electric Power Research Institute (EPRI), as described in Reference 1.

The FIVE methodology consists of a progressive screening evaluation, in which plant fire areas are screened from consideration based on *qualitative information* (Phase I) or by *quantitative analysis* (Phase II). Phase I consists of screening fire areas based on absence of safe shutdown components and lack of plant trip initiators. None of the BFN areas were screened from consideration during this phase. The quantitative analysis (Phase II) then consists of an *initial quantitative evaluation* (see Section 5), followed by a more *detailed quantitative evaluation* (see Section 6) for areas that were not screened, based on a fire-induced core damage frequency of less than 1E-06.

The *initial quantitative evaluation* consists of generating an area-specific fire ignition frequency, then assuming that all fires totally engulf the affected area. Plant components that could be affected by these fires were identified by plant walkdowns. A "conditional core damage frequency" was generated for each area by incorporating the failed components into revision 1 of the PRA plant model. 15 of 34 total areas were screened from further consideration, based on fire ignition frequency, multiplied by the corresponding conditional core damage frequency, being less than 1E-06.

The *detailed quantitative evaluation* was then performed for the remaining areas (i.e. those that were not screened from consideration in Section 5). These areas included the Unit 1, 2 and 3 Reactor Buildings, the Control Building and Turbine Building areas, in addition to 4kV Shutdown Board Rooms A, B, C and D and the Unit 2 Battery and Battery Board Rooms. The Unit 2 Reactor Building was evaluated for fire vulnerabilities by using the conventional fire hazard evaluation techniques identified in Reference 1. Other areas, including the Control Building, were evaluated by using a probabilistic approach to segment the evaluation into individual cases for analysis.

Following the detailed evaluation process, all remaining plant areas were screened from further consideration, confirming that there are no fire-induced vulnerabilities associated with the continued operation of Browns Ferry Unit 2.

1. INTRODUCTION

This report describes the process used to evaluate fire hazards at Browns Ferry Nuclear Plant Unit 2. This evaluation was performed in response to the Individual Plant Examination for External Events (IPEEE) requested by Supplement 4 of Generic Letter 88-20. The methodology used to perform this examination is based on the Fire Induced Vulnerability Evaluation (FIVE) methodology that was developed by the Electric Power Research Institute (EPRI), as described in Reference 1. Section 1.1, below, provides an overview description of the FIVE methodology. Section 1.2 describes the implementation of this guidance for Browns Ferry.

1.1 Overview of the Fire Induced Vulnerability Evaluation (FIVE) Methodology

The EPRI FIVE documentation (Reference 1) was used as a basis for evaluation of fire hazards and for screening fires from further consideration, based on a screening criteria of less than 1E-06 core damage frequency due to fire related initiating events.

The FIVE documentation describes the fire evaluation process in three phases. The steps involved in each of these phases are shown in Figure 1-1 and are described below.

Phase I **Qualitative screening and fire compartment interaction analysis.** During this phase, plant areas can be removed from further consideration based on the absence of safe shutdown equipment and no identified need for plant trip. Also, fire boundaries are reviewed to ensure that a fire could not develop and then spread to other areas that may contain safe shutdown equipment.

Phase II **Quantitative evaluation of plant areas.** This phase accounts for the largest portion of effort for the fire hazard evaluation process. This portion of the fire hazard evaluation consisted of the following three steps:

Phase II (Step1) identified individual and generic plant fire hazards and their associated fire ignition frequencies for the unscreened plant fire areas and zones. These values were used to generate initiating event frequencies for the various plant areas. Within the EPRI FIVE documentation, this value is identified as "F1." If this value is less than 1E-06, the area can be screened from further consideration.

Phase II (Step 2) evaluated the plant model impacts caused by the fires of concern. When taken with the fire damage analysis from Step 3 of Phase II (described below), this generates a "conditional core damage frequency," or "P2" value, as it is identified in the EPRI FIVE documentation.

Phase II (Step 3) supported the development of a PRA model to refine the results of Step 2 of Phase II by identifying the potential plant impacts due to fires in the various areas through identification of plant component locations and cable routing information.

From a quantitative standpoint, if the fire related core damage frequency, or F2 value (= F1 x P2), is less than 1E-06, the area can be screened from further consideration. If the area can NOT be screened from further consideration, the assumptions used during the screening evaluation are reviewed to evaluate the area for relaxation of overly conservative assumptions. The various parts of Phase II are then repeated as necessary to complete the quantitative screening process.

Phase III **Results and Issues.** The final phase of the fire evaluation process consists of documentation of results and identification of any new or remaining issues, including those addressed by the Sandia Fire Risk Scoping Study (NUREG/CR 5088) and the evaluation of containment performance.

1.2 Implementation of the EPRI FIVE Methodology

The implementation of the EPRI FIVE methodology is shown in Figures 1-2 and 1-3. This implementation is also described below.

Phase I The qualitative screening process is described in Section 3 of this report. During this review, all plant fire areas were conservatively assumed to contain safe shutdown components (SSC). Also, a Fire Compartment Interaction Analysis (FCIA) was performed to determine the potential for fire spread from an exposed compartment to an adjacent unexposed compartment. No insignificant compartments were identified through this process. Therefore, no areas were screened from consideration at this point.

Phase II The quantitative evaluation of fire hazard frequency was performed in three steps, as described below. This process is based on the guidance given in the EPRI FIVE documentation (Reference 1).

Phase II (Step 1) used the guidance in the EPRI FIVE documentation directly to generate fire ignition frequencies (i.e. "F1" values). These calculations are based on the plant-specific data listed in Section 2. This process consisted of two sub-steps. The first sub-step allocated a plant area fire ignition frequency, based on the assignment of each plant location to a generic type of area, such as switchgear rooms or cable spreading rooms. The second sub-step then assigned fire ignition frequencies for identified plant-wide components, such as hydrogen recombiners, to each location. The calculation of fire ignition frequency for each plant fire area, fire zone and compartment is shown in Attachment B. This process is described in Section 4.

Phase II (Step 2) performed a screening evaluation for each fire area, zone and compartment. During this step, all fires were assumed to engulf the affected area and result in a plant trip for Unit 2. The probability for redundant/alternate system unavailability, or "conditional core damage frequency" (i.e. "P2" value) was calculated using the PRA plant model by incorporating the potential fire impacts. Areas that had an overall frequency of fire occurring and damaging safe shutdown components (F1 x P2 = F2, as described in the EPRI FIVE documentation) below the screening criteria of 1E-06 were then screened from further consideration, based on the EPRI FIVE guidance. This step is described in Section 5.

Phase II (Step 3) then performed a more detailed evaluation of those areas that could not be screened from further consideration in Step 2 of Phase II. Due to the differences in area geometry, fire sources and targets (i.e. exposed electrical raceways, components, etc.), three methods of evaluation were used.

For Reactor Building areas, where likely fire ignition sources were identified, a detailed review was made of the plant components and cables that could potentially be impacted within the zone of influence (ZOI) of each fire source. EPRI FIVE fire modeling techniques were used to assess the damage potential of each fire source.

Also, EPRI FIVE guidance was used to calculate the probability of target damage due to transient fire sources. This process is described in Section 6.1.

Due to the specific nature of the Control Room, guidance for the evaluation of this area was taken directly from Appendix J of the Fire Risk Analysis Implementation Guide (Reference 4). This evaluation consisted of a review of the control functions that could be affected by potential fires in various locations within the Control Room and included allowance for recovery of the unaffected control functions following fire suppression.

For other plant areas, such as the Control Building and Turbine Building, a probabilistic model of fire behavior (i.e. an "event tree" methodology), was used to segment the area fire frequency into individual cases for evaluation. This evaluation is described in Section 6.2. This section includes a discussion of the evaluation of those plant locations for which multiple area fires were potentially of concern following the Fire Compartment Interaction Analysis (FCIA) performed in Phase I. Deterministic fire hazard assessment techniques, such as those used for Reactor Building areas, were not used for these remaining areas due to the difficulty in establishing specific fire source/target scenarios. Also, the detailed level of evaluation required for deterministic fire modeling was impractical for areas such as the Turbine Building. Also, deterministic methods were not judged to significantly enhance the fire damage assessment. A probabilistic approach was therefore selected as the most efficient method of assessing the fire damage potential for these areas.

Phase III The results of the fire hazard evaluation are shown in Section 7. This section lists the location within this report for the screening evaluation of all plant locations.

Finally, the resolution of outstanding fire-related issues, including response to the issues arising from the Sandia Laboratories Fire Risk Scoping Study (NUREG/CR-5088) is described in Section 8.

Revision 1 to the PRA plant model was used to perform the quantitative portions of this evaluation. This revision incorporates numerous individual changes, primarily in the area of plant response to loss of offsite power, to the Revision 0 plant model that was described in the initial IPE submittal.

FIRE INDUCED VULNERABILITY EVALUATION (FIVE) METHODOLOGY

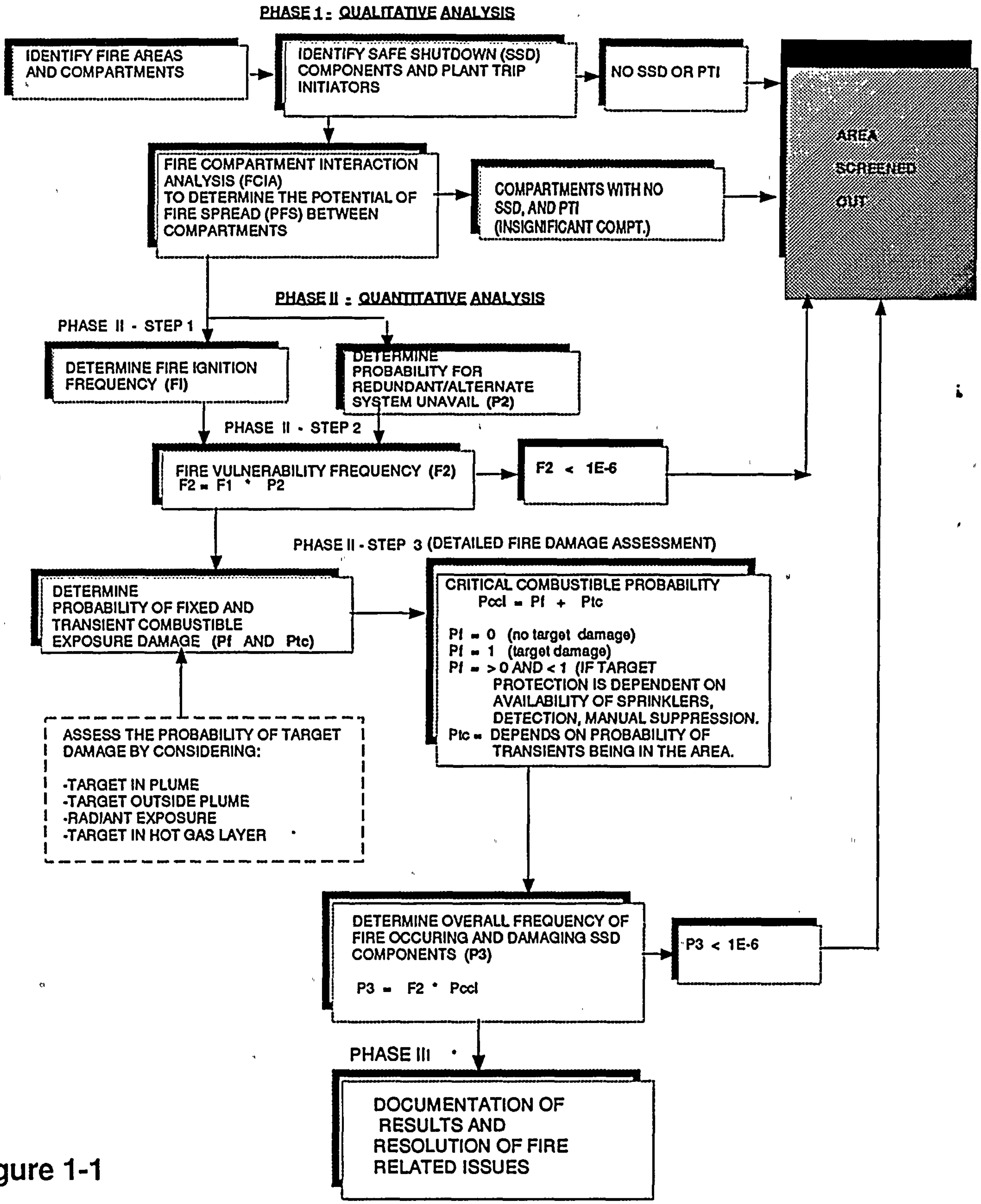


Figure 1-1

five flow diag.-1draw

FIRE INDUCED VULNERABILITY EVALUATION (FIVE)
REPORT ORGANIZATION

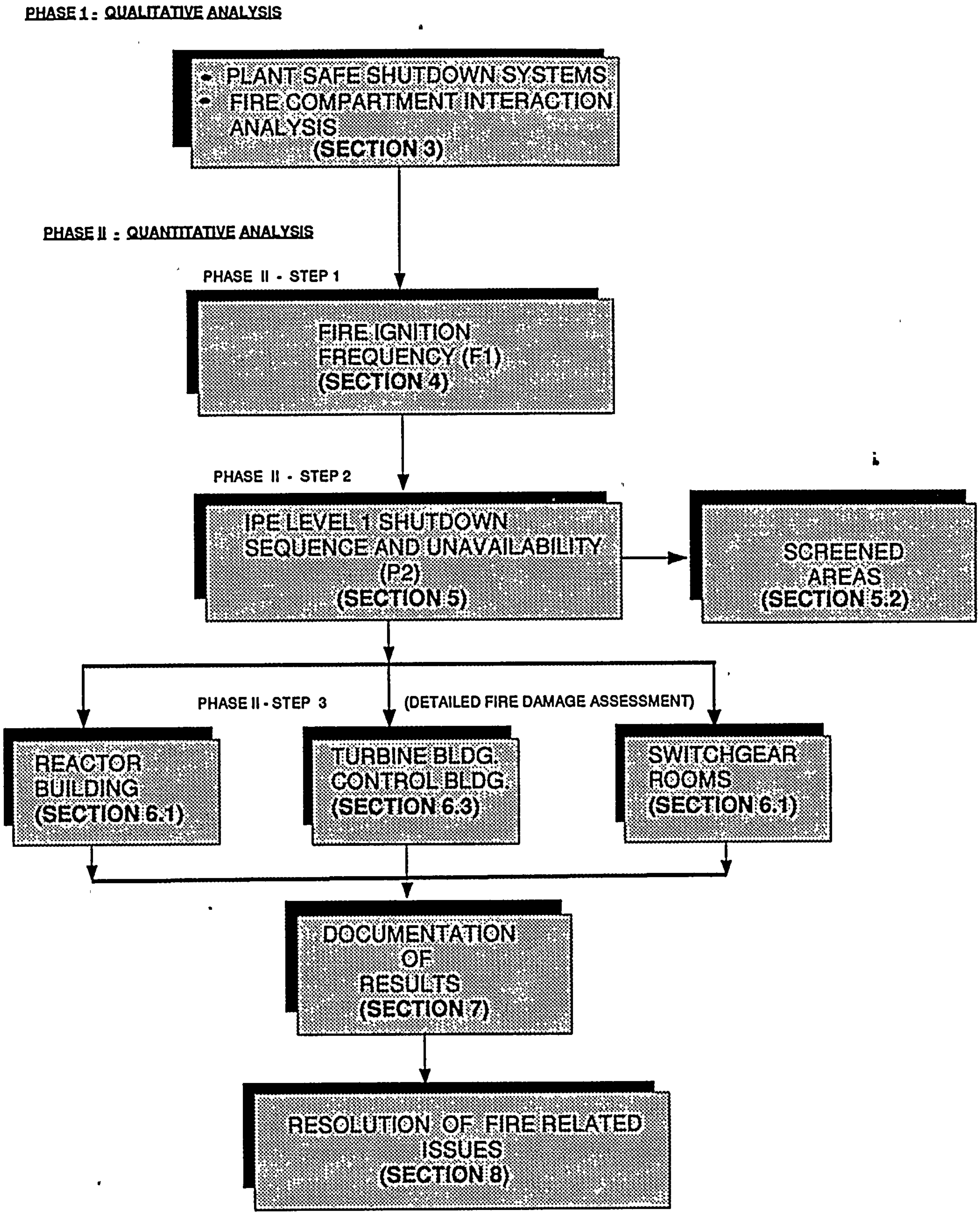


Figure 1-2

FIRE INDUCED VULNERABILITY EVALUATION (FIVE)

IMPLEMENTATION OF FIVE METHOD

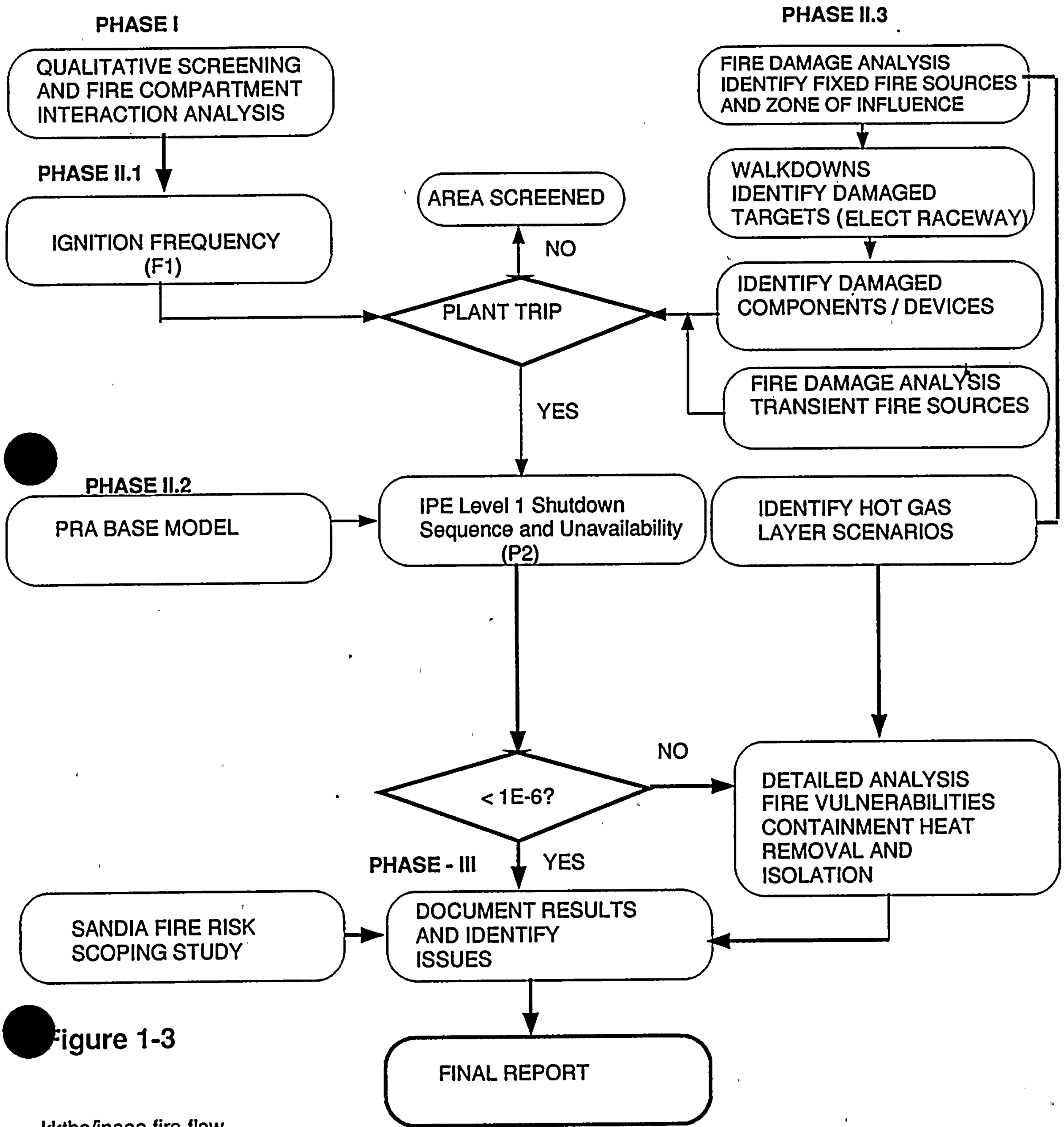


Figure 1-3

kktbo/ipeee fire.flow

2. PLANT SPECIFIC DATA

The Browns Ferry Nuclear Plant is located along the Tennessee River in northern Alabama. The plant data described in this report is specific to Unit 2, but includes potential fire ignition sources that are located in the Unit 1 and Unit 3 Reactor Buildings.

2.1 Number of Units and Plant Locations

The Browns Ferry Nuclear Plant consists of three similar boiling water reactor (BWR) units, which are located adjacent to each other. Each unit has a dedicated Reactor Building and Units 1 and 2 share a common Diesel Generator Building. The Unit 3 Diesel Generator Building is located opposite the Unit 1 and 2 Diesel Generator Building, on the other side of the Unit 3 Reactor Building. All three units share a common Turbine Building, Intake Structure and Switchyard. The common Control Building area is located between the Turbine Building and the Reactor Buildings. All three Control Room areas are located on the same elevation of the Control Building, with the Unit 1 and 2 Control Room areas located in the same room. Two Cable Spreading Rooms are located below the Control Room elevation, as shown in Figure 3-2.

The unit 1 and 2 essential 4kV switchgear is divided among four shutdown board rooms, with shutdown board rooms A and B located in the Unit 1 Reactor Building and shutdown board rooms C and D located in the Unit 2 Reactor Building. Each of these rooms is analyzed as an individual fire area, which is separated from other plant fire areas by rated barriers. The Unit 3 essential 4kV switchgear is located in 4kV shutdown board rooms 3EA, 3EB, 3EC and 3ED. All four of these rooms are located in the Unit 3 Diesel Generator Building.

Essential AC loads are assigned to shutdown boards, which are normally powered from shutdown bus 1 (4kV shutdown boards A and B) or shutdown bus 2 (4kv shutdown boards C and D). Shutdown bus 1 is normally supplied from Unit 1 4kV unit board 1A and shutdown bus 2 is normally supplied from Unit 2 4kV unit board 2A. Unit boards 1B and 2B act as alternate supplies for shutdown buses 2 and 1, respectively, such that each shutdown bus has one supply line from each unit.

The plant unit boards are normally aligned to receive power from the main generator and the 500kV ring bus at each unit, such that a turbine trip will result in a shift to the startup bus for one of the shutdown buses, while the other shutdown bus remains unaffected. The startup buses are supplied from an independent offsite 161 kV source, which is supplied from Athens and Trinity, AL.

The first part of the fire ignition frequency calculation methodology described in the FIVE documentation requires that the various plant areas be assigned to generic types.

The types of generic areas identified within the FIVE methodology and the number of areas of each type identified at Browns Ferry are listed in Table 2-1, below.

Table 2-1
Tabulation of Generic Plant Area Types

Plant Location	Number of Similar Locations
Battery Room	3
Cable Spreading Room	1
Control Room	1
Diesel Generator Room	2
Intake Structure	1
Radwaste Area	1
Reactor Building (BWR)	3
Switchgear Room	15
Transformer Yard	1
Turbine Building	1

2.2 Fire Areas, Fire Zones and Compartments

The Browns Ferry Nuclear Plant Appendix R Analysis (Reference 18) considered 25 separate fire areas at the plant. Each of these areas is separated from any other adjacent fire areas by rated fire barriers. Of these areas, fire area 2, the Unit 2 Reactor Building, was further subdivided into 6 separate fire zones, as shown in Figure 3-1. Due to the availability of non-combustible barriers capable of substantially confining fires within the area, the following two additional fire areas were subdivided into compartments, specifically for this analysis:

Fire Area 16, Control Building, which includes the lower level Computer, Equipment and Auxiliary Instrument Rooms (compartment 16-1), the Cable Spreading Rooms (compartment 16-2) and the Control Room area itself (compartment 16-3). The general layout of the Control Building is shown in Figure 3-2.

Fire Area 25, Turbine Building, which includes the Turbine Building itself (compartment 25-1), the Pipe Tunnel (compartment 25-2) and the Intake Pump Station (compartment 25-3), as shown on Figure 3-3.

For purposes of this analysis, the terms fire area, fire zone and compartment will be used interchangeably to indicate the evaluation of an individual plant area.

Table 2-2
Browns Ferry Fire Areas, Fire Zones and Compartments

Area	Description
1	Unit 1 Reactor Building
2-1	Unit 2 Reactor Building, 519' through 565' Elevation (West side of Torus Area and Main Floor)
2-2	Unit 2 Reactor Building, 519' through 565' Elevations (East side of Torus Area and Main Floor)
2-3	Unit 2 Reactor Building, 593' Elevation, North Side
2-4	Unit 2 Reactor Building, 593' Elevation, South Side and RHR Heat Exchanger Rooms
2-5	Unit 2 Reactor Building, 621' Elevation and North Side of 639' Elevations
2-6	Unit 2 Reactor Building, South Side of 639' Elevation
3	Unit 3 Reactor Building
4	4kV Shutdown Board Room B (Unit 1 Reactor Building, 593' Elevation)
5	4kV Shutdown Board Room A and 250V Battery Room (Unit 1 Reactor Building, 621' Elevation)
6	480V Shutdown Board Room 1A (Unit 1 Reactor Building, 621' Elevation)
7	480V Shutdown Board Room 1B (Unit 1 Reactor Building, 621' Elevation)
8	4kV Shutdown Board Room D (Unit 2 Reactor Building, 593' Elevation)

Table 2-2 Browns Ferry Fire Areas, Fire Zones and Compartments	
Area	Description
9	4kV Shutdown Board Room C and 250V Battery Room (Unit 2 Reactor Building, 621' Elevation)
10	480V Shutdown Board Room 2A (Unit 2 Reactor Building, 621' Elevation)
11	480V Shutdown Board Room 2B (Unit 2 Reactor Building, 621' Elevation)
12	Shutdown Board Room F (Unit 3 Reactor Building, 593' Elevation)
13	Shutdown Board Room E (Unit 3 Reactor Building, 621' Elevation)
14	480V Shutdown Board Room 3A (Unit 3 Reactor Building, 621' Elevation)
15	480V Shutdown Board Room 3B (Unit 3 Reactor Building, 621' Elevation)
16-1	Control Building - 593' Elevation
16-2	Cable Spreading Rooms (Control Building, 606' Elevation)
16-3	Control Rooms (Control Building, 617' Elevation)
17	Unit 1 Battery and Battery Board Rooms (Control Building, 593' Elevation)
18	Unit 2 Battery and Battery Board Rooms (Control Building, 593' Elevation)
19	Unit 3 Battery and Battery Board Rooms (Control Building, 593' Elevation)
20	Unit 1 and 2 Diesel Generator Building
21	Unit 3 Diesel Generator Building
22	4kV Shutdown Board Rooms 3EA and 3EB, (Unit 3 Diesel Generator Building, 583' Elevation)
23	4kV Shutdown Board Rooms 3EC and 3ED (Unit 3 Diesel Generator Building, 583' Elevation)

Table 2-2 Browns Ferry Fire Areas, Fire Zones and Compartments	
Area	Description
24	4kV Bus Tie Board Room (Unit 3 Diesel Generator Building, 565' Elevation)
25-1	Intake Pump Station
25-2	Pipe Tunnel
25-3	Turbine Building

Each of these plant fire areas, fire zones and compartments was then assigned to a generic type of area, as described in Section 2.1, above. The allocation of fire ignition frequency among these areas, based on the type of plant location, is shown in the individual calculation sheet for each area in Attachment B.

Yard area fires, including the potential for propagation to the Turbine Building, were separately considered in Section 6.2.14.

2.3 Plant Wide Components

Following the generation of fire ignition frequencies by generic plant areas (described above), the EPRI FIVE documentation provides guidance for the assignment of fire ignition frequency for specific components that are located throughout the plant, such as electrical transformers, battery chargers, air compressors and ventilation subsystems. The specific plant locations for these components were then used to assign the remainder of the plant fire ignition frequency. These calculations are shown in the individual worksheets for each area shown in Attachment B.

The total number of plant-wide components of each type is summarized in Table 2-3, below.

Table 2-3 Tabulation of Plant-Wide Fire Ignition Sources	
Type of Component	Number
Air Compressors	26
Battery Chargers	34
Fire Protection Panels	40
Non-Qualified Junction Boxes (Allocated by Millions of BTU of Cable)	12,000
Non-Qualified Cable (In Millions of BTU)	12,000
Offgas/Hydrogen Recombiners	3
Motor Generator Sets	31
Transformers (Indoor)	47
Ventilation Subsystems	289

2.4 Cables (Heat of Combustion)

Allocation of combustible loading and fire ignition frequency due to cable insulation among plant areas is shown in Attachment B. In general, cable insulation is distributed among the plant buildings as follows:

Turbine Building	55%
Reactor Buildings	31%
Control Building	13%
Other Areas	11%
Total	100%

2.5 Types of Automatic Fire Suppression Systems

The failure and unavailability rates for the various types of automatic fire suppression systems installed at the Browns Ferry plant are summarized in Table 2-4, below.

Table 2-4 Failure/Unavailability Rates for Automatic Fire Suppression Systems	
Type of Automatic Suppression System	Failure Probability/ Unavailability Rate
Carbon Dioxide (CO_2)	4.0E-02
Halon	5.0E-02
Preaction System	5.0E-02
Wet Pipe Sprinkler System	2.0E-02
Deluge Sprinklers	5.0E-02

2.6 Sprinkler and Fire Detection Device Data

Sprinkler and fire detection device data is summarized in Table 2-5, below, for the devices installed at the Browns Ferry Nuclear Plant.

Table 2-5 Sprinkler and Fire Detection Device Data				
Type	Detector Name	Time Constant	Rated Actuation Temperature	Spacing
Smoke	Ionization/ Photoelectric	10	128° F	~30 feet
Heat	Rate Compensated	83 (RT1)	136° F	Varies
Sprinkler	Standard	100	175° - 286° F	~10 to 12 feet
	Quick Response	30	175° - 200° F 165°F	~10 to 12 feet ~10 feet (Cable Spreading Room)

3. QUALITATIVE SCREENING PROCESS (PHASE I)

During the fire hazard evaluation process, each fire area, fire zone and plant compartment was reviewed for potential impact on safe shutdown components (SSC) by fire. If a given plant area contains no safe shutdown components (SSC) and a plant trip initiator (PTI) does not exist due to fires in the area, the area can be screened from further consideration, provided that there is no potential of a fire spreading (PFS) to another area that does contain safe shutdown equipment or would result in a plant trip. The PFS from one compartment to another is evaluated under the Fire Compartment Interaction Analysis (FCIA), which is described in Section 3.3, below.

For the Browns Ferry Fire Hazard Evaluation, all plant fire areas were retained through this qualitative screening process.

3.1 Plant Safe Shutdown Systems

For purposes of this analysis, the plant safe shutdown systems are defined as those identified in the Level 1 PRA report (Reference 12). Each of these systems is divided into top events, which define the success or failure of a given system function. Partial degradation, such as the loss of one train of components within a multiple train system, is identified by the use of split fractions, which modify the failure rate for the given top event to account for available system components.

3.2 Fire Area Versus Safe Shutdown System Function Evaluation

For the purposes of the qualitative screening process, all plant fire areas, fire zones and compartments were assumed to contain safe shutdown components. Therefore, none of these plant areas were screened from consideration on this basis.

3.3 Fire Compartment Interaction Analysis (FCIA)

The EPRI FIVE guidance gives the following 6 criteria for screening the potential for a fire to spread across a fire boundary from further consideration:

1) Compartments that would have no adverse effect on safe shutdown capability.

2) Area boundary is fire rated at 2 or 3 hours.

3) Area boundary is fire rated at 1 hour with combustible loading below 80,000 BTU/ft^2.

4) The exposing compartment has a low combustible loading (less than 20,000 BTU/ft^2) and automatic fire detection.

5) The exposing and the exposed compartments both have a low combustible loading (less than 20,000 BTU/ft^2) and automatic fire detection.

6) Automatic fire suppression is installed over combustibles in the area and will prevent spread to adjacent compartments.

If a given area was confirmed to not contain safe shutdown components (SSC), did not have the potential to initiate a plant trip, either manual or automatic (PTI) and did not have a potential for fire spread (PFS) into an adjacent area that is not screened, the area can be screened from further consideration, based on qualitative analysis. For purposes of this evaluation, all areas were retained for quantitative evaluation.

For the following plant fire areas, all boundaries that are adjacent to other plant fire areas were confirmed to consist of fire rated boundaries with ratings of 2 to 3 hours. Therefore, the potential for fire spread into or out of these areas can be screened from further consideration, based on screening criteria 2, above.

Fire Area 1	Unit 1 Reactor Building
Fire Area 2*	Unit 2 Reactor Building
Fire Area 3	Unit 3 Reactor Building
Fire Area 4	4kV Shutdown Board Room B
Fire Area 8	4kV Shutdown Board Room D
Fire Area 12	Shutdown Board Room F
Fire Area 16*	Control Building
Fire Area 20	Unit 1 and 3 Diesel Generator Building
Fire Area 21	Unit 3 Diesel Generator Building
Fire Area 25*	Turbine Building

* Fire Areas 2, 16 and 25 were further subdivided into separate fire zones and compartments.

The results of the Fire Compartment Interaction Analysis for remaining plant fire areas, including the fire zones and compartments within fire areas 2, 16 and 25, are summarized in Table 3-1, below.

Table 3-1 Fire Compartment Interaction Analysis (for locations that are not bounded by 2 to 3 hour barriers)						
Fire Area, Fire Zone or Compartment	Adjacent Area	SSC (See Note 1)	PTI (See Note 2)	PFS	Screening Criteria	Comment
2-1	2-2	Yes	Yes	Yes	(Note 6)	5, 8, 9
	2-3	Yes	Yes	No	3, 6	1, 5, 8, 9
	2-4	Yes	Yes	No	3, 6	1, 5, 8, 9
2-2	2-3	Yes	Yes	No	3, 6	1, 5, 8, 9
	2-1	Yes	Yes	Yes	(Note 6)	5, 8, 9
	2-4	Yes	Yes	No	3, 6	1, 5, 8, 9
2-3	2-1	Yes	Yes	No	3, 6	1, 5, 8, 9
	2-2	Yes	Yes	No	3, 6	1, 5, 8, 9
	2-4	Yes	Yes	Yes	(Note 6)	5, 8, 9
	2-5	Yes	Yes	No	3, 6	1, 8, 9
2-4	2-1	Yes	Yes	No	3, 6	1, 8, 9
	2-2	Yes	Yes	No	3, 6	1, 8, 9
	2-3	Yes	Yes	Yes	(Note 6)	8, 9
	2-5	Yes	Yes	No	3, 6	1, 8, 9
2-5	2-3	Yes	Yes	No	3, 6	1, 8, 9
	2-4	Yes	Yes	No	3, 6	1, 8, 9
	2-6	Yes	Yes	Yes	(Note 6)	1, 8, 9
2-6	2-5	Yes	Yes	Yes	(Note 6)	1, 8
5	6	Yes	Yes	No	3	1, 5
	7	Yes	Yes	No	3	1, 5
6	5	Yes	Yes	No	3	1, 5
	7	Yes	Yes	No	3	1, 5

Table 3-1
Fire Compartment Interaction Analysis
(for locations that are not bounded by 2 to 3 hour barriers)

Fire Area, Fire Zone or Compartment	Adjacent Area	SSC (See Note 1)	PTI (See Note 2)	PFS	Screening Criteria	Comment
7	5	Yes	Yes	No	3	1, 5
	6	Yes	Yes	No	3	1, 5
9	10	Yes	Yes	No	3	1, 5, 8
	11	Yes	Yes	No	3	1, 5, 8
10	9	Yes	Yes	No	3	1, 5, 8
	11	Yes	Yes	No	3	1, 5, 8
11	9	Yes	Yes	No	3	1, 5, 8
	10	Yes	Yes	No	3	1, 5, 8
13	14	Yes	Yes	No	3	1, 5, 8
	15	Yes	Yes	No	3	1, 5, 8
14	13	Yes	Yes	No	3	1, 5, 8
	15	Yes	Yes	No	3	1, 5, 8
15	13	Yes	Yes	No	3	1, 5, 8
	14	Yes	Yes	No	3	1, 5, 8
16-1	16-2	Yes	Yes	Yes	(Note 3)	5, 8
	17	Yes	Yes	No	3, 6	1, 5, 8, 9
	18	Yes	Yes	No	3, 6	1, 5, 8, 9
	19	Yes	Yes	No	3, 6	1, 5, 8, 9
16-2	16-1	Yes	Yes	No	6	7, 8, 9
	16-3	Yes	Yes	No	6	7, 8, 9
	17	Yes	Yes	No	6	1, 7, 8, 9
	18	Yes	Yes	No	6	1, 7, 8, 9
	19	Yes	Yes	No	6	1, 7, 8, 9

Table 3-1 Fire Compartment Interaction Analysis (for locations that are not bounded by 2 to 3 hour barriers)						
Fire Area, Fire Zone or Compartment	Adjacent Area	SSC (See Note 1)	PTI (See Note 2)	PFS	Screening Criteria	Comment
16-3	16-2	Yes	Yes	No	(Note 4)	6, 8
17	16-1	Yes	Yes	No	3	1, 5, 8
	16-2	Yes	Yes	No	3	1, 5, 8
18	16-1	Yes	Yes	No	3	1, 5, 8
	16-2	Yes	Yes	No	3	1, 5, 8
19	16-1	Yes	Yes	No	3	1, 5, 8
	16-2	Yes	Yes	No	3	1, 5, 8
22	23	Yes	Yes	No	3	1, 5, 8
	24	Yes	Yes	No	3	1, 5, 8
23	22	Yes	Yes	No	3	1, 5, 8
	24	Yes	Yes	No	3	1, 5, 8
24	22	Yes	Yes	No	3	1, 5, 8
	23	Yes	Yes	No	3	1, 5, 8
25-1	25-2	Yes	Yes	No	2	(Note 5)
	25-3	Yes	Yes	No	2	(Note 5)
25-2	25-1	Yes	Yes	No	2	(Note 5)
	25-3	Yes	Yes	No	2	(Note 5)
25-3	25-1	Yes	Yes	No	2	(Note 5)
	25-2	Yes	Yes	No	2	(Note 5)

Notes:

(1) For purposes of the qualitative screening analysis, all plant compartments were conservatively assumed to contain safe shutdown or IPE plant model components.

(2) For purposes of the qualitative screening analysis, fires in all plant areas were conservatively assumed to result in either manual or automatic plant trip.

(3) The potential for fire spread from compartment 16-1 to 16-2 is discussed in Section 3.3.1, below. The detailed evaluation of this potential multiple area fire is presented in Section 6.2.7.

(4) The potential for fire spread from compartment 16-3 to 16-2 is discussed in Section 3.3.1, below.

(5) Separation between Turbine Building compartments is described in Section 3.3.2, below.

(6) Unit 2 Reactor Building fire zones 2-1/2-2, 2-3/2-4 and 2-5/2-6 are separated by 20 foot boundary areas, as shown in Figure 3-1, but there is no physical boundary between these fire zones. Therefore, heat and products of combustion could propagate from one compartment to the adjacent compartment. Evaluation of fire propagation of various fire sources in these areas is discussed in the detailed analysis presented in Section 6.1.

Comments for Table 3-1 are keyed as follows:

1) 1 hour fire barriers separate compartments.

2) 2 hour fire barriers separate compartments.

3) 3 hour fire barriers separate compartments.

4) Very low combustible loading in exposing compartment (less than 15 minute fire severity).

5) Low combustible loading in exposing compartment (less than 1 hour fire severity).

6) Moderate combustible loading in exposing compartment (between 1 and 2 hour severity).

7) High combustible loading in exposing compartment (over 2 hour fire severity).

8) Automatic fire detection in exposing compartment.

9) Automatic fire suppression in exposing compartment.

10) Very low combustible loading in exposed compartment.

3.3.1 Potential for Fire Spread Between Control Building Compartments

The Control Building consists of 3 main compartments, which are separated by floor elevation. The top elevation comprises the Control Rooms themselves, with the level below containing the Cable Spreading Rooms. The lowest elevation then contains other instrument and computer areas. This layout is shown in Figure 3-2.

Potential for fire spread from compartment 16-1 to 16-2. Compartment 16-1 comprises the 593 foot elevation of the Control Building, with the exception of the Unit 1, Unit 2 and Unit 3 battery and battery board rooms (fire areas 17, 18 and 19, respectively), as shown in Figure 3-2. Compartment 16-2 is the Cable Spreading Room area, which is located above at the 606 foot elevation. Addressable photoelectric smoke detectors are provided for the entire 16-1 compartment, including the MG set rooms, corridor, mechanical equipment room, communication room, computer rooms, auxiliary instrument rooms, process computer room, etc. for early warning fire detection, both locally and in the Control Room. Fire suppression coverage is provided for the majority of the areas that contain any significant level of combustibles. However, manually actuated suppression systems are provided in lieu of automatic systems in most areas to reduce the possibility of inadvertent actuation of toxic fire suppressants into a Control Building environment. Fire suppression systems are provided as follows:

Process computer room	Automatic Halon system
Auxiliary instrument rooms 1, 2 and 3	Manual CO_2 systems
Computer rooms 1, 2 and 3	Manual CO_2 systems

Hose stations and fire extinguishers are available throughout the area. Any fire in this area will be promptly detected due to the area wide detection coverage. If a significant fire did develop in this area, it would be contained and extinguished by the available fire suppression systems. The concrete floor slab separating these two compartments is equivalent to a fire resistance rating of 1.5 hours. However, penetrations exist in the slab which may not be sealed to meet the fire resistance rating of the floor itself. While these penetrations present a minimal potential for fire propagation to the Cable Spreading Room, the potential for this fire is, conservatively, being considered.

The quantitative evaluation of an unsuppressed fire developing in compartment 16-1 and growing to include compartment 16-2 is discussed in Section 6.2.7.

Potential for fire spread from compartment 16-3 to 16-2. Compartment 16-3 (Control Rooms) is located at the 617 foot elevation of the Control Building, directly above compartment 16-2 (Cable Spreading Rooms), which is located at the 606 foot elevation, as shown in Figure 3-2.

Addressable photoelectric smoke detectors are provided throughout compartment 16-3, including detectors located within the control panels themselves. Automatic fire suppression is not provided within the Control Room area. However, hose stations and fire extinguishers are located throughout the area to allow manual fire suppression. This area is occupied by plant operations personnel at all times. A fire is not likely to develop in this area without being detected. Thus, the fire can be quickly controlled and extinguished. Also, a fire will tend to propagate upward and is not likely to propagate down from the 617 foot elevation to the 606 foot elevation and a fire in the Control room (compartment 16-3) propagating to the Cable Spreading Rooms (compartment 16-2) will not have the potential to damage more equipment than a fire in the Control Room alone. Therefore, a fire in compartment 16-3 is not judged likely to propagate to compartment 16-2.

3.3.2 Potential for Fire Spread Between Turbine Building Compartments

The Turbine Building fire area is segmented for this analysis into 3 compartments, as shown on Figure 3-3. The Intake Pump Station and its associated Cable Tunnel comprise compartment 25-1 and the Pipe Tunnel area comprises compartment 25-2. The Turbine Building itself then comprises compartment 25-3.

Potential for fire spread between compartments 25-1 and 25-3. The Intake Pump Station, compartment 25-1, is connected to the Turbine Building, compartment 25-3, through an underground Cable Tunnel, as shown in Figure 3-3. The Cable Tunnel runs approximately 650 feet from the Intake Pump Station at the 550 foot elevation to the electrical cable shaft, which opens into the Turbine Building at the 565 foot elevation. The entire Cable Tunnel and the cable shaft are constructed of reinforced concrete, exceeding a 3 hour fire resistance rating. The cable shaft extends approximately 8 feet above the Turbine Building floor at the 565 foot elevation. The Cable Tunnel is protected with an automatic fire detection system (smoke and linear beam detectors) that provide annunciation in the Control Room. Entrance to the Cable Tunnel shaft is strictly controlled by plant security personnel. The grated steel door entrance to the shaft is kept locked at all times. No combustibles are stored in the Cable Tunnel, therefore no fire exposure hazard is present. The cables in the cable trays are coated with a flame retardant material (Flamastic) or are qualified to IEEE-383 standards.

An internally generated cable tray fire is judged to be unlikely, since the circuits are protected with a fuse or circuit breaker that will actuate to isolate the cable prior to the jacket of a faulted cable reaching its auto-ignition temperature or reaching its insulation damage temperature for all credible low impedance and bolted faults. Therefore, as described above, compartments 25-1 and 25-3 are separated by barriers exceeding 3 hour fire resistance ratings, with the exception of the opening to the Cable Tunnel shaft itself. The unique configuration of this opening, however, as well as the protective features provided, will limit the potential of fire spread from one compartment to the other.

Fire spread between these areas is therefore screened from further consideration based on EPRI FIVE criterion 2.

Potential for fire spread between compartments 25-2 and 25-1 or 25-3. The Pipe Tunnel below the Turbine Building, compartment 25-2, is located at the 565 foot elevation and is separated from compartment 25-1 by a 15 inch reinforced concrete wall that exceeds a 3 hour fire rated construction. Therefore, fire growth between these areas can be screened from further consideration, based on EPRI FIVE criterion 2. Since there are negligible amounts of combustible materials located in the Pipe Tunnel, compartment 25-2, there is minimal potential for fire spread to compartment 25-3. There are two interface points between these compartments, which are located at stairwells 12 and 19, as shown in Figure 3-3.

For a fire to spread from one of these areas to the other, either

1. The fire would have to spread from stairwell 12, down to the backwash receiving room (533.0-T-1), through door 211 and then up the ladder to the entrance of the Pipe Tunnel (point 2).

2. The fire would have to spread from stairwell 19, down to the backwash receiving room (533.0-T-3), up a ladder into the Pipe Tunnel access.

Neither of these propagation paths is judged to be credible. Therefore, the potential for fire spread between these areas along these paths is not separately considered and fire growth between these areas can be screened from further consideration, based on EPRI FIVE criterion 2.

Potential for fire spread between the Turbine Building and the Service/Radwaste Building. The Turbine Building (compartment 25-3) is also adjacent to the Service/Radwaste Building. These areas are separated by a reinforced concrete wall, with the main access between them continuously manned. Doors between these areas are of heavy steel construction. Few penetrations exist in the adjacent wall. Therefore, fire propagation between these areas is judged to be unlikely. It was also noted during this review that the Service/Radwaste Building area does not contain any safe shutdown components or plant trip initiators.

UNIT 2 REACTOR BUILDING
FIRE ZONES

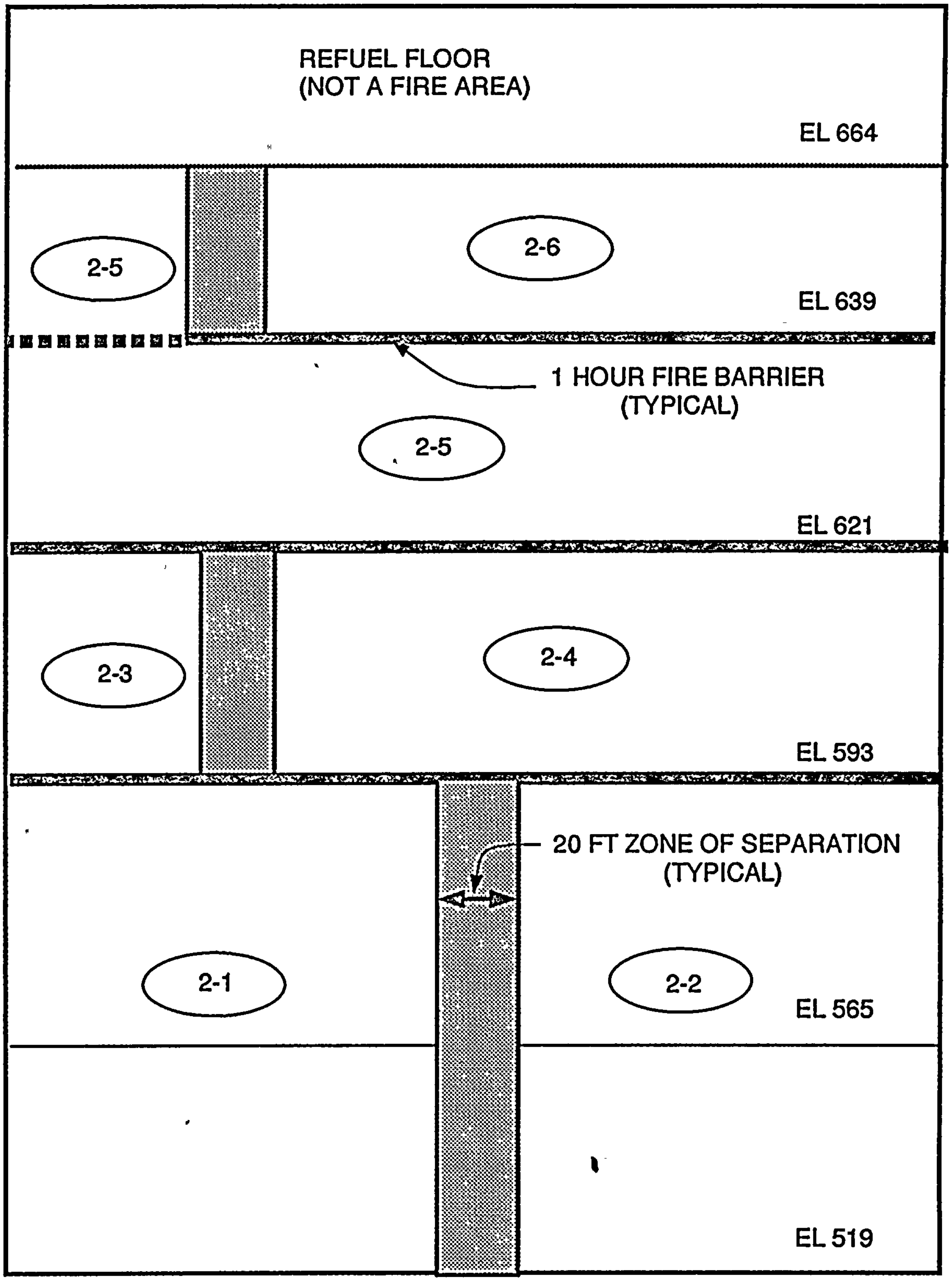


Figure 3-1

CONTROL BUILDING
FIRE AREA 16 COMPARTMENTS

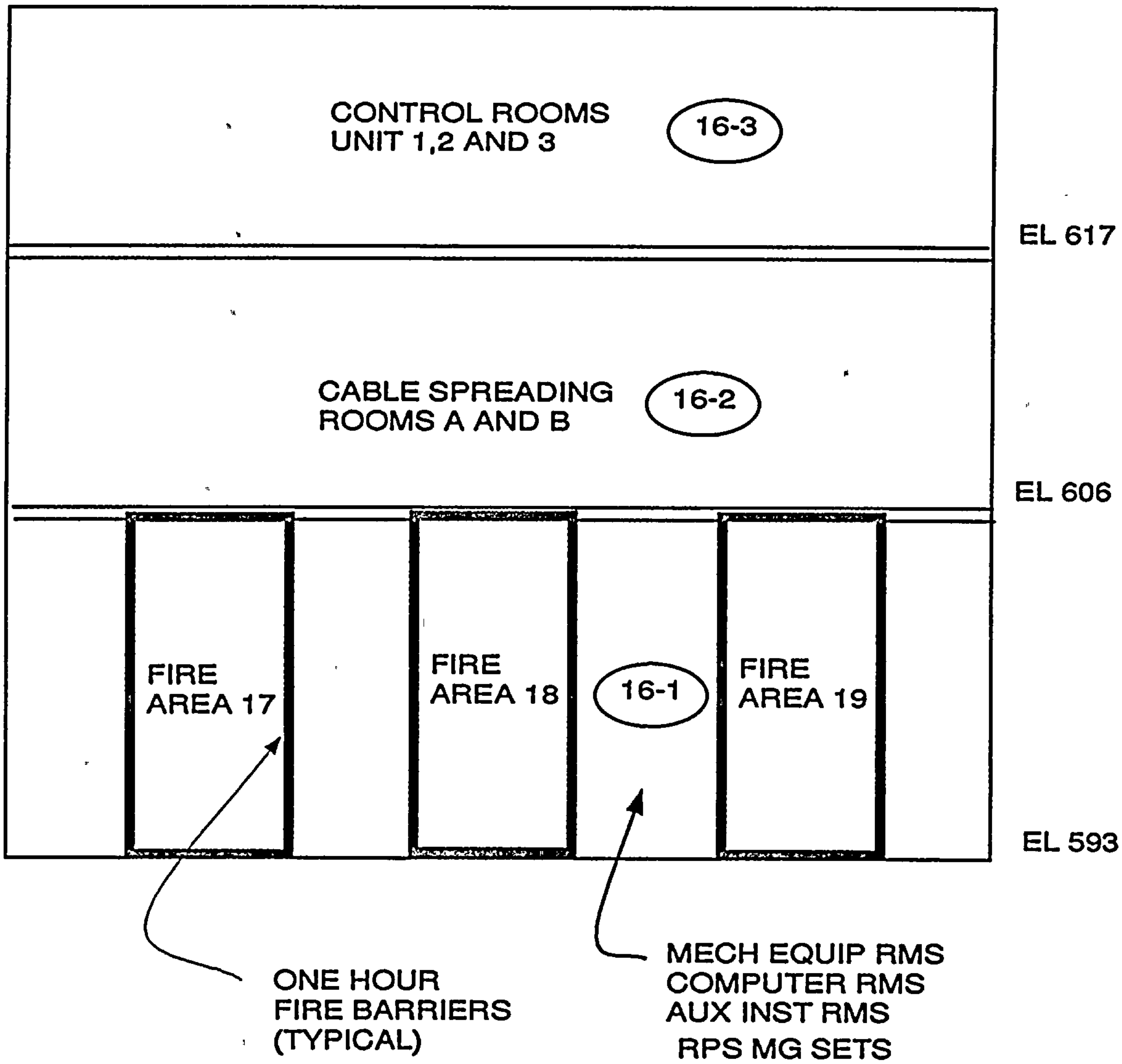
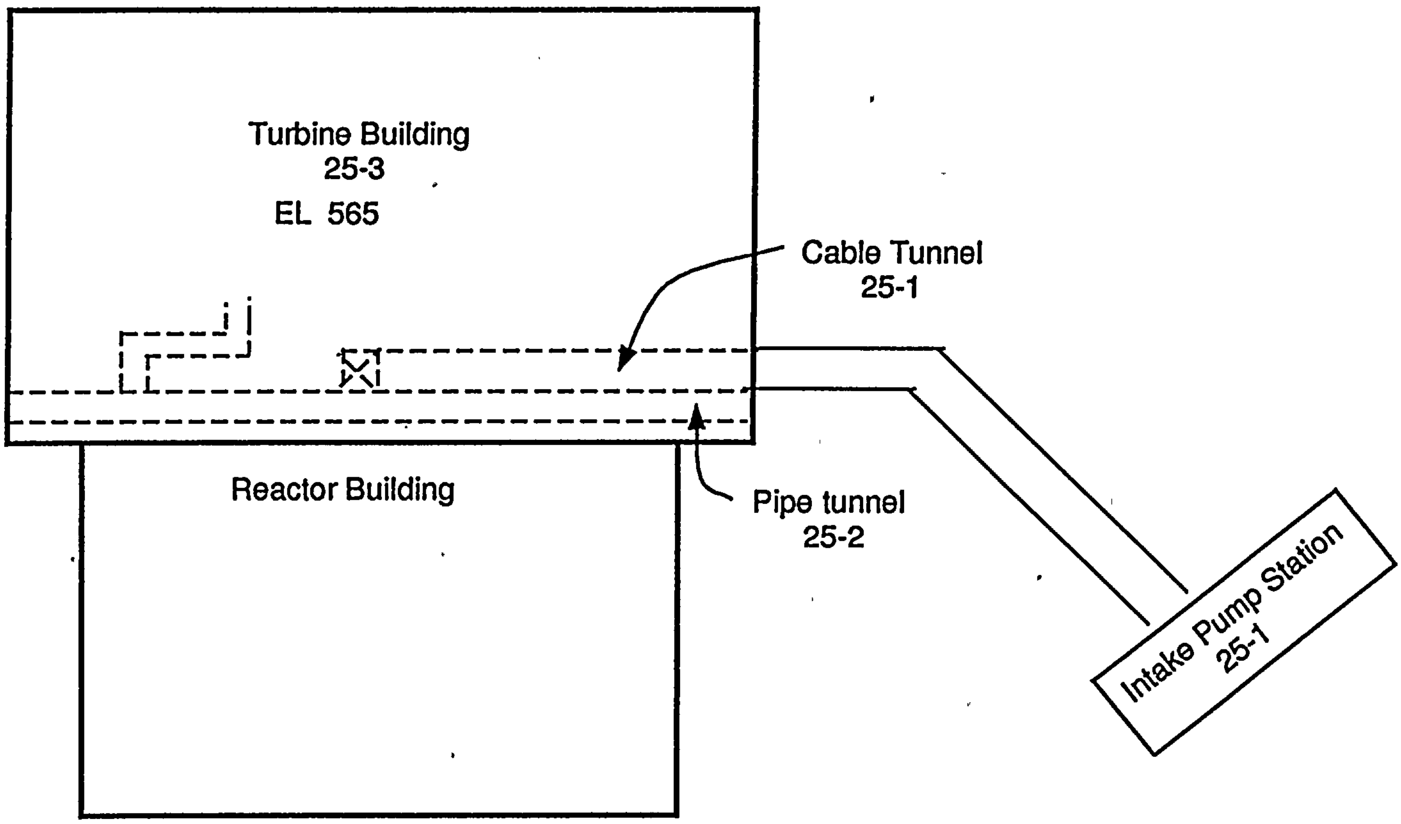
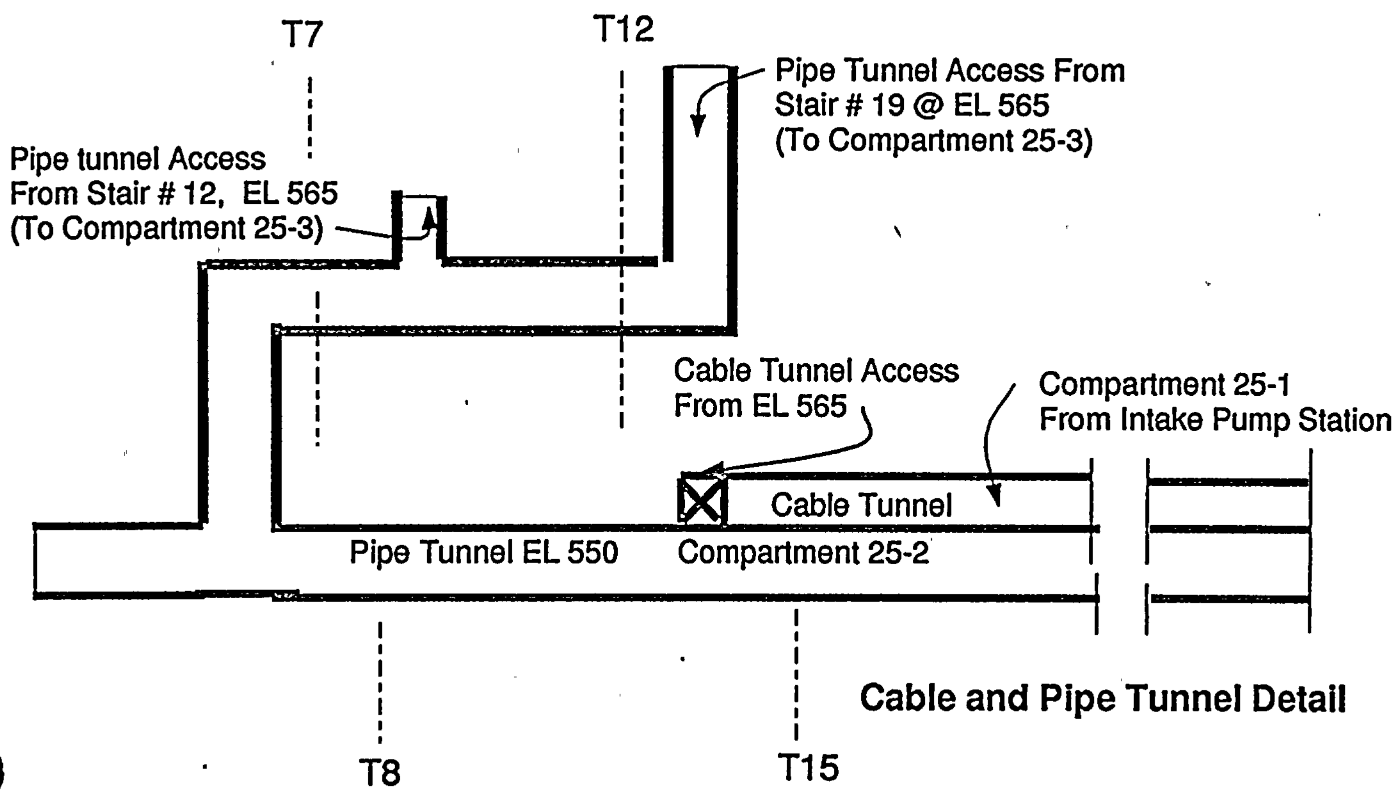


Figure 3-2

control bldg zones

Turbine Building Fire Compartments





Cable and Pipe Tunnel Detail

Figure 3-3

turbine bldg zones

4. PHASE II.1 - CALCULATION OF FIRE IGNITION FREQUENCY (F1)

All plant fire areas were retained through the qualitative screening process for quantitative evaluation in Phase II of the FIVE methodology. In Phase II.1, fire ignition frequencies are generated for each of these areas. These frequencies are generated in the following stepwise fashion:

1. Each plant area is assigned to a generic "type" of area, such as a Reactor Building area or a switchgear room.

2. Based on this assignment, generic plant fire frequencies are assigned to each plant specific location, based on features of the area, such as the number of pumps and panels for Reactor Building areas.

3. Following the allocation of fire frequency by plant area, identified plant wide components, such as elevator motors, are located within the individual areas of the plant and the associated fire ignition frequency is allocated to the area, based on a plant specific weighting for each of the given components. This allocation process is described in the EPRI FIVE documentation

Following quantification of the fire ignition frequency, the area may be screened from further consideration on the following quantitative basis:

> If the fire ignition frequency (F1) for the area is less than 1E-06, the area can be screened from further consideration.

As a practical matter, this screening criteria is rarely used to remove an area from further consideration, due to the relatively high (i.e. compared to 1E-06) fire ignition frequency associated with virtually any fire source in the plant.

The FIVE documentation (Reference 1), then, provides a basis for the generation of fire ignition frequencies for each of the areas, zones and compartments throughout the Browns Ferry plant. In general, this consists of allocation of a "generic" frequency based on either plant location or the presence of certain "plant-wide" components that were identified as fire sources during the EPRI review of the Fire Events Database (FEDB), as described in NSAC/178L.

The actual calculation of fire area ignition frequency is shown in Attachment B to this report. It should be noted that these fire ignition frequencies represent all fires that could be expected to occur in the plant, regardless of fire severity or whether the fire would cause or result in a plant trip. For reference, the fire ignition frequencies generated for the Browns Ferry plant areas are listed in Table 4-1, below.

Table 4-1 Fire Area Ignition Frequencies		
Area	Description	Frequency (All Fires)
1	Unit 1 Reactor Building	9.239E-02
2-1	Unit 2 Reactor Building, 519' through 565' Elevations (West side of Torus Area and Main Floor)	1.870E-02
2-2	Unit 2 Reactor Building, 519' through 565' Elevations (East side of Torus Area and Main Floor)	1.737E-02
2-3	Unit 2 Reactor Building, 593' Elevation, North Side	5.335E-03
2-4	Unit 2 Reactor Building, 593' Elevation, South Side and RHR Heat Exchanger Rooms	2.092E-02
2-5	Unit 2 Reactor Building, 621' Elevation and North Side of 639' Elevations	2.869E-02
2-6	Unit 2 Reactor Building, South Side of 639' Elevation	1.962E-02
3	Unit 3 Reactor Building	9.260E-02
4	4kV Shutdown Board Room B (Unit 1 Reactor Building, 593' Elevation)	6.743E-03
5	4kV Shutdown Board Room A and 250V Battery Room (Unit 1 Reactor Building, 621' Elevation)	8.569E-03
6	480V Shutdown Board Room 1A (Unit 1 Reactor Building, 621' Elevation)	6.644E-03
7	480V Shutdown Board Room 1B (Unit 1 Reactor Building, 621' Elevation)	6.644E-03
8	4kV Shutdown Board Room D (Unit 2 Reactor Building, 593' Elevation)	6.644E-03
9	4kV Shutdown Board Room C and 250V Battery Room (Unit 2 Reactor Building, 621' Elevation)	8.074E-03
10	480V Shutdown Board Room 2A (Unit 2 Reactor Building, 621' Elevation)	6.644E-03

Table 4-1
Fire Area Ignition Frequencies

Area	Description	Frequency (All Fires)
11	480V Shutdown Board Room 2B (Unit 2 Reactor Building, 621' Elevation)	6.644E-03
12	Shutdown Board Room F (Unit 3 Reactor Building, 593' Elevation)	7.247E-03
13	Shutdown Board Room E (Unit 3 Reactor Building, 621' Elevation)	7.148E-03
14	480V Shutdown Board Room 3A (Unit 3 Reactor Building, 621' Elevation)	6.644E-03
15	480V Shutdown Board Room 3B (Unit 3 Reactor Building, 621' Elevation)	6.644E-03
16-1	Control Building - 593' Elevation	5.892E-02
16-2	Control Building - 606' (Cable Spreading Room)	1.344E-02
16-3	Control Building - 617' (Control Room)	3.534E-02
17	Unit 1 Battery and Battery Board Room, Control Building 593' Elevation	2.194E-02
18	Unit 2 Battery and Battery Board Room, Control Building 593' Elevation	2.094E-02
19	Unit 3 Battery and Battery Board Room, Control Building 593' Elevation	2.094E-02
20	Unit 1 and 2 Diesel Generator Building	1.241E-01
21	Unit 3 Diesel Generator Building	1.237E-01
22	4kV Shutdown Board Room 3EA and 3EB, 583' Elevation, Unit 3 Diesel Generator Building	6.674E-03
23	4kV Shutdown Board Room 3EC and 3ED, 583' Elevation, Unit 3 Diesel Generator Building	6.674E-03
24	4kV Bus Tie Board Room, 565' Elevation, Unit 3 Diesel Generator Building	6.660E-03

Table 4-1 Fire Area Ignition Frequencies		
Area	Description	Frequency (All Fires)
25-1	Intake Pump Station	3.581E-02
25-2	Pipe Tunnel	9.875E-06
25-3	Turbine Building	4.500E-01
Total Plant Fire Frequency (for 3 units)		1.305

5. PHASE II.2 - QUANTITATIVE SCREENING

Following the generation of the fire ignition frequencies for each of the fire areas that remain for further quantitative evaluation, each plant area must be evaluated for the probability of core damage, given an assumed engulfing fire in the area and the unavailability of safe shutdown components located outside the area of the fire. Within the EPRI FIVE documentation, this probability is identified as P2.

Plant area walkdowns were performed to confirm the impacts that a potential fire in a given area could have on plant equipment required for safe shutdown, as identified in the IPE plant model. This included any potential impacts on electrical cables. These walkdowns are described in Attachment D.

Once the conditional core damage frequency, or P2 value, has been generated for each of the fire areas under consideration, these values can be combined with the fire ignition frequencies (F1 values) from Table 4-1 to calculate an upper bound core damage frequency (F2 = F1 x P2). If this value is less than 1E-06, the area can be screened from further consideration. The potential for fire-induced containment bypass scenarios for areas that are screened from further consideration with a fire-related core damage frequency above 1E-07 is discussed separately in Section 5.3.

The core damage frequency generated by this process (F2) is considered to be an upper bounding value for the following reasons:

1. All fires in a given area are assumed to either cause and automatic plant trip or result in a manual reactor scram, regardless of fire severity or location. Where equipment failure in an area could possibly result in an automatic plant trip, such as MSIV closure, that form of initiating event was used to quantify the plant model. This is conservative in that many of the fires listed in the EPRI Fire Events Database (NSAC/178L) were suppressed without power reduction or plant trip.

2. All fires in a given area, regardless of severity, location or available suppression and detection systems, are assumed to engulf the area, failing all safe shutdown components and support cables in the area.

The reader should therefore be cautioned against interpreting these values as any more than the results of a bounding analysis, which is performed to enable a screening of less-significant areas from further consideration and identifying those areas for which a detailed analysis of fire hazards is warranted.

The evaluation of each fire area is described in Section 5.1, below. The results of this evaluation are then summarized in Section 5.2. For those areas that were not screened from further consideration in this process, detailed analysis is performed in Section 6.

These areas that were not screened in this section are listed in Table 6-1.

5.1 IPE Level 1 Shutdown Sequence and Unavailability (P2)

Given the plant impacts for an assumed engulfing fire in a given area, the Level 1 plant model from the IPE is used to develop a list of core damage scenarios, based on the likelihood of hardware failure and equipment unavailability. These core damage scenarios are then totalled and normalized to reflect an initiating event frequency of 1.0. This gives what is known as the conditional core damage frequency for the fire event under consideration, which corresponds to the P2 value described in the EPRI FIVE documentation. This conditional value is then multiplied by the fire frequency to generate a fire-related core damage frequency (P2 x F1 = F2).

This evaluation is performed by manually modifying the plant model logic, or rule, structure to incorporate the random and fire-induced failures of given plant components, as reflected in the use of failed of degraded "split fraction" values for the impacted "top events." These top events are used within the plant model logic structure to model the various individual plant system functions. For reference, the top events used within the Browns Ferry Level 1 plant model are listed in Table 3-1. This listing is in the order in which these system functions are questioned within the plant model.

If the cause of plant trip (i.e. loss of offsite power, MSIV closure, etc.) is known, the pre-existing logic structure for this plant trip, or "initiating event," is used to generate a P2 value. If no specific reason for plant trip can be identified, manual reactor trip is conservatively assumed to occur. At this level of analysis, all fires are assumed to result in a plant trip.

In the interest of reducing the length of time required for model quantification at this level of analysis (typically 10 to 12 hours per initiating event), the plant model module that is normally used to assign plant damage states (GTPDS10) was not used. Also, only the core damage top event (NCD) was used from the CNTMT module, which was first copied to a separate module, FIRECD.

It should be noted that, in several plant areas, such as the Control Building and the Reactor Buildings, the conditional core damage frequency resulting from a fire is conservatively assigned a value of 1.0. That is, all fires are assumed to result in core damage. This ensures that these areas will be retained for detailed analysis, as described in Section 6 of this report.

For reference, the conditional core damage frequencies, or P2 values, as they are identified in the EPRI FIVE documentation, for several of the IPE initiating events are shown in Table 5-1 below. These values were taken from the Rev. 1 quantification of the IPE plant model.

Table 5-1 Conditional Core Damage Frequencies for Selected Initiating Events				
Description of Plant Trip	Initiating Event Designation	Initiating Event Frequency (A)	Core Damage Frequency (B)	CCDF (P2 = B/A)
Inadvertent MSIV Closure	CIV	4.34E-01	7.34E-07	1.69E-06
Loss of Main Condenser Vacuum	LOCV	2.72E-01	4.00E-07	1.47E-06
Total Loss of Feedwater	LOFW	3.59E-01	5.18E-07	1.44E-06
Partial Loss of Feedwater	PLFW	3.31E-01	1.13E-07	3.41E-07
Loss of Offsite Power	LOSP	3.39E-02	1.57E-06	4.63E-05
Manual Scram Required	SCRAMR	2.74E-01	8.64E-08	3.15E-07
Turbine Trip	TT	1.42	5.37E-07	3.78E-07

The initiating event frequencies and core damage frequencies shown in Table 5-1 were taken directly from Riskman data files. These values were then "normalized" to reflect an initiating event frequency of 1.0 to obtain the conditional core damage frequency (P2 value), which assumes an initiating event frequency of 1.0 and is later adjusted by a fire frequency value (F1) to generate a fire-related core damage frequency.

For example, if the operator was expected to trip the reactor in response to a given fire, with a fire ignition frequency (F1) of 1E-02 and no plant components damaged by the fire itself, this would be similar to the SCRAMR initiating event, shown above. Table 5-1 shows the core damage frequency that would be reported by the quantification program for the SCRAMR initiating event (8.64E-08). In order to determine the core damage frequency that would result from this new initiating event, the analyst would first determine the core damage frequency for the SCRAMR initiating event, if it had an initiating event frequency of 1.0 (8.64E-08 / 2.74E-01 = 3.15E-07). This value would then be multiplied by the new initiating event frequency (1E-02) to calculate a fire-related core damage frequency of 3.15E-09.

5.1.1 Fire Area 1 - Unit 1 Reactor Building

Browns Ferry Unit 1 is currently a non-operating unit. Also, damage to "unit-specific" (i.e. Unit 1) components due to postulated fire scenarios in the Unit 1 Reactor Building would not be expected to require shutdown of Unit 2 (i.e. a plant trip, or initiating event, would not be expected to occur for Unit 2 due to fires in the Unit 1 Reactor Building). Damage to "unit-common" components, such as power cables, that transit through the Unit 1 Reactor Building, however, may require Unit 2 to be shut down or tripped. A detailed analysis of both unit-specific and common components is provided for the Unit 2 Reactor Building fire zones in the detailed analysis (Section 6.1). This detailed analysis includes evaluation of the probability of critical combustible loading and conditional core damage frequencies due to fires in each of these areas. This detailed evaluation is judged to bound the core damage frequency due to fire-related initiating events in the Unit 1 Reactor Building. Therefore, refer to the detailed analysis of Unit 2 Reactor Building fire zones in Section 6.1 as a bounding case for fires in the Unit 1 Reactor Building.

For a fire in the Unit 1 Reactor Building, it is unlikely that the operator would initiate a reactor trip on Unit 2. For purposes of this screening analysis, though, all Unit 1 Reactor Building fires are conservatively assumed to result in a precautionary trip of Unit 2. At this level of analysis, a conditional core damage frequency of 1.0 is assumed for this fire area. Further discussion of fires in this area is provided in Section 6.1.4.

5.1.2 Fire Area 2 - Unit 2 Reactor Building

The Unit 2 Reactor Building consists of six fire zones, which are analyzed as individual fire areas in volume 1 of the Browns Ferry Fire Protection Report. The schematic layout of these areas is shown in Figure 3-1. Due to the involved nature of the components and support cables located in these areas and the potential for multiple fire zone involvement, these fire zones are evaluated with an assumed conditional core damage frequency of 1.0 for this level of evaluation. These areas are analyzed in more detail in Section 6.1.

The resulting upper bound core damage frequencies for each of these fire zones are shown, for completeness, in Table 5-2, below.

5.1.3 Fire Area 3 - Unit 3 Reactor Building

Browns Ferry Unit 3 is currently a non-operating unit. Also, damage to "unit-specific" (i.e. Unit 3) components due to postulated fire scenarios in the Unit 1 Reactor Building would not be expected to require shutdown of Unit 2 (i.e. a plant trip, or initiating event, would not be expected to occur for Unit 2 due to fire in the Unit 3 Reactor Building). Damage to "unit-common" components however, such as power cables, that transit through the Unit 3 Reactor Building, however, may require Unit 2 to be shut down or tripped.

A detailed analysis of both unit-specific and common components is provided for the Unit 2 Reactor Building fire zones in the detailed analysis (Section 6.1). This detailed analysis includes evaluation of the probability of critical combustible loading and conditional core damage frequencies for fires in each of these areas. This detailed evaluation is judged to bound the core damage frequency due to fire-related initiating events in the Unit 3 Reactor Building. Therefore, refer to the detailed analysis of Unit 2 Reactor Building fire zones in Section 6.1 as a bounding case for the Unit 3 Reactor Building.

As in the case of fires in the Unit 1 Reactor Building, it is unlikely that the operator would initiate plant trip of Unit 2 due to a fire in the Unit 3 Reactor Building. For purposes of analysis, though, all Unit 3 Reactor Building fires are conservatively assumed to result in a precautionary trip of Unit 2. At this level of analysis, a conditional core damage frequency of 1.0 is assumed for this fire area. Further discussion of fires in this area is provided in Section 6.1.4.

5.1.4 Fire Area 4 - 4kV Shutdown Board Room B

Plant walkdowns (see Attachment D) confirm that the following significant plant equipment is located in this fire area:

4kV Shutdown Board B
480V RMOV Board 1B
250V RMOV Board 1B
1-LPNL-925-0541 (ACU 1B Control Panel)
1-TS-031-7205D (Shutdown Board Room ACU 1A)
1-TS-031-7206C (Shutdown Board Room air cooling unit 1B)
Panel 0-PNL-25-45B (4kV Shutdown Board B Logic Relays)
Division I ECCS Analog Trip Unit Inverters (Unit 1 only)
I&C Bus 1B Equipment

The potential fire-related failure of 4kV shutdown board B could also impact the operation of the following additional plant components:

Diesel Generator B	Top Event GB
Shutdown Bus 1	Top Event SHUT1
Shutdown Bus 2	Top Event SHUT2
480V Shutdown Board 2A	Top Event RS
480V RMOV Board 2A	Top Event RH

These walkdowns also evaluated cable routing through this area. During this evaluation, it was determined that cable ES1812-IA supports voltage indication only for 4kV shutdown board A. This circuit was confirmed to provide indication only, is protected by fuses, and does not impact the operation of shutdown bus A.

The Failure Modes and Effects Analysis of Browns Ferry Unit 2 Key Support Systems (Table 3.1.1-2 of the IPE submittal) states that a plant trip would not be expected to occur following loss of 4kV shutdown board B. Therefore, at this level of analysis, a manual reactor trip is assumed to occur. The failure of the components located in this area was modeled by failing top events AB (4kV shutdown board B) and RF (480V RMOV board 1B) in plant model module ELECT12, in addition to the potential fire-related impacts listed above. It was noted during this evaluation that the assumed loss of both shutdown buses is equivalent to a loss of offsite power to the 4kV shutdown boards. This evaluation then generated a conditional core damage frequency of 9.17E-04. Given a fire ignition frequency of 6.74E-03 for this area from Table 4-1, the upper bound conditional core damage frequency for this area can be evaluated as

F2 = 6.74E-03 x 9.17E-04 = **6.20E-06**

Since the upper bound core damage frequency for this evaluation is greater than 1E-06, fires in this area cannot be screened from further consideration at this level of analysis. This evaluation is conservative in that all fires in this area are assumed to result in a manual reactor trip and cause the loss of all plant equipment located in this fire area, regardless of fire severity or manual fire suppression. This equipment includes an assumed failure of both shutdown buses to supply all other 4kV boards, similar to a loss of offsite power. More detailed analysis of this area is provided in Section 6.2.

5.1.5 Fire Area 5 - 4kV Shutdown Board Room A and 250V Battery Room

Plant walkdowns (see Attachment D) confirm that the following major plant equipment is located in this fire area:

4kV Shutdown Board A
480V RMOV Board 1A
250V RMOV Board 1A
Panel 25-32 (Backup Control Panel)
Panel 25-45A (4kV Shutdown Board A Relays)
I&C Bus 1B Equipment
250VDC Battery SB-A and B
250VDC Battery chargers SB-A and B
250VDC Distribution Panel SB-A and B
Shutdown Board Room Emergency Cooling Unit 1-ACU-31-110
I&C Bus 1A Equipment
ATU Inverters - Division II (Unit 1)

These walkdowns also evaluated cable routing through this area. During this review, it was identified that control cables associated with diesel generator B (top event GB) are routed through this area and a fire-related failure of 250VDC distribution panel SB-B could

potentially disable generator B output breaker, disabling the DG.

Also, the fire-related failure of 4kV shutdown board A could potentially impact the operation of the following additional plant components:

Diesel Generator A	Top Event GA
Shutdown Bus 1	Top Event SHUT1
Shutdown Bus 2	Top Event SHUT2

The Failure Modes and Effects Analysis of Browns Ferry Unit 2 Key Support Systems (Table 3.1.1-2 of the IPE submittal) states that a plant trip would not be expected to occur following loss of 4kV shutdown board A. Therefore, a manual reactor trip is conservatively assumed to occur for all fires in this area, at this level of analysis.

The failure of these components was modeled by failing top events AA (4kV shutdown board A), RE (480V RMOV board 1A), DA (250VDC control power for 4kV shutdown board A and 480V shutdown board 1A) and DC (250VDC control power for 4kV shutdown board B and 480V shutdown board 2A) in plant model module ELECT12, in addition to the top events listed above. It was noted during this evaluation that the assumed loss of both shutdown buses is equivalent to a loss of offsite power to the 4kV shutdown boards. This evaluation then generated a conditional core damage frequency of 3.79E-04. Given a fire ignition frequency of 8.57E-03 for this area from Table 4-1, the upper bound conditional core damage frequency for this area can be evaluated as

F2 = 8.57E-03 x 3.79E-04 = **3.24E-06**

Since the upper bound core damage frequency for this evaluation is greater than 1E-06, fires in this area cannot be screened from further consideration at this level of evaluation. This evaluation is conservative in that all fires in this area are assumed to result in a manual reactor trip and cause the loss of all plant equipment located in this fire area, regardless of fire severity or manual fire suppression. Also, this evaluation conservatively assumes loss of power from shutdown buses 1 and 2 to all other 4kV shutdown boards, similar to a loss of offsite power. This area is evaluated in greater detail in Section 6.2.

5.1.6 Fire Area 6 - 480V Shutdown Board Room 1A

Plant walkdowns (see Attachment D) confirm that the following major plant equipment is located in this fire area:

480V Shutdown Board 1A
Panel 1-25-44A-11
Panel 1-25-44B-11

Also, it was confirmed during these walkdowns that no cables traverse this area, other than those associated with 480V shutdown board 1A and the 480V load shed panels.

The Failure Modes and Effects Analysis of Browns Ferry Unit 2 Key Support Systems (Table 3.1.1-2 of the IPE submittal) states that a plant trip would not be expected to occur following loss of 480V shutdown board 1A. At this level of analysis, all fires in this area were therefore conservatively assumed to result in a manual reactor trip, regardless of the actual level of fire severity.

The potential failure of 480V shutdown board 1A due to fire was modeled as a failure of top event RQ in plant model module ELECT12. The potential failure of the 480V load sequencing logic circuits in panels 1-PNL-25-44A-11 and 1-PNL-25-44B-11 was conservatively modeled by failing diesel generators C and D at top events GC and GD. This treatment is conservative in that it fails 4160V switchgear following a loss of offsite power, in addition to the supplied 480V loads.

This evaluation then generated a conditional core damage frequency of 5.22E-07. Given a fire ignition frequency of 6.64E-03 from Table 4-1, the upper bound conditional core damage frequency for this area can be evaluated as

F2 = 6.64E-03 x 5.22E-07 = **3.47E-09**

Since the upper bound core damage frequency for this evaluation is less than 1E-06, fires in this area can be screened from further consideration. This evaluation is conservative in that all fires in this area are assumed to result in a manual reactor trip and cause the loss of all plant equipment located in this fire area, regardless of fire severity or manual fire suppression.

5.1.7 Fire Area 7 - 480V Shutdown Board Room 1B

Plant walkdowns (see Attachment D) confirm that the following major plant equipment is located in this fire area:

480V Shutdown Board 1B
Panel 1-25-44A-12
Panel 1-25-44B-12

These walkdowns also evaluated cable routing through this area. During this review, it was confirmed that control cables for Unit 1 LPCI MG sets 1DN, 1DA, 1EN are located in this area. The potential fire-related failure of these components does not, however, impact the operation of Unit 2.

The Failure Modes and Effects Analysis of Browns Ferry Unit 2 Key Support Systems (Table 3.1.1-2 of the IPE submittal) states that a plant trip would not be expected to occur following loss of 480V shutdown board 1B. Manual reactor trip was therefore conservatively assumed to occur for any and all fires in this area.

The failure of the components in this area was modeled by failing top event RR (480V shutdown board 1B) in plant model module ELECT12. This evaluation then generated a conditional core damage frequency of 5.38E-07. Given a fire ignition frequency for this area of 6.64E-03 from Table 4-1, the upper bound conditional core damage frequency for this area can be evaluated as

F2 = 6.64E-03 x 5.38E-07 = **3.57E-09**

Since the upper bound core damage frequency for this evaluation is less than 1E-06, fires in this area can be screened from further consideration. This evaluation is conservative in that all fires in this area are assumed to result in a manual reactor trip and cause the loss of all plant equipment located in this fire area, regardless of fire severity or manual fire suppression.

5.1.8 Fire Area 8 - 4kV Shutdown Board Room D

Plant walkdowns (see Attachment D) confirm that the following major plant equipment is located in this fire area:

4kV Shutdown Board D
480V RMOV Board 2B
250V RMOV Board 2B
Panel 25-45D
I&C Bus B Equipment
Div I ATU Inverter

Cable routing through this area was also evaluated during these walkdowns to ensure that no other risk-significant components could be impacted by fires in this area.

The Failure Modes and Effects Analysis of Browns Ferry Unit 2 Key Support Systems (Table 3.1.1-2 of the IPE submittal) states that a plant trip would not be expected to occur following loss of 4kV shutdown board D. For this level of analysis, manual reactor trip was therefore conservatively assumed to occur for any and all fires in this area.

The fire-related failure of 4kV shutdown board D could potentially impact the operation of the following additional plant components:

Diesel Generator D	Top Event GD
Shutdown Bus 1	Top Event SHUT1
Shutdown Bus 2	Top Event SHUT2
480V Shutdown Board 2B	Top Event RT
480V RMOV Board 2C	Top Event RJ

The failure of the components that could be impacted by fires in this area was then modeled by failing top events AD (4kV shutdown board D), RI (480V RMOV board 2B), RC (250V RMOV board 2B), DO (120V I&C bus 2B), PX1 (division 1 ATU power supply) and R480 (480V bus recovery) in the plant model, in addition to failing interim variables RIOK (allows recovery of top event RI) and RCOK (allows recovery of top event RC) and the top events noted above. This evaluation then generated a conditional core damage frequency of 7.36E-03. Given a fire ignition frequency of 6.64E-03 from Table 4-1, the upper bound conditional core damage frequency for this area can be evaluated as

F2 = 6.64E-03 x 7.36E-03 = **4.88E-05**

Since the upper bound core damage frequency for this evaluation is more than 1E-06, fires in this area can not be screened from further consideration. This evaluation is conservative in that all fires in this area are assumed to result in plant trip and cause the loss of all plant equipment located in this fire area. In particular, 480VAC RMOV board 2B, which supplies division II RHR suppression pool cooling valves, and 250VDC RMOV board 2B, which supplies division I RHR actuation relays, impact core damage frequency in that the scenarios from this evaluation are dominated by failure of an RPV relief valve to reseat following plant trip, with failure of suppression pool cooling due to fire-induced failures. This area is therefore retained for detailed evaluation in Section 6.2.

5.1.9 Fire Area 9 - 4kV Shutdown Board Room C and 250V Battery Room

Plant walkdowns (see Attachment D) confirm that the following major plant equipment is located in this area:

4kV Shutdown Board C
480V RMOV Board 2A
250V RMOV Board 2A
Unit 2 Panel 25-32 (Remote Shutdown)
Panel 25-45C (4kV Shutdown Board C Relays)
I&C Bus 2A Equipment
250VDC Battery SB-C and D
250VDC Battery chargers SB-C and D
250VDC Distribution Panel SB-C and D
Board Room Emergency Air Conditioner and Dampers
250VDC Battery Exhaust and Supply Fans

Panels 25-42A-1 and B-1 (Common Logic Relays)
Panels 25-42A-2 and B-2 (Common Logic Relays)
ATU Inverters - Division II

Cable routing through this area was also evaluated during these walkdowns to ensure that no other risk-significant components could be impacted by fires in this area.

The Failure Modes and Effects Analysis of Browns Ferry Unit 2 Key Support Systems (Table 3.1.1-2 of the IPE submittal) states that a plant trip would not be expected to occur following loss of 4kV shutdown board C. For this level of analysis, manual reactor trip was therefore conservatively assumed to occur for any and all fires in this area. It should be noted that the potential failure of remote shutdown panel 9-32 in such a way to cause an MSIV closure is subsumed within failure of top event DN (I&C bus 2A). Hot short failure of RPV relief valve control from this panel is not separately considered at this level of analysis.

The fire-related failure of 4kV shutdown board C could potentially impact the operation of the following additional plant components:

Diesel Generator C	Top Event GC
Shutdown Bus 1	Top Event SHUT1
Shutdown Bus 2	Top Event SHUT2
480V Shutdown Board 2A	Top Event RS
480V RMOV Board 2C	Top Event RJ

The failure of the components that could be impacted by fires in this area was then modeled by failing top events AC (4kV shutdown board C), RH (480V RMOV board 2A), RB (250V RMOV board 2A), DB (250VDC control power for shutdown board C and 480V shutdown board 1B), DD (250VDC control power for shutdown board D and 480V shutdown board 2B), DN (120V I&C bus 2A), PX2 and HPI (Division 2 ATU power supply) and R480 (recovery of 480V buses) in the plant model, in addition to failing interim variables RHOK (allows recovery of top event RH) and RBOK (allows recovery of top event RB), in addition to the top events listed above. This evaluation then generated a conditional core damage frequency of 1.08E-02. Given a fire ignition frequency of 8.07E-03 from Table 4-1, the upper bound conditional core damage frequency for this area can be evaluated as

$$F2 = 8.07E\text{-}03 \times 1.08E\text{-}02 = \mathbf{8.71E\text{-}05}$$

Since the upper bound core damage frequency for this evaluation is more than 1E-06, fires in this area can not be screened from further consideration. This evaluation is conservative in that all fires in this area are assumed to result in plant trip and cause the loss of all plant equipment located in this fire area.

In particular, 480VAC RMOV board 2A, which supplies division I RHR suppression pool cooling valves, and 250VDC RMOV board 2A, which supplies division II RHR actuation relays, impact core damage frequency in that the scenarios from this evaluation are dominated by failure of an RPV relief valve to reseat following plant trip, with failure of suppression pool cooling due to fire-induced failures. This area is therefore retained for detailed evaluation in Section 6.2.

5.1.10 Fire Area 10 - 480V Shutdown Board Room 2A

Plant walkdowns (see Attachment D) confirm that the following plant equipment could be affected by a fire in this area:

480V Shutdown Board 2A	Top Event RS
480V Load Sequencing Logic Panel 2-PNL-25-44A-11	See Below
480V Load Sequencing Logic Panel 2-PNL-25-44B-11	See Below

Also, it was confirmed during these walkdowns that no cables traverse this area, other than those associated with 480V shutdown board 2A and the 480V load shed panels.

The potential failure of 480V shutdown board 2A due to fire was modeled as a failure of top event RS in plant model module ELECT12. The potential failure of the 480V load sequencing logic circuits in panels 2-PNL-25-44A-11 and 2-PNL-25-44B-11 was conservatively modeled by failing division I diesel generators A and B at top events GA and GB, in addition to failing shutdown board recovery at top event SDREC. This treatment is conservative in that it fails 4106V switchgear following a loss of offsite power, in addition to the supplied 480V loads.

The Failure Modes and Effects Analysis of Browns Ferry Unit 2 Key Support Systems (Table 3.1.1-2 of the IPE submittal) states that a plant trip may occur due to MSIV closure following loss of 480V shutdown board 2A. Therefore, fires in this area were modeled with RS failure and using the CIV initiating event logic. This quantification resulted in a conditional core damage frequency (P2 value) of 2.52E-06.

Given a fire ignition frequency (F1 value) of 6.64E-03 from Table 4-1, the core damage frequency due to fires in this area can be evaluated as:

F2 = F1 x P2 = 6.64E-03 x 2.52E-06 = 1.67E-08

Since core damage frequency for this evaluation is below 1E-06, fires in this area can be screened from further consideration. Again, this evaluation remains conservative in that all fires are assumed to result in a plant trip due to MSIV closure with failure of 480V Shutdown Board 2A, regardless of fire severity or manual fire suppression.

5.1.11 Fire Area 11 - 480V Shutdown Board Room 2B

Plant walkdowns (see Attachment D) confirm that a significant fire in this area has the potential to impact the operation of the following plant equipment:

480V Shutdown Board 2B	Fails Top Event RT MSIV Closure
LPCI MG Set 2DA	Subsumed in RT
LPCI MG Set 2EN	Subsumed in RT
RBCCW Sectionalizing Valve FCV-70-48	May Impact RBC
HPCI Test Valve FCV-73-35	May impact HPI
480V Load Sequencing Panel 2-PNL-25-44A-12	See Below
480V Load Sequencing Panel 2-PNL-25-44B-12	See Below

Also, it was confirmed during these walkdowns that no cables traverse this area, other than those associated with these components.

The potential failure of 480V shutdown board 2B due to fire was modeled as a failure of top event RT in plant model module ELECT12. As noted above, this subsumes the potential impact of fires in this area on LPCI MG sets 2DA and 2EN by failing the motive power source (480V shutdown board 2B). Also, since the Failure Modes and Effects Analysis of Browns Ferry Unit 2 Key Support Systems (Table 3.1.1-2 of the IPE submittal) states that a plant trip may occur due to MSIV closure following loss of 480V shutdown board 2B, top event IVO (MSIVs remain open) was set to guaranteed failure.

The potential impact on RBCCW of sectionalizing valve FCV-70-48 would be to fail the system following a loss of offsite power. This was conservatively modeled by failing top event RBC for all cases.

The potential impact on HPCI of failing test return valve FCV-73-35 is normally isolated by a second valve (FCV-73-36). Since this minor degradation of the system was not modeled in the system analysis, HPCI was conservatively set to guaranteed failure for all conditions by failing top event HPI.

The potential failure of the 480V load sequencing logic circuits in panels 2-PNL-25-44A-12 and 2-PNL-25-44B-12 was conservatively modeled by failing division I diesel generators C and D at top events GC and GD, in addition to failing shutdown board recovery at top event SDREC. This treatment is conservative in that it fails 4106V switchgear following a loss of offsite power, in addition to the supplied 480V loads.

This quantification resulted in a conditional core damage frequency of 5.51E-06. Given a fire ignition frequency of 6.64E-03 from Table 4-1, the core damage frequency due to fires in this area can be evaluated as:

$$F2 = 6.64E\text{-}03 \times 5.51E\text{-}06 = \mathbf{3.66E\text{-}08}$$

Since core damage frequency for this evaluation is below 1E-06, fires in this area can be screened from further consideration. This evaluation remains conservative in that all fires are assumed to result in a plant trip due to MSIV closure with failure of 480V Shutdown Board 2B, RBCCW and HPCI, regardless of fire severity or manual fire suppression.

5.1.12 Fire Area 12 - Shutdown Board Room F

Physical area walkdowns (see Attachment D) confirm that the following major plant equipment is located in this fire area:

250V RMOV Board 3B
480V HVAC Board B
480V RMOV Board 3B
I&C Bus 3B Equipment
ATU Inverters Division I (Unit 3)
Panel 25-654B

These walkdowns included examination of cable routing and raceway locations. During this examination, it was revealed that the cables supplying 250VDC control power for 4kV shutdown boards 3EA and 3EC are routed through this area. No other Unit 2 related plant components were identified.

While a plant trip would not be expected due to fires in this area, manual reactor trip of Unit 2 has been conservatively assumed for this analysis. The Level 1 plant model was therefore quantified by failing support for top events A3EA and A3EC from top events DE and DG, respectively. This quantification then generated a conditional core damage frequency of 4.17E-07. This value, multiplied by the fire ignition frequency for this area shown in Table 4-1, gives an upper bound core damage frequency of:

$$F2 = 4.17E\text{-}07 \times 7.25E\text{-}03 = \mathbf{3.02E\text{-}09}$$

Therefore, fire hazards within this area can be screened from further consideration, based on an upper bound core damage frequency of less than 1E-06. This evaluation is conservative in that all fires in this area are assumed to result in a manual reactor trip and cause the loss of all plant equipment located in this fire area, regardless of fire severity or manual fire suppression.

5.1.13 Fire Area 13 - Shutdown Board Room E

Plant walkdowns (see Attachment D) confirm that the following major plant equipment is located in this fire area:

250V RMOV Board 3A
480V RMOV Board 3A
Unit 3 Panel 25-32
I&C Bus 3A Equipment
ATU Inverters Division II (Unit 3)

These walkdowns included a review of cables and raceway routing through this area. This review confirmed that control cables associated with the following equipment could be impacted due to fires in this area:

480V RMOV Board 3B	Not modeled
480V Diesel Generator Aux Board 3EB	Top Event RP
480V Shutdown Board 3A	Top Event RX
480V Shutdown Board 3B	Top Event RY

While a plant trip would not be expected due to fires in this area, manual reactor trip of Unit 2 has been conservatively assumed to occur in response to any and all fires in this area for this level of analysis. This was modeled by failing the top events listed above. This evaluation generated a conditional core damage frequency of 7.51E-07. This value, multiplied by the fire ignition frequency for this area shown in Table 4-1, gives an upper bound core damage frequency of:

F2 = 7.51E-07 x 7.15E-03 = **5.37E-09**

Therefore, fire hazards within this area can be screened from further consideration, based on an upper bound core damage frequency of less than 1E-06. This evaluation is conservative in that all fires in this area are assumed to result in a manual reactor trip and cause the loss of all plant equipment located in this fire area, regardless of fire severity or manual fire suppression.

5.1.14 Fire Area 14 - 480V Shutdown Board Room 3A

Plant walkdowns (see Attachment D) confirm that the following major plant equipment is located in this fire area:

480V Shutdown Board 3A

These walkdowns confirm that there is no additional Unit 2 related equipment located in this area. Also, it was confirmed during these walkdowns that no additional Unit 2 related support cables traverse through this area.

The Failure Modes and Effects Analysis of Browns Ferry Unit 2 Key Support Systems (Table 3.1.1-2 of the IPE submittal) states that a plant trip would not be expected to occur following loss of 480V shutdown board 3A.

For this level of analysis, a manual reactor trip was therefore conservatively assumed to occur in response to all fires in this area. The assumed failure of this component is modeled by failing top event RX (480V shutdown board 3A) in plant model module ELECT12. This evaluation then generated a conditional core damage frequency of 4.21E-07. Given a fire ignition frequency of 6.64E-03 from Table 4-1, the upper bound conditional core damage frequency for this area can be evaluated as

F2 = 6.64E-03 x 4.21E-07 = **2.79E-09**

Since the upper bound core damage frequency for this evaluation is less than 1E-06, fires in this area can be screened from further consideration. This evaluation is conservative in that all fires in this area are assumed to result in a manual reactor trip and cause the loss of all plant equipment located in this fire area, regardless of fire severity or manual fire suppression.

5.1.15 Fire Area 15 - 480V Shutdown Board Room 3B

Plant walkdowns (see Attachment D) confirm that the following major plant equipment is located in this fire area:

480V Shutdown Board 3B

These walkdowns confirm that there is no additional Unit 2 related equipment located in this area. Also, it was confirmed during these walkdowns that no additional Unit 2 related support cables traverse through this area.

The Failure Modes and Effects Analysis of Browns Ferry Unit 2 Key Support Systems (Table 3.1.1-2 of the IPE submittal) states that a plant trip would not be expected to occur following loss of 480V shutdown board 3B. For this level of analysis, manual reactor trip was therefore conservatively assumed to occur for any and all fires that occur in this area. The assumed failure of 480V shutdown board 3B component is modeled in the IPE plant model by failing top event RY. This evaluation then generated a conditional core damage frequency of 4.35E-07. Given a fire ignition frequency of 6.64E-03 from Table 4-1, the upper bound conditional core damage frequency for this area can be evaluated as

F2 = 6.64E-03 x 4.35E-07 = 2.89E-09

Since the upper bound core damage frequency for this evaluation is less than 1E-06, fires in this area can be screened from further consideration. This evaluation is conservative in that all fires in this area are assumed to result in a manual reactor trip and cause the loss of all plant equipment located in this fire area, regardless of fire severity or manual fire suppression.

5.1.16 Fire Area 16 - Control Building

The Control Building consists of three compartments, which are analyzed as a single fire area in volume 1 of the Browns Ferry Fire Protection Report. Due to the potential for loss of all plant control functions, requiring possible evacuation of the Control Room itself, this area is evaluated with an assumed conditional core damage frequency of 1.0 for this level of evaluation. The individual compartments within this fire area will be analyzed in more detail in Section 6.2.

The resulting upper bound core damage frequencies for each of these compartments are shown, for completeness, in Table 5-2, below.

5.1.17 Fire Area 17 - Unit 1 Battery and Battery Board Room

Fire area 17 consists of two rooms, the Unit 1 battery room and the Unit 1 battery board room, which are separated by a concrete block wall with an equivalent fire resistance rating of 1.5 hours. A few conduit penetrations exist in this wall. Area wide smoke detection is installed throughout both of these rooms and both areas are protected with manually actuated sprinkler systems. Review of the EPRI Fire Events Database (NSAC/178L) and the construction of the unit battery itself shows battery fires to be of little consequence, beyond potential damage to the battery itself. The battery cells are filled with acid, which provides an instant fire suppressing medium. Heat release intensities, or fire size, will remain small enough so as not to damage the few electrical conduits that traverse the area. Therefore, an engulfing fire will be assumed in the battery board room, damaging all components in the room, in addition to failing the 250VDC batteries in the adjacent battery room.

Plant walkdowns (see Attachment D) confirm that the following major plant equipment could be impacted by a fire in this area:

250VDC Battery 1 (located in the battery room)
Battery Board 1
250VDC Battery Charger 1
Unit Preferred MMG Set 1 and Associated Equipment

24V Neutron Monitoring Batteries and Chargers
48V Annunciator Battery and Charger A
RPS MG Set B
Unit 1 RPS Circuit Protectors
I&C Buses A and B Fused Disconnect Switches
Unit 1 Panel 9-81 (Division 1 only) (FW Inverters)

The Failure Modes and Effects Analysis of Browns Ferry Unit 2 Key Support Systems (Table 3.1.1-2 of the IPE submittal) states that a plant trip would not be expected to occur following loss of the Unit 1 battery board. A manual reactor trip of Unit 2 was therefore conservatively assumed to occur for all fires in this area. It should be noted that the loss of the RPS circuit protectors will, however, result in a reactor trip of Unit 1.

The assumed failure of the components in this fire area is modeled by failing top event DE (Unit 1 battery) in the plant model. This evaluation then generated a conditional core damage frequency of 4.31E-07. Given a fire ignition frequency of 2.19E-02 for this area from Table 4-1, the upper bound conditional core damage frequency for this area can be evaluated as

F2 = 2.19E-02 x 4.31E-07 = **9.44E-09**

Since the upper bound core damage frequency for this evaluation is less than 1E-06, fires in this area can be screened from further consideration.

This evaluation is conservative in that all fires in this area are assumed to result in a manual reactor trip and cause the loss of all plant equipment located in this fire area, regardless of fire severity or manual fire suppression.

5.1.18 Fire Area 18 - Unit 2 Battery and Battery Board Room

Fire area 18 consists of two rooms, the Unit 2 battery room and the Unit 2 battery board room, which are separated by a concrete block wall with an equivalent fire resistance rating of 1.5 hours. A few conduit penetrations exist in this wall. Area wide smoke detection is installed throughout both of these rooms and both areas are protected with manually actuated sprinkler systems. Review of the EPRI Fire Events Database (NSAC/178L) and the construction of the unit battery itself shows battery fires to be of little consequence, beyond potential damage to the battery itself. The battery cells are filled with acid, which provides an instant fire suppressing medium. Heat release intensities, or fire size, will remain small enough so as not to damage the few electrical conduits that traverse the area. Therefore, an engulfing fire will be assumed in the battery board room, damaging all components in the room, in addition to failing the 250VDC batteries in the adjacent battery room.

Plant walkdowns (see Attachment D) confirm that the following risk significant plant equipment is either located in this area or has cables that transit through this area:

250VDC Battery 2 (located in the battery room)	Failed DH, CPREC
Battery Board 2	Subsumed in DH
250VDC Battery Charger 2A and 2B	Subsumed in DH
Unit Preferred MMG Set 2 and Associated Equipment	Subsumed in DH
4kV Shutdown bus 3ED control power	Failed A3ED
Division 2 instrument power (Panels 9-82, 9-88)	Failed PX2
RCIC steam flow indication	Failed RCI
HPCI steam flow indication	Failed HPI
CS/RHR interlock logic II	Subsumed in PX2
RPS circuit protectors	See Below
Manual relief valves 2-PCV-1-4, -18, 23, 41, 42	Failed interim variables PWR6 and PWRALL

The Failure Modes and Effects Analysis of Browns Ferry Unit 2 Key Support Systems (Table 3.1.1-2 of the IPE submittal) states that a plant trip would not be expected to occur following loss of the Unit 2 battery board. A manual reactor trip was therefore conservatively assumed to occur for all fires in this area. It should be noted that a reactor trip will occur on de-energization of RPS circuit protectors. Use of the manual reactor trip (required) logic is conservative in that reactor trip failure (i.e. ATWS) is considered.

The assumed failure of the components located in this fire area is modeled by failing the indicated top events (DH, CPREC, A3ED, PX2, RCI, HPI) and interim variables (PWR6 and PWRALL) in the plant model. It was noted during this review that the potential failure of the RPS circuit protectors would not prevent manual reactor trip. Therefore, this impact is not separately modeled. This evaluation then generated a conditional core damage frequency of 5.79E-05. Given a fire ignition frequency of 2.09E-02 for this area from Table 4-1, the upper bound conditional core damage frequency for this area can be evaluated as

F2 = 2.09E-02 x 5.79E-05 = **1.21E-06**

Since the upper bound core damage frequency for this evaluation is greater than 1E-06, fires in this area can not be screened from further consideration.

This evaluation is conservative in that all fires in this area are assumed to result in a manual reactor trip and cause the loss of all plant equipment located in this fire area, including control cables that transit through the area, regardless of fire severity or manual fire suppression. Also, it should be noted that the fire ignition frequency for this area includes allowance for 25% of the total allowance for all electrical cabinets in a typical reactor building. This is in addition to the ignition frequency for electric cabinets located in the Unit 2 Reactor Building. This area is analyzed in more detail in Section 6.2.

5.1.19 Fire Area 19 - Unit 3 Battery and Battery Board Room

Fire area 19 consists of two rooms, the Unit 3 battery room and the Unit 3 battery board room, which are separated by a concrete block wall with an equivalent fire resistance rating of 1.5 hours. A few conduit penetrations exist in this wall. Area wide smoke detection is installed throughout both of these rooms and both areas are protected with manually actuated sprinkler systems. Review of the EPRI Fire Events Database (NSAC/178L) and the construction of the unit battery itself shows battery fires to be of little consequence, beyond potential damage to the battery itself. The battery cells are filled with acid, which provides an instant fire suppressing medium. Heat release intensities, or fire size, will remain small enough so as not to damage the few electrical conduits that traverse the area. Therefore, an engulfing fire will be assumed in the battery board room, damaging all components in the room, in addition to failing the 250VDC batteries in the adjacent battery room.

Plant walkdowns (see Attachment D) confirm that the following plant equipment could be impacted by a fire in this area:

250VDC Battery 3 (located in battery room)
Battery Board 3
250VDC Battery Charger 3
Unit Preferred MMG Set 3 and Associated Equipment
24V Neutron Monitoring Batteries and Chargers
48V Annunciator Battery Charger B
Unit 3 RPS Circuit Protectors

The Failure Modes and Effects Analysis of Browns Ferry Unit 2 Key Support Systems (Table 3.1.1-2 of the IPE submittal) states that a plant trip would not be expected to occur following loss of the Unit 3 battery board. A manual reactor trip of Unit 2 was therefore conservatively assumed to occur for all fires in this area. It should be noted that a reactor trip of Unit 3 will occur following de-energization of the RPS circuit protectors.

The assumed failure of the components located in this fire area is modeled by failing top events DG (Unit 3 battery) and CPREC (control power recovery) in the plant model. This evaluation then generated a conditional core damage frequency of 4.94E-07.

Given a fire ignition frequency of 2.09E-02 from Table 4-1, the upper bound conditional core damage frequency for this area can be evaluated as

F2 = 2.09E-02 x 4.94E-07 = **1.03E-08**

Since the upper bound core damage frequency for this evaluation is less than 1E-06, fires in this area can be screened from further consideration.

This evaluation is conservative in that all fires in this area are assumed to result in a manual reactor trip and cause the loss of all plant equipment located in this fire area, regardless of fire severity or manual fire suppression.

5.1.20 Fire Area 20 - Unit 1 and 2 Diesel Generator Building

For a fire in the Unit 1 and 2 Diesel Generator Building, it is unlikely that the operator would initiate plant trip on Unit 2, except in the case of severe, unsuppressed fires. For the purposes of this analysis, however, all fires in this area are conservatively assumed to result in a precautionary trip of Unit 2.

All components listed for this area in volume 1 of the Browns Ferry Fire Protection Report and identified during plant walkdowns are associated with the Unit 1 and 2 diesel generators. This potential impact was modeled by failing the following top events:

DG A	GA FA
DG B	GB FB
DG C	GC FC
DG D	GD FD

In addition, common cause top event DIES1 was set to guaranteed failure. Finally, shutdown bus recovery was failed at top event SDREC. It should be noted that these systems are only required following a consequential loss of offsite power, following an assumed reactor trip for Unit 2 following a fire in the Unit 1 and 2 Diesel Generator Building. The conditional core damage frequency for this evaluation is 2.31E-06, which, when multiplied by the fire ignition frequency for this area of 1.24E-01, gives an upper bound core damage frequency of:

F2 = 1.24E-01 x 2.31E-06 = **2.84E-08**

Since the upper bound core damage frequency for this area is less than 1E-06, fires in this area can be screened from further consideration. This evaluation is conservative in that all fires are assumed to result in a manual reactor trip and cause the loss of all plant equipment located in this fire area, regardless of fire severity or automatic or manual fire suppression.

5.1.21 Fire Area 21 - Unit 3 Diesel Generator Building

For a fire in the Unit 3 Diesel Generator Building, it is unlikely that the operator would initiate plant trip on Unit 2, except in the case of severe, unsuppressed fires. For the purposes of this analysis, however, all fires in this area are conservatively assumed to result in a precautionary trip of Unit 2.

All components listed as being in this area in volume 1 of the Browns Ferry Fire Protection Report and identified during plant walkdowns are associated with the Unit 3 diesel generators with the exception of:

4kV Shutdown board 3EB control batteries, battery board and battery charger (SB-3EB)

These impacts were incorporated into the plant model by failing top events DF and CPREC.

The loss of diesel generator availability was modeled by failing the following top events:

DG 3A	GE	FE
DG 3B	GF	FF
DG 3C	GG	FG
DG 3D	GH	FH

Finally, shutdown bus recovery was failed at top event SDREC. It should be noted that these systems are only required to function following a consequential (independent failure) loss of offsite power, following an assumed reactor trip for Unit 2 in response to a fire in the Unit 3 Diesel Generator Building. The conditional core damage frequency for this evaluation is 5.52E-07, which, when multiplied by the fire ignition frequency for this area of 1.24E-01 (see Table 4-1), gives an upper bound core damage frequency of:

F2 = 1.24E-01 x 5.52E-07 = **6.84E-08**

Since the upper bound core damage frequency for this area is less than 1E-06, fires in this area can be screened from further consideration. This evaluation is conservative in that all fires in this area are assumed to result in a manual reactor trip and cause the loss of all plant equipment located in this fire area, regardless of fire severity or automatic or manual fire suppression.

5.1.22 Fire Area 22 - 4kV Shutdown Board Room 3EA and 3EB

Plant walkdowns (see Attachment D) confirm that the following risk-significant equipment could be damaged by a fire in this area:

4kV Shutdown Board 3EA	A3EA
4kV Shutdown Board 3EB	A3EB
Control cables for RHR service water pump A1	SW1A
Control cables for RHR service water pump A3	EA
Control cables for RHR service water pump C1	SW1C
Control cables for RHR service water pump C3	EC

These walkdowns confirmed that there is no additional Unit 2 related equipment located in this area. Also, it was confirmed during these walkdowns that no additional Unit 2 related support cables traverse through this area.

The Failure Modes and Effects Analysis of Browns Ferry Unit 2 Key Support Systems (Table 3.1.1-2 of the IPE submittal) states that a plant trip would not be expected to occur following loss of 4kV shutdown boards 3EA and 3EB individually. For this level of analysis, fires in this area are therefore conservatively modeled by assuming that the Unit 2 operator will trip the reactor for any and all fires in this area.

All fires in this area are then assumed to fail the top events listed by their associated plant components, above. The conditional core damage frequency for this evaluation is 3.12E-06, which, when multiplied by the fire ignition frequency for this area of 6.67E-03 (see Table 4-1), gives an upper bound core damage frequency of:

F2 = 6.67E-03 x 3.12E-06 = **2.08E-08**

Since the upper bound core damage frequency for this area is less than 1E-06, fires in this area can be screened from further consideration. This evaluation is conservative in that all fires in this area are assumed to result in a plant trip and fail both of the shutdown buses that are located in this area, regardless of fire severity or manual fire suppression.

5.1.23 Fire Area 23 - 4kV Shutdown Board Room 3EC and 3ED

Plant walkdowns (see Attachment D) confirm that the following risk significant plant equipment could be affected by a fire in this area:

4kV Shutdown Board 3EC	A3EC
4kV Shutdown Board 3ED	A3ED
Control power cables for 4kV shutdown board 3EA	A3EA

These walkdowns confirm that there is no additional Unit 2 related equipment located in this area. Also, it was confirmed during these walkdowns that no additional Unit 2 related support cables traverse through this area.

The Failure Modes and Effects Analysis of Browns Ferry Unit 2 Key Support Systems (Table 3.1.1-2 of the IPE submittal) states that a plant trip would not be expected to occur following loss of 4kV shutdown boards 3EC and 3ED individually. For this level of analysis, fires in this area are conservatively modeled by assuming that the Unit 2 operator will trip the reactor for any and all fires in this area.

All fires in this area are assumed to fail the components listed above. These impacts are modeled by failing top events A3EA, A3EC and A3ED in the ELECT3 module of the plant model. The conditional core damage frequency for this evaluation is 4.37E-07, which, when multiplied by the fire ignition frequency for this area of 6.67E-03 (see Table 4-1), gives an upper bound core damage frequency of:

F2 = 6.67E-03 x 4.37E-07 = **2.91E-09**

Since the upper bound core damage frequency for this area is less than 1E-06, fires in this area can be screened from further consideration. This evaluation is conservative in that all fires in this area are assumed to result in a plant trip and fail both of the shutdown buses that are located in this area.

5.1.24 Fire Area 24 - 4kV Bus Tie Board Room

Plant walkdowns (see Attachment D) confirm that the following major plant equipment is located in this fire area:

4kV Bus Tie Board

These walkdowns confirm that there is no additional Unit 2 related equipment located in this area. These walkdowns also confirm that there are no associated Unit 2 related cables traversing this area, though the emergency supply cables for 480V diesel generator auxiliary board 3EA are routed through this area. These cables are not separately modeled, since they provide backup power only to standby equipment (i.e. the Unit 3 diesel generators).

Since the Level 1 PRA model does not take credit for electric plant lineups using the 4kV bus tie board, a fire in this area would not have the potential to impact plant response following reactor trip. Also, it is unlikely that a fire in this area would result in a plant trip of Unit 2. During review of the potential failure modes of this board, it was identified that a conceivable failure of shutdown buses 1 and 2 could occur, similar to a loss of offsite power, though offsite power would remain available to the balance of plant loads. For this level of analysis, all fires in this area are therefore conservatively modeled as a loss of all offsite power (initiating event LOSP). This results in a conditional core damage frequency for this area from Table 5-1 of 4.63E-05. The fire ignition frequency for this area is listed in Table 4-1 as 6.66E-03. This evaluation gives an upper bound core damage frequency of:

F2 = 6.66E-03 x 4.63E-05 = **3.08E-07**

Since the upper bound core damage frequency for this area is less than 1E-06, fires in this area can be screened from further consideration.

This evaluation is conservative in that all fires are assumed to result in a loss of 4kV power from the shutdown buses to all shutdown boards, regardless of fire severity or manual fire suppression.

5.1.25 Fire Area 25 - Turbine Building

The Turbine Building consists of three compartments, which are analyzed as a single fire area in volume 1 of the Browns Ferry Fire Protection Report. Due to the potential for loss of all plant cooling (due to loss of intake, including RHR service water and EECW), in addition to a potential loss of offsite power, this fire area is evaluated with an assumed conditional core damage frequency of 1.0 for this level of evaluation. The compartments will be analyzed in more detail in Section 6.2.

The resulting upper bound core damage frequencies for each of these compartments are shown, for completeness, in Table 5-2, below.

5.2 Upper Bound Core Damage Frequencies (F2)

The core damage frequencies resulting from the evaluations described in Section 5.1, above, are considered to be upper bound values and are summarized in Table 5-2, below. These values were generated by multiplying the fire ignition frequency for the affected area (F1) by the conditional core damage frequency (P2). That is, F2 = F1 x P2.

Fire areas that can be screened from consideration, based on an upper bound core damage frequency (F2 value) of less than 1E-06, are shown as shaded in Table 5-2.

Table 5-2 Upper Bound Core Damage Frequencies (F2)				
Fire Area	Description	Fire Ignition Frequency (F1)	Conditional Core Damage Freq. (P2)	Upper Bound Core Damage Frequency (F2=F1xP2)
1	Unit 1 Reactor Building	9.24E-02	1.0 (Assumed)	9.24E-02
2-1	Unit 2 RB, 519' - 565' (West side)	1.87E-02		1.87E-02
2-2	Unit 2 RB, 519' - 565' (East side)	1.74E-02		1.74E-02
2-3	Unit 2 RB, 593', North	5.33E-03		5.33E-03
2-4	Unit 2 RB, 593' South	2.09E-02		2.09E-02
2-5	Unit 2 RB, 621' and North 639' El.	2.87E-02		2.87E-02
2-6	Unit 2 RB, South 639'	1.96E-02		1.96E-02
3	Unit 3 Reactor Building	9.26E-02		9.26E-02
4	4kV Shutdown Board Room B	6.74E-03	9.17E-04	6.20E-06
5	4kV Shutdown Board Room A and 250V Battery Room	8.57E-03	3.79E-04	3.24E-06
6	480V Shutdown Board Room 1A	6.64E-03	5.22E-07	3.47E-09
7	480V Shutdown Board Room 1B	6.64E-03	5.38E-07	3.57E-09
8	4kV Shutdown Board Room D	6.64E-03	7.36E-03	4.88E-05
9	4kV SD Board Room C, 250V Battery Room	8.07E-03	1.08E-02	8.71E-05
10	480V Shutdown Board Room 2A	6.64E-03	2.52E-06	1.67E-08

Table 5-2
Upper Bound Core Damage Frequencies (F2)

Fire Area	Description	Fire Ignition Frequency (F1)	Conditional Core Damage Freq. (P2)	Upper Bound Core Damage Frequency (F2=F1xP2)
11	480V Shutdown Board Room 2B	6.64E-03	5.51E-06	3.66E-08
12	Shutdown Board Room F	7.25E-03	4.17E-07	3.02E-09
13	Shutdown Board Room E	7.15E-03	7.51E-07	5.37E-09
14	480V Shutdown Board Room 3A	6.64E-03	4.21E-07	2.79E-09
15	480V Shutdown Board Room 3B	6.64E-03	4.35E-07	2.89E-09
16-1	Control Bay - 593' Elevation	5.89E-02	1.0 (Assumed)	5.89E-02
16-2	Cable Spreading Room	1.34E-02		1.34E-02
16-3	Control Room	3.53E-02		3.53E-02
17	Unit 1 Battery and Battery Board Room	2.19E-02	4.31E-07	9.44E-09
18	Unit 2 Battery and Battery Board Room	2.09E-02	5.79E-05	1.21E-06
19	Unit 3 Battery and Battery Board Room	2.09E-02	4.94E-07	1.03E-08
20	Unit 1/2 DG Building	1.24E-01	2.31E-06	2.84E-07
21	Unit 3 DG Building	1.24E-01	5.52E-07	6.84E-08
22	4KV Shutdown Board Room 3EA and 3EB	6.67E-03	3.12E-06	2.08E-08
23	4kV Shutdown Board Room 3EC and 3ED	6.67E-03	4.37E-07	2.91E-09

Table 5-2 Upper Bound Core Damage Frequencies (F2)				
Fire Area	Description	Fire Ignition Frequency (F1)	Conditional Core Damage Freq. (P2)	Upper Bound Core Damage Frequency (F2=F1xP2)
24	4kV Bus Tie Board Room	6.66E-03	4.63E-05	3.08E-07
25-1	Intake Pump Station	3.58E-02	1.0 (Assumed)	3.58E-02
25-2	Pipe Tunnel	9.88E-06		9.88E-06
25-3	Turbine Building	4.50E-01		4.50E-01

The values shown in this table were generated in the following manner:

The fire ignition frequency (F1) was calculated, based on guidance given in the EPRI FIVE documentation. This process is described under Phase II.1 (see Table 4-1).

The conditional core damage frequency for the area (P2), was calculated, as described in the appropriate portions of Section 5.1, above.

The upper bound core damage frequency (F2) was then calculated by multiplying F1 and P2.

5.3 Consideration of Potential Fire-Induced Containment Bypass Scenarios

Although not a specific requirement of the FIVE methodology, the fire PRA procedure guide (Reference 4) states that an area can be screened from consideration only if fire-related core damage frequency is less than 1E-07 or less than 1E-06 with no potential for containment bypass or isolation failure due to fire. This condition potentially applies to the two areas listed in Table 5-2 with fire-related core damage frequencies between 1E-06 and 1E-07 (i.e. fire area 20, Unit 1 and 2 Diesel Generator Building and fire area 24, 4 kV Bus Tie Board Room).

The IPE report identified the following 4 types of containment bypass scenarios (see description of endstates OJA and NJA in Table 4.6-4 of Reference 12):

1. Plant trip due to stuck open relief valve, followed by failure of turbine trip and failure of MSIV isolation.

2. Unisolated break outside containment with failure of HPCI, RCIC, RHR and core spray.

3. Other plant trip events with failure of turbine trip and failure of MSIV isolation, with a concurrent loss of condensate/feedwater and CRD injection.

4. Interfacing system LOCA.

These scenarios were examined to assess the possibility of the path being affected by a fire. The following criteria were used to screen the paths with no fire-induced potential or one that is more likely or severe than that analyzed in the IPE.

No fire susceptible components in the path. This criterion is used to screen the unisolated break outside containment scenario from consideration.

Two or more non-fire susceptible valves in series. This criterion does not apply to the remaining scenarios.

Therefore, the following scenarios remain for potential consideration of containment bypass:

Plant trip due to stuck open relief valve with unisolated steam flow through the main turbine to the main condenser.

Plant trip with unisolated steam flow to the main condenser, with a concurrent loss of condensate/feedwater and CRD injection.

Interfacing system LOCA, such as inadvertent opening of a low pressure/high pressure discharge valve. This form of scenario was separately addressed in the Fire Protection Report (see Section 5.4.3 of Reference 18).

Review of these scenarios for potential applicability to fire areas 20 and 24 shows that the equipment that could be impacted by fires in these areas does not have the potential to impact the frequency of either of these scenarios. Therefore, these areas can continue to be screened from further consideration, as shown in Table 5-2.

6. PHASE II.3 - DETAILED AREA ANALYSIS

This section provides a discussion of the methods used to apply the detailed quantitative screening process guidelines described above to each of the unscreened plant fire areas. Due to the wide variance in fire sources, potentially damaged targets and area geometry, the methods used to evaluate each area will also vary. The methods used to evaluate each of the remaining areas is described below. The results of this screening evaluation are provided in the discussion of Phase III in Section 7.

The areas that were not screened from further consideration in Section 5 (see Table 5-2) are listed, with their associated fire ignition frequencies in Table 6-1, below.

The form of evaluation performed for each of the areas listed in Table 6-1 is based on the type of area and the types of fire ignition sources and fire hazards present in the area.

- The Unit 2 Reactor Building (fire area 2) consists of an extremely large volume, with individual fire zones of about 10,000 ft^2 on each of 5 major elevations, with few major combustibles. Therefore, a detailed review of fire sources and potential hazards was performed for this area. This evaluation is described in Section 6.1, below.

- The Unit 1 and Unit 3 Reactor Buildings (fire areas 1 and 3) are similar to Unit 2, but represent currently non-operational units. Therefore, a comparative review of the analysis of the Unit 2 Reactor Building was performed to ensure that this analysis conservatively bounds the cases for fires in either of the adjoining Reactor Buildings. This evaluation is described in Section 6.1.4, below.

- Fires in 4kV Shutdown Board Rooms A and B (fire areas 5 and 4, respectively) were evaluated by using a case-based approach to identify fire impacts. This was required due to the conservative nature of the initial evaluation, which assumed that all fires result in loss of both 4kV shutdown buses. The evaluation of each of these areas is described in the following sections:

4kV Shutdown Board Room B (Fire Area 4)	Section 6.2.2
4kV Shutdown Board Room A (Fire Area 5)	Section 6.2.3

- Fires in 4kV Shutdown Board Rooms C and D (fire areas 9 and 8, respectively) were also evaluated by using a case-based approach to identify fire impacts. This was required due to the proximity of two particular plant RMOV boards in each area and the relatively small volume involved. The evaluation of each of these areas is described in the following sections:

4kV Shutdown Board Room D (Fire Area 8)	Section 6.2.4
4kV Shutdown Board Room C (Fire Area 9)	Section 6.2.5

- Fires in the Control Building (fire area 16) were evaluated on the basis of individual cases for each of the elevations of this area. An event tree approach was used to develop the individual cases for the evaluation of fires in the Control Room area itself. A detailed fire source and target damage analysis is not practical in this area due to the configuration of equipment, confined space and low ceilings. The significant fire sources in this area are the electrical cabinets and panels themselves. Since these must be considered as fire sources, as well as targets, a source/target geometry cannot be established. The evaluation of each of these areas is described in the following sections:

Control Building - 593' Elevation (Compartment 16-1)	Section 6.2.6
Cable Spreading Rooms (Compartment 16-2)	Section 6.2.8
Control Rooms (Compartment 16-3)	Section 6.2.9

  The special case of a potential fire spread from the 593 foot elevation (Compartment 16-1) to involve the Cable Spreading Rooms (Compartment 16-2) is evaluated in Section 6.2.7.

- The Unit Battery and Battery Board Room area (fire area 18) consists of two distinct rooms, which are separated by a concrete wall. Individual cases for evaluation of this area were developed using the severity factors discussed in Section 6.2.1. This evaluation is described in Section 6.2.10.

- Finally, the Turbine Building (fire area 25, compartment 25-3) consists of a large common area, with little separation between the units, particularly on the Turbine Operating Deck elevation. Due to the level of combustibles and the range of suppression systems and large distances between targets, this area was analyzed using an event tree approach to generate the individual cases used for evaluation. The evaluation of the Turbine Building areas is described in the following sections:

Intake Pump Station (Compartment 25-1)	Section 6.2.11
Pipe Tunnel (Compartment 25-2)	Section 6.2.12
Turbine Building (Compartment 25-3)	Section 6.2.13
Yard Area Fires	Section 6.2.14

For reference, the plant areas that were not removed from further consideration in Section 5 are listed in Table 6-1, below.

Table 6-1 Unscreened Areas	
Area	Description
1	Unit 1 Reactor Building
2-1	Unit 2 Reactor Building, 519' through 565' Elev. (West side of Torus Area and Main Floor)
2-2	Unit 2 Reactor Building, 519' through 565' Elevations (East side of Torus Area and Main Floor)
2-3	Unit 2 Reactor Building, 593' Elevation, North Side
2-4	Unit 2 Reactor Building, 593' Elevation, South Side and RHR Heat Exchanger Rooms
2-5	Unit 2 Reactor Building, 621' Elevation and North Side of 639' Elevations
2-6	Unit 2 Reactor Building, South Side of 639' Elevation
3	Unit 3 Reactor Building
4	4kV Shutdown Board Room B
5	4kV Shutdown Board Room A and 250V Battery Room
8	4kV Shutdown Board Room D
9	4kV Shutdown Board Room C and 250V Battery Room
16-1	Control Bay - 593' Elevation
16-2	Control Bay - 606' Elevation (Cable Spreading Room)
16-3	Control Bay - 617' Elevation (Control Room)
18	Unit 2 Battery and Battery Board Rooms
25-1	Intake Pump Station
25-2	Pipe Tunnel
25-3	Turbine Building

6.1 Unit 2 Reactor Building - Evaluation of Critical Combustible Loading Probability

For the plant areas that have not yet been screened from further consideration, the EPRI FIVE documentation provides guidance for the evaluation of transient (steps 3.4 through 3.8) and fixed (step 3.1) ignition sources.

Transient ignition sources are evaluated by reviewing plant transient combustible control procedures to determine the level of transient combustible loading that may be expected during plant operation. Credit may then be taken for non-exposure of transients due to administrative controls (p), the effective surface area over which the source may be located, relative to the effective surface area of targets in the area and the surface area of the target itself (u) and presence of combustibles in violation of administrative procedures, as indicated by area inspection reports (w). The impact of these potential fire sources is then evaluated for various potential sources (i.e. trash bags, oil cans) and for damage due to plume effects, as well as radiant exposure. The evaluation of transient ignition sources in the Unit 2 Reactor Building is shown in Section 6.1.1, below.

Fixed ignition sources are evaluated by using the heat release rates shown in Attachment A and generating the height above which component damage will not occur due to plume effects and the horizontal distance beyond which component damage will not occur due to radiant effects. These distances are then used to generate a "Zone of Influence" (ZOI) for each fixed ignition source identified in Attachment C. Each of these potential ignition sources was then walked down to identify any plant components within the ZOI. This process is described in Attachment D. The evaluation of fixed ignition sources in the Unit 2 Reactor Building is shown in Section 6.1.2, below.

Significant fire sources within the Unit 2 Reactor Building were identified by reviewing combustible loading calculations (Reference 3), through plant walkdowns and use of the guidance provided in the EPRI Fire Events Database (NSAC/178L). Insignificant fire sources (i.e., those which do not have the potential to cause component damage due to hot gas layer, fire plume or radiant exposure) were screened from further consideration. This process is described in Section C.1 of Attachment C.

Throughout this evaluation, electrical conduit, cable and other components are conservatively assumed to be damaged based on either a non-qualified cable damage threshold temperature of 425 degrees Fahrenheit or a radiant heat exposure of 0.5 BTU/sec/ft^2, whichever is more limiting. This is conservative when compared to the customary damage criteria for qualified cable of 700 degrees Fahrenheit or a radiant exposure of 1.0 BTU/sec/ft^2, as given in the EPRI FIVE documentation.

Non-qualified cables are not separately considered as independent fire sources for this portion of the fire risk analysis for the following reasons:

- At Browns Ferry, these cables are coated with a flame retardant throughout the plant. This suppresses the initial fire development and prevents fire growth between cables. When considering cables as potential fire sources, one must assume that a fire initiating event can occur at any location within the exposed cables (i.e. cable trays) within the plant. The worst case fire scenario could be a fire that ignites at the lowermost cable tray in a stack and propagates to ignite cables in the upper trays. Cable tray fire exposure testing of non-rated, flame retardant coated cables performed by Sandia Laboratories, as described in NUREG/CR-5384 (SAND89-1359) shows that, under relatively severe fire exposure test conditions, it takes approximately 12 minutes to ignite a lower cable tray and, eventually, achieve burn lengths of up to 6 feet. The exposure fire conditions for these tests were indicative of severe fire conditions. That is, no barriers were placed between lower and upper trays during the burning of a diesel fuel exposure fire. The diesel fuel pool was then allowed to burn continuously for 13 minutes. During these tests, 3 of the 5 coatings evaluated prevented the propagation of the fire to the upper tray, even under these severe conditions. Fire spread to the upper tray was observed for the other two coatings, which involved approximately 7 feet of cable tray. Therefore, in all likelihood, the flame retardant coatings in use at Browns Ferry will limit a fire to the initial cable tray. If the fire were to involve a second cable tray, a total length of approximately 13 feet of cable tray could eventually become involved in the fire. Reference 23 (Cable Tray Combustible Loading Calculation) shows that cable tray combustible loading varies from a maximum of 234,000 BTU/ft for control or low to medium level signal cable trays to 117,000 BTU/ft for 480V cable trays. Using the maximum tray loading, the total heat of combustion for a 13 foot section of cable tray is calculated to be approximately 1 million BTU. On a frequency basis, the potential for ignition of a fire in a section of cable of this size is

$$F1 = 1.95E\text{-}03 \times 3/1{,}200 = \mathbf{4.88E\text{-}06}$$

Where

1.95E-03 Represents the total ignition frequency for non-qualified cables in the Unit 2 Reactor Building (see Attachment B).

3/1,200 Represents the ratio of a given cable tray segment heat capacity to the total BTU loading due to unqualified cables for the Unit 2 Reactor Building.

This shows that the cable ignition frequency within any given segment of cable tray location is very low. At this level of frequency, it is judged that this fire ignition source can be neglected (i.e. screened from further evaluation).

- Concerning the fire frequency for non-qualified cable, this frequency is based on 8 fires in the industry. Of these, 3 occurred within 2 months of commercial operation at San Onofre, Unit 1 (2/7, 3/9 and 3/12/68). The descriptions for these fires indicate an "infant mortality" process, where overheating and other aspects of design and construction become evident. Since Browns Ferry has long since passed through this process, these events do not apply.

- The remaining fire events in the EPRI database (entries 231, 282, 301, 398 and 716) indicate minor fires with minimal damage, beyond the failed cable. Of the 3 entries that indicate fire duration, one was self extinguishing (immediate) and the other two were put out with portable extinguishers within 2 and 8 minutes. Of the two remaining entries, one indicated that no power reduction was necessary (breaker trip apparently isolated the fire, allowing it to burn out).

- Finally, if one arbitrarily assumes that all fires in unqualified cables could potentially lead to a total loss of offsite power, the core damage frequency from fires in unqualified cables can be bounded as less than:

  $$F2 = 1.95E\text{-}03 \times 4.63E\text{-}05 = 8.97E\text{-}08$$

  Where

1.95E-03	Represents the total fire frequency for unqualified cables in the Unit 2 Reactor Building.
4.63E-05	Is the conditional core damage frequency, from Table 5-1, for the total loss of offsite power (LOSP) initiating event.

  This evaluation is conservative in that, while it is unlikely that any single fire in an unqualified cable tray could result in a total loss of offsite power, the consequence (i.e. conditional core damage frequency) for this initiating event is judged to bound the potential for damage to other plant components.

Given these considerations, unqualified cables are not considered as separate ignition sources for this portion of the fire risk analysis, though they will continue to be evaluated as potential fire targets. For completeness, though, unqualified cables are listed, with the associated values shown above, in Table 6-3.

6.1.1 Unit 2 Reactor Building - Evaluation of Transient Combustible Sources

Transient combustibles are analyzed for the Unit 2 Reactor Building by considering the building as a single area. These fire ignition sources consist of transients (5.735E-04), cable fires due to welding (4.500E-04) and transient fires due to welding (2.735E-03), with a total fire ignition frequency of 3.76E-03 for each individual fire zone. Since there are 6 fire zones identified in the Unit 2 Reactor Building (fire area 2) and each fire zone is assigned the same likelihood of transient fire ignition, transient fire frequency for the Unit 2 Reactor Building as a whole can be calculated as

$$6 \times 3.76E\text{-}03 = 2.26E\text{-}02$$

Plant reviews have confirmed that effective combustible control procedures are in place and are enforced at the Browns Ferry Nuclear Plant. The probability of storing transient combustibles within a damaging range of plant targets can therefore be determined by using the guidelines provided in the EPRI FIVE documentation. For these analyses, a full 32 gallon trash bag and a 5 gallon oil drum are used to bound the range of transient combustible fire sources that could be expected in the plant. The zone of exposure calculations for these fire sources are shown in Figures 6-1 and 6-2.

These transient combustibles can cause fires that impact plant components, which are considered to be targets, in one of two ways, either by the fire plume itself or through the effects of radiant exposure. Therefore, both of these cases are analyzed below. These calculations are also shown in Figures 6-3 and 6-4.

Plume Effects of Transient Combustibles

The frequency of target damage due to plume effects is determined through a calculation (provided in the EPRI FIVE documentation) that uses the following three factors:

1. The probability of combustibles being exposed (p), which can be assumed to be equal to 0.10, provided that the plant transient combustible control program has storage of flammable and combustible liquids in approved containers, ordinary combustibles or WRP clothing enclosed in metal cabinets or metal containers with fusible link actuated covers (WRP clothing is not stored in the Reactor Buildings at Browns Ferry) and all transient combustibles are removed at the completion of work unless otherwise approved. For this review, then,

   $p = 0.10$

2. Calculation of an area ratio (u) to determine the probability of transient combustibles being located within a "damaging effect" range of a susceptible plant component, cable or other target.

This value is generated from the "footprint area" of the target and the fire source, divided by the total floor area under consideration. For the Unit 2 Reactor Building, the total floor area is listed as 69,277 ft^2.

a. Case 1 - 32 gallon trash bag. Review of the Unit 2 Reactor Building shows that the majority of electrical raceways are located well above the plume damage height of 12 feet (see Figure 6-1).

b. Case 2 - 5 gallon oil can. Review of the Unit 2 Reactor Building shows that the majority of electrical raceways are located well above the plume damage height of 10 feet (see Figure 6-2).

Since the 32 gallon trash bag represents the more restrictive case (i.e. larger range of plume damage), this case is used to evaluate all transient combustibles. The total exposed surface area of cable trays and conduits within the 12 foot damage height over open floor area, where transients could be placed, is estimated to be 1000 ft^2. Assuming that 20% of the Reactor Building floor area is occupied up by plant hardware, the area ratio, u, for transient fire sources can be calculated as

u = (1000) / (69,277 x 0.80)
u = (1000) / (55,422)
u = 0.018

3. Calculation of a probability that the critical amount of combustible will be present between inspections (w). For the Browns Ferry plant, fire hazard inspections are conducted on no less than a weekly basis. Based on EPRI guidance, no less than one noncompliance is conservatively assumed to occur per year, even if none have been recorded. This gives a constant value of w for all Unit 2 Reactor Building areas of

y = 1/52
w = (y/2) x (ln(1/y))
w = (9.615E-03) x (3.951)
w = 3.80E-02

Given these values, the probability of damage to targets (cable trays) in the Unit 2 Reactor Building due to the plume effects of transient fire sources can be calculated as:

Ptc, plume exposure= (Pfst) x (u) x (p) x (w)
Ptc, plume exposure= 1 x (0.018) x (0.10) x (3.80E-02) = 6.84E-05
Probability of target damage due to plume effects of transients
$(Ptc)_{plume}$ = 2.26E-02 x 6.84E-05 = **1.55E-06**

As noted above, this represents the likelihood of damage to any cable trays in the Reactor Building, which are arbitrarily assumed to contain cables of concern. This value will then be added to the likelihood of damage from radiant effects (Ptc), which is calculated below.

Radiant Effects of Transient Combustibles

Target damage due to radiant exposure is calculated in much the same way as the effects of plume damage are evaluated, except that the effective area ratio (u) must be recalculated. Again, the Unit 2 Reactor Building, floor area is 69,277 ft^2. Review of the potential plant targets gives an effective target area of approximately 1,500 ft^2.

a. Case 1 - 32 gallon trash bag. With the transient combustible assumed to consist of a full 32 gallon trash bag, which is expected to bound most other transient fire sources, particularly those that can be left unattended, this gives an effective zone of influence for this area of 5 feet radius (see Figures 6-1 and 6-3).

b. Case 2 - 5 gallon oil can. With the transient combustible assumed to consist of a 5 gallon oil can, which is expected to bound most transient liquid fire sources, particularly those that can be left unattended, this gives an effective zone of influence for this area of 4 feet radius (see Figure 6-2), or 64 square feet total.

Since the 32 gallon trash bag is the more restrictive case (i.e. larger radius of damage area), this case will be used to evaluate all transient combustibles. For this analysis, the effective 5 foot radius surface area is area is rounded up to 100 ft^2.

Assuming that 20% of the Reactor Building floor area is taken up by hardware, the effective area ratio, u, can be calculated for damage due to radiant effects as

$$u = (1{,}500 + 100) / (69{,}277 \times 0.80)$$
$$u = (1{,}600) / (55{,}422)$$
$$u = 0.0289$$

Therefore, $Ptc_{radiant}$ due to radiant effects of transient fire ignition sources can be calculated as:

Ptc,rad exposure = (Pfst) x (u) x (p) x (w)
Ptc,rad exposure= 1 x (0.0289) x (0.10) x (3.80E-02) = 1.098E-04

The probability of target damage due to radiant effects of transients is then:

$Ptc_{radiant}$ = (2.26E-02) x (1.098E-04) = **2.48E-06**

Finally, the total probability of target damage due to transient fire sources can then be calculated as:

$$Ptc_{total} = Ptc_{plume} + Ptc_{radiant}$$
$$Ptc_{total} = 1.55E\text{-}06 + 2.48E\text{-}06$$
$$Ptc_{total} = \mathbf{4.03E\text{-}06}$$

Since this value is greater than 1E-06, these fire sources can not be screened from further consideration at this level of evaluation. Due to the potential consequences of fires of this type, this fire ignition frequency is conservatively included with the fixed fire source ignition frequency for 240V lighting transformer TL2A, as described in Section 6.1.2. This fire source was selected because it has the highest conditional core damage frequency for fixed fire ignition sources in the Unit 2 Reactor Building (see Table 6-3).

Given this conditional core damage (see Table 6-3), the potential core damage due to all transient fire sources in the Unit 2 Reactor Building can be estimated as no higher than:

P2 = 4.03E-06 x 1.43E-03 = **3.76E-08**

6.1.2 Unit 2 Reactor Building - Evaluation of Fixed Ignition Sources

This process consists of a detailed fire growth and propagation analysis and a subsequent assessment of fire damage that could result from fires in the fixed ignition sources identified in Attachment C. For each of the fire zones under consideration, significant fire sources are identified, using the fire source selection guidance provided in references 1 and 3. These fire sources were analyzed in Attachment C and are summarized in Table 6-2, below.

Given these fire ignition sources, fire growth and propagation analyses are then performed based on the FIVE worksheets and heat transfer equations. For each fire source, the critical radial distance and damage height is calculated. All electrical components and raceways within this "zone of influence" (ZOI) are then considered to be damaged by the fire. For each identified fire source, a calculation is also made to determine if there is enough combustible material present to cause damage due to the development of a hot gas layer or due to ceiling jet effects.

The heat release rates and combustible loadings for each of these sources are shown in Attachment A.

Following this evaluation, a determination is made as to whether a plant trip would occur, given the occurrence of a fire. For example, unless other plant equipment becomes involved, it is unlikely that the Unit 2 operator would trip the reactor due to a fire in the primary containment Hydrogen/Oxygen analyzer.

Table 6-2 Unit 2 Reactor Building Fixed Fire Sources	
Fire Zone	Fire Source
2-1	250V RMOV Board 2C
	2-PNLA-25-340 (H2/O2 Analyzer)
	Drywell/Torus Compressor
	Core Spray Pumps 2A and 2C
	RHR Pumps 2A and 2C
2-2	480V RMOV Board 2C
	480V RB Vent Board 2B
	2-PNLA-25-341 (H2/O2 Analyzer)
	Core Spray Pumps 2B and 2D
	RHR Pumps 2B and 2D
2-3	Unit 2 Preferred AC Transformer
2-4	480V RMOV Board 2D
	RBCCW Pump 2A
	RBCCW Pump 2B
	RWCU Room Monitor
	Shutdown Board Room HVAC Compressor Motor

Table 6-2 Unit 2 Reactor Building Fixed Fire Sources	
Fire Zone	Fire Source
2-5	480V RMOV Board 2E
	LPCI MG Sets 2DN and 2EA
	4kV/480V Transformers TS2A and TS2B
	2-LPNL-025-0031 (RCIC Aux Control Panel)
	4kV RPT Board 2-1, Panel 1 and 2
	4kV RPT Board 2-2, Panel 1 and 2
	Panel 25-3 (Filter demin)
	Panel 25-9 (Sample panel)
	240V Lighting Board 2A
	240V Lighting Transformer TL2A (Oil)
	SLC Pumps A and B
2-6	Recirc Pump MG Sets 2A and 2B
	4kV/480V Transformer 2A and 2B
	LPCI MG Sets 2DA and 2EN
	LPNL-25-23 and 24
	240V Lighting Board 2B
	240V Lighting Transformer TL2B

It may be noted that several of the fire ignition sources identified in Attachment B are not listed in Table 6-2, above, as significant fire sources. These plant components include fire protection panels, non-qualified cables, some HVAC components, small pumps and panels containing minor levels of combustibles, as described in Attachment C.

Each of the identified potential fire sources is described below. This discussion includes the results of plant walkdowns for the specific fire source and reviews of the impact on plant operation and the Level 1 PRA plant model of fires within the given component.

250V RMOV Board 2C (Fire Zone 2-1)

Electrical components and raceways within the zone of influence of potential fires in 250V RMOV board 2C, as shown in Attachment C.2, have been evaluated, based on the walkdown information described in Attachment D. While automatic plant trip would not be expected due to fires in this board, manual trip may occur, since this component is modeled in the PRA (top event RD). The failure of 250V RMOV board 2C is incorporated by failing top event RD in the plant model. This evaluation gives a conditional core damage frequency (P2 value) of 4.60E-07. Given a fire ignition frequency for this source of 3.97E-03 (total area ignition frequency for fire zone 2-1 due to electric cabinets in Attachment B is 7.937E-03 and this represents 1 of 2 cabinets in the area that are analyzed as potentially significant fire sources), this results in a core damage frequency due to fires in 250V RMOV board 2C of

F2 = 3.97E-03 x 4.60E-07 = **1.83E-09**

2-PNLA-25-340 (H2/O2 Analyzer) (Fire Zone 2-1)

Electrical components and raceways within the zone of influence of potential fires in panel 2-PNLA-25-340, as shown in Attachment C.2, have been evaluated, based on the walkdown information described in Attachment D. This evaluation confirmed that potential damage to this panel and any components within the zone of influence would not result in an automatic plant trip. Also, these components are not required for the safe shutdown of the plant and are not included in the PRA equipment list. Therefore, fires in 2-PNLA-25-340 can be screened from further consideration.

Drywell/Torus Compressor (Fire Zone 2-1)

Electrical components and raceways within the zone of influence of potential fires in the Drywell/Torus compressor, as shown in Attachment C.2, have been evaluated, based on the walkdown information described in Attachment D. This review confirmed that a fire in the Drywell/Torus compressor could impact conduits that contain cables associated with core spray pump 2C and RCIC turbine control circuitry. Due to this potential impact on the operation of safety related equipment, manual reactor trip is assumed to occur in response to any and all fires in this compressor. The potential impact on core spray pump 2C is incorporated by conservatively failing both trains of core spray in top event CS. The potential loss of RCIC control is incorporated by failing top event RCI within the plant model. This evaluation gives a conditional core damage frequency (P2 value) of 6.56E-06. Given a fire ignition frequency for this source of 5.42E-04 (total area ignition frequency for fire zone 2-1 from Attachment B due to compressors), this results in a core damage frequency due to fires in the drywell/torus compressor of

F2 = 5.42E-04 x 6.56E-06 = **3.55E-09**

Core Spray Pumps 2A and 2C (Fire Zone 2-1)

Electrical components and raceways within the zone of influence of potential fires in core spray pumps 2A and 2C, as shown in Attachment C.3, have been evaluated, based on the walkdown information described in Attachment D. This review confirmed that a fire in the core spray pumps could impact RCIC operation, in addition to the pumps themselves. Due to the potential loss of safety related equipment, manual reactor trip is assumed to occur. The failure of core spray pumps 2A and 2C is modeled by conservatively failing both trains of core spray in top event CS, in addition to failing RCIC top event RCI (see discussion of fire in core spray pumps 2B and 2D, below) in the plant model. This evaluation gives a conditional core damage frequency (P2 value) of 6.56E-06. Given a fire ignition frequency for this source of 2.08E-03 (total area ignition frequency from Attachment B due to pumps is 5.208E-03 and these potential fire sources represent 2 of 5 pumps in the area), this results in a core damage frequency due to fires in core spray pumps 2A and 2C of

F2 = 2.08E-03 x 6.56E-06 = **1.36E-08**

RHR Pumps 2A and 2C (Fire Zone 2-1)

Electrical components and raceways within the zone of influence of potential fires in RHR pumps 2A and 2C, as shown in Attachment C.3, have been evaluated, based on the walkdown information described in Attachment D. Due to the potential loss of safety related equipment, manual reactor trip is assumed to occur. The failure of RHR pumps 2A and 2C is incorporated by failing top events RPA and RBC in the plant model. This evaluation gives a conditional core damage frequency (P2 value) of 2.26E-06. Given a fire ignition frequency for this source of 2.08E-03 (total area ignition frequency from Attachment B due to pumps is 5.208E-03 and these potential fire sources represent 2 of 5 pumps in the area), this results in a core damage frequency due to fires in RHR pumps 2A and 2C of

F2 = 2.08E-03 x 2.26E-06 = **4.70E-09**

480V RMOV Board 2C (Fire Zone 2-2)

Electrical components and raceways within the zone of influence of potential fires in 480V RMOV board 2C, as shown in Attachment C.2, have been evaluated, based on the walkdown information described in Attachment D. Due to the potential loss of control power to a number of flow control valves, manual reactor trip is assumed to occur in response to any and all fires in this component. The failure of 480V RMOV board 2C is incorporated by failing top event RJ in the plant model. This evaluation gives a conditional core damage frequency (P2 value) of 4.77E-07.

Given a fire ignition frequency for this source of 2.91E-03 (total area ignition frequency for fire zone 2-2 due to cabinets and panels from Attachment B is 8.730E-03 and this represents 1 of 3 cabinets in the area that were identified as potentially significant fire sources), this results in a core damage frequency due to fires in 480V RMOV board 2C of

F2 = 2.91E-03 x 4.77E-07 = **1.39E-09**

480V RB Vent Board 2B (Fire Zone 2-2)

Electrical components and raceways within the zone of influence of potential fires in 480V Reactor Building vent board 2B, as shown in Attachment C.2, have been evaluated, based on the walkdown information described in Attachment D. This evaluation confirmed that the zone of influence of a fire in this panel has the potential to impact the following system functions:

480V RB Vent Board 2B	RBI
CAD pressure relief	CAD
CAD/Drywell control air crosstie	DCA
FCV-74-53 (RHR I LPCI inject/	LPC
shutdown cooling return)	SP (division I)

The associated top event for each of these system functions is shown above. In the case of FCV-74-53, the loss of control to this valve would only degrade (as opposed to fail) the LPC and SP functions. For this evaluation, top event LPC was conservatively set to guaranteed failure. Also, Reactor Building isolation has no impact on scenario outcome, besides assignment of a contained endstate following core damage. That is, failure of RB isolation will not impact the frequency of core damage. The conditional core damage frequency (P2 value) from this evaluation is 4.90E-06. Given a fire ignition frequency for this source of 2.91E-03 (total area ignition frequency for fire zone 2-2 due to cabinets and panels from Attachment B is 8.730E-03 and this represents 1 of 3 cabinets in the area that were identified as potentially significant fire sources), this results in a core damage frequency due to fires in 480V RB vent board 2B of

F2 = 2.91E-03 x 4.90E-06 = **1.43E-08**

2-PNLA-25-341 (H2/O2 Analyzer) (Fire Zone 2-2)

Electrical components and raceways within the potential zone of influence of a fire in 2-PNLA-25-341, as shown in Attachment C.2, have been evaluated, based on the walkdown information described in Attachment D. Since a fire in this panel would not result in a plant trip and no components are influenced that are required for safe shutdown of the plant or are contained in the PRA equipment list, fires in 2-PNLA-25-341 can be screened from further consideration.

Core Spray Pumps 2B and 2D (Fire Zone 2-2)

Electrical components and raceways within the zone of influence of potential fires in core spray pumps 2B and 2D, as shown in Attachment C.3, have been evaluated, based on the walkdown information described in Attachment D. Due to the potential loss of safety related equipment, manual reactor trip is assumed to occur. The failure of core spray pumps 2B and 2D is incorporated by conservatively failing both trains of core spray in top event CS and the potential loss of RCIC control is modeled by failing top event RCI in the plant model. This evaluation gives a conditional core damage frequency (P2 value) of 6.56E-06. Given a fire ignition frequency for this source of 2.08E-03 (total area ignition frequency from Attachment B due to pumps is 4.167E-03 and these potential fire sources represent 2 of 4 pumps in the area), this results in a core damage frequency due to fires in core spray pumps 2B and 2D of

F2 = 2.08E-03 x 6.56E-06 = **1.36E-08**

RHR Pumps 2B and 2D (Fire Zone 2-2)

Electrical components and raceways within the zone of influence of potential fires in RHR pumps 2B and 2D, as shown in Attachment C.3, have been evaluated, based on the walkdown information described in Attachment D. This review showed that fires in the RHR pumps could potentially damage HPCI control circuitry, in addition to the pumps themselves. Due to the potential loss of safety related equipment, manual reactor trip is assumed to occur for any fires in the RHR pumps. The failure of RHR pumps 2B and 2D is modeled by failing top events RPB and RPD. The potential impact on HPCI operation is incorporated by failing HPCI top event HPI in the plant model. This evaluation gives a conditional core damage frequency (P2 value) of 3.26E-06. Given a fire ignition frequency for this source of 2.08E-03 (total area ignition frequency from Attachment B due to pumps is 4.167E-03 and these potential fire sources represent 2 of 4 pumps in the area), this results in a core damage frequency due to fires in RHR pumps 2B and 2D of

F2 = 2.08E-03 x 3.26E-06 = **6.78E-09**

Unit 2 Preferred AC Transformer (Fire Zone 2-3)

Electrical components and raceways within the zone of influence of potential fires in the Unit 2 preferred AC transformer, as shown in Attachment C.2, have been evaluated, based on the walkdown information described in Attachment D. This evaluation confirmed that a fire in this transformer has the potential to impact the following system functions:

Unit 2 preferred AC transformer	See below
DG B breaker to 4kV shutdown board B	GB
DG C breaker to 4kV shutdown board C	GC

DG D breaker to 4kV shutdown board D	GD
4kV shutdown board 3ED control power	A3ED
Battery charger 2A	DH
RPV water level B	LT2
RPV pressure B	NH2
Drywell pressure II	DW (train 2)
Main steamline flow instrumentation	IVO
RBCCW sectionalizing valve 2-FCV-70-47	RBC
HPCI 120VAC power	HPI
FSV-84-49 (Drywell N2 line from tank B)	CAD
Aux controls - RHR pump D	RPD
Core spray discharge valves	CS
SDC supply isolation valve FCV-74-47	SDC
RPS scram pilot valves	See Below
RPS backup scram valves	See Below

The associated top event for each of these system functions is shown above. In the case of battery charger 2A, this impact would be similar to loss of support from 480V shutdown board 2A (top event RS). This review also identified the following cables related to analog trip unit (ATU) operation at panels 9-81 and 9-82:

Cable 2A5170 is contained within conduit 2ES1040-I. This cable supplies trouble annunciator indication for the division I ECCS ATU inverter. While failure of this cable would reduce the level of indication available, ATU system operation would not be affected.

Cables 2A5171 and 2PC4964-II are associated with the division II ECCS ATU inverter and are contained within conduit 2ES3421-II, which is located 1.5 to 2.5 feet outside the zone of influence of potential fires in this transformer.

Otherwise, the indicated top events were conservatively set to guaranteed failure. The transformer itself is used to provide an alternate source of preferred AC power and is not modeled in the PRA. Also, due to the timing (i.e. immediate action) and the diverse nature of failures required (i.e. hot shorts to all scram pilot valves and open cables for both sets of backup scram valves) to fail the RPS function, this impact is not separately modeled. The conditional core damage frequency (P2 value) from this evaluation is 3.17E-05. Since this is the only significant potential fire source that was identified for fire zone 2-3, all of the fire ignition frequency for this area, or 5.04E-04 from Attachment B, is conservatively assigned to this source. This results in a core damage frequency due to fires in Unit 2 Preferred AC Transformer of:

F2 = 5.04E-04 x 3.17E-05 = **1.60E-08**

480V RMOV Board 2D (Fire Zone 2-4)

Electrical components and raceways within the zone of influence of potential fires in 480V RMOV board 2D, as shown in Attachment C.2, have been evaluated, based on the walkdown information described in Attachment D. Due to the potential loss of control power to a number of flow control valves, manual reactor trip is assumed to occur. The failure of 480V RMOV board 2D is incorporated by failing top event RK in the plant model. This evaluation gives a conditional core damage frequency (P2 value) of 4.35E-07. Given a fire ignition frequency for this source of 5.16E-03 (total area ignition frequency for fire zone 2-4 for panels and cabinets from Attachment B is 1.032E-02 and this represents 1 of 2 cabinets in the area that were identified as potentially significant fire sources), this results in a core damage frequency due to fires in 480V RMOV board 2D of

F2 = 5.16E-03 x 4.35E-07 = **2.24E-09**

RBCCW Pump 2A (Fire Zone 2-4)

Electrical components and raceways within the zone of influence of potential fires in RBCCW pump 2A, as shown in Attachment C.2, have been evaluated, based on the walkdown information described in Attachment D. Since the success criteria for the RBCCW system is modeled as two of two trains of cooling available following plant trip (one of two following a loss of offsite power), this is modeled as a plant trip with failure of top event RBC. This evaluation gives a conditional core damage frequency of 2.13E-06. Given a fire ignition frequency for this source of 2.08E-03 (total area ignition frequency for fire zone 2-4 due to pumps from Attachment B is 4.167E-03 and this represents 1 of 2 pumps in the area that have been identified as potentially significant fire sources), this results in a core damage frequency due to fires in RBCCW pump 2A of

F2 = 2.08E-03 x 2.13E-06 = **4.44E-09**

RBCCW Pump 2B (Fire Zone 2-4)

Electrical components and raceways within the zone of influence of potential fires in RBCCW pump 2B, as shown in Attachment C.2, have been evaluated, based on the walkdown information described in Attachment D. This review identified HPCI control circuits that are located in conduit 2ES2884-II, which is located slightly outside the zone of influence. To ensure conservative treatment, HPCI was therefore failed for this evaluation. Also, since the success criteria for RBCCW within the plant model is modeled as two of two trains required following plant trip (one of two following a loss of offsite power), this is modeled as a plant trip with failure of top events RBC and HPI. This evaluation gives a conditional core damage frequency of 4.94E-06.

Given a fire ignition frequency for this source of 2.08E-03 (total area ignition frequency for fire zone 2-4 due to pumps from Attachment B 4.167E-03 and this represents 1 of 2 pumps in the area that have been identified as potentially significant fire sources), this results in a core damage frequency due to fires in RBCCW pump 2B of

F2 = 2.08E-03 x 4.94E-06 = **1.03E-08**

RWCU Room Monitor (Fire Zone 2-4)

Electrical components and raceways within the zone of influence of a fire in the RWCU room monitor, as shown in Attachment C.2, have been evaluated, based on the walkdown information described in Attachment D. Since damage to the monitor and any associated components within the zone of influence would not be expected to result in a plant trip and these components are not required for safe shutdown of the plant and are not included in the PRA equipment list, fires in the RWCU room monitor can be screened from further consideration.

Shutdown Board Room HVAC Compressor Motor (Fire Zone 2-4)

Electrical components and raceways within the zone of influence from a fire in the shutdown board room HVAC compressor motor, as shown in Attachment C.2, have been evaluated, based on the walkdown information described in Attachment D. This evaluation confirmed that a significant fire in this component has the potential to damage to the motor and cables associated with the following components modeled in the PRA:

4kV Shutdown Board D control power	AC
480V RMOV Board 2B	RI
RHR Service Water Pump D1 auto start	SW1D
RHR Service Water Pump D3 control	ED
FCV-23-52 (RHR heat exchanger D outlet)	HXD
FCV-70-47 (RBCCW drywell isolation)	RBC
Core Spray Pump 2D control	CS
Core Spray NE corner room cooler fan control	CS

The impacted top events are shown for each of the potentially failed components. This fire source was then evaluated by failing all of the top events (system functions) listed above. This evaluation generated a conditional core damage frequency of 2.42E-04. Given a fire ignition frequency for this source of 1.085E-03 (total area ignition frequency for fire zone 2-4 due to compressors from Attachment B), this results in a core damage frequency due to fires in the shutdown board room HVAC compressor motor of

F2 = 2.42E-04 x 1.085E-03 = **2.63E-07**

480V RMOV Board 2E (Fire Zone 2-5)

Electrical components and raceways within the zone of influence from a fire in 480V RMOV board 2E, as shown in Attachment C.2, have been evaluated, based on the walkdown information described in Attachment D. Due to the potential loss of control power to a number of flow control valves, manual reactor trip is assumed to occur. The failure of 480V RMOV board 2E is incorporated by failing top event RL in the plant model. This evaluation gives a conditional core damage frequency (P2 value) of 4.18E-07. Given a fire ignition frequency for this source of 2.27E-03 (total area ignition frequency for fire zone 2-5 due to electric cabinets from Attachment B is 1.587E-02 and this represents 1 of 7 cabinets and panels in the area that were judged to be significant fire sources), this results in a core damage frequency due to fires in 480V RMOV board 2E of

F2 = 2.27E-03 x 4.18E-07 = **9.49E-10**

LPCI MG Sets 2DN and 2EA (Fire Zone 2-5)

Electrical components and raceways within the zone of influence of a fire in LPCI MG set 2DN, as shown in Attachment C.2, have been evaluated, based on the walkdown information described in Attachment D. Damage to the MG set and any components within the zone of influence would not be expected to result in plant trip, though the associated LPCI bus (480V RMOV board 2D) would shift to the alternate power source. Since this board is modeled in the PRA (top event RK), manual reactor trip is assumed to occur for any fires in this MG set. This evaluation is similar to that shown for 480V RMOV board 2D, which is described above. Given a fire ignition frequency for this source of 5.33E-04 (total fire frequency for fire zone 2-5 due to MG sets is 1.065E-03 and this represents 1 of 2 MG sets in this area), the core damage frequency due to fires in LPCI MG set 2DN can be calculated as

F2 = 5.33E-04 x 4.35E-07 = **2.32E-10**

Electrical components and raceways within the zone of influence of potential fires in LPCI MG set 2EA, as shown in Attachment C.2, have also been evaluated, based on the walkdown information described in Attachment D. This evaluation confirmed that, while it is possible for a fire in LPCI MG Set 2EA to impact LPCI MG Set 2DN circuitry, this would only require 480V RMOV board 2D to shift to its alternate power source. Since this board is modeled in the PRA (top event RK), manual reactor trip is assumed to occur for any fires in this MG set. The simultaneous failure of both of these power sources is subsumed in the failure of 480V RMOV bus 2A (top event RS), which is described above. Given a fire ignition frequency for this source of 5.33E-04 (total fire frequency for fire zone 2-5 due to MG sets is 1.065E-03 and this represents 1 of 2 MG sets in this area), the core damage frequency due to fires in LPCI MG set 2EA can be calculated as

F2 = 5.33E-04 x 2.39E-06 = **1.27E-09**

4kV/480V Transformers TS2A and TS2B (Fire Zone 2-5)

Electrical components and raceways within the zone of influence of potential fires in 4kV/480V transformer TS2A, as shown in Attachment C.2, have been evaluated, based on the walkdown information described in Attachment D. This evaluation confirmed that a fire in this component has the potential to impact 480V shutdown board 2A and HPCI control circuits. The Failure Modes and Effects Analysis from the IPE report shows that this could result in an MSIV closure initiating event. Therefore, fires in this transformer were modeled with the CIV initiating event logic and failing top events RS and HPI. This evaluation generated a conditional core damage frequency of 5.27E-06. The fire ignition frequency for this component (see Attachment B) is 5.04E-04 (1.513E-03 total ignition frequency due to transformers in fire zone 2-5 and this represents one of three transformers in this area). This information was used to generate a core damage frequency due to fires in transformer TS2A of

F2 = 5.27E-06 x 5.04E-04 = **2.66E-09**

Electrical components and raceways within the zone of influence of potential fires in transformer TS2B, as shown in Attachment C.2, have also been evaluated, based on the walkdown information described in Attachment D. This evaluation confirmed that a fire in this component has the potential to impact 480V shutdown board 2B and HPCI control circuits in addition to control circuits of LPCI MG sets 2DN and 2EA. The Failure Modes and Effects Analysis from the IPE report shows that this could result in an MSIV closure initiating event. Therefore, fires in this transformer were modeled with the CIV initiating event logic and failing top events RT and HPI. The LPCI MG sets were conservatively modeled by failing 480V shutdown board 2A at top event RS. This evaluation generated a conditional core damage frequency of 7.08E-06. The fire ignition frequency for this component (see Attachment B) is 5.04E-04 (1.513E-03 total ignition frequency due to transformers in fire zone 2-5 and this represents one of three transformers in this area). This information was used to generate a core damage frequency due to fires in transformer TS2B of

F2 = 7.08E-06 x 5.04E-04 = **3.57E-09**

2-LPNL-025-0031 (RCIC Aux Control Panel) (Fire Zone 2-5)

Electrical components and raceways within the zone of influence of potential fires in the RCIC aux control panel, as shown in Attachment C.2, have been evaluated, based on the walkdown information described in Attachment D. This evaluation confirmed that the zone of influence of a fire in this panel has the potential to impact the following system functions:

RCIC system control	RCI
HPCI system control	HPI

FCV-74-67 (RHR II LPCI inject/ shutdown cooling return)	LPC SP (division II)

The associated top event for each of these system functions is shown above. In the case of FCV-74-67, the loss of control to this valve would only degrade (as opposed to fail) the low pressure injection and suppression pool cooling (top events LPC and SP) function of one train of the RHR system. For this evaluation, top event LPC was conservatively set to guaranteed failure. The conditional core damage frequency (P2 value) from this evaluation is 1.27E-04. Given a fire ignition frequency for this source of 2.27E-03 (total area ignition frequency for fire zone 2-5 due to electric cabinets from Attachment B is 1.587E-02 and this represents 1 of 7 cabinets and panels in the area that were judged to be significant fire sources), this results in a core damage frequency due to fires in 2-LPNL-025-0031 of

F2 = 2.27E-03 x 1.27E-04 = **2.88E-07**

4kV RPT Board 2-1, Panel 1 and 2 (Fire Zone 2-5)

Electrical components and raceways within the zone of influence of potential fires in RPT board 2-1, as shown in Attachment C.2, have been evaluated, based on the walkdown information described in Attachment D. Due to the potential loss of one loop of reactor recirculation, following a potential trip of these circuit breakers, manual reactor trip is conservatively assumed to occur for any and all fires in these boards. This evaluation gives a conditional core damage frequency (P2 value) of 3.15E-07. Given a fire ignition frequency for this source of 2.27E-03 (total area ignition frequency from Attachment B due to electric cabinets is 1.587E-02 and this represents 1 of 7 cabinets in the area that were judged to be potentially significant fire sources), this results in a core damage frequency due to fires in 4kV RPT board 2-1 of

F2 = 2.27E-03 x 3.15E-07 = **7.15E-10**

4kV RPT Board 2-2, Panel 1 and 2 (Fire Zone 2-5)

Electrical components and raceways within the zone of influence of potential fires in RPT board 2-2, as shown in Attachment C.2, have been evaluated, based on the walkdown information described in Attachment D. This evaluation confirmed that a fire in 4kV RPT board 2-2 only has the potential to impact the operation of RHR inboard isolation valve 2-FCV-74-67, beyond the RPT board itself. Due to the potential loss of one loop of reactor recirculation, following a potential trip of these circuit breakers, manual reactor trip is conservatively assumed to occur for any and all fires in these boards. This evaluation gives a conditional core damage frequency (P2 value) of 3.15E-07.

Given a fire ignition frequency for this source of 2.27E-03 (total area ignition frequency from Attachment B due to electric cabinets is 1.587E-02 and this represents 1 of 7 cabinets in the area that were judged to be potentially significant fire sources), this results in a core damage frequency due to fires in 4kV RPT board 2-2 of

F2 = 2.27E-03 x 3.15E-07 = **7.15E-10**

Panel 25-3 (Filter demin) (Fire Zone 2-5)

Electrical components and raceways within the zone of influence of a fire in filter demineralizer panel 25-3, as shown in Attachment C.2, have been evaluated, based on the walkdown information described in Attachment D. This evaluation confirmed that the zone of influence of a fire in panel 25-3 has the potential to impact circuitry for recirculation pumps 2A and 2B. Due to the potential loss of reactor recirculation, manual reactor trip is conservatively assumed to occur for any and all fires in these panels. This evaluation gives a conditional core damage frequency (P2 value) of 3.15E-07. Given a fire ignition frequency for this source of 2.27E-03 (total area ignition frequency from Attachment B due to electric cabinets is 1.587E-02 and this represents 1 of 7 cabinets in the area that were judged to be potentially significant fire sources), this results in a core damage frequency due to fires in 4kV RPT board 2-2 of

F2 = 2.27E-03 x 3.15E-07 = **7.15E-10**

Panel 25-9 (Sample panel) (Fire Zone 2-5)

Electrical components and raceways within the zone of influence of a fire in sample panel 25-9, as shown in Attachment C.2, have been evaluated, based on the walkdown information described in Attachment D. This evaluation confirmed that the zone of influence of a fire in panel 25-9 has the potential to impact circuitry for recirculation pumps 2A and 2B. Due to the potential loss of reactor recirculation, manual reactor trip is conservatively assumed to occur for any and all fires in these panels. This evaluation gives a conditional core damage frequency (P2 value) of 3.15E-07. Given a fire ignition frequency for this source of 2.27E-03 (total area ignition frequency from Attachment B due to electric cabinets is 1.587E-02 and this represents 1 of 7 cabinets in the area that were judged to be potentially significant fire sources), this results in a core damage frequency due to fires in 4kV RPT board 2-2 of

F2 = 2.27E-03 x 3.15E-07 = **7.15E-10**

240V Lighting Board 2A (Fire Zone 2-5)

Electrical components and raceways within the zone of influence of a fire in 240V lighting board 2A, as shown in Attachment C.2, have been evaluated, based on the walkdown information described in Attachment D.

Since a severe fire in this board could impact RCIC and HPCI control cables, manual plant trip was assumed to occur in response to all fires in this potential fire source with concurrent failure of top events RCI (RCIC) and HPI (HPCI). The conditional core damage frequency (P2 value) from this evaluation is 8.60E-07. Given a fire ignition frequency for this source of 2.27E-03 (total area ignition frequency for fire zone 2-5 due to electric cabinets from Attachment B is 1.587E-02 and this represents 1 of 7 cabinets and panels in the area that were judged to be significant fire sources), this results in a core damage frequency due to fires in 240V lighting board 2A

F2 = 2.27E-03 x 8.60E-07 = **1.95E-09**

<u>240V Lighting Transformer TL2A (Fire Zone 2-5)</u>

Electrical components and raceways that could be affected by a fire in transformer TL2A, as shown in Attachment C.2, have been reviewed for impacts on plant operation. As shown in Attachment C.3, the hot gas layer temperatures produced during a severe transformer oil fire would not damage any electrical components other than those located directly in the plume or within the radiant exposure range. Due to the number of cables that could potentially be impacted by the plume, this fire source was evaluated by conservatively assuming that only a single train of equipment, which has been shown to be routed outside fire zone 2-5, was available following a fire in this transformer. The availability of this equipment is documented in safe shutdown instruction 2-SSI-2-5. It should be noted that this procedure does not take credit for other plant systems that may not have been impacted by the fire and assumes a loss of offsite power. For purposes of plant model quantification, the following top events were set to guaranteed failure:

Offsite power	OG5, OG16
Diesel generators (C and D noted as being "potentially unreliable due to fire")	GC, GD
MSIVs	IVO
HPCI	HPI
RCIC	RCI
Main condensate	CP
RHR pumps	RPA, RPD
Control rod drive	CRD
Core spray	CS

This evaluation gives a conditional core damage frequency (P2 value) of 1.42E-03. Given a fire ignition frequency for this source of 5.04E-04 (total area ignition frequency for this fire zone for transformers from Attachment B 1.513E-03 and this represents 1 of 3 transformers in this area, this results in a core damage frequency due to fires in 240V lighting transformer TL2A of

$$F2 = F1 \times P2$$
$$F2 = 5.04E\text{-}04 \times 1.42E\text{-}03$$
$$\mathbf{F2 = 7.17E\text{-}07}$$

Since this value is less than 1E-06, this fire source can be screened from further consideration. This evaluation remains conservative in that all fires are assumed to be severe and lead to a total drainage of transformer coolant into the diked area prior to fire ignition. Also, no credit is taken for manual or automatic suppression. Finally, an extremely limited set of equipment was evaluated as being available for this fire, whereas no credit was taken for any additional equipment that would actually be available, even in this severe case, beyond the single train of equipment used in the safe shutdown instruction.

Due to the significant level of component damage that could potentially result from a fire in this transformer, this fire source was judged to bound the potential consequences of transient fire sources in the Unit 2 Reactor Building (see Section 6.1.1).

<u>SLC Pumps A and B (Fire Zone 2-5)</u>

Electrical components and raceways within the zone of influence of potential fires in SLC pumps A and B, as shown in Attachment C.2, have been evaluated, based on the walkdown information described in Attachment D. Since these components are modeled in the PRA, manual reactor trip is conservatively assumed to occur for any and all fires in these pumps. This evaluation gives a conditional core damage frequency (P2 value) of 2.12E-05. Given a fire ignition frequency for this source of 5.21E-03 (total area ignition frequency for fire zone 2-5 from Attachment B due to pumps), this results in a core damage frequency due to fires in SLC pumps of

$$\mathbf{F2} = 2.27E\text{-}03 \times 2.12E\text{-}05 = \mathbf{4.81E\text{-}08}$$

<u>Reactor Recirculation Pump MG Sets 2A and 2B (Fire Zone 2-6)</u>

Electrical components and raceways that could be affected by fires in reactor recirculation pump MG sets 2A and 2B have been evaluated, based on the walkdown information described in Attachment D. Due to the potential size of the fire from this source, this zone of influence comprises essentially all of fire zone 2-6. This area is separated from the Unit 1 and Unit 3 Reactor Buildings by 3 hour rated fire barriers and by 1 hour rated barriers to fire zone 2-5. Also, sprinklers are installed over the MG sets to preclude the development of a severe fire. For this evaluation, however, credit is not taken for fire suppression. The evaluation of potential impacts due to recirc MG set fires confirmed that a severe fire has the potential to impact RBCCW system operation, PSC head tank operation, LPCI MG set 2EN and 2DA power to 480V RMOV boards 2D and 2E and 250VDC control power to 4kV shutdown board C.

Fires in the MG sets were therefore modeled by failing top event RBC and failing RS support to top event RK (480V RMOV board 2D) and RT support to top event RL (480V RMOV board 2E) and control power to top event AC (4kV shutdown board C). Given a total recirc MG set fire ignition frequency of 2.13E-03 (total MG set fire frequency for fire zone 2-6 from Attachment B) and a conditional core damage frequency from this evaluation of 1.61E-05, the fire-related core damage frequency for fires in the MG sets can be evaluated as

F2 = 2.13E-03 x 1.61E-05 = **3.42E-08**

This evaluation subsumes the impacts of the other potential fire sources in fire zone 2-6 and remains conservative in that all fires are modeled as severe and no credit is taken for sprinkler protection of these potential fire sources.

<u>4160/480V Emergency Transformer TS2E (Fire Zone 2-6)</u>

The evaluation of components that could be impacted by a fire in transformer TS2E is similar to the evaluation of reactor recirculation pump MG sets, as discussed above. Given a fire ignition frequency of 5.04E-04 (total fire ignition frequency due to transformers in fire zone 2-6 is 1.01E-03 and this represents 1 of 2 transformers in the area), this results in a fire-related core damage frequency for this component of

F2 = 5.04E-04 x 1.61E-05 =**8.11E-09**

<u>LPCI MG Sets 2DA and 2EN (Fire Zone 2-6)</u>

Electrical components and raceways within the zone of influence of potential fires in LPCI MG set 2DA, as shown in Attachment C.2, have been evaluated, based on the walkdown information described in Attachment D. While 480V RMOV board 2D is normally supplied from LPCI MG Set DN and this board is modeled in the PRA (top event RK), manual reactor trip is assumed to occur for any fires in this MG set. This evaluation is similar to that shown for 480V RMOV board 2D, which is described above. Given a fire ignition frequency for this source of 5.32E-04 (total fire frequency for fire zone 2-6 due to MG sets is 2.129E-03 and this represents 1 of 4 MG sets in this area), the core damage frequency due to fires in LPCI MG set 2DA can be calculated as

F2 = 5.32E-04 x 4.35E-07 = **2.31E-10**

Electrical components and raceways within the zone of influence of potential fires in LPCI MG set 2EN, as shown in Attachment C.2, have been evaluated, based on the walkdown information described in Attachment D. This evaluation confirmed that, while it is possible for a fire in LPCI MG Set 2EN to impact LPCI MG Set 2DA circuitry, this form of failure would only require 480V RMOV board 2E to shift to its alternate power source.

Without a LPCI signal or requirement for low pressure injection, no loads would be operating and plant trip would not be expected to occur. Since this board is modeled in the PRA (top event RL), manual reactor trip is assumed to occur for any fires in this MG set. The potential loss of both of these power sources is similar to that shown for 480V RMOV board 2B (top event RT), which is described above. Given a fire ignition frequency for this source of 5.32E-04 (total fire frequency for fire zone 2-6 due to MG sets is 2.029E-03 and this represents 1 of 4 MG sets in this area), the core damage frequency due to fires in LPCI MG set 2EN can be calculated as

F2 = 5.32E-04 x 2.67E-06 = **1.42E-09**

<u>LPNL-25-23 and 24 (Fire Zone 2-6)</u>

Electrical components and raceways within the zone of influence of potential fires in panels LPNL-25-23 and LPNL-25-24, as shown in Attachment C.2, have been evaluated, based on the walkdown information described in Attachment D. A fire in either or both of these panels would not be expected to result in plant trip. Also, these components are not required for the safe shutdown of the plant and are not included in the PRA equipment list. Therefore, fires in LPNL-25-23 and LPNL-25-24 can be screened from further consideration.

<u>240V Lighting Board 2B (Fire Zone 2-6)</u>

Electrical components and raceways within the zone of influence of potential fires in 240V lighting board 2B, as shown in Attachment C.2, have been evaluated, based on the walkdown information described in Attachment D. Fires in this board would not be expected to result in plant trip. Also, these components are not required for the safe shutdown of the plant and are not included in the PRA equipment list. Therefore, fires in 240V lighting board 2B can be screened from further consideration.

<u>240V Lighting Transformer TL2B (Fire Zone 2-6)</u>

The calculations shown in Attachment C.2 indicate that this fire source has the potential to form a hot gas layer in the immediate area. More detailed calculations performed in Attachment C.3 show that the hot gas layer temperatures will not be high enough to damage components that are located in the area. Plant walkdowns, as described in Attachment D, show that there is only one conduit (2ES3925) in the vicinity of transformer TL2B. This conduit which contains the normal supply cables for 480V RMOV board 2E from LPCI MG set 2EN. Since the loads on this board are not normally energized or demanded to operate during plant operation, plant trip would not be expected due to fires in this location. However, since this board is modeled in the PRA (top event RL), manual reactor trip is assumed to occur for any fires in this MG set. This evaluation is similar to that shown for 480V RMOV board 2E (i.e. failure of top event RL), which is described above.

Given a fire ignition frequency for this source of 5.04E-04 (total fire frequency for fire zone 2-6 due to transformers is 1.01E-03 and this represents 1 of 2 transformers in this area), the core damage frequency due to fires in 240V lighting transformer TL2B can be calculated as

$$F2 = 5.04E\text{-}04 \times 4.18E\text{-}07 = \mathbf{2.10E\text{-}10}$$

6.1.3 Summary of Unit 2 Reactor Building Ignition Sources

Table 6-3, below, summarizes the evaluation of fixed ignition sources in the Unit 2 Reactor Building. As noted in Section 6.1.1, above, transient fire sources have been treated as fires in 240V lighting transformer TL2A. This fire source was selected as a modeling basis for the transient fire sources due to the high conditional core damage frequency. Also, unqualified cables are listed, for completeness, as a fire source, in Table 6-3. As noted in the prior discussion, this evaluation assumes that any and all of these fires result in a total loss of offsite power.

Table 6-3
Unit 2 Reactor Building Fire Sources

Area	Fire Source	Ignition Freq. (F1)	Conditional Core Damage Freq. (P2)	Core Damage Frequency (F2)
2-1	250V RMOV Board 2C	3.97E-03	4.60E-07	1.83E-09
	2-PNLA-25-340	Screened		
	Drywell/Torus Compressor	5.42E-04	6.56E-06	3.55E-09
	Core Spray Pumps 2A, 2C	2.08E-03	6.56E-06	1.36E-08
	RHR Pumps 2A, 2C	2.08E-03	2.26E-06	4.70E-09
2-2	480V RMOV Board 2C	2.91E-03	4.77E-07	1.39E-09
	480V RB Vent Board 2B	2.91E-03	4.90E-06	1.43E-08
	2-PNLA-25-341	Screened		
	Core Spray Pumps 2B, 2D	2.08E-03	6.56E-06	1.36E-08
	RHR Pumps 2B and 2D	2.08E-03	3.26E-06	6.78E-09
2-3	Unit 2 Pref. AC Transformer	5.04E-04	3.17E-05	1.60E-08

Table 6-3
Unit 2 Reactor Building Fire Sources

Area	Fire Source	Ignition Freq. (F1)	Conditional Core Damage Freq. (P2)	Core Damage Frequency (F2)
2-4	480V RMOV Board 2D	7.94E-04	4.35E-07	3.45E-10
	RBCCW Pump 2A	2.08E-03	2.13E-06	4.44E-09
	RBCCW Pump 2B	2.08E-03	4.94E-06	1.03E-08
	RWCU Room Monitor	Screened		
	Shutdown Board Room HVAC Compressor Motor	1.08E-03	2.42E-04	2.63E-07
2-5	480V RMOV Board 2E	2.27E-03	4.18E-07	9.49E-10
	LPCI MG Set 2DN	5.33E-04	4.35E-07	2.32E-10
	LPCI MG Set 2EA	5.32E-04	2.39E-06	1.27E-09
	4kV/480V Trans. TS2A	5.04E-04	5.27E-06	2.66E-09
	4kV/480V Trans. TS2B	5.04E-04	7.08E-06	3.57E-09
	2-LPNL-025-0031	2.27E-03	1.27E-04	2.88E-07
	RPT Board 2-1	2.27E-03	3.15E-07	7.15E-10
	RPT Board 2-2	2.27E-03	3.15E-07	7.15E-10
	Panel 25-3	2.27E-03	3.15E-07	7.15E-10
	Panel 25-9	2.27E-03	3.15E-07	7.15E-10
	240V Lighting Board 2A	2.27E-03	8.60E-07	1.95E-09
	240V Lighting Trans. TL2A	5.04E-04	1.42E-03	7.17E-07
	SLC Pumps A and B	5.21E-03	2.12E-05	4.81E-08

Table 6-3 Unit 2 Reactor Building Fire Sources				
Area	Fire Source	Ignition Freq. (F1)	Conditional Core Damage Freq. (P2)	Core Damage Frequency (F2)
2-6	Recirc MG Sets 2A, 2B	2.13E-03	1.61E-05	3.42E-08
	4kV/480V Trans. TS2E	5.04E-04	1.61E-05	8.11E-09
	LPCI MG Set 2DA	5.32E-04	4.35E-07	2.31E-10
	LPCI MG Set 2EN	5.32E-04	2.67E-06	1.42E-09
	LPNL-25-23 and 24	Screened		
	240V Lighting Board 2B	Screened		
	240V Lighting Trans. TL2B	5.04E-04	4.18E-07	2.10E-10
Transient Sources		4.03E-06	1.42E-03	5.79E-09
Unqualified Cable		1.18E-02	4.63E-05	5.47E-07

Since the fire related core damage frequency for each of these sources is less than 1E-06, these potential fire sources can be screened from further consideration.

6.1.4 Comparison of Unit 2 with Unit 1 and Unit 3 Reactor Buildings

Browns Ferry Units 1 and 3 are currently not operating and appear in the Unit 2 analysis in a support capacity only. Fires in any given plant location for Units 1 or 3 would, by definition, be more severe if such a fire occurred at the operating unit. Given this consideration, a bounding case run was evaluated for potential fires in the Unit 1 and Unit 3 Reactor Buildings. In each of these cases, it was assumed that the equipment located in any fire areas within the Reactor Buildings themselves could potentially fail, since the associated cables may transit through the affected Reactor Building to Unit 2. The fire areas containing plant components that could potentially be damaged due to fire-induced cable damage from fires in the Unit 1 and Unit 3 Reactor Buildings are shown in Table 6-4, below. This table includes the Diesel Generator Buildings, since the power cables for diesel generators C and D may transit through the Unit 1 Reactor Building on their way to shutdown boards C and D. The Unit 3 Diesel Generator Building is then included for completeness.

The same reasoning applies to 4kV shutdown board rooms A and B, which are assumed to fail all cables that traverse the Unit 1 Reactor Building and 4kV shutdown board rooms 3EA, 3EB, 3EC and 3ED, which are assumed to fail all cables that traverse the Unit 3 Reactor Building.

Table 6-4
Fire Areas Located in Unit 1 and 3 Reactor Buildings
(Includes Diesel Generator Buildings)

Fire Area	Description	Impacted Top Events
Unit 1 Reactor Building		
4	4kV Shutdown Board Room B, 593' Elevation	AB, RF
5	4kV Shutdown Board Room A and 250V Battery Room (621' Elevation)	AA, RE, DA, DC
6	480V Shutdown Board Room 1A, 621' Elevation	RQ
7	480V Shutdown Board Room 1B, 621' Elevation	RR
20	Unit 1 and 2 Diesel Generator Building	GA, GB,.GC, GD, SDREC
Unit 3 Reactor Building		
12	Shutdown Board Room F, 593' Elevation	A3EA, A3EC
13	Shutdown Board Room E, 621' Elevation	
14	480V Shutdown Board Room 3A, 621' Elevation	RX
15	480V Shutdown Board Room 3B, 621' Elevation	RY
21	Unit 3 Diesel Generator Building	GE, GF, GG, GH, SDREC, DF, CPREC

Table 6-4 Fire Areas Located in Unit 1 and 3 Reactor Buildings (Includes Diesel Generator Buildings)		
Fire Area	Description	Impacted Top Events
22	4kV Shutdown Board Rm 3EA, 3EB, 583' Elevation, Unit 3 DG Building	A3EA, A3EB
23	4kV Shutdown Board Rm 3EC, 3ED, 583' Elevation, Unit 3 DG Building	A3EC, A3ED
24	4kV Bus Tie Board Room, 565' Elevation, Unit 3 DG Building	

Therefore, based on the information shown in Table 6-4, above, a bounding case fire was evaluated for the Unit 1 Reactor Building by assuming failure of the following top events:

AA, AB, DA, DC, RE, RF, RQ, RR, GA, GB, GC, GD and SDREC

This evaluation generated a conditional core damage frequency of 5.62E-07. Given a fire ignition frequency from Table 4-1 for this area of 9.24E-02, the fire-related core damage frequency for this area can be evaluated as being no higher than:

F2 = P1 x F1 = 5.62E-07 x 9.24E-02 = **5.19E-08**

Since this value is less than 1E-06, fires in the Unit 1 Reactor Building can be screened from further consideration. This evaluation is conservative in that all fires in this area are assumed to damage all of the cables that are assumed to transit through the area, regardless of fire severity or manual or automatic suppression.

In a similar fashion, a bounding case fire was evaluated for the Unit 3 Reactor Building by assuming failure of the following top events:

A3EA, A3EB, A3EC, A3ED, DF, CPREC, RX, RY, GE, GF, GG, GH and SDREC

This evaluation generated a conditional core damage frequency of 1.16E-06. Given a fire ignition frequency from Table 4-1 for this area of 9.26E-02, the fire-related core damage frequency for this area can be evaluated as being no higher than:

F2 = P1 x F1 = 1.16E-06 x 9.26E-02 = **1.06E-07**

Since this value is less than 1E-06, fires in the Unit 3 Reactor Building can be screened from further consideration. This evaluation is conservative in that all fires in this area are assumed to damage all of the cables that are assumed to transit through the area, regardless of fire severity or manual or automatic suppression.

6.2 Use of Event Trees

During the quantitative screening process described in Section 5, several plant areas were arbitrarily assigned a conditional core damage frequency of 1.0. By definition, each of these areas was then retained for detailed analysis. The detailed evaluation of Reactor Building areas is described in Section 6.1. The remaining areas are:

Fire Area 4	4kV Shutdown Board Room B
Fire Area 5	4kV Shutdown Board Room A
Fire Area 8	4kV Shutdown Board Room D
Fire Area 9	4kV Shutdown Board Room C
Compartment 16-1	Control Building, 593 Foot Elevation
Compartment 16-2	Cable Spreading Rooms
Compartment 16-3	Control Rooms
Fire Area 18	Unit 2 Battery and Battery Board Rooms
Compartment 25-1	Intake Structure
Compartment 25-2	Pipe Tunnel
Compartment 25-3	Turbine Building

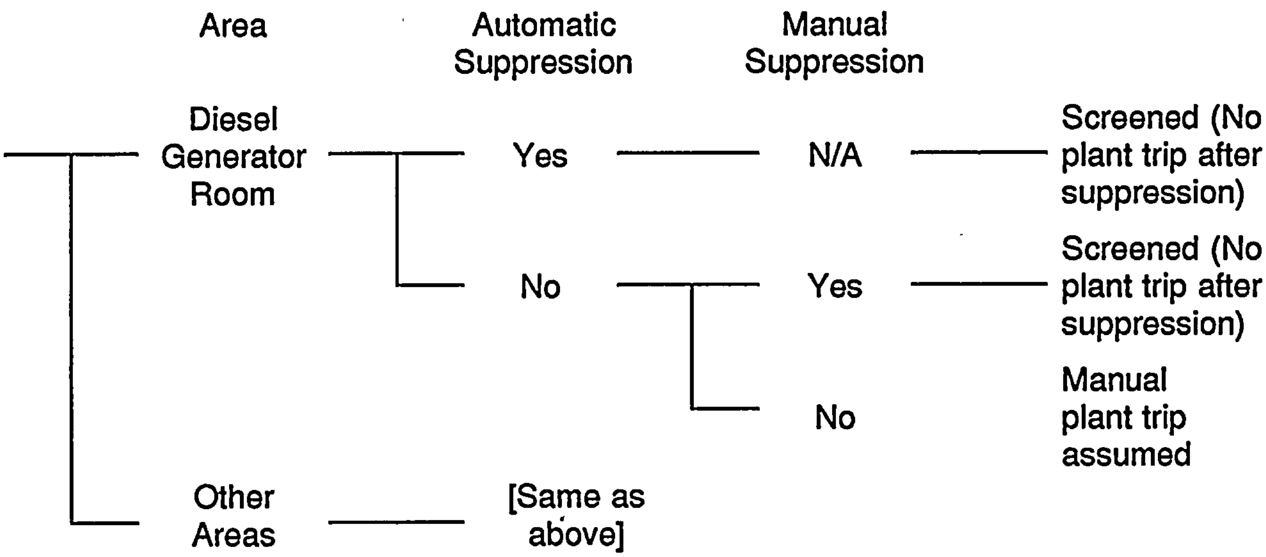
Of these areas, fire areas 4, 5, 8, 9 and 18 were not screened following evaluation in Section 5. This evaluation of all fires as engulfing generated a fire-related core damage frequency for each of these areas of more than 1E-06. Therefore, more detailed analysis of these areas was warranted. The form of evaluation selected for these areas utilizes an "event tree" approach to identify individual cases for evaluation.

During the screening evaluation described in Section 5, all fires, regardless of actual fire severity, were arbitrarily assumed to engulf the affected area. By default, this form of treatment evaluates all fires as "high frequency/high consequence" events. In the use of an event tree approach to the analysis of fires, the fire frequency for the remaining areas is segmented into a range of cases. These cases are selected to cover the scale between "high frequency/low consequence" events, such as fires that result in a turbine trip only, to "low frequency/high consequence" events, such as Control Room evacuation. In the area of low consequence fires, if there is no reason to believe that an automatic plant trip would result or that the operator would trip Unit 2, an initiating event can be avoided altogether. In areas that contain Unit 2 equipment, plant trip is conservatively assumed to occur for all fires. It should be noted that plant trip of Unit 2 would not be expected following a number of fires that could occur in the plant, such as fires in the Unit 3 Hydrogen recombiners.

Each of the areas that remain for detailed evaluation is described in the appropriate section below. For each of these areas, a description of the area, including any significant fire sources, targets and mitigating features, is provided. Where a large number of cases are used, a graphic "event tree" is shown to assist in identifying the particular cases for each area.

Each of the cases to be evaluated is then described in the text and the results of the evaluation are summarized in a table. Each section then closes with a discussion of the resulting evaluation, including comments on why the analysis remains conservative.

To illustrate the use of an event tree, one can assume that a fire occurs within the Unit 3 Diesel Generator Building (Fire Area 21). In this case, there is no reason to expect that a plant trip would result for Unit 2, unless the fire spreads outside the initial room. That is, a fire in DG 3A would have to spread to DG 3B or to the pipe chase area before the operator would be expected to trip Unit 2. At this point, however, the operator may be expected to trip the unit, based on the potential loss of all diesels. Instead of arbitrarily assuming that all fires totally engulf the building (see discussion for this area in Section 5.1.21), credit can now be taken for a number of plant design features, including automatic suppression. Since damage to all equipment located in the affected room is conservatively assumed to occur, response time for the fire brigade is based on preventing the spread of the fire to other rooms, instead of "time to damage" calculations for equipment located in the area. Graphically, the event tree for this evaluation can be shown as:



This graphic representation can be read as:

1. All of the fire ignition frequency is segmented into those fires that occur in the diesel generator rooms themselves and those fires that occur within the other rooms of the Diesel Generator Building.

2. Automatic suppression for the affected area is then evaluated. If automatic suppression is successful, plant trip would be avoided. It should be noted that automatic suppression is not credited where it is not installed. In this case, the "other areas" may have automatic fire suppression systems installed, but some do not.

3. For those cases where automatic suppression is either not modeled or is unsuccessful, manual suppression of the fire before it spreads to an adjacent room is questioned. It should be noted that this condition still assumes that all equipment located in the room is damaged, regardless of fire severity or suppression. If manual suppression is successful, plant trip is avoided, though an LCO condition, such as 7 day operation, may be imposed by plant technical specifications, due to the equipment that was assumed to be damaged.

4. In the cases where the fire was not suppressed, it is conservatively assumed that the fire has the strength to breach adjacent walls into a second area. It is further assumed that the operator will initiate a reactor trip at this point.

At this point, the fire ignition frequency from Table 4.1 has been used to generate a fire-related initiating event frequency. This use of an event tree also allows the identification of specific equipment damage, due to fire location or other factors. In the example above, it is conservatively assumed that all diesel generators are failed by any unsuppressed fires. This information is then used, in conjunction with the Level 1 PRA plant model, to generate a conditional core damage frequency for this fire-related initiating event. If the fire-related initiating event frequency, multiplied by the conditional core damage frequency is less than 1E-06, the area can be screened from further consideration.

Given the plant model impacts for a given fire, or for a specific case for a given area, the Level 1 plant model from the IPE is used to develop a list of core damage scenarios, based on the likelihood of hardware failure and equipment unavailability. These core damage scenarios are then totalled to give a conditional core damage frequency for the fire event under consideration. This conditional value is then multiplied by the fire frequency for the current case to generate a fire-related core damage frequency for that case. The fire-related core damage frequency for each case is then totalled to generate the fire-related core damaged frequency for the entire area. If this value is less than 1E-06, the area can be screened from further consideration.

6.2.1 Review of the EPRI Fire Events Database (NSAC/178L)

In order to expand the quantitative screening process described in the EPRI FIVE documentation from the evaluation of all fires as "high frequency/high consequence" events, a separate review of the fire events database was performed, specifically to augment the use of an event tree approach, as described above. The use of an event tree approach to fire hazard evaluation provides the framework within which to segment the fire ignition frequency into "minor" and "severe" cases. The data source used to generate this information is the fire events database developed by EPRI (NSAC/178L).

During the development and evaluation of plant model impacts for each of the various fire areas under consideration, assumptions must be made concerning the population of fires that can occur. Specific questions concerning the likelihood that a given fire will have the potential to develop into a severe event must be answered before one can effectively evaluate the plant risk due to fire hazards. This process begins with a detailed review of the fire events in the fire events database. This database is described in EPRI document NSAC/178L and is the same pool of information that was used to develop the fire ignition frequencies shown in Table 4-1. This information then provides a consistent reference point, from which to glean insights concerning industry experience with fires.

A two-step process is used during this review. The first review of the fire events database consists of a review of the means used to suppress the fire (i.e. use of hose streams or installed suppression systems). The use of a hose stream or installed system to suppress a fire indicates the presence of a significant fire, as opposed to fires that may have been suppressed by use of portable extinguishers or allowed to burn out. The second review evaluates the text descriptions for data entries that may not have this information filled in. This second step is performed in order to ensure that the data entries that do contain this information do, in fact, represent the rest of the population of fire events as a whole.

A total of 753 fire events are documented in NSAC/178L. Of these database entries, 577 have specific entries describing the equipment that was used to suppress the fire. This information appears in the "EQUIP_USED" data field. These 577 fire events can be divided by fire ignition source and separated into categories (i.e. hose stream or installed system) as shown in Table 6-5, below:

Table 6-5 Means of Suppression for Various Fire Sources			
Fire Source	Means of Suppression		Total Entries
	Hose Stream or Installed System	Other Means	
Not Specified	23	52	75
Welding during construction	10	48	58
Diesel Generator	12	36	48
Dryers	3	8	11
Elevator Motors	1	4	5
Offgas	3	30	33
Main Feed Pumps	4	5	9
Other Pumps	8	30	38
Gas Turbine	1	3	4
Motor Control Centers (applicable to switchgear areas)	6	34	40
T/G Oil	4	8	12
T/G Hydrogen	3	2	5
Other Hydrogen	1	5	6
Yard Transformers	14	5	19
Other Transformers	3	6	9
Other	12	193	205
Total	108	469	577

Of the 12 significant fires that are attributed to "other" sources, 5 were due to transient fire sources, 3 were due to fires in junction boxes or cables, 2 were due to welding and 2 were panel or cabinet fires. For purposes of this analysis, then, the following fire severity factors can be used:

For fires in shutdown board A, B, C and D rooms, fires caused by motor control centers and other sources apply. For these fire sources, (6 + 12 =) 18 fires were suppressed with hose streams or by installed systems, of (40 + 205 =) 245 total fires from these sources. In other words, (18/245 =) 7.3% of the fires in these areas are judged to be severe.

For fires in cable tray areas, only the "other" fire sources apply. For these fires, (12/205 =) 5.9% are judged to be severe.

Of the (753 - 577 =) 176 entries that had no entry describing the equipment used to suppress the fire (i.e. no listing under the "EQUIP_USED" data field), the text descriptions were reviewed to provide an indication of the severity of the fire. From this review, few (less than 10%) of the descriptive entries indicated that the fire had the potential to develop into a severe fire. In other words, the remaining fires in the database do not represent a significantly higher rate of potentially significant fires than those entries that provided the information used to generate Table 6-5.

6.2.2 Fire Area 4 - 4kV Shutdown Board Room B

Review of the results from the screening quantification of this area (see Section 5.1.4) revealed that the results were dominated by the assumed failure of both shutdown buses due to an assumed catastrophic fire in this area. Further review of the layout of this area confirms that the cubicles for these interties are separated by a distance of 20 to 25 feet. Also, only one shutdown bus is used to supply the shutdown board at any time, with the other circuit breaker open.

Given this situation, the most credible failure of a shutdown bus would be for the circuit breaker from shutdown bus 1 to fail to trip, causing shutdown board A to shift to shutdown bus 2. This form of failure would require the fire to fail the DC control power to trip the tie breaker, followed by failure of the buswork inside shutdown board B itself, which would require an extensive and severe fire.

Based on a the review of fire severity shown in the Fire Events Database, approximately 93% of the fires that would be expected to occur in this area can be extinguished with portable equipment or allowed to burn out. If necessary, this response would be carried out by the first responder on the scene from the fire brigade. Due to the low level of severity for these fires, damage to either of the bus ends for either shutdown bus is unlikely to occur.

For the remaining fires (7.3% of total ignition frequency), all fires are modeled as becoming engulfing, similar to the analysis shown in Section 5.1.4.

The revised evaluation of this area therefore consists of evaluating 2 cases:

Case 1 A minor fire starts in 4kV shutdown board room B. This fire is then either suppressed by the first responder on the scene or is allowed to burn out. 4kV shutdown board B (top event AB) and 480V RMOV board 1B (top event RF) are conservatively assumed to fail prior to fire suppression. As noted above, this case is assigned 93% of total area fire ignition frequency or (6.74E-03 x 0.93 =) 6.27E-03.

Case 2 A fire starts anywhere in fire area 4 and is eventually suppressed with hose streams. This fire is then conservatively assumed to spread to envelop and damage all components in the area, including the interties with shutdown buses 1 and 2 (i.e. similar to the evaluation shown for this area in Section 5.1.4). This case is assigned 7.3% of total area fire frequency, as described in Section 6.2.1, above, or (6.74E-03 x 0.073 =) 4.92E-04

The evaluation of each of these cases is shown in Table 6-6, below.

Table 6-6
Evaluation of Fires in Fire Area 4

Case	Description	Case Frequency (F1)	Probability of Core Damage (P2)	Core Damage Frequency (F1 x P2)
Case 1	Minor fire, suppressed	6.27E-03	7.38E-06	4.63E-08
Case 2	Severe fire, assumed to become engulfing and fail both shutdown buses	4.92E-04	9.17E-04	4.51E-07
Total		6.76E-03		4.97E-07

Since the total core damage frequency for both of these cases is less than 1E-06, fires in this area can be screened from further consideration. This evaluation is judged to remain conservative in that all fires, regardless of size or location, are assumed to result in plant trip, with significant damage to plant components. Also, no credit is taken for fire suppression, beyond the response of the initial fire brigade member on the scene.

6.2.3 Fire Area 5 - 4kV Shutdown Board Room A and 250V Battery Room

Review of the results from the screening quantification of this area (see Section 5.1.4) revealed that the results were dominated by the assumed failure of both shutdown buses due to an assumed catastrophic fire in this area. Further review of the layout of this area confirms that the cubicles for these interties are separated by a distance of 20 to 25 feet. Also, only one shutdown bus is used to supply the shutdown board at any time, with the other circuit breaker open.

Given this situation, the most credible failure of a shutdown bus would be for the circuit breaker from shutdown bus 1 to fail to trip, causing shutdown board B to shift to shutdown bus 2. This form of failure would require the fire to fail the DC control power to trip the tie breaker, followed by failure of the buswork inside shutdown board A itself, which would require an extensive and severe fire.

Based on a the review of fire severity shown in the Fire Events Database, approximately 93% of the fires that would be expected to occur in this area can be extinguished with portable equipment or allowed to burn out. If necessary, this response would be carried out by the first responder on the scene from the fire brigade. Due to the low level of severity for these fires, damage to either of the bus ends for either shutdown bus is judged to be unlikely to occur.

For the remaining fires (7.3% of total ignition frequency), all fires are modeled as becoming engulfing, similar to the analysis shown in Section 5.1.5.

The revised evaluation of this area therefore consists of evaluating 2 cases:

Case 1 A minor fire starts in 4kV shutdown board room A. This fire is then either suppressed by the first responder on the scene or is allowed to burn out. 4kV shutdown board A (top event AA), 480V RMOV board 1B (top event RE), and 250VDC control power (top events DA and DC) are conservatively assumed to fail prior to fire suppression. As noted above, this case is assigned 93% of total area fire ignition frequency or (8.57E-03 x 0.93 =) 7.97E-03.

Case 2 A fire starts anywhere in fire area 4 and is eventually suppressed with hose streams. This fire is then conservatively assumed to spread to envelop and damage all components in the area, including the interties with shutdown buses 1 and 2 (i.e. similar to the evaluation shown for this area in Section 5.1.5). This case is assigned 7.3% of total area fire frequency, as described in Section 6.2.1, above, or (8.57E-03 x 0.073 =) 6.26E-04

The evaluation of each of these cases is shown in Table 6-7, below.

Table 6-7 Evaluation of Fires in Fire Area 5				
Case	Description	Case Frequency (F1)	Probability of Core Damage (P2)	Core Damage Frequency (F1 x P2)
Case 1	Minor fire, suppressed	7.97E-03	2.14E-06	1.71E-08
Case 2	Severe fire, assumed to become engulfing and fail both shutdown buses	6.26E-04	3.79E-04	2.37E-07
Total		8.60E-03		2.54E-07

Since the total core damage frequency for these cases is less than 1E-06, fires in this area can be screened from further consideration. This evaluation is judged to remain conservative in that all fires, regardless of size or location, are assumed to result in plant trip, with significant damage to plant components. Also, no credit is taken for fire suppression, beyond the response of the initial fire brigade member on the scene.

6.2.4 Fire Area 8 - 4kV Shutdown Board Room D

Review of the results from the screening quantification of this area (see Section 5.1.8) revealed that the results were dominated by failure of containment heat removal through suppression pool cooling following failure of relief valve closure after an assumed MSIV closure due to loss of support for the drywell air system. The conservative nature of this evaluation was due to the following three main factors:

1. All fires in this area were assumed to lead to irrecoverable damage to 480V RMOV board 2B, which is required to support suppression pool cooling by providing motive force for the torus return valves for division II (RHR pumps 2B and 2D).

2. All fires in this area were assumed to lead to irrecoverable damage to 250V RMOV board 2B. This panel supplies power to the logic relays that are required to remotely operate the torus return valves for suppression pool cooling with the division I RHR pumps (2A and 2C).

3. Conservative treatment of support system requirements, both explicit and implicit, was found within the Level 1 IPE plant model. When all components were assumed to fail for all fires in this area during the screening evaluation, this included 120 VAC I&C bus 2B, which is normally supplied from power sources in this room. Top event DO was therefore failed during the screening analysis, whereas review of the electric power system analysis (IPE documentation - BFN Electric Power System, figure on page 2-24 of the system analysis) shows that a degraded failure rate, as indicated by use of split fraction DO3, should have been used. It should be noted that I&C bus 2B is set to guaranteed failure for these evaluations, due to the assumed loss of both normal and alternate power supplies (i.e. due to "cascaded dependencies").

   This explicitly conservative treatment of a single top event then brought about a number of implicit conservative assumptions within the logic structure of the plant model. First, the drywell air system is assumed to fail following failure of top event DO, due to isolation of suction valve 2-FCV-32-63. This leads to an assumed immediate closure of MSIVs, since accumulators are conservatively not modeled. Following the assumed loss of 480V RMOV 2B and 250V RMOV board 2B due to the fire, suppression pool cooling was failed, with recovery conservatively not modeled, even though the operator would have a significant amount of time available in which to restore cooling. Since MSIV closure was assumed to occur within the logic structure of the model, RPV relief valves were required to lift following plant trip. The independent failure of valve reclosure then leads directly to an assumed failure of the containment, and, subsequently, to assumed core damage.

Review of the layout of this area shows that the two primary targets of concern, 480V RMOV board 2B and 250V RMOV board 2B, are separated from each other by a walkway approximately 3 feet wide. These targets are also separated from other fire sources by an access walkway approximately 6 feet wide.

There is no automatic fire suppression in this area, but addressable (analog) smoke detectors are used to provide area wide coverage. The initial fire brigade responder is expected to be on the scene and capable of suppressing a minor fire before damage to other plant components occur. Since the fire sources in this area include motor control centers, as well as other, predominantly transient, fire sources, a minor fire fraction of (100% - 7.3% =) 93% is applied to cases 1 and 2, as described in Section 6.2.1.

The revised evaluation of this area therefore consists of evaluating 3 cases:

Case 1 A minor fire starts in 250V RMOV board 2B. This fire is then conservatively assumed to envelop and damage all equipment in the area, with the exception of 480V RMOV board 2B.

This case is assigned 1/3 of the panel frequency of this area, and adjusted to include minor fires only as 3.00E-03/3 x (0.93) = 9.30E-04.

Case 2 A minor fire starts anywhere in fire area 8, except in 250V RMOV board 2B. This fire is then conservatively assumed to spread to envelop and damage all components in the area, with the exception of 250V RMOV board 2B. This case is assigned the remaining frequency from Case 1 and adjusted to include minor fires only as (6.64E-03-(3.00E-03/3)) x (0.93) = 5.25E-03.

Case 3 A fire starts anywhere in the area and grows to envelop and damage all components in the area. This case is similar to the initial screening evaluation, except that I&C bus 2B is not assumed to fail, only lose power from sources that are located in this area. The fire ignition frequency assigned to this case is 6.64E-03 x 0.073 = 4.85E-04.

The evaluation of each of these cases is shown in Table 6-8, below.

Table 6-8
Evaluation of Fires in Fire Area 8

Case	Description	Case Frequency (F1)	Probability of Core Damage (P2)	Core Damage Frequency (F1 x P2)
Case 1	Minor fire in 250V RMOV board 2B, assumed to fail all equipment but 480V RMOV board 2B	9.30E-04	4.46E-05	4.15E-08
Case 2	Minor fire anywhere but 250V RMOV board 2B, assumed to fail all equipment but 250V RMOV board 2B	5.25E-03	3.42E-05	1.80E-07
Case 3	Engulfing fire, assumed to fail all equipment and cables in fire area 8	4.85E-04	3.98E-04	1.93E-07
Total		6.64E-03		4.15E-07

Since the total core damage frequency for all three of these cases is less than 1E-06, fires in this area can be screened from further consideration. This evaluation is judged to remain conservative in that all fires, regardless of size or location, are assumed to result in plant trip, with significant damage to numerous and non-adjacent components. Also, no credit is taken for manual fire suppression for any of these cases.

6.2.5 Fire Area 9 - 4kV Shutdown Board Room C and 250V Battery Room

Review of the results from the screening quantification of this area (see Section 5.1.9) revealed that the results were dominated by failure of containment heat removal through suppression pool cooling following safety valve failure to reclose after an assumed MSIV closure due to loss of support for the drywell air system. This evaluation was due to the following three main factors:

1. All fires were assumed to lead to irrecoverable damage to 480V RMOV board 2A. This panel is required to support suppression pool cooling by providing motive force for the torus return valves for division I (RHR pumps 2A and 2C).

2. All fires in this area were assumed to lead to irrecoverable damage to 250V RMOV board 2A, which is required to support the logic relays that are required to remotely operate the torus return valves for suppression pool cooling with the division II RHR pumps (2B and 2D).

3. Support system requirements are treated conservatively within the Level 1 IPE plant model. When all components were assumed to fail for all fires in this area during the screening evaluation, this included 120 VAC I&C bus 2A, which is normally supplied from power sources in this room. Top event DN was therefore failed during the screening analysis, whereas review of the electric power system analysis (see IPE documentation, BFN Electrical Power System, figure on page 2-24 of the system analysis) shows that a degraded failure rate, as indicated by use of split fraction DN3, should have been used. This explicitly conservative treatment of a single top event then invoked a number of implicit conservative assumptions within the plant model. First, the drywell air system is assumed to fail following failure of top event DN, due to isolation of suction valve 2-FCV-32-62. This leads to an assumed immediate closure of MSIVs, since accumulators are conservatively not modeled. Following the assumed loss of 480V RMOV 2A and 250V RMOV board 2A due to the fire, suppression pool cooling was failed, with recovery conservatively not modeled, even though the operator would have a significant amount of time available in which to restore cooling.

Since MSIV closure is assumed to occur within the logic structure of the model, RPV relief valves are required to lift following plant trip. The independent failure of valve reclosure then leads directly to an assumed failure of the containment and, subsequently, to assumed core damage.

Review of the layout of this area, including physical walkdown of the area, as documented in Attachment D, shows that the two primary targets of concern, 480V RMOV board 2A and 250V RMOV board 2A are separated from each other by a walkway approximately 3 feet wide. These targets are also separated from other fire sources by an accessway approximately 6 feet wide.

There is no automatic fire suppression in this area, but addressable (analog) smoke detectors are used to provide area wide coverage. The initial fire brigade responder is expected to be on the scene and capable of suppressing a minor fire before damage to other plant components occur. Since the fire sources in this area include motor control centers, as well as other, predominantly transient, fire sources, a minor fire fraction of (100% - 7.3% =) 93% is applied to cases 1 and 2, as described in Section 6.2.1.

The revised evaluation for this area therefore consists of evaluating 3 cases:

Case 1 A minor fire starts in 250V RMOV board 2A. This fire is then conservatively assumed to envelop and damage all equipment in the area, with the exception of 480V RMOV board 2A. This case is assigned 1/3 of the panel frequency of this area, and adjusted to include minor fires only as 3.00E-03/3 x (0.93) = 9.30E-04.

Case 2 A minor fire starts anywhere in fire area 9, except in 250V RMOV board 2A. This fire is then conservatively assumed to spread to envelop and damage all components in the area, with the exception of 250V RMOV board 2A. This case is assigned the remaining frequency from Case 1 and adjusted to include minor fires only as (8.074E-03-(3.00E-03/3)) x (0.93) = 6.58E-03.

Case 3 A fire starts anywhere in the area and grows to envelop and damage all components in the area. This case is similar to the initial screening evaluation, except that I&C bus 2A is not failed, only degraded by loss of sources in this area. It should be noted that this case subsumes the conceivable failure of MSIV or RPV relief valve control circuitry that is located in remote shutdown panel 9-32 (assumed to become damaged by a hot gas layer, which is assumed to develop over the entire fire area, despite the efforts of the fire brigade). The fire ignition frequency of this case is 8.074E-03 x 0.073 = 5.89E-04.

The evaluation of each of these cases is shown in Table 6-9, below.

Table 6-9 Evaluation of Fires in Fire Area 9				
Case	Description	Case Frequency (F1)	Probability of Core Damage (P2)	Core Damage Frequency (F1 x P2)
Case 1	Minor fire in 250V RMOV board 2A, assumed to fail all equipment but 480V RMOV board 2A	9.30E-04	6.41E-07	5.96E-10
Case 2	Minor fire anywhere but 250V RMOV board 2A, assumed to fail all equipment but 250V RMOV board 2A	6.58E-03	2.37E-05	1.56E-07
Case 3	Engulfing fire, assumed to fail all equipment and cables in fire area 9	5.89E-04	7.24E-04	4.26E-07
Total		8.07E-03		4.51E-07

Since the total core damage frequency for all three of these cases is less than 1E-06, fires in this area can be screened from further consideration. This evaluation is judged to remain conservative in that all fires, regardless of size or location, are assumed to result in plant trip, with significant damage to numerous and non-adjacent components.

6.2.6 Fire Compartment 16-1, Control Building, Elevation 593 (Equipment Areas)

This area is not separated from upper elevations of the Control Building by rated fire barriers, though the ceiling/floor interface with the Cable Spreading Rooms (compartment 16-2) and the walls between rooms are of substantial construction, using non-combustible materials that are equivalent to a fire rating of 1.5 hours. Fire detection for this area is provided by area-wide addressable (analog) detectors, which alarm locally and in the Control Room.

The 593 foot elevation of the Control Building is laid out as a series of individual rooms, which are located on either side of the unit battery and battery board rooms (fire areas 17, 18 and 19), each of which is enclosed with rated fire barriers.

A single corridor, running the entire length of the Control Building (approximately 450 feet), serves as the access path to all of these areas. There are no significant combustibles located in this corridor area. The rated fire boundaries of fire areas 17, 18 and 19 act to segment the remaining rooms on this elevation into four groups. Running from west to east, these rooms are:

Process Computer Room	Group 1
Fire Area 17 (Unit 1 Battery Room)	Rated Fire Barrier
Unit 1 Auxiliary Instrument Room	Group 2
Unit 1 and 2 Computer Room	Group 2
Unit 2 Auxiliary Instrument Room	Group 2
Fire Area 18 (Unit 2 Battery Room)	Rated Fire Barrier
Communication Room	Group 3
Unit 3 Computer Room	Group 3
Unit 3 Auxiliary Instrument Room	Group 3
Fire Area 19 (Unit 3 Battery Room)	Rated Fire Barrier
Mechanical Equipment Room	Group 4

The first segment of this elevation consists of the Process Computer Room only. This room is located at the west end of the elevation and is separated from other rooms on this elevation by fire area 17. This area contains no safe shutdown equipment and failure of the process computer does not result in a plant trip. Also, this area is protected by an automatic Halon suppression system and its boundaries are of 2 hour fire rated construction.

The second group of rooms on this elevation consists of the Unit 1 and 2 Auxiliary Instrument and Computer Rooms.

> The Unit 1 Auxiliary Instrument Room is on the other side of fire area 17 from the Process Computer Room. This area contains Unit 1 relay panels, with no Unit 2 safe shutdown components and is protected by a manually actuated CO_2 fire suppression system. This area is also adjacent to the Unit 1 and 2 Computer Room.

The Unit 1 and 2 Computer Room is located between the Unit 1 and the Unit 2 Auxiliary Instrument Rooms. This area contains equipment that supports the operation of Unit 2 balance of plant equipment, but does not impact the operability of ECCS equipment or its associated functions. Fire protection is provided by a manually actuated CO_2 fire suppression system.

The Unit 2 Auxiliary Instrument Room is located between the Unit 1 and 2 Computer Room and fire area 18. This area is protected by a manually actuated CO_2 fire suppression system and has dimensions of approximately 35 by 30 feet, with a floor area of approximately 1,000 square feet. The following risk-significant panels are located in this room:

2-9-18	Feedwater
2-9-29	Feedwater
2-9-30	Safety Relief Valves
2-9-32	Division I (A and C) RHR, Core Spray and HPCI
2-9-33	Division II (B and D) RHR, Core Spray and HPCI
2-9-39	HPCI Relay Auxiliary Panel
2-9-42	MSIV
2-9-43	MSIV Closure
2-9-48	Feedwater
2-9-49	Feedwater
2-9-50	Feedwater

The third group of rooms on this elevation consists of the Communication Room and the Unit 3 Auxiliary Instrument and Computer Rooms. There are no Unit 2 safe shutdown components in these areas and Unit 2 plant trip would not be expected to occur due to fires in these areas.

The Communication Room is located between fire area 18 and the Unit 3 Computer Room.

The Unit 3 Computer Room is located between the Communication Room the Unit 2 Auxiliary Instrument Room. Fire protection is provided by a manually actuated CO_2 fire suppression system.

The Unit 3 Auxiliary Instrument Room is located between the Unit 3 Computer Room and fire area 19. This area is protected by a manually actuated CO_2 fire suppression system.

The final segment of this area consists of the Mechanical Equipment Room only. This area is located on the opposite side of Fire Area 19 from the Unit 3 Auxiliary Instrument Room. This area contains various Control Building HVAC equipment. Plant trip would not be expected in response to fires in this area.

The fire ignition frequency for this area is listed in Table 4-1 as 5.892E-02. The worksheet in Attachment B for this area shows that over 95% of this ignition frequency is due to electrical cabinets. Due to the general nature of the instrumentation and controls on the panels in each of the rooms on this elevation, this fire frequency is divided evenly among the 8 rooms. Due to the limited level of combustibles present, none of this frequency is allocated to the corridor. Therefore, the fire ignition frequency for each room on this elevation is assigned as 7.36E-03.

Due to the presence of area wide detection coverage, the initial fire brigade responder is expected to be on the scene and capable of suppressing a minor fire before damage to other plant components occur. Since the fire sources in this area are predominantly due to electrical panels and transient sources, a minor fire fraction of (100% - 5.9% =) 94% is applied to this area, as described in Section 6.2.1. For purposes of this analysis, then, 5.9% (= 12/205) of all fires in this area are conservatively assumed to require Control Room evacuation if not suppressed.

The primary means of fire suppression for the Unit 1/Unit 2 Auxiliary Instrument Rooms and for the Unit 1 and 2 Computer Room is through manual actuation of an area-wide CO_2 suppression system. This manual action is assigned a failure rate of 0.1.

Following failure to suppress the fire immediately with the installed CO_2 system, manual fire suppression by the fire brigade is questioned for severe fires. Following fire brigade failure to suppress the fire, Control Room evacuation is conservatively assumed to be required due to loss of control functions. Therefore, fire brigade suppression of severe fires following failure of CO_2 actuation, and before Control Room evacuation is required, is assigned a failure rate of 0.1.

Given this information, the event tree for this area can be shown graphically as:

```
                   Location        Severity           Suppression
                                                   CO2        Fire
                                                              Brigade
5.89E-02    (=7.36E-03)      (0.94)
------------ Unit 2 ----------- Minor -------------------        Case 1 - Total Loss of Feedwater
          |  Auxiliary  |    (0.059)       (0.9)                                        (=6.92E-03)
          |  Instrument ----- Severe- ------- Yes ----          Case 2 - MSIV Closure
          |  Room                      |  (0.1)                                         (=3.91E-04)
          |                            --- No ----- Yes --      Case 3 - MSIV Closure/HPCI Failure
          |                                      |  (0.1)                               (=3.91E-05)
          |                                      -- No ---      Case 4 - Control Room Evacuation
          |  (=7.36E-03)      (0.94)                                                    (=4.34E-06)
          |-- Unit 1/2 ---------- Minor ------------------      Case 1 - Total Loss of Feedwater
          |  Computer  |     (0.059)       (0.9)                                        (=6.92E-03)
          |  Room      ------ Severe -------Yes------           Case 2 - MSIV Closure
          |                            |  (0.1)                                         (=2.65E-04)
          |                            --- No ------            Case 3 - MSIV Closure/HPCI Failure
          |  (=4.42E-02)                                                                (=4.34E-05)
         --- Other Areas--------------------------------------  Screened (No Unit 2 Trip)
                                                                                        (=4.42E-02)
                                                                                        (=5.89E-02)
```

This information was then used to generate the following cases:

Case 1 Total loss of main feedwater, following a minor fire in either the Unit 2 Aux Instrument Room or the Unit 1 and 2 Computer Room. Due to the number of panels in the Aux Instrument Room that contain feedwater controls, total loss of main feedwater was conservatively assumed to occur following any fire in either of these areas, regardless of severity.

Case 2 MSIV closure, following successful suppression of a severe fire (i.e. any fire that was not put out with portable equipment from the fire events database) in either the Unit 2 Aux Instrument Room or the Unit 1 and 2 Computer Room. Damage to control circuits, resulting in MSIV closure before manual fire suppression with the installed CO_2 suppression system, is conservatively assumed to occur for all of these fires. It should be noted that, since Browns Ferry has steam driven main feedwater pumps, this case subsumes any impacts from Case 1, above.

Case 3 MSIV closure with HPCI failure, following failure to suppress a severe fire in either the Unit 2 Aux Instrument Room or the Unit 1 and 2 Computer Room with the installed CO_2 system. This case assumes failure of control circuitry for both MSIV and HPCI operation, requiring RCIC operation to maintain high pressure injection or emergency depressurization to enable use of low pressure injection systems

Case 4 Evacuation of the Control Building is conservatively assumed to be required for all severe fires in the Unit 2 Aux Instrument Room that were not suppressed by either actuation of the installed CO_2 system or by the fire brigade. Due to the predominance of operator contributions to core damage under this situation, a conditional core damage frequency of 0.10 was assigned to this case, regardless of plant equipment that remains operable from panel 2-25-32. This value compares to the 0.074 and 0.064 values for Control Room evacuation given in NUREG/CR-4550 for Peach Bottom Unit 2 and Surry Power Station, respectively.

The remaining path for this event tree, fire in one of the 6 remaining rooms on this elevation, were screened from further consideration, primarily due to separation by rated fire barriers of fire areas 17 and 18. In other words, plant trip of Unit 2 would not be expected to occur in response to fires in these areas.

The results of this evaluation are summarized in Table 6-10, below.

Table 6-10
Evaluation of Fires on Control Building Elevation 593

Case	Description	Case Frequency (F1)	Probability of Core Damage (P2)	Core Damage Frequency (F1 x P2)
Case 1	Total Loss of Feedwater	1.38E-02	1.44E-06	1.99E-08
Case 2	MSIV Closure	7.82E-04	1.69E-06	1.32E-09
Case 3	MSIV Closure/HPCI	7.89E-05	2.33E-06	1.82E-10
Case 4	Control Building Evacuation	4.34E-06	0.10	4.34E-07
Screened	Other Areas	4.42E-02	N/A	N/A
Total		5.89E-02		4.73E-07

Since the total core damage frequency for all of these cases is less than 1E-06, this area can be screened from further consideration.

It should be noted that Case 4, which assumes that abandonment of the Control Room is required, is judged to bound the conceivable case where a fire would initiate on this elevation and then propagate through non-fire rated barriers to the Cable Spreading Room, which is located above this area.

This evaluation is conservative in that all fires in areas that could potentially impact Unit 2 operation are assumed to result in plant trip. Also, for severe fires in the Unit 2 Aux Instrument Room that are not suppressed by the fire brigade, Control Room evacuation is assumed to be required.

6.2.7 Fire Propagation from Compartment 16-1 to Compartment 16-2

As noted in Section 3.3, there is the potential for a multiple area fire that would develop on the 593 foot elevation of the Control Building and propagate to include the Cable Spreading Room, which is located above this elevation. Though the ceiling is constructed of reinforced concrete, with an equivalent fire rating of 1.5 hours, penetrations exist, through which an unsuppressed fire could conceivably spread to damage equipment in the Cable Spreading Room.

The analysis of potential fires in the Unit 2 Aux Instrument Room is shown in Attachment C (Page C.3-10). This analysis assumed that the fire engulfs up to 8 adjacent panels within the room. The hot gas layer temperature for this fire was well below the damage/ignition temperature of electrical components. Therefore, any hot gases escaping through ceiling penetrations into the Cable Spreading Room would therefore not have the potential to damage equipment in the Cable Spreading Room. This evaluation is typical of other areas on the 593 foot elevation that contain electrical panels and cabinets. It should be noted that the areas that are protected by Halon and CO_2 suppression systems have penetration seals that are not fire rated, but will provide protection against the propagation of smoke and hot gases.

In addition to the calculations shown in Attachment C, a separate evaluation of this multiple area fire was performed using two event trees. The first event tree was used to develop the frequency of fires that have the potential to breach through the unsealed penetrations in the interface between the 593 foot elevation and the Cable Spreading Rooms. The second event tree was used to develop the individual cases to be evaluated for this potential multiple area fire.

As noted above, the total fire ignition frequency for the 593' elevation (compartment 16-1) is listed as 5.892E-02. Again, this fire frequency is assumed to be divided evenly among the 8 rooms on this elevation. Therefore, each room is assigned a fire ignition frequency of 7.36E-03.

The fires that could occur in this area are categorized as minor or severe, as discussed in Section 6.2.1 and the corresponding probability factors are used. Due to the geometry of the rooms on this elevation, with potential fire propagation through an unsealed ceiling penetration, minor fires are assigned a probability factor of 0.94 and severe fires have a probability factor of 0.05. For this evaluation, minor fires are taken to represent fires that would not propagate beyond the initial room. Severe fires are then assumed to require suppression with installed systems or by the full fire brigade prior to ceiling breach. Otherwise, fire growth to the Cable Spreading Rooms is conservatively assumed to occur.

For areas with manually actuated fire suppression systems installed, this form of suppression is assigned a failure rate of 0.1. Following failure to suppress the fire immediately with an installed system, manual fire suppression by the fire brigade prior to fire growth to the Cable Spreading Room is assigned a failure rate of 0.1. Following fire brigade failure to suppress the fire, fire growth to envelop the Cable Spreading Room is, again, conservatively assumed to occur.

Given this information, the event tree for fire growth from the 593 foot elevation to the Cable Spreading Rooms can be shown graphically as:

```
                Location       Severity         Suppression
                                              CO2 or      Fire
                                              Halon      Brigade
                7.36E-03       (0.94)
5.892E-02-------- Mech ------------- Minor --------------------------------------  Contained
             |  Equipt.     |  (0.059)                      (0.90)
             |  Room        ----- Severe -------------------------Yes-----   Contained
             |                                            |  (0.10)
             |                                            ----- No ----   Propagates
             |  7.36E-03       (0.94)                                      (= 4.34E-05)
             |---Commun.--------- Minor ---------------------------------------- Contained
             |  Room        |  (0.059)                      (0.90)
             |              ----- Severe ---------------------------Yes ----  Contained
             |                                            |  (0.10)
             |                                            ----- No ----   Propagated
             |  4.42E-02       (0.94)                                      (= 4.34E-05)
             --- Other -------------- Minor --------------------------------------  Contained
                Areas       |  (0.059)           (0.90)
                            ---- Severe --------------- Yes ----------------   Contained
                                          |      (0.10)    (0.90)
                                          --------- No ----- Yes -----   Contained
                                                          |  (0.10)
                                                          ---- No -----   Propagates
                                                                           (= 2.61E-05)
                     Total Ignition Frequency                                5.89E-02
                     Total Propagation Frequency                             1.13E-04
```

Using the values noted above, this gives a frequency for fires that are assumed to propagate to the Cable Spreading Room of 1.13E-04. Once the fire has grown to affect the Cable Spreading Room, however, fire propagation will be prevented by the fire suppression system installed in the spreading room. For purposes of this evaluation, even suppressed fires are assumed to lead to a total loss of all offsite power. Following failure of spreading room suppression, Control Room Evacuation is conservatively assumed to be required. As above, this is assigned a conditional core damage frequency of 0.10.

```
                Automatic
               Suppression

                  (0.96)
1.13E-04   ------------- Yes -----    Total LOSP assumed to occur
                |                          (= 1.08E-04)
                |  (0.04)
             ------ No ------    Control Room Evacuation assumed to be required
                                           (= 4.52E-06)
```

The evaluation of each of these cases is shown in Table 6-11, below.

Table 6-11
Fire Growth from 593' to Cable Spreading Rooms

Case Description	Case Frequency (F1)	Probability of Core Damage (P2)	Core Damage Frequency (F1 x P2)
Suppressed Fire (Assumed Total LOSP)	1.08E-04	1.57E-06	1.70E-10
Unsuppressed Fire Control Building Evacuation	4.52E-06	0.10	4.52E-07
Total	1.13E-04		4.52E-07

Since the total core damage frequency for these cases is less than 1E-06, this potential multiple area fire can be screened from consideration. This evaluation remains conservative in that all medium and severe fires that occur on the 593' elevation and remain unsuppressed are assumed to breach the ceiling boundary into the Cable Spreading Room. This boundary consists of reinforced concrete, which has an equivalent fire rating of 1.5 hours, but has penetrations that may not be sealed to this fire rating equivalency. Therefore, fire propagation to the Cable Spreading Room is conservatively assumed to occur.

Also, it should be noted that unsuppressed severe fires in the Unit 2 Auxiliary Instrument Room are assumed to require Control Room abandonment (see Table 6-7, above). This case also bounds the potential multiple area fire.

6.2.8 Fire Compartment 16-2, Cable Spreading Rooms (CSR)

This area is located below the Control Rooms, at the 606 foot elevation. The Cable Spreading Room dimensions are approximately 30 by 450 feet, with a total floor area of approximately 13,000 square feet. Although the floor/ceiling interface with the 593 foot elevation (compartment 16-1) and the ceiling/floor interface with the Control Rooms (compartment 16-3) and the walls between rooms are not fire rated, these area boundaries are of substantial construction, using non-combustible materials that are equivalent to a fire rating of 1.5 hours.

This area is protected by an automatic preaction sprinkler system that utilizes closely spaced, high density design Quick Response Sprinkler (QRS) heads.

Review of the EPRI Fire Events Database (NSAC/178L) shows that there have been four Cable Spreading Room fires in the commercial nuclear industry. All of these fires occurred inside electrical cabinets, with one inside an inverter cabinet, one affecting an actuation relay and the other two affecting circuit and alarm printed circuit cards only. Unit shutdown was only indicated for one of the fires and there is no indication of plant trip in any of these instances. Due to the sparse nature of this data (4 entries in approximately 1200 years of reactor experience), this information is used only as an indication of the nature of fires in the Cable Spreading Room and as an indication of the level of conservatism introduced by assuming component damage and plant trip for all fires in this area.

It should be noted that Browns Ferry has had a significant fire that developed from the Cable Spreading Room. Within the Fire Events Database, this event was assigned as a transient fire source. While the polyurethane that was used for penetration seals at the time of the fire has been removed from consideration as a fire source, the impact of this fire is conservatively evaluated for this analysis as Case 2, which is described below. For further description of the fire itself, see "Cable Fire At Browns Ferry Nuclear Power Plant" in the July 1976 issue of Fire Journal.

Since this area has fire detectors installed to provide area wide coverage, the initial fire brigade responder is expected to be on the scene and capable of suppressing a minor fire before damage to other plant components occur. Since the fire sources in this area are predominantly transient fire sources, a minor fire fraction of (100% - 5.9% =) 94% is applied to fires in this area, as described in Section 6.2.1. For purposes of this analysis, then, 5.9% (= 12/205) of all fires in this area are conservatively assumed to require Control Room evacuation if not suppressed.

The fire ignition frequency calculated for this area in Attachment B is 1.344E-02 and the preaction sprinkler failure rate (from the EPRI documentation) of 0.05. Following failure of automatic suppression for severe fires, manual suppression (through use of hose streams or other response equipment) prior to Control Room evacuation is assigned a failure rate of 0.10. Given this information, the event tree for this area can be graphically shown as:

```
     Severity              Suppression
                     Automatic      Fire
                    (Sprinklers)   Brigade

1.344E-02   (0.94)
------------ Minor ---------------------------------   Case 1 - Total Loss of Feedwater
     |      (0.059)       (0.95)                                            (=1.26E-02)
     ---------- Major ---------- Yes -----------------   Case 2 - MSIV closure with
                          |  (0.05)    (0.9)                      failure of HPCI/
                          ------- No ------- Yes ----   Case 2 -        RCIC/CS
                                        |  (0.1)                               (=7.87E-04)
                                        ----- No ----   Case 3 - Control Room Evacuation
                                                                               (=3.95E-06)
                                                                               (=1.34E-02)
```

This information can be used to generate the following cases:

Case 1 Fires that are contained to a single cable tray (i.e. minor fires, capable of being suppressed with portable extinguishers). Due to the plant component damage that is conservatively assumed to occur for all fires in this area, this is modeled as a total loss of feedwater, which requires successful actuation of HPCI or RCIC to maintain high pressure RPV injection. The Sandia studies shown in NUREG/CR-5384 (SAND89-1359) list an ignition time of 12 minutes for fire retardant coated cables following test fires in a lower cable tray using diesel fuel or natural gas burners. This time is adequate to ensure an initial manual response with portable fire extinguishers prior to fire growth to include a second cable tray for minor fires, as described above. This case fails the primary means of high pressure injection, questioning HPCI and RCIC operation to maintain RPV water level.

Case 2 Fire growth to include a second cable tray, following a severe fire with successful suppression by either the installed automatic preaction system or by the fire brigade. This is modeled as an MSIV closure with failure of HPCI, RCIC and low pressure ECCS injection with core spray. It should be noted that this case assumes failure of all high pressure injection sources, except for control rod drive hydraulics, in addition to failing all low pressure injection sources except main condensate and RHR. This set of impacts was selected because it models the Unit 1 control functions that were eventually lost during the cable fire that occurred at Browns Ferry on March 22, 1975.

Case 3 Fire growth to include more than two cable trays. This condition is assumed to occur for all fires that were suppressed with installed systems or by hose streams (i.e. more severe fires), but for which suppression by the automatic system and by the fire brigade was unsuccessful.

It should be noted that, for most fire retardants tested in the Sandia studies, there was no ignition of the upper trays. Following fire brigade failure to suppress the fire at an assigned frequency of 0.10 (i.e. the fire brigade fails to suppress 10% of these fires), this case is modeled as requiring evacuation of the Control Building. Due to the predominance of operator contributions to core damage under this situation, a conditional core damage frequency of 0.10 was assigned, regardless of the control functions that remain operable from panel 2-25-32 and other locations within the plant itself. This value compares to the 0.074 and 0.064 values for Control Room evacuation given in NUREG/CR-4550 for Peach Bottom Unit 2 and Surry Power Station, respectively.

The evaluation of each of these cases is shown in Table 6-12, below.

Table 6-12
Evaluation of Cable Spreading Room Fires

Case	Description	Case Frequency (F1)	Probability of Core Damage (P2)	Core Damage Frequency (F1 x P2)
Case 1	Total Loss of Feedwater	1.26E-02	1.44E-06	1.81E-08
Case 2	MSIV Closure with HPCI, RCIC and CS Failure	7.87E-04	4.39E-05	3.45E-08
Case 3	Control Building Evacuation	3.95E-06	0.10	3.95E-07
Total		1.34E-02		4.48E-07

Since the total core damage frequency for all of these cases is less than 1E-06, this area can be screened from further consideration.

It should be noted that Case 3, which assumes that abandonment of the Control Room is required, is judged to bound the conceivable case where a fire would initiate in the Cable Spreading Room and then propagate through non-fire rated barriers to the Control Room or to the 593 foot elevation of the Control Building fire area.

This evaluation remains conservative in that a total loss of main feedwater is assumed to occur for any and all fires in this area. Following successful manual or automatic suppression of a severe fire, the fire is conservatively assumed to approximate the severe fire from March, 1975, despite the plant changes that have been incorporated since then and the fact that this fire burned for more than seven hours before finally being suppressed. Following failure of manual suppression, the actions of the fire brigade to suppress the fire prior to Control Room evacuation are conservatively not modeled. Finally, Control Room evacuation is assumed to be required for any severe, unsuppressed fires in this area.

6.2.9 Fire Compartment 16-3, Control Rooms

Compartment 16-3 is a large common area that runs approximately 450 feet along the length of the Control Building. The Unit 1 and Unit 2 Control Rooms share a common area and are separated from the Unit 3 Control Room by the Relay Room and the Technical Support Center, which has automatic sprinkler fire suppression installed. Therefore, fire propagation from the Unit 3 Control Room into the Unit 1/Unit 2 Control Room area is not judged to be feasible.

The Unit 2 control area is laid out in a "U" shape, with the main generator and other associated controls located immediately to the left of the entrance. Following the control panel sections around to the right, the other balance of plant and main feedwater controls are located on panel 2-9-8, to the left of the main core map area. To the right of the core map area is panel 2-9-3, which contains the following controls, looking from left to right:

MSIV Controls
Primary Containment Isolation

RCIC
SRV Actuation/ADS
Division I Core Spray
Division I RHR

Division II RHR
Division II Core Spray
HPCI

Internal barriers exist between the panel section that controls primary containment isolation and RCIC and the panel section that controls Division I and Division II ECCS functions.

During reviews of the boundaries between this area and the Cable Spreading Rooms (located below), it was determined that a fire is unlikely to propagate through the Control Room floor into the Cable Spreading room. This is discussed in Section 6.3.1.

In general, if a fire occurs in the Unit 1 Control Room area, this will have no impact on Unit 2 operation. In the case of a severely involved fire, though, the Unit 2 Control Room may eventually have to be evacuated due to smoke and other products of combustion.

Review of industry experience reveals that there have only been 12 Control Room fires. Of these, only 2 resulted in plant trip (one due to an electrical fault in the main turbine EHC system and a second due to manual reactor trip from hot shutdown during control rod testing upon actuation of a single fire alarm in the relay room). Both of these events occurred in 1985, at different plants. This review also showed that, of the 7 Control Room fire entries that specify fire duration and suppression times, only 2 burned for more than 2 minutes. Both of these fires were put out within 5 minutes. Of these fires, 5 were put out with portable extinguishers, while the other 2 did not require suppression before burning out on their own. This review provides the following general indications:

> Only 1 in 6 Control Room fires is expected to be severe enough to warrant a plant trip. This includes one entry where the operator tripped the unit based on indication of a fire alarm only.
>
> In general, Control Room fires do not burn as long or as hot as the Sandia studies indicate would be required (540 seconds, or 9 minutes, to obscure vision at eye level due to smoke, from Sandia test 23 results) to force Control Room evacuation. Also, the Sandia test assumes a much more severe fire than industry experience has ever recorded for the Control Room. The short durations for Control Room fires that have been seen in the industry may also be due to the continuous presence of personnel. The Sandia test also assumes that the fire will not be detected until 30 seconds after ignition. While this response time may be indicative of installed fire detection systems installed, only 2 of the 8 Fire Events Database (NSAC/178L) Control Room fire entries that identify the means of detection list fire detection equipment. The remaining 6 entries identify plant personnel as the means of fire detection.

While the industry data indicates that a large share of Control Room fires do not even warrant plant trip, this experience is of such a sparse nature (12 Control Room fires in more than 1200 reactor years of experience) that it is used to provide an indication of the level of conservatism in this evaluation, rather than as a basis for quantitative analysis.

The evaluation of fires in the Control Room area is based on the guidance given in Appendix J of the EPRI Fire Risk Analysis Implementation Guide (draft report, January 1994).

In essence, this process consists of evaluating the Control Room panels for any potential impact on plant operation due to loss of controls and a review of plant systems for those that would not continue to operate following an assumed Control Room abandonment beginning 15 minutes after the fire and lasting until 60 minutes after fire ignition for unsuppressed fires. Fire suppression is credited by the operating shift personnel using values from the Human Cognitive Reliability (HCR) correlation with a best estimate of 3.4E-03.

As noted above, the most significant Control Room panel, from the aspect of potential impact on plant operation, is panel 2-9-3. A fire in this panel can induce plant trip through either MSIV closure or through inadvertent SRV operation. Also, there is the potential for a fire in this panel to fail high pressure injection and the operator's ability to depressurize the plant, though such a fire would have to breach two sets of panel section boundaries. Based on this evaluation, the analysis of Control Room fires will center on the impacts of fires in panel 2-9-3. It should be noted that fires in this panel automatically subsume those other fires that could result in plant trip or total loss of main feedwater, through MSIV closure.

The total fire ignition frequency for all three plant control room areas was calculated in Attachment B as 3.534E-02. By dividing this fire frequency evenly among the three Control Rooms, a frequency of 1.178E-02 per Control Room is generated. Manual fire suppression by authorized personnel, using available fire extinguishers, before a fire spreads outside the initial panel of concern, is assigned a failure rate of 3.4E-03, as noted above.

Graphically, the event tree for this area can be shown as:

	Location	Manual Suppression	
3.534E-02	1.18E-02 Unit 1	Yes (0.997)	Screened (no Unit 2 impact) (=1.18E-02)
		No (3.4E-03)	Case 1 - Unit 2 Abandoned, no fire impacts (=4.01E-05)
	1.18E-02 Unit 2	Yes (0.997)	Case 2 - MSIV Closure/RCIC Failure (=1.18E-02)
		No (3.4E-03)	Case 3 - Unit 2 evacuation with fire impacts (=4.01E-05)
	1.18E-02 Unit 3		Screened (no Unit 2 impact) (=1.18E-02)
			(=3.55E-02)

This information can be used to generate the following cases:

Case 1 Unit 2 abandonment due to smoke from an unsuppressed fire in the Unit 1 control area. While this case has no fire-induced failures, long term operation of main feedwater and HPCI is conservatively assumed to fail prior to Control Room re-entry. Manual control of RCIC injection is available from the remote shutdown panel, as directed by the Control Room Abandonment procedure (2-AOI-100-2). This form of recovery is conservatively not modeled.

Case 2 MSIV closure with RCIC failure, following fire suppression of a fire in the Unit 2 control area. Damage to plant components is assumed to occur, resulting in loss of all main feedwater (due to loss of steam flow), which fails the primary means of high pressure injection. This sequence of events then questions HPCI operation to maintain RPV water level.

Case 3 Unit 2 abandonment with fire induced failures (MSIV closure with RCIC failure and stuck open SRV), following failure of manual suppression for any fire in the Unit 2 control area. In this case, the assumed damage results in loss of main feedwater and high pressure injection with RCIC (as in Case 2, above) and induces a stuck open SRV condition through an assumed hot short, similar to an induced small break LOCA condition. Recovery from the stuck open SRV (through de-energizing the associated circuit from the respective unit battery or 250V RMOV board) is conservatively not modeled. Due to the assumed failure of only one SRV in an open condition, RPV depressurization is conservatively required prior to injection with low pressure sources. HPCI failure is assumed to occur during the evacuation period, prior to Control Room entry, as described in Case 1, above.

The evaluation of each of these cases is shown in Table 6-13, below.

Table 6-13 Evaluation of Control Room Fires				
Case	Description	Case Frequency (F1)	Probability of Core Damage (P2)	Core Damage Frequency (F1 x P2)
Case 1	MSIV Closure	4.01E-05	4.39E-05	1.76E-09
Case 2	MSIV Closure/RCIC	1.18E-04	2.35E-06	2.77E-08
Case 3	Control Room Evacuation with MSIV Closure, RCIC failure and Stuck Open SRV	4.01E-05	1.88E-04	7.54E-09
Suppressed Unit 1 Fire		1.18E-02	N/A	N/A
Unit 3 Control Room Fire		1.18E-02	N/A	N/A
Total		3.55E-02		3.70E-08

Since the total core damage frequency for all of these cases is less than 1E-06, this area can be screened from further consideration.

It should be noted that Case 3, which assumes that Control Room evacuation is required, is judged to bound the conceivable case where a fire would initiate in the Control Room area and then propagate through non-fire rated barriers to the Cable Spreading Rooms, which are located below this elevation.

This evaluation remains conservative in that all Unit 2 Control Room fires are assumed to result in component damage and plant trip, where industry experience has shown that few Control Room fires result in plant trip at all. Also, Control Room evacuation is assumed to be required for any unsuppressed fires in this area, including those that occur in the Unit 1 Control Room area.

6.2.10 Fire Area 18 - Unit 2 Battery and Battery Board Rooms

Review of the initial analysis of this area shown in Section 5.1.18 showed that the upper bound core damage frequency for this area was slightly over the screening criteria of 1E-06. Based on the review of fire severity shown in the Fire Events Database see Section 6.2.1, approximately 93% of the fires that would be expected to occur in this type of area would be put out with portable equipment or allowed to burn out. This response would be carried out by the first responder on the scene from the fire brigade, who would be expected to arrive shortly after initial detection of the fire.

For the remaining fires (7.3% of total ignition frequency), all fires are modeled as becoming engulfing, similar to the analysis shown in Section 5.1.18.

The revised evaluation of this area therefore consists of evaluating 2 cases:

Case 1 A minor fire starts in the battery or battery board rooms. This fire is then either suppressed by the first responder on the scene or is allowed to burn out. The Unit 2 battery is assumed to fail prior to fire suppression. As noted above, this case is assigned 93% of total area fire ignition frequency or (2.09E-02 x 0.93 =) 1.94E-02.

Case 2 A fire starts anywhere in fire area 18 and is eventually suppressed with hose streams. This fire is conservatively assumed to spread to envelop and damage all components in the area, similar to the evaluation shown in 5.1.18. This case is assigned 7.3% of total area fire frequency, as described in Section 6.2.1, above, or (2.09E-02 x 0.073 =) 1.53E-03

The evaluation of each of these cases is shown in Table 6-14, below.

Table 6-14 Evaluation of Fires in Fire Area 18				
Case	Description	Case Frequency (F1)	Probability of Core Damage (P2)	Core Damage Frequency (F1 x P2)
Case 1	Minor fire, suppressed	1.94E-02	2.39E-05	4.64E-07
Case 2	Significant fire, assumed to become engulfing	1.53E-03	5.79E-05	8.86E-08
Total		2.09E-02		5.53E-07

Since the total core damage frequency for all three of these cases is less than 1E-06, fires in this area can be screened from further consideration. This evaluation is judged to remain conservative in that all fires, regardless of size or location, are assumed to result in plant trip, with significant damage to plant components. Also, no credit is taken for fire suppression, beyond the response of the initial fire brigade member on the scene.

6.2.11 Compartment 25-1 - Intake Pump Station

A large share (80%) of the fire frequency for this area is due to fires in electric cabinets, fire pumps and in other pumps (primarily circulating water, EECW and RHR service water). Two cases are analyzed for this area. A second case was necessary to evaluate a particular set of cable routings near the interface with the cable tunnel area, in the northwest corner of the pump station. This case is discussed in detail below.

Plant trip would not be expected to occur for a fire in the EECW or RHR service water pump rooms, even if more than one of these pumps could be affected by a fire. Fire in a condenser circulating water pump area, particularly if the fire was severe enough to affect an adjacent pump, could, however, result in a plant trip due to loss of condenser vacuum.

Fires in this compartment were therefore evaluated by conservatively assuming that all fires lead to a plant trip on low condenser vacuum. This gives a fire-related core damage frequency for this case of:

$$F2_{Case\ 1} = 3.58E\text{-}02 \times 1.47E\text{-}06 = \mathbf{5.26E\text{-}08}$$

As noted above, plant area walkdowns identified a particular cable arrangement near the interface with the cable tunnel, in which the power cables for RHR service water pumps powered by division 1 and by division 2 power are routed approximately 6 feet from each other. In the event of an unsuppressed, severe fire, it is conceivable that such a fire could result in a loss of all RHR service water and also damage Unit 2 condenser circulating water pump cables, resulting in a loss of both primary means of removing decay heat from the plant through failure of the ultimate heat sink.

This case is evaluated by reviewing the appropriate ignition sources for this area from Attachment B. This information, when taken with the area layout information, reveals that there are no ignition sources within approximately 20 to 30 feet of this area, except for cable and junction box ignition sources.

Cable and junction box ignition sources are assigned a total ignition frequency for this area of 5.352E-04. Due to the potential severity of this case, 10% of this frequency is arbitrarily assigned to this case.

Transient fire sources for this area have a total ignition frequency of 3.988E-03. Due to the nature of transient sources, only those occurring within approximately 20 feet of the area of concern are judged to have the potential to apply to this case. Due to the geometry of this area, this equates to approximately 400 square feet. Since this section of the intake structure has a floor area of approximately 360 feet by 50 feet, or 18,000 ft^2, only transient ignition sources occurring over about (400/18,000 =) 2.22% of the floor area would potentially apply to this case.

This is conservative in that it only considers the adjacent floor area, and not the other elevations of this structure or the RHR service water pump area, as a total effective floor area for transient ignition sources.

The total potential ignition frequency for this case can then be calculated as:

$$(5.352E\text{-}04 \times 10\%) + (3.988E\text{-}03 \times 2.22\%) = 1.42E\text{-}04$$

This location is protected by a preaction fire suppression system, which, from the EPRI FIVE documentation, has an effective failure rate of 0.05. Finally, only severe fires (i.e. those put out by hose streams or installed systems), as described in Section 6.2.1, are judged to have the potential to develop to the size required to threaten both trains of RHR service water. Therefore, a severity factor of 0.059 is applied. The total fire frequency for this case can now be calculated as:

$$F1_{Case\ 2} = 1.42E\text{-}04 \times 0.059 \times 0.05 = \mathbf{4.19E\text{-}07}$$

Due to the potential severity of this case (i.e. threatening all available means of decay heat removal, except for the containment vent), a conditional core damage frequency of 1.0 is applied to this case.

Total fire-related core damage frequency for the intake pump station can therefore be estimated as no higher than:

F2 = 5.26E-08 + 4.19E-07 = **4.72E-07**

Since the total value for both of these cases is less than 1E-06, this area can be screened from further consideration. This evaluation remains conservative in that all fires in this area are assumed to result in a plant trip with loss of main condenser vacuum, whereas a significant fraction of fires in this area would not be expected to result in plant trip at all.

6.2.12 Fire Compartment 25-2 - Pipe Tunnel

Due to a lack of plant components in this area, plant trip due to fires in this area would not be expected. Also, the area has an extremely low fire frequency (9.875E-06), primarily due a small number of cables that transit through the area.

Since a plant trip would not be expected following any fire in this area, the area can be conservatively evaluated by assuming a manual reactor trip for all fires in this area. This gives a fire-related core damage frequency of

F2 = 9.875E-06 x 3.15E-07 = **<1E-10**

Since this value is less than 1E-06, this area can be screened from further consideration. This evaluation remains conservative in that all fires in this area are assumed to result in a plant trip, whereas a significant fraction of fires in this area would not be expected to result in plant trip at all.

6.2.13 Fire Compartment 25-3 - Turbine Building

The Turbine Building itself has the highest fire frequency of all areas at the Browns Ferry site, at 4.50E-01 fires per year. This is primarily due to fires in the Hydrogen recombiners, which contribute a fire ignition frequency of 2.58E-01.

While a recombiner fire may result in a plant trip for the affected unit, this would be similar to a turbine or reactor trip, as opposed to a loss of condenser vacuum, loss of offsite power or loss of feedwater transient. The recombiner components are located in the Turbine Building basement, in individual compartments for each unit.

Access to each of these compartments is through a set of offset doorways, preventing fire growth from one area to another. While these boundaries are not fire rated, they are of substantial commercial construction, consisting of reinforced concrete, except for the recombiner tube removal area, which consists of a concrete block wall. Since a fire in the Unit 1 or Unit 3 recombiners would not be expected to impact operation of Unit 2, a Unit 2 turbine trip is assumed to result from 1 out of every 3 Hydrogen recombiner fires.

Turbine generator lube oil fires have a total fire ignition frequency of 3.90E-02 and could be expected to lead to a plant trip for all fires at Unit 2 or any unsuppressed fires at Unit 1 or Unit 3. These areas are supplied with deluge water spray systems, which have a failure rate, from the EPRI FIVE documentation, of 0.05. Since turbine generator oil fires contribute a total of 3.90E-02 to fire ignition frequency for this area, unsuppressed fires would have a frequency of (3.90E-02 x 0.05 =) 1.95E-03. These unsuppressed fires are evaluated with the total loss of offsite power (LOSP) conditional core damage frequency, even though manual fire suppression has not been credited and the offsite feed lines for the 3 units are widely spaced (approximately 100 to 150 feet between adjacent units).

Of the remaining lube oil fire ignition frequency, 2.48E-02 is due to Unit 1 or Unit 3 fires which are screened out, and 1.22E-02 is due to suppressed oil fires, which are modeled as leading to a trip of Unit 2. These are evaluated, again, as resulting in a turbine trip.

All fires on the turbine operating deck, regardless of unit, are expected to result in a turbine trip for Unit 2, if only as a precaution. These fires have a frequency of (1.20E-2 + 1.65E-02 =) 2.85E-02, from turbine generator exciter and hydrogen sources, respectively.

Of the remaining Turbine Building fire ignition frequency (1.25E-01), only cases where the fire is unsuppressed are expected to significantly impact the plant availability of equipment. These events are modeled as leading to a total loss of offsite power. Otherwise, suppressed fires at Unit 1 or Unit 3 would not be expected to lead to trip of Unit 2. Therefore, using a manual suppression factor of 0.1, total loss of offsite power is modeled as occurring due to fires in the Turbine Building at a frequency of (1.25E-01 x 0.1 =) 1.25E-02.

Graphically, the event tree for this area can be shown as follows:

```
        Location              Suppression      Unit

4.50E-01   (2.58E-01)                          (0.67)
  ---------- Recombiners ----------------------- Unit 1/3 ----    Screened (No Unit 2 Trip)
     |                                         | (0.33)                                      (=1.73E-01)
     |                                         -- Unit 2  ----    Turbine Trip (Case 1)
     |      (3.90E-02)        (0.95)           (0.67)                                        (=8.51E-02)
     |------- Lube Oil ----- Suppressed ----- Unit 1/3 ---        Screened (No Unit 2 Trip)
     |                   |                     | (0.33)                                      (=2.48E-02)
     |                   |                     -- Unit 2  ----    Turbine Trip (Case 2B)
     |                   |    (0.05)                                                         (=1.22E-02)
     |                   -- Unsuppressed   ---------------        LOSP (Case 2A)
     |      (2.85E-02)                                                                       (=1.95E-03)
     |---- Turbine Deck -------------------------------------     Turbine Trip (Case 3)
     |      (1.25E-01)        (0.9)            (0.67)                                        (=2.85E-02)
     ----- Other Areas ---- Suppressed ----- Unit 1/3 ---         Screened (No Unit 2 Trip)
                         |                     | (0.33)                                      (=7.54E-02)
                         |                     -- Unit 2   ---    Turbine Trip (Case 4B)
                         |    (0.1)                                                          (=3.71E-02)
                         -- Unsuppressed   ---------------        LOSP (Case 4A)
                                                                                             (=1.25E-02)
                                                                                             (=4.50E-01)
```

The total fire-related core damage frequency for the Turbine Building can then be evaluated as shown in Table 6-15, below.

Table 6-15 Evaluation of Turbine Building Fires				
Case	Raw Ignition Frequency	Adjusted Frequency (F1)	Probability of Core Damage (P2)	Core Damage Frequency (F1 x P2)
Case 1 - Unit 2 recombiner	2.58E-01	8.51E-02	3.78E-07	3.22E-08
Case 2A - Unsuppressed lube oil fire	3.90E-02	1.95E-03	4.63E-05	9.03E-08
Case 2B - Suppressed lube oil fire at Unit 2		1.22E-02	3.78E-07	4.61E-09
Case 3 - Turbine deck fire	2.85E-02	2.85E-02	3.78E-07	1.08E-08
Case 4A - LOSP due to unsuppressed fire	1.25E-01	1.25E-02	4.63E-05	5.78E-07
Case 4B - Turbine trip due to other fires		3.71E-02	3.78E-07	1.40E-08
Total	4.50E-01	1.77E-01		7.30E-07

Since the total core damage frequency for all cases is less than 1E-06, this area can be screened from further consideration.

It can also be noted, from comparison of the values in the "Raw" and the "Adjusted" fire ignition frequency columns in Table 6-15, that only about 40% of all Turbine Building fires result in a plant trip of Unit 2. Of the cases evaluated, unsuppressed fires (cases 2A and 4A, which are assumed to lead to a total loss of offsite power) contribute approximately 91% of the total core damage frequency for this area.

6.2.14 Yard Area Fires

The EPRI FIVE documentation gives a separate fire ignition frequency for yard area fires, which are dominated by catastrophic failure of main transformers. Due to the potential for fire growth to the Turbine Building or initiation of a loss of all offsite power, these fires are separately considered.

The 3 cases of this type of fire described in the FIVE documentation are:

1. Yard fire propagating to the Turbine Building. The fire ignition frequency for a single unit plant is given as 4.0E-03, so this is adjusted to 1.20E-02 for Browns Ferry. This form of fire is modeled as an unsuppressed lube oil fire (see Section 6.2.10, above), which is modeled as resulting in a total loss of offsite power.

2. Yard fire resulting in a loss of offsite power. The fire ignition frequency given for this type of fire at a single unit station is 1.6E-03. This value is therefore adjusted to 4.80E-03 for Browns Ferry. As indicated, this is modeled as a total loss of all offsite power.

3. Yard - other. This category is used to model those events, primarily main transformer failures, that do not result in a loss of offsite power. The fire ignition frequency given for this type of fire at a single station is 1.5E-02, so this is adjusted to 4.50E-02 for Browns Ferry. Since no material degradation beyond main transformer failure is indicated, this is modeled as a turbine trip.

The total fire-related core damage frequency for yard area fires can then be evaluated as shown in Table 6-16, below.

Table 6-16
Evaluation of Yard Area Fires

Case	Frequency (F1)	Probability of Core Damage (P2)	Core Damage Frequency (F1 x P2)
1. Propagates to Turbine Bldg	1.20E-02	4.63E-05	5.56E-07
2. LOSP	4.80E-03	4.63E-05	2.22E-07
3. Other Yard Area Fires	4.50E-02	3.78E-07	1.70E-08
Total	6.18E-02		7.95E-07

Since the total core damage frequency for all cases is less than 1E-06, yard area fires can be screened from further consideration.

		TRANSIENT COMBUSTIBLE EXPOSURE				transient combustible
		Reactor Building All. fire zones				
		Scenario Descriptio	Trash bag located close to a fixed combustible source			
			SCREENING CRITERIA			
IGNITION SOURCES	HEAT RELEASE RATE (q) BTU/SEC (3)	LOCATION FACTOR (1, 2 OR 4)	EFFECTIVE HEAT RELEASE RATE (qeff) BTU/SEC	COMBUSTIBLE LOAD BTU (4)	DAMAGE HEIGHT (H) FT (1)	CRITICAL RADIAL DISTANCE FT (2)
32 gallon Trash Fire	380	1	380	NA	11.07	4.92
		(away from wall)				
	Notes:	1. Critical Heat flux (Q"crit) = 0.5 btu/sec/ft2 (for non-qualified cabl;es)				
		2. Damage temperature = 425 F (For non-qualified cable)				
		3. Rcrit = [q * 0.4 / 4 * pi * Q"crit]^1/2				
		4. H = [340 * qeff^2/3 / Tcrit]^3/5 (Tcrit = 425 -100 = 325 F)				
		5. Ambient Temperature = 100F				
		6. HRR = 380 Btu/sec (Reference 1, Figure 4)				
		AUTOMATIC SPRINKLER SYSTEM EVALUATION				
TIME FOR CRITICAL DAMAGE (SEC)				TIME FOR SPRINKLER SYSTEM ACTUATION (SEC)		
IGNITION SOURCE:	Trash Bag	Target: Elect. Cabinet				
TARGET THERMAL RESPONSE PARAMETER		30	electrical cabinet	SPRINKLER TEMP. RATING (Tsprink)		200
PEAK HEAT RELEASE RATE (q)		380		RADIAL DISTANCE SOURCE TO SPRINKLER (r)		6
EFFECTIVE HEAT RELEASE RATE (qeff)		380		DISTANCE FROM SOURCE TO CEILING (H)		26
RADIANT HEAT RELEASE RATE (qrad)		152	q * 0.4	SPRINKLER TIME CONSTANT		100
LINE OF SIGHT DISTANCE FROM SOURCE (R)		5		GAS TEMP. RISE AT CEILING (DTgas,plume)'		79
RADIAL DISTANCE (r)		5		LESSER OF DTgas' OR 1600 (DTgas,plume)		79
TARGET HEIGHT (Z)		0		GAS TEMP. RISE AT SPRINKLER (DTgas,sprink)		63
RADIANT HEAT FLUX AT TARGET (Qrtarget)		0.48	qrad/4*3.14*R^2	SPRINKLER TEMP RISE/GAS TEMP RISE		1.59
CONVECTIVE HEAT FLUX IN PLUME (Qcplume)		NA*	0.3*qeff/z^2	DIMENSIONLESS ACTUATION TIME OF SPRINK.		NA
TOTAL HEAT FLUX (Qtotal)		0.48	Qrtarget + Qctarget	ESTIMATED SPRINKLER ACTUATION TIME (SEC)		Sprinkler does not
TIME TO CRITICAL DAMAGE (SEC)		3015	(3.14/4)(TRP/Qtotal)^2			activate
PROBABILITY OF FIRE PLUME EXPOSURE DUE TO TRANSIENT COMBUSTIBLE SOURCE :						
Ptc = Pfst * u * p * w						
Pfst =	1.000					
u =	0.018	(from Figure 6.3)				
w =	0.038	"				
p =	0.100	"				
Ptc (plume effects) =	6.840E-05					
Total transient fire frequency	2.260E-02	(Section 6.1.1)		* Target not in plume, ceiling jet or hot gas layer		
Probability of target damage	1.55E-06			Targets located in plume are assumed to be damaged prior to sprinkler actua		
				and are therefore not evaluated.		
			FIGURE 6 - 1			

		TRANSIENT COMBUSTIBLE EXPOSURE				transient combustible-1
		Reactor Building All. fire zones				
		Scenario Descriptio	5 gallon lube oil located close to a fixed combustible source			
			SCREENING CRITERIA			
IGNITION SOURCES	HEAT RELEASE RATE (q) BTU/SEC (3)	LOCATION FACTOR (1, 2 OR 4)	EFFECTIVE HEAT RELEASE RATE (qeff) BTU/SEC	COMBUSTIBLE LOAD BTU (4)	DAMAGE HEIGHT (H) FT (1)	CRITICAL RADIAL DISTANCE FT (2)
5 gallon lube oil drum	250	1	250	NA	9.36	3.99
Exposed surface area = 2.25 ft2		(away from wall)				
	Notes:	1. Critical Heat flux (Q"crit) = 0.5 btu/sec/ft2 (for non-qualified cabl;es)				
		2. Damage temperature = 425 F (For non-qualified cable)				
		3. Rcrit = [q * 0.4 / 4 * pi * Q"crit]^1/2				
		4. H = [340 * qeff^2/3 / Tcrit]^3/5 (Tcrit = 425 -100 = 325 F)				
		5. Ambient Temperature = 100F				
		6. HRR = 110 btu/sec/ft2 * 2.25 ft2 = 250 btu/sec (Reference 1)				

		AUTOMATIC SPRINKLER SYSTEM EVALUATION			
TIME FOR CRITICAL DAMAGE (SEC)				TIME FOR SPRINKLER SYSTEM ACTUATION (SEC)	
IGNITION SOURCE:	Lube Oil	Target: Elect. Cabinet			
TARGET THERMAL RESPONSE PARAMETER		30	electrical cabinet	SPRINKLER TEMP. RATING (Tsprink)	200
PEAK HEAT RELEASE RATE (q)		250		RADIAL DISTANCE SOURCE TO SPRINKLER (r)	6
EFFECTIVE HEAT RELEASE RATE (qeff)		250		DISTANCE FROM SOURCE TO CEILING (H)	26
RADIANT HEAT RELEASE RATE (qrad)		100	q * 0.4	SPRINKLER TIME CONSTANT	100
LINE OF SIGHT DISTANCE FROM SOURCE (R)		5		GAS TEMP. RISE AT CEILING (DTgas,plume)'	60
RADIAL DISTANCE (r)		5		LESSER OF DTgas' OR 1600 (DTgas,plume)	60
TARGET HEIGHT (Z)		0		GAS TEMP. RISE AT SPRINKLER (DTgas,sprink)	48
RADIANT HEAT FLUX AT TARGET (Qrtarget)		0.32	qrad/4*3.14*R^2	SPRINKLER TEMP RISE/GAS TEMP RISE	2.09
CONVECTIVE HEAT FLUX IN PLUME (Qcplume)		NA*	0.3*qeff/z^2	DIMENSIONLESS ACTUATION TIME OF SPRINK.	NA
TOTAL HEAT FLUX (Qtotal)		0.32	Qrtarget + Qctarget	ESTIMATED SPRINKLER ACTUATION TIME (SEC)	Sprinkler does not activate
TIME TO CRITICAL DAMAGE (SEC)		6966	(3.14/4)(TRP/Qtotal)^2		

PROBABILITY OF RADIANT ENERGY EXPOSURE DUE TO TRANSIENT COMBUSTIBLE SOURCE :				
Ptc= Pfst * u * p * w				
Pfst =	1.000			
u =	0.029	(from attached worksheets)		
w =	0.038	"		
p =	0.100	"		
Ptc (radiant effects) =	1.098E-04	(Section 6.1.1)		* Target not in plume, ceiling jet or hot gas layer
Total transient fire frequency	2.260E-02			Targets located in plume are assumed to be damaged prior to sprinkler actua
Probability of target damage	2.48E-06			and are therefore not evaluated.

FIGURE 6 - 2

Plant	BFN	**By**	
Compartment	Unit 2 Reactor Building	**Date**	
Scenario	Targets within plume region of 32 gallon trash bag fire	**Page**	

PROBABILITY OF TRANSIENT COMBUSTIBLE FIRE EXPOSURE, P_{tc}

Note: fixed ignition sources should be evaluated before performing these calculations. If a fixed ignition source is shown to be able to cause damage, and there is no automatic suppression in the area, and manual suppression is not credited, the probability of critical combustible loading (P_{ccl}) for this compartment is = 1.0, and there is no need to consider P_{tc}. Next, an attempt should be made to screen the compartment with *u, p,* and/or *w* conservatively set to 1.0. If unsucessful, calculate one or more of the following to screen the compartment. Additional guidance can be found in Section 6.3.7, Steps 3.5 - 3.8 "Fire-Induced Vulnerabiltiy Evaluation (FIVE) (EPRI TR-100370, April 1992).

CALCULATE *p*, the probability of combustibles being exposed.

P can be assumed equal to 10% if the plant transient combustible control program has features similar to the following:

- Flammable and combustible liquids in the compartment are stored in approved containers;
- Ordinary combustibles or WRP clothing are stored in enclosed metal cabinets or metal containers with fusible link actuated covers or FM approved self-extinguishing lids;
- All exposed transient combustibles used by plant personnel are removed upon completion of the work unless otherwise approved.

Otherwise, a plant specific value can be obtained by performing a walkdown of significant areas of the plant to determine the following:

1.	Number of instances where plant combustibles are found exposed	P_1	$P_1 =$
2.	Number of instances where plant combustibles are found not exposed	P_2	$P_2 =$
3.	Calculate p	$P = \frac{P_1}{P_1 + P_2}$	$P =$ 0.1

CALCULATE *u*, the probability of transient combustibles being located in range of the target (i.e., area ratio).

1.	Determine the surface area of targets facing the floor (e.g. tray width times length)	$A_s =$ ___ x ___	$A_s =$ 1000
2.	Determine the critical separation distance (from radiant energy exposure using Worksheet 3 or CIIIe).	A_{sr} from Worksheet 3 or CIIIe	$A_{sr} =$ N/A
3.	Determine the net area of floor space where combustibles could be stored in the compartment (total floor area minus 20% equipment area)	Total area, ___ * .8	Net Area = 56,000
4.	Calculate *u*	$u = \frac{(A_s + A_{sr})}{\text{Net Area}}$	$u =$ .018

CALCULATE *w*, the probability of critical amounts of combustibles being present between inspections.

Critical quantity of transient combustible (from Worksheet 1, 2, or 3, or CIIIe) ________

If this amount is allowed during power operation without a permit, then:

$w = 1.0.$

If this amount is not allowed during power operation without a permit, then calculate *w* as follows:

1.	Frequency of critical combustible loading is the number of times the critical quantity was found present in violation of procedures. Use inspection reports from the NRC, QA, audits, housekeeping, TCC inspections, FP tours, OP inspections, etc. Divide the total number of incidents by the number of years for which the data was gathered.	F_{ccl} = findings/yr	$F_{ccl} =$ 1
2.	Determine the highest frequency of inspection for transient combustibles or plant housekeeping from information on the reverse side. F_w may be conservatively set to 1.0/year.	F_w = inspections/yr 52	$F_w =$ 52
3.	Calculate *x*	$x = F_{ccl}/F_w$	$x =$ 0.019
4.	Probability of critical amounts of transient combustible being present between inspections	$w = (x/2) * \ln(1/x)$	$w =$ 0.038

FIGURE 6.3

Plant	BFN	**By**	
Compartment	Unit 2 Reactor Building	**Date**	
Scenario	Targets within radiant exposure region of 32 gallon trash bag fire	**Page**	

PROBABILITY OF TRANSIENT COMBUSTIBLE FIRE EXPOSURE, P_{tc}

Note: fixed ignition sources should be evaluated before performing these calculations. If a fixed ignition source is shown to be able to cause damage, and there is no automatic suppression in the area, and manual suppression is not credited, the probability of critical combustible loading (P_{ccl}) for this compartment is = 1.0, and there is no need to consider P_{tc}. Next, an attempt should be made to screen the compartment with u, p, and/or w conservatively set to 1.0. If unsucessful, calculate one or more of the following to screen the compartment. Additional guidance can be found in Section 6.3.7, Steps 3.5 - 3.8 "Fire-Induced Vulnerabiltiy Evaluation (FIVE) (EPRI TR-100370, April 1992).

CALCULATE *p*, the probability of combustibles being exposed.

P can be assumed equal to 10% if the plant transient combustible control program has features similar to the following:

- Flammable and combustible liquids in the compartment are stored in approved containers;
- Ordinary combustibles or WRP clothing are stored in enclosed metal cabinets or metal containers with fusible link actuated covers or FM approved self-extinguishing lids;
- All exposed transient combustibles used by plant personnel are removed upon completion of the work unless otherwise approved.

Otherwise, a plant specific value can be obtained by performing a walkdown of significant areas of the plant to determine the following:

1.	Number of instances where plant combustibles are found exposed	P_1	$P_1 =$
2.	Number of instances where plant combustibles are found not exposed	P_2	$P_2 =$
3.	Calculate p	$P = \frac{P_1}{P_1 + P_2}$	$P =$ 0.1

CALCULATE *u*, the probability of transient combustibles being located in range of the target (i.e., area ratio).

1.	Determine the surface area of targets facing the floor (e.g. tray width times length)	$A_s =$ ___ x ___	$A_s =$ 1500 ft^2
2.	Determine the critical separation distance (from radiant energy exposure using Worksheet 3 or CIIIe). CRITICAL RADIAL DIST = 5.0'	A_{sr} from Worksheet 3 or CIIIe	$A_{sr} =$ 100 ft^2
3.	Determine the net area of floor space where combustibles could be stored in the compartment (total floor area minus 20% equipment area)	Total area, ___ * .8	Net Area = 56,000
4.	Calculate *u*	$u = \frac{(A_s + A_{sr})}{\text{Net Area}}$	$u =$ 0.0289

CALCULATE *w*, the probability of critical amounts of combustibles being present between inspections.

Critical quantity of transient combustible (from Worksheet 1, 2, or 3, or CIIIe) ______

If this amount is allowed during power operation without a permit, then:

$w = 1.0.$

If this amount is not allowed during power operation without a permit, then calculate *w* as follows:

1.	Frequency of critical combustible loading is the number of times the critical quantity was found present in violation of procedures. Use inspection reports from the NRC, QA, audits, housekeeping, TCC inspections, FP tours, OP inspections, etc. Divide the total number of incidents by the number of years for which the data was gathered.	F_{ccl} = findings/yr	$F_{ccl} =$ 1
2.	Determine the highest frequency of inspection for transient combustibles or plant housekeeping from information on the reverse side. F_w may be conservatively set to 1.0/year.	F_w = inspections/yr 52	$F_w =$ 52
3.	Calculate *x*	$x = F_{ccl}/F_w$	$x =$ 0.019
4.	Probability of critical amounts of transient combustible being present between inspections	$w = (x/2) * \ln(1/x)$	$w =$ 0.038

FIGURE 6.4

7. DOCUMENTATION OF RESULTS (PHASE III)

In keeping with the requirements of Supplement 4 to Generic Letter 88-20 (NUREG 1407) and the guidance provided by the EPRI FIVE documentation, this evaluation has confirmed that there are no fire-induced vulnerabilities associated with the continued operation of Browns Ferry Unit 2.

The screening evaluation of fire hazards that were performed in the course of this plant evaluation are summarized in Table 7-1, below. This table shows, as shaded, the level of analysis within the EPRI FIVE process at which any given plant area was screened from further consideration and the results from the associated section of this report that addresses the evaluation. In the case of Unit 2 Reactor Building fire zones, the analysis was performed by evaluation of individual fire ignition sources, as described in Note 1 below Table 7-1.

Since this evaluation represents the result of a progressive screening analysis, the reader should be cautioned against summing the fire-related core damage frequency values developed for any of the individual plant areas described in this report in an attempt to determine a "total" value for plant risk due to fires. Due to the conservative nature of this evaluation, these values presented in this report should be considered as upper bounding values only. That is, this evaluation has shown that the total core damage frequency due to fire-initiated plant trips for each of the plant areas at Browns Ferry Unit 2 is no higher than the value listed in Table 7-1. Due to the conservative nature of this evaluation, the "actual" core damage frequency due to fire-related initiating events is judged to be considerably lower than these values.

Also, due to the progressive nature of this evaluation, the various individual plant areas and potential fire sources have been screened from further consideration at significantly different levels of detail in the analysis. For example, the areas that were screened from further consideration in Section 5 were evaluated by assuming that any and all fires are severe, engulfing the entire fire area and damaging all plant equipment and electrical cables in the area. The evaluations described in this section take no credit whatsoever for automatic or manual fire suppression. The plant areas that are evaluated in a more detailed analysis in Section 6 are still judged to be conservative, though the level of conservatism in these evaluations is not as drastic as that used in the initial evaluation.

Table 7-1 Summary of Results				
Area	Description of Area	Areas Screened During Qualitative Analysis (Phase I)	Areas Screened During Quantitative Analysis (Phase II)	
			Initial Screening (Section 5)	Detailed Analysis (Section 6)
1	Unit 1 Reactor Building			5.19E-08
2-1	Unit 2 Reactor Building, West 519' - 565' Elevations			See Note 1
2-2	Unit 2 Reactor Building, East 519' - 565' Elevations			See Note 1
2-3	Unit 2 Reactor Building, North 593' Elevation			See Note 1
2-4	Unit 2 Reactor Building, 593' South and RHR Heat Exchanger Rooms			See Note 1
2-5	Unit 2 Reactor Building, 621' and North Side of 639'			See Note 1
2-6	Unit 2 Reactor Building, South 639' Elevation			See Note 1
3	Unit 3 Reactor Building			1.06E-07
4	4kV Shutdown Board Room B			4.97E-07
5	4kV Shutdown Board Room A and 250V Battery Room			2.54E-07
6	480V Shutdown Board Room 1A		3.47E-09	
7	480V Shutdown Board Room 1B		3.57E-09	
8	4kV Shutdown Board Room D			4.15E-07

Table 7-1 Summary of Results				
Area	Description of Area	Areas Screened During Qualitative Analysis (Phase I)	Areas Screened During Quantitative Analysis (Phase II)	
			Initial Screening (Section 5)	Detailed Analysis (Section 6)
9	4kV Shutdown Board Room C, 250V Battery Room			4.51E-07
10	480V Shutdown Board Room 2A		1.67E-08	
11	480V Shutdown Board Room 2B		3.66E-08	
12	Shutdown Board Room F		3.02E-09	
13	Shutdown Board Room E		5.37E-09	
14	480V Shutdown Board Room 3A		2.79E-09	
15	480V Shutdown Board Room 3B		2.89E-09	
16-1	Control Bay - 593' Elevation			4.73E-07
16-2	Cable Spreading Room			4.48E-07
16-3	Control Rooms			3.70E-08
17	Unit 1 Battery and Battery Board Room		9.44E-09	
18	Unit 2 Battery and Battery Board Room			5.53E-07
19	Unit 3 Battery and Battery Board Room		1.03E-08	
20	Unit 1 and 2 Diesel Generator Building		2.84E-07	
21	Unit 3 Diesel Generator Building		6.84E-08	

Table 7-1 Summary of Results				
Area	Description of Area	Areas Screened During Qualitative Analysis (Phase I)	Areas Screened During Quantitative Analysis (Phase II)	
			Initial Screening (Section 5)	Detailed Analysis (Section 6)
22	4kV Shutdown Board Room 3EA, 3EB		2.08E-08	
23	4kV Shutdown Board Room 3EC, 3ED		2.91E-09	
24	4kV Bus Tie Board Room		3.08E-07	
25-1	Intake Pump Station			4.72E-07
25-2	Pipe Tunnel			<1E-10
25-3	Turbine Building			7.30E-07

Notes:

1. The Unit 2 Reactor Building areas were analyzed by individual evaluation of potential fire sources within the individual fire zones. Only the following sources were evaluated with a core damage frequency above 1E-07:

Shutdown Board Room HVAC Panel	2.63E-07
2-LPNL-025-0031	2.88E-07
240V Lighting Transformer TL2A	7.17E-07

   For completeness, unqualified cables were also included in Table 6-3, with an upper bound core damage frequency of 5.47E-07

2. No fire area or zone was assigned to the yard area, though, for completeness, potential fires in this area are evaluated in Section 6.2.14. This evaluation gave a total upper bound core damage frequency for this area of 7.95E-07.

Review of these results show that only the following potential fire ignition sources/fire area evaluations resulted in upper bound core damage frequencies that were within a factor of 2 of the screening criteria of 1E-06:

Yard Area Fires, which have an upper bound core damage frequency of 7.95E-07. This is primarily due to the level of component damage that is assumed to occur for all fires that breach into the Turbine Building (i.e. total loss of offsite power). No credit was taken in this evaluation for fire suppression by installed systems or the fire brigade. Also, no credit was taken for any potential recovery actions.

Turbine Building (Compartment 25-3), which has an upper bound core damage frequency of 7.30E-07. This is primarily due to the level of component damage (i.e. loss of all offsite power) that is assumed to occur for all unsuppressed fires. Credit was only taken for fire suppression by installed systems. Fire suppression by the fire brigade and potential recovery actions were not considered in this evaluation.

240V Lighting Transformer TL2A (Located in Fire Zone 2-5), which has an upper bound core damage frequency of 7.17E-07. This is primarily due to the level of component damage that is assumed to occur for all fires in this ignition source. No credit was taken in this evaluation for fire suppression by installed systems or the fire brigade. Also, no credit was taken for any potential recovery actions.

Unit 2 Battery and Battery Board Room (Fire Area 18), which has an upper bound core damage frequency of 5.53E-07. This is primarily due to the level of component damage that is assumed to occur for all fires, even those that were evaluated as minor. No credit was taken in this evaluation for fire suppression by installed systems or the fire brigade. Also, no credit was taken for any potential recovery actions.

While no vulnerabilities were identified in the course of this evaluation, several items of interest were noted:

- In general, essential switchgear rooms were noted to have low conditional core damage frequencies. This is due to the large amount of partitioning between divisions and trains at the Browns Ferry plant. For example, RHR pumps A, B, C and D are each supplied from a different 4kV shutdown board, each of which is located in a different fire area. The four core spray pumps are supplied in a similar fashion. This design prevents the failure of a single shutdown board, whether due to fire or due to independent hardware failure, from failing an entire division of a given ECCS system. Also, the unit battery boards are set up to allow the maximum level of flexibility and redundancy between the three units.

- Several potential fire sources were noted to be located in close proximity to risk-significant components and cables. Most notable of these were the shutdown board room HVAC compressor motor, which is mounted above the shutdown board room and is located below cable runs that could potentially impact the operation of a number of safety related components, and the Unit 2 preferred AC transformer, which is also located in such a way that a number of safety-related cables fall within the potential zone of influence.

- Division 1 and 2 related switchgear was noted to be in close proximity (i.e. separated by a three to four foot wide walkway) in two cases (4kV shutdown board rooms C and D).

- As expected, oil filled transformers have the potential to generate significant amounts of heat and cause extensive damage to components in the area. In a similar fashion, the recirc MG sets can act as significant fire sources due to the large amount of lubricating oil present. While many of the transformers have been replaced with air cooled units, the remaining oil filled units still pose the threat of developing a severe fire, even though the plant design will protect against core damage. Plant training should therefore continue to ensure that fire brigade members are cognizant of these hazards.

8. NEW AND REMAINING ISSUES (PHASE III)

This includes response to and resolution of the Sandia Fire Risk Scoping Study (NUREG 5088) issues and the evaluation of containment isolation and heat removal. Also, the individual requirements for performance and documentation of a fire IPEEE, as specified in NUREG 1407, are addressed.

8.1 Evaluation of Containment Heat Removal and Isolation

The Phase II analysis concluded that the likelihood of loss of safe shutdown capability for all Browns Ferry fire areas and compartments is less than 1E-06 per reactor year (i.e. the core damage frequency from a particular fire-initiated event is negligible). Therefore, a separate analysis of containment performance and potential degradation due to the impact of fire-related component damage is not necessary.

A separate discussion of the potential for fire-induced containment bypass scenarios is provided in Section 5.

8.2 Treatment of Sandia Fire Risk Scoping Study Issues

The EPRI FIVE documentation discusses the following six issues to be addressed.

1. Seismic/fire interactions.

2. Fire barrier qualification.

3. Manual fire fighting effectiveness.

4. Total environment equipment survival.

5. Control systems interaction.

6. Improved analytical codes.

These issues, which were originally taken from the Fire Risk Scoping Study (NUREG/CR-5088) performed by Sandia Laboratories (the Sandia Fire Risk Scoping Study Issues) are discussed below. The specific responses for each of these concerns for the Browns Ferry Unit 2 analysis are listed in italics directly below the description of the Sandia issue.

8.2.1 Seismic/Fire Interactions

The issue of seismic/fire interactions centers on the following 3 areas of interest:

- Seismically induced fires. In particular, this concern centers on fires caused by flammable gas or liquid storage containers or systems that could rupture during a seismic event.

- Seismic actuation of fire suppression systems. In particular, this concern centers on the failure of electrical or other components due to water sprays.

- Seismic degradation of fire suppression systems. In particular, this concern reviews the plant design for fragility of fire suppression systems to a seismic event.

Each of these areas of interest is described in detail below.

8.2.1.1 Seismically Induced Fires

As part of the seismic assessment walkdown, verify hydrogen or other flammable gas or liquid storage vessels in areas with seismic safe shutdown or safety related equipment are not subject to leakage under seismic conditions. Examples would be improperly anchored hydrogen or oxygen bottles, hydrogen tanks used for primary coolant chemistry control, etc.

Response *Hydrogen or flammable gas/liquid storage vessels are not kept on a permanent basis in the Reactor Building, Diesel Generator Buildings, Control Building or the Intake Pump Station. Site standard practice 12.7 (Reference 26), Housekeeping/Temporary Equipment Control, provides the requirements for this type of combustible, including the requirement that compressed gas cylinders be tied to permanent structural features, using methods as described in the standard practice.*

In addition, the seismic walkdown required for the seismic portion of the IPEEE will identify any potential for seismic class II components affecting seismic class I components in safety related areas.

8.2.1.2 Seismic Actuation of Fire Suppression Systems

As part of the seismic assessment, verify that the design of the water suppression system considers the effects, if appropriate, of inadvertent suppression system actuation and discharge on that equipment credited as part of the seismic safe shutdown path in a margins assessment that was not previously reviewed relative to the internal flooding analysis or concerns such as those discussed in NRC I&E Notice 83-41.

Response *This issue was also addressed by Information Notice 94-12, Effects of Fire Suppression System Actuation on Safety Related Systems. The Browns Ferry response to these issues was as follows:*

1. *Mercury Relays. No mercury relays are present in the fire protection control systems.*

2. *Seismic Dust/Smoke Detectors. Smoke and/or heat detectors are used at Browns Ferry to actuate fire suppression systems in various areas of the plant. The CO_2 systems are actuated by heat detectors or by a combination of smoke and heat detectors. Therefore, dust particles created during a seismic event alone will not activate the CO_2 systems.*

   Most safety related areas in the plant are protected with fusible link (closed head) preaction sprinkler systems. If the preaction sprinkler system is inadvertently actuated (due to a seismic event), there will still be no water discharge due to the closed head sprinklers. The only safety related areas where open head spray systems area used are in the Unit 1 Reactor Building cable trays and the Unit 3 Diesel Generator Building cable and pipe tunnel area cable trays. The Unit 1 spray system is planned to be decommissioned prior to restart and the pertinent areas of the Unit 3 DG building do not contain any components that are susceptible to water damage. As part of the Appendix R analysis, fire suppression damage evaluations have been made. It has been concluded that spurious discharge of water from fire suppression systems will have no adverse impact on the safe shutdown capability of the plant.

3. *Water Deluge Systems. As noted above, open head deluge systems are only used for cable tray protection in two areas of the plant that contain safety related equipment. These systems do not provide protection for electrical cabinets or non-spray proof components.*

4. *Fire Suppressant Availability during a Seismic Event. Halon systems are not used to protect areas that contain safety related equipment. The CO_2 systems are seismically qualified, with the exception of the refrigeration system, which is not required except for prolonged periods. The water suppression system used three electric motor driven pumps and one diesel driven fire pump. The pumps and associated 4kV shutdown boards are located in seismic class 1 structures.*

5. *Switchgear Fires. There are few cases where electrical cables and raceways are located close to the top of electrical cabinets and could become directly involved in a fire. These cases are evaluated in Section 6.1 of this report.*

6. *Electro-Mechanical Components in Cable Spreading Rooms. No electric cabinets are present in these areas at Browns Ferry. HVAC equipment and control panels in these areas are installed such that tipping or sliding is prevented.*

8.2.1.3 <u>Seismic Degradation of Fire Suppression Systems</u>

As part of the seismic assessment walkdown, verify that plant fire suppression systems have been structurally installed in accordance with good industrial practice and reviewed for seismic considerations, such that suppression system piping and components will not fail and damage safe shutdown path components, nor is it likely that leaking or cascading of the suppressant will result.

<u>Response</u> *The fire protection system piping is designed to maintain pressure boundary integrity where spray damage to safety related components would affect the safe shutdown capability of the plant. The fire protection system piping is designed at a minimum for position retention (seismic II/I design criteria). Additionally, the seismic portion of the IPEEE analysis will identify any potential outliers, where seismic class II components could damage seismic class I components.*

8.2.2 Fire Barrier Qualifications

The concern for fire barrier qualification centers on the following 4 areas of interest:

- Fire barrier surveillance program.
- Inspection and maintenance of fire doors.
- Installation, inspection, surveillance and maintenance of penetration seal assemblies.
- Inspection, testing and maintenance of fire dampers.

Each of these areas of interest is described in detail below.

8.2.2.1 Fire Barriers

Fire barriers and components such as fire dampers, fire penetration seals and fire doors for fire barriers are included in the plant surveillance program.

Response *Fire barriers are included in the Browns Ferry plant surveillance program. Surveillance instruction 0-SI-4.11.G.1a, Visual Inspection of Fire Rated Barriers (Floors, Walls and Ceiling), is performed to verify the functional status of required fire rated barriers, including mechanical pipe fire rated penetration seals and external electrical conduit fire rated seals by performing a visual inspection.*

8.2.2.2 Fire Doors

A fire door inspection and maintenance program should be implemented at the plant.

Response *The inspection of fire doors is addressed by surveillance instruction 0-SI-4.11.G.2.b, Fire Door Inspection.*

8.2.2.3 Penetration Seal Assemblies

a. A penetration seal inspection and surveillance program should be implemented at the plant.

Response *The surveillance and inspection of penetration seals is addressed in surveillance instructions 0-SI-4.11.G.1.a, Visual Inspection of Fire Rated Barriers (Floors, Walls and Ceiling), and 0-SI-4.11.G.1.c (2), Visual Inspection of Cable Tray Penetrations in Fire-Rated Barriers.*

b. Fire barrier penetration seals have been installed and maintained to address concerns such as those identified in NRC Information Notice 88-04.

Response *Fire barrier penetration seals at the Browns Ferry Nuclear Plant have been installed and are maintained in compliance with the relevant Appendix R requirements, as described in Volume 1 of the Browns Ferry Fire Protection Report.*

8.2.2.4 Fire Dampers

a. An inspection and maintenance program for fire dampers should be implemented at the plant.

Response *The inspection and testing of fire dampers is addressed by surveillance instructions 0-SI-4.11.G.1.b, Visual Inspection/Test of Appendix R, Unit 2, System 64 Fire Dampers (Unit 2 Reactor Building), 0-SI-4.11.G.1.b (1), Visual Inspection/Test of Appendix R System 30 Fire Dampers (Radwaste Building), 0-SI-4.11.G.1.b (2), Visual Inspection/Test of Appendix R System 31 and 39 Fire Dampers (Control Bay) and 0-SI-4.11.G.1.b (4), Visual Inspection/Test of Appendix R Fire Dampers (other areas).*

b. Damper installations address concerns such as those identified in NRC Information Notice 89-52, "Potential Fire Damper Operational Problems," dated June 8, 1989 and NRC Information Notice 83-69, "Improperly Installed Fire Dampers at Nuclear Power Plants," dated October 21, 1983.

Response *Fire dampers at the Browns Ferry Nuclear Plant are installed to meet the Appendix R compartmentation requirements. These dampers are inspected as described in Volume 1 of the Browns Ferry Fire Protection Report. Recent fire damper installations in the Unit 3 Reactor Building are of the "dynamic" type. That is, these dampers are designed to close under rated air flow conditions. For other areas of the plant, procedures are in place to shut down the HVAC systems for fires in those areas, enabling the fire dampers to close.*

All Appendix R fire dampers are tested by removing the fusible links and ensuring that the dampers close properly ("drop test").

8.2.3 Manual Fire Fighting Effectiveness

The concern for manual fire fighting effectiveness centers on the following 6 areas of interest:

- Fire reporting, including the use and availability of portable fire extinguishers and plant procedures for reporting fires, including plant communication.
- Fire brigade makeup and equipment.
- Fire brigade training in the classroom.
- Fire brigade practice in hands-on structural fire training and in the use of equipment.
- Fire brigade drills.
- Fire brigade training records.

Each of these areas of interest is described in detail below.

8.2.3.1 Reporting Fires

a. Appropriate plant personnel are knowledgeable in the use of portable fire extinguishers.

Response *Plant personnel and fire brigade members receive regular training in the use of portable fire extinguishers.*

b. Portable extinguishers are located throughout the plant.

Response *Portable fire extinguishers are placed at key locations throughout the plant. These locations are identified in the pre-fire plans shown in Volume 2 of the Browns Ferry Fire Protection Report, Section IV.*

c. A plant procedure is in use for reporting fires in the plant.

Response *EPIP 21, Fire Emergency Procedure, directs the notifications required in the event of a plant fire, including fire brigade members and offsite contacts.*

d. A plant communication system that includes contact to the control room is operable at the plant.

Response *All plant personnel are directed, during initial and refresher General Employee Training, to contact the Control Room in the event of a fire in the plant. This notification may be by telephone, from one of the internal plant communication stations or by plant operations/fire brigade radio.*

8.2.3.2 Fire Brigade Makeup and Equipment

8.2.3.2.1 A fire brigade that is made up of at least 5 trained people on each shift should be maintained at the plant.

Response *This requirement (1 brigade leader and at least 4 other members) is specified in Volume 2 of the Browns Ferry Fire Protection Report, Section II (5.1.1).*

8.2.3.2.2 The fire brigade leader and at least two other brigade members on each brigade shift should be knowledgeable in plant systems and operations.

Response *This requirement is specified in Volume 2 of the Browns Ferry Fire Protection Report, Section II (5.1.2).*

8.2.3.2.3 Each brigade member should receive an annual review of physical condition to evaluate his ability to perform fire fighting activities.

Response *This requirement is specified in Volume 2 of the Browns Ferry Fire Protection Report, Section II (5.1.3).*

8.2.3.2.4 A minimum amount of equipment should be provided for the on site fire brigade:

a. Personal protective equipment should be provided such as SCBA, turnout coats, boots, gloves, and hard hats.

Response *This requirement is specified in Volume 2 of the Browns Ferry Fire Protection Report, Section II (6.1.1.1).*

b. Emergency communications equipment should be provided for fire brigade use.

Response *This requirement is specified in Volume 2 of the Browns Ferry Fire Protection Report, Section II (6.1.1.4, 6.1.1.6 and 6.1.1.7).*

c. Portable lights should be provided for fire brigade use.

Response *This requirement is specified in Volume 2 of the Browns Ferry Fire Protection Report, Section II (6.1.1.4 and 6.1.1.5).*

d. Portable ventilation equipment should be provided for fire brigade use.

Response *This requirement is specified in Volume 2 of the Browns Ferry Fire Protection Report, Section II (6.1.1.3).*

e. Portable extinguishers should be provided for fire brigade use.

Response *The locations of portable and other fire extinguishers for fire brigade use are specified in the pre-fire plans shown in Volume 2 of the Browns Ferry Fire Protection Report, Section IV.*

8.2.3.3 Fire Brigade Training

Brigade members should receive an initial classroom instruction program consisting of the following:

a. A review of the plant fire fighting plan and identification of each individual's responsibilities.

b. Identification of typical fire hazards and associated types of fires that may occur in the plant.

c. Identification of the location of fire fighting equipment and familiarization with the layout of the plant, including access and egress routes.

d. Training on the proper use of available fire fighting equipment and the correct method of fighting each type of fire. The types of fires covered should include fires in energized electrical equipment, fires in cables and cable trays and fires involving flammable and combustible liquids and gases.

e. Training on the proper use of communication, lighting, ventilation and emergency breathing equipment.

f. Training on techniques for fighting fires inside buildings and confined spaces.

g. A review of fire fighting strategies and procedures.

Response *Fire Brigade training requirements, including those listed in items (a) through (g), above, are specified in Volume 2 of the Browns Ferry Fire Protection Report, Section III.*

8.2.3.4 Fire Brigade Practice

Fire brigade members should receive hands-on structural fire fighting training at least once a year to provide experience in actual fire extinguishment and the use of emergency breathing apparatus.

Response *Fire Brigade practice and drill requirements, including annual requirements, such as actual fire extinguishment and the use of emergency breathing apparatus, are specified in Volume 2 of the Browns Ferry Fire Protection Report, Section III.*

8.2.3.5 Fire Brigade Drills

a. Fire brigade drills are performed in the plant so that each fire brigade shift can practice as a team.

Response *Fire brigade drill requirements, including practice as a team, are specified in Volume 2 of the Browns Ferry Fire Protection Report, Section III (Appendix B).*

b. Drills should be performed at regular intervals for each shift fire brigade.

Response *Appendix B of Section III (Volume 2 of the Browns Ferry Fire Protection Report) requires drills to be scheduled at least 1 drill per shift per quarter, not to exceed 92 days between drills.*

c. At least one unannounced fire drill for each shift fire brigade should be performed per year.

Response *Unannounced drills are to be scheduled on an annual basis, not closer than 4 weeks apart, as specified by Appendix B of Section III (Volume 2 of the Browns Ferry Fire Protection Report).*

d. At least one drill per year should be performed on a "backshift" for each shift fire brigade.

Response *This requirement is specified in Volume 2 of the Browns Ferry Fire Protection Report, Section III (Appendix B).*

e. Drills should be preplanned to establish training objectives and critiqued to determine how well the training objectives have been met.

Response *This requirement is specified in Volume 2 of the Browns Ferry Fire Protection Report, Section III (Appendix D).*

f. At least triennially, an unannounced drill should be performed for and critiqued by qualified individuals, independent of the licensee's staff.

Response *This requirement is specified in Volume 2 of the Browns Ferry Fire Protection Report, Section III (Appendix B).*

g. Pre-fire plans should be developed for safety related areas of the plant (as a minimum).

Response *Pre-fire plans are provided in Volume 2 of the Browns Ferry Fire Protection Report, Section IV. These plans include area access, combustibles in the area, locations of fire suppression equipment, including hose stations, and radiological hazards. These plans have been developed for all safety related plant areas, in addition to other plant areas.*

h. The pre-fire plans should be updated and used as part of the brigade training.

Response *This requirement is specified in Volume 2 of the Browns Ferry Fire Protection Report, Section III (Appendix B).*

i. Fire brigade equipment is maintained in good condition and ready for use by the fire brigade.

Response *Quarterly Inspection of Emergency Equipment (FPO 000 INS 005) specifies the inspection procedures for fire brigade equipment, such as that contained in equipment cages, lockers, stretcher cabinets and carts. Also, equipment operability is verified prior to storage after each drill.*

8.2.3.6 Fire Brigade Training Records

Records are provided for each fire brigade member, demonstrating the minimum level of training and refresher training has been provided.

<u>Response</u> *Fire brigade training records are required to be maintained, as specified in Volume 2 of the Browns Ferry Fire Protection Report, Section IV.*

<u>8.2.4 Total Environment Equipment Survival</u>

The general issue of total environmental equipment survival centers on the following 3 areas of interest:

- Adverse effects of combustion products on plant equipment.
- Spurious or inadvertent fire suppression system actuation.
- Impact on effectiveness of operator actions.

Each of these areas of interest is discussed in detail below.

8.2.4.1 <u>Potential Adverse Effects on Plant Equipment by Combustion Products</u>

a. The FIVE methodology does not currently provide for an evaluation of non-thermal environmental effects of smoke on equipment. See Section 4.2.2 of EPRI TR-100370, Fire-Induced Vulnerability Evaluation (FIVE).

<u>Response</u> *During the screening evaluation, all equipment in the affected area was assumed to be damaged by the fire. More specific plant model impacts were modeled during the detailed analysis. This treatment is judged to conservatively bound the impact of non-thermal environmental effects on plant equipment. Also, these non-thermal effects, such as corrosion or degradation due to soot or other smoke products occur over a much longer period than that required to establish cold shutdown conditions. These impacts on plant equipment, such as control circuitry and switchgear, would be addressed during the ensuing plant outage period, as part of corrective maintenance following the fire.*

b. Plant staff should be aware of and sensitive to the potential impact of smoke and products of combustion on human performance in safe shutdown operations in application of FIVE.

Response *Plant operations personnel receive regular training in the effective use of SCBA equipment. Also, operator actions were considered to fail for fires in a given area within the plant model by failing the associated plant equipment.*

8.2.4.2 Spurious or Inadvertent Fire Suppression Activation

Verify that the design of fire suppression systems considers the effects, if appropriate, of inadvertent suppression system actuation and discharge on equipment credited for safe shutdown for concerns such as those discussed in NRC I&E Information Notice 83-41.

Response *This issue was also addressed by Information Notice 94-12, Effects of Fire Suppression System Actuation on Safety Related Systems. The Browns Ferry response to these issues is discussed under Section 8.2.1.2, above.*

8.2.4.3 Operator Action Effectiveness

a. There are safe shutdown procedures that identify the steps for planned shutdown when necessary, in the event of a fire.

Response *Safe shutdown instructions have been developed to address the fires that could develop in each area of the plant. These procedures provide detailed instructions to direct the control room operator's response to the potential loss of equipment and support cables located in each area of the plant.*

b. Operators should receive training on the safe shutdown procedures.

Response *Discussions with plant operators have confirmed that they regularly receive training in the use of the safe shutdown instructions.*

c. If, in performance of these procedures, operators are expected to pass through or perform manual actions in areas that may contain fire or smoke suitable SCBA equipment and other protective equipment are available for operators to perform their function.

Response *SCBA equipment is located in key locations throughout the plant, in addition to the equipment that is located in the fire brigade lockers. Plant operators receive regularly scheduled training in the effective use of this equipment.*

8.2.5 Control Systems Interactions

This issue centers on the concern that safe shutdown circuits are physically independent of, or can be isolated from, the control room for a fire in the control room fire area.

Response *The remote shutdown system provides for plant monitoring and control stations from which to perform a safe shutdown of the plant from outside the control bay in the event of control system damage due to a fire in the Control Room, Cable Spreading Room or the 593 foot elevation of the Control Building. This capability is described in Section 7.18 of the Updated Final Safety Analysis Report. The implementation of this capability is directed by Abnormal Operating Instruction 2-AOI-100-2, Control Room Abandonment.*

8.2.6 Improved Analytical Codes

The issue of analytical codes centers on the fire modelling techniques that have been incorporated into the FIVE methodology. These modeling techniques, which are derived from the basic correlations used in the COMPBRN IIIe fire modelling program, have been reviewed for use in the modeling of fire progression.

Response *The correlations shown in the FIVE documentation were used to generate the zones of influence that were used during the detailed analysis of Reactor Building areas in Section 6.1.*

These correlations are based on updated fire modeling techniques from those reviewed in the Sandia study.

8.3 Requirements of NUREG-1407

The analysis described in this report was performed in order to meet the informational requirements of NUREG-1407. In particular, NUREG-1407 specifies the submittal of documentation for the following areas of interest (Appendix C, Section C.3):

1. A description of the methodology and key assumptions used in performing the fire IPEEE and a discussion of the status of Appendix R modifications.

Response *The fire IPEEE methodology consists of a progressive screening analysis, based on the EPRI FIVE methodology, as described in EPRI report TR-100370.*

Browns Ferry Unit 2 is currently in compliance with all Appendix R related requirements.

2. A summary of walkdown findings and a concise description of the walkdown team and the procedures used. This should include a description of the efforts to ensure that cable routing used in the analysis represents as-built information and the treatment of any existing dependence between remote shutdown and control room circuitry.

<u>Response</u> *The walkdown findings and procedures are described in Attachment D. In general, this process confirmed the existing Appendix R documentation. Cable routing information was confirmed during this process by physical area walkdown and review of plant documentation.*

The remote shutdown capability was only credited for severe fires in the Control Bay, which were conservatively assumed to require Control Room evacuation (see Section 6.2). This system was specifically designed to provide an independent control capability for identified plant systems and functions, including any required control circuitry. The remote shutdown capability system is described in Section 7.18 (Backup Control System) of the UFSAR.

3. A discussion of the criteria used to identify critical fire areas and a list of critical areas, including (a) single areas in which equipment failures represent a serious erosion of safety margin, and (b) same as (a), but for double or multiple areas that share common barriers, penetration seals, HVAC ducting, etc.

<u>Response</u> *Critical fire areas are considered to be those areas that contain either any components that are modeled in the Level 1 PRA plant model or any associated support circuitry. During the qualitative screening analysis (see Section 3.3), all plant areas were conservatively assumed to contain safe shutdown equipment or associated support cables. All plant fire areas were therefore retained for quantitative analysis.*

Each of the individual fire areas was then evaluated on a quantitative basis, assuming that any and all fires would totally engulf the area and result in a plant trip. If the resulting core damage frequency was less than 10^{-6}, further quantitative analysis was judged to be unnecessary and the area was screened from further consideration. This process is described in Section 5.

Detailed area analysis was then performed for the Reactor Building, Control Building and Turbine Building areas, in addition to shutdown board rooms C and D. This analysis is described in Section 6. The results of this evaluation are summarized in Section 7.

Fire hazards that could extend to include multiple fire areas were screened from further consideration, based on the fire barrier screening guidelines given in the EPRI FIVE documentation. This is discussed in Section 3.3. The potential for a multiple area fire developing on the 593 foot elevation of the Control Building and propagating to the Cable Spreading Room, above, was not screened from consideration through this process. This potential fire is separately evaluated in Section 6.2.7.

4. A discussion of the criteria used to determine fire size and duration and the treatment of cross-zone fire spread and associated major assumptions.

<u>Response</u> *Fire size was conservatively assumed to be engulfing for all fires analyzed in the screening analysis described in Section 5. Fires were assumed to entirely consume the fire source for all Unit 2 Reactor Building fire sources (see Section 6.1).*

The Fire Events Database (NSAC/178L) was used as a basis for fire size for fires analyzed in Section 6.2. Fire duration was as required to consume the source. Cross-zone spread of fires was evaluated using the EPRI FIVE criteria, as described in Section 3.3. A potential multiple fire, developing on the 593 foot elevation of the Control Building and propagating to the Cable Spreading Room, above, was identified through this process. This potential fire is separately evaluated in Section 6.2.7.

5. A discussion of the fire initiating event database, including the plant specific database used. Provide documentation in each case where the plant specific data is less conservative than the data used in the approved fire vulnerability methodologies. Describe methods for handling data, including major assumptions, the role of expert judgement, and the identification and evaluation of sources of data uncertainty.

Response *The EPRI Fire Events Database was used to generate fire ignition frequencies, as described in the EPRI FIVE documentation. Review of plant experience shows plant specific data to be no less conservative than the data given in the FIVE documentation.*

Due to the use of a progressive screening analysis, data uncertainty was not explicitly modeled. For each of the fires that remained for more detailed analysis, a qualitative discussion of conservative assumptions and potential recovery actions is given in Section 6. It should be noted that, with the exception of the use of the remote shutdown capability for selected severe fires in the Control Building, recovery of equipment from fire-induced damage is conservatively not credited in this analysis.

6. A discussion of the treatment of fire growth and spread, the spread of hot gases and smoke, and the analysis of detection and suppression and their associated assumptions, including the treatment of suppression induced damage to equipment.

Response *Fire growth between areas is addressed by using the EPRI FIVE criteria, as described in Section 3.3. Detection and suppression are not evaluated as mitigating any fires in the screening evaluation that was performed in Section 5 and are only credited for selected cases, on a case-by-case basis, in the detailed analysis, as described in Section 6.*

Suppression-induced damage is addressed under the associated Sandia issue in Section 8.2.1.2

7. A discussion of fire damage modeling, including the definition of fire-induced failures related to fire barriers and control systems and fire induced damage to cabinets. A discussion of how human intervention is treated and how fire induced and non-fire induced failures are combined. Identify recovery actions and types of fire mitigating actions for which credit is taken in these sequences.

Response *Fire barrier effectiveness was evaluated using the EPRI FIVE criteria, as described in Section 3.3 and documented in EPRI report TR-100370. For this analysis, control systems were assumed to fail in such a way as to fail the function of the affected system. It should be noted that this analysis conservatively assumes that "hot short" failures occur whenever necessary to fail the system function.*

Cabinet damage was conservatively assumed to occur for all fires in the area, with the exception of those areas evaluated in Section 6, where component damage was typically assumed to occur, based on the individual case under consideration.

Human intervention is conservatively not credited in the screening analysis (Section 5) and is only credited on a case-by-case basis in the detailed analysis (Section 6), for fire suppression only. Non-fire induced failures are combined with fire-related impacts through use of the Level 1 PRA plant model. With the exception of selected Control Room evacuation scenarios, where use of the remote shutdown system is modeled, no credit is taken in this analysis for possible recovery from fire-induced failures. In other words, all fire-induced failures are conservatively assumed to be irrecoverable.

8. Discuss the treatment of fire detection and suppression, including fire fighting procedures, fire brigade training and adequacy of existing fire brigade equipment and treatment of access routes versus existing barriers.

__Response__ *Fire suppression was only considered in the detailed analysis in Section 6, and only on a case-by-case basis. Fire brigade training, equipment availability and procedures are described under the associated Sandia issue in Section 8.2.*

9. All functional and systemic event trees associated with fire-initiated sequences.

__Response__ *The plant model and associated event trees are as described in the Level 1 PRA/IPE report. Fire-initiated scenarios were incorporated by failing individual top events within the Level 1 plant model. The individual event trees that were used to segment fire ignition frequency into individual cases, where this technique was used, are shown in Section 6.2.*

10. A description of dominant functional and systemic sequences leading to core damage, along with their frequencies and percentage contribution to overall core damage frequency due to fire. Sequence selection criteria are as provided in Generic Letter 88-20 and NUREG-1335. The description of the sequences should include a discussion of specific assumptions and human recovery actions.

Response *The results of the fire risk analysis are summarized and discussed in Section 7. Due to the use of a progressive screening approach, as described in the EPRI FIVE documentation, individual scenarios are not listed for areas that were screened from further consideration, based on fire-related core damage frequency of less than 1E-06.*

11. The estimated core damage frequency, the timing of the associated core damage, a list of analytical assumptions, including their bases, and the sources of uncertainty.

Response *The results of this analysis are shown in Section 7. The analytical assumptions used to evaluate each plant area are provided with the discussion in the associated text. Due to the use of a screening analysis, plant damage states would only be evaluated for unscreened areas. Also, a separate analysis of data uncertainty was not performed due to use of a screening analysis.*

12. Any fire induced containment failures identified as being different from those identified in the internal events analysis.

Response *Containment failure due to fire-induced damage was addressed in Section 8.1. This review concluded that no significant containment failures were introduced by the analysis of internal fires.*

13. Documentation with regard to the decay heat removal function and Fire Risk Scoping Study issues addressed by the submittal, the basis and assumptions used to address these issues, and a discussion of the findings and conclusions. Evaluation results and potential improvements should be specifically highlighted. Specifically, NUREG-1407 (Section 4) specifies that the submittal should address the following Fire Risk Scoping Study issues:

- Seismic/fire interactions.
- Effect of fire suppressant systems on safety equipment.
- Control system interactions.

Response *The issues raised in the Fire Risk Scoping Study (NUREG/CR-5088) are addressed in Section 8.2.*

14. When an existing PRA is used to address the fire IPEEE, the licensee should describe sensitivity studies related to the use of the initial hazard, supplemental plant walkdown results and subsequent evaluations. The licensee should examine the above list to fill in those items missed in the existing fire PRA.

Response *Only the plant model was used from the Level 1 PRA. In particular, this model was used specifically to capture the non-fire induced failures that could occur and to model plant response, following the incorporation of fire-induced failures.*

9. REFERENCES

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2. Electric Power Research Institute, "Fire Events Database for U.S. Nuclear Power Plants," NSAC/178L, Revision 1, January 1993.

3. Tennessee Valley Authority, Combustible Loading Tables, Calculation MD-N0026-910163.

4. Electric Power Research Institute, "Fire Risk Analysis Implementation Guide," EPRI Draft Report 3385-01, January 1994.

5. Sandia National Laboratories, "An Experimental Investigation of Internally Ignited Fires in Nuclear Power Plant Control Cabinets," Part I and II, Prepared for the U.S. NRC, NUREG/CR-4527, SAND86-0336, November 1988.

6. Sandia National Laboratories, "Analysis of Core Damage Frequency: Surry Power Station, Unit 1 - External Events," Prepared for the U.S. NRC, NUREG/CR-4550, SAND86-2084, Volume 3, Part 3, Revision 1, December 1990.

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8. Sandia National Laboratories, "Fire Risk Scoping Study: Investigation of Nuclear Power Plant Fire Risk, Including Previously Unaddressed Issues," Prepared for the U.S. NRC, NUREG/CR-5088, SAND88-0177, January 1989.

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10. Society of Fire Protection Engineers, "Handbook of Fire Protection Engineering," First Edition.

11. Tennessee Valley Authority, "Browns Ferry Nuclear Plant Unit 1 Probabilistic Risk Assessment," September, 1987.

12. Tennessee Valley Authority, "Browns Ferry Nuclear Plant Unit 2 Probabilistic Risk Assessment Individual Plant Examination," Revision 1, August 12, 1994.

13. Tennessee Valley Authority, "Browns Ferry Nuclear Plant Updated Final Safety Analysis Report," Amendment 10.

14. Tennessee Valley Authority, "Fire Emergency Procedure," EPIP-21, Revision 0, March 7, 1989.

15. Tennessee Valley Authority, "Quarterly Inspection of Emergency Equipment," FPO 000 INS 005.

16. Tennessee Valley Authority, "Control Room Abandonment," Browns Ferry Nuclear Plant Abnormal Operating Instruction 2-AOI-100-2, Revision 27, March 15, 1993.

17. Tennessee Valley Authority, "Fire Protection," Site Standard Practice 12.15, Revision 9, March 18, 1994.

18. Tennessee Valley Authority, Fire Protection Report - Browns Ferry Nuclear Plant, Volume 1, Revision 9, November, 1994.

19. Tennessee Valley Authority, Browns Ferry Safe Shutdown Instructions, 2-SSI-001, 2-SSI-002 through 2-SSI-25.

20. Tewarson, A., "Heat Release in Fires," Fire and Materials, Volume 4, 1980, pages 185-191.

21. Drysdale, D., "An Introduction to Fire Dynamics," 1987, pages 14, 315, 21 (Table 1.13) and 172 (Table 5.9).

22. Society of Fire Protection Engineers, "Fire Protection Handbook," 17th Edition, Table A-2.

23. Tennessee Valley Authority, "Cable Tray Combustible Loading Calculations," MD-N0026-910155.

24. Tennessee Valley Authority, "NRC Generic Letter 88-20, Supplement 4 (NUREG-1407), IPEEE Internal Fires Engineering Analysis Plant Configuration Data Walkdowns," Walkdown Instruction Number EEB-010, RIMS R14 940706 204, Revision 0, July 5, 1994.

25. Tennessee Valley Authority, "Browns Ferry Nuclear Plant (BFN) - IPEEE Internal Fires Potentially Disabled Equipment Evaluations - NRC Generic Letter 88-20, Supplement 4," RIMS R92 940929 974, October 3, 1994.

26. Tennessee Valley Authority, "Housekeeping/Temporary Equipment Control," Site Standard Practice SSP-12.7, Revision 7, November 9, 1994.

ATTACHMENT A

HEAT RELEASE RATES

ATTACHMENT A
HEAT RELEASE RATES (HRR)

1. Electrical Cabinets:

The electrical cabinet fire represents a naturally ventilated fire scenario (ventilation controlled). Vent geometry controls the fire size because vent geometry limits the amount of air entering and hence limit the amount of oxygen that may combine with the fuel. The very early stages of fire may however see fuel limited burning. The BFN electrical cabinets/boards are composed of several individual cubicles. These cubicles are separated from each other by internal barriers. Conduit penetrations through the top of the cabinet are sealed by appropriate metallic fittings. All doors are kept closed and most cabinets have no ventilation openings. Mostly unqualified cables are used in electrical cabinets.

The heat heat release rate for a fire can be calculated by using the following equation (Reference 21, page 14):

$$Q = m \ \Delta H \chi$$

Where Q is the heat release rate (kJ/s), m is the mass loss rate (kg/s), ΔH is the heat of complete combustion of the fuel (kJ/kg), and χ is the combustion efficiency factor. Reference 20 (also Table 5.9, Reference 21) reported that χ lies within a relatively narrow range between 0.4 and 0.7. For PVC, χ is given as .357. The value of χ will be conservatively chosen as 0.5. The above equation states that the heat release rate can be calculated by multiplying the fuel mass loss rate by the heat of combustion. However, the mass of fuel vapors actually participating in combustion is controlled by the availability of air to the fuel and follows the reaction:

$$1 \text{ kg fuel} + r \text{ kg air} \Rightarrow (1 + r) \text{ kg products.}$$

where r is the stoichiometric air/fuel mass ratio, i.e. r kg of air required to completely burn 1 kg of fuel. Hence, in order to calculate the maximum HRR, one should know the value of m_{air}, or the amount of air entering the electrical cabinet through the ventilation openings. The size of the opening determines the flow of air towards the fuel. Reference 21 page 315 reports that:

$$m_{air} = 0.52 \ A_o \sqrt{H_o} \text{ kg/s}$$

where m_{air} is the mass of the air entering (kg/s), A_o is the area of opening and H_o is opening height (m).

Assuming stoichiometric burning of fuel, the maximum heat release rate is calculated as:

$$Q_{max} = m_{air}\ [\Delta H / r]\ \chi\ \times\ 10^3 kW$$

The above two equations can be combined to give the maximum heat release rate of:

$$Q_{max} = 0.52\ A_o \sqrt{H_o}\ [\Delta H / r]\ \chi\ \times\ 10^3 kW$$

Reference 21 Table 1.13 lists the stoichiometric ratio of 2.98 kJ/g of air for polyvinyl chloride. The mass of air can conservatively be calculated based on a vent opening in the electrical cabinet of .305m wide x .305m high (12" x 12" with base of the opening located at floor level) as .027 kg/s. (note that most of the MCC's in the reactor building have no vents, see attachment C for details)

Therefore the heat release rate is calculated to be 56 kW or 53 btu/sec.

The above heat release rate therefore is based on a conservative approach of selecting vent openings which either do not exist in electrical cabinets or are larger than the ones present.

Actual electrical cabinet fire tests have also been performed and will be examined for comparison purposes to the above calculated values. Heat release rates (HRR) were determined for several vertical cabinet fire tests conducted by Sandia National Labs for NRC (Reference 5). Test PCT # 1, utilized unqualified cables in a vertical electrical cabinet (3x5x7.5 ft), with closed doors, ventilation grilles (~ 4 sq ft) and in situ combustible loading of 740,000 Btu. The closed cabinet test PCT # 1 shows a peak HRR of 175 Btu/sec (includes the transient fuel source of 68 btu/sec). Open cabinet fire tests show much larger HRR. The closed cabinet HRR is approximately twice (accounting for the transient fire source) the above calculated value.

The validity of the calculated method can be verified by using the test cabinet vent geometry in the above equations. There are 4 vent openings of .369m x .344m. Assuming the average vent opening height to be 1 meter from the base and 30% free area, the mass air flow rate is calculated to be .079 kg/s. The corresponding HRR with a 50% combustion efficiency is then 118 kw or 112 btu/s. The test results show approximately the same HRR when the transient fuel source contribution is neglected. This shows that cabinet fires are ventilation controlled fires and smaller vent openings will result in lower HRR. Therefore,

the calculated HRR is in agreement with the test results based on smaller vent openings and still conservative due to lack of vent openings on most electrical cabinets.

Fires may either damage adjacent cabinets/cubicles or possibly, propagate to adjacent cabinets. Due to the relatively low combustible loading and presence of internal barriers between cabinets, the fire is not likely to propagate to adjacent cabinets. The fire tests in Reference 5 indicate the potential of propagation in cabinets sharing common walls for large fires (~ 1 million Btu) due to auto-ignition of cables in adjacent cabinet. However, the combustible loading in individual cabinets/cubicles at BFN is substantially less (see specific electric board evaluation) to develop auto ignition temperatures (600 C) in adjacent cabinet. Even if the fire does propagate to an adjacent cabinet, there is a time delay of ~ 15 minutes and that no significant heat release occurs from the adjacent cabinet for 15 minutes in which time the original cabinet would have consumed its combustibles. Since the potential of fire spread to outside the burning cabinet is dependent on many variables including fire growth rate, cable configurations, location of adjacent cabinet, barriers between cabinets, etc., it will be conservatively assumed that a fire simultaneously spreads to at least one additional (for electrical cabinet with no vents). Fire will be assumed to spread to a third cabinet if ventilated and cabinets are stacked. Therefore, the HRR for an electrical board/cabinet fire will be estimated as:

Heat Release Rate per Electrical Board/Cabinet = 2 x HRR per cabinet/cubicle (for non-vent cabinets)

OR

= 3 x HRR per cabinet/cubicle (for ventilated cabinets and stacked, neglecting small cubicles with negligible combustibles).

2. Pumps and Transformers:

There are no current fire test data available for pump motors and transformer HRR. These components are generally located in open spaces. The combustion process is likely to be fuel controlled rather than ventilation controlled like the electrical cabinets. The HRR will therefore be based on fuel loading. Again, electrical cabinet test data will be used for guidance. Data available in Reference

5 and reviewed in Reference 4 suggests that HRR for closed electrical cabinet with combustible loading of 1million Btu (non-qualified cables) be taken as 400 Btu/sec.

Pumps and transformers with no oil/grease (neglect minor quantities) will be considered similar to a closed non-qualified electrical cabinet. HRR will be computed based on the combustible loading of the component at 400 Btu/sec for 1 million Btu loading or 40 Btu/sec per 100,000 Btu.

Following is an estimate of HRR for electrical boards/cabinets, pumps and transformers based upon the above discussion:

480V RMOV BD 2C (Unit 2 Reactor Building EL 565) page 5

ignition freq (five)/HRR.WKZ

Number of Cubicles =	29	
Combustible Loading (Btu) =	5.39E+06	
Combustible Loading per Cubicle =	1.86E+05	
Heat Release Rate per Cubicle =	53 Btu/sec	
No vent openings are present in this boar		
Heat Release rate per Electrical Board =	106	(2 x HRR/cubicle)
Combustible Loading for 2 cubicles =	3.72E+05	

480V RB Vent Bd 2B (Unit 2 Reactor Building EL 565)

Number of Cubicles =	24	
Combustible Loading (Btu) =	4.90E+06	
Combustible Loading per Cubicle =	2.04E+05	
Heat Release Rate per Cubicle =	53 btu/sec	
No vent openings are present in this board		
Heat Release rate per Electrical Board =	106	(2 x HRR/cubicle)
Combustible Loading for 2 cubicles =	4.08E+05	

250V RMOV BD 2C (Unit 2 Reactor Building EL 565)

Number of Cubicles =	18	
Combustible Loading (Btu) =	4.90E+06	
Combustible Loading per Cubicle =	2.72E+05	
Heat Release Rate per Cubicle =	53 btu/sec	
No vent openings are present in this board		
Heat Release rate per Electrical Board =	106	(2 x HRR/cubicle)
Combustible Loading for 2 cubicles =	5.44E+05	

480V RMOV BD 2D (Unit 2 Reactor Building EL 593)

Number of Cubicles =	10	
Combustible Loading (Btu) =	2.31E+06	
Combustible Loading per Cubicle =	2.31E+05	
Heat Release Rate per Cubicle =	53 BTU/sec	
No vent openings are present in this board		
Heat Release rate per Electrical Board =	106	(2 x HRR/cubicle)
Combustible Loading for 2 cubicles =	4.62E+05	

Unit 2 Preferred AC Transformer (Unit 2 Reactor Building EL 593) **page 6**

Number of components =	1
Combustible Loading (Btu) =	2.80E+05
Heat Release rate per Component	= 40 Btu/sec per 100,000 Btu
Heat Release rate =	112 Btu/sec

RBCCW pump 2A (Unit 2 Reactor Building EL 593)

Number of components =	1
Combustible Loading (Btu) =	7.50E+04
Heat Release rate per Component	= 40 Btu/sec per 100,000 Btu
Heat Release rate =	30 Btu/sec

RBCCW pump 2B (Unit 2 Reactor Building EL 593)

Number of components =	1
Combustible Loading (Btu) =	7.50E+04
Heat Release rate per Component	= 40 Btu/sec per 100,000 Btu
Heat Release rate =	30 Btu/sec

Shutdown Bd Rm 2C/2D A/C Units (Unit 2 reactor Building EL 604)

Number of components =	1
Combustible Loading (Btu) =	7.00E+04
Heat Release rate per Component	= 40 Btu/sec per 100,000 Btu
Heat Release rate =	30 Btu/sec

480V RMOV BD 2E (Unit 2 Reactor Building EL 621)

Number of Cubicles =	12
Combustible Loading (Btu) =	9.80E+05
Combustible Loading per Cubicle	8.17E+04
Heat Release Rate per Cubicle =	53 btu/sec

Vent openings are present in this board, cubicles stacked 2 high

Heat Release rate per Electrical B	106	(2 x HRR/cubicle)
Combustible Loading for 2 cubicl	1.63E+05	

MG Sets 2DN and 2DE (Unit 2 Reactor Building EL 621)

Number of components = 1
Combustible Loading (Btu) = 2.80E+05
Heat Release rate per Component = 40 Btu/sec per 100,000 Btu
Heat Release rate = 112 Btu/sec

4KV-480V Transformer (Unit 2 Reactor Building EL 621)

Number of components = 1
Combustible Loading (Btu) = 2.80E+05
Heat Release rate per Component = 40 Btu/sec per 100,000 Btu
Heat Release rate = 112 Btu/sec

2-LPNL-025-0031 (Unit 2 Reactor Building EL 621)

Number of Cubicles = 3
Combustible Loading (Btu) = 9.80E+05
Combustible Loading per Cubicle = 3.27E+05
Heat Release Rate per Cubicle = 53 btu/sec
No vent openings are present in this board
Heat Release rate per panel = 106 (2 x HRR/cubicle)
Combustible Loading for 2 cubicles = 6.53E+05

4KV RPT BD 2-1 and 2-2 (Unit 2 Reactor Building EL 621)
Number of components = 1
Combustible Loading (Btu) = 2.80E+05
Heat Release rate per Component = 53 Btu/sec
Since vents are present in this panel, use twice the HRR
Heat Release rate = 106 Btu/sec

Panel 25-3 and Panel 25-9(Unit 2 Reactor Building EL 621)
Number of components = 1
Combustible Loading (Btu) = 1.40E+04
These are low voltage instrument panels and have very limited combustibles.
However, assume a HRR of 50 btu/sec per panel.

240V Lighting Board 2A (Unit 2 Reactor Building EL 621)

Number of Cubicles = 10
Combustible Loading (Btu) = 5.60E+05
Combustible Loading per Cubicle 5.60E+04
Heat Release Rate per Cubicle = 53 btu/sec
No vent openings are present in this board
Heat Release rate per Electrical E 106 (2 x HRR/cubicle)
Combustible Loading for 2 cubicl 1.12E+05

MG set 2DA and 2EN (Unit 2 reactor building El 621)
Number of components = 1
Combustible Loading (Btu) = 2.80E+05
Heat Release rate per Component = 40 Btu/sec per 100,000 Btu
Heat Release rate = 112 Btu/sec

LPNL-25-23 and 24 (Unit 2 Reactor Building EL 621)

Number of Cubicles = 4
Combustible Loading (Btu) = 1.26E+06
Combustible Loading per Cubicle 3.15E+05
Heat Release Rate per Cubicle = 53 btu/sec
Vent openings are present in one of the compartments.
Heat Release rate per panel = 106 (2 x HRR/cubicle)
Combustible Loading for 2 cubicl 6.30E+05

240V Lighting Board 2B (Unit 2 Reactor Building EL 639 -south side)
Number of Cubicles = 10
Combustible Loading (Btu) = 5.60E+05
Combustible Loading per Cubicle 5.60E+04
Heat Release Rate per Cubicle = 53 btu/sec
No vent openings are present in this board
Heat Release rate per Electrical E 106 (2 x HRR/cubicle)
Combustible Loading for 2 cubicl 1.12E+05

Drywell Torus Compressor (Reactor Building Unit 2, EL 565)

Oil:
Spill Area: 54.00 ft2/gal (Table 3, Reference 5)
Pool Area for 1/2 pint spill: 3.38 ft2 (54 ft2/gal * 1gal/16 pints)
From reference 5, Table 2E, Thermophysical properties similar to transformer oil:
Heat of Combustion (Hc): 19998.00 Btu/lbm
Ideal unit mass Loss rate (n 0.008 lbm/sec ft2
Combustion efficiency (%): 0.85

Unit Heat Release rate (q") Hc * m" * %
Unit Heat Release rate q" 19998 * .008 * .85 135.99 Btu/sec/ft2
Peak Heat Release rate = Unit Heat Release rate * Spill Area
136 * 3.3 = 450 Btu/sec

Motor:
Number of components = 1
Combustible Loading (Btu) = 1.00E+05
Heat Release rate per Component = 40 Btu/sec per 100,000 Btu
Heat Release rate = 40 Btu/sec

SLC Pumps A and B Motor (Unit 2 Reactor Building EL 639- north side)

Motor:
Number of components = 1
Combustible Loading (Btu) = 2.80E+05
Heat Release rate per Component = 40 Btu/sec per 100,000 Btu
Heat Release rate = 112 Btu/sec

Oil:
Spill Area: 54.00 ft2/gal (Table 3, Reference 5)
Pool Area for 1/2 pint spill: 3.38 ft2 (54 ft2/gal * 1gal/16 pints)
From reference 5, Table 2E, Thermophysical properties similar to transformer oil:
Heat of Combustion (Hc): 19998.00 Btu/lbm
Ideal unit mass Loss rate (n 0.008 lbm/sec ft2
Combustion efficiency (%): 0.85

Unit Heat Release rate (q") Hc * m" * %
Unit Heat Release rate q" 19998 * .008 * .85 135.99 Btu/sec/ft2
Peak Heat Release rate = Unit Heat Release rate * Spill Area
136 * 3.3 = 450 Btu/sec

ATTACHMENT B

FIRE AREA/COMPARTMENT IGNITION FREQUENCY CALCULATIONS

Attachment B

Fire Area/Compartment Ignition Frequency Calculations

This attachment contains the detailed fire area/compartment ignition frequency calculations. The input required for these calculations includes:

- Number of various plant locations, as shown in Table 2-1.
- Plant Wide Components. This information was extracted from Reference 3, related system flow diagrams and plant walkdowns.
- Cables - Heat of Combustion, which was extracted from Reference 3.

The generic fire frequencies and weighting factors used in these calculations are in accordance with the EPRI FIVE documentation, as described in EPRI report TR-100370. Where specific fire frequencies were not provided for plant areas, such as computer rooms, mechanical equipment rooms, etc., conservative assumptions were made.

The performance of these calculations consists of two main steps:

First, fire ignition frequency that can be assigned to specific plant areas, such as switchgear area fires, is allocated to similar areas within the plant.

Second, identified plant-wide components, such as battery chargers, are located and the associated fire ignition frequency is assigned to the respective area.

The calculation of fire ignition frequency for each plant area is then shown on an individual sheet. These values are summarized in Table 4-1 of the main report.

The fire ignition frequencies that are assigned to each of the identified plant areas are based on the data from the Fire Events Database (EPRI report NSAC/178L), which identifies 753 plant fires over approximately 1200 reactor years of experience.

BROWNS FERRY NUCLEAR PLANT
PLANT WIDE COMPONENTS

FIRE PROTECTION PANELS:

RB 1 (FA-1)	- 5	
RB 2 (FA-2)	- 9	(includes 3 new panels per DCN W17907 and existing 6 panels which are abandoned in place)
RB 3 (FA-3)	- 11	(includes 3 new panels being installed under DCN W17908)
DGB 1/2 (FA-20)	- 2	(FP panel and CO_2 switch panel)
DBG 3 (FA-21)	- 2	(FP panel and CO_2 switch panel)
IPS (FA-25-1)	- 1	
CB (FA-16)	- 6	(includes 1 on El 617, 2 on El 606, 1 on El 593, 1 in process computer room and 1 in Unit 3 computer room)
TB (FA-25-3)	- 4	
TOTAL	40	

BATTERY CHARGERS:

DGB 1/2	- 8	(FA-20)
DGB 3	- 9	(FA-21)
BAT & BBR 1	- 3	(250V, 48V and 24V neutron) (FA-17)
BAT & BBR 2	- 3	(250V, 48V and 24V neutron) (FA-18)
BAT & BBR 3	- 3	(250V, 48V and 24V neutron) (FA-19)
Communication BAT BD Room	- 3	(FA-16)
SDBR - C	- 2	(FA-9)
SDBR - A	- 1	(El 586) (FA-25-3)
TOTAL	34	

TRANSFORMERS:

BAT BD Room 1	- 2	(dry type) (FA-17)
DGB 1/2	- 3	(FA-20)
Elect BD Room Unit 3-621	- 1	(dry type transformer) (FA-13)
Elect BD Room Unit 3-593	- 1	(dry type transformer) (FA-12)
SDBR-A-621	- 1	(FA-5)
RB-519, Unit 1	- 1	(FA-1)
RB-519, Unit 3	- 1	(FA-3)
RB-593, Unit 2	- 1	(Unit 2 preferred AC transformer) (FA-2)
RB-621, Unit 1	- 3	(4KV/480 and 240V Lighting Board) (FA-1)
RB-621, Unit 2	- 3	(4KV/480 and 240V Lighting Board) (FA-2)
RB-621, Unit 3	- 3	(4KV/480 and 240V Lighting Board) (FA-3)
RB-639, Unit 1	- 1	(4KV/480) (FA-1)
RB-639, Unit 2	- 2	(4KV/480 and 240V Lighting Board) (FA-2)
RB-639, Unit 3	- 1	(4KV/480) (FA-3)
TB 617	- 1	(GE Transformer) (FA-25-3)
TB 604	- 18	(FA-25-3)
TB 584	- 1	(FA-25-3)
Intake Pump STA	- 3	(FA-25-1)
TOTAL	47	

BROWNS FERRY NUCLEAR PLANT
PLANT WIDE COMPONENTS

AIR COMPRESSORS:

RB, Unit 1	-	3 (FA-1)
RB, Unit 2	-	3 (FA-2)
RB, Unit 3	-	3 (FA-3)
TB (604)	-	6 (FA-25-3)
TB (565)	-	10 (FA-25-3)
CB (606, MER)	-	1 (FA-16)
TOTAL		26

VENTILATION SUBSYSTEMS:

RB Unit 1	-	19 (FA-1)
RB Unit 2	-	20 (includes 4 for SDBR C/D) (FA-2)
RB Unit 3	-	21 (includes 4 for SDBR E/F) (FA-3)
Turbine Bldg 1	-	36 (includes 1 booster fan) (FA-25-3)
Turbine Bldg 2	-	38 (includes AHU for CB) (FA-25-3)
Turbine Bldg 3	-	37 (includes record storage Elect BD Room) (FA-25-3)
Control Bldg	-	45 (30-El 617, FA-16)
		(2-El 606, FA-16)
		(1-El 593, FA-4)
		(1-El 593, FA-12)
		(3-El 593, FA-16)
		(4-El 621, FA-5)
		(4-El 621, FA-9)
Radwaste	-	17 (FA-25-3)
DGB 1/2	-	20 (FA-20)
DGB 3	-	28 (FA-21)
Intake Pump Sta	-	8 (FA-25-1)
TOTAL		289

RPS MG Sets:

Control Bldg	-	6 (FA-16)
Turbine 617	-	2 (FA-25-3)
RB-1 639	-	4 (FA-1)
RB-2 639	-	4 (FA-2)
RB-3 639	-	2 (FA-3)
RB-1 621	-	4 (FA-1)
RB-2 621	-	2 (FA-2)
RB-3 621	-	4 (FA-3)
MG Set Room	-	1 (FA-17)
MG Set Room	-	1 (FA-18)
MG Set Room	-	1 (FA-19)
TOTAL		31

OFFGAS RECOMBINERS:

Unit 1	-	1 (FA-25-3)
Unit 2	-	1 (FA-25-3)
Unit 3	-	1 (FA-25-3)
TOTAL		3

CABLES
HEAT OF COMBUSTION

Ignition frequency (five)/cables heat of comb

DESCRIPTION	BTU	TOTAL (BTU)
REACTOR BUILDING, UNIT 2 (FIRE AREA 2)		
EL 639	63,375,144	
EL 621	181,402,998	
EL 593	557,653,714	
	8,962,800	
EL 565	427,923,910	
		1,239,318,566
REACTOR BUILDING, UNIT 1 (FIRE AREA 1)		1,239,318,566
(Assume combustible heat load similar to unit 2)		
REACTOR BUILDING, UNIT 3 (FIRE AREA 3)		1,239,318,566
(Assume combustible heat load similar to unit 2)		
CONTROL BUILDING (FIRE AREA 16)		
Unit 1/2 main control room	0	
Mechanical equipment room EL 617	30,000,000	
Cable spreading room A	749,713,692	
Cable spreading room B	621,427,433	
Stairway C4 El 606	5,571,720	
Stairway C2 El 606	13,676,040	
Stairway C6 El 606	112,000	
Auxiliary instrument room 1	18,065,880	
Auxiliary instrument room 2	19,078,920	
Auxiliary instrument room 3	20,598,480	
Unit 1/2 computer room	11,842,436	
Unit 3 computer room	13,507,200	
Communications room	12,995,573	
		1,516,589,374
4KV SHUTDOWN BD C (FIRE AREA 9)		5,000,000
BATTERY BOARD ROOM 2 (FIRE AREA 18)		506,520
DIESEL GENERATOR BUILDING UNIT 1/2 (FIRE AREA 20)		
Pipe tunnel		21,093,135
DIESEL GENERATOR BUILDING UNIT 3 (FIRE AREA 21)		
480V Diesel auxiliary board 3EB	6,470,100	
Pipe tunnel	24,237,990	
Stairs	4,479,300	
		35,187,390
4KV SHUTDOWN BD 3EA AND 3EB (FIRE AREA 22)		15,279,390
4KV SHUTDOWN BD 3EC AND 3ED (FIRE AREA 23)		14,582,610
4KV BUS TIE BD (FIRE AREA 24)		7,614,810
TURBINE BUILDING (FIRE AREA 25)		
Unit 1,2,3 turbine building (fire compt. 25-3)	6,165,729,000	
Pipe tunnel (fire compt. 25-2)	5,000,000	
Intake pump station (fire compt. 25-1)	271,630,557	
Radwaste building	167,463,262	
		6,609,822,819
TOTAL CABLES COMBUSTIBLE LOAD		11,943,631,746

Note: Cables heat of combustion is only identified for those areas which have exposed cables.
Other areas either do not have any exposed cables or cables are in conduits.

BROWNS FERRY NUCLEAR PLANT UNIT 2

FIRE INDUCED VULNERABILITY EVALUATION

FIRE AREA/COMPARTMENT FIRE FREQUENCY

PHASE 11 (STEP 1) EVALUATION

		FIRE AREA / COMPARTMENT IGNITION FREQUENCY				
		PHASE II (STEP 1) EVALUATION				
		PLANT LOCATION: REACTOR BUILDING (FIRE AREA 1)				
Components	Generic Fire Frequency	Location Weighting Factor (WL)	Number of Components in Fire Area	Total Number In All Plant Locations	Ignition Source Weighting Factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location						
Electric Cabinets	5.00E-02	1				5.000E-02
Pumps	2.50E-02	1				2.500E-02
Plant Wide						
Fire Prot. Panel	2.40E-03	3	5	40	1.25E-01	9.000E-04
RPS MG Set	5.50E-03	3	8	31	2.58E-01	4.258E-03
Non-Qual. Cable	6.30E-03	3	1239	12000	1.03E-01	1.951E-03
Non-Qual. JB	1.60E-03	3	1239	12000	1.03E-01	4.956E-04
Transformers	7.90E-03	3	5	47	1.06E-01	2.521E-03
Battery Chargers	4.00E-03	3		34		0.000E+00
Air Compressors	4.70E-03	3	3	26	1.15E-01	1.627E-03
Vent. Subsystem	9.50E-03	3	19	289	6.57E-02	1.874E-03
			Sum of Ignition Sources WF for Transient source	Total Number of Compartments		
Transients	1.30E-03	3	5	34	1.47E-01	5.735E-04
Cable fire-Welding	5.10E-03	3	1	34	2.94E-02	4.500E-04
Transient Fire-weld	3.10E-02	3	1	34	2.94E-02	2.735E-03
				TOTAL		**9.239E-02**

		FIRE AREA / COMPARTMENT IGNITION FREQUENCY				
		PHASE II (STEP 1) EVALUATION				
		PLANT LOCATION: REACTOR BUILDING (FIRE ZONE 2-1)				
Components	Generic Fire Frequency	Location Weighting Factor (WL)	Number of Components in Fire Zone	Total Number In All Plant Locations*	Ignition Source Weighting Factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location						
Electric Cabinets	5.00E-02	1	10	63	1.59E-01	7.937E-03
Pumps	2.50E-02	1	5	24	2.08E-01	5.208E-03
Plant Wide						
Fire Prot. Panel	2.40E-03	3	3	40	7.50E-02	5.400E-04
RPS MG Set	5.50E-03	3		31	0.00E+00	0.000E+00
Non-Qual. Cable	6.30E-03	3	214	12000	1.78E-02	3.371E-04
Non-Qual. JB	1.60E-03	3	214	12000	1.78E-02	8.560E-05
Transformers	7.90E-03	3		47	0.00E+00	0.000E+00
Battery Chargers	4.00E-03	3		34	0.00E+00	0.000E+00
Air Compressors	4.70E-03	3	1	26	3.85E-02	5.423E-04
Vent. Subsystem	9.50E-03	3	3	289	1.04E-02	2.958E-04
			Sum of Ignition Source WF For Transients	Total Number of Compartments		
Transients	1.30E-03	3	5	34	1.47E-01	5.735E-04
Cable fire-Welding	5.10E-03	3	1	34	2.94E-02	4.500E-04
Transient Fire-weldin	3.10E-02	3	1	34	2.94E-02	2.735E-03
				TOTAL		1.870E-02
Note:						
* For "Plant Locations", the total number represents components in the Fire Area 2.						

		FIRE AREA / COMPARTMENT IGNITION FREQUENCY				
		PHASE II (STEP 1) EVALUATION				
		PLANT LOCATION: REACTOR BUILDING (FIRE ZONE 2-2)				
Components	Generic Fire Frequency	Location Weighting Factor (WL)	Number of Components In Fire Zone	Total Number In All Plant Locations*	Ignition Source Weighting Factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Electric Cabinets	5.00E-02	1	11	63	1.75E-01	8.730E-03
Pumps	2.50E-02	1	4	24	1.67E-01	4.167E-03
Plant Wide						
Fire Prot. Panel	2.40E-03	3		40	0.00E+00	0.000E+00
RPS MG Set	5.50E-03	3		31	0.00E+00	0.000E+00
Non-Qual. Cable	6.30E-03	3	214	12000	1.78E-02	3.371E-04
Non-Qual. JB	1.60E-03	3	214	12000	1.78E-02	8.560E-05
Transformers	7.90E-03	3		47	0.00E+00	0.000E+00
Battery Chargers	4.00E-03	3		34	0.00E+00	0.000E+00
Air Compressors	4.70E-03	3		26	0.00E+00	0.000E+00
Vent. Subsystem	9.50E-03	3	3	289	1.04E-02	2.958E-04
			Sum of Ignitior Source WF For Transients	Total Number of Compartments		
Transients	1.30E-03	3	5	34	1.47E-01	5.735E-04
Cable fire-Welding	5.10E-03	3	1	34	2.94E-02	4.500E-04
Transient Fire-weldin	3.10E-02	3	1	34	2.94E-02	2.735E-03
				TOTAL		1.737E-02
Note:						
* For "Plant Locations", the total number represents components in the Fire Area 2.						

FIRE AREA / COMPARTMENT IGNITION FREQUENCY

PHASE II (STEP 1) EVALUATION

PLANT LOCATION: REACTOR BUILDING (FIRE ZONE 2-3)

Components	Generic Fire Frequency	Location Weighting Factor (WL)	Number of Components in Fire Zone	Total Number In All Plant Locations*	Ignition Source Weighting Factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location						
Electric Cabinets	5.00E-02	1	1	63	1.59E-02	7.937E-04
Pumps	2.50E-02	1		24	0.00E+00	0.000E+00
Plant Wide						
Fire Prot. Panel	2.40E-03	3		40	0.00E+00	0.000E+00
RPS MG Set	5.50E-03	3		31	0.00E+00	0.000E+00
Non-Qual. Cable	6.30E-03	3	141	12000	1.18E-02	2.221E-04
Non-Qual. JB	1.60E-03	3	141	12000	1.18E-02	5.640E-05
Transformers	7.90E-03	3	1	47	2.13E-02	5.043E-04
Battery Chargers	4.00E-03	3		34	0.00E+00	0.000E+00
Air Compressors	4.70E-03	3		26	0.00E+00	0.000E+00
Vent. Subsystem	9.50E-03	3		289	0.00E+00	0.000E+00
			Sum of Ignition Source WF For Transients	Total Number of Compartments		
Transients	1.30E-03	3	5	34	1.47E-01	5.735E-04
Cable fire-Welding	5.10E-03	3	1	34	2.94E-02	4.500E-04
Transient Fire-weldin	3.10E-02	3	1	34	2.94E-02	2.735E-03
				TOTAL		5.335E-03

Note:

* For "Plant Locations", the total number represents components in the Fire Area 2.

FIRE AREA / COMPARTMENT IGNITION FREQUENCY
PHASE II (STEP 1) EVALUATION
PLANT LOCATION: REACTOR BUILDING (FIRE ZONE 2-4)

Components	Generic Fire Frequency	Location Weighting Factor (WL)	Number of Components In Fire Zone	Total Number In All Plant Locations*	Ignition Source Weighting Factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location						
Electric Cabinets	5.00E-02	1	13	63	2.06E-01	1.032E-02
Pumps	2.50E-02	1	4	24	1.67E-01	4.167E-03
Plant Wide						
Fire Prot. Panel	2.40E-03	3	2	40	5.00E-02	3.600E-04
RPS MG Set	5.50E-03	3		31	0.00E+00	0.000E+00
Non-Qual. Cable	6.30E-03	3	426	12000	3.55E-02	6.710E-04
Non-Qual. JB	1.60E-03	3	426	12000	3.55E-02	1.704E-04
Transformers	7.90E-03	3		47	0.00E+00	0.000E+00
Battery Chargers	4.00E-03	3		34	0.00E+00	0.000E+00
Air Compressors	4.70E-03	3	2	26	7.69E-02	1.085E-03
Vent. Subsystem	9.50E-03	3	4	289	1.38E-02	3.945E-04
			Sum of Ignition Source WF For Transients	Total Number of Compartments		
Transients	1.30E-03	3	5	34	1.47E-01	5.735E-04
Cable fire-Welding	5.10E-03	3	1	34	2.94E-02	4.500E-04
Transient Fire-weldin	3.10E-02	3	1	34	2.94E-02	2.735E-03
				TOTAL		2.092E-02

Note:

* For "Plant Locations", the total number represents components in the Fire Area 2.

		FIRE AREA / COMPARTMENT IGNITION FREQUENCY				
		PHASE II (STEP 1) EVALUATION				
		PLANT LOCATION: REACTOR BUILDING (FIRE ZONE 2-5)				
Components	Generic Fire Frequency	Location Weighting Factor (WL)	Number of Components in Fire Zone	Total Number In All Plant Locations*	Ignition Source Weighting Factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location						
Electric Cabinets	5.00E-02	1	20	63	3.17E-01	1.587E-02
Pumps	2.50E-02	1	5	24	2.08E-01	5.208E-03
Plant Wide						
Fire Prot. Panel	2.40E-03	3	4	40	1.00E-01	7.200E-04
RPS MG Set	5.50E-03	3	2	31	6.45E-02	1.065E-03
Non-Qual. Cable	6.30E-03	3	181	12000	1.51E-02	2.851E-04
Non-Qual. JB	1.60E-03	3	181	12000	1.51E-02	7.240E-05
Transformers	7.90E-03	3	3	47	6.38E-02	1.513E-03
Battery Chargers	4.00E-03	3		34	0.00E+00	0.000E+00
Air Compressors	4.70E-03	3		26	0.00E+00	0.000E+00
Vent. Subsystem	9.50E-03	3	2	289	6.92E-03	1.972E-04
			Sum of Ignition Source WF For Transients	Total Number of Compartments		
Transients	1.30E-03	3	5	34	1.47E-01	5.735E-04
Cable fire-Welding	5.10E-03	3	1	34	2.94E-02	4.500E-04
Transient Fire-weldin	3.10E-02	3	1	34	2.94E-02	2.735E-03
				TOTAL		2.869E-02
Note:						
* For "Plant Locations", the total number represents components in the Fire Area 2.						

FIRE AREA / COMPARTMENT IGNITION FREQUENCY
PHASE II (STEP 1) EVALUATION
PLANT LOCATION: REACTOR BUILDING · (FIRE ZONE 2-6)

Components	Generic Fire Frequency	Location Weighting Factor (WL)	Number of Components ir Fire Zone	Total Number In All Plant Locations*	Ignition Source Weighting Factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location						
Electric Cabinets	5.00E-02	1	8	63	1.27E-01	6.349E-03
Pumps	2.50E-02	1	6	24	2.50E-01	6.250E-03
Plant Wide						
Fire Prot. Panel	2.40E-03	3		40	0.00E+00	0.000E+00
RPS MG Set	5.50E-03	3	4	31	1.29E-01	2.129E-03
Non-Qual. Cable	6.30E-03	3	63	12000	5.25E-03	9.923E-05
Non-Qual. JB	1.60E-03	3	63	12000	5.25E-03	2.520E-05
Transformers	7.90E-03	3	2	47	4.26E-02	1.009E-03
Battery Chargers	4.00E-03	3		34	0.00E+00	0.000E+00
Air Compressors	4.70E-03	3		26	0.00E+00	0.000E+00
Vent. Subsystem	9.50E-03	3		289	0.00E+00	0.000E+00
			Sum of Ignitior Source WF For Transients	Total Number of Compartments		
Transients	1.30E-03	3	5	34	1.47E-01	5.735E-04
Cable fire-Welding	5.10E-03	3	1	34	2.94E-02	4.500E-04
Transient Fire-weldin	3.10E-02	3	1	34	2.94E-02	2.735E-03
				TOTAL		1.962E-02

Note:

* For "Plant Locations", the total number represents components in the Fire Area 2.

		FIRE AREA / COMPARTMENT IGNITION FREQUENCY				
		PHASE II (STEP 1) EVALUATION				
		PLANT LOCATION: UNIT 3 REACTOR BUILDING (FIRE AREA 3)				
Components	Generic Fire Frequency	Location Weighting Factor (WL)	Number of Components In Fire Area	Total Number In All Plant Locations	Ignition Source Weighting Factor (WI)	Fire Compartme Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location						
Electric Cabinets	5.00E-02	1	1	1	1.00E+00	5.000E-02
Pumps	2.50E-02	1	1	1	1.00E+00	2.500E-02
Plant Wide						
Fire Prot. Panel	2.40E-03	3	11	40	2.75E-01	1.980E-03
RPS MG Set	5.50E-03	3	6	31	1.94E-01	3.194E-03
Non-Qual. Cable	6.30E-03	3	1239	12000	1.03E-01	1.951E-03
Non-Qual. JB	1.60E-03	3	1239	12000	1.03E-01	4.956E-04
Transformers	7.90E-03	3	5	47	1.06E-01	2.521E-03
Battery Chargers	4.00E-03	3		34		0.000E+00
Air Compressors	4.70E-03	3	3	26	1.15E-01	1.627E-03
Vent. Subsystem	9.50E-03	3	21	289	7.27E-02	2.071E-03
			Sum of Ignition Sources WF for Transient source	Total Number of Compartments		
Transients	1.30E-03	3	5	34	1.47E-01	5.735E-04
Cable fire-Welding	5.10E-03	3	1	34	2.94E-02	4.500E-04
Transient Fire-weld	3.10E-02	3	1	34	2.94E-02	2.735E-03
				TOTAL		9.260E-02

		FIRE AREA / COMPARTMENT IGNITION FREQUENCY				
		PHASE II (STEP 1) EVALUATION				
		PLANT LOCATION: SWITCHGEAR ROOM (FIRE AREA 4)				
Components	Generic Fire Frequency	Location Weighting Factor (WL)*	Number of Components in Fire Area	Total Number In All Plant Locations	Ignition Source Weighting Factor (WI)	Fire Compartme Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location						
Electric Cabinets	1.50E-02	0.2				3.000E-03
Plant Wide						
Fire Prot. Panel	2.40E-03	3		40	0.00E+00	0.000E+00
RPS MG Set	5.50E-03	3		31	0.00E+00	0.000E+00
Non-Qual. Cable	6.30E-03	3		12000	0.00E+00	0.000E+00
Non-Qual. JB	1.60E-03	3		12000	0.00E+00	0.000E+00
Transformers	7.90E-03	3		47	0.00E+00	0.000E+00
Battery Chargers	4.00E-03	3		34	0.00E+00	0.000E+00
Air Compressors	4.70E-03	3		26	0.00E+00	0.000E+00
Vent. Subsystem	9.50E-03	3	1	289	3.46E-03	9.862E-05
			Sum of Ignitior Sources WF for Transient source	Total Number of Compartments		
Transients	1.30E-03	3	4	34	1.18E-01	4.588E-04
Cable fire-Welding	5.10E-03	3	1	34	2.94E-02	4.500E-04
Transient Fire-weld	3.10E-02	3	1	34	2.94E-02	2.735E-03
				TOTAL		**6.743E-03**

NOTE:

* For Plant Locations the weighting factor is the number of units (3) divided by number of switchgear rooms (15).

		FIRE AREA / COMPARTMENT IGNITION FREQUENCY				
		PHASE II (STEP 1) EVALUATION				
		PLANT LOCATION: SWITCHGEAR ROOM (FIRE AREA 5)				
Components	Generic Fire Frequency	Location Weighting Factor (WL)*	Number of Components in Fire Area	Total Number In All Plant Locations	Ignition Source Weighting Factor (WI)	Fire Compartme Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location						
Electric Cabinets	1.50E-02	0.2				3.000E-03
Batteries (1)	3.20E-03	0.1				3.20E-04
Plant Wide						
Fire Prot. Panel	2.40E-03	3		40	0.00E+00	0.000E+00
RPS MG Set	5.50E-03	3		31	0.00E+00	0.000E+00
Non-Qual. Cable	6.30E-03	3		12000	0.00E+00	0.000E+00
Non-Qual. JB	1.60E-03	3		12000	0.00E+00	0.000E+00
Transformers	7.90E-03	3	1	47	2.13E-02	5.043E-04
Battery Chargers	4.00E-03	3	2	34	5.88E-02	7.059E-04
Air Compressors	4.70E-03	3		26	0.00E+00	0.000E+00
Vent. Subsystem	9.50E-03	3	4	289	1.38E-02	3.945E-04
			Sum of Ignition Sources WF for Transient source	Total Number of Compartments		
Transients	1.30E-03	3	4	34	1.18E-01	4.588E-04
Cable fire-Welding	5.10E-03	3	1	34	2.94E-02	4.500E-04
Transient Fire-weld	3.10E-02	3	1	34	2.94E-02	2.735E-03
				TOTAL		8.569E-03

Note: (1) Due to the presence of few batteries in the area, they are being included as contribution to the ignition frequency. 10% of the ignition frequency of a typical unit battery room has been added.

* For Plant Locations the weighting factor is the number of units (3) divided by number of switchgear rooms (15).

		FIRE AREA / COMPARTMENT IGNITION FREQUENCY				
		PHASE II (STEP 1) EVALUATION				
		PLANT LOCATION: SWITCHGEAR ROOM (FIRE AREA 6)				
Components	Generic Fire Frequency	Location Weighting Factor (WL)*	Number of Components In Fire Area	Total Number In All Plant Locations	Ignition Source Weighting Factor (WI)	Fire Compartme Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location						
Electric Cabinets	1.50E-02	0.2				3.000E-03
Plant Wide						
Fire Prot. Panel	2.40E-03	3		40	0.00E+00	0.000E+00
RPS MG Set	5.50E-03	3		31	0.00E+00	0.000E+00
Non-Qual. Cable	6.30E-03	3		12000	0.00E+00	0.000E+00
Non-Qual. JB	1.60E-03	3		12000	0.00E+00	0.000E+00
Transformers	7.90E-03	3		47	0.00E+00	0.000E+00
Battery Chargers	4.00E-03	3		34	0.00E+00	0.000E+00
Air Compressors	4.70E-03	3		26	0.00E+00	0.000E+00
Vent. Subsystem	9.50E-03	3		289	0.00E+00	0.000E+00
			Sum of Ignitior Sources WF foi Transient source	Total Number of Compartments		
Transients	1.30E-03	3	4	34	1.18E-01	4.588E-04
Cable fire-Welding	5.10E-03	3	1	34	2.94E-02	4.500E-04
Transient Fire-weld	3.10E-02	3	1	34	2.94E-02	2.735E-03
				TOTAL		**6.644E-03**

NOTE: No Plant Wide ignition sources (except transients) exist in this area.

* For Plant Locations the weighting factor is the number of units (3) divided by number of switchgear rooms (15).

		FIRE AREA / COMPARTMENT IGNITION FREQUENCY				
		PHASE II (STEP 1) EVALUATION				
		PLANT LOCATION: SWITCHGEAR ROOM (FIRE AREA 7)				
Components	Generic Fire Frequency	Location Weighting Factor (WL)*	Number of Components In Fire Area	Total Number In All Plant Locations	Ignition Source Weighting Factor (WI)	Fire Compartme Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location						
Electric Cabinets	1.50E-02	0.2				3.000E-03
Plant Wide						
Fire Prot. Panel	2.40E-03	3		40	0.00E+00	0.000E+00
RPS MG Set	5.50E-03	3		31	0.00E+00	0.000E+00
Non-Qual. Cable	6.30E-03	3		12000	0.00E+00	0.000E+00
Non-Qual. JB	1.60E-03	3		12000	0.00E+00	0.000E+00
Transformers	7.90E-03	3		47	0.00E+00	0.000E+00
Battery Chargers	4.00E-03	3		34	0.00E+00	0.000E+00
Air Compressors	4.70E-03	3		26	0.00E+00	0.000E+00
Vent. Subsystem	9.50E-03	3		289	0.00E+00	0.000E+00
			Sum of Ignition Sources WF for Transient source	Total Number of Compartments		
Transients	1.30E-03	3	4	34	1.18E-01	4.588E-04
Cable fire-Welding	5.10E-03	3	1	34	2.94E-02	4.500E-04
Transient Fire-weld	3.10E-02	3	1	34	2.94E-02	2.735E-03
				TOTAL		**6.644E-03**

NOTE: No Plant Wide ignition sources (except transients) exist in this area.

* For Plant Locations the weighting factor is the number of units (3) divided by number of switchgear rooms (15).

		FIRE AREA / COMPARTMENT IGNITION FREQUENCY				
		PHASE II (STEP 1) EVALUATION				
		PLANT LOCATION: SWITCHGEAR ROOM (FIRE AREA 8)				
Components	Generic Fire Frequency	Location Weighting Factor (WL)*	Number of Components in Fire Area	Total Number In All Plant Locations	Ignition Source Weighting Factor (WI)	Fire Compartme Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location						
Electric Cabinets	1.50E-02	0.2				3.000E-03
Plant Wide						
Fire Prot. Panel	2.40E-03	3		40	0.00E+00	0.000E+00
RPS MG Set	5.50E-03	3		31	0.00E+00	0.000E+00
Non-Qual. Cable	6.30E-03	3		12000	0.00E+00	0.000E+00
Non-Qual. JB	1.60E-03	3		12000	0.00E+00	0.000E+00
Transformers	7.90E-03	3		47	0.00E+00	0.000E+00
Battery Chargers	4.00E-03	3		34	0.00E+00	0.000E+00
Air Compressors	4.70E-03	3		26	0.00E+00	0.000E+00
Vent. Subsystem	9.50E-03	3		289	0.00E+00	0.000E+00
			Sum of Ignition Sources WF for Transient source	Total Number of Compartments		
Transients	1.30E-03	3	4	34	1.18E-01	4.588E-04
Cable fire-Welding	5.10E-03	3	1	34	2.94E-02	4.500E-04
Transient Fire-weld	3.10E-02	3	1	34	2.94E-02	2.735E-03
				TOTAL		**6.644E-03**

NOTE: No Plant Wide ignition sources (except transients) exist in this area.

* For Plant Locations the weighting factor is the number of units (3) divided by number of switchgear rooms (15).

		FIRE AREA / COMPARTMENT IGNITION FREQUENCY				
		PHASE II (STEP 1) EVALUATION				
		PLANT LOCATION: SWITCHGEAR ROOM (FIRE AREA 9)				
Components	Generic Fire Frequency	Location Weighting Factor (WL)*	Number of Components in Fire Area	Total Number In All Plant Locations	Ignition Source Weighting Factor (WI)	Fire Compartme Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location						
Electric Cabinets	1.50E-02	0.2				3.000E-03
Batteries (1)	3.20E-03	0.1				3.20E-04
Plant Wide						
Fire Prot. Panel	2.40E-03	3		40	0.00E+00	0.000E+00
RPS MG Set	5.50E-03	3		31	0.00E+00	0.000E+00
Non-Qual. Cable	6.30E-03	3	5	12000	4.17E-04	7.875E-06
Non-Qual. JB	1.60E-03	3	5	12000	4.17E-04	2.000E-06
Transformers	7.90E-03	3		47	0.00E+00	0.000E+00
Battery Chargers	4.00E-03	3	2	34	5.88E-02	7.059E-04
Air Compressors	4.70E-03	3		26	0.00E+00	0.000E+00
Vent. Subsystem	9.50E-03	3	4	289	1.38E-02	3.945E-04
			Sum of Ignition Sources WF for Transient source	Total Number of Compartments		
Transients	1.30E-03	3	4	34	1.18E-01	4.588E-04
Cable fire-Welding	5.10E-03	3	1	34	2.94E-02	4.500E-04
Transient Fire-weld	3.10E-02	3	1	34	2.94E-02	2.735E-03
				TOTAL		**8.074E-03**

Note: (1) Due to the presence of few batteries in the area, they are being included as contribution to the ignition frequency. 10% of the ignition frequency of a typical unit battery room has been added.

* For Plant Locations the weighting factor is the number of units (3) divided by number of switchgear rooms (15).

		FIRE AREA / COMPARTMENT IGNITION FREQUENCY				
		PHASE II (STEP 1) EVALUATION				
		PLANT LOCATION: SWITCHGEAR ROOM (FIRE AREA 10)				
Components	Generic Fire Frequency	Location Weighting Factor (WL)*	Number of Components in Fire Area	Total Number In All Plant Locations	Ignition Source Weighting Factor (WI)	Fire Compartme Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location						
Electric Cabinets	1.50E-02	0.2				3.000E-03
Plant Wide						
Fire Prot. Panel	2.40E-03	3		40	0.00E+00	0.000E+00
RPS MG Set	5.50E-03	3		31	0.00E+00	0.000E+00
Non-Qual. Cable	6.30E-03	3		12000	0.00E+00	0.000E+00
Non-Qual. JB	1.60E-03	3		12000	0.00E+00	0.000E+00
Transformers	7.90E-03	3		47	0.00E+00	0.000E+00
Battery Chargers	4.00E-03	3		34	0.00E+00	0.000E+00
Air Compressors	4.70E-03	3		26	0.00E+00	0.000E+00
Vent. Subsystem	9.50E-03	3		289	0.00E+00	0.000E+00
			Sum of Ignition Sources WF for Transient source	Total Number of Compartments		
Transients	1.30E-03	3	4	34	1.18E-01	4.588E-04
Cable fire-Welding	5.10E-03	3	1	34	2.94E-02	4.500E-04
Transient Fire-weld	3.10E-02	3	1	34	2.94E-02	2.735E-03
				TOTAL		**6.644E-03**

NOTE: No Plant Wide ignition sources (except transients) exist in this area.

* For Plant Locations the weighting factor is the number of units (3) divided by number of switchgear rooms (15).

		FIRE AREA / COMPARTMENT IGNITION FREQUENCY				
		PHASE II (STEP 1) EVALUATION				
		PLANT LOCATION: SWITCHGEAR ROOM (FIRE AREA 11)				
Components	Generic Fire Frequency	Location Weighting Factor (WL)*	Number of Components in Fire Area	Total Number In All Plant Locations	Ignition Source Weighting Factor (WI)	Fire Compartme Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location						
Electric Cabinets	1.50E-02	0.2				3.000E-03
Plant Wide						
Fire Prot. Panel	2.40E-03	3		40	0.00E+00	0.000E+00
RPS MG Set	5.50E-03	3		31	0.00E+00	0.000E+00
Non-Qual. Cable	6.30E-03	3		12000	0.00E+00	0.000E+00
Non-Qual. JB	1.60E-03	3		12000	0.00E+00	0.000E+00
Transformers	7.90E-03	3		47	0.00E+00	0.000E+00
Battery Chargers	4.00E-03	3		34	0.00E+00	0.000E+00
Air Compressors	4.70E-03	3		26	0.00E+00	0.000E+00
Vent. Subsystem	9.50E-03	3		289	0.00E+00	0.000E+00
			Sum of Ignition Sources WF for Transient source	Total Number of Compartments		
Transients	1.30E-03	3	4	34	1.18E-01	4.588E-04
Cable fire-Welding	5.10E-03	3	1	34	2.94E-02	4.500E-04
Transient Fire-weld	3.10E-02	3	1	34	2.94E-02	2.735E-03
				TOTAL		**6.644E-03**

NOTE: No Plant Wide ignition sources (except transients) exist in this area.

* For Plant Locations the weighting factor is the number of units (3) divided by number of switchgear rooms (15).

		FIRE AREA / COMPARTMENT IGNITION FREQUENCY				
		PHASE II (STEP 1) EVALUATION				
		PLANT LOCATION: SWITCHGEAR ROOM (FIRE AREA 12)				
Components	Generic Fire Frequency	Location Weighting Factor (WL)*	Number of Components In Fire Area	Total Number In All Plant Locations	Ignition Source Weighting Factor (WI)	Fire Compartme Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location						
Electric Cabinets	1.50E-02	0.2				3.000E-03
Plant Wide						
Fire Prot. Panel	2.40E-03	3		40	0.00E+00	0.000E+00
RPS MG Set	5.50E-03	3		31	0.00E+00	0.000E+00
Non-Qual. Cable	6.30E-03	3		12000	0.00E+00	0.000E+00
Non-Qual. JB	1.60E-03	3		12000	0.00E+00	0.000E+00
Transformers	7.90E-03	3	1	47	2.13E-02	5.043E-04
Battery Chargers	4.00E-03	3		34	0.00E+00	0.000E+00
Air Compressors	4.70E-03	3		26	0.00E+00	0.000E+00
Vent. Subsystem	9.50E-03	3	1	289	3.46E-03	9.862E-05
			Sum of Ignitior Sources WF foi Transient sourc	Total Number of Compartments		
Transients	1.30E-03	3	4	34	1.18E-01	4.588E-04
Cable fire-Welding	5.10E-03	3	1	34	2.94E-02	4.500E-04
Transient Fire-weld	3.10E-02	3	1	34	2.94E-02	2.735E-03
				TOTAL		**7.247E-03**

NOTE: No Plant Wide ignition sources (except transients) exist in this area.

* For Plant Locations the weighting factor is the number of units (3) divided by number of switchgear rooms (15).

		FIRE AREA / COMPARTMENT IGNITION FREQUENCY				
		PHASE II (STEP 1) EVALUATION				
		PLANT LOCATION: SWITCHGEAR ROOM (FIRE AREA 13)				
		n				
Components	Generic Fire Frequency	Location Weighting Factor (WL)*	Number of Components in Fire Area	Total Number In All Plant Locations	Ignition Source Weighting Factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location						
Electric Cabinets	1.50E-02	0.2				3.000E-03
Plant Wide						
Fire Prot. Panel	2.40E-03	3		40	0.00E+00	0.000E+00
RPS MG Set	5.50E-03	3		31	0.00E+00	0.000E+00
Non-Qual. Cable	6.30E-03	3		12000	0.00E+00	0.000E+00
Non-Qual. JB	1.60E-03	3		12000	0.00E+00	0.000E+00
Transformers	7.90E-03	3	1	47	2.13E-02	5.043E-04
Battery Chargers	4.00E-03	3		34	0.00E+00	0.000E+00
Air Compressors	4.70E-03	3		26	0.00E+00	0.000E+00
Vent. Subsystem	9.50E-03	3		289	0.00E+00	0.000E+00
			Sum of Ignition Sources WF for Transient source	Total Number of Compartments		
Transients	1.30E-03	3	4	34	1.18E-01	4.588E-04
Cable fire-Welding	5.10E-03	3	1	34	2.94E-02	4.500E-04
Transient Fire-weld	3.10E-02	3	1	34	2.94E-02	2.735E-03
				TOTAL		**7.148E-03**
* For Plant Locations the weighting factor is the number of units (3) divided by number of switchgear rooms (15).						

		FIRE AREA / COMPARTMENT IGNITION FREQUENCY				
		PHASE II (STEP 1) EVALUATION				
		PLANT LOCATION: SWITCHGEAR ROOM (FIRE AREA 14)				
Components	Generic Fire Frequency	Location Weighting Factor (WL)*	Number of Components in Fire Area	Total Number In All Plant Locations	Ignition Source Weighting Factor (WI)	Fire Compartme Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location						
Electric Cabinets	1.50E-02	0.2				3.000E-03
Plant Wide						
Fire Prot. Panel	2.40E-03	3		40	0.00E+00	0.000E+00
RPS MG Set	5.50E-03	3		31	0.00E+00	0.000E+00
Non-Qual. Cable	6.30E-03	3		12000	0.00E+00	0.000E+00
Non-Qual. JB	1.60E-03	3		12000	0.00E+00	0.000E+00
Transformers	7.90E-03	3		47	0.00E+00	0.000E+00
Battery Chargers	4.00E-03	3		34	0.00E+00	0.000E+00
Air Compressors	4.70E-03	3		26	0.00E+00	0.000E+00
Vent. Subsystem	9.50E-03	3		289	0.00E+00	0.000E+00
			Sum of Ignition Sources WF for Transient source	Total Number of Compartments		
Transients	1.30E-03	3	4	34	1.18E-01	4.588E-04
Cable fire-Welding	5.10E-03	3	1	34	2.94E-02	4.500E-04
Transient Fire-weld	3.10E-02	3	1	34	2.94E-02	2.735E-03
				TOTAL		**6.644E-03**
NOTE: No Plant Wide ignition sources (except transients) exist in this area.						
* For Plant Locations the weighting factor is the number						
of units (3) divided by number of switchgear rooms (15).						

		FIRE AREA / COMPARTMENT IGNITION FREQUENCY				
		PHASE II (STEP 1) EVALUATION				
		PLANT LOCATION: SWITCHGEAR ROOM (FIRE AREA 15)				
Components	Generic Fire Frequency	Location Weighting Factor (WL)*	Number of Components in Fire Area	Total Number In All Plant Locations	Ignition Source Weighting Factor (WI)	Fire Compartme Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location						
Electric Cabinets	1.50E-02	0.2				3.000E-03
Plant Wide						
Fire Prot. Panel	2.40E-03	3		40	0.00E+00	0.000E+00
RPS MG Set	5.50E-03	3		31	0.00E+00	0.000E+00
Non-Qual. Cable	6.30E-03	3		12000	0.00E+00	0.000E+00
Non-Qual. JB	1.60E-03	3		12000	0.00E+00	0.000E+00
Transformers	7.90E-03	3		47	0.00E+00	0.000E+00
Battery Chargers	4.00E-03	3		34	0.00E+00	0.000E+00
Air Compressors	4.70E-03	3		26	0.00E+00	0.000E+00
Vent. Subsystem	9.50E-03	3		289	0.00E+00	0.000E+00
			Sum of Ignition Sources WF for Transient source	Total Number of Compartments		
Transients	1.30E-03	3	4	34	1.18E-01	4.588E-04
Cable fire-Welding	5.10E-03	3	1	34	2.94E-02	4.500E-04
Transient Fire-weld	3.10E-02	3	1	34	2.94E-02	2.735E-03
				TOTAL		**6.644E-03**

NOTE: No Plant Wide ignition sources (except transients) exist in this area.

* For Plant Locations the weighting factor is the number of units (3) divided by number of switchgear rooms (15).

		FIRE AREA / COMPARTMENT IGNITION FREQUENCY				
		PHASE II (STEP 1) EVALUATION				
PLANT LOCATION: CONTROL BUILDING EL 593 (FIRE AREA 16-CONTROL BAY)						COMPT. 16-1
Components	Generic Fire Frequency	Location Weighting Factor (WL)	Number of Components in Fire Area	Total Number In All Plant Locations	Ignition Source Weighting Factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location						
Electric Cabinets	5.00E-02	1				5.000E-02
Pumps*						
Plant Wide						
Fire Prot. Panel	2.40E-03	3	3	40	7.50E-02	5.400E-04
RPS MG Set	5.50E-03	3	6	31	1.94E-01	3.194E-03
Non-Qual. Cable	6.30E-03	3	96	12000	8.00E-03	1.512E-04
Non-Qual. JB	1.60E-03	3	96	12000	8.00E-03	3.840E-05
Transformers	7.90E-03	3		47	0.00E+00	0.000E+00
Battery Chargers	4.00E-03	3	3	34	8.82E-02	1.059E-03
Air Compressors	4.70E-03	3		26	0.00E+00	0.000E+00
Vent. Subsystem	9.50E-03	3	3	289	1.04E-02	2.958E-04
			Sum of Ignition Sources WF for Transient source	Total Number of Compartments		
Transients	1.30E-03	3	4	34	1.18E-01	4.588E-04
Cable fire-Welding	5.10E-03	3	1	34	2.94E-02	4.500E-04
Transient Fire-weld	3.10E-02	3	1	34	2.94E-02	2.735E-03
					TOTAL	**5.892E-02**

Note: This location in the control building is not categorized as a Plant Location/Building in TABLE 1.2 of the methodology.
The ignition frequency attributed due to electrical cabinets is assumed similar to a reactor building.
* Only a few small HVAC pumps with negligible combustibles are located in this area. Therefore, impact of pumps on the fire frequency is being neglected.

		FIRE AREA / COMPARTMENT IGNITION FREQUENCY				
		PHASE II (STEP 1) EVALUATION				
PLANT LOCATION: CABLE SPREADING ROOMS A AND B (FIRE AREA 16-CONTROL BAY						COMPT. 16-2
Components	Generic Fire Frequency	Location Weighting Factor (WL)*	Number of Components in Fire Area	Total Number In All Plant Locations	Ignition Source Weighting Factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location						
Electric Cabinets	3.20E-03	3				9.600E-03
Plant Wide						
Fire Prot. Panel	2.40E-03	3	2	40	5.00E-02	3.600E-04
RPS MG Set	5.50E-03	3		31	0.00E+00	0.000E+00
Non-Qual. Cable	6.30E-03	3	1390	12000	1.16E-01	2.189E-03
Non-Qual. JB	1.60E-03	3	1390	12000	1.16E-01	5.560E-04
Transformers	7.90E-03	3		47	0.00E+00	0.000E+00
Battery Chargers	4.00E-03	3		34	0.00E+00	0.000E+00
Air Compressors	4.70E-03	3	1	26	3.85E-02	5.423E-04
Vent. Subsystem	9.50E-03	3	2	289	6.92E-03	1.972E-04
			Sum of Ignition Sources WF for Transient source	Total Number of Compartments		
Transients	1.30E-03	3		34	0.00E+00	0.000E+00
Cable fire-Welding	5.10E-03	3		34	0.00E+00	0.000E+00
Transient Fire-weld	3.10E-02	3		34	0.00E+00	0.000E+00
					TOTAL	1.344E-02

*Note: There are two cable spreading rooms (CSR) in the plant. Units 1 and 2 CSR's are combined into one room, while unit 3 has its own CSR. Both CSR's are located on the same floor (EL 606) of the control bay. Even though suppression and detection is provided in these areas, the two rooms are not separated by fire rated barriers. For calculation purposes the two CSR's will be considered as one room. Therefore the weighting factor will be 3 (3 units/1 CSR).

Contribution due to transient combustibles is being neglected due to restrictions imposed by plant procedures.

		FIRE AREA / COMPARTMENT IGNITION FREQUENCY				
		PHASE II (STEP 1) EVALUATION				
	PLANT LOCATION: CONTROL ROOMS (FIRE AREA 16-CONTROL BAY,					COMPT. 16-3
Components	Generic Fire Frequency	Location Weighting Factor (WL)*	Number of Components in Fire Area	Total Number In All Plant Locations	Ignition Source Weighting Factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location						
Electric Cabinets	9.50E-03	3				2.850E-02
Plant Wide						
Fire Prot. Panel	2.40E-03	3	1	40	2.50E-02	1.800E-04
RPS MG Set	5.50E-03	3		31	0.00E+00	0.000E+00
Non-Qual. Cable	6.30E-03	3	30	12000	2.50E-03	4.725E-05
Non-Qual. JB	1.60E-03	3	30	12000	2.50E-03	1.200E-05
Transformers	7.90E-03	3		47	0.00E+00	0.000E+00
Battery Chargers	4.00E-03	3		34	0.00E+00	0.000E+00
Air Compressors	4.70E-03	3		26	0.00E+00	0.000E+00
Vent. Subsystem	9.50E-03	3	30	289	1.04E-01	2.958E-03
			Sum of Ignition Sources WF for Transient source	Total Number of Compartments		
Transients	1.30E-03	3	4	34	1.18E-01	4.588E-04
Cable fire-Welding	5.10E-03	3	1	34	2.94E-02	4.500E-04
Transient Fire-weld	3.10E-02	3	1	34	2.94E-02	2.735E-03
					TOTAL	**3.534E-02**

* For Plant Locations, the calculated weighting factor should be 1 (3 units/3 control rooms). However, since the three control rooms are located in one area without being separated by a fire rated barrier (i.e. there is potential of fire spread between control rooms), the 3 control rooms can be considered as one room and the weighting factor will be 3 (3 units/1 control room).

		FIRE AREA / COMPARTMENT IGNITION FREQUENCY				
		PHASE II (STEP 1) EVALUATION				
	PLANT LOCATION: BATTERY AND BATTERY BOARD RMS UNIT 1 (FIRE AREA 17)					
Components	Generic Fire Frequency	Location Weighting Factor (WL)	Number of Components In Fire Area	Total Number In All Plant Locations	Ignition Source Weighting Factor (WI)	Fire Compartme Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location						
Electric Cabinets	5.00E-02	0.25				1.250E-02
Batteries	3.20E-03	1				3.20E-03
Plant Wide						
Fire Prot. Panel	2.40E-03	3		40	0.00E+00	0.000E+00
RPS MG Set	5.50E-03	3	1	31	3.23E-02	5.323E-04
Non-Qual. Cable	6.30E-03	3		12000	0.00E+00	0.000E+00
Non-Qual. JB	1.60E-03	3		12000	0.00E+00	0.000E+00
Transformers	7.90E-03	3	2	47	4.26E-02	1.009E-03
Battery Chargers	4.00E-03	3	3	34	8.82E-02	1.059E-03
Air Compressors	4.70E-03	3		26	0.00E+00	0.000E+00
Vent. Subsystem	9.50E-03	3		289	0.00E+00	0.000E+00
			Sum of Ignition Sources WF for Transient source	Total Number of Compartments		
Transients	1.30E-03	3	4	34	1.18E-01	4.588E-04
Cable fire-Welding	5.10E-03	3	1	34	2.94E-02	4.500E-04
Transient Fire-weld	3.10E-02	3	1	34	2.94E-02	2.735E-03
				TOTAL		**2.194E-02**

Note: This area comprises of 2 rooms, one containing low voltage (250V DC) equipment and the other is a battery room. The room containing 250V equipment will be designated as Reactor Building location and contain the contributions from the cabinets and plant wide components. The cabinet contribution will be asumed as 25% of the overall reactor building cabinets ignition frequency. The other room will be designated as Battery Room location and contain only the contribution from batteries. This calculation shows the combined ignition frequency of the two rooms.

FIRE AREA / COMPARTMENT IGNITION FREQUENCY

PHASE II (STEP 1) EVALUATION

PLANT LOCATION: BATTERY AND BATTERY BOARD RMS UNIT 2 (FIRE AREA 18)

Components	Generic Fire Frequency	Location Weighting Factor (WL)	Number of Components in Fire Area	Total Number In All Plant Locations	Ignition Source Weighting Factor (WI)	Fire Compartme Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location						
Electric Cabinets	5.00E-02	0.25				1.250E-02
Batteries	3.20E-03	1				3.20E-03
Plant Wide						
Fire Prot. Panel	2.40E-03	3		40	0.00E+00	0.000E+00
RPS MG Set	5.50E-03	3	1	31	3.23E-02	5.323E-04
Non-Qual. Cable	6.30E-03	3	1	12000	8.33E-05	1.575E-06
Non-Qual. JB	1.60E-03	3	1	12000	8.33E-05	4.000E-07
Transformers	7.90E-03	3		47	0.00E+00	0.000E+00
Battery Chargers	4.00E-03	3	3	34	8.82E-02	1.059E-03
Air Compressors	4.70E-03	3		26	0.00E+00	0.000E+00
Vent. Subsystem	9.50E-03	3		289	0.00E+00	0.000E+00
			Sum of Ignition Sources WF for Transient source	**Total Number of Compartments**		
Transients	1.30E-03	3	4	34	1.18E-01	4.588E-04
Cable fire-Welding	5.10E-03	3	1	34	2.94E-02	4.500E-04
Transient Fire-weld	3.10E-02	3	1	34	2.94E-02	2.735E-03
				TOTAL		**2.094E-02**

Note: This area comprises of 2 rooms, one containing low voltage (250V DC) equipment and the other is a battery room. The room containing 250V equipment will be designated as Reactor Building location and contain the contributions from the cabinets and plant wide components. The cabinet contribution will be asumed as 25% of the overall reactor building cabinets ignition frequency. The other room will be designated as Battery Room location and contain only the contribution from batteries. This calculation shows the combined ignition frequency of the two rooms.

		FIRE AREA / COMPARTMENT IGNITION FREQUENCY				
		PHASE II (STEP 1) EVALUATION				
	PLANT LOCATION: BATTERY AND BATTERY BOARD RMS UNIT 3 (FIRE AREA 19)					
Components	Generic Fire Frequency	Location Weighting Factor (WL)	Number of Components In Fire Area	Total Number In All Plant Locations	Ignition Source Weighting Factor (WI)	Fire Compartme Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location						
Electric Cabinets	5.00E-02	0.25				1.250E-02
Batteries	3.20E-03	1				3.20E-03
Plant Wide						
Fire Prot. Panel	2.40E-03	3		40	0.00E+00	0.000E+00
RPS MG Set	5.50E-03	3	1	31	3.23E-02	5.323E-04
Non-Qual. Cable	6.30E-03	3		12000	0.00E+00	0.000E+00
Non-Qual. JB	1.60E-03	3		12000	0.00E+00	0.000E+00
Transformers	7.90E-03	3		47	0.00E+00	0.000E+00
Battery Chargers	4.00E-03	3	3	34	8.82E-02	1.059E-03
Air Compressors	4.70E-03	3		26	0.00E+00	0.000E+00
Vent. Subsystem	9.50E-03	3		289	0.00E+00	0.000E+00
			Sum of Ignition Sources WF for Transient source	Total Number of Compartments		
Transients	1.30E-03	3	4	34	1.18E-01	4.588E-04
Cable fire-Welding	5.10E-03	3	1	34	2.94E-02	4.500E-04
Transient Fire-weld	3.10E-02	3	1	34	2.94E-02	2.735E-03
				TOTAL		**2.094E-02**

Note: This area comprises of 2 rooms, one containing low voltage (250V DC) equipment and the other is a battery room. The room containing 250V equipment will be designated as Reactor Building location and contain the contributions from the cabinets and plant wide components. The cabinet contribution will be asumed as 25% of the overall reactor building cabinets ignition frequency. The other room will be designated as Battery Room location and contain only the contribution from batteries. This calculation shows the combined ignition frequency of the two rooms.

		FIRE AREA / COMPARTMENT IGNITION FREQUENCY				
		PHASE II (STEP 1) EVALUATION				
	PLANT LOCATION: UNIT 1/2 DIESEL GENERATOR BUILDING (FIRE AREA 20)					
Components	Generic Fire Frequency	Location Weighting Factor (WL)*	Number of Components in Fire Area	Total Number In All Plant Locations	Ignition Source Weighting Factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location						
Diesel Generators	2.60E-02	4	1	1	1.00E+00	1.040E-01
Electrical cabinets	2.40E-03	4	1	1	1.00E+00	9.600E-03
Plant Wide						
Fire Prot. Panel	2.40E-03	3	2	40	5.00E-02	3.600E-04
RPS MG Set	5.50E-03	3		31	0.00E+00	0.000E+00
Non-Qual. Cable	6.30E-03	3	21	12000	1.75E-03	3.308E-05
Non-Qual. JB	1.60E-03	3	21	12000	1.75E-03	8.400E-06
Transformers	7.90E-03	3	3	47	6.38E-02	1.513E-03
Battery Chargers	4.00E-03	3	8	34	2.35E-01	2.824E-03
Air Compressors	4.70E-03	3		26	0.00E+00	0.000E+00
Vent. Subsystem	9.50E-03	3	20	289	6.92E-02	1.972E-03
			Sum of Ignitior Source WF For Transients	Total Number of Compartments		
Transients	1.30E-03	3	5	34	1.47E-01	5.735E-04
Cable fire-Welding	5.10E-03	3	1	34	2.94E-02	4.500E-04
Transient Fire-weldin	3.10E-02	3	1	34	2.94E-02	2.735E-03
				TOTAL		**1.241E-01**

*Note: BFN has two separate diesel buildings, one for units 1 and 2 and other one for unit 3. Each diesel building has four diesel generators and associated elctrical and mechanical systems. Thus fire frequency for one BFN diesel generator building will be four times as much as the generic fire frequency (i.e. a Plant Location weighting factor of 4).

		FIRE AREA / COMPARTMENT IGNITION FREQUENCY				
		PHASE II (STEP 1) EVALUATION				
	PLANT LOCATION: UNIT 3 DIESEL GENERATOR BUILDING (FIRE AREA 21)					
Components	Generic Fire Frequency	Location Weighting Factor (WL)*	Number of Components in Fire Area	Total Number In All Plant Locations	Ignition Source Weighting Factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location						
Diesel Generators	2.60E-02	4	1	1	1.00E+00	1.040E-01
Electrical cabinets	2.40E-03	4	1	1	1.00E+00	9.600E-03
Plant Wide						
Fire Prot. Panel	2.40E-03	3	2	40	5.00E-02	3.600E-04
RPS MG Set	5.50E-03	3		31	0.00E+00	0.000E+00
Non-Qual. Cable	6.30E-03	3	35	12000	2.92E-03	5.513E-05
Non-Qual. JB	1.60E-03	3	35	12000	2.92E-03	1.400E-05
Transformers	7.90E-03	3		47	0.00E+00	0.000E+00
Battery Chargers	4.00E-03	3	9	34	2.65E-01	3.176E-03
Air Compressors	4.70E-03	3		26	0.00E+00	0.000E+00
Vent. Subsystem	9.50E-03	3	28	289	9.69E-02	2.761E-03
			Sum of Ignition Source WF For Transients	Total Number of Compartments		
Transients	1.30E-03	3	5	34	1.47E-01	5.735E-04
Cable fire-Welding	5.10E-03	3	1	34	2.94E-02	4.500E-04
Transient Fire-weldin	3.10E-02	3	1	34	2.94E-02	2.735E-03
				TOTAL		1.237E-01

*Note: BFN has two separate diesel buildings, one for units 1 and 2 and other one for unit 3. Each diesel building has four diesel generators and associated elctrical and mechanical systems. Thus fire frequency for one BFN diesel generator building will befour times as much as the generic fire frequency (i.e. a Plant Location weighting factor of 4).

		FIRE AREA / COMPARTMENT IGNITION FREQUENCY				
		PHASE II (STEP 1) EVALUATION				
	PLANT LOCATION: SWITCHGEAR ROOM (UNIT 3 4KV SDBR, FIRE AREA 22)					
Components	Generic Fire Frequency	Location Weighting Factor (WL)*	Number of Components in Fire Area	Total Number In All Plant Locations	Ignition Source Weighting Factor (WI)	Fire Compartme Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location						
Electric Cabinets	1.50E-02	0.2				3.000E-03
Plant Wide						
Fire Prot. Panel	2.40E-03	3		40	0.00E+00	0.000E+00
RPS MG Set	5.50E-03	3		31	0.00E+00	0.000E+00
Non-Qual. Cable	6.30E-03	3	15	12000	1.25E-03	2.363E-05
Non-Qual. JB	1.60E-03	3	15	12000	1.25E-03	6.000E-06
Transformers	7.90E-03	3		47	0.00E+00	0.000E+00
Battery Chargers	4.00E-03	3		34	0.00E+00	0.000E+00
Air Compressors	4.70E-03	3		26	0.00E+00	0.000E+00
Vent. Subsystem	9.50E-03	3		289	0.00E+00	0.000E+00
			Sum of Ignition Sources WF for Transient source	Total Number of Compartments		
Transients	1.30E-03	3	4	34	1.18E-01	4.588E-04
Cable fire-Welding	5.10E-03	3	1	34	2.94E-02	4.500E-04
Transient Fire-weld	3.10E-02	3	1	34	2.94E-02	2.735E-03
				TOTAL		**6.674E-03**

* For Plant Locations the weighting factor is the number of units (3) divided by number of switchgear rooms (15).

		FIRE AREA / COMPARTMENT IGNITION FREQUENCY				
		PHASE II (STEP 1) EVALUATION				
	PLANT LOCATION: SWITCHGEAR ROOM (UNIT 3 4KV SDBR, FIRE AREA 23)					
Components	Generic Fire Frequency	Location Weighting Factor (WL)*	Number of Components In Fire Area	Total Number In All Plant Locations	Ignition Source Weighting Factor (WI)	Fire Compartme Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location						
Electric Cabinets	1.50E-02	0.2				3.000E-03
Plant Wide						
Fire Prot. Panel	2.40E-03	3		40	0.00E+00	0.000E+00
RPS MG Set	5.50E-03	3		31	0.00E+00	0.000E+00
Non-Qual. Cable	6.30E-03	3	15	12000	1.25E-03	2.363E-05
Non-Qual. JB	1.60E-03	3	15	12000	1.25E-03	6.000E-06
Transformers	7.90E-03	3		47	0.00E+00	0.000E+00
Battery Chargers	4.00E-03	3		34	0.00E+00	0.000E+00
Air Compressors	4.70E-03	3		26	0.00E+00	0.000E+00
Vent. Subsystem	9.50E-03	3		289	0.00E+00	0.000E+00
			Sum of Ignitior Sources WF foi Transient source	Total Number of Compartments		
Transients	1.30E-03	3	4	34	1.18E-01	4.588E-04
Cable fire-Welding	5.10E-03	3	1	34	2.94E-02	4.500E-04
Transient Fire-weld	3.10E-02	3	1	34	2.94E-02	2.735E-03
				TOTAL		6.674E-03

* For Plant Locations the weighting factor is the number of units (3) divided by number of switchgear rooms (15).

FIRE AREA / COMPARTMENT IGNITION FREQUENCY

PHASE II (STEP 1) EVALUATION

PLANT LOCATION: SWITCHGEAR ROOM (UNIT 3 4KV BUS TIE BD ROOM, FIRE AREA 24)

Components	Generic Fire Frequency	Location Weighting Factor (WL)*	Number of Components In Fire Area	Total Number In All Plant Locations	Ignition Source Weighting Factor (WI)	Fire Compartme Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location						
Electric Cabinets	1.50E-02	0.2				3.000E-03
Plant Wide						
Fire Prot. Panel	2.40E-03	3		40	0.00E+00	0.000E+00
RPS MG Set	5.50E-03	3		31	0.00E+00	0.000E+00
Non-Qual. Cable	6.30E-03	3	8	12000	6.67E-04	1.260E-05
Non-Qual. JB	1.60E-03	3	8	12000	6.67E-04	3.200E-06
Transformers	7.90E-03	3		47	0.00E+00	0.000E+00
Battery Chargers	4.00E-03	3		34	0.00E+00	0.000E+00
Air Compressors	4.70E-03	3		26	0.00E+00	0.000E+00
Vent. Subsystem	9.50E-03	3		289	0.00E+00	0.000E+00
			Sum of Ignitior Sources WF for Transient source	Total Number of Compartments		
Transients	1.30E-03	3	4	34	1.18E-01	4.588E-04
Cable fire-Welding	5.10E-03	3	1	34	2.94E-02	4.500E-04
Transient Fire-weld	3.10E-02	3	1	34	2.94E-02	2.735E-03
				TOTAL		6.660E-03

* For Plant Locations the weighting factor is the number of units (3) divided by number of switchgear rooms (15).

		FIRE AREA / COMPARTMENT IGNITION FREQUENCY				
		PHASE II (STEP 1) EVALUATION				
	PLANT LOCATION: INTAKE PUMP STATION (FIRE COMPARTMENT 25-1)					
Components	Generic Fire Frequency	Location Weighting Factor (WL)*	Number of Components In Fire Area	Total Number In All Plant Locations	Ignition Source Weighting Factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location						
Electric Cabinets	2.40E-03	3				7.200E-03
Fire Pumps	4.00E-03	3				1.200E-02
Others	3.20E-03	3				9.600E-03
Plant Wide						
Fire Prot. Panel	2.40E-03	3	1	40	2.50E-02	1.800E-04
RPS MG Set	5.50E-03	3		31	0.00E+00	0.000E+00
Non-Qual. Cable	6.30E-03	3	271	12000	2.26E-02	4.268E-04
Non-Qual. JB	1.60E-03	3	271	12000	2.26E-02	1.084E-04
Transformers	7.90E-03	3	3	47	6.38E-02	1.513E-03
Battery Chargers	4.00E-03	3		34	0.00E+00	0.000E+00
Air Compressors	4.70E-03	3		26	0.00E+00	0.000E+00
Vent. Subsystem	9.50E-03	3	8	289	2.77E-02	7.889E-04
				Sum of Ignitior Source WF For Transients		
Transients	1.30E-03	3	7	34	2.06E-01	8.029E-04
Cable fire-Welding	5.10E-03	3	1	34	2.94E-02	4.500E-04
Transient Fire-weldin	3.10E-02	3	1	34	2.94E-02	2.735E-03
				TOTAL		3.581E-02

*Note: The Plant Location weighting factor is 3 (3units/1 IPS). The Plant Wide Location weighting factor is 3 to account for three (3) units.

		FIRE AREA / COMPARTMENT IGNITION FREQUENCY				
		PHASE II (STEP 1) EVALUATION				
		PLANT LOCATION: PIPE TUNNEL (FIRE COMPARTMENT 25-2)				
Components	Generic Fire Frequency	Location Weighting Factor (WL)	Number of Components in Fire Area	Total Number In All Plant Locations	Ignition Source Weighting Factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location						
		NO PLANT LOCATION COMPONENTS				
Plant Wide						
Non-Qual. Cable	6.30E-03	3	5	12000	4.17E-04	7.875E-06
Non-Qual. JB	1.60E-03	3	5	12000	4.17E-04	2.000E-06
				TOTAL		9.875E-06

		FIRE AREA / COMPARTMENT IGNITION FREQUENCY				
		PHASE II (STEP 1) EVALUATION				
		PLANT LOCATION: TURBINE BUILDING (FIRE COMPARTMENT 25-3)				
Components	Generic Fire Frequency	Location Weighting Factor (WL)	Number of Components in Fire Area	Total Number In All Plant Locations	Ignition Source Weighting Factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location						
Boiler	1.60E-03	3	1	1	1.00E+00	4.80E-03
Electrical Cabinets	1.30E-02	3	1	1	1.00E+00	3.90E-02
Feedwater Pumps	4.00E-03	3	1	1	1.00E+00	1.20E-02
Other Pumps	6.30E-03	3	1	1	1.00E+00	1.89E-02
T/G Excitor	4.00E-03	3	1	1	1.00E+00	1.20E-02
T/G Oil	1.30E-02	3	1	1	1.00E+00	3.90E-02
T/G Hydrogen	5.50E-03	3	1	1	1.00E+00	1.65E-02
Plant Wide						
Fire Prot. Panel	2.40E-03	3	4	40	1.00E-01	7.200E-04
RPS MG Set	5.50E-03	3	2	31	6.45E-02	1.065E-03
Non-Qual. Cable	6.30E-03	3	6165	12000	5.14E-01	9.710E-03
Non-Qual. JB	1.60E-03	3	6165	12000	5.14E-01	2.466E-03
Transformers	7.90E-03	3	20	47	4.26E-01	1.009E-02
Battery Chargers	4.00E-03	3	1	34	2.94E-02	3.529E-04
Air Compressors	4.70E-03	3	16	26	6.15E-01	8.677E-03
Vent. Subsystem	9.50E-03	3	128	289	4.43E-01	1.262E-02
Off-Gas/H2 Recomb.	8.60E-02	3	3	3	1.00E+00	2.580E-01
			Sum of Ignition Sources WF for Transient source	Total Number of Compartments		
Transients	1.30E-03	3	8	34	2.35E-01	9.176E-04
Cable fire-Welding	5.10E-03	3	1	34	2.94E-02	4.500E-04
Transient Fire-weldi	3.10E-02	3	1	34	2.94E-02	2.735E-03
				TOTAL		**4.500E-01**

ATTACHMENT C

FIRE DAMAGE ANALYSIS/ CRITICAL COMBUSTIBLE LOADING CALCULATIONS

Attachment C

Fire Damage Analysis/Critical Combustible Loading Calculations

This attachment is subdivided into the following sections:

C.1 Consideration of plant wide and plant location components for zone of influence (ZOI) calculations.

C.2 Zone of influence (ZOI) calculations.

C.3 Detailed evaluation of fire sources that have the potential to develop hot gas layers in the associated plant area.

C.1 Consideration of plant wide and plant location components for zone of influence (ZOI) calculations.

This document provides the calculations used to generate the fire damage envelope, or zone of influence, for fixed fire sources in the Unit 2 Reactor Building. The heat release rates and combustible values used in these calculations are taken from Attachment A, Heat Release Rates.

Based on the EPRI Fire Events Database (NSAC/178/L), a list of "Plant Wide Components" were identified. These components showed a similar history of fire involvement (see EPRI report TR-100370, Table 12). Similar "Plant Wide Components" for the Browns Ferry site are listed in Table 2-3 of this report. The specific number of each of these components within each zone in the Unit 2 Reactor Building are identified in Attachment B. A selected number of these fire sources were then analyzed for critical combustible loading calculations.

These calculations are based on the correlations that are described in the EPRI FIVE documentation (EPRI report TR-100370).

In addition to the calculations, a figure is provided for each of the components analyzed in this attachment. These figures show a graphic representation of the zone of influence for each component.

The selection of fire sources for analysis was based on the potential for the component to ignite and release heat without the presence of an exposing fire. The total combustible loading and heat release rates were significant factors in the selection of these fire sources, since these parameters define the fire size.

Plant Wide Components

The selection of plant-wide components for evaluation is described below.

Transformers

All 6 transformers listed in Attachment B and the Plant Wide Component List have been identified as potential fire sources.

Unit 2 Preferred AC Transformer	(Fire zone 2-3)
4kV/480V Transformers TS2A and TS2B	(Fire zone 2-5)
240V Lighting Transformer TL2A	(Fire zone 2-5)
4kV/480V Emergency Transformers TS1E, 2A and 2B	(Fire zone 2-6)
240V Lighting Transformer TL2B	(Fire zone 2-6)

RPS MG Sets

All 6 plant MG sets listed in Attachment B have been identified as potentially significant fire sources.

LPCI MG Sets 2DN and 2EA	(Fire zone 2-5)
Recirc Pump MG Sets 2A and 2B	(Fire zone 2-6)
LPCI MG Sets 2DA and 2EN	(Fire zone 2-6)

Air Compressors

Three air compressors listed in Attachment B have been identified as potential fire sources. Two air compressors on the 593 foot elevation are void of combustibles (see Reference 3). The remaining compressor is evaluated as a potential fire source.

Drywell/Torus Compressor	(Fire zone 2-1)

Fire Protection Panels

Nine fire protection panels were identified in Attachment B. Of these panels, 6 have been abandoned in place. The remaining 3 units are remote field panels for microprocessor-based addressable systems. These panels contain CPU, analog device and indicating device microprocessor cards, along with the associated power supply. These devices operate on 24V and are very low amperage circuits. Due to the presence of negligible amounts of combustibles and the existence of low power circuits, any fire that could develop within these panels would be of an insignificant nature and would not present an exposure fire hazard for other plant components.

Ventilation Subsystems

Twenty ventilation subsystems have been identified in Attachment B, Plant Wide Component List. Eight of these ventilation systems comprising the Reactor Building supply and exhaust fans, are located outside the building and, therefore, do not contribute to fire hazards within the building. The 6 air cooling units associated with the RHR and Core Spray Pumps are evaluated along with the respective pumps. The remaining 6 systems are evaluated as potential fire sources.

Shutdown Board Room HVAC Compressor Motor (2) and Fan Motor (2)	(Fire zone 2-4)
CRD Repair Room Fan (1)	(Fire zone 2-5)
Primary Containment Purge Filter Unit (1)	(Fire zone 2-5)

Plant Location Components

Electrical cabinets and pumps are categorized as plant location components, as opposed to plant wide components, for the Reactor Building. A generic fire frequency is assigned for these components as a group, based on the EPRI FIVE documentation. The following electrical panels were considered as potential fire sources based on their combustible characteristics (i.e. switchgear, motor control centers, large control panels, etc.). The components that were not separately considered for detailed evaluation at this point contain limited combustibles and have low electrical energy. These components were therefore judged as unlikely to present an exposure fire hazard.

Electric Panels

Fire Zone 2-1	250V RMOV Board 2C 2-PNLA-25-340 (H2/O2 Analyzer)
Fire Zone 2-2	480V RMOV Board 2C 480V RB Vent Board 2B 2-PNLA-25-341 (H2/O2 Analyzer)
Fire Zone 2-4	480V RMOV Board 2D RWCU Room Monitor
Fire Zone 2-5	480V RMOV Board 2E 2-LPNL-025-0031 (RCIC Aux Control Panel) 4kV RPT Board 2-1, Panel 1 and 2 4kV RPT Board 2-2, Panel 1 and 2 Panel 25-3 (Filter demin) Panel 25-9 (Sample panel) 240V Lighting Board 2A
Fire Zone 2-6	LPNL-25-23 and 24 240V Lighting Board 2B

Pumps

Reactor Building pumps were considered as potential fire sources based on the size of their associated motor (over 5 horsepower) and significant oil and grease content. Pumps were also screened from further consideration based on location (i.e. sump pumps, pumps located in isolated rooms, etc.), mode of operation (i.e. normally de-energized), small motor size and low combustible loading (i.e. limited oil and grease content). The following pumps remained for detailed analysis following this review:

Fire Zone 2-1	Core Spray Pumps 2A and 2C RHR Pumps 2A and 2C

Fire Zone 2-2	Core Spray Pumps 2B and 2D RHR Pumps 2B and 2D HPCI Pump
Fire Zone 2-3	RCW Booster Pump
Fire Zone 2-4	RBCCW Pump 2A RBCCW Pump 2B RWCU Pumps
Fire Zone 2-6	SLC Pumps A and B

C.2 FIRE DAMAGE ENVELOP OR ZONE OF INFLUENCE (ZOI) CALCULATIONS

The FIVE methodology provides guidance to perform preliminary evaluation of areas with respect to their fire hazards potential. The screening methodology requires identification of credible fire scenarios defining potential targets and fire sources and their geometric relationships. the following general scenarios will be considered:

- Targets located in the plume, directly above the fire source.
- Targets located next to fire source, exposed to heating by thermal radiation.
- Targets located in the hot gas layer or in the ceiling jet (outside the plume).

The following parameters are used in this evaluation:

- Damage Temperature = 425 ^{0}F (Based on non-qualified cables)
- Critical Heat Flux = 0.5 Btu/sec/ft^2 (Based on non-qualified cables)
- Heat Loss Fraction = 0.85
- Area Geometry (See accompanying spread sheets)
- Heat Release Rates (See accompanying spread sheets)
- Heat of combustion (From Reference 3)

The fire sources have to be analyzed for their potential to form hot gas layer which may cause damage to all components in the area; or the potential to create ceiling jet sub layer causing damage beyond the fire plume zone. Detailed evaluations will be required for these scenarios. The "No-Damage" distances are pre-calculated for all credible fire sources. Once the fire damage envelop (zone of influence) is determined, the components likely to be damaged are thus identified. To facilitate the analysis, a spread sheet format is used to calculate the damage height, critical radial distance and potential for the targets being in the hot gas layer or ceiling jet.

The hot gas layer formation can readily be observed if the fire source combustible load (Btu) exceeds the calculated critical energy release (Q_{tot}). The ceiling jet sub layer can form if the plume height exceeds the ceiling height. The damage height (H) and the critical radial distance (R_{crit}) are calculated based on FIVE guidance. It should be noted that the damage height is based on the combined affects of the ambient temperature rise (smoke layer temperature rise) due to complete pyrolysis of the fire source and the direct affect of the fire plume. The critical temperature rise T_{crit} (Damage temp. - Ambient temp.) is adjusted to account for the smoke layer temperature rise.

The figures shown in Attachment D provide zones of influence for various fire sources.

BUILDING: REACTOR BUILDING UNIT 2, EL 565			Page 1 of 11		/home05/ignition freq (five)/walkdown.2-565	
FIRE AREA/ZONE: 2-1 and 2-2						
CEILING HEIGHT:(above source)	22	DAMAGE TEMP:	425	CRIT. HEAT FLUX (Q"crit)	0.50	BTU/SEC/FT2
FLOOR AREA (FT2)	16500	VOLUME (V): FT3	363000	AMBIENT TEMP (Tamb):	100	
HEAT ADDITION PER UNIT VOL. (Q/V) TO FORM HOT GAS LAYER :			4.5	(FROM TABLE 7E @ 325 F)	CRIT. TEMP RISE (Tcrit):	325
CRITICAL NET ENERGY ADDITION - BTU (Qnet)		1633500	CRITICAL TOTAL ENERGY RELEASE - BTU (Qtot)		10890000	(Qnet/(1-0.85))
			SCREENING CRITERIA			
IGNITION SOURCES	HEAT RELEASE RATE (q) BTU/SEC (3)	LOCATION FACTOR (1, 2 OR 4)	EFFECTIVE HEAT RELEASE RATE (qeff) BTU/SEC	COMBUSTIBLE LOAD BTU (4)	DAMAGE HEIGHT (H) FT (1)	CRITICAL RADIAL DISTANCE FT (2)
480V RMOV BD 2C	106	1	106	372000	6.77	2.60
Electrical cabinet		(>2 ft from wall)				
Fire Zone 2-2						
Notes: 1.	Electrical cabinets: No Vents, closed doors					
2.	HRR: See Attachment A					
3.	Qnet/ft3 (for combustible load contribution)		372000 x .15 /363000	0.15		
	Smoke layer temp rise = 10 F (from Table 7E, Ref 1); Effectve Tcrit			315		
480V RB Vent BD 2B	106	1	106	408000	6.77	2.60
Electrical cabinet		(>3 ft from wall)				
Fire Zone 2-2						
Notes: 1.	Electrical cabinets: No Vents, closed doors					
2.	HRR: See Attachment A					
3.	Qnet/ft3 (for combustible load contribution)		408000 x .15 /363000	0.17		
	Smoke layer temp rise = 10 F (from Table 7E, Ref 1); Effectve Tcrit			315		
250V RMOV BD 2C	106	1	106	544000	6.82	2.60
Electrical cabinet		(4 ft from wall)				
Fire Zone 2-1						
Notes: 1.	Electrical cabinets: No Vents, closed doors					
2.	HRR: Based on closed cabinet for non-qualified cable, See Attachment A					
3.	Qnet/ft3 (for combustible load contribution)		544000 x .15 /363000	0.22		
	Smoke layer temp rise = 14 F (from Table 7E, Ref 1); Effectve Tcrit			311		
(1) H = [340 * qeff^2/3 / Effective Tcrit]^3/5						
(2) Rcrit = [q * 0.4 / 4 * pi * Q"crit]^1/2						

BUILDING: REACTOR BUILDING UNIT 2, EL 565			Page 2 of 11		/home05/ignition freq (five)/walkdown.2-565i	
FIRE AREA/ZONE: 2-1 and 2-2						
CEILING HEIGHT: (above sou	22	DAMAGE TEMP:	425	CRIT. HEAT FLUX (Q"crit)	0.50	BTU/SEC/FT2
FLOOR AREA (FT2)	16500	VOLUME (V): FT3	363000	AMBIENT TEMP (Tamb):	100	
HEAT ADDITION PER UNIT VOL. (Q/V) TO FORM HOT GAS LAYER :			4.5	(FROM TABLE 7E @ 325 F)	CRIT. TEMP RISE (Tcrit):	325
CRITICAL NET ENERGY ADDITION - BTU (Qnet)		1633500	CRITICAL TOTAL ENERGY RELEASE - BTU (Qtot)		10890000	(Qnet/(1-0.85))
			SCREENING CRITERIA			
IGNITION SOURCES	HEAT RELEASE RATE (q) BTU/SEC (3)	LOCATION FACTOR (1, 2 OR 4)	EFFECTIVE HEAT RELEASE RATE (qeff) BTU/SEC	COMBUSTIBLE LOAD BTU (4)	DAMAGE HEIGHT (H) FT (1)	CRITICAL RADIAL DISTANCE FT (2)
2-PNLA-25-340	100	1	100	700000	6.70	2.52
Oxygen/Hydrogen analyzer		(away from wall)				
Fire Zone 2-1						
Notes: 1.	Cabinets: Two small Vents, closed doors					
2.	HRR: Enclosed steel cabinet, Small voltage electronic instruments, small HP motor, solenoids, etc.					
	EPRI fire events data base does not identify this equipment to be a viable fire source.					
	However, assume HRR = 100 Btu/sec (Apprximating 2 non-qual. electrical cubicle)					
3.	Qnet/ft3 (for combustible load contribution) = 700,000 x .15/363000			0.29		
	Smoke layer temp rise = 17 F (from Table 7E, Ref 1); Effective Tcri			308		
2-PNLA-25-341	100	1	100	700000	6.70	2.52
Oxygen/Hydrogen analyzer		(away from wall)				
Fire Zone 2-2						
Notes: 1.	Cabinets: Two small Vents, closed doors					
2.	HRR: Enclosed steel cabinet, Small voltage electronic instruments, small HP motor, solenoids, etc.					
	EPRI fire events data base does not identify this equipment to be a viable fire source.					
	However, assume HRR = 100 Btu/sec (Apprximating 1 non-qual. electrical cubicle)					
3.	Qnet/ft3 (for combustible load contribution) = 700,000 x .15/363000			0.29		
	Smoke layer temp rise = 17 F (from Table 7E, Ref 1); Effective Tcri			308		
Drywell Torus Compressor						
Fire Zone 2-1						
Oil fire	450	1	450	145000	11.85	5.35
Motor fire	40	1	40	100000	4.50	1.60
Notes: 1.	Motor: 40 HP No Vents, Totally enclosed		Compressor: Totally enclosed, no visible oil leaks			
2.	HRR: Based on 1/2 pint oil spill and motor HRR similar to electrical cabinet, See Attachment A					
3.	Smoke layer temperature rise is neglected due to the minimum amount of combustible loading.					
(1) H = [340 * qeff^2/3 / Tcrit]^3/5						
(2) Rcrit = [q * 0.4 / 4 * pi * Q"crit]^1/2						

BUILDING: REACTOR BUILDING UNIT 2, EL 593			Page 3 of 11		/home05/ignition freq (five)/walkdown.2-593i	
FIRE AREA/ZONE: 2-3 and 2-4						
CEILING HEIGHT:(above source)	22	DAMAGE TEMP:	425	CRIT. HEAT FLUX (Q"crit)	0.50	BTU/SEC/FT2
FLOOR AREA (FT2)	12800	VOLUME (V): FT3	281600	AMBIENT TEMP (Tamb):	100	
HEAT ADDITION PER UNIT VOL. (Q/V) TO FORM HOT GAS LAYER :			4.5	(FROM TABLE 7E @ 325 F)	CRIT. TEMP RISE (Tcrit):	325
CRITICAL NET ENERGY ADDITION - BTU (Qnet)		1267200	CRITICAL TOTAL ENERGY RELEASE - BTU (Qtot)		8.45E+06	[Qnet/(1-0.85)]
			SCREENING CRITERIA			
IGNITION SOURCES	HEAT RELEASE RATE (q) BTU/SEC (3)	LOCATION FACTOR (1, 2 OR 4)	EFFECTIVE HEAT RELEASE RATE (qeff) BTU/SEC	COMBUSTIBLE LOAD BTU (4)	DAMAGE HEIGHT (H) FT (1)	CRITICAL RADIAL DISTANCE FT (2)
480V RMOV BD 2D	106	1	106	4.62E+05	6.82	2.60
Electrical cabinet		(> 4 ft from wall)				
Fire Zone 2-4						
Notes: 1.	Electrical cabinets: No Vents, closed doors					
2.	HRR: See Attachment A					
3.	Qnet/ft3 (for combustible load contribution) = 462,000 x .15/281600			0.25		
	Smoke layer temp rise = 14 F (from Table 7E, Ref 1); Effective Tcri			311		
Unit 2 Preferred AC	112	2	224	2.80E+05	9.13	3.78
Transformer		(<2 ft from wall)				
Fire Zone 2-3						
Notes: 1.	Transformer: Dry type, enclosed no vents					
2.	HRR: See Attachment A					
3.	Qnet/ft3 (for combustible load contribution) = 280,000 x .15/281600			0.15		
	Smoke layer temp rise = 10 F (from Table 7E, Ref 1); Effective Tcri			315		
RBCCW pump 2A	30	1	30	7.50E+04	4.01	1.38
Fire Zone 2-4		(away from wall)				
RBCCW pump 2B	30	1	30	7.50E+04	4.01	1.38
Fire Zone 2-4		(away from wall)				
Notes: 1.	Pump: 75 HP, Class 1E, no oil, limited ventilation at bottom.					
2.	HRR: Based on closed electrical cabinet with non-qualified cable, See Attachment A					
3.	Smoke layer temperature rise is neglected due to minimum amount of combustible loading.					
(1) H = [340 * qeff^2/3 / Effective Tcrit]^3/5						
(2) Rcrit = [q * 0.4 / 4 * pi * Q"crit]^1/2						

BUILDING: REACTOR BUILDING UNIT 2, EL 593			Page 4 of 11		/home05/ignition freq (five)/walkdown.2-593ii	
FIRE AREA/ZONE: 2-3 and 2-4						
CEILING HEIGHT:(above source)	22	DAMAGE TEMP:	425	CRIT. HEAT FLUX (Q"crit)	0.50	BTU/SEC/FT2
FLOOR AREA (FT2)	12800	VOLUME (V): FT3	281600	AMBIENT TEMP (Tamb):	100	
HEAT ADDITION PER UNIT VOL. (Q/V) TO FORM HOT GAS LAYER :			4.5	(FROM TABLE 7E @ 325 F)	CRIT. TEMP RISE (Tcrit):	325
CRITICAL NET ENERGY ADDITION - BTU (Qnet)		1267200	CRITICAL TOTAL ENERGY RELEASE - BTU (Qtot)		8.45E+06	[Qnet/(1-0.85)]
			SCREENING CRITERIA			
IGNITION SOURCES	HEAT RELEASE RATE (q) BTU/SEC (3)	LOCATION FACTOR (1, 2 OR 4)	EFFECTIVE HEAT RELEASE RATE (qeff) BTU/SEC	COMBUSTIBLE LOAD BTU (4)	DAMAGE HEIGHT (H) FT (1)	CRITICAL RADIAL DISTANCE FT (2)
Fire Protection Battery No. 2	Only used as a junction box. Old fire protection system has been decommissioned. No fire impact.					
and Charger						
Fire Zone 2-3						
Fire Protection Panel	Only used as a junction box. Old fire protection system has been decommissioned. No fire impact.					
DC Supply- JB 5664						
Fire Zone 2-3						
RCW Booster Pump	Normally Denergized, no oil, Totally enclosed motor, Neglect any fire impact.					
Fire Zone 2-3						
(1) H = [340 * qeff^2/3 / Tcrit]^3/5						
(2) Rcrit = [q * 0.4 / 4 * pi * Q"crit]^1/2						

BUILDING: REACTOR BUILDING UNIT 2, EL 593 (SDBR Roof, EL 604)			Page 5 of 11		/home05/ignition freq (five)/walkdown.2-593iii	
FIRE AREA/ZONE: 2-3 and 2-4						
CEILING HEIGHT:(above source)	15	DAMAGE TEMP:	425	CRIT. HEAT FLUX (Q"crit)	0.50	BTU/SEC/FT2
FLOOR AREA (FT2)	12800	VOLUME (V): FT3	192000	AMBIENT TEMP (Tamb):	100	
HEAT ADDITION PER UNIT VOL. (Q/V) TO FORM HOT GAS LAYER :			4.5	(FROM TABLE 7E @ 325 F)	CRIT. TEMP RISE (Tcrit):	325
CRITICAL NET ENERGY ADDITION - BTU (Qnet)		864000	CRITICAL TOTAL ENERGY RELEASE - BTU (Qtot)		5.76E+06	[Qnet/(1-0.85)]
			SCREENING CRITERIA			
IGNITION SOURCES	HEAT RELEASE RATE (q) BTU/SEC (3)	LOCATION FACTOR (1, 2 OR 4)	EFFECTIVE HEAT RELEASE RATE (qeff) BTU/SEC	COMBUSTIBLE LOAD BTU (4)	DAMAGE HEIGHT (H) FT (1)	CRITICAL RADIAL DISTANCE FT (2)
Shutdown Bd Rm 2C/2D AC units						
Fire Zone 2-3/2-4 (EL 604)						
Compressor	Totally enclosed and sealed oil contents. No fire impact					
	No oil changes are required precluding any spill potential.					
Motor	30	1	30	7.00E+04	4.01	1.38
	(See Attachment A)			(Based on 5 lb of plastic		
				@ 14000 Btu/lb)		
	Smoke layer temperature rise is neglected due to minimum amount of combustible loading.					
Fan/Motor	The fan/motor is totally enclosed within a large steel housing.					
	Neglighible amount of combustibles within the housing.					
	Any heat release will have minimal impact outside the fan/motor housing.					
RWCU Pump Room Monitor	130	2	260	2.80E+05	9.69	4.07
Fire Zone 2-4		(<2 ft from wall)		(Based on 5 lb of plastic		
(EL 593,)				@ 14000 Btu/lb)		
Ceil Ht. 22' above fire source						
Notes: 1.	HRR: Based on Television set burning, Reference 4.					
	Table 2-1.7, Test 2.					
2.	Qnet/ft3 (for combustible load contribution) = 280,000 x .15/281,60(			0.15		
	Smoke layer temp rise = 10 F (from Table 7E, Ref 1); Effective Tcri			315		
Radio Repeater	No fire impact.					
UPS - F4	Small voltage electronic instruments. EPRI fire events Data base does not identify this equipment to be a viable fire source.					
Fire zone 2-4, EL 593						
Ceil Ht. 22, above fire source.						
(1) H = [340 * qeff^2/3 / Effective Tcrit]^3/5						
(2) Rcrit = [q * 0.4 / 4 * pi * Q"crit]^1/2						

BUILDING: REACTOR BUILDING UNIT 2, EL 621			Page 6 of	11	/home05/ignition freq (five)/walkdown.2-621i	
FIRE AREA/ZONE: 2-5						
CEILING HEIGHT:(above source)	13	DAMAGE TEMP:	425	CRIT. HEAT FLUX (Q"crit)	0.50	BTU/SEC/FT2
FLOOR AREA (FT2)	9640	VOLUME (V): FT3	125320	AMBIENT TEMP (Tamb):	100	
HEAT ADDITION PER UNIT VOL. (Q/V) TO FORM HOT GAS LAYER :			4.5	(FROM TABLE 7E @ 325 F)	CRIT. TEMP RISE (Tcrit):	325
CRITICAL NET ENERGY ADDITION - BTU (Qnet)		563940	CRITICAL TOTAL ENERGY RELEASE - BTU (Qtot)		3.76E+06	[Qnet/(1-0.85)]
			SCREENING CRITERIA			
IGNITION SOURCES	HEAT RELEASE RATE (q) BTU/SEC (3)	LOCATION FACTOR (1, 2 OR 4)	EFFECTIVE HEAT RELEASE RATE (qeff) BTU/SEC	COMBUSTIBLE LOAD BTU (4)	DAMAGE HEIGHT (H) FT (1)	CRITICAL RADIAL DISTANCE FT (2)
480V RMOV BD 2E	106	4	424	1.63E+05	11.81	2.60
Electrical cabinet		(in corner)				
Notes: 1.	Electrical cabinets: non-qualified, closed doors					
2.	HRR: Based on closed cabinet for non-qualified cable , See Attachment A					
3.	Qnet/ft3 (for combustible load contribution) = 163,000 x .15/125320			0.19		
	Smoke layer temp rise = 11 F (from Table 7E, Ref 1); Effective Tcrit			314		
MG Sets 2DN and 2EA	112	1	112	2.80E+05	7.04	2.67
		(away from wall)		(For each MG Set)		
Notes: 1.	EPRI fire events database shows that most fires associated with MG sets were due to dust buildup on motors.					
	Therefore, the fire will be considered similar to a motor fire. As described in Attachment A, the HRR for motors will be					
	based on closed non-qualified electrical cabinet and adjusted for its combustible loading.					
2.	Qnet/ft3 (for combustible load contribution) = 280,000 x .15/125320			0.34		
	Smoke layer temp rise = 19 F (from Table 7E, Ref 1); Effective Tcrit			306		
4KV-480V Transformer	112	2	224	2.80E+05	9.29	3.78
TS2A AND TS2B		(close to wall)				
Notes: 1.	Transformer: Dry Type, concrete wall partition between two transformers.					
2.	HRR: Similar to closed electrical cabinet with non-qualified cable, See Attachment A					
3.	Qnet/ft3 (for combustible load contribution) = 280,000 x .15/125320			0.34		
	Smoke layer temp rise = 19 F (from Table 7E, Ref 1); Effective Tcrit			306		
(1) H = [340 * qeff^2/3 /Effective Tcrit]^3/5						
(2) Rcrit = [q * 0.4 / 4 * pi * Q"crit]^1/2						

BUILDING: REACTOR BUILDING UNIT 2, EL 621			Page 7 of 11		/home05/ignition freq (five)/walkdown.2-621iii	
FIRE AREA/ZONE: 2-5						
CEILING HEIGHT:(above source)	13	DAMAGE TEMP:	425	CRIT. HEAT FLUX (Q"crit)	0.50	BTU/SEC/FT2
FLOOR AREA (FT2)	9640	VOLUME (V): FT3	125320	AMBIENT TEMP (Tamb):	100	
HEAT ADDITION PER UNIT VOL. (Q/V) TO FORM HOT GAS LAYER :			4.5	(FROM TABLE 7E @ 325 F)	CRIT. TEMP RISE (Tcrit):	325
CRITICAL NET ENERGY ADDITION - BTU (Qnet)		563940	CRITICAL TOTAL ENERGY RELEASE - BTU (Qtot)		3.76E+06	[Qnet/(1-0.85)]
			SCREENING CRITERIA			
IGNITION SOURCES	HEAT RELEASE RATE (q) BTU/SEC (3)	LOCATION FACTOR (1, 2 OR 4)	EFFECTIVE HEAT RELEASE RATE (qeff) BTU/SEC	COMBUSTIBLE LOAD BTU (4)	DAMAGE HEIGHT (H) FT (1)	CRITICAL RADIAL DISTANCE FT (2)
2-LPNL-025-0031	106	1	106	6.53E+05	7.26	2.60
RCIC Aux Control panel		(away from wall)				
Notes: 1.	Control Panel: non-qualified, closed doors, no vents					
2.	HRR: See Attachment A					
3.	The control panel is <3' away from the transfomer. Fire is not likely to propagate to the transformer due to adequate distance. and closed doors.					
	Radiant exposure may cause damage to the transformer, however, temperatures will not reach auto ignition point.					
4.	Qnet/ft3 (for combustible load contribution) = 653,000 x .15/125320			0.78		
	Smoke layer temp rise = 45 F (from Table 7E, Ref 1); Effective Tcri			280		
4KV RPT BD 2-1, Panel 1 and 2	106	1	106	2.80E+05	6.90	2.60
Recirc Pump 2A RPT Breaker		(away from wall)				
Notes: 1.	Panel: Non-qualified, closed door					
2.	HRR: Based on closed cabinet for non-qualified cable , See Attachment A					
3.	Combustible load: From Reference 3.					
4.	Qnet/ft3 (for combustible load contribution) = 280000 x .15/125320			0.34		
	Smoke layer temp rise = 20 F (from Table 7E, Ref 1); Effective Tcri			305		
4KV RPT BD 2-2, Panel 1 and 2	106	1	106	2.80E+05	6.90	2.60
Recirc Pump 2B RPT Breaker		(away from wall)				
Notes: 1.	Panel: Non-qualified, closed door					
2.	HRR: Based on closed cabinet for non-qualified cable , See Attachment A					
3.	Qnet/ft3 (for combustible load contribution) = 280000 x .15/125320			0.34		
	Smoke layer temp rise = 20 F (from Table 7E, Ref 1); Effective Tcri			305		
Panel 25-3 (F/D Influent,bypass)	50	1	50	1.40E+04	4.92	1.78
Notes: 1.	Control panel: non-qualified, closed door					
2.	HRR: See Attachment A					
4.	Fire is not likely to propagate to Panel 25-9 due to closed cabinet steel construction and adequate separation.					
	Radiant exposure may cause damage , however temperatures will not be high enough for auto ignition.					
	Smoke layer temperature rise is neglected due to minimum amount of combustible loading.					
Panel 25-9	50	1	50	1.40E+04	4.92	1.78
Sample panel	(Similar to Panel 25-3)					
(1) H = [340 * qeff^2/3 / Effective Tcrit]^3/5		(2) Rcrit = [q * 0.4 / 4 * pi * Q"crit]^1/2				

BUILDING: REACTOR BUILDING UNIT 2, EL 621			Page 8 of 11		/home05/ignition freq (five)/walkdown.2-621ii	
FIRE AREA/ZONE: 2-5						
CEILING HEIGHT:(above source)	13	DAMAGE TEMP:	425	CRIT. HEAT FLUX (Q"crit)	0.50	BTU/SEC/FT2
FLOOR AREA (FT2)	9640	VOLUME (V): FT3	125320	AMBIENT TEMP (Tamb):	100	
HEAT ADDITION PER UNIT VOL. (Q/V) TO FORM HOT GAS LAYER :			4.5	(FROM TABLE 7E @ 325 F)	CRIT. TEMP RISE (Tcrit):	325
CRITICAL NET ENERGY ADDITION - BTU (Qnet) Q/V*V	563940		CRITICAL TOTAL ENERGY RELEASE - BTU (Qtot)		3.76E+06	[Qnet/(1-0.85)]
			SCREENING CRITERIA			
IGNITION SOURCES	HEAT RELEASE RATE (q) BTU/SEC (3)	LOCATION FACTOR (1, 2 OR 4)	EFFECTIVE HEAT RELEASE RATE (qeff) BTU/SEC	COMBUSTIBLE LOAD BTU (4)	DAMAGE HEIGHT (H) FT (1)	CRITICAL RADIAL DISTANCE FT (2)
CRD Repair Rm, Filter/Fan	Negligible combustible loading, Neglect any fire impact					
Janitor Closet	Small amounts of combustibles (mop, plastic,cans) kept in a locked metal closet.					
	Any fire will be contained within the closet. Neglect fire impact outside the closet.					
Primary Containment Purge	The purge unit is used to purge containment during plant shutdown. It is not operated continously.					
Filter Unit	The heating of the charcoal beds due to any radioactive decay heat build up will be negligible. Therefore , a fire					
	is unlikely. Even if the fire occurs, it will be contained within the stainless steel sealed housing.					
	The purge unit is not required for the safe shutdown of the plant.					
	Therefore, there will be no fire impact. due to the purge unit.					
240V Lighting Board 2A	106	1	106	1.12E+05	6.75	2.60
Electrical Cabinet		(away from wall)			(note 4)	
Notes: 1.	Electrical Cabinet: Non-qualified, closed door					
2.	HRR: See Attachment A					
3.	Qnet/ft3 (for combustible load contribution) =112,000 x .15/125320			0.13		
	Smoke layer temp rise = 9 F (From table 7E, Ref 1); Effective Tcrit			316		
4.	Possibility of forming ceiling jet sub layer - See detailed evaluation					
240V Lighting Transformer		1	6075	7.35E+06	33.57	19.67
Transformer TL-2A						
	The combustion of transformer will form hot gas layer on this floor.					
	See detailed evaluation.					
(1) H = [340 * qeff^2/3 /Tcrit]^3/5						
(2) Rcrit = [q * 0.4 / 4 * pi * Q"crit]^1/2						

BUILDING: REACTOR BUILDING UNIT 2, EL 639 (south side)			Page 9 of 11		/home05/ignition freq (five)/walkdown.2-639i	
FIRE AREA/ZONE: 2-6						
CEILING HEIGHT:(above source)	18	DAMAGE TEMP:	425	CRIT. HEAT FLUX (Q"crit)	0.50	BTU/SEC/FT2
FLOOR AREA (FT2)	8600	VOLUME (V): FT3	154800	AMBIENT TEMP (Tamb):	100	
HEAT ADDITION PER UNIT VOL. (Q/V) TO FORM HOT GAS LAYER :			4.5	(FROM TABLE 7E @ 325 F)	CRIT. TEMP RISE (Tcrit):	325
CRITICAL NET ENERGY ADDITION - BTU (Qnet)		696600	CRITICAL TOTAL ENERGY RELEASE - BTU (Qtot)		4.64E+06	[Qnet/(1-085)]
			SCREENING CRITERIA			
IGNITION SOURCES	HEAT RELEASE RATE (q) BTU/SEC (3)	LOCATION FACTOR (1, 2 OR 4)	EFFECTIVE HEAT RELEASE RATE (qeff) BTU/SEC	COMBUSTIBLE LOAD BTU (4)	DAMAGE HEIGHT (H) FT (1)	CRITICAL RADIAL DISTANCE FT (2)
Recirculation Pump Motor				5.22E+08		
Generator Sets 2A and 2b						
Notes:	The combustion of MG sets oil contents will cause hot gas layer in the area.					
	See detailed evaluati					
4KV - 480V Emergency				1.65E+07		
Transformer 2A and 2B						
Notes:	The combustion of transformer oil contents will cause hot gas layer in the area. Approximately 9' x 9' curb around the transformer					
	See detailed evaluation.					
MG Sets 2DA and 2EN	112	1	112	2.80E+05	6.99	2.67
		(away from wall)		(Per MG Set)		
Notes: 1.	EPRI fire events database shows that most fires associated with MG sets were due to dust buildup on motors.					
	Therefore, the fire will be considered similar to a motor fire. As described in Attachment A, the HRR for motors will be					
	based on closed non-qualified electrical cabinet and adjusted for its combustible loading.					
2.	Combustible load: From Reference 3.					
3.	The MG Sets are approximately 40" from each other. One Mg Set will not ignite the other due to adequate separation					
	No conduits or cable trays within the zone of influence.					
4.	Qnet/ft3 (for combustible load contribution) = 280,000 x .15/154800			0.27		
	Smoke layer temp rise = 15 F (from Table 7E, Ref 1); Effective Tcri			310		
LPNL-25-23 and 24	106	1	106	6.30E+05	7.13	2.60
Recirc Pump control cabinets						
Notes: 1.	Control Cabinet: non-qualified, closed door					
2.	HRR: Similar to closed electrical cabinet with non-qualified cable, See Attachment A					
3.	The two control cabinets are ~2' apart. The fire from one cabinet will not propagate to the other due to					
	closed doors, steel cabinet construction and adequate separation. The radiant heat however, may cause damage to internal components.					
4.	Qnet/ft3 (for combustible load contribution) = 630,000 x .15/154800			0.61		
	Smoke layer temp rise = 36 F (from Table 7E, Ref 1); Effective Tcri			289		
(1) H = [340 * qeff^2/3 /Effective Tcrit]^3/5						
(2) Rcrit = [q * 0.4 / 4 * pi * Q"crit]^1/2						

BUILDING: REACTOR BUILDING UNIT 2, EL 639 (south side)			Page 10 of 11		/home05/ignition freq (five)/walkdown.2-639ii	
FIRE AREA/ZONE: 2-6						
CEILING HEIGHT:(above source)	18	DAMAGE TEMP:	425	CRIT. HEAT FLUX (Q"crit)	0.50	BTU/SEC/FT2
FLOOR AREA (FT2)	8600	VOLUME (V): FT3	154800	AMBIENT TEMP (Tamb):	100	
HEAT ADDITION PER UNIT VOL. (Q/V) TO FORM HOT GAS LAYER :			4.5	(FROM TABLE 7E @ 325 F)	CRIT. TEMP RISE (Tcrit):	325
CRITICAL NET ENERGY ADDITION - BTU (Qnet)		696600	CRITICAL TOTAL ENERGY RELEASE - BTU (Qtot)		4.64E+06	[Qnet/(1-0.85)]
			SCREENING CRITERIA			
IGNITION SOURCES	HEAT RELEASE RATE (q) BTU/SEC (3)	LOCATION FACTOR (1, 2 OR 4)	EFFECTIVE HEAT RELEASE RATE (qeff) BTU/SEC	COMBUSTIBLE LOAD BTU (4)	DAMAGE HEIGHT (H) FT (1)	CRITICAL RADIAL DISTANCE FT (2)
240V Lighting Board 2B	106	1	106	1.12E+05	6.74	2.60
Electrical Cabinet		(away from wall)				
Notes: 1.	Electrical Cabinet: Non-qualified, closed door					
2.	HRR: See Attachment A					
3.	See detailed evaluation for consideration of adjacent combustibles.					
4.	Qnet/ft3 (for combustible load contribution) = 112,000 x .15/154800			0.11		
	Smoke layer temp rise = 8 F (from Table 7E, Ref 1); Effective Tcrit			317		
240V Lighting Transformer				7.35E+06		
TL 2B					1.00	
	The combustion of transformer will form hot gas layer on this floor.					
	See detailed evaluation.					
SLC Pumps A and B						
Motor	112	1	112	2.80E+05	6.79	2.67
Oil	450	1	450	2.80E+05	12.19	5.35
Fire Zone 2-5						
Notes: 1.	Approximately 20 lbs of plastic is estimated for each motor which equates to a combustible loading of 280,000 Btu.					
	HRR is estmated based on one closed electrical cabinet and adjusted for combustible loading. See Attachment A.					
2.	The pump oil is in a sealed enclosure. No visible leakage was found. However, assume 1/2 pint oil spill. See Attachment A for HRR rate calculat					
3.	Qnet/ft3 (for combustible load contribution) = 280,000 x .15/154800			0.78		
	Smoke layer temp rise = 15 F (from Table 7E, Ref 1); Effective Tcri			310		
(1) H = [340 * qeff^2/3 / Tcrit]^3/5						
(2) Rcrit = [q * 0.4 / 4 * pi * Q"crit]^1/2						

BUILDING: REACTOR BUILDING UNIT 2, EL 519			Page 11 of 11		/home05/ignition freq (five)/walkdown.2-519l	
FIRE AREA/ZONE: 2-1and 2-2						
CEILING HEIGHT:(above source)	varies	DAMAGE TEMP:	425	CRIT. HEAT FLUX (Q"crit)	0.50	BTU/SEC/FT2
FLOOR AREA (FT2)	na	VOLUME (V): FT3	na	AMBIENT TEMP (Tamb):	100	
HEAT ADDITION PER UNIT VOL. (Q/V) TO FORM HOT GAS LAYER :			4.5	(FROM TABLE 7E @ 325 F)	CRIT. TEMP RISE (Tcrit):	325
CRITICAL NET ENERGY ADDITION - BTU (Qnet)		na	CRITICAL TOTAL ENERGY RELEASE - BTU (Qtot)		na	(Qnet/(1-0.85))
			SCREENING CRITERIA			
IGNITION SOURCES	HEAT RELEASE RATE (q) BTU/SEC (3)	LOCATION FACTOR (1, 2 OR 4)	EFFECTIVE HEAT RELEASE RATE (qeff) BTU/SEC	COMBUSTIBLE LOAD BTU (4)	DAMAGE HEIGHT (H) FT (1)	CRITICAL RADIAL DISTANCE FT (2)
RHR Pumps 2A/2C	The RHR and core spray pumps are located in four separate quadrants on elevation 519 (torus area) of the reactor building. Each quadrant is widely separated					
RHR Pumps 2B/2D	from other quadrants. The amount of combustibles in the Torus area is negligible and does not provide continuity for propagation of fire. A fire in any quadrant					
Core Spray Pumps 2A/2C	will thus be confined to that area and will not affect the redundant divisions. The oil contained in these pumps					
Core Spray Pumps 2B/2D	are in sealed housings. There is no indication of oil leakage in the area. The HVAC equipment for these pumps are also located in the same					
	quadrants up close to the ceiling. The fan motors are located outside the housing and are relatively small HP. The fire hazards due to these					
Associated Miscellaneous	motors is negligible. A walkdown of this area indicates that all equipment located in these quadrants are associated with the respective pumps.					
Equipment including HVAC	Therefore, a fire in this area will only affect RHR or core spray systems. Note that HPCI room is located next to RHR pump room 2B/2D and will be considered					
	involved in the same fire for conservatism. Similarly RCIC pump will be considered involved in a core spray pump 2B/2D fire.					

(1) H = [340 * qeff^2/3 / Tcrit]^3/5

(2) Rcrit = [q * 0.4 / 4 * pi * Q"crit]^1/2

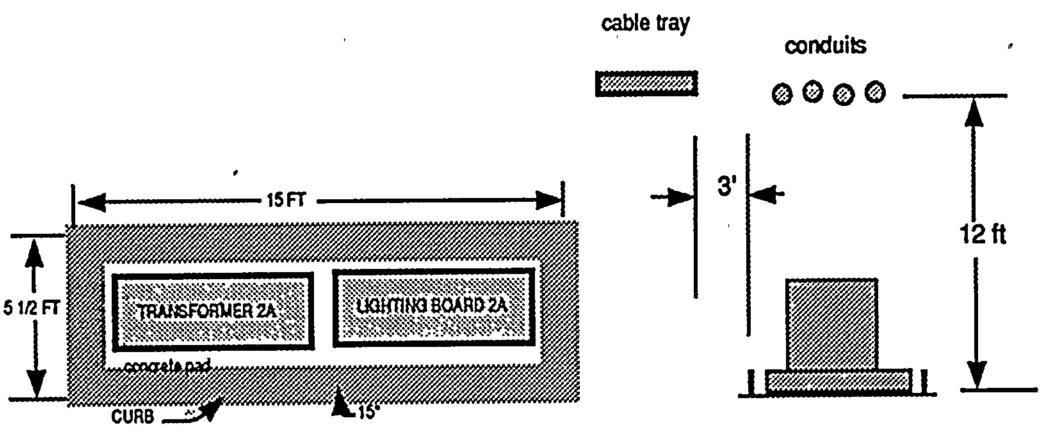
C.3 Detailed Evaluation

The following components were identified for detailed evaluation due to the potential for hot gas layer build up assuming consumption of all combustible materials:

- 240V Lighting transformer TL2A, Reactor Building EL 621, Fire Zone 2-5
- 240V Lighting transformer TL2B, Reactor Building EL 639, Fire Zone 2-6
- Recirculation Pump Motor Generator sets 2A and 2B, Reactor Building EL 639, Fire Zone 2-6
- 4KV-480V Emergency Transformer TS-2E, Reactor Building EL 639, Fire Zone 2-6
- RHR pumps, Reactor Building EL 519, Fire Zone 2-1 and 2-2
- -- Core spray pumps, Reactor Building EL 519, Fire Zone 2-1 and 2-2

Each one of these components are analyzed as follows:

- 240V Lighting transformer TL2A, Reactor Building EL 621, Fire Zone 2-5



The transformer is located in the NE section of EL 621 and is connected to 240V Lighting Board 2A. The lighting board has been individually analyzed Attachment C.2. It was determined that ceiling jet sub layer is likely to form, because the fire plume of 6.75 ft. above the lighting board 2A12/14/94 (8 ft. high) is within the ceiling jet space. However, since the cabinet has no unsealed top penetrations, the base of the fire will be located much lower and the ceiling jet

sub layer is not likely to form. The fire scenario associated with the transformer and the lighting board is analyzed in detail as follows:

As shown in the above figure, the transformer and the lighting board are located on a concrete pad. A curb spaced ~ 15" from the concrete pad encloses the transformer and the board. Oil spill from the transformer will be contained in the curbed area. A fire will therefore be limited to the surface area within the curbed space. Heat release rates are calculated based on this exposed surface area. Any heat release contribution due to the lighting board is neglected. Enough conservatism is built in the assumption that sufficient oil will spill to fill up the entire curbed space. Also the lighting cabinet housing is of steel construction and doors are kept closed. A fire outside may damage the components within (some possible ignition) but is not likely to add significantly to the overall heat release rates. The accompanying calculations show that the conduits, being in the fire plume, will be damaged prior to sprinkler activation. However, the cable trays are in the hot gas layer. The hot gas layer temperature was calculated to be below the target (cable tray) damage threshold.

The hot gas layer temperature was estimated using the NIST fire model CFAST Version 2.0 for a single compartment. To run the model, the room size was conservatively assumed to be 80' x 80' and a vent opening was assumed to be 5' x 5'. Minor door cracks were also modeled. The doors remained closed during the simulation. A constant heat release rate of 6000 btu/sec (see page C.3-4) was used to define the fire size. The hot gas layer temperature was calculated to be approximately 500 K or 440 F for the duration. This is borderline damage temperature for non-qualified cables. However, it is expected that the actual temperature will be much lower due to conservatism used in the model i.e., smaller compartment size, entire curbed area filled with oil, instantaneous heat contribution and no credit for suppression. Also note that 425 F cable damage temperature was used for the initial screening, and that actual damage temperatures will be much higher. Therefore, apart from the conduits in the fire plume or within range of radiant exposure, no other conduits or cables will be subjected to damaging temperatures. The P2 values calculated in section 6.1.2 are however based on a more conservative assumption of only single train of components, routed outside the fire zone, are available. Detailed calculations are shown on sheet C.3-4. The CFAST fire model run is shown on sheet C.3-5 through C.3-9.

- 240V Lighting transformer TL2B, Reactor Building EL 639, Fire Zone 2-6

This transformer and the curbing are very similar to the one evaluated above. The heat release rate based on the curbed surface area will also be approximately the same as evaluated above (6000 btu/sec), This area has detection coverage but no suppression. The ceilings are high (~ 24' as opposed to 17' for transformer 2A), consequently the hot gas layer temperatures will be much less than 410 F as calculated above. Therefore, there is no liklihood of any components damage in the area unless in the immediate vicinity. There is only one conduit (2ES3925) in the vicinity (no cable trays or conduits in the overhead), and will be assumed damaged along with the lighting board.

- Recirculation Pump Motor Generator sets 2A and 2B, Reactor Building EL 639 Fire Zone 2-6; 4KV-480V Emergency Transformer TS-2E, Reactor Building EL 639, Fire Zone 2-6

Due to the potential size of the fire from this fire source, this zone of influence comprises all of fire zone 2-6. Damage to components and potential impact on core damage frequency is evaluated in section 6.1.2.

- RHR pumps, Reactor Building EL 519, Fire Zone 2-1 and 2-2
Core spray pumps, Reactor Building EL 519, Fire Zone 2-1 and 2-2

RHR and core spray pumps are located on EL 519 of the reactor building. Each division of the RHR and core spray pumps are located in separate quadrants, well isolated from each other by the torus area space. The large separation between pump quadrants and lack of combustibles in the torus area will prevent fire from propagating between quadrants. Thus a fire in any pump area will be confined to that area. These pumps contain oil within sealed housings. There is no evidence of any leakage or spill. The pumps are on standby and are normally not operating. Therefore a fire is not likely to originate in this area. Plant walkdowns have shown that all electrical cables and conduits in the pump quadrants are associated with the equipment located within the area. As a conservative approach, the entire pump quadrant will be considered within the fire zone of influence. The P2 values calculated in section 6.1.2 are based on this conservative assumption.

240V Lighting Transformer: (TL2A)

This transformer was identified to cause hot gas layer if all the combustibles were consumed. The transformer contains large quantity of oil. The oil fire exposure hazard is limited by the amount of spill. Curbes are provided around the transformer limiting the spread of . oil spill.

The spill (curbed) area is approx. (ft2)	45	(15 * 1.25 * 2 + 3 * 1.25 * 2)
Heat release rate (btu/sec/ft2) =	135	(Reference 1, Table 3)
Heat release (btu/sec) =	6075	

This is significant amount of heat release and equates to a damage height of 34 ft and critical radial distance of 20 ft. Due to the ceiling height being approximately 17' at the fire source location, the fire plume will impinge on the ceiling and form ceiling jet layer, thus causing additional damage.

AUTOMATIC SPRINKLER SYSTEM EVALUATION

TIME FOR CRITICAL DAMAGE (SEC)			TIME FOR SPRINKLER SYSTEM ACTUATION (SEC)		
Assess the damage potential of conduits located 12' directly above the transformer prior to sprinkler actuation:					
IGNITION SOURCE: Transformer	**Target:**	conduit			
TARGET THERMAL RESPONSE PARAMETER	20	Cable in Conduit	SPRINKLER TEMP. RATING (Tsprink)	200	F
PEAK HEAT RELEASE RATE (q)	6000		RADIAL DISTANCE SOURCE TO SPRINKLER	6	Ft
EFFECTIVE HEAT RELEASE RATE (qeff)	6000		DISTANCE FROM SOURCE TO CEILING (H)	16	Ft
RADIANT HEAT RELEASE RATE (qrad)	2400	q * 0.4	SPRINKLER TIME CONSTANT	100	
LINE OF SIGHT DISTANCE FROM SOURCE (R)	12		GAS TEMP. RISE AT CEILING (DTgas,plume)'	1127	340 * [(qeff)^.67/H^1.67]
RADIAL DISTANCE (r)	0		LESSER OF DTgas' OR 1600 (DTgas,plume)	1127	
TARGET HEIGHT (Z)	12		GAS TEMP. RISE AT SPRINKLER (DTgas,sprin	650	DTgas,plume * 0.3/((r/H)^0.667
RADIANT HEAT FLUX AT TARGET (Qrtarget)	1.33	qrad/4*3.14*R^2	SPRINKLER TEMP RISE/GAS TEMP RISE	0.15	(Tsprink - Tamb) / DTgas,sprink
CONVECTIVE HEAT FLUX IN PLUME (Qcplume)	12.50	0.3*qeff/z^2	DIMENSIONLESS ACTUATION TIME OF SPRI	0.17	-ln[1-(DTsprink/Dtgas,sprink)]
LESSER OF Qcplume OR 9	9.00				
CONVEC. HEAT FLUX AT TARGET (Qctarget)	na	Qcplume*0.13/((r/Z)^.3			
TOTAL HEAT FLUX (Qtotal)	10.33	Qrtarget + Qctarget	**ESTIMATED SPRINKLER ACTUATION TIME (S**	17	Dimensioless act * Time const.
TIME TO CRITICAL DAMAGE (SEC)	3	(3.14/4)(TRP/Qtotal)^2			

Assess the damage potential of cable trays located 12' above the transformer and 3' radially, prior to sprinkler actuatio					
IGNITION SOURCE: Transformer	**Target: cable tra**				
TARGET THERMAL RESPONSE PARAMETER	20	Cable tray			
PEAK HEAT RELEASE RATE (q)	6000				
EFFECTIVE HEAT RELEASE RATE (qeff)	6000				
RADIANT HEAT RELEASE RATE (qrad)	2400	q * 0.4			
LINE OF SIGHT DISTANCE FROM SOURCE (R)	13				
RADIAL DISTANCE (r)	3				
TARGET HEIGHT (Z)	12				
RADIANT HEAT FLUX AT TARGET (Qrtarget)	1.13	qrad/4*3.14*R^2			
CONVECTIVE HEAT FLUX IN PLUME (Qcplume)	na	0.3*qeff/z^2			
LESSER OF Qcplume OR 9	na				
CONVEC. HEAT FLUX AT TARGET (Qctarget) (1)	0.42	target in hot gas layer	**(1) Convective heat flux for targets in hot gas**		
TOTAL HEAT FLUX (Qtotal)	1.55	Qrtarget + Qctarget	Qctarget = 0.025(Thgl - To) KW/m^2-K =	4.73	(0.42 btu/sec/ft2)
TIME TO CRITICAL DAMAGE (SEC)	131	(3.14/4)(TRP/Qtotal)^2	Hot gas layer temperature (K) =	500	(from CFAST fire model run)
			Ambient temperature (K) =	311	(100 F)

This analysis shows that the conduits directly above the transformer will be damaged due to a fire prior to sprinkler actuation. Some conduits located directly behind the transformer (not evaluated) will also be damaged due to radiant heat flux. However, cable trays located above the transformer (in hot gas layer) will not be damaged prior to sprinkler activation.

DATA: TRANS.DAT

240 V Lighting Transformer, EL 621, fire Zone 2-5
Compartment Size = 80′ x 80′
Ceiling Height = 17 ft.
Vent Opening = 25 square ft
Door Opening = 5′ x 1′
Fire Size = 6000/btu/sec

```
VERSN     2 240V LIGHTING TRANSFORMER, EL 621 (2A)
TIMES    1200      60     60     60      0
TAMB    311.   101300.       0.
EAMB    314.   101300.       0.
HI/F      0.00
WIDTH    24.38
DEPTH    24.38
HEIGH     5.18
HVENT    1  2  1   1.524   0.305   0.000   0.000
CVENT    1  2  1   1.00    1.00    1.00    1.00    1.00    1.00    1.00
VVENT    2   1  2.32   2
CEILI CONCRETE
WALLS CONCRETE
FLOOR CONCRETE
CHEMI     16.     0.   10.0    41867304.   300.  388.    0.
LFBO    1
LFBT    2
FPOS    15.00  15.00   0.00
FTIME      20.     50.     50.    100.    100.    400.
FMASS   0.0000 0.1512 0.1512 0.1512 0.1512 0.1512 0.1512
FHIGH     0.00    0.00    0.00    0.00    0.00    0.00    0.00
FAREA     4.18    4.18    4.18    4.18    4.18    4.18    4.18
FQDOT   0.00      6.33E+06 6.33E+06 6.33E+06 6.33E+06 6.33E+06 6.33E+0
CJET ALL
HCR    0.333 0.333 0.333 0.333 0.333 0.333 0.333
CO     0.020 0.020 0.020 0.020 0.020 0.020 0.020
OD     0.020 0.020 0.020 0.020 0.020 0.020 0.020
STPMAX  5.00
DUMPR TRANS.HIS
DEVICE 1
WINDOW  0     0. -100. 1280. 1024. 1100.
GRAPH   1  100.    50.     0.  600.  475.    10.   3 TI HEIG
GRAPH   2  100.   550.     0.  600.  940.    10.   3 TI CELSI
GRAPH   3  720.    50.     0. 1250.  475.    10.   3 TI FIRE SIZE(k
GRAPH   4  720.   550.     0. 1250.  940.    10.   3 TI O|D2]O(
HEAT   0 0 0 0 3   1 U
TEMPE  0 0 0 0 2   1 U
INTER  0 0 0 0 1   1 U
O2     0 0 0 0 4   1 U
```

RESULT:
Hot Gas Layer ~ 500 OK (440 OF)

```
**      CFAST  Version   2.0  Run  9/27/94        **
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```

Time =   0.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	311.0	311.0	5.2	0.000	0.000
Outside					0.000

Time =   60.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	458.4	308.7	2.9	0.151	6.330E+06
Outside					0.000

Time =   120.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	465.9	308.0	1.6	0.151	6.330E+06
Outside					0.000

Time =   180.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	472.1	309.8	0.80	0.151	6.330E+06
Outside					0.000

Time =   240.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	479.4	314.4	0.36	0.151	6.330E+06
Outside					0.000

Time = 300.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	486.2	319.6	0.17	0.151	6.330E+06
Outside					0.000

Time = 360.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	490.8	322.8	0.10	0.151	6.330E+06
Outside					0.000

Time = 420.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	493.3	324.2	8.41E-02	0.151	6.330E+06
Outside					0.000

Time = 480.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	494.9	325.0	8.05E-02	0.151	6.330E+06
Outside					0.000

Time = 540.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	496.0	325.8	7.98E-02	0.151	6.330E+06
Outside					0.000

Time = 600.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	497.0	326.5	7.97E-02	0.151	6.330E+06
Outside					0.000

Time = 660.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	498.0	327.4	7.99E-02	0.151	6.330E+06
Outside					0.000

Time = 720.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	498.9	328.2	8.01E-02	0.151	6.330E+06
Outside					0.000

Time = 780.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	499.7	328.9	8.03E-02	0.151	6.330E+06
Outside					0.000

Time = 840.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	500.6	329.7	8.05E-02	0.151	6.330E+06
Outside					0.000

Time = 900.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	501.5	330.5	8.07E-02	0.151	6.330E+06
Outside					0.000

Time = 960.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	502.3	331.3	8.09E-02	0.151	6.330E+06
Outside					0.000

Time = 1020.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	503.2	332.0	8.11E-02	0.151	6.330E+06
Outside					0.000

Time = 1080.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	504.0	332.8	8.12E-02	0.151	6.330E+06
Outside					0.000

Time = 1140.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	504.8	333.5	8.14E-02	0.151	6.330E+06
Outside					0.000

Time = 1200.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	505.6	334.3	8.16E-02	0.151	6.330E+06
Outside					0.000

DATA:

AUXROOM.DAT

Unit 2 Auxiliary Room, EL 593 Control Building, Compartment 16-1
Compartment Size = 25'x 25'x 12 (room dimensions reflect area of room less 35%)
Heat of Combustion = 14000 Btu/lb; **Oxygen Limit** = 10%
Fire Size: Based on fire affecting an equivalent of 8 electrical cabinets with a peak HRR of 400 Btu/sec (Approx. 8 x 50 btu/s per Attachment A). Fire is postulated to grow from 0 to 400 btu/s in ~480 sec and thereafter remain steady for 1200 sec(20 min).

```
VERSN    2 UNIT 2 AUXILIARY INSTRUMENT ROOM
TIMES    1200    60     60     60     0
TAMB    300.  101300.      0.
EAMB    314.  101300.      0.
HI/F     0.00
WIDTH    7.62
DEPTH    7.62
HEIGH    3.66
HVENT   1  2  1   1.524   0.076   0.000   0.000
CVENT   1  2  1   1.00    1.00    1.00    1.00    1.00    1.00    1.00    1.00    1.00
CEILI  CONCRETE
WALLS  CONCRETE
FLOOR  CONCRETE
CHEMI    16.     0.   10.0    32563460.  300. 388.    0.
LFBO    1
LFBT    2
FPOS     3.00    3.00    1.52
FTIME     20.     50.     50.    100.    100.    138.    175.    250.    400.
FMASS  0.0000 0.0016 0.0024 0.0032 0.0049 0.0065 0.0081 0.0097 0.0113 0.0130
FHIGH    0.00    0.00    0.00    0.00    0.00    0.00    0.00    0.00    0.00    0.00
FAREA    4.18    4.18    4.18    4.18    4.18    4.18    4.18    4.18    4.18    4.18
FQDOT  0.00      5.28E+04 7.91E+04 1.06E+05 1.58E+05 2.11E+05 2.64E+05 3.17E+05 3
CJET OFF
HCR    0.333 0.333 0.333 0.333 0.333 0.333 0.333 0.333 0.333 0.333
CO     0.020 0.020 0.020 0.020 0.020 0.020 0.020 0.020 0.020 0.020
OD     0.020 0.020 0.020 0.020 0.020 0.020 0.020 0.020 0.020 0.020
STPMAX  5.00
DUMPR AUXROOM.HIS
DEVICE 1
WINDOW  0     0. -100. 1280. 1024. 1100.
GRAPH   1  100.   50.    0.   600.   475.    10.   3 TI HEIG
GRAPH   2  100.  550.    0.   600.   940.    10.   3 TI CELSI
GRAPH   3  720.   50.    0.  1250.   475.    10.   3 TI FIRE_SIZE(k
GRAPH   4  720.  550.    0.  1250.   940.    10.   3 TI O|D2|O(
HEAT  0  0  0  0  3   1 U
TEMPE 0  0  0  0  2   1 U
INTER 0  0  0  0  1   1 U
O2    0  0  0  0  4   1 U
```

RESULT:
Hot Gas Layer ~ 400 ^{0}K (260 ^{0}F)

```
**       CFAST  Version   2.0  Run 11/8/94          **
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Time = 0.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	300.0	300.0	3.7	0.000	0.000
Outside					0.000

Time = 60.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	346.1	300.1	2.8	3.540E-03	1.166E+05
Outside					0.000

Time = 120.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	383.3	300.3	2.0	7.342E-03	2.388E+05
Outside					0.000

Time = 180.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	419.2	300.9	1.5	9.807E-03	3.205E+05
Outside					0.000

Time = 240.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	442.2	302.0	1.4	1.109E-02	3.621E+05
Outside					0.000

Time = 300.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	452.4	303.3	1.3	1.187E-02	3.867E+05
Outside					0.000

Time = 360.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	458.4	304.7	1.3	1.255E-02	4.079E+05
Outside					0.000

Time = 420.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	463.2	306.2	1.2	1.300E-02	4.220E+05
Outside					0.000

Time = 480.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	464.3	307.5	1.2	1.300E-02	4.220E+05
Outside					0.000

Time = 540.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	435.8	309.3	1.3	1.300E-02	4.683E+04
Outside					0.000

Time = 600.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	393.1	310.0	1.5	1.300E-02	1.488E+05
Outside					0.000

Time = 660.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	387.2	309.9	1.5	1.300E-02	1.361E+05
Outside					0.000

Time = 720.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	382.0	309.7	1.5	1.300E-02	1.243E+05
Outside					0.000

Time = 780.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	377.3	309.4	1.5	1.300E-02	1.137E+05
Outside					0.000

Time = 840.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	373.0	309.2	1.5	1.300E-02	1.042E+05
Outside					0.000

Time = 900.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	369.1	308.9	1.5	1.300E-02	9.569E+04
Outside					0.000

Time = 960.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	365.5	308.7	1.5	1.300E-02	8.797E+04
Outside					0.000

Time = 1020.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	362.2	308.5	1.5	1.300E-02	8.094E+04
Outside					0.000

Time = 1080.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	359.0	308.3	1.5	1.300E-02	7.449E+04
Outside					0.000

Time = 1140.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	356.1	308.1	1.5	1.300E-02	6.854E+04
Outside					0.000

Time = 1200.0 seconds.

Compartment	Upper Temp. (K)	Lower Temp. (K)	Inter. Height (m)	Pyrol Rate (kg/s)	Fire Size (W)
1	353.3	308.0	1.5	1.300E-02	6.301E+04
Outside					0.000

ATTACHMENT D

PLANT WALKDOWNS

Attachment D

Plant Walkdowns

Plant walkdowns were performed to confirm the locations of potential fire sources and to identify any electrical raceways and components which can be affected by a postulated fire. The extent of fire damage was either based on an "engulfing" fire, such as the switchgear rooms or board rooms, or focused on a defined "zone of influence" (ZOI) for individual components in the Unit 2 Reactor Building.

Each of the plant switchgear rooms (shutdown board rooms) was walked down to identify plant components and cables routed through the affected area. This information then served to confirm the screening evaluations performed for these areas in Section 5.

Each of the fire sources identified in Attachment C were walked down to identify electrical raceways and other components located within the zone of influence. The results of this evaluation are implemented in Section 6.1.

This attachment is therefore subdivided into the following sections:

D.1 General instructions and detailed procedures developed to support the walkdown process.

D.2 Individual area and component zone of influence walkdown documentation sheets. This walkdown information was used to identify the potentially disabled equipment for a given fire location (see Reference 25).

D.3 Sample sheets from the evaluation of potentially disabled equipment (Reference 25).

D.1 WALKDOWN INSTRUCTION REQUIREMENTS

QA Record

WALKDOWN INSTRUCTION REQUIREMENTS

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT

WALKDOWN INSTRUCTION

NUMBER EEB-~~009~~ 010 WLA 7/5/94

REVISION 0

TITLE : NRC GENERIC LETTER 88-20, SUPPLEMENT 4 (NUREG-1407), IPEEE INTERNAL FIRES ENGINEERING ANALYSIS PLANT CONFIGURATION DATA WALKDOWNS

PREPARED BY : WILLIAM L. ALDREDGE WLA 7/5/94 DATE : 6-21-94

RESPONSIBLE ORGANIZATION : ENGINEERING AND MATERIALS

TECHNICAL REVIEW : RASHID ABBAS R Abbas DATE : 6-30-94

QA REVIEW : [signature] DATE : 7-1-94

APPROVED BY : for [signature] DATE : 7/5/94

EFFECTIVE DATE : 7/5/94

1. PURPOSE :

This procedure is to control the plant walkdown efforts necessary to support the "Internal Fires" portion of BFNP's response to NRC Generic Letter 88-20, Supplement 4 (NUREG-1407).

2. SCOPE :

Plant walkdowns controlled by this instruction are required to support BFNPs response to NUREG-1407 for Internal Fires. These walkdowns will involve equipment identified by the Engineering and Material's Nuclear/Mechanical Group (lead group for NUREG-1407).

2.1 BACKGROUND :

This procedure will be used to obtain the actual plant configuration, which is required to support BFNP's response to NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities". The specific walkdown information of concern will be that necessary to support the "Internal Fires" portion of BFNP's response to the NUREG. The walkdown requirement will be to determine the additional equipment (particularly electrical panels, conduits and cable trays), which will be involved or disabled by an exposure fire. This will be accomplished by:

1) Mechanical/Nuclear Engineering Group identifying the equipment which is the originating fire source, and a spatial fire damage "zone of influence" (ZOI), which is to be walked-down. This will then require a plant walkthrough to identify required plant support (if any). Based upon the plant configuration and the possible existance of intervening combustables, the required walkdown ZOI (see Att. B) will be finalized.
2) A walkdown team will be assembled and the walkdown team, using this procedure, will determine the plant equipment which is located within the ZOI. The information gathered by this walkdown will be used to determine, and identify, the full scope of equipment involved or disabled by the subject fire. Based upon this effort, the knowledge of involvement, and extent of severity, for the fire of concern will be determined.

The evaluation being performed to support BFNP's response to NUREG-1407 will be a one time effort, thus, this procedure will be deleted once this walkdown effort is completed. Additionally, the effort to support the NUREG response is not required to comply with the QA requirements imposed by 10CFR50 Appendix B. While not required by the NUREG, 0-TI-238 requires conformance with BFNP's QA program, thus, compliance with QA requirements will be applicable for this effort.

The walkdowns conducted using this procedure will primarly be recording conduits, cable trays, and possibly panels which are located in the ZOI of concern. Therefore, this procedure will deliberately lack detail instructions, since the walkdown team will have sufficient expertise to respond to unexpected or unique plant conditions.

2.2 WALKDOWN SCOPE :

The walkdown boundary for each individual piece of equipment will be identified by the Mechanical/Nuclear Group of Engineering and Materials. This will consist of the equipment considered as the initial fire source, and it's spatial ZOI (see Attachment B for applicable form).

3. DEFINITIONS :

3.1. Walkdown (W/D) - A QA documented information-gathering activity to verify and record the as-built status of plant areas or equipment.

3.2. Walkthrough - An activity that precedes the implementation of a walkdown of a plant area (or equipment), to determine plant support required during the walkdown.

3.3. Walkdown Package (WDP) - The issued package which documents the data gathered by the walkdown effort..

4. PRECAUTIONS :

The walkdowns conducted under the authorization of this instruction must comply with all plant procedures and instructions related to the activities being performed. The following safety practices are imposed in other documents but in the interest of safety will be repeated here.

1. Always exercise precaution in the vicinity of unenergized equipment, since it may be subject to automatic starting.
2. Always assure that any RAD CON surveys of areas to be walked-down are still applicable.
3. Remain aware of your surroundings and other work activities being conducted nearby.
4. Observe plant requirements concerning proper clothing and safety equipment.
5. Never climb on conduits or instrument tubing.

5. WALKDOWN REQUIREMENTS :

The walkdown boundary, or ZOI for each piece of equipment identified must be walked-down to determine the actual equipment located in the ZOI space. This will involve identifying all equipment which involves electrical power or controls, which is located within the ZOI space. While certain equipment could be excluded from this information gathering effort, in the interest of assuring the adequacy and completness of the data collected, all field installed equipment will be recorded. This will leave decisions concerning the use of this data with the evaluation effort, and not with the walkdown team. Also, to assure that the walkdown data adequately reflects the plant equipment, site engineering personnel must perform the first and second party data collection. This will assure that the data collected is of adequate quality and has been collected by individuals familiar with electrical equipment. This will also provide continuity between the data collection and data use.

6. **WALKDOWN TEAM/TEAM MEMBERS REQUIREMENTS :**

6.1 Team Leader : When plant support has been identified by the plant walkthrough, the requirements of reference 10.3 must be observed. The walkdown team leader is responsible for coordinating and scheduling this identified plant support.

6.2 Each walkdown team must as a minimum involve at least two individuals. Preferable, one individual from the electrical discipline and one from the mech/nuclear discipline, although the use of two individuals from the electrical discipline is acceptable. The walkdown team must be capable of providing two party data collection (second party verification) for data collected. Thus, if the walkdown team is comprised of only two individuals, these individuals must be capable of providing this function. Second party verification must be independently obtained (each individual must observe the field data collected).

6.3 Each walkdown team member must be knowledgable of this instruction and have a reading knowledge of SSP-7.2 (Outage Management), SSP-12.50 (Unit Separation for Recovery Activities), and 0-TI-238 (Systems, Components or Equipment Walkdown). Compliance with this requirement will be demonstrated by maintaining a training log which reflects training for SSP-7.2, SSP-12.50, and 0-TI-238. Once this walkdown effort is completed, the training log will be intered into RIMS with a cross-reference to this walkdown instruction.

6.4 Each walkdown team member must have at least 4 years of on the job experience relating to various design activities involving similar equipment to that which will be encountered. Their general knowledge and experience with electrical equipment, design considerations, and their engineering background will thus be adequate to assure the integrity of this data collection effort. This is equivalent to ANSI N45.2.6 level II training, as it relates to this activity.

6.4 Each walkdown team member will be responsible for complying with plant procedures and practices as outlined in section 4 of reference 10.3 (0-TI-238).

6.5 PDDs must be initiated for any instances, identified during walkdowns, where the actual plant configuration differs from the issued as-constructed drawings. (see section 7.1.13 of reference 10.3, 0-TI-238).

7. **ACCEPTANCE REQUIREMENTS/CRITERIA :**

The objective of these walkdowns is to identify equipment which will be impacted or disabled by the postulated fire sources. Therefore, conservative determinations must be used to identify equipment within the applicable ZOIs. This can best be assured by including margins in the identification of the ZOIs, and requiring knowledgable individuals to conduct the walkdowns (thus assuring that questionable equipment is included in the data collection).

7.1 Measuring instruments :
The identified ZOIs have incorporated sufficient margins in their defined boundaries, that the use of uncaliberated, commerical grade measuring devices is acceptable. The identified distances should be considered as having only positive tolerances.

7.2 Field measurements to identify ZOI boundary limits :
The ZOI boundaries have incorporated sufficient margins to assure that measurement accuracies are not critical, if the walkdown team members have sufficient knowledge to recognize nearby equipment which should be included. Questionable equipment which is included to assure conservative data collection will be noted on the data sheets. Thus, the acceptance requirement will be to use judgement (based upon the team members knowledge and experience) to include any questionable equipment.

7.3 Walkdown Packages review and approval :
WDPs shall receive a review by the responsible supervisor and an independent review for technical adequacy, as defined in section 7.2.2 of reference 10.3 (0-TI-238), prior to being sent to RIMS. The WDP cover sheet (see attachment B of ref. 10.3) has provisions to document these reviews.

8. **REQUIRED DOCUMENTATION :**

The following requirements represent the minimum documentation requirements acceptable for this effort.

8.1 Each piece of equipment walkdown to determine the extent of involvement of its associated fire will be documentated as an individual walkdown package. These walkdown packages will, as a minimum, contain the following :

A. Cover sheet (see Attachment B, ref. 10.3).
B. List of individuals compromising the walkdown team, with the team leader being identified (see attachment A, this instruction).
C. A copy of any applicable Work Requests or labling requests initiated as a result of the walkdown (see ref. 10.3).
D. Walkdown Scope Identification/Field Verification Form (see Attachment B, this instruction).

8.2 Auditable training records must be maintained for each walkdown team member to clearly reflect their compliance with the required training necessary for participation in this effort.

8.3 A listing of WDP number assignments which provides a cross reference to this controlling instruction document must be maintained.

9. **AREAS OF RESPONSIBILITY :**

9.1. **Engineering and Materials-Mech/Nuc Group :**

A. Identification of equipment considered as the source of origin for the fire, complete with it's ZOI. This will be used for scoping a plant walkthrough which will establish the required plant support for the walkdown.
B. Personnel to conduct the walkthrough. The walkthrough will not only determine necessary plant support, but will also identify intervening combustibles and allow finalization of the ZOI field verification form. Thus, this individual(s) must be sufficiently knowledgable of the fire protection criteria and program to properly execute this responsibility.
B. Walkdown Team Leader personnel. This responsibility includes the initial coordination and planning with appropriate plant organizations (plant walkthrough) to assure that the walkdown can be performed. The actual walkdown may be performed by other individuals where this would not lead to quality deficiencies or violate other requirements.

C. Support personnel for the walkdown team. This support can be provided by the walkdown team leader, if this doesn't conflict with other requirements.

9.2. **Engineering and Materials-Electrical Group :**
A. Support personnel for the walkdown team. This support must be sufficiently familiar with electrical equipment to assure that all applicable equipment is identified.

9.3 **Document Control-Engineering Records Processing :**
A. Walkdown Instruction issue and distribution.
B. Walkdown Package Microfilming and Storage.
C. Cross reference listing of the WDP's (see Appendix C, ref. 10.3).

10. REFERENCES :

10.1 SSP-7.2, "Outage Management"
10.2 SSP-12.50, "Unit Separation for Recovery Activities".
10.3 0-TI-238, "Systems, Components or.Equipment Walkdown"
10.4 Patrick P. Carier's Licensing Information Request to R.J. McMahon, dated July 12, 1991 (R08 910712 923). This request contains both the NRC Generic Letter 88-20, Supplement 4 and NUREG -1407.

ATTACHMENTS :

A. Field Walkdown Data Cover Sheet Form.
B. Walkdown Scope Identification/Field Verification Form.

(ATTACHMENT A)

(FIELD WALKDOWN DATA COVER SHEET)

PAGE 1

WALKDOWN PACKAGE : ____________

CONTROLLING INSTRUCTION : ____________

EQUIPMENT WALKED-DOWN :

____

WALKDOWN TEAM :

TEAM LEADER : ____________________ ORG ____________
MEMBER : ____________________ ORG ____________
MEMBER : ____________________ORG ____________
MEMBER : ____________________ ORG ____________
MEMBER : ____________________ ORG ____________

WALKDOWN DATE : ____________

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE

page ___ of ___

fire area/zone ___________

Ignition Source ______________________

cable trays in fire damage zone: ______________________

conduits in fire damage zone: ______________________

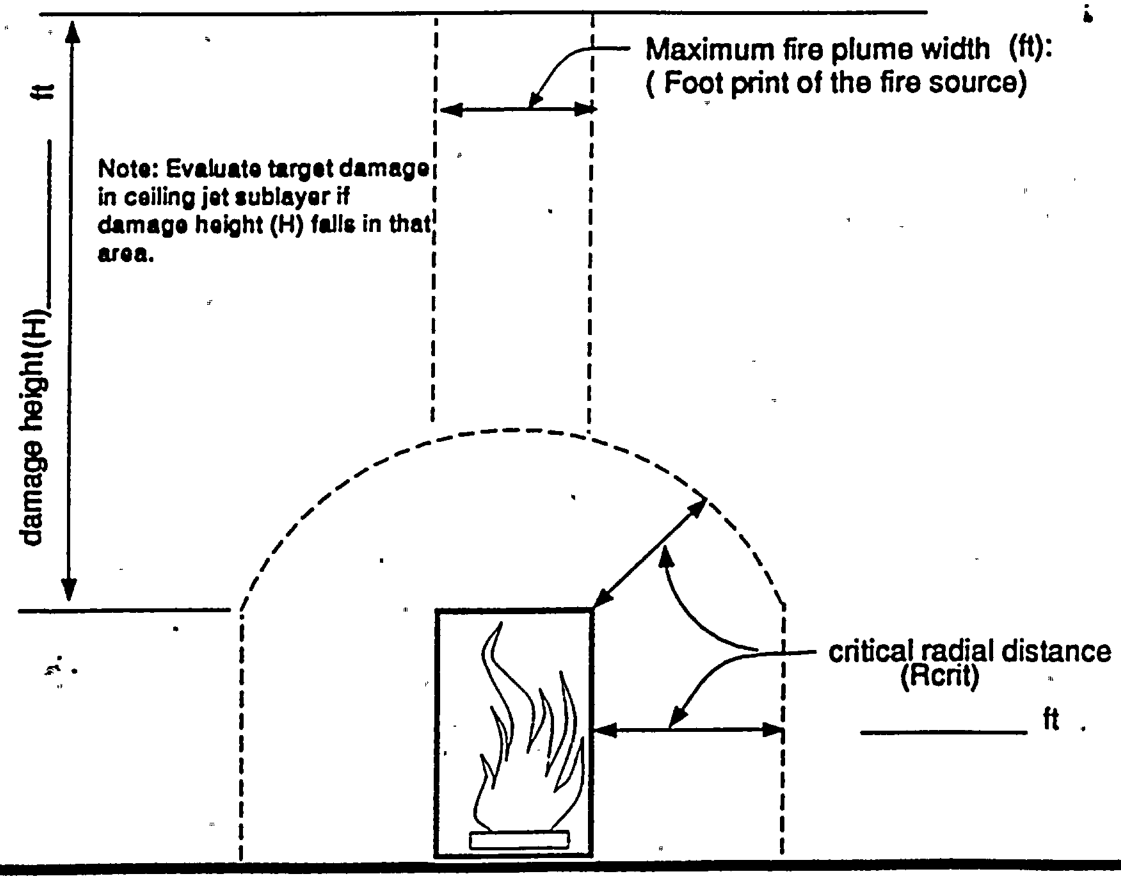
____

____

____

____

other components in fire damage zone: ______________________



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party ______________ Date ____________

Walkdown Second Party ______________ Date ____________

D.2 INDIVIDUAL COMPONENT ZONES OF INFLUENCE

QA Record

R21 941207 001

TITLE : SYSTEM, COMPONENT, OR EQUIPMENT WALKDOWNS
UNIT 0, 0-TI-238, APPENDIX B

0-TI-238

WALKDOWN PACKAGE

Package No EEB 010 - 1 Revision REV 1 R1

Title NRC GENERIC LETTER 88-20 , SUPPLEMENT 4 (NUREG-1407) , IPEEE INTERNAL FIRES ENGINEERING ANALYSIS PLANT CONFIGURATION DATA WALKDOWNS

Reference TSD (if applicable) N/A

Applicable Walkdown Instruction EEB 010

Note : Revision 1 to this Walkdown Package has revised the data for the following equipment, because of changes in the identified sphere of influence :
Panel 25-3 and 25-9; 480V RMOV BD 2E; SLC Pumps A & B; U-2 Unit Preferred Transformer; 240V Ltg Bds 2A & 2B; and 4KV-480V Transformers TS2A & TS2B.
[Pages: 13, 20, 23, 25, 31, 32, 33, & 41] R1

Prepared By : William L Aldredge

Approvals / Review :

Kim Mundy Phone 7695 Date 12-7-94
Responsible SE Supervisor

RAH Wright Phone 7479 Date 12-6-94
independent Reviewer

cc : Shift Operations Supervisor
Plant Manager
Site Quality Surveillance

WALKDOWN DATA COLLECTION CLARIFICATION

1. Cable tray or conduit numbers for equipment whose involvement in the critical distance, or damage area, is dedicated to the ignition source, will not be recorded because the end device (ignition source) is already considered involved and disabled.
2. When conduit numbers are difficult to obtain as a result of the physical installation, but the conduits are clearly recognizable as circuitry identifiable by its dedication to a particular junction box (JB) or panel, the JB or panel may be recorded and all circuitry involved with the JB or panel considered to be failed. Similarly, if the circuitry is clearly for lighting, fire detectors, power receptacles, paging system, or some other non-critical circuitry, simply identifying the application when the conduit tagging is not readily identifiable will be acceptable.
3. When a conduit number is questionable, or not readily available during the field walkdown, available data will be collected. Thus, rather than a conduit number the data collected may be an associated penetration number. This data will allow the conduit number to be determined by conducting a cross-reference check of the penetration number. Performing this cross-reference to determine the conduit number will not be a part of the walkdown effort.
4. Miscellaneous piping, gas cylinder storage, and other non-electrical equipment may be tabulated by the walkdown team, if the information is considered to have potential usefulness in responding to the IPEEE issue. Since there is no requirement for this information, its inclusion is strictly voluntary and may or may not be used in the evaluation based upon the walkdowns data.

JUSTIFICATION :

The data collected by these walkdowns is being used to evaluate the potential consequences of a fire bounded by the identified ignition source and its resulting critical distance, or damage area. The fire consequence is not considered to be of concern, if it does not result in an automatic unit trip. Thus, many design features can be quickly dispositioned as not creating an automatic unit trip simply by the potential impact of the plant design. Some of these features are : plant lighting and associated lighting panel circuitry; paging; fire detection; sound powered phone circuitry; and localized alarm circuitry.

Also, many local panels have a limited number of conduits associated with them and are dedicated to specific functions or the operation of a specific piece of equipment. Thus, the plant response resulting from disabling the panel is the same as its response to a loss of the specific equipment or function. Examples of such panels are local MG set control panels, 480V Shutdown Logic panels, and various other dedicated function panels. Therefore, identifying these panels is sufficient to allow the potential involvement and impact to be determined without the individual conduit numbers. Based upon the discussion provided, the justification for the walkdown effort clarification and the data collected has been demonstrated and the integrity of the walkdown data and its use will not be compromised.

TRAINING LOG FOR WALKDOWN INSTRUCTION EEB 010

INDIVIDUAL	SSP-7.2	SSP-12.50	0-TI-238	WI EEB010	YEARS OF EXPER.
1 WILLIAM L. ALDREDGE William L Aldredge	WLA 7-6-94	WLA 7-6-94	WLA 7-6-94	WLA 7-6-94	20 YRS NUCLEAR DESIGN
2 TURNER J. HOWARD Turner J. Howard	TJH 7-6-94	TJH 7-6-94	TJH 7-6-94	TJH 7-6-94	18 YRS NUC DESIGN 2 YR NUC CONST.
3 John A ELMERICK John A Elmerick	JAE 7-6-94	JAE 7-6-94	JAE 7-6-94	JAE 7-6-94	8 yr. Nuclear DESIGN
4 RASHID ABBAS Rashid Abbas	RA 7-6-94	RA 7-6-94	RA 7-6-94	RA 7-6-94	20 YRS Nuclear DES.
5 PERRY ZIMMERMAN Perry Zimmerman	PZ 7-27-94	PZ 7-26-94	PZ 7-26-94	PZ 7-26-94	9 YRS NUCLEAR DES.
6 DAVID FINCHER David Fincher	7-27-94	7-27-94	7-27-94	7-26-94	18 YRS NUCLEAR DSG.
7					
8					
9					
10					

WALKDOWN PACKAGE : EEB -010 - 1

CONTROLLING INSTRUCTION : WI EEB 010

EQUIPMENT WALKED DOWN :

1. 4KV-480V TRANSFORMER TS2A
2. 4KV-480V TRANSFORMER TS2B
3. 4KV-480V TRANSFORMER TS2E
4. 4KV RPT BD 2-I
5. 4KV RPT BD 2-II
6. 480V RB VENT BD 2B
7. 480V RMOV BD 2C
8. 480V RMOV BD 2D
9. 480V RMOV BD 2E
10. 250V RMOV BD 2C
11. 240V LTG TRANSFORMER TL2A
12. 240V LTG BD 2A
13. 240V LTG BD 2B
14. DRYWELL TORUS COMPR.
15. U2- UNIT PREF TRANSFORMER
16. MG SET 2DN
17. MG SET 2DA
18. MG SET 2EN
19. MG SET 2EA
20. PANEL 25-3
21. PANEL 25-9
22. LPNL-25-23
23. LPNL-25-24
24. 2-PNLA-25-31
25. 2-PNLA-25-340
26. 2-PNLA-25-341
27. RBCCW PUMP 2A
28. RBCCW PUMP 2B
29. RWCU PUMP RM MONITOR
30. SDBR A/C UNIT COMPR. MOTOR 2C
31. SDBR A/C UNIT COMPR. MOTOR 2D
32. SLC PUMP A
33. SLC PUMP B
34. 480V SD BD RM 1A
35. 480V SD BD RM 1B
36. 480V SD BD RM 2A
37. 480V SD BD RM 2B
38. 480V SD BD RM 3A
39. 480V SD BD RM 3B
40. 4KV SD BD RM A
41. 4KV SD BD RM A- BATT. RMs
42. 4KV SD BD RM B
43. 4KV SD BD RM C
44. 4KV SD BD RM C- BATT. RMs
45. 4KV SD BD RM D
46. 4KV SD BD RM 3EA
47. 4KV SD BD RM 3EB
48. 4KV SD BD RM 3EC
49. 4KV SD BD RM 3ED
50. 4KV BUSS TIE BD
51. SD BD RM E
52. SD BD RM F
53. U1- BATTERY RM [250V]
54. U1- BATTERY BD RM [250V]
55. U2- BATTERY RM [250V]
56. U2- BATTERY BD RM [250V]
57. U3- BATTERY RM [250V]
58. U3- BATTERY BD RM [250V]
59. RECIRC PUMP MG-SET 2A
60. RECIRC PUMP MG-SET 2B

WALKDOWN TEAM :

TEAM LEADER : RASHID ABBAS ORG E&M M/N
MEMBER : WILLIAM L ALDREDGE ORG E&M EE
MEMBER : TURNER J HOWARD ORG E&M EE
MEMBER : JOHN A ELMERICK ORG E&M M/N
MEMBER : PERRY ZIMMERMAN ORG E&M EE [c]
MEMBER : DAVID E FINCHER ORG E&M EE [c]

[c]... Contract empolyee

WALKDOWN DATEs : JULY thru SEPT 1994

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION P3 9-26-94

FIRE DAMAGE ZONE OF INFLUENCE

fire area/zone 2-1

EEB-010-1 PG 6 OF 88

Ignition Source 250 V RMOV BD. 2C

*1 .... CONDUITS AND CABLES ASSOCIATED WITH THE BOARD.

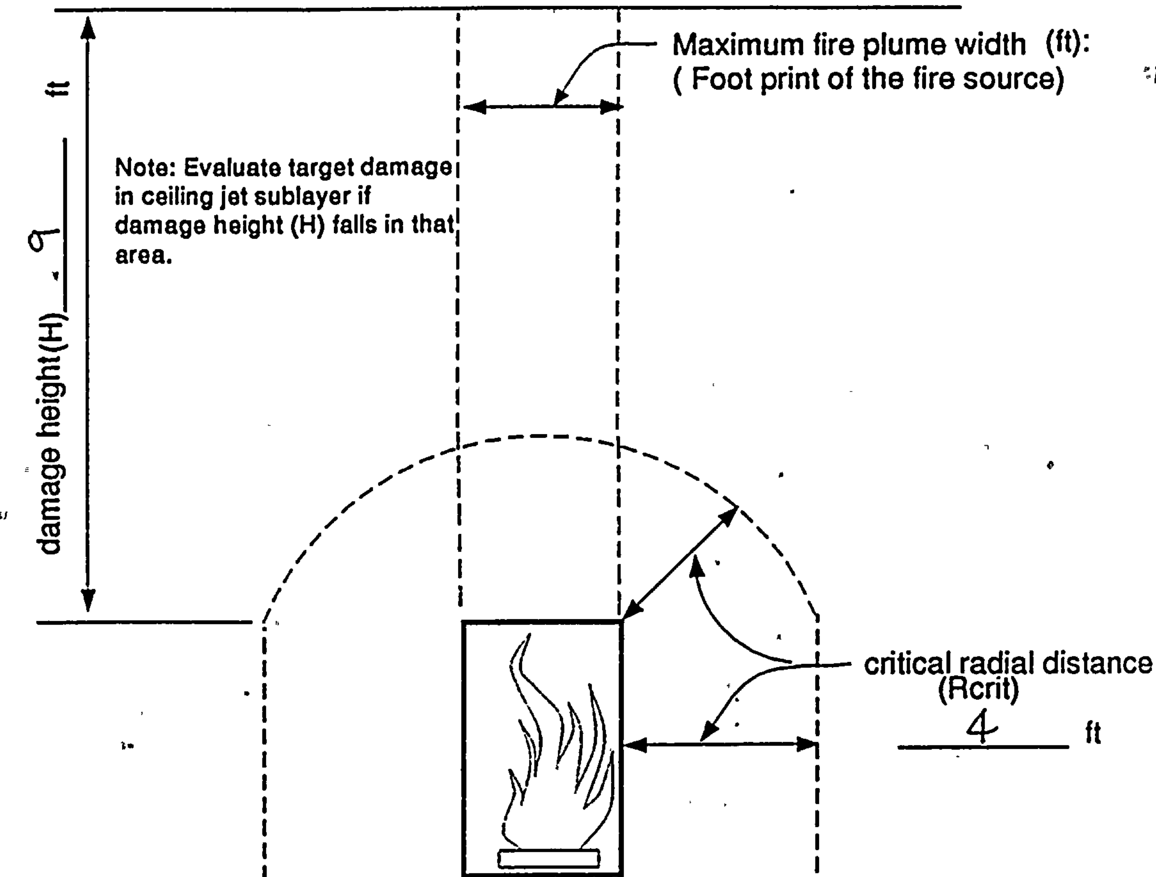
cable trays in fire damage zone: NONE

conduits in fire damage zone: NONE, *1

OTHER EQUIPMENT IN FIRE DAMAGE ZONE:

UTILITY OUTLET WITH CONDUIT

other components in fire damage zone: SEE ABOVE



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party Turner J Howard JKE Date 7-7-94 7-11-94

Walkdown Second Party William L Aldredge Date 7-7-94

kktb0/ignition frequency (five)/walkdown.source loc1

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE

fire area/zone 2-1

Ignition Source 2-PNLA-25-340

*1.... LOCATED JUST OUTSIDE CRITICAL RADIAL DISTANCE.

cable trays in fire damage zone: NONE

conduits in fire damage zone: 2V1431*1, 2V1430*1, *2

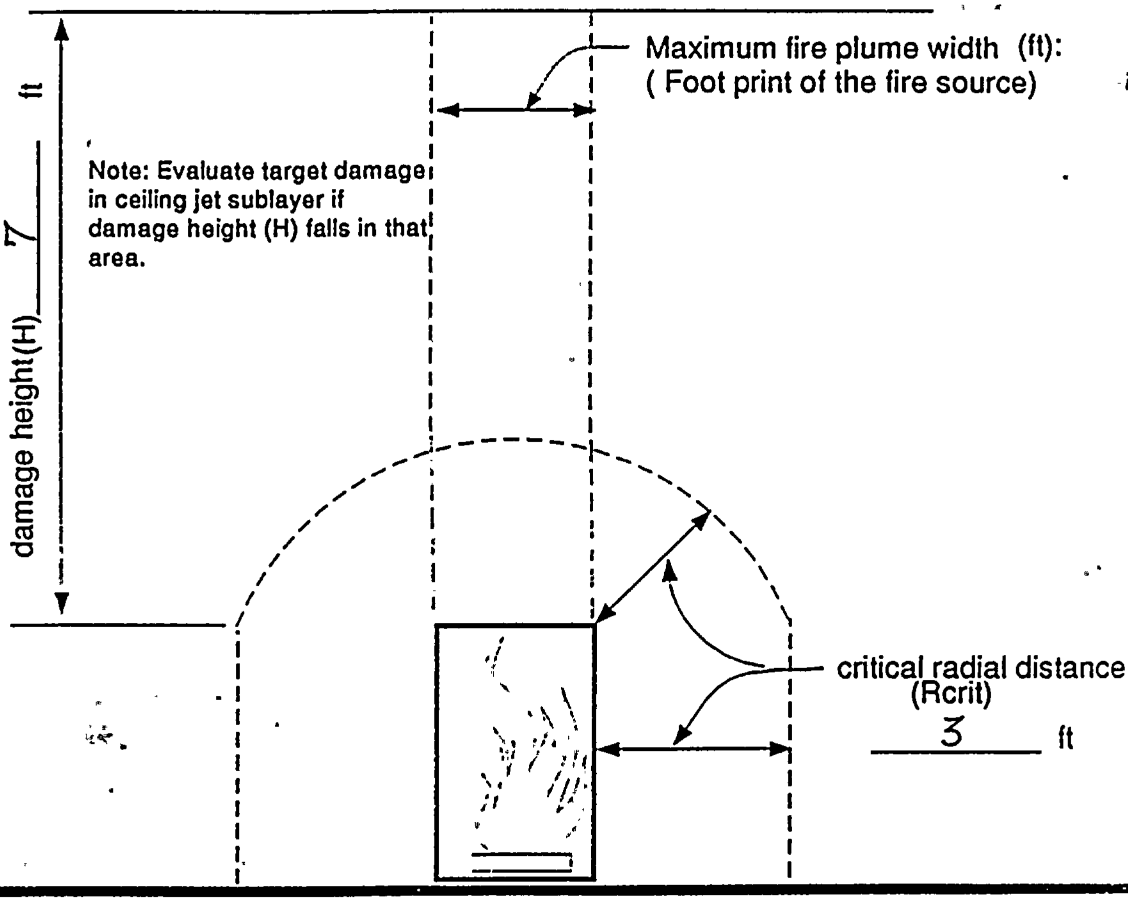
OTHER EQUIPMENT IN THE FIRE DAMAGE ZONE:

LC201, JB3996 (CONDUITS 2V1430 & 2V1431)*1, AND

TEST*1, H_2*1, & O_2*1 TANKS

*2.... ALL OTHER CONDUITS ASSOCIATED WITH PANEL 25-340.

other components in fire damage zone: SEE ABOVE



Use additional sheets if necessary to identify components in the zone of influence

JAE 7-11-94

Walkdown First Party William L Aldredge Date 7-7-94

Walkdown Second Party Turner J. Havard Date 7-7-94

kktb0/ignition frequency (five)/walkdown.source loc1

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE

fire area/zone 2-1 EEB-010-1 PG 8 OF 88

Ignition Source DRYWELL TORUS COMPRESSOR

cable trays in fire damage zone: NONE

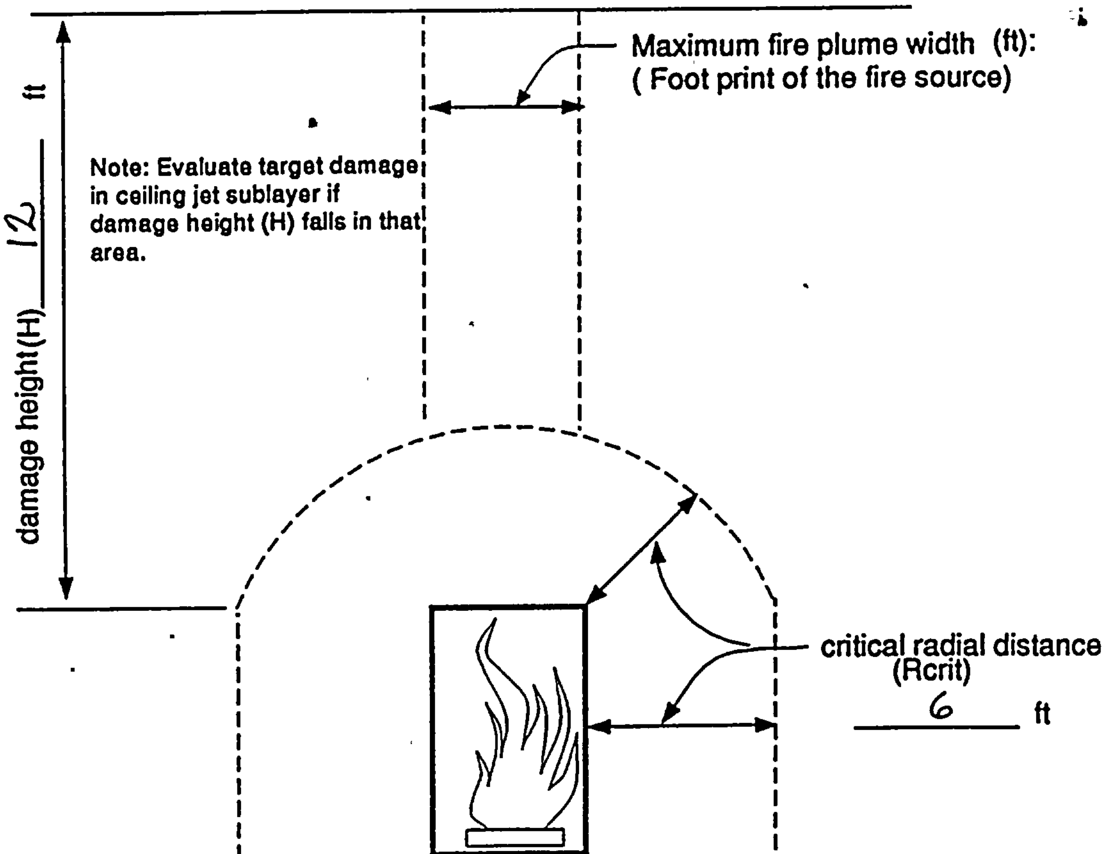
conduits in fire damage zone:

- ES50-I *1, 2C1447 *1, 2ES1356 *1

*1 .... LOCATED ON PERIPHERY OF DAMAGE ZONE

- DRYWELL TORUS COMPRESSOR

other components in fire damage zone: NONE



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party William L Aldridge Date 7-7-94

Walkdown Second Party Turner J. Howard Date 7-7-94

kktb0/ignition frequency (five)/walkdown.source loc1

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE

fire area/zone 2-2 EEB-010-1 PG 9 OF 88

Ignition Source 480V RMOV BD. 2C

*1 .... CONDUITS AND CABLES ASSOCIATED WITH THE BOARD.

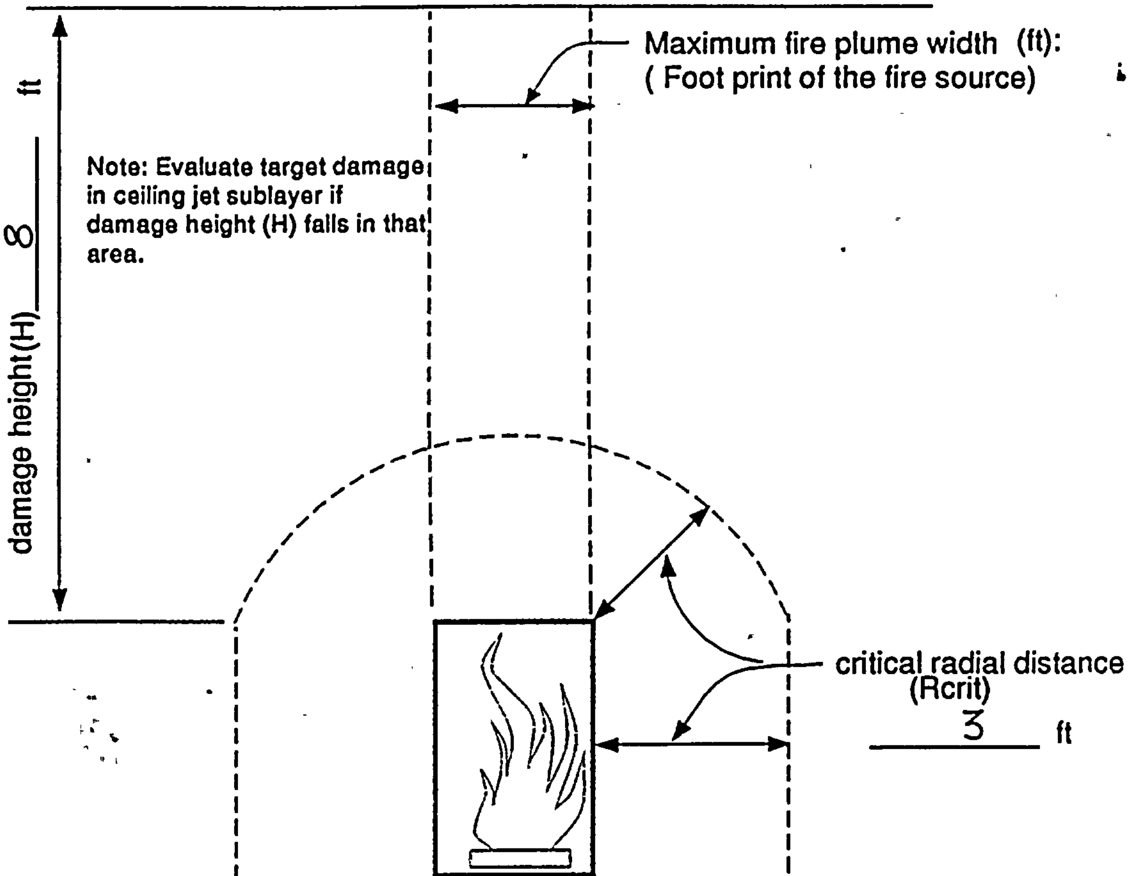
cable trays in fire damage zone: NONE

conduits in fire damage zone: 2K904, 2R1647, 2R885, *1

OTHER EQUIPMENT IN THE FIRE DAMAGE ZONE:

EMERGENCY LIGHT 101, 2-JB-244-9225, SOUND POWERED PHONE JACK (2 EA.), NITROGEN TANK (2 EA.)

other components in fire damage zone: SEE ABOVE



Use additional sheets if necessary to identify components in the zone of influence

JAE 7-11-94

Walkdown First Party William L Aldredge Date 7-7-94

Walkdown Second Party Turner J. Howard Date 7-7-94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE

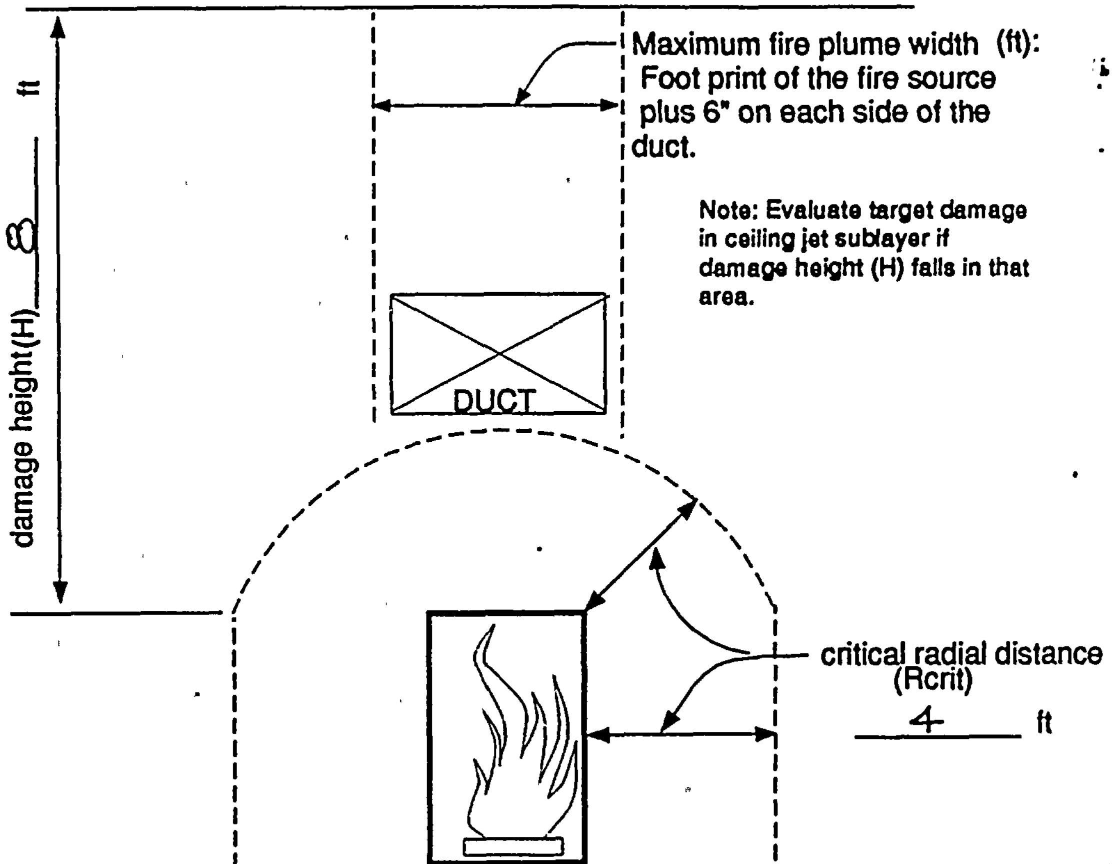
fire area/zone 2-2

Ignition Source 480V RB VENT BD 2B

cable trays in fire damage zone: NONE

conduits in fire damage zone: 2RM56, 2ES3918-II, K442-I (OTHER CONDUIT IN THE AREA COULD NOT BE IDENTIFIED DUE TO INACCESABILITY OR NOT BEING TAGGED).

other components in fire damage zone: SEE PAGE 2



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party [signature] Date 9-22-94

Walkdown Second Party [signature] Date 9-22-94

kktb0/ignition frequency (five)/walkdown.source loc1

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION
(CONTINUATION)

1. IGNITION SOURCE : 480V RB VENT BD 2B

2. CABLE TRAYS IN FIRE ZONE : (CONTINUED)

N/A

3. CONDUITS IN FIRE DAMAGED ZONE : (CONTINUED)

(- SEE PAGE 1 -)

4. OTHER COMPONENTS IN THE FIRE DAMAGE ZONE : (CONTINUED)

PNL-25-338, 2-RE-90-134A, 2-PNL-25-338A, 2-PI-84-50, 2-RFV-84-615, 2-PREG-84-54, 2-SHV-084-0613, 2-SHV-084-0614, 2-SHV-084-0050, 2-RTV-084-0616, 2-AMP-244-6115, FIRE ALARM, OVERHEAD LIGHTS, PNL-25-336*, 2-RE-90-132A*, 2-PNL-25-336A*.

* INDICATES COMPONENTS ARE SLIGHTLY BEYOND THE FIRE ZONE, HOWEVER THEY ARE LOCATED WITHIN 6"-12" OF PANEL-25-338.

WALKDOWN FIRST PARTY Terry Zimmerman DATE 9-22-94

WALKDOWN SECOND PARTY William L Aldridge DATE 9-22-94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

P3 9-26-94

FIRE DAMAGE ZONE OF INFLUENCE

fire area/zone 2-2

Ignition Source 2-PNLA-25-341

*1 .... CONDUIT AND CABLE ASSOCIATED WITH 2-PNLA-25-341

cable trays in fire damage zone: NONE

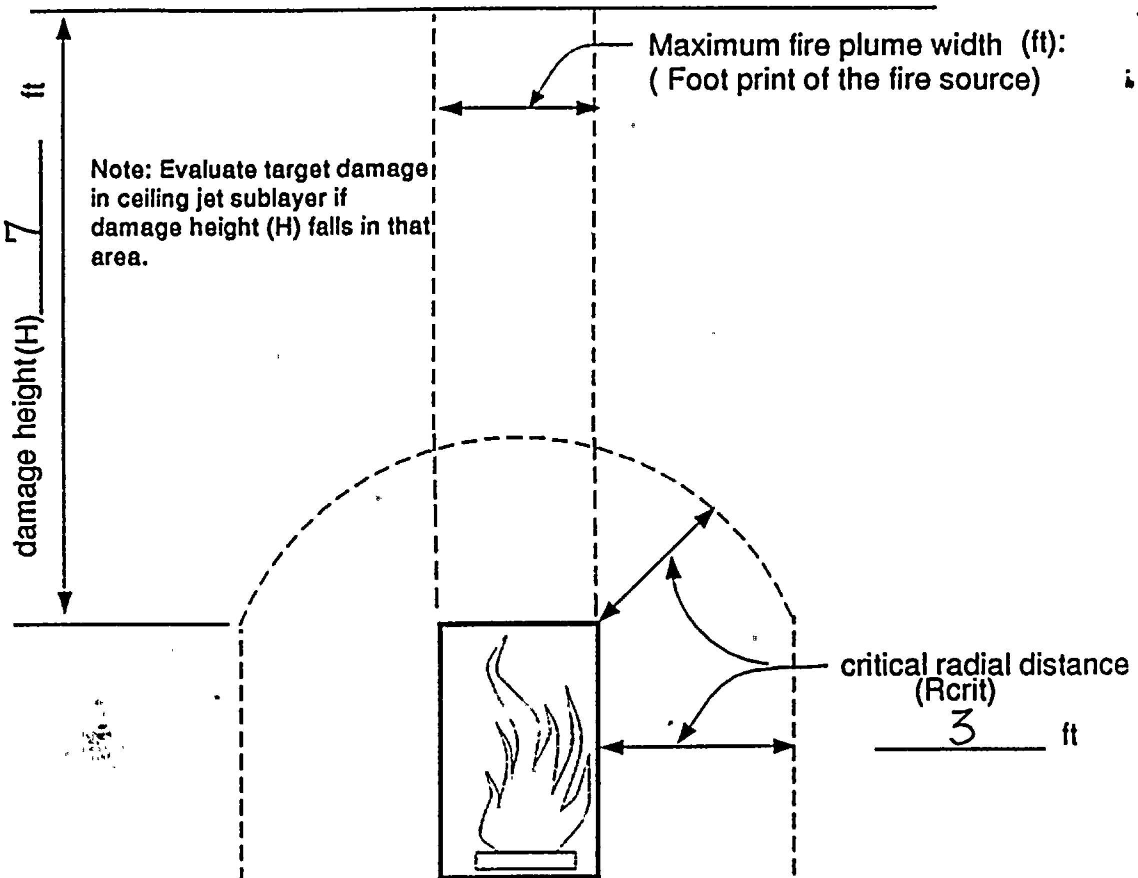
conduits in fire damage zone: *1

OTHER EQUIPMENT IN THE FIRE DAMAGE ZONE:

JB6156, TEST GAS TANK, H_2 TANK AND

O_2 TANK.

other components in fire damage zone: SEE ABOVE



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party William L Aldredge Date ~~7-7-94~~ 7-11-94 JAE

Walkdown Second Party Turner J. Howard Date 7-7-94

kktb0/ignition frequency (five)/walkdown.source loc1

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE

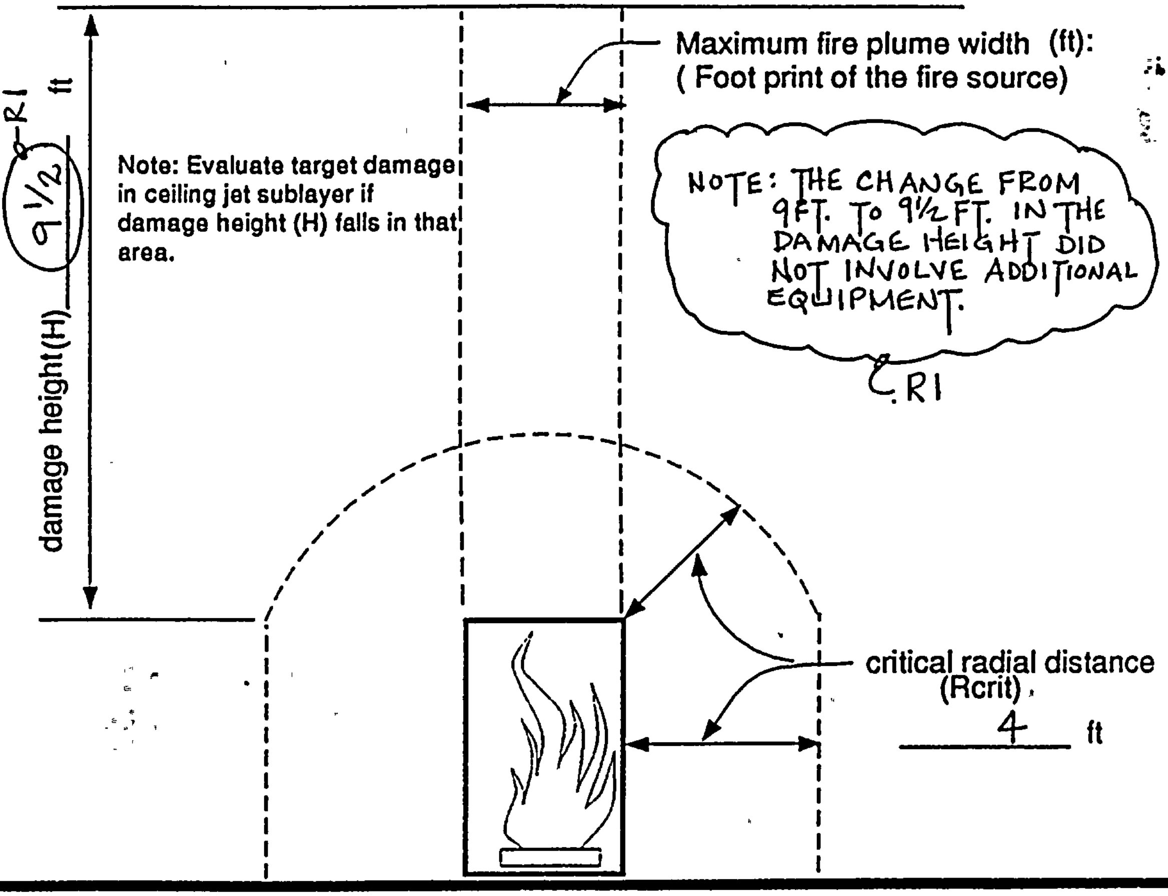
fire area/zone 2-3

Ignition Source U2 - UNIT PREFERRED TRANSFORMER

cable trays in fire damage zone: N/A

conduits in fire damage zone: 2R2338, 2R2340, 2R2337, 2V1438, 2V1437, 2ES3405-II, 2ES4609-II, 2ES3443, 2ES3442, ES5656-IIC, 2ES5726-II, ES5657-IID, 2PC1494-II, A1703, 2R1749, 2ES507-I, ES5640-IB, 2PC994-II, 2PC5071-II, 2PL435-B2, 2ES2109, ES5566-IID, ES5561-IIC, 3B193, 3B349, (CONT'D ON PG. 2)

other components in fire damage zone: SEE PAGE 2



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party	Perry Zimmerman	Date 9-14-94	RO	William L. Aldridge 12/1/94
Walkdown Second Party	David E. Fincher	Date 9-14-94	R	R Aker 12/1/94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION
(CONTINUATION) page 2 of 2

1. IGNITION SOURCE :
U2-UNIT PREFERRED TRANSFORMER

2. CABLE TRAYS IN FIRE ZONE : (CONTINUED)
N/A

3. CONDUITS IN FIRE DAMAGED ZONE : (CONTINUED)
2RP1824-IB, 2RP1724-IA, 2PC4950, 2PC893, 2PC892, 2ES1040-I, 2ES4085, 2RP2009-IIA, 2PC894-II, 2RP2021-IIB, 2ES2528, 2ES3021-II, 2ES3020-II, 2V1092, 2ES3019-II*, 2ES3022-II*, 2ES3023-II*, 2ES3591-II, 2RP219-IIIA, 2RP528-G4, 2RP518-G3, 2RP508-G2, 2RP498-G1, 2RP432-IIIB, 2PC124-IB, 2PC49-IA, 2PC174-IIB, 2PC99-IIA, 2RP1810-IIA* 2RP1822-IIB*, 2PC920-II*, 2PL2345*, 2PL5229, 2PL5254, 2M7, 2ES407-I, 2ES3260-II, 2ES1427-IS2, 2ES3421-II*, 2V2453*
(* INDICATES CONDUIT IS APPROX 1'-6" TO 2'-6" OUTSIDE OF FIRE ZONE)

4. OTHER COMPONENTS IN THE FIRE DAMAGE ZONE : (CONTINUED)
PNL 25-6-001, 2-PT-3-59, 2-PT-3-79, 2-FSV-075-0072, 2-HS-075-051B, 2-HS-075-053B.

WALKDOWN FIRST PARTY Barry Zimmerman DATE 9-14-94

WALKDOWN SECOND PARTY David E. Fricker DATE 9-14-94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE

fire area/zone 2-4 EEB-010-1 PG. 15 OF 88

Ignition Source 480V RMOV BD. 2D

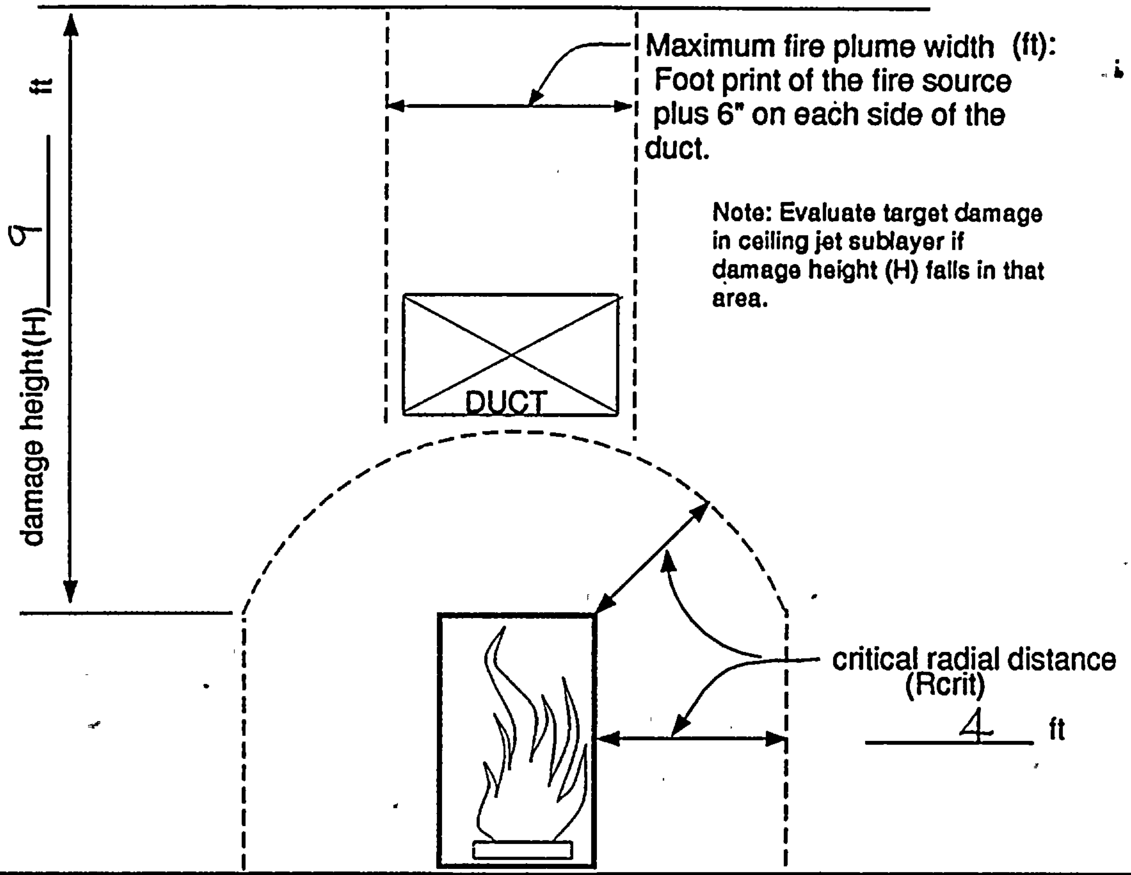
*1 .... CONDUITS & CABLES ASSOCIATED WITH THE BOARD.

cable trays in fire damage zone: NONE

conduits in fire damage zone: 2 CKTS FROM LC-203 (NO COND. NO.), *1

• SYS 26 EQUIPMENT ABANDON IN PLACE (NON ELECTRICAL)

other components in fire damage zone: SEE ABOVE



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party William L Aldredge Date 7-14-94

Walkdown Second Party Turner J. Howard Date 7-14-94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE

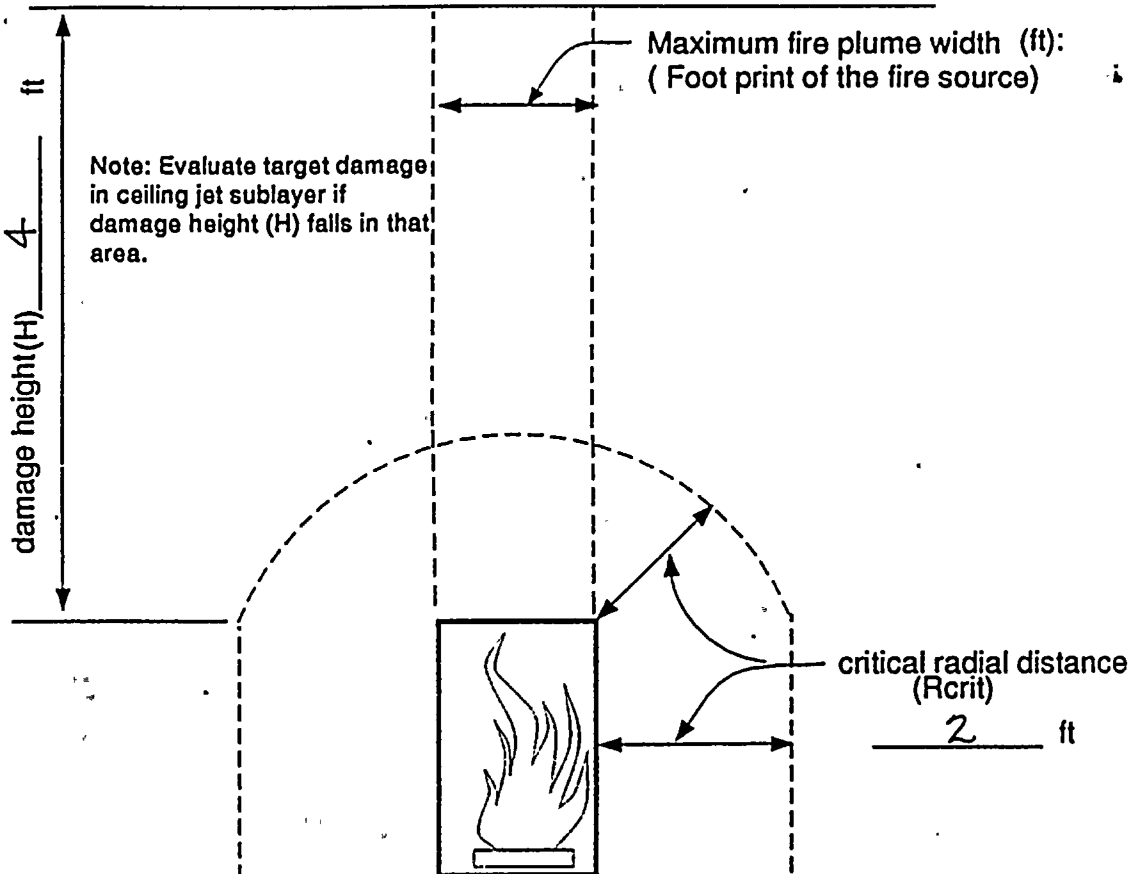
fire area/zone 2-4

Ignition Source RBCCW PUMP 2A

cable trays in fire damage zone: NONE

conduits in fire damage zone: NONE (EXCEPT THOSE ASSOCIATED WITH THE RBCCW PUMP)

other components in fire damage zone: NONE



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party William L Aldredge Date ~~7-7-94~~ 7-11-94 JAE

Walkdown Second Party Turner M Howard Date 7-7-94

kktb0/ignition frequency (five)/walkdown.source loc1

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE

fire area/zone 2-4

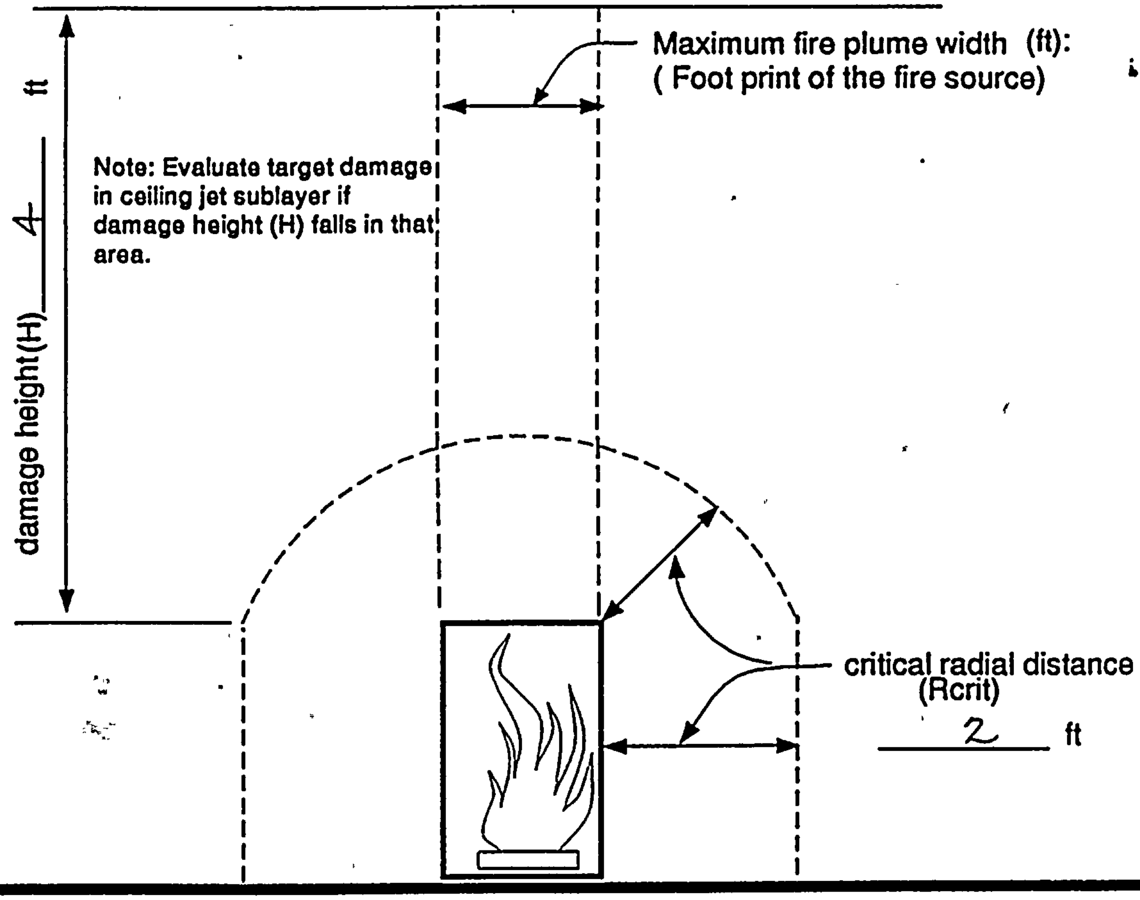
EEB-010-1 PG 17 OF 88

Ignition Source RBCCW PUMP 2B

cable trays in fire damage zone: NONE

conduits in fire damage zone: 2ES 2884-II *1, ALL OTHER CONDUITS ASSOCIATED WITH RBCCW PUMP 2B.
*1 .... LOCATED JUST OUTSIDE CRITICAL RADIAL DISTANCE

other components in fire damage zone: NONE



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party William L Aldredge JAE Date ~~7-7-94~~ 7-11-94

Walkdown Second Party Turner J. Howard Date 7-7-94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

PB 9-26-94

FIRE DAMAGE ZONE OF INFLUENCE

page ~~12~~ 1 of ~~24~~ 1

fire area/zone 2-4 EEB-010-1 PG 18 OF 88

Ignition Source RWCU PUMP RM MONITOR

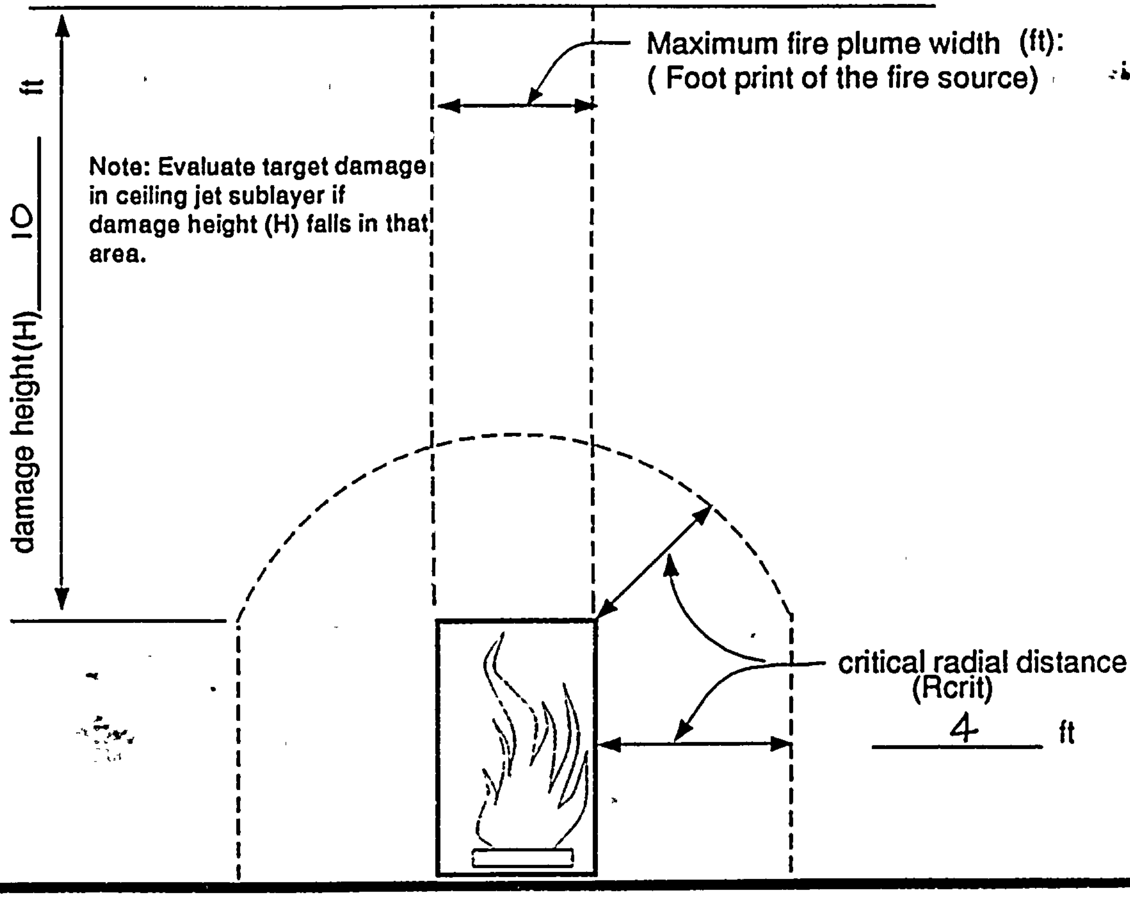
cable trays in fire damage zone: NONE

conduits in fire damage zone: NONE (EXCEPT FOR RWCU PUMP MOT.)

OTHER EQUIPMENT IN THE FIRE DAMAGE ZONE:

FIRE DETECTION CONDUITS FOR PANEL 25-290, AND PANEL 25-290, WHICH HAVE BEEN ABANDON IN PLACE

other components in fire damage zone: SEE ABOVE



Use additional sheets if necessary to identify components in the zone of influence

JAE 7-11-94

Walkdown First Party William L. Aldredge Date 7-7-94

Walkdown Second Party Turner J. Howard Date 7-7-94

kktb0/ignition frequency (five)/walkdown.source loc1

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE

page 1 of 1

fire area/zone ~~2-3~~ PB 9-26-94 / 2-4   EEB-010-1 PG 19 OF 88

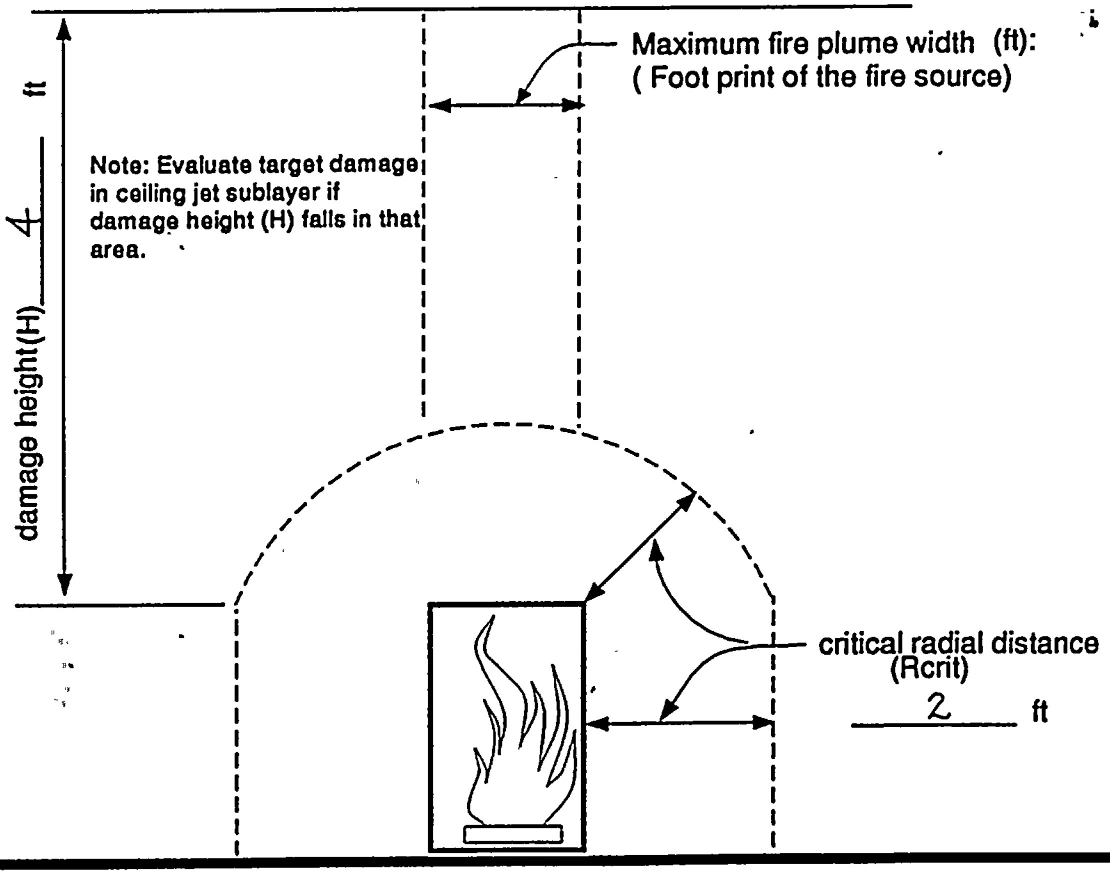
Ignition Source SDBR A/C UNIT, COMPRESSOR MOTOR 2C & 2D

cable trays in fire damage zone: SAM - ES II (≈ 5' ABOVE MOTOR 2C)

conduits in fire damage zone: 2ES624-I, 2PL5714-I, 2PL5702-I, 2PL* (*...EITHER 5741, 5704, OR 5701)

- CONDUITS ASSOCIATED WITH PENETRATIONS: R26045587, R26045586, R26045588, R26045589, R26045590, R26045591, R26045566, R26045565, R26045564, R26045563, ~~R260454~~ WRA 7/12/94 R26045562, R26045623

other components in fire damage zone: EMERGENCY LIGHT (NO. 124)



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party William L Aldredge Date 7-12-94

Walkdown Second Party Turner [illegible] Date 7-12-94

kktb0/ignition frequency (live)/walkdown.source loc1

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

B9-2694

FIRE DAMAGE ZONE OF INFLUENCE

fire area/zone 2-5

EEB-010-1 PG. 20 OF 88

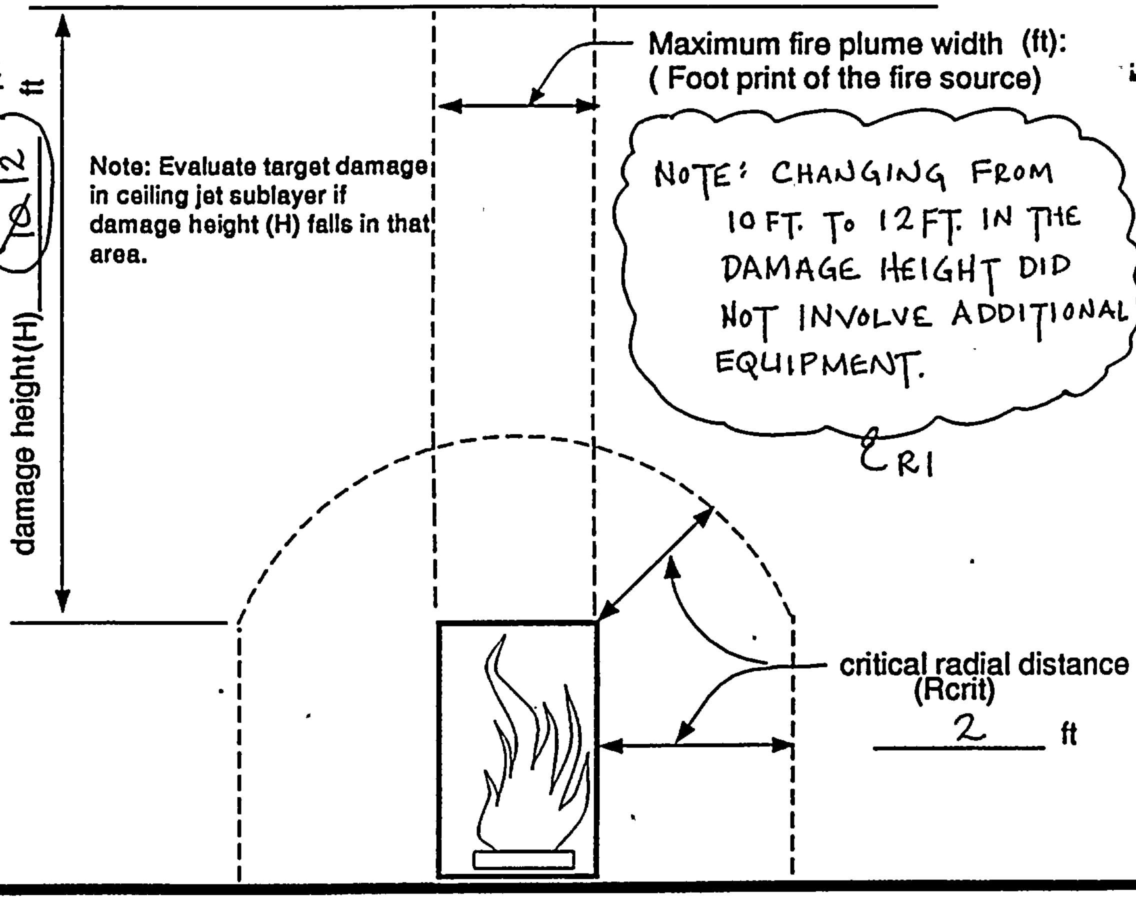
Ignition Source 480V RMOV BD. 2E

cable trays in fire damage zone: NONE

conduits in fire damage zone: 2LT40, 2LT41, 2FE451, 2K873

- FIRE PROTECTION PANELS 25-333 AND XA-26-96B2 AND THEIR ASSOCIATED CONDUITS.
- 2-JBOX-26-9622 WITH 2 CONDUITS (MAY BE OUTSIDE 10'?)
- OVERHEAD LIGHTS.

other components in fire damage zone: SEE ABOVE



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party William L Aldredge Date 7-12-94

Walkdown Second Party John A Elmore Date 7-12-94

R0

William L Aldredge 12/1/94

R Dunas 12/1/94

R1

kktb0/ignition frequency (five)/walkdown.source loc1

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

PB 9-26-94

FIRE DAMAGE ZONE OF INFLUENCE

fire area/zone 2-5

EEB-010-1 PG 21 OF 88

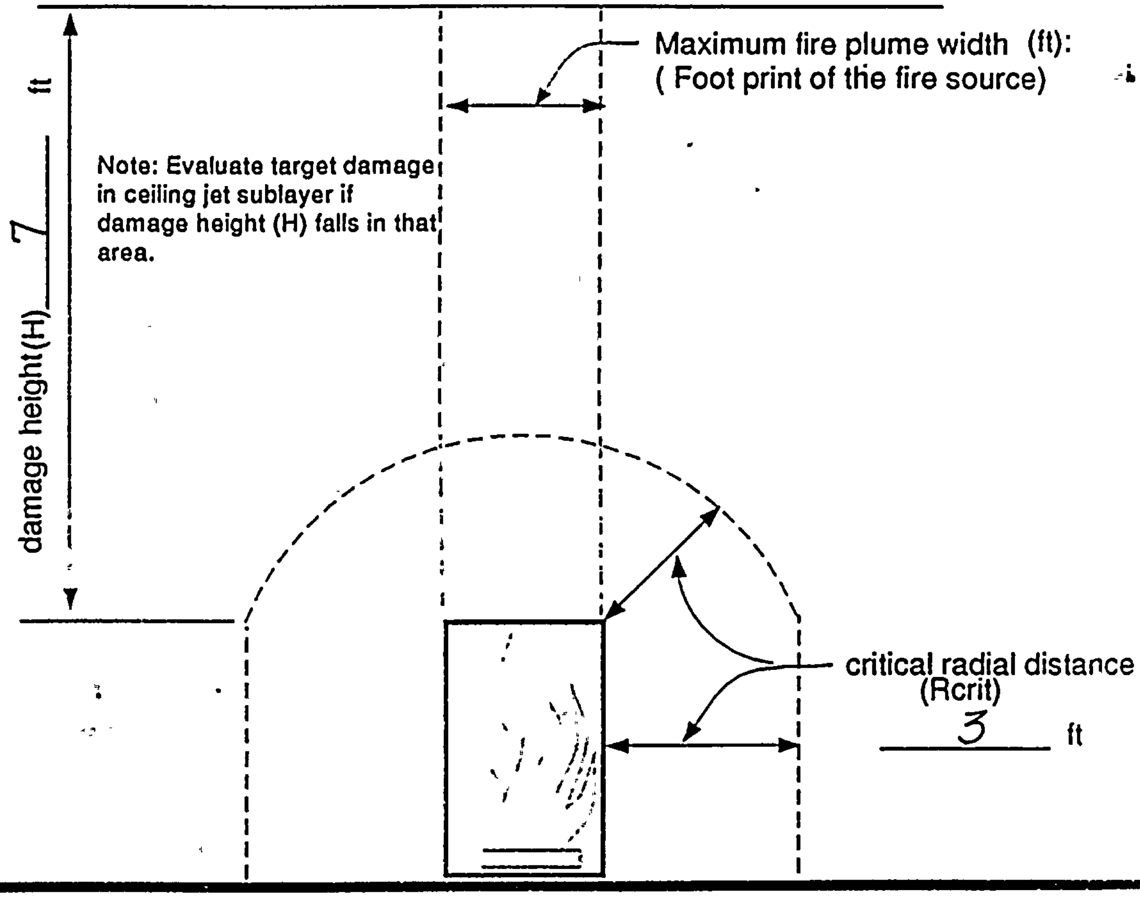
Ignition Source (LPCI) MG SET 2DN

*.... APPROXIMATELY 3FT. FROM THE FIRE IGNITION SOURCE.

cable trays in fire damage zone: NONE

conduits in fire damage zone: 2E32850-II *

other components in fire damage zone: NONE



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party William Y Aldredge Date 7-12-94

Walkdown Second Party John R Elwist Date 7-12-94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE

fire area/zone 2-5

EEB-010-1 PG 22 OF 88

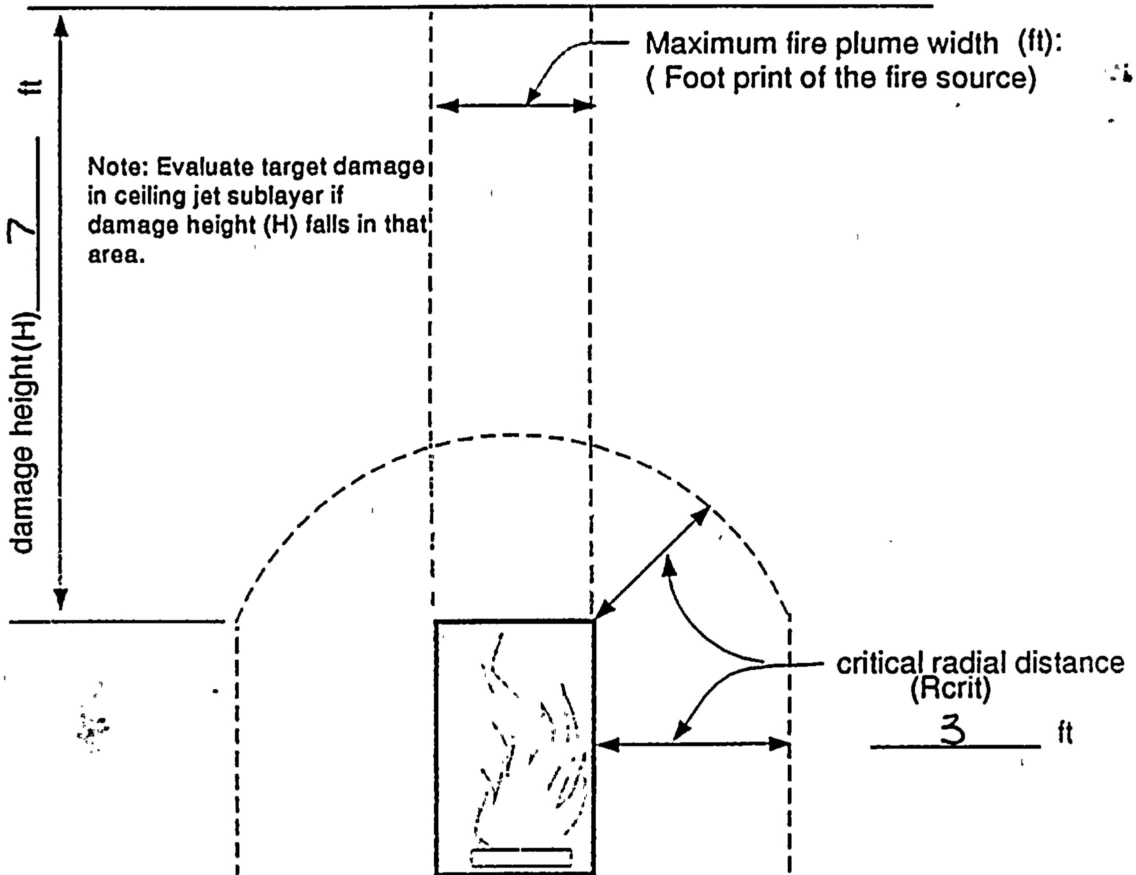
Ignition Source (LPCI) MG SET 2EA

cable trays in fire damage zone: NONE

conduits in fire damage zone:

THE CONDUITS FOR MG SET 2DN ARE IN THE CRITICAL RADIAL DISTANCE OF MG SET 2EA.

other components in fire damage zone: NONE



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party William L Aldredge Date 7-12-94

Walkdown Second Party John A Elmits Date 7-12-94

kktb0/ignition frequency (five)/walkdown.source loc1

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE

fire area/zone 2-5

Ignition Source 4KV-480V TRANSFORMER TS2A

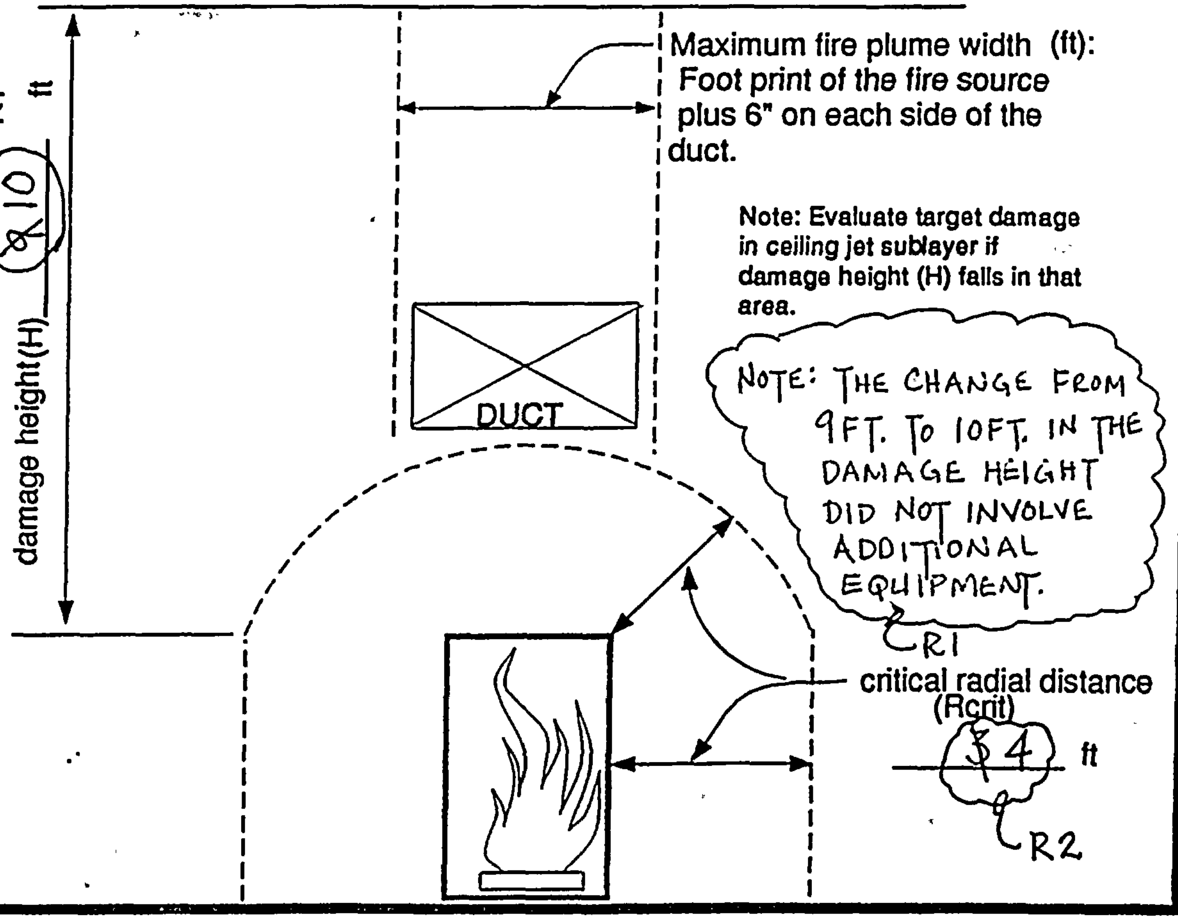
cable trays in fire damage zone: NONE

conduits in fire damage zone:

• CONDUITS 2PL460-I, 2PL1491, 2FE475, 2A3173, 2PL3873, 2ES2832-II, L217 (TS2B FAN AND ALARM CIRCUIT POWER SOURCE).

(CONTINUED ON PAGE 2)

other components in fire damage zone: SEE PAGE 2



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party William L Aldredge Date 7-13-94

Walkdown Second Party Turner H. Howard Date 7-13-94

R0

William L Aldredge 12/1/94

R Arnas 12/1/94

R1

kktb0/ignition frequency (five)/walkdown.source loc1

William L Aldredge 7/7/95

James L. Langford 7-7-95

R2

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION
(CONTINUATION) page 2 of 2

1. IGNITION SOURCE : 4KV-480V TRANSFORMER TS2A

2. CABLE TRAYS IN FIRE ZONE : (CONTINUED)

NONE

3. CONDUITS IN FIRE DAMAGED ZONE : (CONTINUED)

- UN-NUMERED CONDUIT FOR JB5951 (DISCARD THESE CONDUITS 2A3172, 2A3173, 2ES312-I, AND 2PL3857)
- CONDUITS FOR PENETRATIONS: R26215213, R26212142, R26212143, R26212144.
- ALL OTHER CONDUITS ASSOCIATED WITH TRANSFORMER TS2A.
- CONDUITS 3B390, 3B391, 3B393 |R2

4. OTHER COMPONENTS IN THE FIRE DAMAGE ZONE : (CONTINUED)

- FIRE DETECTOR CIRCUITRY.
- JB5951 (MG SET 2DN)
- OVERHEAD LIGHTS

WALKDOWN FIRST PARTY William L Aldredge DATE 7-13-94

WALKDOWN SECOND PARTY Turner J. Howard DATE 7-13-94

William L Aldredge 7/7/95
James L. Langford 7-7-95 } R2

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE

fire area/zone 2-5

Ignition Source 4KV-480V TRANSFORMER TS2B

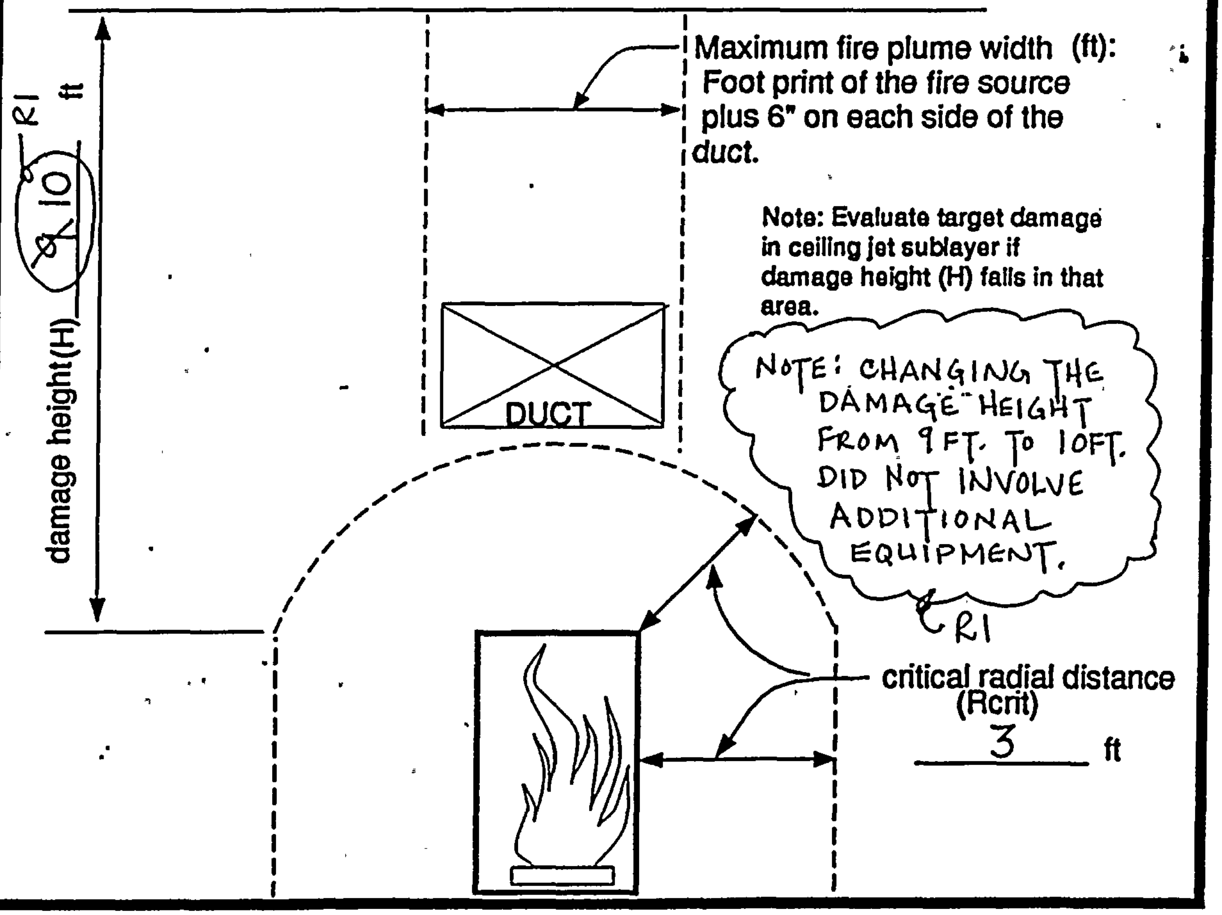
cable trays in fire damage zone: NONE

conduits in fire damage zone:

- CONDUIT 2P1491, 2FE475, 2A3173, 2PL3873, AND UN-NUMBERED CONDUIT FOR JB5951 (DISCARD CONDUITS 2A3172, 2A3173, 2ES312-I, AND 2PL3857).

(CONTINUED ON PAGE 2)

other components in fire damage zone: SEE PAGE 2



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party William L Aldredge Date 7-13-94 William L Aldredge 12/1/94

Walkdown Second Party Thurmond Howard Date 7-13-94 RO R Adkins 12/1/94

kktb0/ignition frequency (live)/walkdown.source loc1

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION
(CONTINUATION)
page 2 of 2

1. IGNITION SOURCE : 4KV-480V TRANSFORMER TS2B

2. CABLE TRAYS IN FIRE ZONE : (CONTINUED)

NONE

3. CONDUITS IN FIRE DAMAGED ZONE : (CONTINUED)

- CONDUITS FOR PENETRATIONS: R262152I3, R262I2I42, R262I2I43 AND R262I2I44.
- ALL OTHER CONDUITS ASSOCIATED WITH TRANSFORMER TS2B.

4. OTHER COMPONENTS IN THE FIRE DAMAGE ZONE : (CONTINUED)

- FIRE DETECTOR CIRCUITRY.
- JB 5951 (MG SET 2DN).

WALKDOWN FIRST PARTY William L Aldredge DATE 7-13-94

WALKDOWN SECOND PARTY Turner J. Howard DATE 7-13-94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE

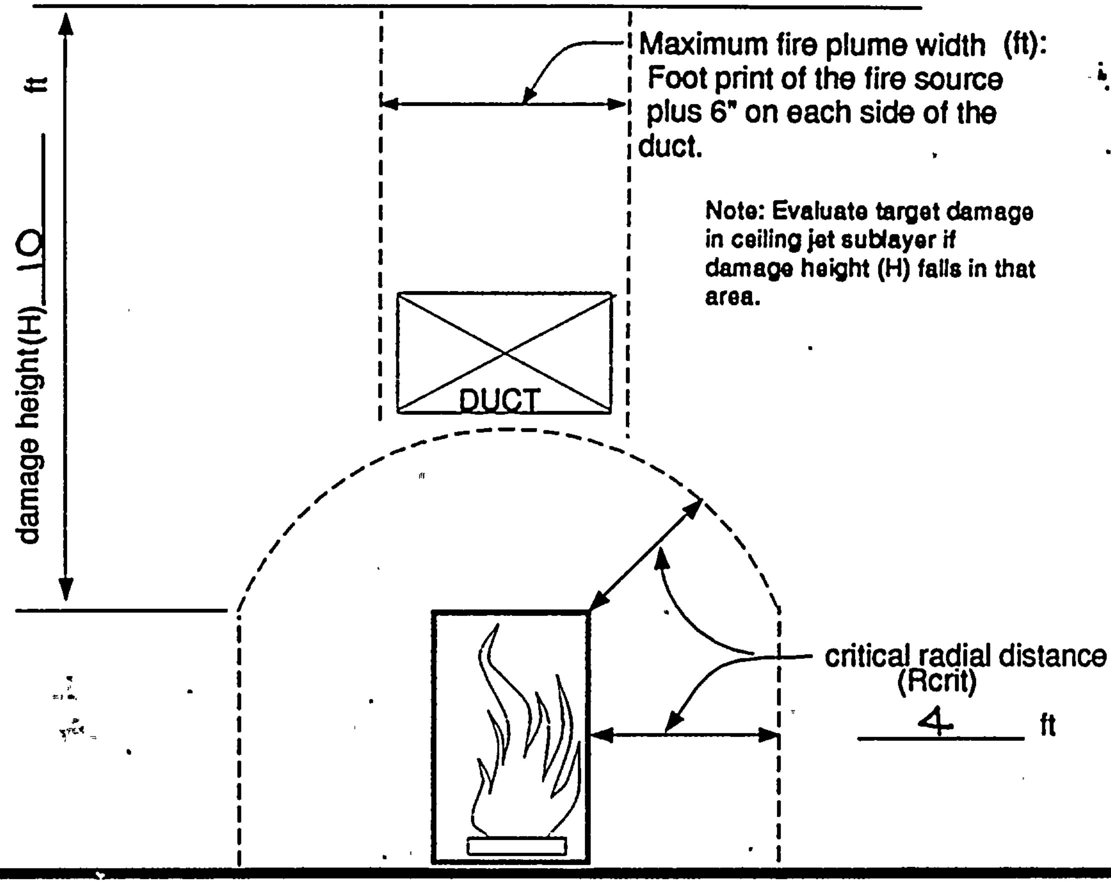
fire area/zone 2-5

Ignition Source 2-LPNL-25-31 (RCIC SYS AUX PNL)

cable trays in fire damage zone: CL, GC (CONT'D ON PAGE 2)

conduits in fire damage zone: 2AT15, 2AT3, 2A2022, 2A2017,
2ES2774-II, 2R1471, PP633, 2ES3906-IIA, C167,

other components in fire damage zone: SEE PAGE 2



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party Perry Zimmerman Date 9-20-94

Walkdown Second Party David E. Fincher Date 9-20-94

kktb0/ignition frequency (five)/walkdown.source loc1

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION
(CONTINUATION) page 2 of 2

1. IGNITION SOURCE :
2-LPNL-25-31 (RCIC SYS AUX PANEL)

2. CABLE TRAYS IN FIRE ZONE : (CONTINUED)

BW, NO, MN (THESE THREE TRAYS ARE APPROX. 1FT. BEYOND THE FIRE ZONE).

3. CONDUITS IN FIRE DAMAGED ZONE : (CONTINUED)

– SEE PAGE 1 –

4. OTHER COMPONENTS IN THE FIRE DAMAGE ZONE : (CONTINUED)
2-PNLA-071-25-425, 240V LTG. BD 2A VOLTAGE REGULATOR, 240V LTG. BD 2A XFRMR-TL2A*, 240V LTG. BD 2A*, SOUND POWERED PHONE JACK, OVERHEAD LIGHTS

*- COMPONENTS ARE LOCATED SLIGHTLY BEYOND THE FIRE ZONE

WALKDOWN FIRST PARTY Perry Zimmerman DATE 9-20-94

WALKDOWN SECOND PARTY David E. [illegible] DATE 9-20-94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE

PB 9-26-94

fire area/zone 2-5

EEB-010-1 PG 29 OF 88

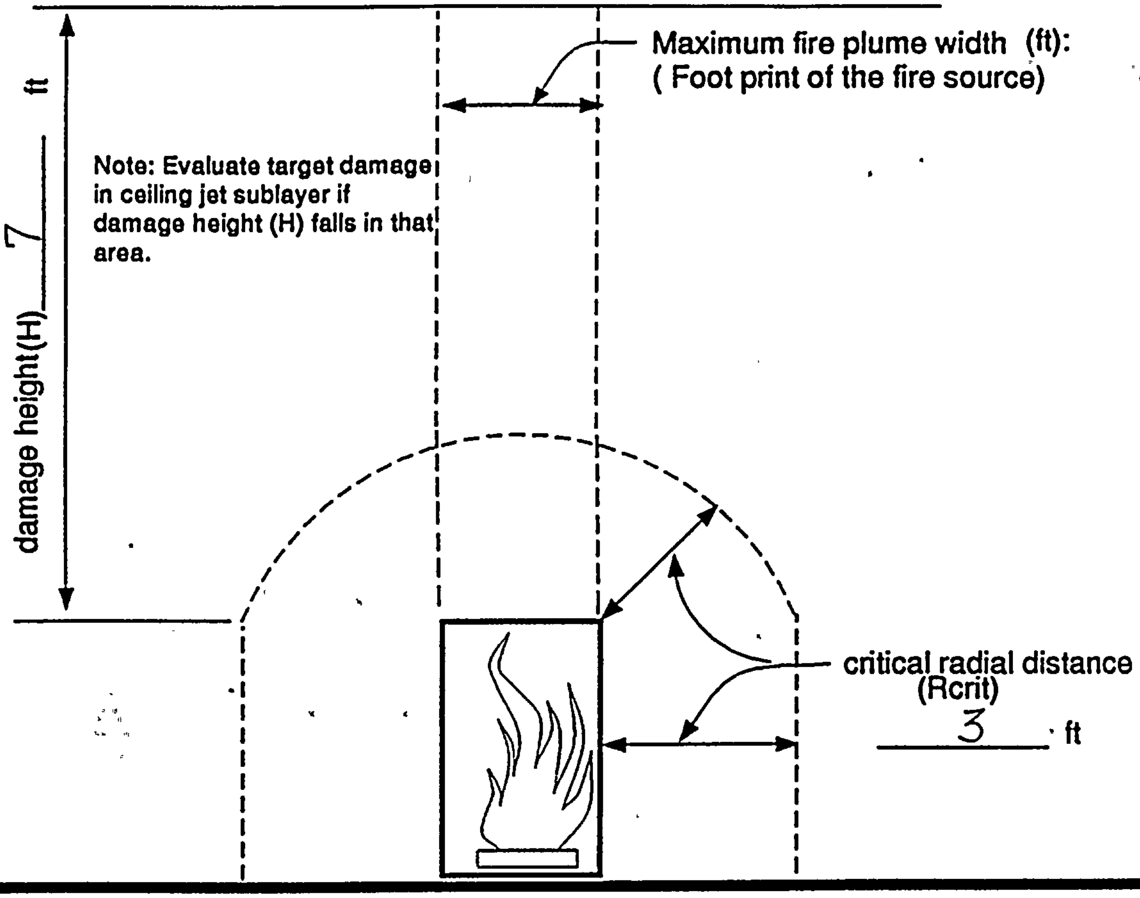
Ignition Source 4KV RPT BD 2-I

cable trays in fire damage zone: ALL ARE FOR THE RPT BD.

conduits in fire damage zone: ALL ARE FOR THE RPT BD. EXCEPT FOR 2ATR FOR PANEL 2-PNLA-68-25/418 AND THE EMERGENCY FEEDER TO 480V R MOV BD. 2E (FROM MG SET 2EA).

- PANEL 2-PNLA-68-25/418
- 2-8" MG OIL DRAIN LINES (2-77-617 & 3-77-617 AND THEIR CHK VALVES)

other components in fire damage zone: SEE ABOVE



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party [signature] Date 7-12-94

Walkdown Second Party [signature] Date 7-12-94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION
FIRE DAMAGE ZONE OF INFLUENCE

fire area/zone 2-5   EEB-010-1 PG 30 OF 88

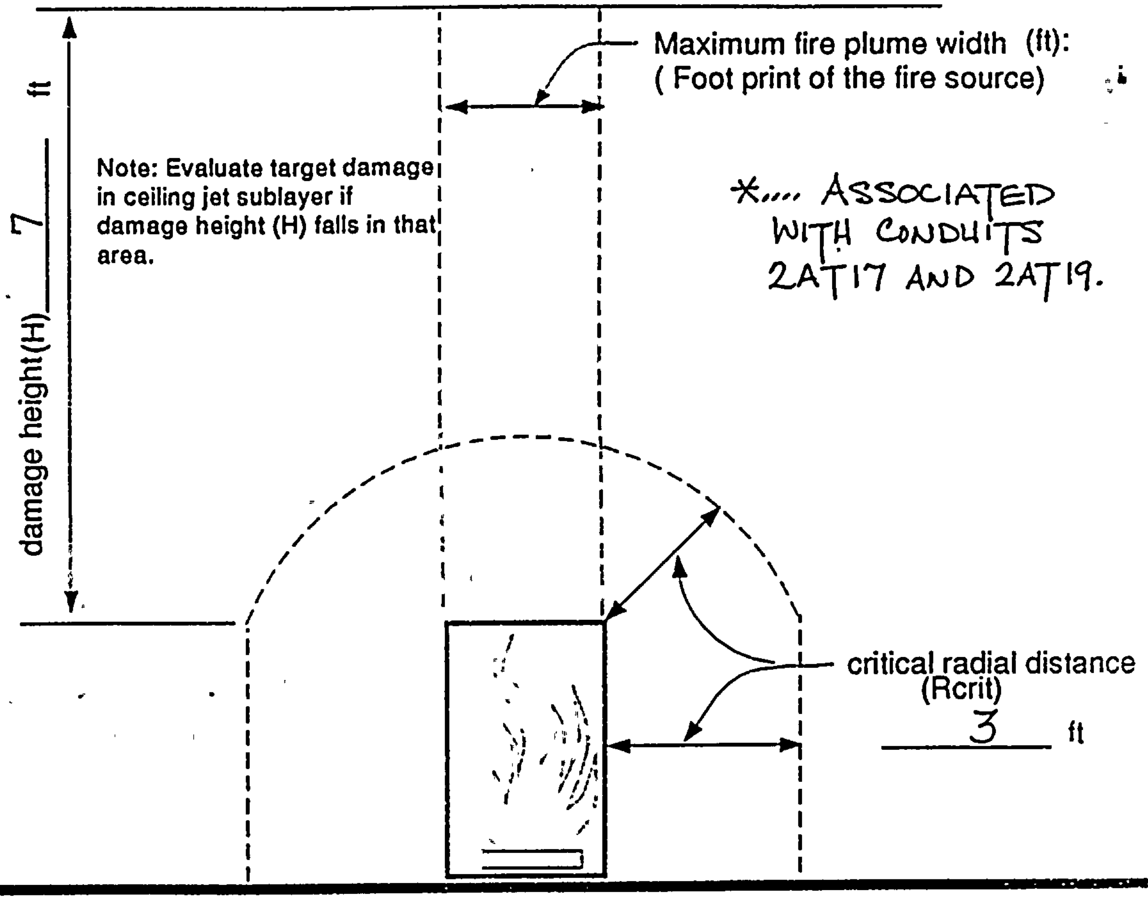
Ignition Source 4KV RPT BD. 2-II

cable trays in fire damage zone: ALL ARE ASSOCIATED WITH THE RPT BD.

conduits in fire damage zone: 2K954, 2FE7586, 2K956, 2AT19, 2AT17, 2K877, 2A524, 2ES4471, 2ES4472, 2ES5691-II, 2ES3906-II, 2FE8576 AND ONE COND. NO.=? [FROM C-ZONE STA. ENCLOSURE AT FRONT-SIDE OF RPT BD.]

● ALL OTHER CONDUITS ASSOCIATED WITH THE RPT BD.

other components in fire damage zone: PNLA-68-25/419 (ATWS)*



*.... ASSOCIATED WITH CONDUITS 2AT17 AND 2AT19.

Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party William L Aldridge   Date 7-12-94

Walkdown Second Party John A Ehrlich   Date 7-12-94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION RS 9-26-94

FIRE DAMAGE ZONE OF INFLUENCE

page ~~15~~ 1 of ~~24~~ 1

fire area/zone 2-5

EEB-010-1 PG 31 OF 88

Ignition Source PANEL 25-3 AND 25-9

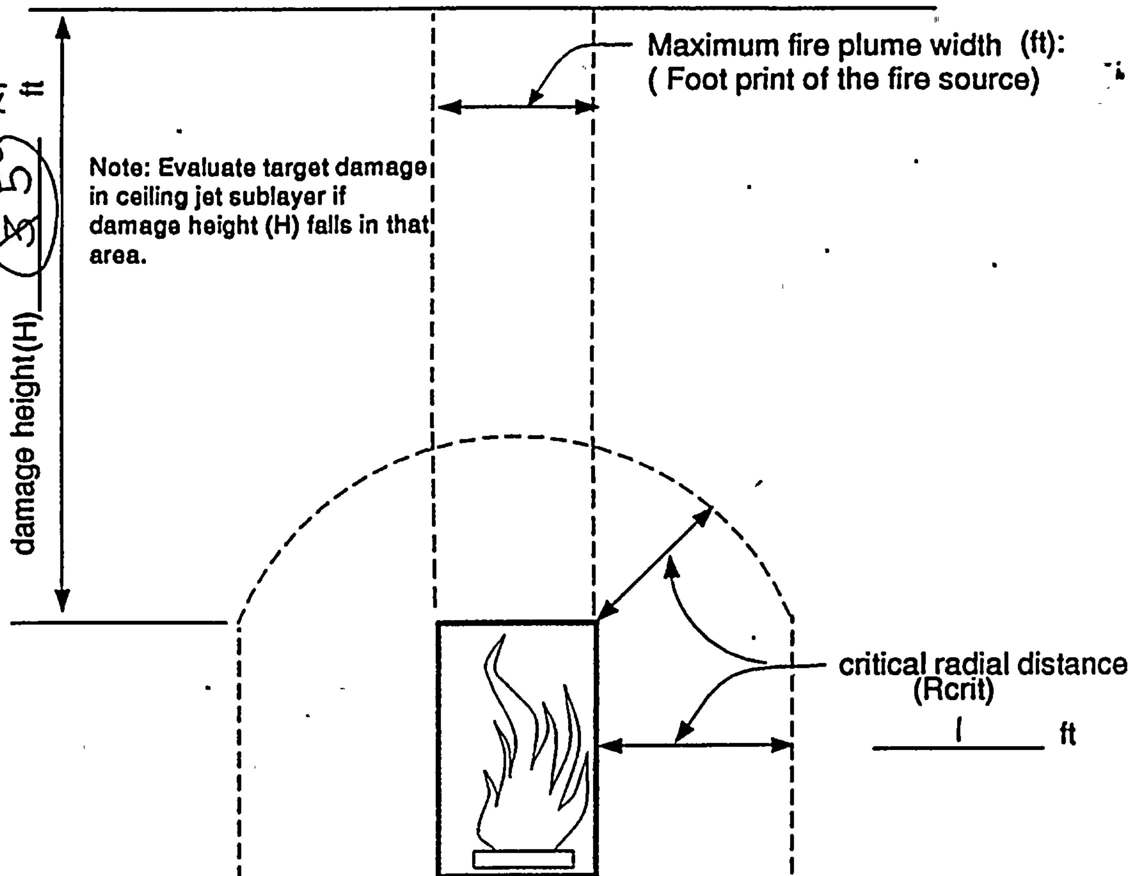
cable trays in fire damage zone: ~~NONE~~ "BY", "DW", & "DV" R1

conduits in fire damage zone:

• ALL CONDUITS ARE ASSOCIATED WITH THE TWO PANELS.

NOTE: TRAY "BY" MAY NOT BE WITHIN THE DAMAGE HEIGHT, BUT IS INCLUDED TO BE CONSERVATIVE. R1

other components in fire damage zone: NONE



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party [signature] Date 7-12-94

Walkdown Second Party [signature] Date 7-12-94

[signature] William L. Aldredge

RO [signature] 12/1/94

kktb0/ignition frequency (five)/walkdown.source loc1

WALKDOWN SCOPE IDENTIFICATION · FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE

fire area/zone 2-5

Ignition Source 240V LTG BD 2A/240V LTG. BD 2A XFRMR TL2A

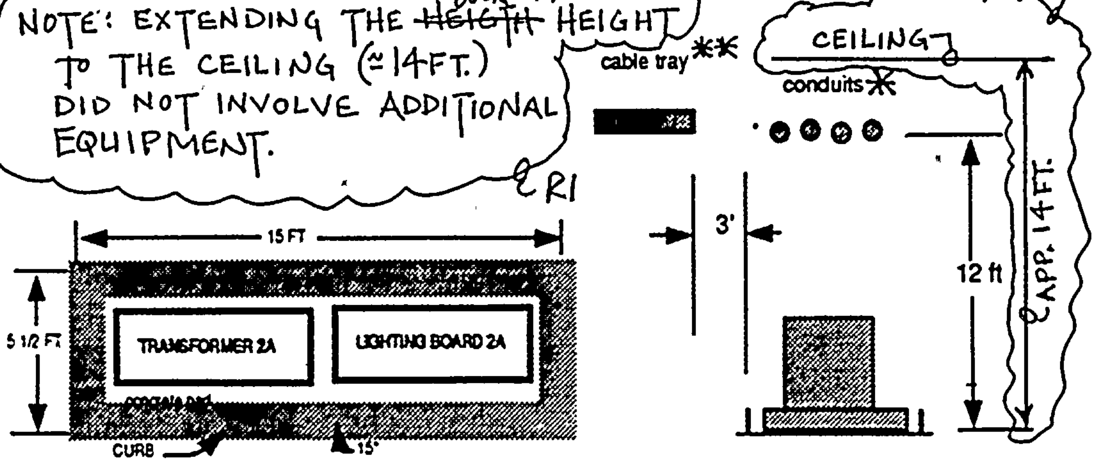
cable trays in fire damage zone: CL (TOP), GC (BOT.)**

conduits in fire damage zone: 2T15, 2T3, 2A2022, 2A2017, 2ES2774-II, 2B1471

*CONDUITS INDICATED IN DIAGRAM BELOW ARE APPROX. ~~17-20'~~ 14' RI ABOVE FLOOR & HAVE BEEN IDENTIFIED AS FIRE DET./ANNUNCIATOR SYSTEMS.

other components in fire damage zone: 240V LTG. BD 2A VOLT. REG, 2-LPNL-025-0031 (SLIGHTLY BEYOND FIRE ZONE)

NOTE: EXTENDING THE ~~HEIGTH~~ HEIGHT TO THE CEILING (≈14FT.) DID NOT INVOLVE ADDITIONAL EQUIPMENT. RI WRA 12/1/94



** CABLE TRAYS SHOWN IN DIAGRAM APPEAR TO BE OUTSIDE THE FIRE ZONE. TRAYS ON THE NEAR SIDE CL (TOP), GC (BOTTOM) ARE ≈3' FROM THE CURB AND ≈15' FROM THE FLOOR (TO BOTTOM TRAY). FAR SIDE TRAYS, BW (TOP), NO (MIDDLE) AND MN (BOTTOM) ARE ≈5' FROM THE CURB AND ≈12'/13' FROM THE FLOOR (TO BOTTOM TRAY)

Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party Perry Zimmerman Date 9-20-94 William L Aldredge 12/1/94

Walkdown Second Party David E. Fincher Date 9-20-94 R Abbas 12/1/94

orbC ignition frequency fire/walkdown source loc

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

PB 9-26-94

FIRE DAMAGE ZONE OF INFLUENCE

page ~~24~~ 1 of ~~24~~ 1

EEB-010-1
PG. 33 OF 88

fire area/zone 2-5

Ignition Source SLC PUMPS A & B*

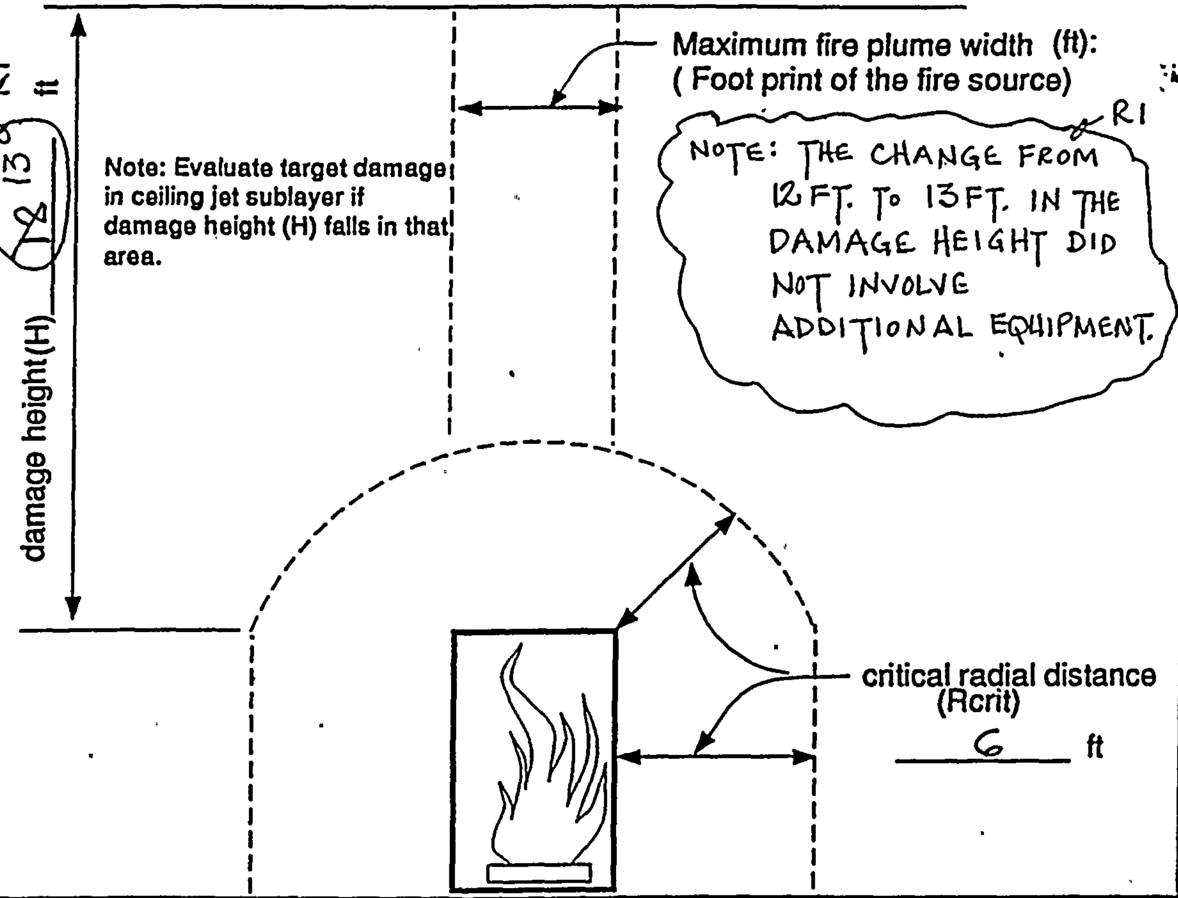
*.... SLC PUMPS A & B ARE WITHIN 6 FEET OF EACH OTHER.

cable trays in fire damage zone: NONE

conduits in fire damage zone: 2A337, 2A1024, 2A1468, 2A1472, 2A1471, 2A2649, SP1277, SP1273

- NITROGEN TANK • 2-JBOX-63-1028 • POWER OUTLET RECPT.
- VARIOUS SLC SYS. COMPONENTS (SURY VLV., MIX TANKS, DISCH. ACCUMLATOR, PULSATION DAMPERS, STORAGE TANK, SQUIB VLV.)

other components in fire damage zone: SEE ABOVE



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party Turner J. Howard Date 7-13-94 William L Aldredge 12/1/94

Walkdown Second Party John A Elimik Date 7-14-94 RO R Alihas 12/1/94

kktb0/ignition frequency (live)/walkdown.source loc1

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE

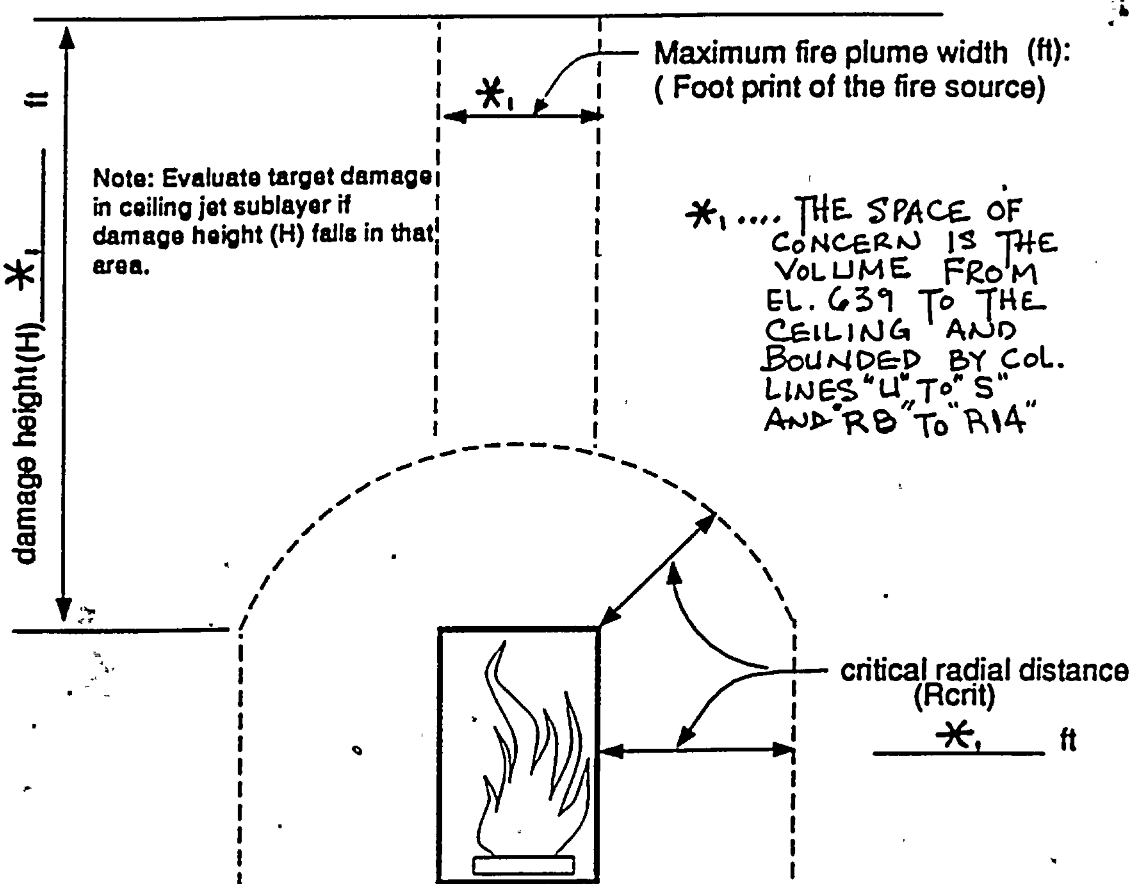
fire area/zone 2-6 EEB-010-1 PG. 34 OF 88

Ignition Source RECIRC. MG SET 2A & 2B

cable trays in fire damage zone: FU, NN, MO

conduits in fire damage zone: B518, 2A2242

other components in fire damage zone: See PAGE 2



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party David E. Fincher Date 9-15-94

Walkdown Second Party Perry Zimmerman Date 9-15-94

kktb0/ignition frequency (five)/walkdown.source loc1

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION
(CONTINUATION)

1. IGNITION SOURCE : RECIRC. MG SET 2A & 2B

2. CABLE TRAYS IN FIRE ZONE : (CONTINUED)

N/A

3. CONDUITS IN FIRE DAMAGED ZONE : (CONTINUED)

4. OTHER COMPONENTS IN THE FIRE DAMAGE ZONE : (CONTINUED)

RECIRC PUMPS
MG SET 2A

MG SET 2B

2-HS-96-48B · EMERGENCY OIL PUMP MG SET B
4KV-480V SHUTDOWN BOARDS 2A AND 2B
2-FCV-70-1

2-FSV-70-1

2-LS-70-2A

2-LS-70-2B

JB5953

JB5992

JB5952

JB5991

(CON'T ON PAGE 3)

WALKDOWN FIRST PARTY David E. [signature] DATE 9-15-94

WALKDOWN SECOND PARTY Perry Zimmerman DATE 9-15-94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION
(CONTINUATION)
page 3 of 3

1. IGNITION SOURCE : RECIRC MG SET 2A & 2B

2. CABLE TRAYS IN FIRE ZONE : (CONTINUED)

N/A

3. CONDUITS IN FIRE DAMAGED ZONE : (CONTINUED)

4. OTHER COMPONENTS IN THE FIRE DAMAGE ZONE : (CONTINUED)

MG SET 2EN
MG SET 2DA
LPNL-25-24
LPNL-25-23
JB 3110 (ON PNL-2-25-180)
JB 3111 (ON PNL-2-25-181)
JB 2919
2-LAC-247-0209
2-RI-90-4B
240V LIGHTING BD 2B
240V LIGHTING XFMR
240V LIGHTING BD 2B VOLTAGE REG
2-LS-075-0078A
2-LS-075-0078B
2-LS-075-0078C
2-LS-075-0078D
2-HS-78-62B
2-HS-96-48A
2-FCV-78-62

OVERHEAD, EVACUATION, EMERGENCY LIGHTS, FIRE PMP START, SPEAKERS, SMOKE DETECTORS, BELLS, OUTLETS, S.P. JACKS, SIREN

WALKDOWN FIRST PARTY David E. Linder DATE 9-15-94

WALKDOWN SECOND PARTY Perry Zimmerman DATE 9-15-94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

PB 9-26-94

FIRE DAMAGE ZONE OF INFLUENCE

page ~~14~~ 1 of ~~24~~ 1

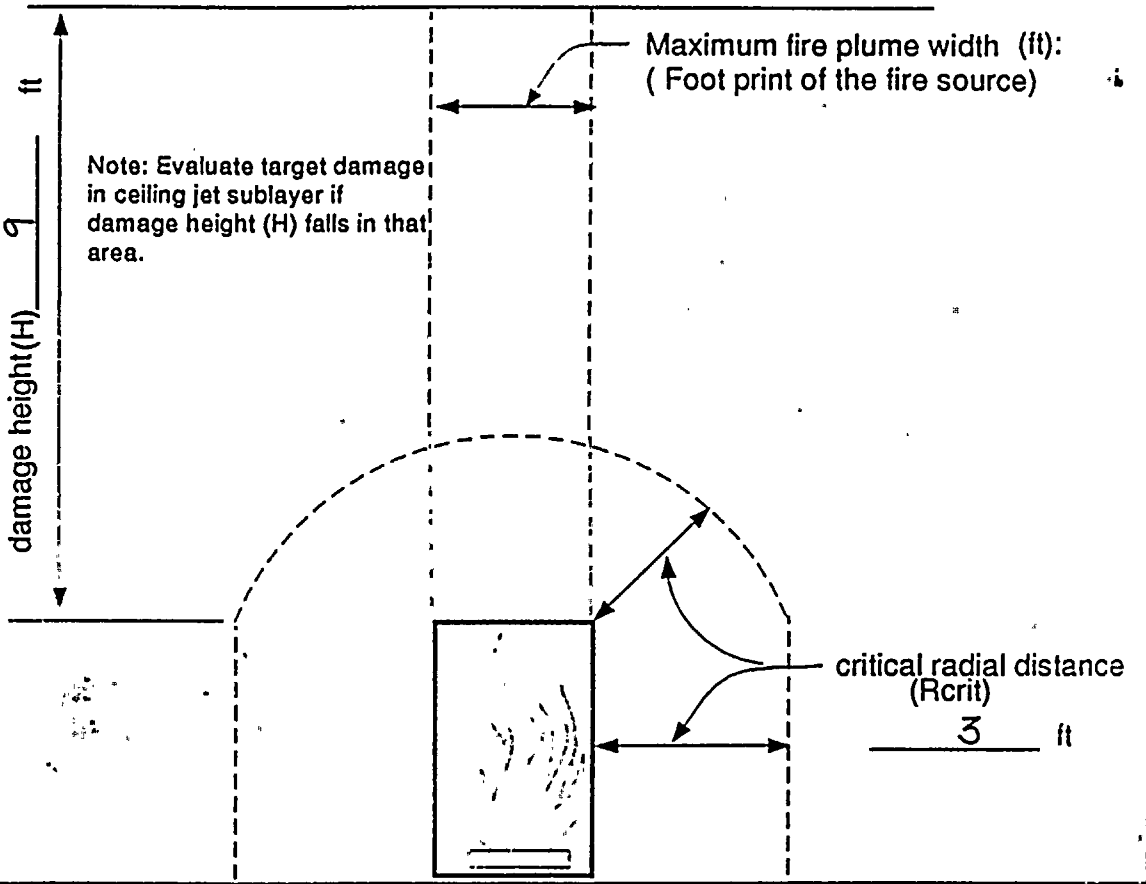
fire area/zone 2-6 (EL. 639) EEB-010-1 PG 37 OF 88

Ignition Source 4KV-480V TRANSFORMER TS2E

cable trays in fire damage zone: NONE

conduits in fire damage zone: NONE (EXCEPT THOSE ASSOCIATED WITH THE TRANSFORMER)

other components in fire damage zone: NONE



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party William T Aldredge Date 7-12-94

Walkdown Second Party John A Elwell Date 7-12-94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION PB 9-26-94

FIRE DAMAGE ZONE OF INFLUENCE

page ~~17~~ 1 of ~~24~~ 1

fire area/zone 2-6

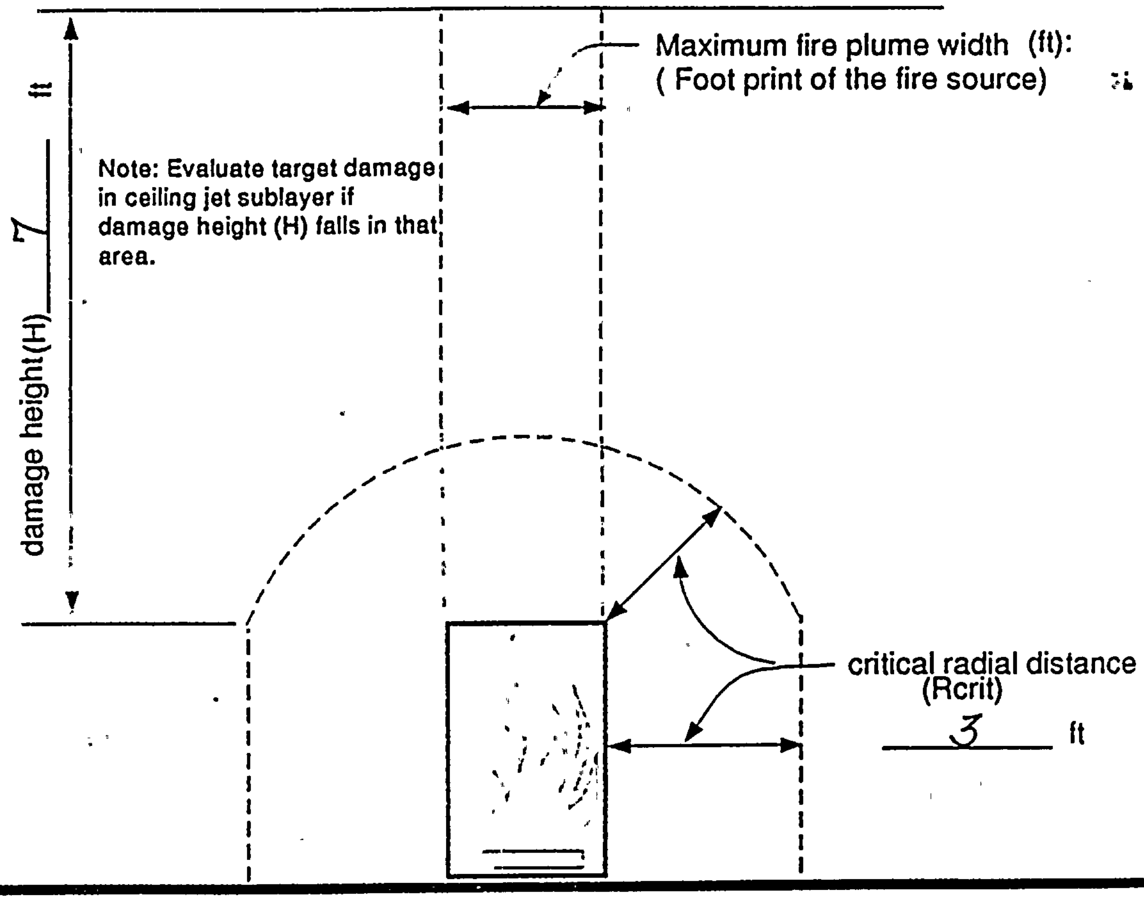
EEB-010-1 PG 38 OF 88

Ignition Source (LPCI) MG SET 2DA

cable trays in fire damage zone: NONE

conduits in fire damage zone: NONE (EXCEPT THOSE ASSOCIATED WITH THE MG SET)

other components in fire damage zone: NONE



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party William L Aldredge Date 7-12-94

Walkdown Second Party John A Elmit Date 7-12-94

kktb0/ignition frequency (five)/walkdown.source loc1

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

PB 9-26-94

FIRE DAMAGE ZONE OF INFLUENCE

fire area/zone 2-6

Ignition Source (LPCI) MG SET 2EN

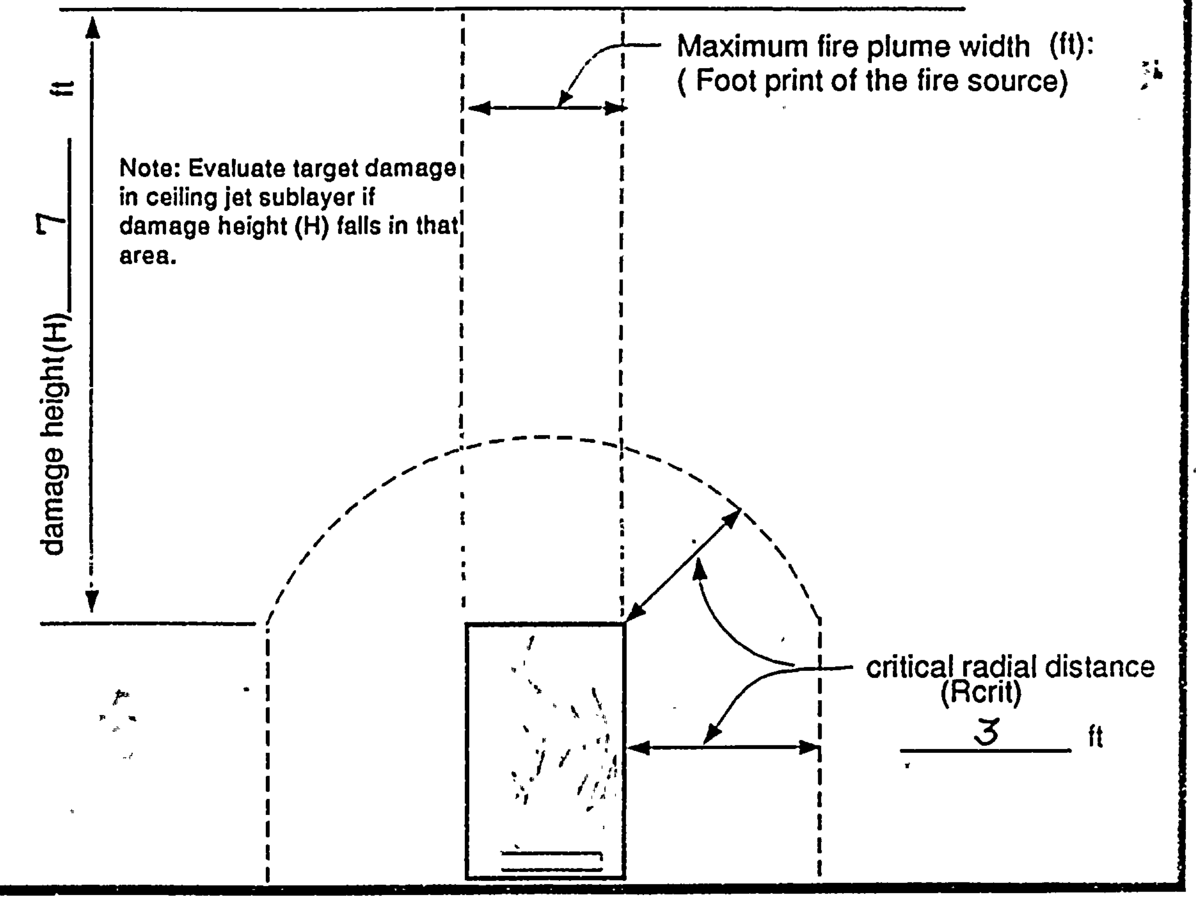
cable trays in fire damage zone: NONE

conduits in fire damage zone: —

BOTH INPUT AND OUTPUT POWER FOR LPCI MG SET 2DA.

• ALL OTHER CONDUITS ARE ASSOCIATED WITH MG SET 2EN.

other components in fire damage zone: NONE



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party William L Aldridge Date 7-12-94

Walkdown Second Party John A Elliott Date 7-12-94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE

fire area/zone 2-6

EEB-010-1 PG 40 OF 88

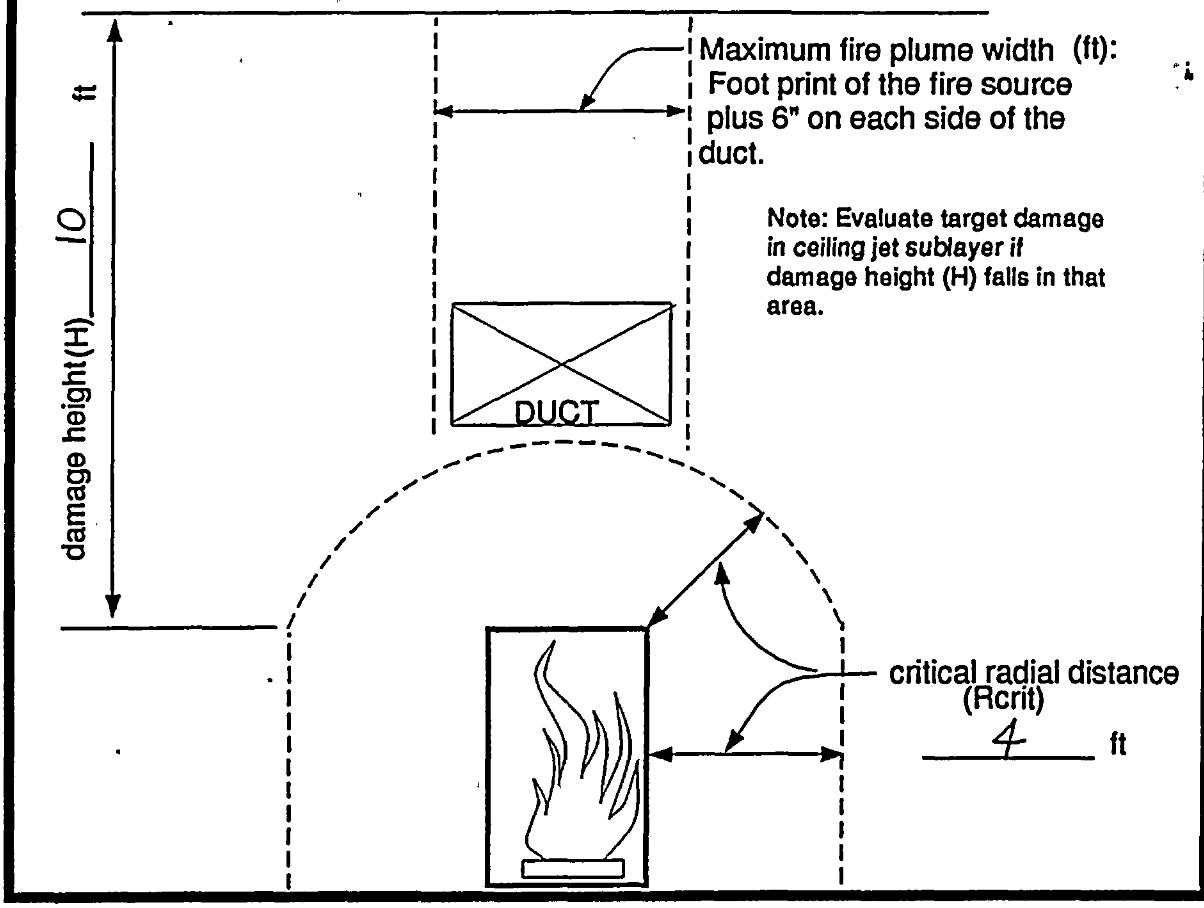
Ignition Source LPNL-25-23 AND -24

cable trays in fire damage zone: NONE

conduits in fire damage zone:

- CONDUITS ASSOCIATED WITH PANELS 25-23 & -24.
- 2LT25 & 2LT35..... SPEAKER SYSTEM (ASSOCIATED WITH 2-AMP-244-6131 & -6132)
- PANELS 2-25-180 (JB3110) AND 2-25-181 (JB3111)

other components in fire damage zone: SEE ABOVE



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party William L Aldredge Date 7-13-94

Walkdown Second Party Turner L Howard Date 7-13-94

kktb0/ignition frequency (five)/walkdown.source loc1

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE

fire area/zone 2-6

Ignition Source 240V LIGHTING BOARD 2B

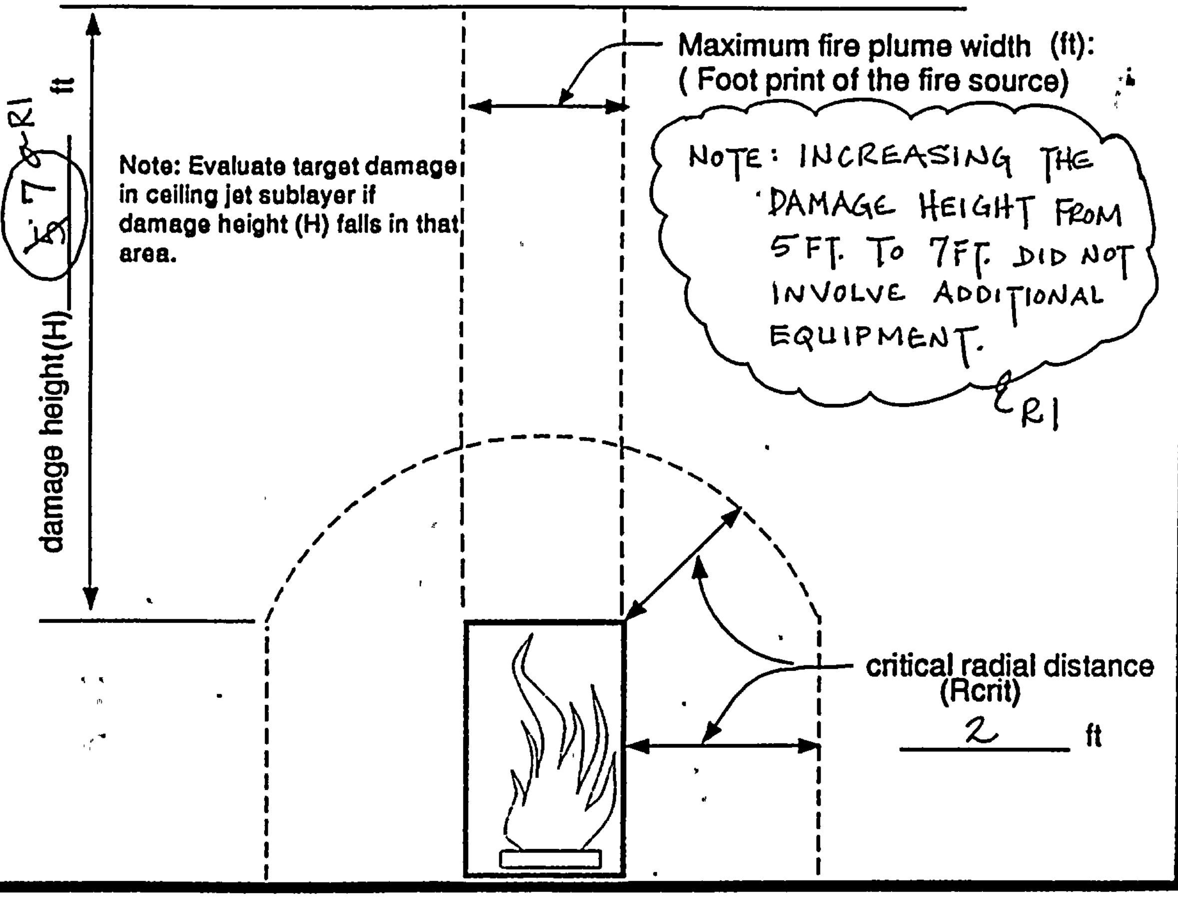
*.... LOCATED APP. 3½ FT. FROM THE IGNITION SOURCE.

cable trays in fire damage zone: NONE

conduits in fire damage zone: 2FE565, 2FE564, 2FE129, 2FE130 AND 2ES3925-II* (ALL OTHERS ASSOCIATED WITH THE LTG. BD.).

- FIRE PUMP START SWITCH (ASSOCIATED WITH 2FE129 & 2FE130).
- UTILITY OUTLET RECPT. (FED FROM LC-209).
- COMMUNICATION PHONE JACK (PAX 218).

other components in fire damage zone: SEE ABOVE



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party *William L. Aldredge* Date 7-12-94] *William L. Aldredge 12/1/94*]

Walkdown Second Party *[signature]* Date 7-12-94] RO *R Alonas 12/1/94*]

kktb0/ignition frequency (five)/walkdown.source loc1

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE

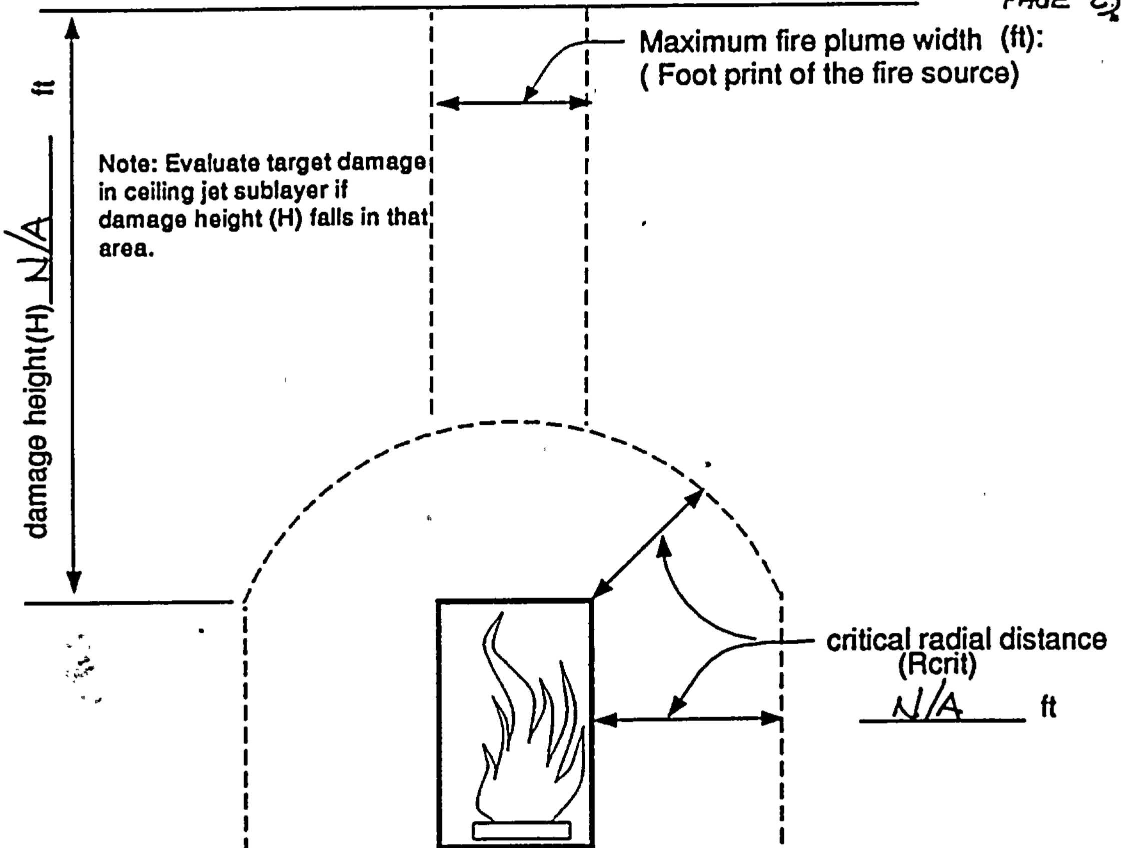
fire area/zone 4 EEB-010-1 PG 42 OF 88

Ignition Source ENGULFING FIRE - 4KV SHUTDOWN BOARD ROOM B

cable trays in fire damage zone: N/A

conduits in fire damage zone: ES1817-IA

other components in fire damage zone: 4KV SHUTDOWN BD B (CONT ON PAGE 2)



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party Perry Zimmerman Date 8-26-94

Walkdown Second Party [illegible] Date 8-31-94

kktb0/ignition frequency (five)/walkdown.source loc1

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION
(CONTINUATION) page 2 of 2

1. IGNITION SOURCE :
ENGULFING FIRE - 4KV SD BD Rm B

2. CABLE TRAYS IN FIRE ZONE : (CONTINUED)
N/A

3. CONDUITS IN FIRE DAMAGED ZONE : (CONTINUED)
- SEE PAGE 1 -

4. OTHER COMPONENTS IN THE FIRE DAMAGE ZONE : (CONTINUED)
480V RMOV BD 1B, 250V DC RMOV BD 1B, 1-LPNL-925-0541, 1-TS-031-7205D, 1-TS-031-7206C, PNL 25-45B, DIV. I ECCS ATU INVERTERS, 1-JBOX-253-6459, 1-JBOX-253-8865, 1-XFA-253-001B, 1-JBOX-253-6458, I/C BUS B TRANSFORMER, EMERGENCY LIGHTS, OVERHEAD LIGHTS, SOUND POWERED PHONES, FIRE ALARM

WALKDOWN FIRST PARTY Perry Zimmerman DATE 8-26-94

WALKDOWN SECOND PARTY David E. Linder DATE 8-31-94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE

page 1 of 2

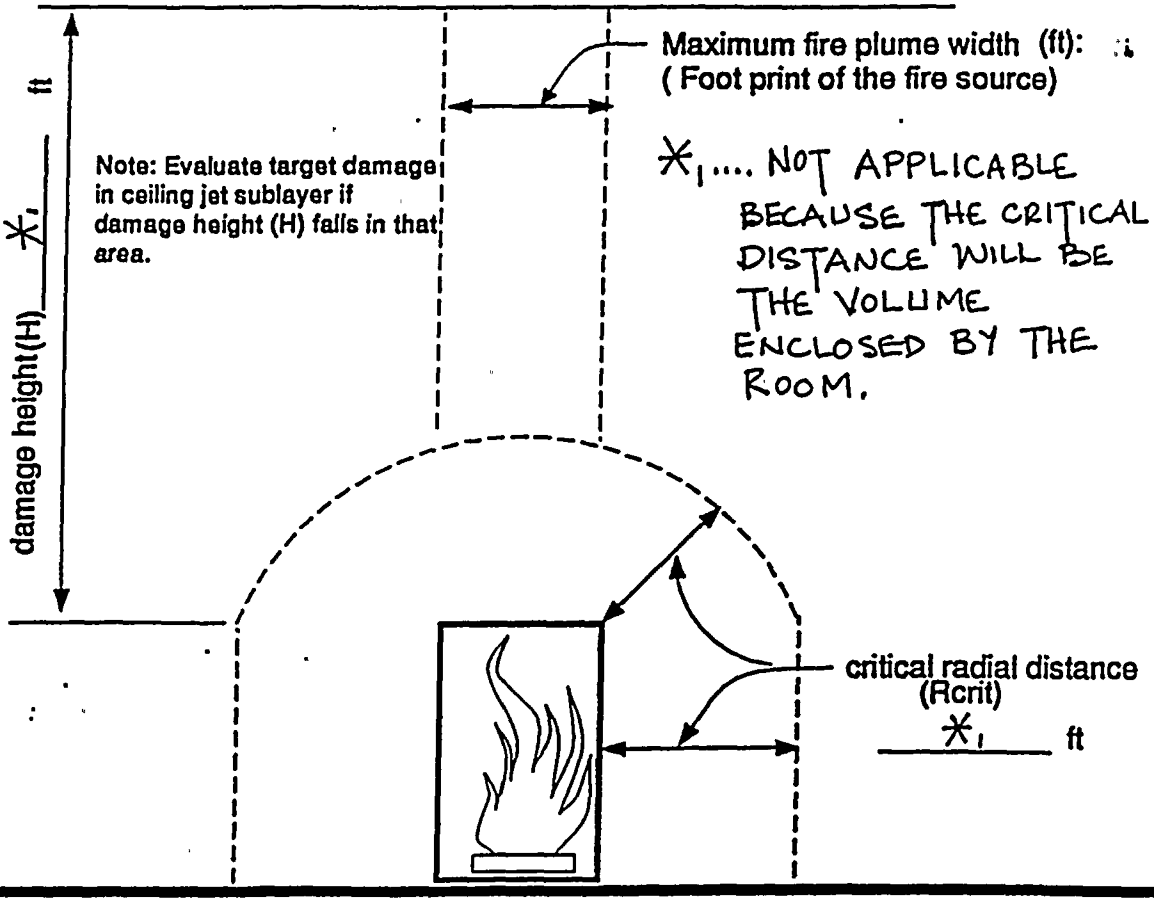
fire area/zone 5 EEB-010-1 PG. 44 OF 88

Ignition Source ENGULFING FIRE - 4KV SHUTDOWN BOARD ROOM A AND 250V BATT. ROOM A&B

cable trays in fire damage zone: N/A

conduits in fire damage zone: ES3868-II, 2B284-IE, 1B283-IE, 1LS380-A2, 1PL455-I, 1K930, 1K934, 1K936, PP626

other components in fire damage zone: SEE PAGE 2



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party [signature] Date 9-1-94

Walkdown Second Party [signature] Date 9-1-94

kktb0/ignition frequency (live)/walkdown.source loc1

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION
(CONTINUATION) page 2 of 2

1. IGNITION SOURCE : ENGULFING FIRE – 4KV SHUTDOWN BOARD ROOM A & 250V BATTERY ROOM A & B

2. CABLE TRAYS IN FIRE ZONE : (CONTINUED)

N/A

3. CONDUITS IN FIRE DAMAGED ZONE : (CONTINUED)

SEE SHEET 1

4. OTHER COMPONENTS IN THE FIRE DAMAGE ZONE : (CONTINUED)

4KV SHUTDOWN BOARD A DIV I, 480V REACTOR MOV BOARD 1A, 250V DC REACTOR MOV BOARD 1A, PANEL 25-45A, I&C BUS A TRANSFORMER, 1-JBOX-253-6455, 1-XFA-253-001A, 1-JBOX-253-6456, 1-JBOX-253-8862, ECCS ATU INVERTERS, BACKUP CONTROL PANEL 25-32, 1-LPNL-925-0539, 1-LPNL-925-0542, 1-TS-31-7206, 1-LPNL-925-0610, 1-TS-031-7205C
PNL 1-25-232, 0-FS-31-166A, 0-FCO-31-160A, 0-FCO-31-160B, 0-FS-31-167, 0-FS-31-168, 0-FS-31-169, 0-FS-31-170, 0-TS-31-162, 0-HS-31-160, 0-HS-31-161, 250V DC DIST. PANELS SB-A & SB-B, DISC. SWITCH (480V PWR TO BATT CHGR FOR 4KV SD BD B), 250V DC DIST. PANELS SB-A & SB-B, DISC. SWITCH (480V PWR TO BATT CHGR FOR 4KV SD BD A), EVAC. LIGHT, EMERG. LIGHTS, OVERHEAD LIGHTS, SOUND POWERED PHONE JACKS, OUTLETS, SWITCHES.

WALKDOWN FIRST PARTY [signature] DATE 9-1-94

WALKDOWN SECOND PARTY [signature] DATE 9-1-94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE

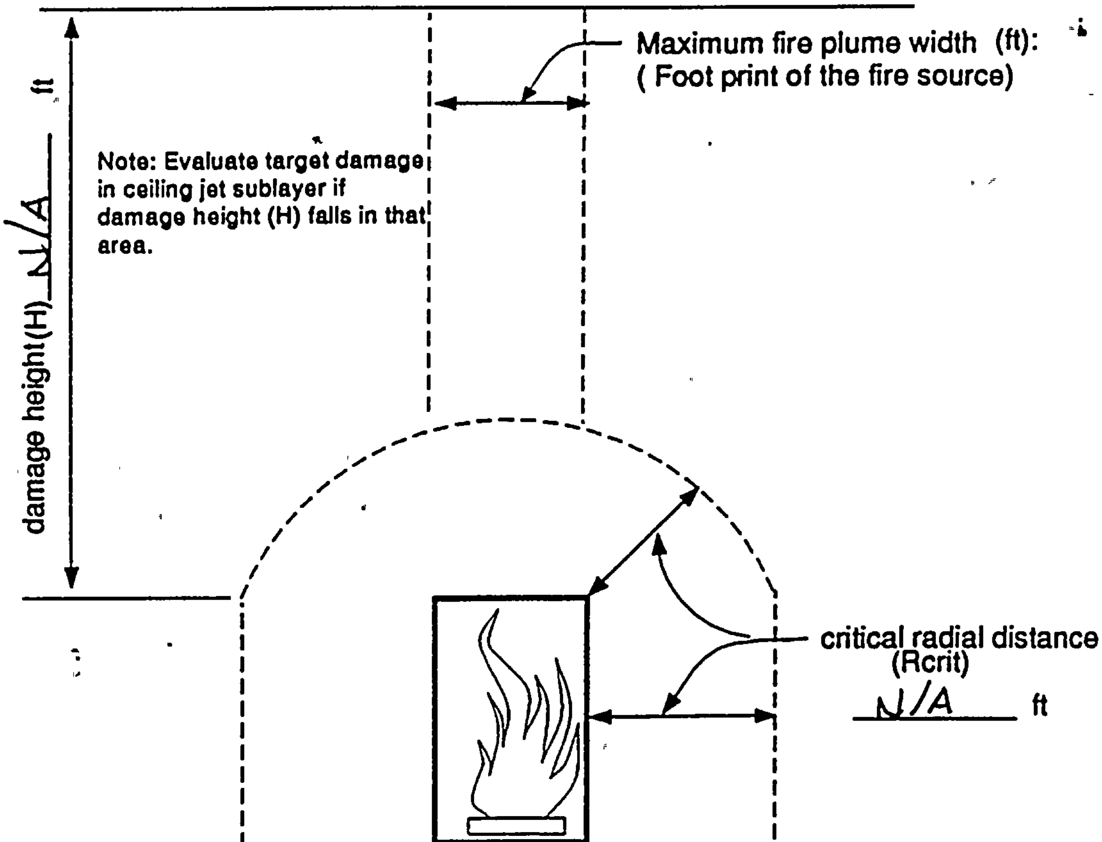
fire area/zone 6 EEB-010-1 PG 46 OF 88

Ignition Source 480V SHUTDOWN BOARD 1A (ENGULFING FIRE)

cable trays in fire damage zone: N/A

conduits in fire damage zone: ES2027-IA, PP625

other components in fire damage zone: SEE PAGE 2



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party David E. Tinder Date 9-9-94

Walkdown Second Party Perry Zimmerman Date 9-9-94

kktb0/ignition frequency (live)/walkdown.source loc1

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION
(CONTINUATION)
page 2 of 2

1. IGNITION SOURCE :
ENGULFING FIRE- 480V SHUTDOWN BOARD 1A

2. CABLE TRAYS IN FIRE ZONE : (CONTINUED)

3. CONDUITS IN FIRE DAMAGED ZONE : (CONTINUED)

- SEE PAGE 1 -

4. OTHER COMPONENTS IN THE FIRE DAMAGE ZONE : (CONTINUED)

480V SHUTDOWN BD 1A
PANEL 25-44A-11 (480V UNIT LOGIC PANEL)
PANEL 25-44B-11 (480V UNIT LOGIC PANEL)
VOLTAGE REGULATOR 1EA (MG SET CONTROL)
VOLTAGE REGULATOR 1DN (MG SET CONTROL)
EMERGENCY / EVACUATION LIGHT, PHONE JACKS, RECEPTACLE

WALKDOWN FIRST PARTY David E. [illegible] DATE 9-9-94
WALKDOWN SECOND PARTY Perry Zimmerman DATE 9-9-94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE

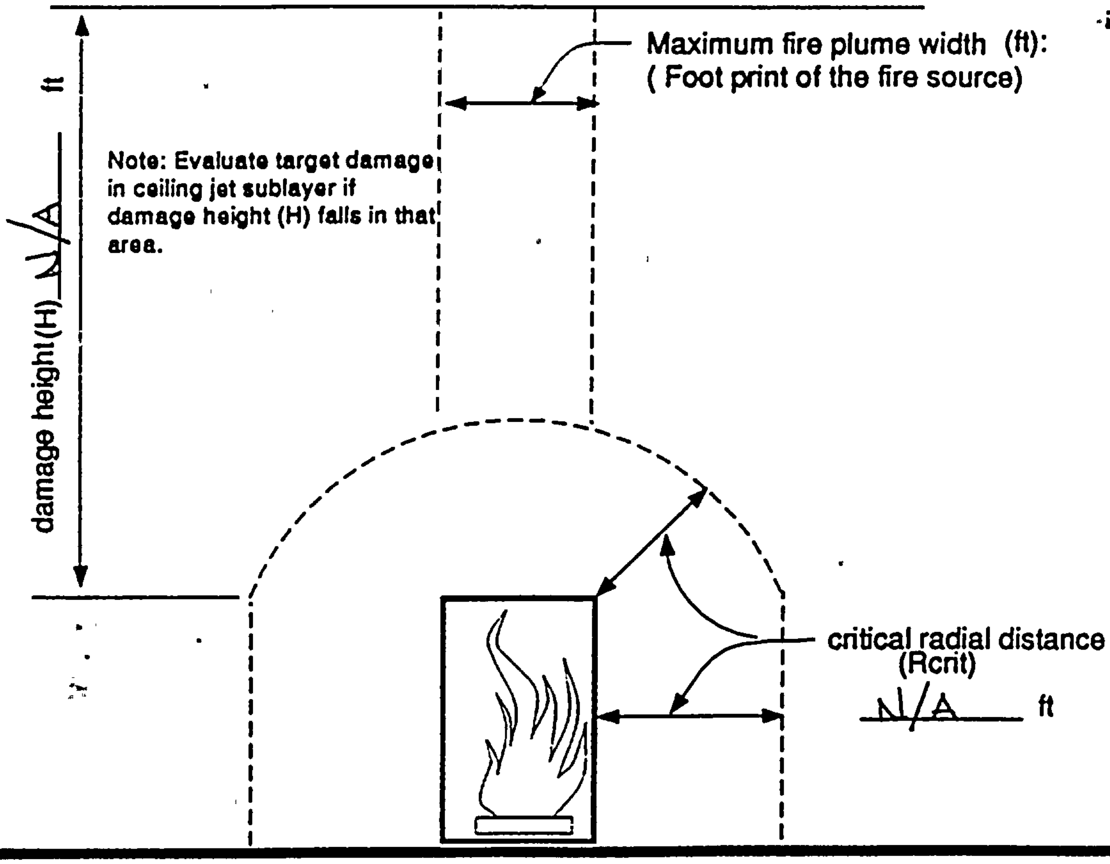
fire area/zone 7

Ignition Source ENGULFING FIRE 480V SD BD 1B

cable trays in fire damage zone: N/A

conduits in fire damage zone: 1PL763, 1PL378, 1PL762, 1PL556, 1PL470, 1ES3191, 1ES340*, 1ES339*, 1ES341*, 1ES312*, 1ES3950*, 1ES330*, 1ES329*, 1ES313*, CONDUITS ASSOC. W/PEN. S16214150, 51, 52, 53 AND S16212124.
(*, CONDUITS IDENTIFIED AT 480V SD BD Rm 1A)

other components in fire damage zone: SEE PAGE 2



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party Perry Zimmerman Date 8-17-94

Walkdown Second Party Turner V. Howard Date 8-17-94

kktb0/ignition frequency (live)/walkdown.source loc1

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION
(CONTINUATION)

1. IGNITION SOURCE :
480V SD BD 1B - ENGULFING FIRE

2. CABLE TRAYS IN FIRE ZONE : (CONTINUED)
N/A

3. CONDUITS IN FIRE DAMAGED ZONE : (CONTINUED)
- SEE PAGE 1 -

4. OTHER COMPONENTS IN THE FIRE DAMAGE ZONE : (CONTINUED)
480V SD BD 1B, VR BOX 1EN, VR BOX 1DA, PNL 25-44A-12, PNL 25-44B-12, OVERHEAD LIGHTS, EMERG. LIGHTS, SWITCHES, RECEPTICLES, EVAC. LIGHT

WALKDOWN FIRST PARTY Perry Zimmerman DATE 8-17-94

WALKDOWN SECOND PARTY Turner A. Howard DATE 8-17-94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE page 1 of 2

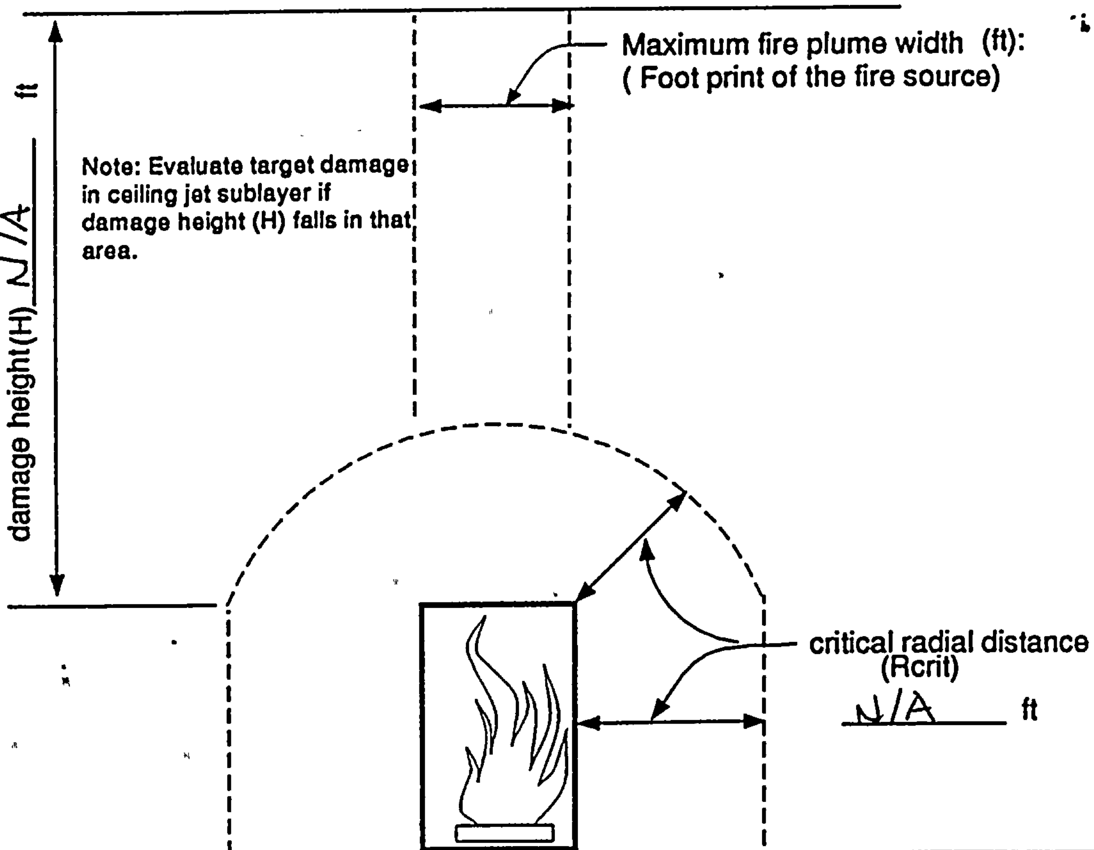
fire area/zone 8 EEB-010-1 PG 50 OF 88

Ignition Source ENGULFING FIRE - 4KV SHUTDOWN BOARD ROOM D

cable trays in fire damage zone: N/A

conduits in fire damage zone: 2ES2775-II, ES4591, ES4593, 2ES1040, 2M501-I

other components in fire damage zone: CONT. ON PAGE 2



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party Perry Zimmerman Date 8-26-94

Walkdown Second Party David E. [illegible] Date 8-29-94

kktb0/ignition frequency (five)/walkdown.source loc1

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION
(CONTINUATION)
page 2 of 2

1. IGNITION SOURCE :
ENGULFING FIRE - 4KV SHUTDOWN BOARD ROOM D

2. CABLE TRAYS IN FIRE ZONE : (CONTINUED)

N/A

3. CONDUITS IN FIRE DAMAGED ZONE : (CONTINUED)

— SEE PAGE 1 —

4. OTHER COMPONENTS IN THE FIRE DAMAGE ZONE : (CONTINUED)
4KV SD BD D, 480V RMOV BD 2B, 250V DC RMOV BD 2B, DIV. I ECCS ATU INVERTER, I/C BUS B TRANSFORMER, 2-JBOX-253-7195, I/C BUS B REGULATOR DIV. II, 2-JBOX-253-7196, 0-LPNL-0045D, 2-LPNL-925-8614, 2-LPNL-925-0541, 2-LPNL-925-0539, 2-LPNL-925-8613, 2-JBOX-256-9722, JB8271, FIRE ALARMS, EVAC. LIGHT, EMERGENCY LIGHTING, SMOKE DETECTORS, ISOLATION INDICATORS, SOUND POWERED PHONES/JACKS, OUTLETS, SWITCHES, OVERHEAD LIGHTS

WALKDOWN FIRST PARTY Perry Zimmerman DATE 8-26-94

WALKDOWN SECOND PARTY David E. [illegible] DATE 8-29-94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE

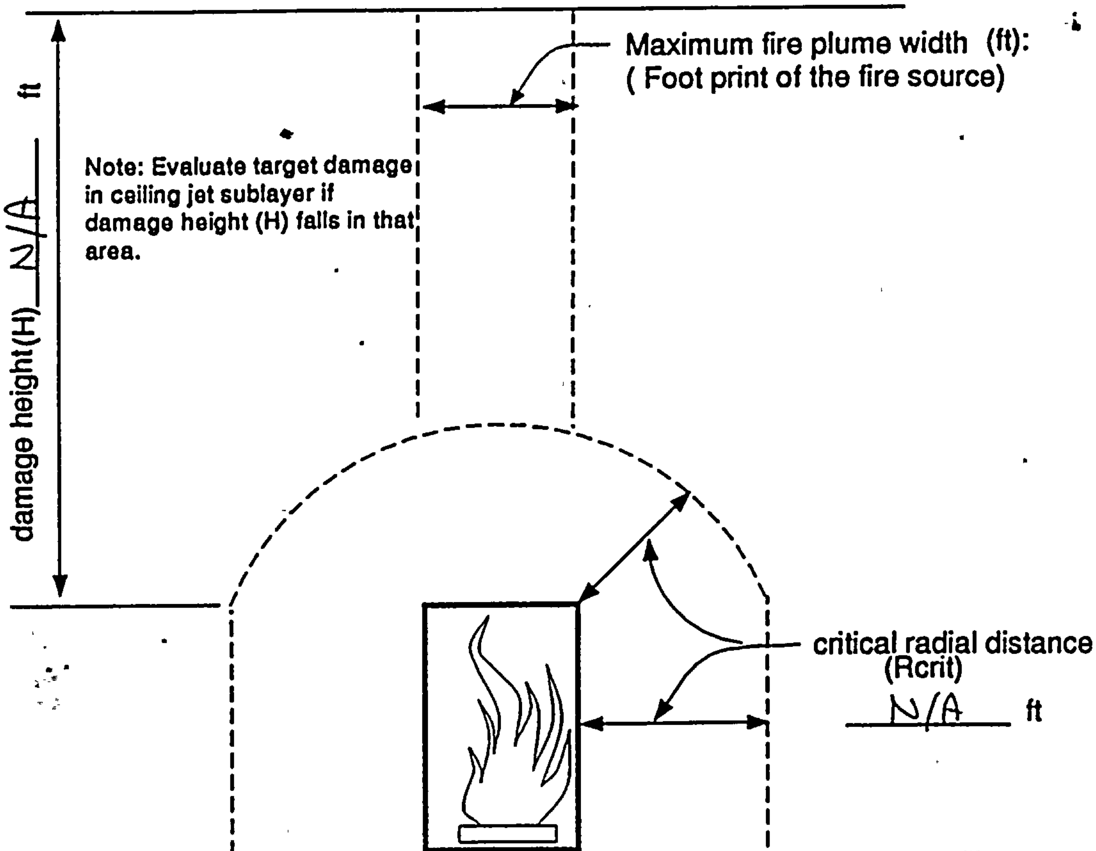
fire area/zone 9 EEB-010-1 PG 52 OF 88

Ignition Source ENGULFING FIRE- AKV SHUTDOWN BD ROOM C

cable trays in fire damage zone: N/A

conduits in fire damage zone: 2LS308-A1, 2LS309-A1, 2PL769
2PL461-II, ES3868-II, ES2674-II, 2B284-IE

other components in fire damage zone: CON'T ON PAGE 2



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party David E. Fincher Date 9-7-94

Walkdown Second Party Perry Zimmerman Date 9-7-94

kktb0/ignition frequency (five)/walkdown.source loc1

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION
(CONTINUATION)
page 2 of 2

1. IGNITION SOURCE :
ENGULFING FIRE - 4KV SHUTDOWN BD ROOM C

2. CABLE TRAYS IN FIRE ZONE : (CONTINUED)

3. CONDUITS IN FIRE DAMAGED ZONE : (CONTINUED)

- SEE PAGE 1 -

4. OTHER COMPONENTS IN THE FIRE DAMAGE ZONE : (CONTINUED)

4KV SHUTDOWN BOARD C, 480 V RMOV BD 2A, 250 V RMOV BD 2A, DIV II ECCS ATU INVERTER, 4KV LOGIC RELAY PANEL, 480 V COMMON LOGIC RELAY PANELS, 2-JBOX-031-8270, PANEL 2-25-32, PANEL 2-25-232, 2-LPNL-025-0416, 2-LPNL-025-0417, I&C BUS XFMR A, I&C BUS A REGULATOR, I&C BUS A CKT BREAKER BOX, 2-JBOX-253-7192, JB 4120, JB 3253
OVERHEAD LIGHTS, EMERGENCY LIGHTS, EVACUATION LIGHT, SMOKE
PB ~~DETECTORS~~, SIREN, PHONE JACKS, CARD READER
9-22-94 DETECTORS

WALKDOWN FIRST PARTY David E. Linder DATE 9-7-94

WALKDOWN SECOND PARTY Barry Zimmerman DATE 9-7-94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE

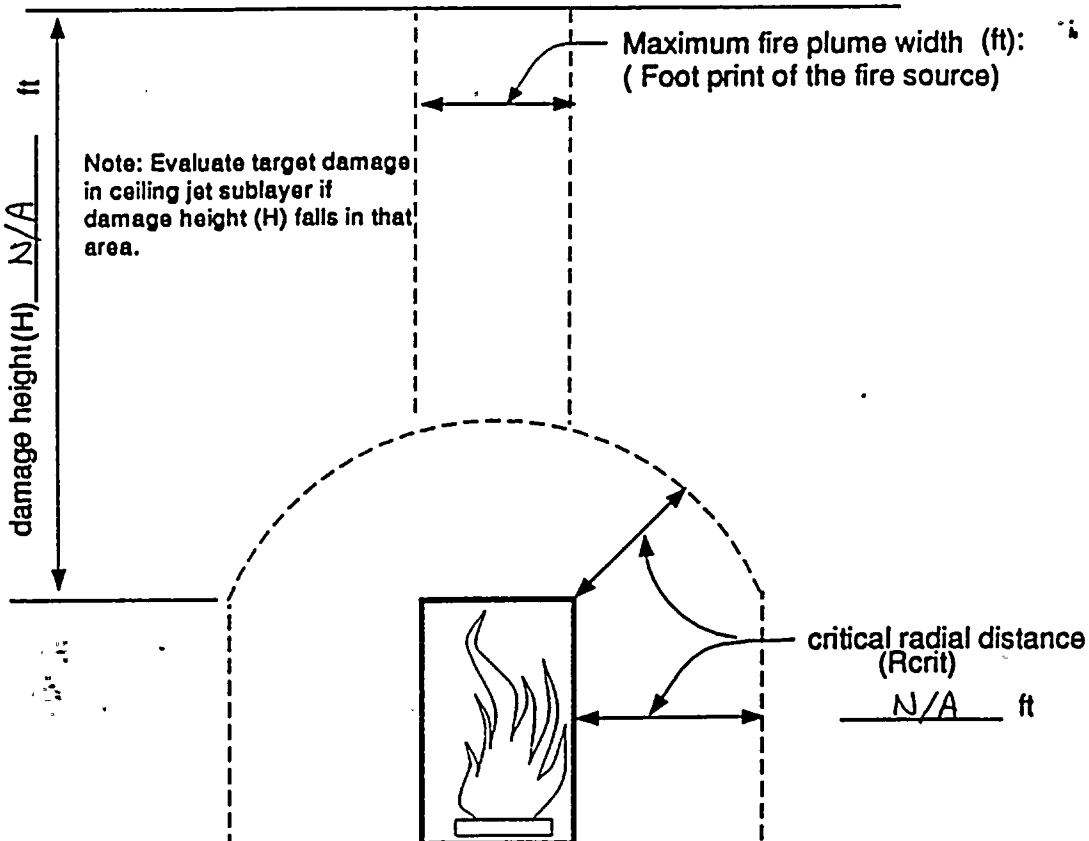
fire area/zone 9 EEB-010-1 PG 54 OF 88

Ignition Source ENGULFING FIRE – 250 V BATTERY BOARD RM C & D

cable trays in fire damage zone: N/A

conduits in fire damage zone: NONE

other components in fire damage zone: CON'T ON PAGE 2



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party David E. Fincher Date 9-7-94

Walkdown Second Party Gary Zimmerman Date 9-7-94

kktb0/ignition frequency (five)/walkdown.source loc1

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION
(CONTINUATION)

1. IGNITION SOURCE :
ENGULFING FIRE 250 V BATTERY BOARD RM C & D

2. CABLE TRAYS IN FIRE ZONE : (CONTINUED)
N/A

3. CONDUITS IN FIRE DAMAGED ZONE : (CONTINUED)
NONE

4. OTHER COMPONENTS IN THE FIRE DAMAGE ZONE : (CONTINUED)

- 0-FCO-31-163A, 0-FCO-31-163B, 0-FCO-31-164A,
- 0-FCO-31-164B, 0-TS-31-165, 0-HS-31-1033,
- 250V DC DISTRIBUTION PANEL SB-C
- 250 V DC DISTRIBUTION PANEL SB-D
- 250 V DC BATTERY CHARGER SB-C
- 250 V DC BATTERY CHARGER SB-D
- DISCONNECT SW - FOR 4KV SHUTDOWN BD C
- DISCONNECT SW - FOR 4KV SHUTDOWN BD D
- OVERHEAD LIGHTS, EMERGENCY LIGHTS, EVAC LIGHT, RECEPTACLES, SMOKE DETECTORS, PHONE JACKS

WALKDOWN FIRST PARTY David E. [signature] DATE 9-7-94

WALKDOWN SECOND PARTY Barry Zimmerman DATE 9-7-94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION
FIRE DAMAGE ZONE OF INFLUENCE

fire area/zone 10   EEB-010-1 PG 56 OF 88

Ignition Source 480V SHUTDOWN BOARD 2A

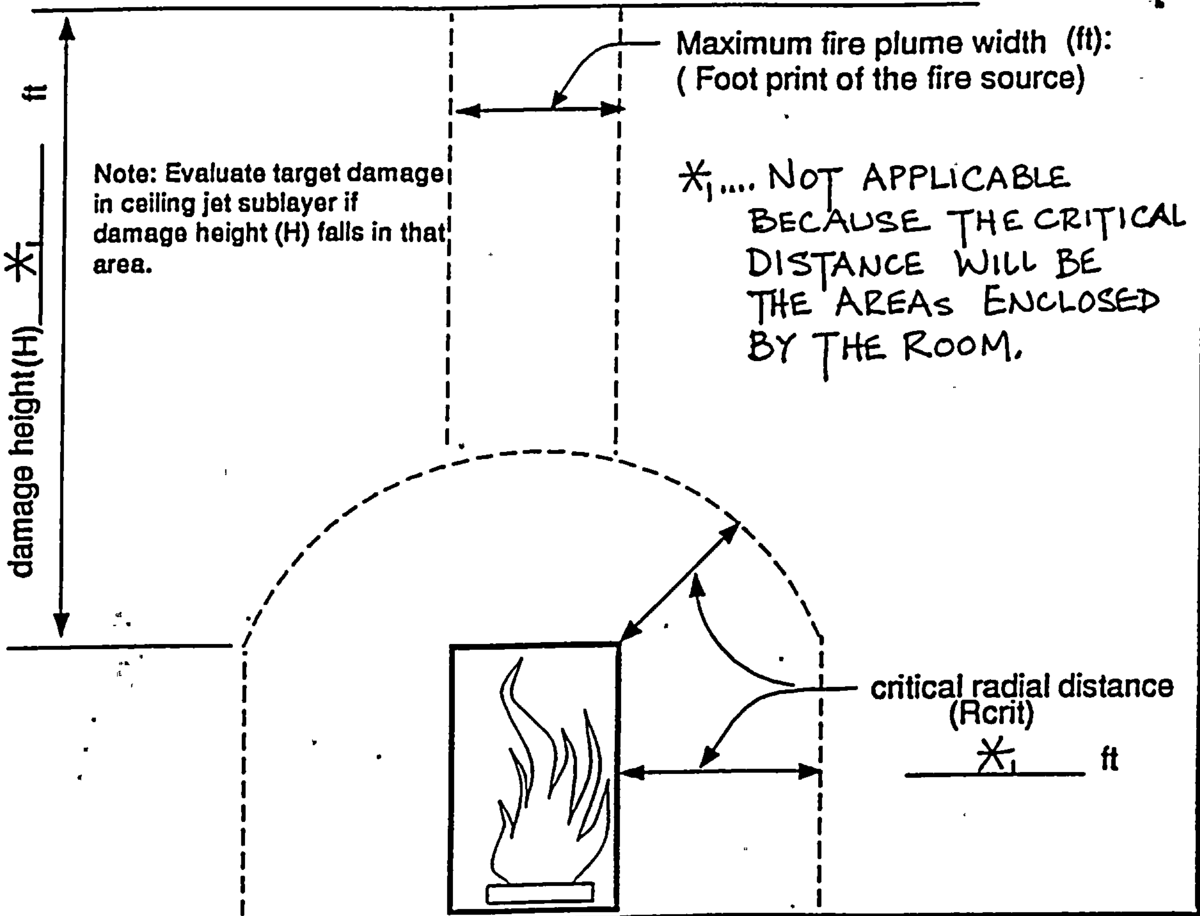
*1 .... CONDUITS AND CABLES ASSOCIATED WITH 480V SD BD. 2A

cable trays in fire damage zone: NONE

conduits in fire damage zone: 2A960; 2LS757-A1; 2LS347-B1; *1; 2LS767-B1; CONDUITS FOR PENET. NOs.: S26212212 & S26212213.

- JB4749 (ALL CKT. FOR SD BD. 2A)
- PANELS 2-25-44A-11 & 2-25-44B-11
- OVERHEAD LIGHTS, EMERGENCY LIGHTS, FIRE ALARM, BLUE LIGHT, SOUND POWERED PHONES AND POWER OUTLET RECPT.

other components in fire damage zone: SEE ABOVE



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party *William L Aldredge*   Date 7-22-94

Walkdown Second Party *Turner J. Howard*   Date 7-25-94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION
FIRE DAMAGE ZONE OF INFLUENCE

fire area/zone 11

Ignition Source 480V SHUTDOWN BOARD 2B

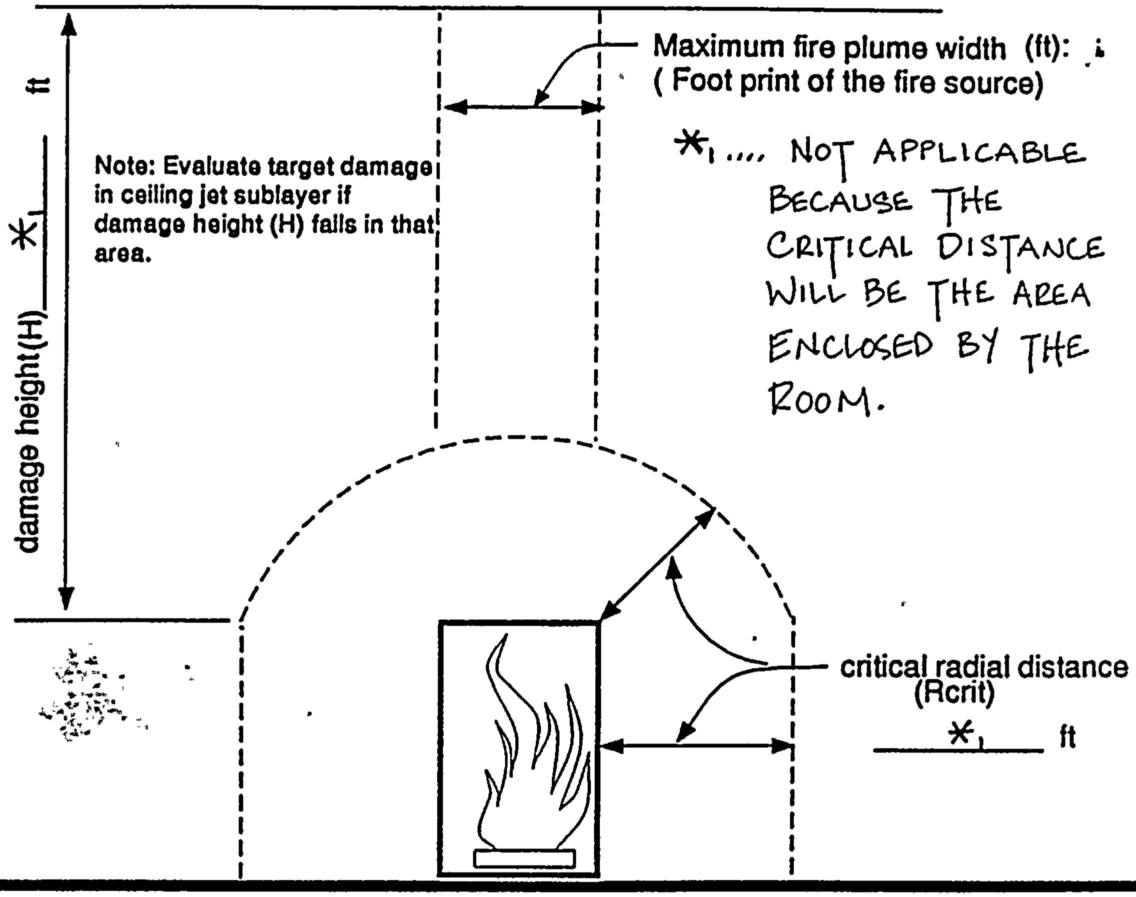
*2 .... CONDUIT/CABLES ASSOCIATED WITH THE 480V SD BD.

cable trays in fire damage zone: NONE

conduits in fire damage zone: 2ES2788-II, 2T205, 2LS382-AII, 2LS778-A2, 2LS433-BII, 2LS787-B2, 2ES3959-2, 2ES345-I, 2ES3958-2, 2ES3928, 2ES3968-2, 2ES3970-2, 2ES3969-2, 2ES3927, *2

FOR CONDUITS WITH PENETRATION NUMBERS SEE PAGE 2

other components in fire damage zone: SEE PAGE 2



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party William L Aldredge Date 7-22-94

Walkdown Second Party Turner J. Howard Date 7-25-94

WALKDOWN SCOPE IDENTIFICATION/FIELD VERIFICATION
FIRE DAMAGE ZONE OF INFLUENCE

FIRE AREA / ZONE : **11**

IGNITION SOURCE : **480V SHUTDOWN BOARD 2B**

CONDUITS IN FIRE DAMAGED ZONE (CONTINUED) :

CONDUITS ASSOCIATED WITH PENETRATION NUMBERS :

S26214212, S26214213 (S26211247), S26214214, S26211865, S26211863, S26214150, S26214151, S26214152, S26214153, S26214220, S26213870, S26214209, S26214210, S26214211, S26215055, S26215057 (JB 4750), S26211231, S26212701.

OTHER COMPONENTS IN THE FIRE DAMAGE ZONE (CONTINUED)

PANELS 25-44A-12 AND 25-44B-12 , PLUS THE FOLLOWING : OVERHEAD LIGHTS, EMERGENCY LIGHTS, BLUE LIGHT, FIRE DETECTOR CIRCUITRY, AND POWER OUTLET RECEPT.

WALKDOWN FIRST PARTY : William L Aldredge DATE : 7-22-94

WALKDOWN SECOND PARTY : Thomas J. Howard DATE : 7-25-94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE

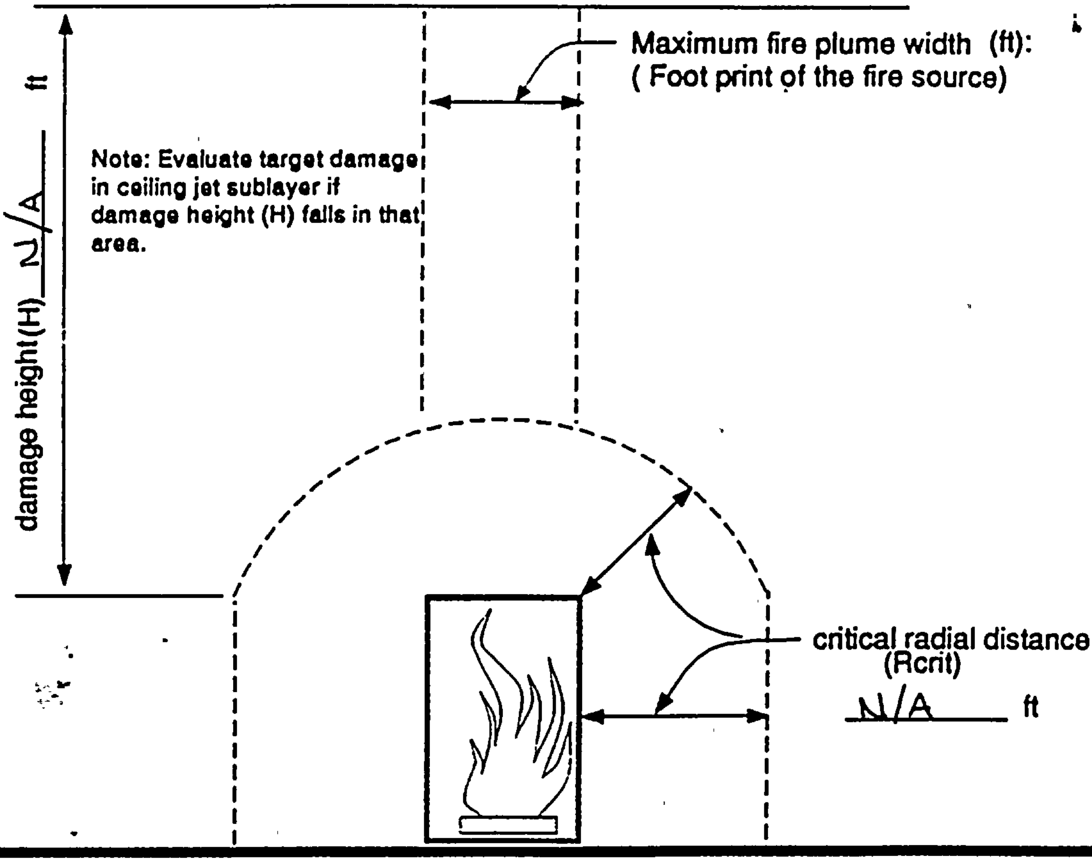
fire area/zone 12 EEB-010-1 PG 59 OF 88

Ignition Source ENGULFING FIRE-SHUTDOWN BD Rm F

cable trays in fire damage zone: 37HCT-ESII (EMPTY)

conduits in fire damage zone: 3ES931-I, 3ES1548-I,
3PC5302, 3ES9384-I, 3B180-B1, 3B188-B3

other components in fire damage zone: SEE PAGE 2



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party Barry Zimmerman Date 8-31-94

Walkdown Second Party David E. Linden Date 8-31-94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION
(CONTINUATION)

1. IGNITION SOURCE :
ENGULFING FIRE- SHUTDOWN BOARD ROOM F

2. CABLE TRAYS IN FIRE ZONE : (CONTINUED)

- SEE PAGE 1-

3. CONDUITS IN FIRE DAMAGED ZONE : (CONTINUED)

- SEE PAGE 1-

4. OTHER COMPONENTS IN THE FIRE DAMAGE ZONE : (CONTINUED)
480V RMOV BD 3B, 250V DC RMOV BD 3B, 3-LPNL-925-0658, 3-LPNL-925-656B, 3-LPNL-925-0541, 3-JBOX-031-9634, I/C BUS B XFRMR, 3-JBOX-253-7161, 3-XFA-253-3B, 3-JBOX-253-7162, 3-JBOX-253-8868, 3-INVT-256-0001, 3-ECAB-231-003A, 3-ECAB-231-003B, 3-LPNL-925-654B, 3-TS-031-7206C, 3-TS-031-7205D, OVERHEAD LIGHTS, EMERGENCY LIGHTS, EVAC LIGHT, FIRE ALARM, SOUND POWERED PHONE, SMOKE DETECTORS, ISOL. INDICATORS

WALKDOWN FIRST PARTY Perry Zimmerman DATE 8-31-94

WALKDOWN SECOND PARTY David E. Linder DATE 9-7-94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE

page 1 of 2

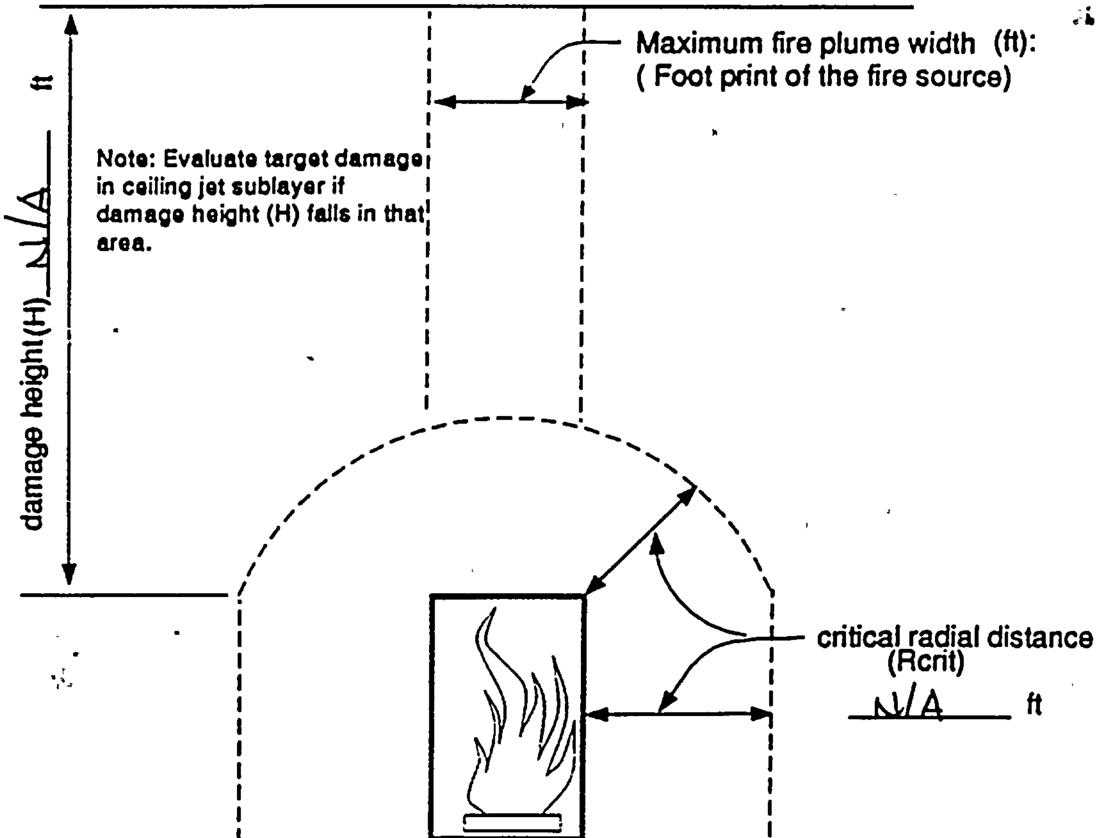
fire area/zone 13 EEB-010-1 PG 61 OF 88

Ignition Source SHUTDOWN BD. ROOM 'E' - ENGULFING FIRE

cable trays in fire damage zone: N/A

conduits in fire damage zone: 3PL6266-II, 3PL6364-II, 3PL769, 3PL5427-II, 3PL5428-II, 3B377-B3, 3B376-B1, 3PL6342-I,

other components in fire damage zone: SEE PAGE 2



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party Barry Zimmerman Date 9-15-94 (CONST. STILL ON-GOING)

Walkdown Second Party David E. [illegible] Date 9-15-94

kktb0/ignition frequency (five)/walkdown.source loc1

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION
(CONTINUATION) page 2 of 2

1. IGNITION SOURCE :
SHUTDOWN BD Room 'E' - ENGULFING FIRE

2. CABLE TRAYS IN FIRE ZONE : (CONTINUED)
N/A

3. CONDUITS IN FIRE DAMAGED ZONE : (CONTINUED)
- SEE PAGE 1 -

4. OTHER COMPONENTS IN THE FIRE DAMAGE ZONE : (CONTINUED)
480V RMOV BD 3A, 250V DC RMOV BD 3A, 3-LPNL-925-656A
3-LPNL-925-0416, 3-LPNL-925-0613, 3-LPNL-925-0539,
3-LPNL-925-0542, 3-TS-031-7206D, 3-TS-031-7205C,
3-INVT-256-0002, 3-JBOX-253-8866, 3-JBOX-253-7159,
3-XFA-253-3A, 3-JBOX-253-7158, I/C BUS A XFRMR, 3-25-32,
3-LPNL-925-654A, OVERHEAD LIGHTS, EMERG. LIGHTS,
EVAC. LIGHTS, SMOKE DET., PHONES AND JACKS.

WALKDOWN FIRST PARTY Perry Zimmerman DATE 9-16-94 (CONT. STILL ON-GOING)
WALKDOWN SECOND PARTY David E. Tinder DATE 9-16-94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE

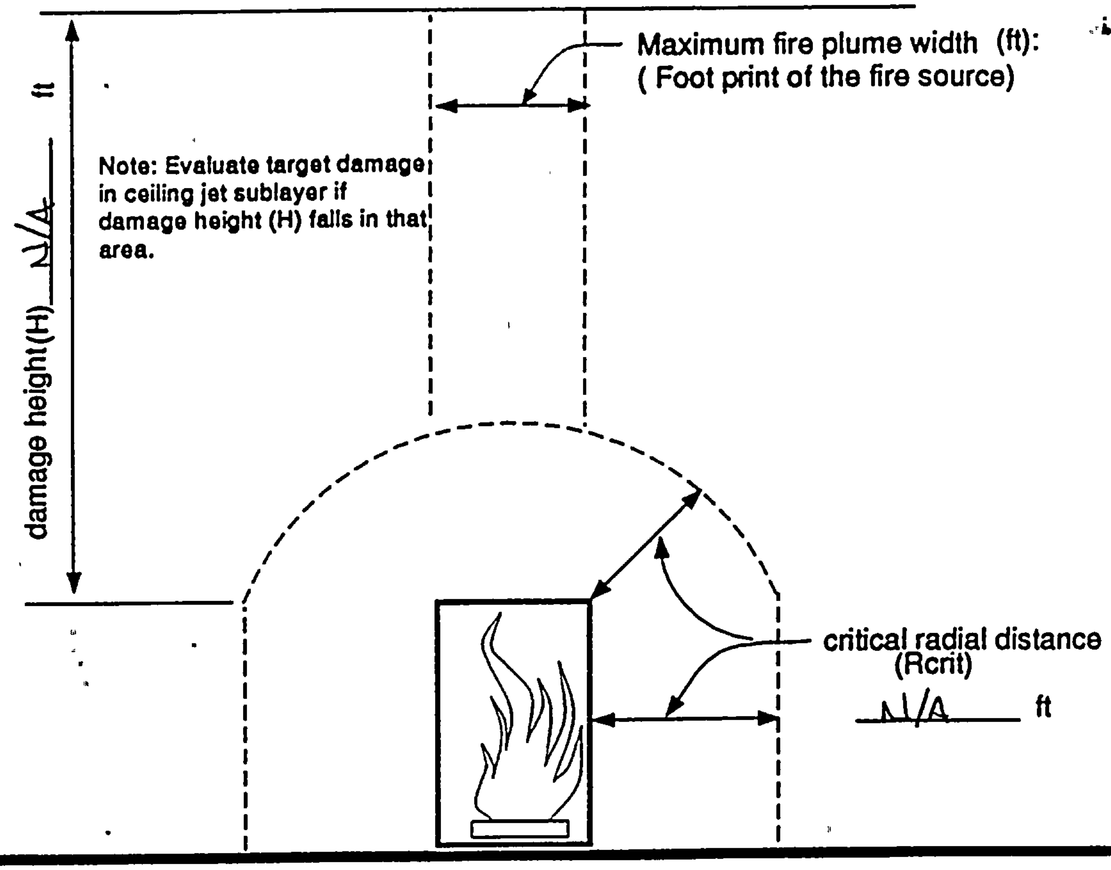
fire area/zone 14 EEB-010-1 PG 63 OF 88

Ignition Source 480V SD BD 3A - ENGULFING FIRE

cable trays in fire damage zone: N/A

conduits in fire damage zone: 3PL393, 3PL385, 3PL704,
3PL392, 3PL386

other components in fire damage zone: SEE PAGE 2



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party Perry Zimmerman Date 8-18-94 (CONSTRUCTION STILL ON-GOING)

Walkdown Second Party Turner Fitzgerald Date 8-18-94

kktb0/ignition frequency (five)/walkdown.source loc1

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION
(CONTINUATION) page 2 of 2

1. IGNITION SOURCE :
480V SD BD 3A

2. CABLE TRAYS IN FIRE ZONE : (CONTINUED)
N/A

3. CONDUITS IN FIRE DAMAGED ZONE : (CONTINUED)
– SEE PAGE 1 –

4. OTHER COMPONENTS IN THE FIRE DAMAGE ZONE : (CONTINUED)
480V SD BD 3A, VR BOX 3DN, VR BOX 3EA, 3-LPNL-925-655A, EVACUATION LIGHT, EMERGENCY LIGHTING, SOUND POWERED PHONE JACKS, SMOKE DETECTORS, FIRE ALARM, OVERHEAD LIGHTS RECEPTICLES, OUTLETS

WALKDOWN FIRST PARTY Barry Zimmerman DATE 8-18-94
WALKDOWN SECOND PARTY Turner J. Howard DATE 8-18-94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE

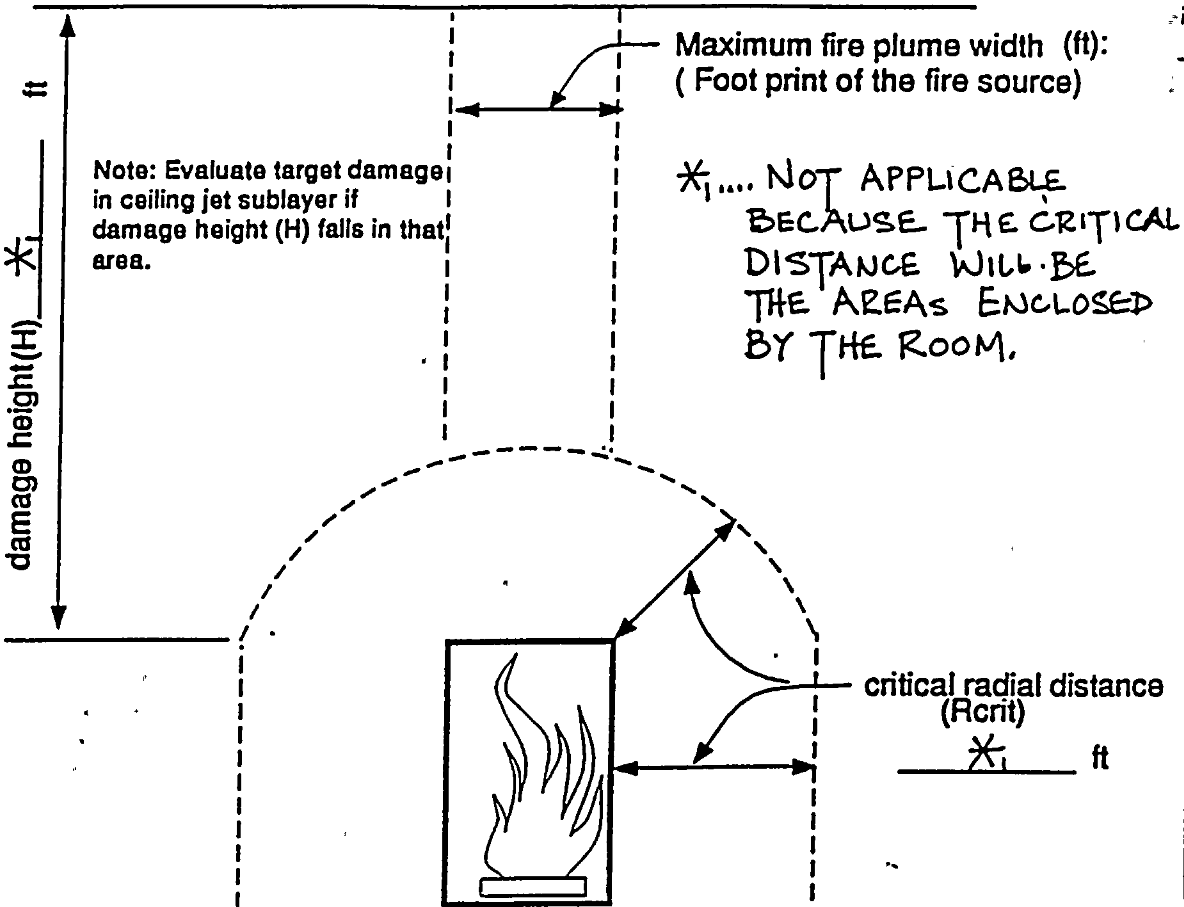
fire area/zone 15

Ignition Source 480V SHUTDOWN BOARD 3B

cable trays in fire damage zone: N/A

conduits in fire damage zone: N/A

other components in fire damage zone: SEE PAGE 2



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party [signature] Date 8-18-94 (CONSTRUCTION STILL ONGOING)

Walkdown Second Party [signature] Date 8-18-94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION
(CONTINUATION) page 2 of 2

1. IGNITION SOURCE : 480V SD BD 3B

2. CABLE TRAYS IN FIRE ZONE : (CONTINUED)

N/A

3. CONDUITS IN FIRE DAMAGED ZONE : (CONTINUED)

N/A

4. OTHER COMPONENTS IN THE FIRE DAMAGE ZONE : (CONTINUED)

480V SD BD 3B, VR BOX 3DA, VR BOX 3EN, 3-LPNL-925-655B, OVERHEAD LIGHTS, EMERG. LIGHTS, EVAC. LIGHT, SMOKE DETECTORS, OUTLETS, SWITCHES, S.P. PHONE JACK, FIRE ALARM

WALKDOWN FIRST PARTY Perry Zimmerman DATE 8-18-94

WALKDOWN SECOND PARTY Turner J. Howard DATE 8-18-94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE

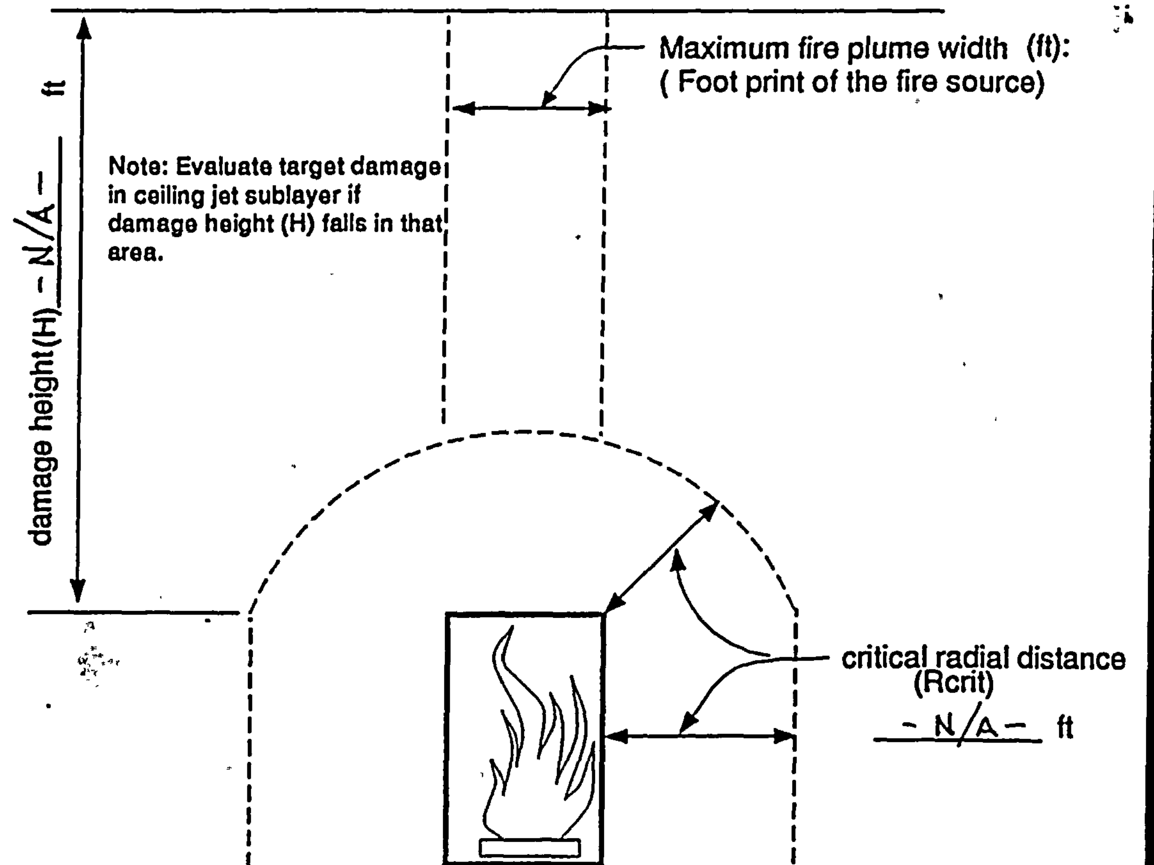
fire area/zone 17

Ignition Source ENGULFING FIRE FOR 250V BATTERY ROOM 1 (250V STATION BATTERY 1)

cable trays in fire damage zone: NONE

conduits in fire damage zone: (EXCLUDING CONDUITS DEDICATED TO EQUIPMENT IN THE ROOM) 2M1-II, 1ES920, AND CONDUITS ASSOCIATED WITH PENETRATIONS: B15933107, B15934095, B15934096, B15934101, B15934102

other components in fire damage zone: SEE PAGE 2



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party William L Aldredge Date 8-31-94

Walkdown Second Party Thomas J. Howard Date 8-31-94

kktb0/ignition frequency (five)/walkdown.source loc1

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION
(CONTINUATION)

1. IGNITION SOURCE : 250V BATTERY ROOM 1
(250V STATION BATTERY 1)

2. CABLE TRAYS IN FIRE ZONE : (CONTINUED)

N/A

3. CONDUITS IN FIRE DAMAGED ZONE : (CONTINUED)

SEE PAGE 1

4. OTHER COMPONENTS IN THE FIRE DAMAGE ZONE : (CONTINUED)

- 250V MAIN BATT. 1
- ± 24V NEUTRON MONITORING BATT. (UNIT 1 - CHANNEL A)
- ± 24V NEUTRON MONITORING BATT. (UNIT 1 - CHANNEL B)
- 48V ANNUNCIATOR BATT. A
- OVERHEAD LIGHTS, SMOKE DETECTORS, FIRE ALARM, SWITCHES, SECURITY DETECTION SYS, RECEPTICLES

WALKDOWN FIRST PARTY Thomas J. Howard DATE 9-2-94

WALKDOWN SECOND PARTY Perry J. Zimmerman DATE 9-2-94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE

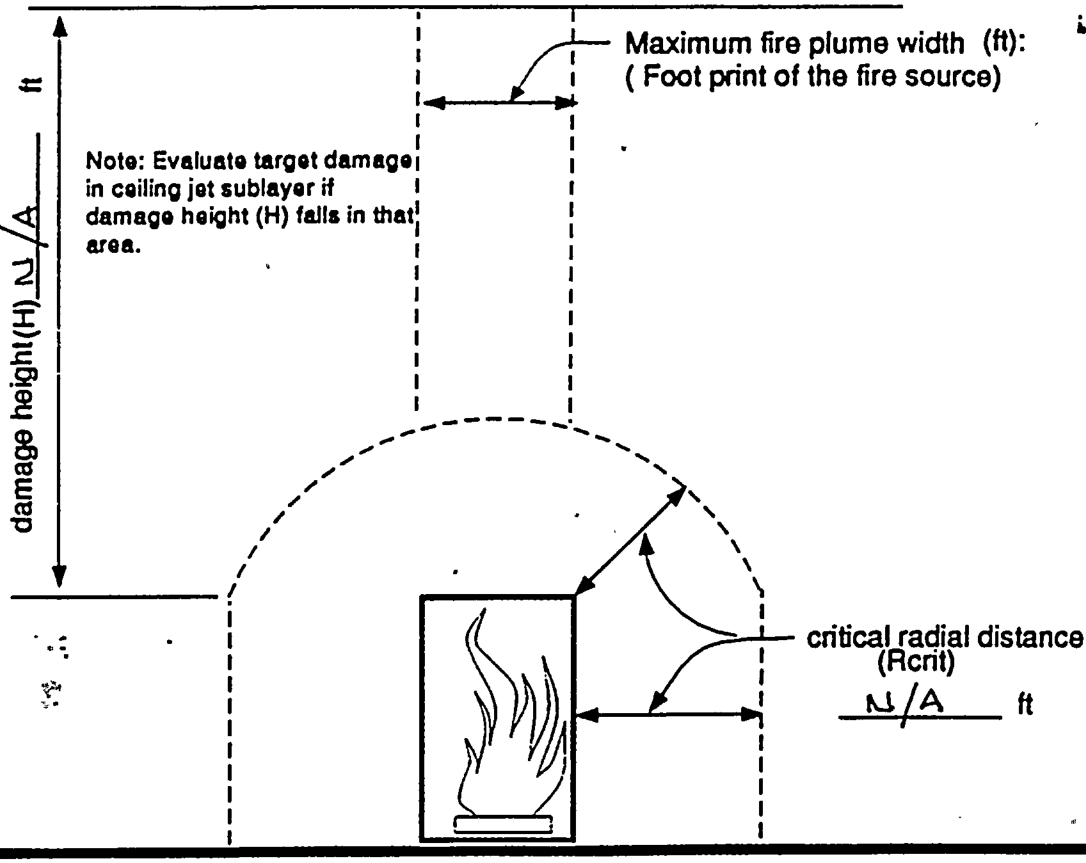
fire area/zone 17 EEB-010-1 PG 69 OF 88

Ignition Source ENGULFING FIRE-U1 BATT BD ROOM (250V BATTERY BD 1)

cable trays in fire damage zone: N/A

conduits in fire damage zone: 1ES920, 1PC909-II, 1PC908-II, PL429, PL427, 1A862, UNTAGGED CONDUIT

other components in fire damage zone: SEE PAGE 2



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party Perry Zimmerman Date 9-15-94

Walkdown Second Party David E. Fowler Date 9.15.94

kktb0/ignition frequency (five)/walkdown.source loc1

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION
(CONTINUATION)
page 2 of 2

1. IGNITION SOURCE :
ENGULFING FIRE - U1 BATTERY BD Rm

2. CABLE TRAYS IN FIRE ZONE : (CONTINUED)

N/A

3. CONDUITS IN FIRE DAMAGED ZONE : (CONTINUED)

— SEE PAGE 1 —

4. OTHER COMPONENTS IN THE FIRE DAMAGE ZONE : (CONTINUED)
250V DC BATTERY BD 1, 250V BATTERY CHRGR 1, RPS CONTROL PANEL-CKT PROTECTOR 1A1, 1A2, 1B1, 1B2, 1C1 AND 1C2, 250V BATT CHRGR 1 OUTPUT XFER SW., 1-JBOX-253-6457, 48V ANN. BATT CHRGR A, 1-JBOX-253-6460, FDS-1 RPS BUS ALT. SOURCE XFRMR DISC. SWITCH, TRP-1 RPS BUS ALT SOURCE REG XFRMR, TUP-1 UNIT PREF XFRMR, UNIT 1 RPS A&B ALT SOURCE (PREF XFRMR SAFETY SW), ±24V NEUTRON MON. BATT. CHRGR A1-1, A2-1, B1-1 AND B2-2. OVERHEAD LIGHTS, EMER. LIGHTS, SMOKE DETECTORS, FIRE ALARM, PHONE, SOUND POWERED PHONE JACK, SECURITY SYSTEM, RECEPTICLES.

WALKDOWN FIRST PARTY Perry Zimmerman DATE 9-15-94

WALKDOWN SECOND PARTY David E. [illegible] DATE 9-15-94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE

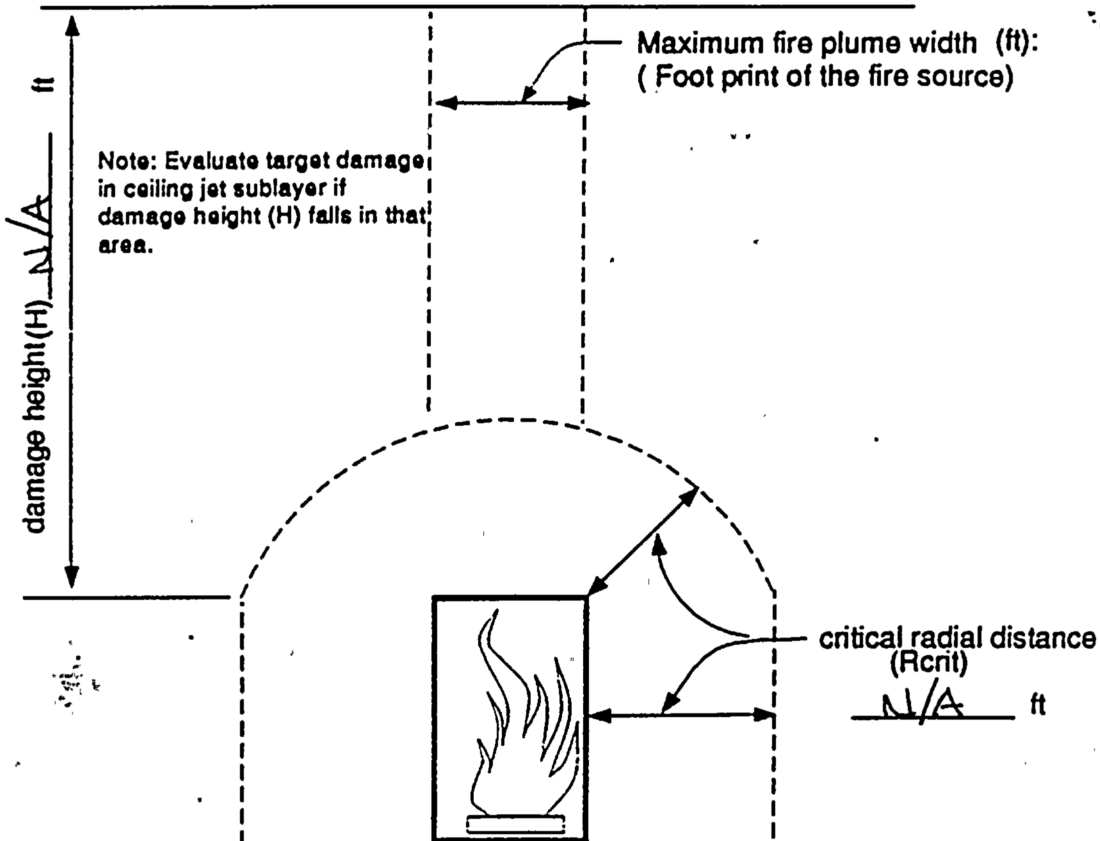
fire area/zone 18 EEB-010-1 PG 71 OF 88

Ignition Source ENGULFING FIRE- 112 BATTERY BD ROOM (250V BATTERY BD 2)

cable trays in fire damage zone: N/A

conduits in fire damage zone: 2RP40-IIA, 2ES3401-II, 2RP41-IIB, 2ES3422-II, 2PC917-I, 2PC918-I, 2PC943-I, 2ES2111, 2PC932-I, C119, 2SG2065, 2SG2067, 2LS415, EE3831, EE3830, CONDUITS ASSOCIATED WITH PEN., B25932791, B25932795, B25933143 (CONT'D ON PG 2)

other components in fire damage zone: SEE PAGE 2



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party [signature] Date 9-15-94

Walkdown Second Party [signature] Date 9-15-94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION
(CONTINUATION) page 2 of 2

1. IGNITION SOURCE :
ENGULFING FIRE-U2 BATTERY BD Rm

2. CABLE TRAYS IN FIRE ZONE : (CONTINUED)
N/A

3. CONDUITS IN FIRE DAMAGED ZONE : (CONTINUED)
B25933357, B25933358
~~SEE PAGE 1~~ AB 9-26-94

4. OTHER COMPONENTS IN THE FIRE DAMAGE ZONE : (CONTINUED)
250V DC BATTERY BD 2, RPS CONTROL PNL, CKT PROTECTOR 2A1, 2A2, 2B1, 2B2, 2C1 AND 2C2, 250V DC BATTERY CHGR 2A, 120V AC PLANT NONPREF. DIST., PLANT NONPREF. AUTOTRANSFER SW., 120V AC U2 PREF. SYS MIMIC BUS, UNIT PFD SYNC & CTRL PNL & 120V AC U2 PREF DIST., 2-JBOX-253-7194, 2-JBOX-253-7197, U2 110V AC BACKUP SPLY TO RPS DISC. SW., ± 24V NEUTRON MONITORING BATT CHRG. A1-2, A2-2, B1-2 & B2-2, EMERG. LIGHTS, OVERHEAD LIGHTS, SOUND POWERED PHONE JACKS, PHONE, SECURITY SYS., FIRE ALARM, SMOKE DET.,

WALKDOWN FIRST PARTY Barry Zimmerman DATE 9-15-94

WALKDOWN SECOND PARTY David E. Linden DATE 9-15-94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE

page 1 of 2

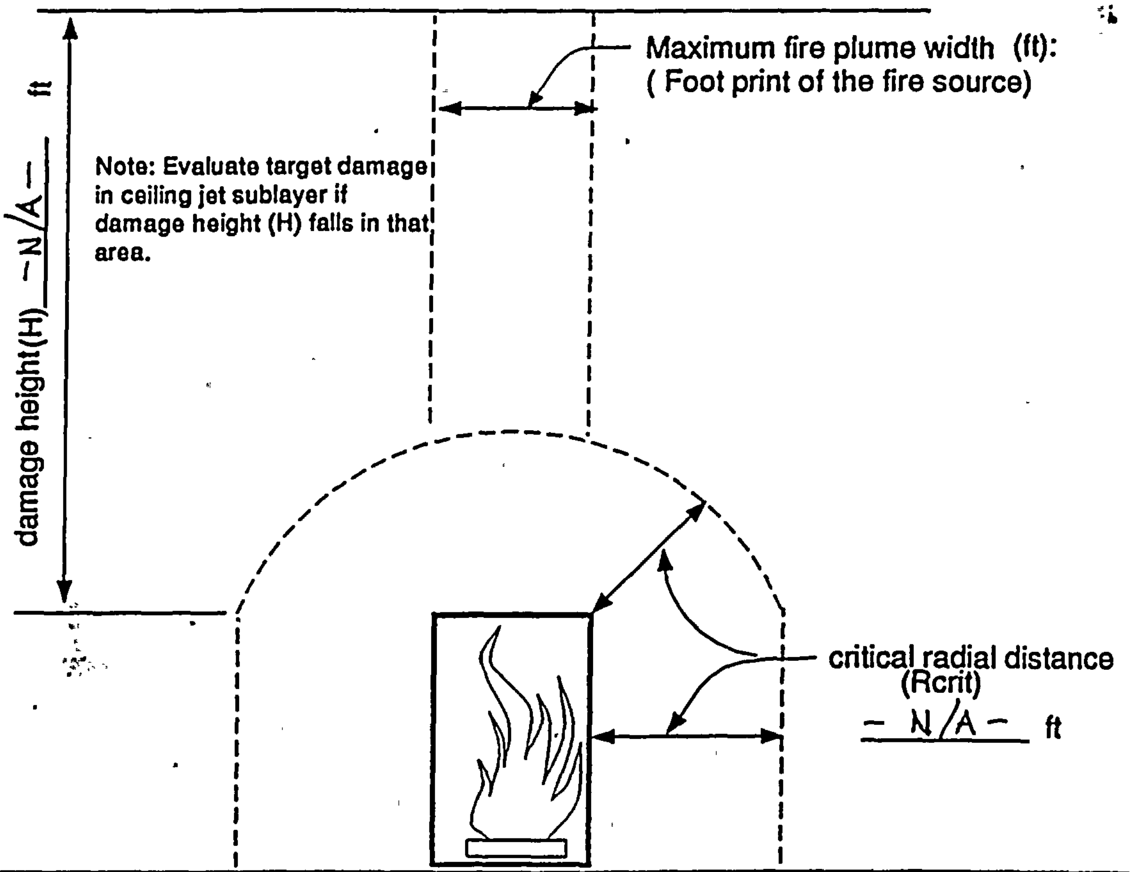
fire area/zone 18 EEB-010-1 PG 73 OF 88

Ignition Source ENGULFING FIRE FOR BATTERY ROOM 2 (250V STATION BATTERY 2)

cable trays in fire damage zone: NONE

conduits in fire damage zone: (EXCLUDING CONDUITS DEDICATED TO EQUIPMENT IN THE ROOM) 3B193, 2M327, 2M7, 2R885, FE7774, 2ES174-I, 2M12-I AND THE CONDUITS ASSOCIATED WITH PENETRATIONS: B25935198, B25934170, B25934167, B25934171, B25934173, B25934174.

other components in fire damage zone: SEE PAGE 2



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party William L Aldredge Date 8-31-94

Walkdown Second Party Thomas J. Howard Date 8-31-94

kktb0/ignition frequency (live)/walkdown.source loc1

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION
(CONTINUATION)
page 2 of 2

1. IGNITION SOURCE : BATTERY ROOM 2
(250 V STATION BATTERY 2)

2. CABLE TRAYS IN FIRE ZONE : (CONTINUED)

N/A

3. CONDUITS IN FIRE DAMAGED ZONE : (CONTINUED)

SEE PAGE 1

4. OTHER COMPONENTS IN THE FIRE DAMAGE ZONE : (CONTINUED)

- 250V UNIT BATT. NO. 2 (0-BATA-248-0002)
- ± 24V NEUTRON MONITORING BATT. (UNIT 2 - CHANNEL A)
- ± 24V NEUTRON MONITORING BATT. (UNIT 2 - CHANNEL B)
- OVERHEAD LIGHTS, SMOKE DETECTORS, FIRE ALARM, SWITCHES, SECURITY DETECTION SYS., RECEPTICLES

WALKDOWN FIRST PARTY [signature] DATE 9-2-94

WALKDOWN SECOND PARTY Barry J. Zimmerman DATE 9-2-94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE page 1 of 2

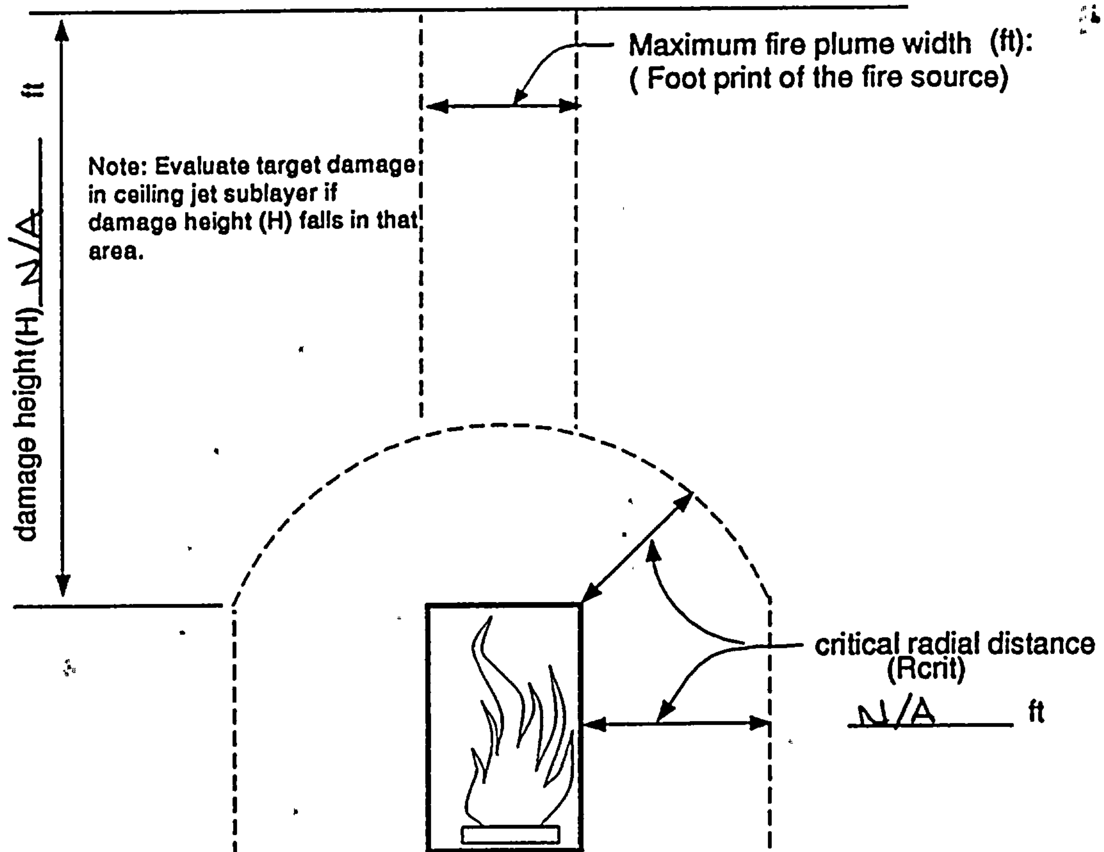
fire area/zone 19 EEB-010-1 PG 75 OF 88

Ignition Source ENGULFING FIRE FOR 250V BATTERY ROOM 3 (250V STATION BATTERY 3)

cable trays in fire damage zone: N/A

conduits in fire damage zone: 3ES9474-IS1, 3RP1365-A, 3RP1435-B, 3RP1445-B, 3A862, 3RP1390-B

other components in fire damage zone: SEE PAGE 2



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party Turner J. Howard Date 9-2-94

Walkdown Second Party Barry Zimmerman Date 9-2-94

kktb0/ignition frequency (five)/walkdown.source loc1

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION
(CONTINUATION) page 2 of 2

1. IGNITION SOURCE : 250V BATTERY ROOM 3
(250V STATION BATTERY 3)

2. CABLE TRAYS IN FIRE ZONE : (CONTINUED)

N/A

3. CONDUITS IN FIRE DAMAGED ZONE : (CONTINUED)
SEE PAGE 1

4. OTHER COMPONENTS IN THE FIRE DAMAGE ZONE : (CONTINUED)

- 250V UNIT BATT. NO. 3 (0-BATA-248-0003)
- ± 24V NEUTRON MONITORING BATT. (UNIT 3-CHANNEL A)
- ± 24V NEUTRON MONITORING BATT. (UNIT 3 - CHANNEL B) THIS IDENTIFICATION DERIVED FROM SET ROUTE AND CONDUITS 3NM1282, 3NM1281, AND 3NM1280.
- JB4543 (CARD READER SYSTEM)
- OVERHEAD LIGHTS, SMOKE DET., FIRE ALARMS

WALKDOWN FIRST PARTY Turner L. Howard DATE 9-2-94

WALKDOWN SECOND PARTY Perry Zimmerman DATE 9-2-94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE

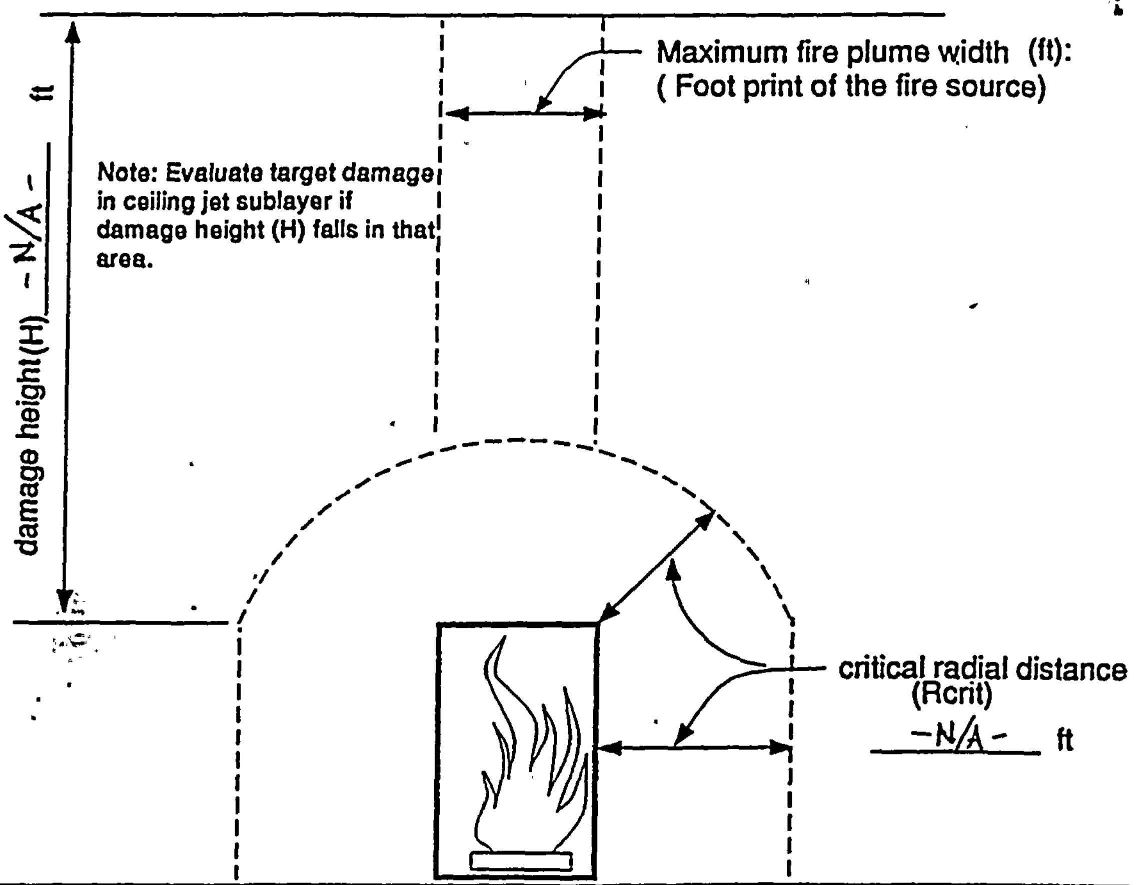
fire area/zone 19

Ignition Source ENGULFING FIRE FOR BATTERY BOARD ROOM 3 (250V BATTERY BOARD 3)

cable trays in fire damage zone: NONE

conduits in fire damage zone: (EXCLUDING CONDUITS DEDICATED TO THE EQUIPMENT IN THE ROOM)
K909, K2922, 3A864, 3ES9166-II, 3RP1808-IIA, 3RP1820-IIB, AND 2ES3440-II

other components in fire damage zone: SEE PAGE 2



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party William L Aldredge Date 8-31-94

Walkdown Second Party Turner A Howard Date 8-31-94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION
(CONTINUATION) page 2 of 2

1. IGNITION SOURCE :
BATTERY BOARD ROOM 3 (250V BATTERY BOARD 3)

2. CABLE TRAYS IN FIRE ZONE : (CONTINUED)
- N/A -

3. CONDUITS IN FIRE DAMAGED ZONE : (CONTINUED)
THERE ARE VARIOUS MISC. SMALL CONDUITS ROUTED ACROSS THE CEILING WITHOUT IDENTIFICATION TAGS. THESE APPEAR TO BE FIRE DETECTOR CIRCUITRY AND SIMILAR EQUIPMENT.

4. OTHER COMPONENTS IN THE FIRE DAMAGE ZONE : (CONTINUED)
- RPS CIRCUIT PROTECTORS (6 TOTAL)
- UNIT 3 PREF. AC BUS TRANSFORMER (37.5 KVA)
- RPS ALT SUPPLY TRANSFER (TRP-3)
- ±24VDC NEUTRON MONITORING CHARGERS B1-3, B2-3, A1-3 & A2-3
- 48V ANN. BATTERY CHARGER B
- 250V BATTERY BOARD 3
- 250V BATTERY CHARGER 3 TRANSFER SW's.
- OVERHEAD LIGHTS
- TELEPHONE
- FIRE DETECTORS
- 3-JB-3-6494 (CONDUITS 3RP1808-IIA, 3RP1820-IIA & 2ES3440-II)

WALKDOWN FIRST PARTY William L Aldredge DATE 8-31-94

WALKDOWN SECOND PARTY Turner A. Howard DATE 8-31-94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE

page 1 of 2

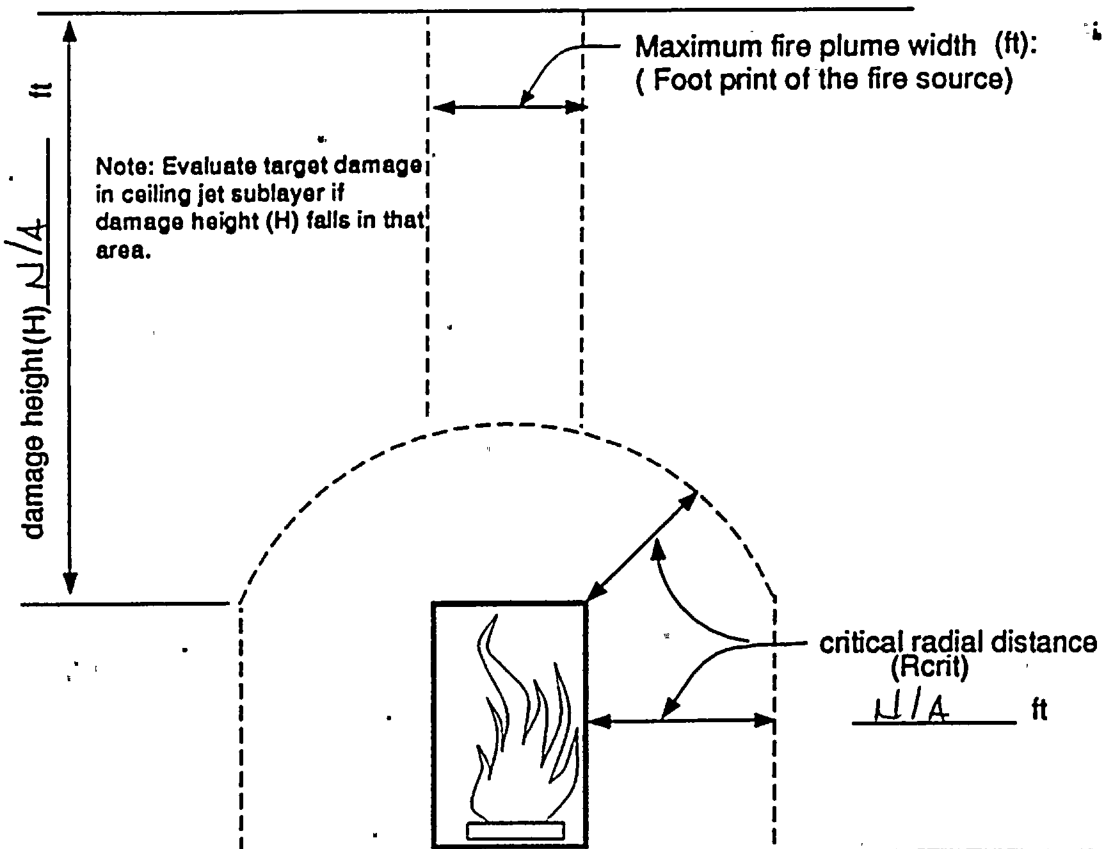
fire area/zone 22 EEB-010-1 PG 79 OF 88

Ignition Source ENGULFING AREA - 4KV BD Rm 3EA

cable trays in fire damage zone: N/A

conduits in fire damage zone: 3SP0040, 3PL621-I, 3PL620-I, EE7575, EE7576, PL4848, PL4847, PL4844, PL4849, PL4845, PL4846, 3PP900, 3PP901, 3PP902,
DEF PB 9-21-94 ~~3PP459~~, PP460, 3PP733, PP465, PP466, K1070, 3A1540, 3ES2308, COAX CABLE M2010/M2054, CONT. ON PAGE 2

other components in fire damage zone: SEE PAGE 2



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party [signature] Date 8-31-94

Walkdown Second Party [signature] Date 9-1-94

kktb0/ignition frequency (five)/walkdown.source loc1

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION
(CONTINUATION) page 2 of 2

1. IGNITION SOURCE :
ENGULFING FIRE - 4KV BD ROOM 3EA

2. CABLE TRAYS IN FIRE ZONE : (CONTINUED)
N/A

3. CONDUITS IN FIRE DAMAGED ZONE : (CONTINUED)
3B247* (@ PEN D35832078 TO UNLABLED FLOOR PEN)

* CONDUIT WAS NOT TAGGED AT EITHER PEN. IN THE 4KV BD RM 3EA, IT WAS IDENTIFIED BY REVIEWING THE OPPOSITE SIDE PEN (D35834078) IN THE DIESEL AUX ROOM WHERE IT WAS TAGGED AND AT THE 250V DC BATT BD 3EB.

4. OTHER COMPONENTS IN THE FIRE DAMAGE ZONE : (CONTINUED)
4KV SD BD 3EA, JB4094, JB4565, JB4400, JB4401, JB4402, JB4403, 3-CMOD-039-0090A, 3-JBOX-039-9605, 3-JBOX-039-9606, SMOKE DETECTORS, SOUND POWERED PHONE/JACK, FIRE ALARM, OVERHEAD LIGHTS, EMERG. LIGHTS, BELL

WALKDOWN FIRST PARTY Perry Zimmerman DATE 8-31-94

WALKDOWN SECOND PARTY David E. Findley DATE 9.1.94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE

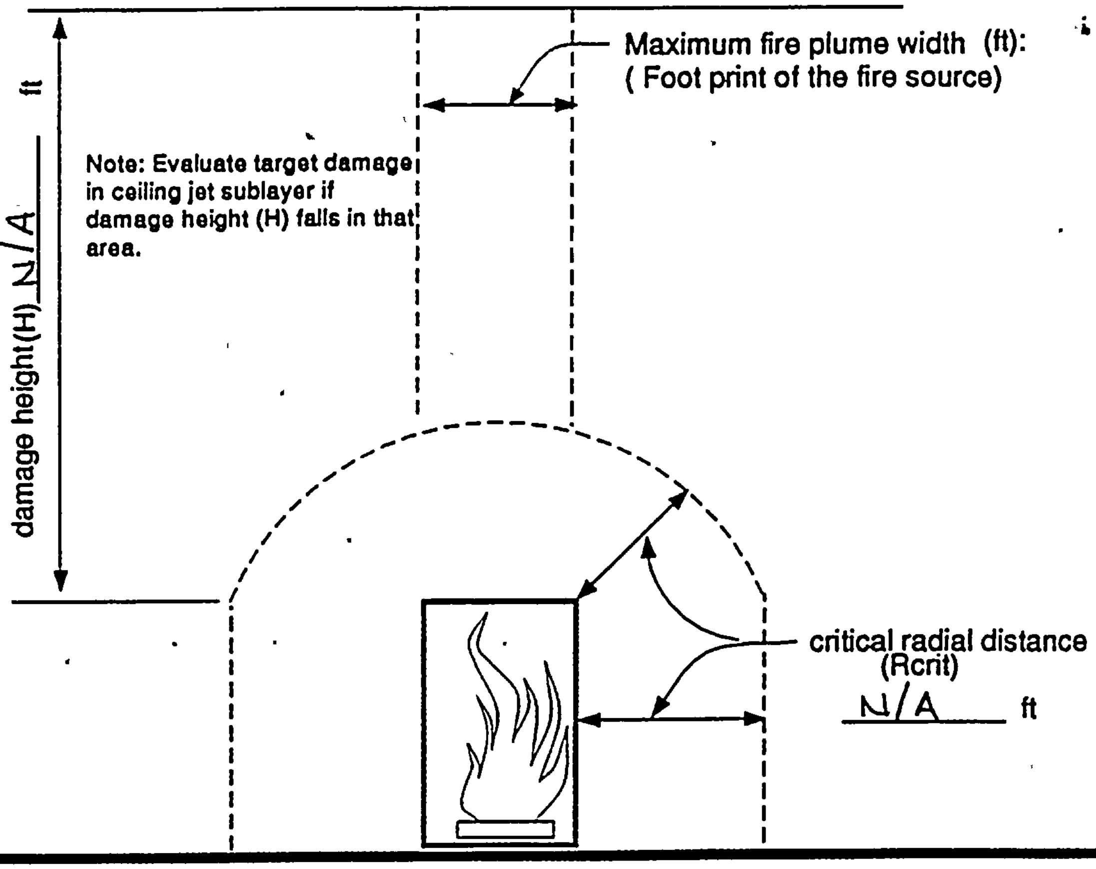
fire area/zone 22 EEB-010-1 PG 81 OF 88

Ignition Source ENGULFING FIRE - 4KV BD RM 3EB

cable trays in fire damage zone: CE-ESI, DL, EAJ, KKU-ESI, (CONT ON PG2)

conduits in fire damage zone: 3ES2057, 3ES2077, MC566, PP454-ID, 3PP680, 3PP681, 3PL661, 3ES1580, 3PP634 (TWO UNTAGGED), 3B200, 3A3882, 3ES1540, 3PP739, PP459-IA, PP460-IA, PL4846, 3A3930, 3PP914, PL410, PL415, PL421, PL422, ES86, 3PP779, 3PP959, (CONT. ON PAGE 2)

other components in fire damage zone: SEE PAGE 2



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party Perry Zimmerman Date 8-31-94

Walkdown Second Party David E. [illegible] Date 8-31-94

kktb0/ignition frequency (five)/walkdown.source loc1

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION
(CONTINUATION) page 2 of 2

1. IGNITION SOURCE :
ENGULFING FIRE - 4KV BD Rm 3EB

2. CABLE TRAYS IN FIRE ZONE : (CONTINUED)

VAJ, KKS; KKV-ESI, BAK

3. CONDUITS IN FIRE DAMAGED ZONE : (CONTINUED)
PL4846, PL4849, PL4845, PL4848, PL4847, PL4844

4. OTHER COMPONENTS IN THE FIRE DAMAGE ZONE : (CONTINUED)
4KV SD BD 3EB, JB5778, JB4095, JB4566
EMERG. LIGHTS, OVERHEAD LIGHTS, SMOKE DETECTORS, SWITCHES,
OUTLETS, SOUND POWERED PHONE JACKS.

WALKDOWN FIRST PARTY Perry Zimmerman DATE 8-31-94
WALKDOWN SECOND PARTY David E. Tinder DATE 8-31-94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE

page 1 of 2

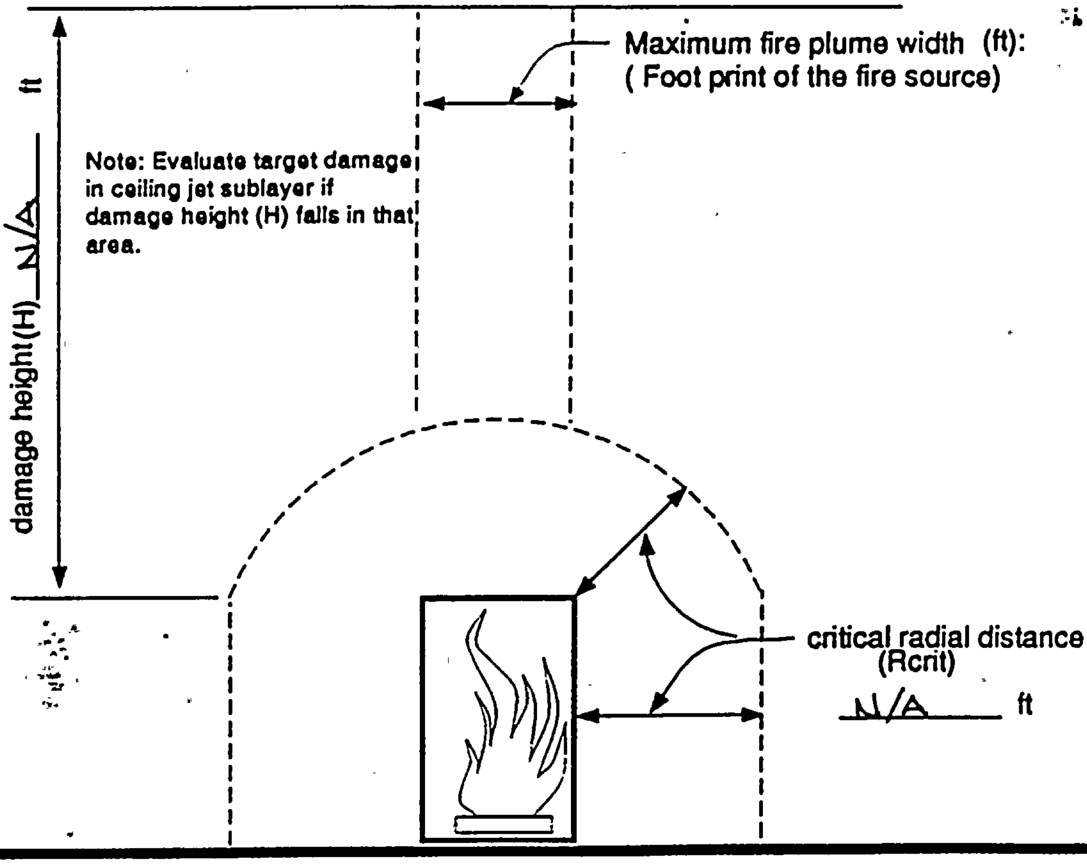
fire area/zone 23 EEB-010-1 PG 83 OF 88

Ignition Source ENGULFING FIRE- 4KV BD RM 3EC

cable trays in fire damage zone: N/A

conduits in fire damage zone: 3M237, 3M417, 3ES4719, 3B180-B1, 3ES4808, 3A1550, UNTAGGED CONDUIT FROM CEILING TO FLOOR, UNTAGGED FROM CEILING TO PEN. DS5831989 TO TRAY WE (A 1" UNTAGGED CONDUIT 'TEES' OF THIS CONDUIT & TERMINATES @ LTG. PNL LC45).

other components in fire damage zone: SEE PAGE 2



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party Barry Zimmerman Date 8-31-94

Walkdown Second Party David E. Fincher Date 8.31.94

kktb0/ignition frequency (five)/walkdown.source loc1

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION
(CONTINUATION)
page 2 of 2

1. IGNITION SOURCE :
ENGULFING FIRE - 4KV BD RM 3EC

2. CABLE TRAYS IN FIRE ZONE : (CONTINUED)
N/A

3. CONDUITS IN FIRE DAMAGED ZONE : (CONTINUED)
- SEE PAGE 1 -

4. OTHER COMPONENTS IN THE FIRE DAMAGE ZONE : (CONTINUED)
4KV SD BD 3EC, EMERGENCY LIGHTING PNL LD-5, LIGHTING CAB. LC-45, JB4096, JB4567, 3-JBOX-026-9603, 3-JBOX-039-9609, 3-JBOX-039-9785, JB4404, JB4405, JB4406, JB4407, JB4335, OVERHEAD LIGHTS, SOUND POWERED PHONE/JACK, EMERG. LIGHTS, SMOKE DETECTORS, FIRE ALARM, BELL

WALKDOWN FIRST PARTY Perry Zimmerman DATE 8-31-94
WALKDOWN SECOND PARTY David E. Tucker DATE 8-31-94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE

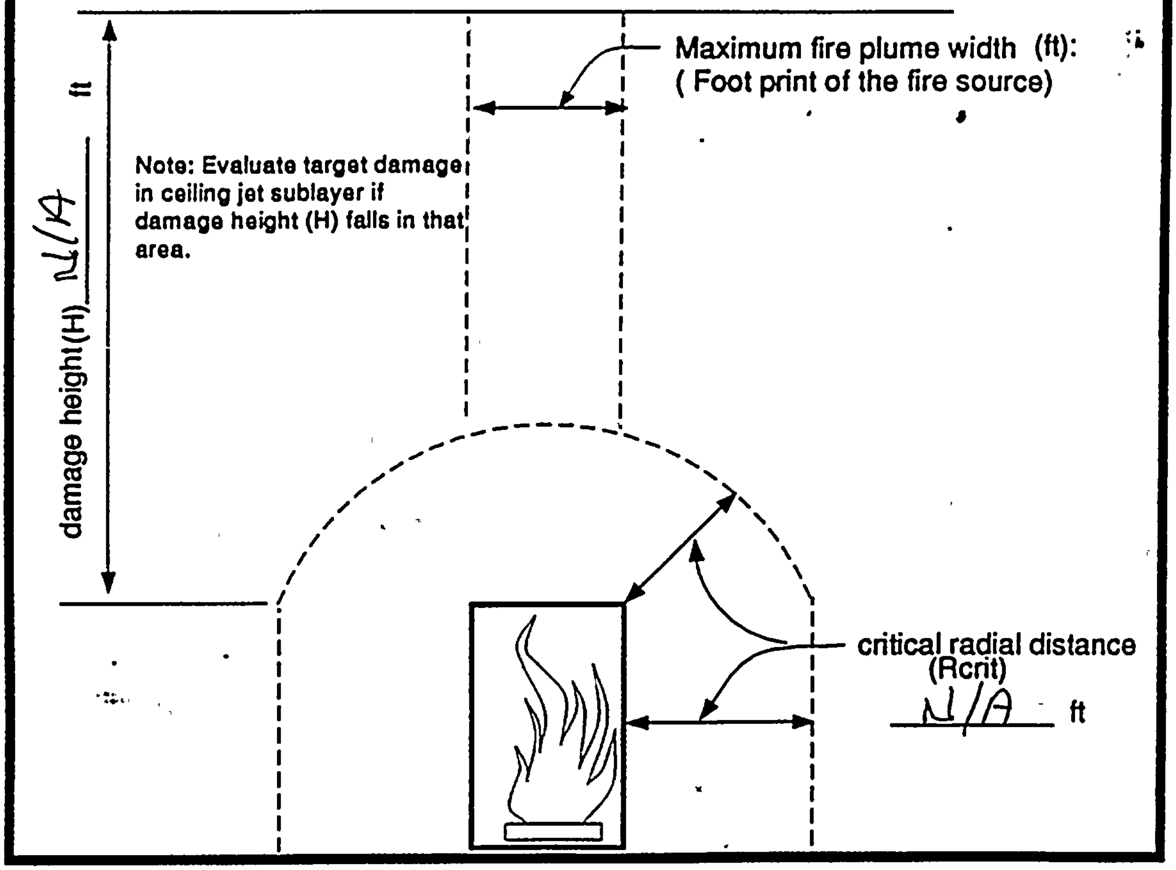
fire area/zone 23 EEB-010-1 PG 85 OF 88

Ignition Source 4KV SHUTDOWN BOARD 3ED - ENGULFING FIRE

cable trays in fire damage zone: DJ-ESIII, DL-ESII, EAJ, KKT-ESII, VAJ,

conduits in fire damage zone: 3ES4049, 3PP851, 3ES4080-II, 3A3872, 3A3958, 3PP687, 3PP686, 3PL5428-II, 3PL5427-II, 3PL1925, 3ES4578, 3PP9858, 3PP471-IIC, 3PP472-IIC, 3PP920, 3PP921, 3PP922, 3PP857, LS

other components in fire damage zone: CON'T ON PAGE 2



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party David E. Fischer Date 9-1-94

Walkdown Second Party Jerry Zimmerman Date 9-1-94

kktb0/ignition frequency (five)/walkdown.source loc1

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION
(CONTINUATION) page 2 of 2

1. IGNITION SOURCE : 4KV SHUTDOWN BOARD 3ED-ENGULFING FIRE

2. CABLE TRAYS IN FIRE ZONE : (CONTINUED)

EAG, KKK, KKX-ESII,

3. CONDUITS IN FIRE DAMAGED ZONE : (CONTINUED)

UNMARKED CONDUIT TO TRAY KKK AND TRAY DL

4. OTHER COMPONENTS IN THE FIRE DAMAGE ZONE : (CONTINUED)

JB 4292

JB 4568

JB 5777

JB 4229

5 KV METALCLAD SWGR PANEL

OVERHEAD LIGHTS, SMOKE DETECTORS, PHONE, SOUND POWERED PHONE/JACK, BELL, FIRE ALARM, SPEAKER, RECEPTICLES, EMERG. LIGHTS, SWITCHES

WALKDOWN FIRST PARTY David E. Lindner DATE 9-1-94

WALKDOWN SECOND PARTY Terry Zimmerman DATE 9-1-94

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION

FIRE DAMAGE ZONE OF INFLUENCE

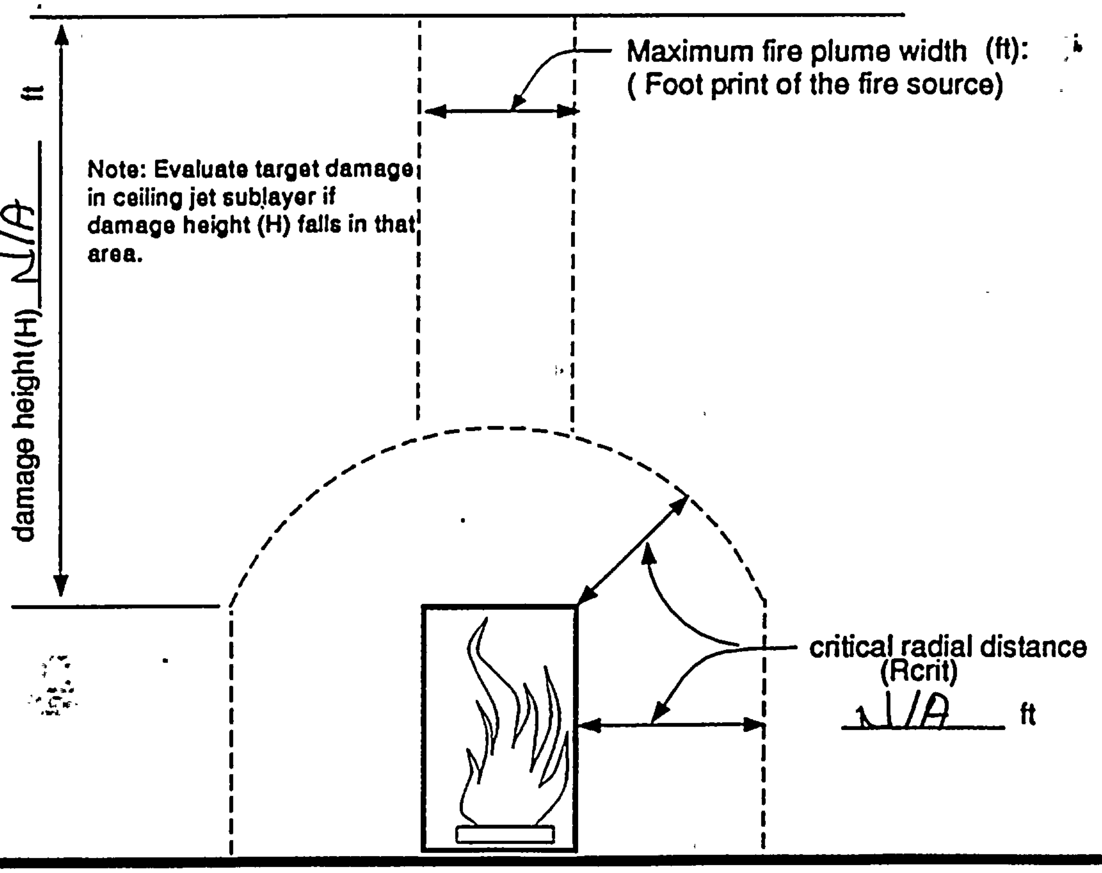
fire area/zone 24 EEB-010-1 PG 87 OF 88

Ignition Source 4 KV BUS TIE BOARD ROOM - ENGULFING FIRE

cable trays in fire damage zone: EAJ, DL, VAJ, DH, DG, CE, KKT, KKU

conduits in fire damage zone: 3PL5427-II, 3K80, 3PP285, 3A1540, 3ES1560, 3ES2308, 3ES4808, 3A1550, 3PP858, SP0040 (SPARE)

other components in fire damage zone: CON'T ON PAGE 2



Use additional sheets if necessary to identify components in the zone of influence

Walkdown First Party Perry Zimmerman Date 9-1-94

Walkdown Second Party David E. Sinclair Date 9-1-94

kktb0/ignition frequency (five)/walkdown.source loc1

WALKDOWN SCOPE IDENTIFICATION / FIELD VERIFICATION
(CONTINUATION) page 2 of 2

1. IGNITION SOURCE :
ENGULFING FIRE - 4KV BUS TIE BOARD ROOM

2. CABLE TRAYS IN FIRE ZONE : (CONTINUED)

- SEE PAGE 1 -

3. CONDUITS IN FIRE DAMAGED ZONE : (CONTINUED)

- SEE PAGE 1 -

4. OTHER COMPONENTS IN THE FIRE DAMAGE ZONE : (CONTINUED)

4KV BUS TIE BOARD
OVERHEAD LIGHTS, SMOKE DETECTORS, PHONE, BELL, EMERG. LIGHT, RECEPTICLE, SOUND POWERED PHONE JACK.

WALKDOWN FIRST PARTY Perry Zimmerman DATE 9-1-94

WALKDOWN SECOND PARTY David E. Lincoln DATE 9-1-94

D.3 SAMPLE SHEETS FROM THE EVALUATION OF POTENTIALLY DISABLED EQUIPMENT

EQUIPMENT POTENTIALLY DISABLED BY THE IGNITION SOURCE

1. IGNITION SOURCE : 240V LIGHTING BOARD 2B PG 1 OF 1

2. FIRE DAMAGE ZONE COMPONENTS :

- 240V LIGHTING BOARD 2B
- 2FE565
- 2FE564
- 2FE129 (**)
- 2FE130 (**)
- 2ES3925-II (*)
- UTILITY OUTLET RECPT.
- COMMUNICATION PHONE JACK

3. ASSOCIATED EQUIPMENT WHICH MAY BE DISABLED :

- 240V LIGHTING BD. 2B :
  - EMER THROW-OVER FOR LD1
  - LTG. CABS. LC110, LC111, LC210, LC211
  - LTG. CABS. LD-1, LD-2
  - EMER THROW-OVER FOR LD2
  - LTG. CABS. LC310, LC311
  - LTG. CAB. LD-3
  - EMER. THROW-OVER FOR LD3
- 2FE565: 2FE321: FIRE DET. SIG.
  2FE330: FIRE DET. IND.
- 2FE564: 2FE320: FIRE DET. SIG.
  2FE330: FIRE DET. IND.
- FE129 (**): FIRE PUMP START CONTROL
- FE130 (**): FIRE PUMP START CONTROL
- 2ES3925-II (*): 480V RMOV BD 2E, NOR SUPPLY
- UTILITY OUTLET RECPT.
- COMMUNICATION PHONE JACK

4. AUTOMATIC UNIT TRIP INITIATED : [ ] YES, [✓] NO

5. COMMENTS :

(*) LOCATED JUST OUTSIDE THE CRITICAL DISTANCE.

(**) AFTER REVIEWING DWGS. 2-45N814-4, 2-45N812-4, 45N816-2 AND 45C800-FE-6, IT WAS DETERMINED THAT CONDUIT/CABLES LISTED 2FE129 & 2FE130 ARE ACTUALLY FE129 & FE130.

EQUIPMENT POTENTIALLY DISABLED BY THE IGNITION SOURCE

1. IGNITION SOURCE : 480V SHUTDOWN BOARD 2A

2. FIRE DAMAGE ZONE COMPONENTS :

- 480V SHUTDOWN BOARD 2A.
- PANELS 25-44A-11 AND 25-44B-11.
- 2A960, 2LS757-A1, 2LS347-B1, 2LS767-B1.
- CONDUIT ASSOCIATED WITH PENETRATION NOS: 326212212 AND 326212213.
- OVERHEAD LIGHTS, EMERGENCY LIGHTS, FIRE ALARM, BLUE LIGHT, SOUND POWERED PHONE JACK, POWER OUTLET RECPT.

3. ASSOCIATED EQUIPMENT WHICH MAY BE DISABLED :

- 480V SHUTDOWN BOARD 2A.
- 480V LOAD SHEDDING LOGIC PANELS 25-44A-11 AND 25-44B-11.
- 2A960 : 2A953; 480V LOAD SHEDDING LOGIC PANELS 25-44; POWER FAILURE ALARM
  GR335; GR336; GR337; GR338 : GROUND WIRE FOR PANELS 25-44A-11, -44A-12, -44B-11, AND -44B-12
- 2LS757-A1 : 2LS755-A1; CONT BAY WATER CHILLER B 480V LOAD SHED LOGIC
  2LS756-A1; CONT BAY WATER CHILLER 3A 480V LOAD SHED LOGIC
- 2LS767-B1 : 2LS765-B1; CONT BAY WATER CHILLER B 480V LOAD SHED LOGIC
  2LS766-B1; CONT BAY WATER CHILLER 3A 480V LOAD SHED LOGIC
- 2LS347-B1 (FROM PANEL 25-44B-11) : 480V LOAD SHED LOGIC
  2LS16-B1; 250V BATT. CHARGER 2A
  2LS84-B1; DRYWELL BLOWER 2B-3
  2LS86-B1; DRYWELL BLOWER 2B-3
  2LS88-B1; FCV-70-48
  2LS90-B1; DRYWELL BLOWER 2B-4
  2LS92-B1; DRYWELL BLOWER 2A-5
  2LS94-B1; DRYWELL BLOWER 2A-5
  2LS96-B1; RECIRC MG SET OIL PUMP 2A-3
  2LS98-B1; DRYWELL BLOWER 2B-5

(SEE CONT. ON PAGE 2)

4. AUTOMATIC UNIT TRIP INITIATED : [✓] YES , [ ] NO
   (CONSERVATIVE DETERMINATION)

5. COMMENTS :
   LOSS OF 480V SD BD. 2A WILL RESULT IN LOSS OF POWER FOR PANELS 25-44A-11 & 25-44A-12 WITH PANEL 25-44B-11 BEING IMPACTED THUS A COMPLETE LOSS OF 480V LOAD SHEDDING LOGIC CAN BE POSTULATED.

EQUIPMENT POTENTIALLY DISABLED BY THE IGNITION SOURCE

1. IGNITION SOURCE : 480V SHUTDOWN BOARD 2A

2. FIRE DAMAGE ZONE COMPONENTS (CONTINUED) :

N/A

3. ASSOCIATED EQUIPMENT WHICH MAY BE DISABLED (CONTINUED) :

- CONDUITS ASSOCIATED WITH PENETRATIONS S26212212 & S26212213.

S26212212: 2LS363-B1:
(PNL 25-44B-11)
2LS20-B1 PNL 25-44B-11 LOGIC POWER SUPPLY ([480V SD BD 2B])
2LS51-B1 DRYWELL BLOWER 2B-1
2LS53-B1 DRYWELL BLOWER 2B-1
2LS55-B1 DRYWELL BLOWER 2B-2
2LS57-B1 RAW COOLING WATER BOOSTER PUMP 2A
2LS59-B1 CLOSED COOLING WATER PUMP 2B
2LS61-B1 CLOSED COOLING WATER PUMP 2B
2LS63-B1 FUEL POOL COOLING PUMP 2B

S26212213: 2LS448-B11:
(PNL 25-44B-12)
2LS171-B2 DRYWELL BLOWER 2A-1
2LS173-B2 DRYWELL BLOWER 2A-2
2LS176-B2 DRYWELL BLOWER 2A-2
2LS180-B2 CONT. & SER. AIR COMPR. D
2LS182-B2 CLOSED COOLING WATER PUMP 2A
2LS184-B2 FUEL POOL COOLING PUMP 2A

480V LOAD SHEDDING LOGIC

- OVERHEAD LIGHTS, EMERGENCY LIGHTS, FIRE ALARM, BLUE LIGHT, SOUND POWER PHONES JACK, POWER OUTLET RECPT.

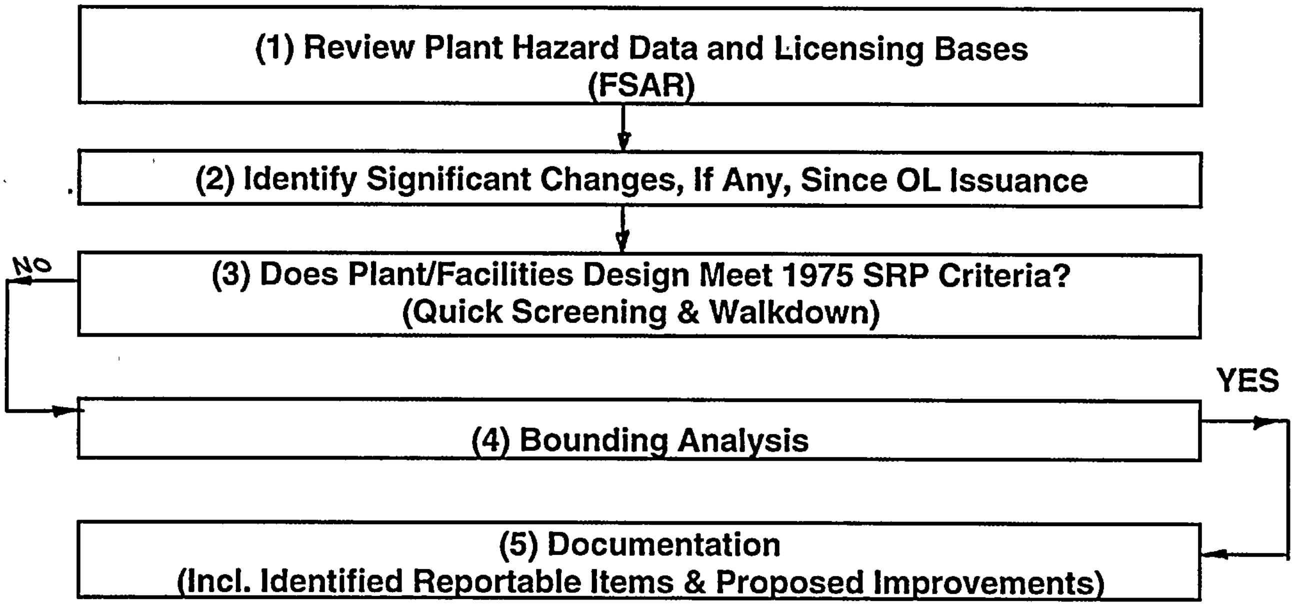
5. HIGH WINDS, FLOODS, AND OTHERS

5.1 HIGH WINDS

5.1.1 METHODOLOGY

For the purposes of an IPEEE, TVA followed a progressive screening approach to identify potential vulnerabilities at Browns Ferry Nuclear Plant due to high winds and tornadoes as outlined in NUREG-1407.

The steps shown in the figure below represent a series of steps followed in increasing level of detail, effort, and resolution.



Optional steps were bypassed because though the 1975 Standard Review Plan (SRP) (NUREG-75/087) criteria were not met, the potential vulnerabilities were either identified or demonstrated to be insignificant based on Bounding Analysis.

IPE for internal events have been completed. No plant-unique accident sequences different from those determined by the IPE for internal events were predicted or identified. Therefore, no additional containment performance assessment is needed.

5.1.2 REVIEW OF PLANT-SPECIFIC HAZARD DATA AND LICENSING BASES

The IPEEE team reviewed in detail the information on plant design hazard and the licensing bases for high winds and tornadoes. The review was based on a detailed study of the latest version of the Browns Ferry Nuclear Plant Final Safety Analysis

Report (FSAR). Some relevant FSAR information pertaining only to winds and tornadoes are documented in Appendix A.

5.1.3 IDENTIFICATION OF SIGNIFICANT CHANGES SINCE OL ISSUANCE

The IPEEE team reviewed the Browns Ferry site for identifying any significant changes with respect to high winds, since the operating license (OL) was issued and not reported per 10CFR 50.71(e). This was accomplished by a) comparing the FSAR issued at the time of issuance of the operating license to the latest version of the FSAR and b) by walkdown of the site area and noting the difference between original site layout plan and present as-built condition.

The following observations were made:

1) The nearest missile testing facility is located on Redstone Arsenal about 25 miles east of the plant. The maximum range of any tested missile is about 15 kilometers (about 9 miles). Testing restrictions preclude testing any missile with a range that could carry it beyond the reservation boundaries. In addition, the testing facility has the capability of destroying an errant missile.

2) The nearest industrial facility is located 5.5 miles from the Browns Ferry site.

3) There have not been any significant changes in the design bases for winds and tornado which could affect the original design condition.

4) After the site walkdown the following additional structures or components were noted:

   1) Plant Administration Building 2) Sewage Lift Station 3) Plant Engineering Building 4) Gasoline Pump 5) Trash Compactor 6) Diesel HPFP House 7) Two Additional Condensate Water Storage Tanks 8) East Access Facility 9) Drywell Chillers and Transformer 10) Dewatering Facility 11) Acid Storage Area 12) Hypochlorite Building 13) Trash Compactor 14) Paint and Sandblasting Shop 15) Security Diesel Building 16) Liquid Nitrogen Storage Tank. 17) New Security Building 18) Maintenance Building 19) 480 Volt Motor Operated Control Center.

   Other items or structures which are temporary or semi- permanent in nature and located in owner controlled or protected area are:

   1) A number of trailers being used as temporary worker's facility 2) Radwaste storage area 3) Portable toilets 4) Tool boxes 5) Loose construction materials 6) Garbage drums 7) Semi-permanent office buildings.

5.1.4 DETERMINATION IF THE PLANT/FACILITIES DESIGN MEETS 1975 SRP CRITERIA

The approach for determination was based on 1) Study of 1975 SRP Criteria relevant to high winds and tornadoes, 2) Comparison of information obtained from the review discussed in Sections 5.1.2 and 5.1.3 for conformance to 1975 SRP criteria, and 3) Performing a confirmatory walkdown of the plant.

5.1.4.1 STUDY OF 1975 SRP CRITERIA

To perform a comprehensive study the following sections of the Standard Review Plan (SRP) NUREG-75/087 and relevant Regulatory Guides were reviewed and important design bases were noted:

a)	Section 2.3.1	Regional Climatology
b)	Section 2.3.2	Local Meteorology
c)	Section 3.3.1	Wind Loadings
d)	Section 3.3.2	Tornado Loadings
e)	Section 3.5.1.4	Missiles Generated by Natural Phenomena
f)	Section 3.5.1.5	Site Proximity Missiles (Except Aircraft)
g)	Section 3.5.2	Structures, Systems, and Components to be Protected From Externally Generated Missiles
h)	Section 3.5.3	Barrier Design Procedures
i)	Section 3.8.4	Other Seismic Category I Structures

j) Regulatory Guide 1.76 "Design Basis Tornado for Nuclear Power Plants".

k) Regulatory Guide 1.117 "Tornado Design Classification".

- General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," of Appendix A, "General Design Criteria for Nuclear Power Plants", to 10CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires, in part, that structures, systems, and components important to safety be designed to withstand the effects of natural phenomena such as tornadoes

without loss of capability to perform their safety functions. Criterion 2 also requires that the design bases for these structures, systems, and components reflect (1) appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding region, with sufficient margin for the limited accuracy and quantity of the historical data and the period of time in which data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena, and (3) the importance of the safety functions to be performed.

- Regulatory Guide 1.76 (Reference 5.1.3), Design Basis Tornado (DBT) for Region I, has a maximum wind speed of 360 mph (290 mph of rotational and 70 mph of translational) and pressure drop of 3.0 psi @ 2psi/sec.

- The procedures utilized to transform the wind velocity into an effective pressure to be applied to structures and parts and portions of structures are given by the following:

  For a design wind velocity of V_{30} (mph) specified at a height of 30 feet above the ground, the velocity pressure, q_{30}, is given by $q_{30} = 0.00256\ (V_{30})^2$ psf. [Reference 5.1.8]

- The following shall apply for transforming the tornado wind velocity into an effective pressure applied to structures:

  For a maximum tornado velocity, V (mph), the maximum velocity pressure, $p = 0.00256\ V^2$ psf. [Reference 5.1.8]

- The failure of any structure or component not designed for tornado loads will not affect the capability of other structures or components to perform necessary safety functions.

- The probability per year of damage to the total of all important structures, system, and components due to a specific design basis natural phenomena capable of generating a missile should be smaller than the acceptable probability of 10^{-7} per year as stated in Regulatory Guide 1.117 (Reference 5.1.4).

- All plants are required to be designed to protect safety-related structures and component against damage from a missile which might be generated by the design basis tornado for that plant. The effect of postulated missiles that include at least three objects: a massive high kinetic energy missile which deforms on impact, a rigid missile to test penetration resistance, and a small rigid missile of a size sufficient to just pass through any openings in protective barriers. Until more definitive guidelines are established, these missiles may be assumed to be an 1800 Kg automobile, a 125 Kg 8" armor piercing artillery shell, and a 1" solid steel sphere, all impacting at 35% of the maximum horizontal wind speed of the design basis tornado or alternatives suggested by the National Bureau of Standards. Vertical velocities of 70% of the postulated horizontal velocities are acceptable for both spectra (as mentioned in reference 5.1.9) except for the small missile.

5.1.4.2 COMPARISON OF INFORMATION OBTAINED FROM THE REVIEW DISCUSSED IN SECTION 5.1.2 AND 5.1.3 FOR CONFORMANCE TO 1975 SRP CRITERIA

- As required by the SRP all structures, systems, and components important to safety have been designed to withstand the effects of wind and tornado without loss of capability to perform their safety functions with following exceptions: 1) Top 320 feet of 600 feet high reinforced concrete chimney located south of the class I Off-gas treatment building and approximately 365 feet from the class I Diesel Generator and Standby Gas Treatment buildings has been designed to fall in the event of a tornado and not impact any nearby class I structures (Refer FSAR Section 12.2.4.1). 2) The superstructure of the secondary containment above the refueling floor is a structural steel frame which supports metal roof decking, stepped fascia panels, and insulated metal siding panels. Some of the panels are designed as relief panel and permitted to blow off during a design basis tornado in order to prevent the over pressurization of the secondary containment system. 3) Diesel Generator Building walls above El. 583.5 does not meet the minimum thickness requirement (24" vertical and 21" horizontal) for tornado generated missile protection. A probabilistic analysis of the Diesel Generator Building walls was performed (Reference 5.1.16) to show that the frequency of occurrence of tornado generated missile strikes is less than or equal to 1.0×10^{-7} per year which meets the SRP criteria. This has been documented in FSAR section 12.2.8.1.

- Maximum wind speed of 300 mph has been used for the Browns Ferry Design Basis Tornado (DBT) instead of 360 mph specified by Regulatory Guide 1.76.

- Formula $q = 0.002558\ V^2 \simeq 0.00256\ V^2$ psf is equivalent to SRP formula for determining the dynamic pressure due to 100 mph and the 300 mph tornado wind velocity (Refer FSAR section 12.2.2.9.1).

- The structures and or components not designed for tornado loads were not considered to perform any safety-related function or importance to the continued safe operation of the plant. Although the potential exists for a portion of these structures or components to become tornado generated missiles, it was judged that any such missiles are represented by the existing postulated missiles considered in the design of the safety-related structures, systems or components.

- Design Basis missiles as defined in FSAR Section 12.2.2.9.2 differ from the requirement of SRP Section 3.5.1.4.

- Structures of Browns Ferry have not been evaluated for tornado generated missile resulting from vertical wind speed of tornado as required by SRP.

5.1.4.3 WALKDOWN OF THE PLANT

The walkdown effort concentrated on outdoor facilities that could be affected by high winds and tornadoes. Additional structures or components since issuance of the operating license have been noted in section 5.1.3 observation item 4. The following paragraphs discuss those additional structures or components:

- Out of 19 additional structures noted, item 1 through 16 have been reported per 10CFR 50.71(e).

- Item 17 is part of the security upgrade and still under construction. A safety assessment has been performed to show it does not have any adverse effect on class I structures, systems or components.

- Items 18 and 19 are non essential class II structures, and the failure of these structures or components thereof will not affect the capability of other safety related structures, systems or components..

- For other temporary or semi-permanent structures located inside the owner controlled or protected area, it has been determined that in case of a collapse due to high winds or tornado, no component thereof acts as a missile which is not represented by the design basis missile for the Browns Ferry Nuclear Plant. Determination is based on engineering judgement and analysis of some of the temporary structures of similar nature (Reference 5.1.15).

- It was observed during walkdown of the plant that one external door (door number 484) located at the east end of the control bay at elevation 593 is not designed for tornado missile protection.

5.1.5 <u>IDENTIFICATION OF OUTLIERS</u>

The following outliers were identified based on comparison between SRP NUREG-75/087 and Browns Ferry nuclear plant design basis:

1) Tornado Missile impact due to vertical velocity of tornado wind has not been taken into account for design of safety-related structures, systems or components at BFNP.

2) Control Bay door number 484 (external door located at the east end of control bay) is not designed to withstand tornado wind or tornado missile impact.

3) Design basis tornado missile as specified in SRP NUREG-75/087 Section 3.5.1.4 differs from design basis tornado missile used for BFNP (Refer FSAR Section 12.2.2.9.2).

4) Tornado wind speed of 300 mph used in the BFNP design is less than the maximum tornado wind speed of 360 mph specified by Regulatory Guide 1.76.

5) For both the Diesel Generator buildings cloistered air intake and plenums above elevation 595, and exhaust stacks above elevation 602 are unprotected from tornado generated missile strike.

5.1.6 RESOLUTION OF OUTLIERS

1) The vulnerable areas in safety related structure for impact due to vertical wind speed are Reactor building refueling floor at El. 664, Control Bay roof and Diesel Generator building roof. Effect of vertical wind is analyzed below for both cases.

a) Control Bay Roof and Diesel Generator Building Roof:

The effect of vertical wind velocity has been evaluated in calculation CD-Q0000-940307 (Reference 5.1.17) and it has been demonstrated that roof slabs of Diesel Generator building and Control Bay are adequate for tornado generated missile protection.

b) Reactor Building Refueling Floor at EL.664:

The fuel storage pool provides a specially designed underwater storage space for the irradiated spent fuel assemblies, which require shielding and cooling during storage and handling. The dryer separator storage pool provides an underwater storage space for the irradiated reactor vessel steam dryer, steam separator assembly and grid plates during a routine refueling outage. All storage pools are lined with stainless steel liner plates to minimize the potential of pool leakage.

The potential effects of a tornado generated missile striking the fuel storage pool of a Boiling Water Reactor was investigated by General Electric (GE) and documented in GE Topical Report "Tornado protection for the spent fuel storage pool," APED-5696, November 1968 (Reference 5.1.23). Two key concerns were examined to determine i) whether sufficient water could be removed from the pool to prevent cooling of the fuel, and ii) whether missiles could potentially enter the pool and damage the stored fuel.

The GE Topical report demonstrates that the fuel pool is designed with substantial capability for withstanding these effects due to a tornado.

2) It has been demonstrated in calculation CD-Q0000-940307 (Reference 5.1.17) that probability of Control Bay door number 484 or equivalent open area being hit by a design basis tornado generated missile is 7.563×10^{-8} per year. The frequency is lower than the NRC established credibility criterion of 10^{-7} per year as given in Regulatory Guide 1.117.

3) Design Basis missile as defined in FSAR Section 12.2.2.9.2 differs from missiles defined by NUREG-0800 and NUREG-75/087 Section 3.5.1.4 and Section 3.5.3. It has been demonstrated in calculation CD-Q0000-940307 (Reference 5.1.17) that Browns Ferry design basis tornado generated missile spectrum is conservative compared to NUREG-0800 Section 3.5.1.4 spectrum II and Section 3.5.3 Table 2, and is acceptable.

4) As per Regulatory Guide 1.76 (Reference 5.1.3), design basis tornado for region I has a maximum wind speed of 360 mph. Maximum wind speed is the sum of the rotational speed of 290 mph and maximum translational speed of 70 mph. Browns Ferry design has been based on tornado wind speed of 300 mph and no breakdown of rotational and translational wind speed has been mentioned in any design basis document. This value has been justified in calculation CD-Q0000-940307 (Reference 5.1.17) based on probability of Browns Ferry structure being hit by a tornado with wind velocity more than 300 mph to be 1.08×10^{-8} per year. The frequency is lower than the NRC established credibility criterion of 10^{-7} per year as given in Regulatory Guide 1.117 and reference 5.1.21.

For BFNP, effect due to tornado generated missile is based on 300 mph wind velocity, whereas SRP NUREG-75/087 section 3.5.1.4 specifies maximum horizontal wind velocity of 288 mph (0.8 times total tornado velocity).

5) It has been demonstrated in calculation XD-Q0000-890002 (Reference 5.1.16) that probability of cloistered air intake and exhaust plenums above elevation 595 and exhaust stacks above elevation 602 being hit by a design basis tornado generated missile is at least 7.8×10^{-8} per year. The frequency is lower than the NRC established credibility criterion of 10^{-7} per year as given in Regulatory Guide 1.117.

5.1.7 BOUNDING ANALYSIS

Design of BFN for wind and tornado loadings is described in the FSAR Section 12.2. The original design dates before the Regulatory Guides and Standard Review Plan were issued, therefore, strict conformance to Regulatory Guides and the 1975 Standard Review Plan Criteria does not exist.

In calculation CD-Q0000-940307 (Reference 17) a bounding analysis was performed to assess the overall effect on vulnerable areas identified as outliers due to multiple missile effects.

The areas considered vulnerable to tornado missile damage which has been included in the bounding analysis are:

a) Diesel Generator Building roof openings (unit 1,2 & 3) - openings of cloistered air intake and exhaust plenums above El. 595'.

b) Diesel Generator Building wall (El. 583' to 595') - East & West wall between El. 583.5' to 594.0' (inadequate wall thickness).

c) Diesel Generator Building doors - 10 metal doors at El. 565.

d) Reactor Building piping penetrations - 13 pipes of different diameters at north and south side wall.

e) Vent Tower exhaust & supply fan - exhaust and supply fan vents located at Reactor Building vent tower.

f) Control Bay door - door no 484 (8'-0" x 6'-4") - not designed for tornado.

g) Intake Pumping Station RHRSW pumps - 9 pumps covered by grating at top and 3 other openings.

Total area of openings and/or inadequate concrete thickness calculated to be 5107 sq.ft.

Total probability of missiles penetrating the target area was calculated to be 1.85×10^{-6} based on a hypothetical multiple threat of 6000 available missiles and ψ value of 8.64×10^{-11} (ψ = Frequency of impact / missile / unit target area / tornado point strike frequency)

The calculated probability value is conservatively high for the following reasons:

- Only a fraction of the missiles will have the trajectories that would allow the

missiles to penetrate through critical openings or strike the critical concrete walls that could impact the equipment.

- Only a small fraction of the missiles will exceed the velocity needed to damage the impacted equipment.
- Damage to individual components does not usually lead to a high risk of core damage. Also, recoverable actions and redundant components provide means to limit the risk of core damage.

Therefore, the probability of multiple missiles penetrating any safety-related structure through openings or thin concrete walls and causing sufficient damage to equipment resulting in an increase in core damage is estimated to be less than 1×10^{-6} per year.

This probability is acceptably low for the purposes of screening the event according to NUREG-1407.

5.1.8 DISCUSSION ON NRC INFORMATION NOTICE 93-53 SUPPLEMENT1

NRC Information Notice 93-53, Supplement 1 "Effect of Hurricane Andrew on Turkey Point Nuclear Generating Station and Lessons Learned", was studied by the IPEEE team and the following observations were made:

- Turkey Point Nuclear Generating Station was hit by hurricane storms with maximum wind speeds of 180 mph.
- The wind speeds used in the design of safety-related structures of east-coast plants vary from 110 to 130 mph with a load factor of 1.7, so plant's safety-related structures are just adequate to resist a category 5 hurricane.
- Some of the non-safety structures and equipment were severely damaged without jeopardizing safety function of any structure or equipment.

In light of the effect of Hurricane Andrew, the following determinations were made for Browns Ferry Nuclear Plant:

- The Browns Ferry site is located in an area occasionally traversed by cyclonic storms. Wind speeds in excess of 40 mph are occasionally reported, but wind speeds in excess of 75 mph are rare.
- In spite of the low probability of high winds, a structural design capable of withstanding loadings resulting from a 100-mph sustained wind has been considered appropriate, and all class I structures and equipment have been

designed to maintain their integrity when subjected to loading resulting from a 300-mph tornado.

- An assessment has been made to assure that the consequences of failure of non-safety structures and equipment would not disable the safety functions of safety-related structures, systems and components (also termed as II over I consideration).

5.1.9 CONCLUSION

Based on the comparison of Section 5.1.2 and 5.1.3 to Section 5.1.4 and resolution of outliers, it is concluded that the Browns Ferry Plant/Facilities design is robust in relation to the 1975 SRP Criteria, and the walkdown reveals no potential vulnerabilities which could not be demonstrated as insignificant. Also, TVA developed Abnormal Operating Instruction O-AOI-100-7 (Reference 5.1.24) to provide specific actions to be taken during tornado watches and warnings to ensure all personnel are evacuated to a safe area and to place buildings/equipment in a configuration which will be more capable of withstanding the effects of a Tornado. It is judged that the contribution from the high winds hazard to core damage frequency is less than 10^{-6} per year, and the IPEEE screening criteria is met.

No other plant-unique external event is known that poses any significant threat of severe accident within the context of the screening approach for "High Winds".

5.1.10 REFERENCES FOR HIGH WINDS

5.1.1 General Design Criteria BFN-50-C-7101 Revision R1.

5.1.2 Design Criteria BFN-50-C-7100 Revision R9.

5.1.3 U.S. Atomic Energy Commission (AEC) Regulatory Guide 1.76 "Design Basis Tornado for Nuclear Power Plants".

5.1.4 U.S. NRC Regulatory Guide 1.117 "Tornado Design Classification".

5.1.5 U.S NRC Standard Review Plan (SRP) NUREG-75/087 Section 2.3.1 "Regional Climatology".

5.1.6 U.S NRC Standard Review Plan (SRP) NUREG-75/087 Section 2.3.2 "Local Meteorology".

5.1.7 U.S NRC Standard Review Plan (SRP) NUREG-75/087 Section 3.3.1 "Wind Loadings".

5.1.8 U.S NRC Standard Review Plan (SRP) NUREG-75/087 Section 3.3.2 "Tornado Loadings".

5.1.9 U.S NRC Standard Review Plan (SRP) NUREG-75/087 and NUREG-0800 Section 3.5.1.4 "Missile Generated by Natural Phenomena".

5.1.10 U.S NRC Standard Review Plan (SRP) NUREG-75/087 Section 3.5.1.5 "Site Proximity Missiles (Except Aircraft)".

5.1.11 U.S NRC Standard Review Plan (SRP) NUREG-75/087 Section 3.5.2 "Structures, Systems, and Components to be Protected from Externally Generated Missiles".

5.1.12 U.S NRC SRP NUREG-0800 Section 3.5.3 "Barrier Design Procedures".

5.1.13 U.S NRC NUREG-1407 "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities".

5.1.14 "Tornado Protection for the Spent Fuel Storage Pool," GE APED-5696, November 1968.

5.1.15 Calculation CD-Q0303-890654 "Tornado Generated Missile Protection".

5.1.16 Calculation XD-Q0000-890002 "Frequency of Occurrence of Tornado generated Missile Strike on Vulnerable Diesel Generator Building Areas".

5.1.17 Calculation CD-Q0000-940307 "Verification calculation for individual plant examination of external events (IPEEE) for high winds".

5.1.18 Calculation MD-Q0031-87011 "Tornado protection requirements for HVAC equipment in the Reactor building vent towers".

5.1.19 Calculation XD-Q0000-900002 "Frequency of occurrence of Tornado generated missile strike on vulnerable piping penetrating the Reactor building".

5.1.20 Browns Ferry Nuclear Plant Final Safety Analysis Report Amendment 11.

5.1.21 "Technical Basis For Interim Regional Tornado Criteria," By E. H. Markee, Jr., WASH-1300, USAEC, May 1974.

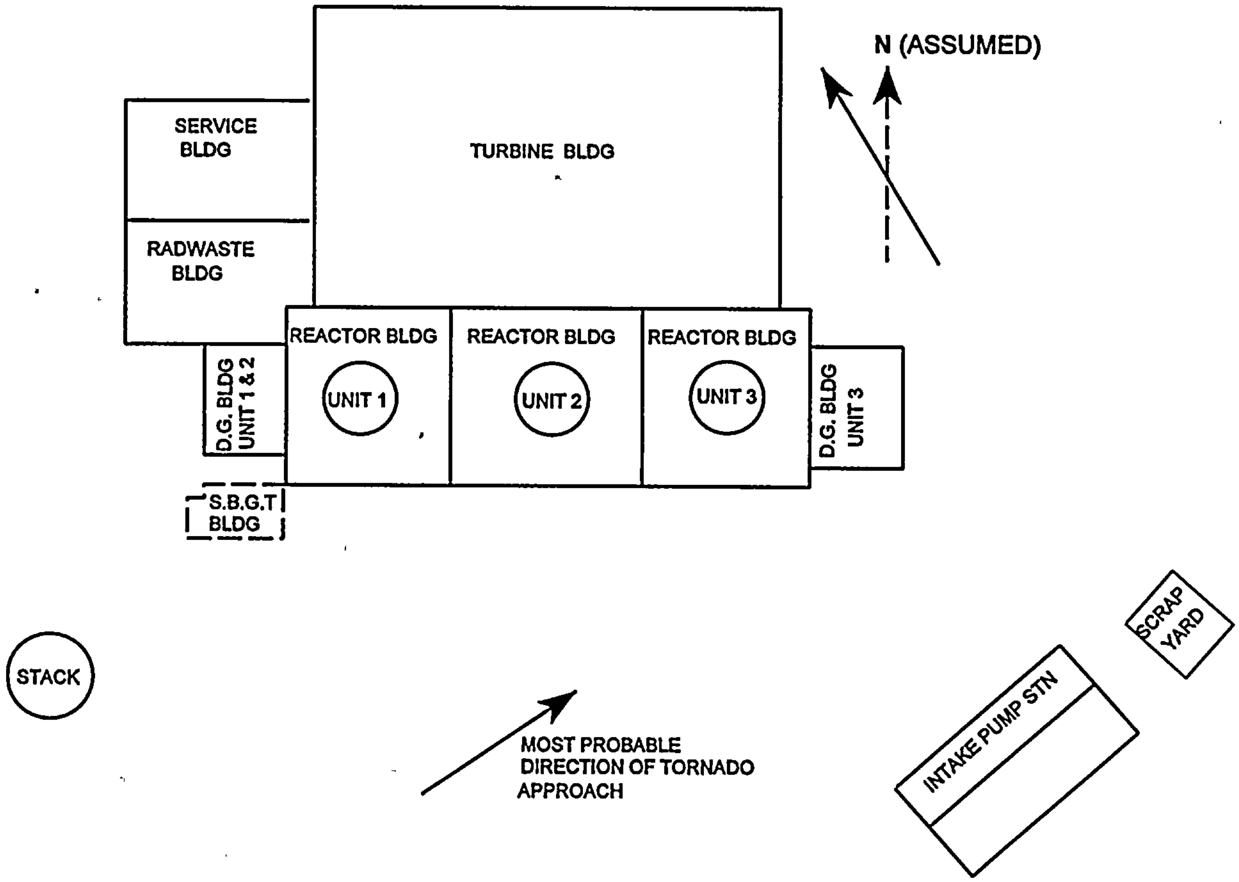
5.1.22 "Tornado Resistant Design of Nuclear Power Plant Structures," By J. R. McDonald, K. C. Mehta and J. E. Minor, Nuclear Safety, Vol. 15, No. 4, July-August 1974.

5.1.23 "Tornado Protection for the Spent Fuel Storage Pool" by D. R. Miller and W. A. Williams, GE Topical Report APED-5696, November 1968.

5.1.24 BFNP Abnormal Operating Instruction 0-AOI-100-7 "Tornado".

5.1.25 "Tornado Missile Risk Analysis" By Carolina Power and Light Company, EPRI NP-768

5.1.26 "Shutdown Decay Heat Removal Analysis of a Westinghouse 2-Loop Pressurized Water Reactor" - NUREG/CR-4458, March 1987.



LAYOUT OF BROWNS FERRY NUCLEAR PLANT

APPENDIX A

SECTION 1.6.1.1.10

DESIGN BASES DEPENDENT UPON THE SITE AND ENVIRONS

c. Wind Loading Design

A structural design capable of withstanding loadings resulting from a 100-mph sustained wind is considered appropriate. All Class I structures and equipment are designed to maintain their integrity when subjected to loading resulting from a 300-mph tornado.

SECTION 2.3.6.1

THUNDERSTORMS

During 1968-1980, there were about 57 days annually on which thunderstorms were reported in the Huntsville area. Thunderstorms occurred most frequently in July, August, June, and May. November and December had the smallest number of thunderstorms, with an average of one thunderstorm day each month.

Windstorms (often associated with thunderstorms) may occur several times a year, particularly in winter, spring, and summer, with winds occasionally exceeding 40 mph. In 1964, 95 mph winds, with rain and hail, were reported at the Redstone Arsenal, 25 miles east-southeast of the site. Also in April 1958 and July 1963, winds were reported in excess of 70 mph in the Huntsville area.

SECTION 2.3.6.2

TORNADOES

There were four tornadoes reported in Limestone County during the 50-year period 1916-1965. In the adjacent counties, Morgan and Madison, 18 tornadoes were reported during the same period. The bordering Southern Tennessee counties, Giles and Lincoln, reported 13 and 5 tornadoes, respectively, during the 55-year period 1916-70.

Tornado data compiled by the severe local storms (SELS) unit of the National Weather Service for the period 1955-67 were used for evaluating the tornado probability for the Browns Ferry site. In the 1-degree latitude/longitude square containing the site (about 3,930 mi^2), there were 31 tornadoes reported during the 13-year period, or about 2.38 tornadoes per year. Thom's value for the mean tornado path area (2.82 mi^2) was used in calculating an annual point probability for the site of 1.71×10^{-3}; this is equivalent to a

mean recurrence interval of about 600 years.

The National Severe Storms Forecast Center in Kansas City, Missouri calculated the tornado return probability for the Browns Ferry site based on tornado occurrences within a 30 nautical mile (nm) radius during 1950-1986. A circle of 30 nm radius has an area comparable to one degree latitude-longitude square. Based on 48 tornado occurrences having path size estimates in the 37-year period, the return probability is 6.979×10^{-4} and the mean return interval is 1,433 years. The annual tornado occurrence in the 30 nm radius circle is 1.81 (based on 67 tornadoes reported).

SECTION 2.3.8

CONCLUSIONS

The Browns Ferry site is located in an area occasionally traversed by cyclonic storms. Wind speeds in excess of 40 mph are occasionally reported, but wind speeds in excess of 75 mph are rare. The estimated probability of a tornado occurrence at the Browns Ferry site in any one year is 6.979×10^{-4}, or about one occurrence in 1,433 years should be expected. In spite of the low probability, the plant is designed to withstand tornado forces.

SECTION 11.6.4

The intake pumping station is a watertight structure below the top deck which is at EL. 565.

The Condenser Circulating Water pumps and valves are not designed to Class I design considerations. The circulating water pump motors and travelling screens are the only parts exposed above grade. The pumps have been analyzed and determined to be stable under tornado wind conditions. They are subject to individual missile damage, but there is virtually no possibility of all nine pump motors being damaged simultaneously by missiles.

One circulating water pump has more than adequate capacity to dissipate the shutdown heat for the three units. A cross-tie with electric-operated butterfly valves is provided between the three circulating water tunnels so that any one pump in emergency, can supply water to all units.

SECTION 12.2.1

Loading Considerations for Structures, Foundations, Equipment and Systems

Structures and Foundations

Wind loads - 100 mph

Tornado wind loads - 300 mph
Tornado depressurization (3 psi at the rate of 0.6 psi)
Tornado generated missiles (see section 12.2.2.9.2)

Reactor Building Crane

Tornado wind loads - 300 mph

SECTION 12.2.2.3.1

EXTERIOR WALLS

The exterior walls are designed to resist stresses from shears, moments, and deflections resulting from .... normal wind loads, tornado wind loads, and a rapid depressurization as a result of a tornado.

Wind loads are determined as described in paragraph 12.2.2.9. For a tabulation of stresses resulting from 300-mph wind and rapid depressurization. For a discussion of the capability for resisting penetration by tornado generated missiles, see paragraph 12.2.2.9.2.

SECTION 12.2.2.4

STEEL STRUCTURE ABOVE EL. 664

Exterior insulated siding is suspended 12 feet off the center of the frame columns for architectural appearance by a girt system of channels and box beams. The siding is stepped in at the top and base of the frame in two-foot steps. Girts are designed to remain in place under tornado wind loading.

The average design wind pressure is obtained by applying a shape factor of 1.3 to the dynamic wind pressure, which is the product of one-half the air density and the square of the design velocity. The value 1.3 is the shape coefficient for a typical building with vertical walls normal to the wind direction, consisting of windward side coefficient of 0.9 and leeward side coefficient of -0.4. The normal design wind of 100 mph corresponds to an average design pressure of 33 lb/ft^2 (consisting of 23 lb/ft^2 on windward side and 10 lb/ft^2 on leeward side) and the tornado wind to a design pressure of 207 lb/ft^2(consisting of windward and leeward sides combined).

SECTION 12.2.2.5

REACTOR BUILDING CRANE

SECTION 12.2.2.5.1

DESCRIPTION

The crane was designed to withstand the DBE when loaded and a 300-mph wind when unloaded. Stresses do not exceed 0.9 of the yield point (Fy) for the wind condition.

SECTION 12.2.2.9

WIND LOAD

SECTION 12.2.2.9.1

PRESSURE MAGNITUDE AND DISTRIBUTION

The magnitude and distribution of wind pressures, both for the 100 mph wind and the 300 mph tornado wind, are determined by following the recommendations of ASCE paper no. 3269 "Wind Forces on Structures".

The dynamic pressure is determined by

$q = 0.002558V^2$
q = dynamic pressure, psf
V = wind velocity, mph

SECTION 12.2.2.9.2

TORNADO GENERATED MISSILES

The potential missiles that could be generated at any point, with respect to the Reactor Building, and which serve as the design basis missiles are:

a. A 2-inch x 4-inch x 12-foot board weighing 40 pounds/cu ft, end on;

b. A cross-tie, 7 inches x 9 inches x 8-1/2 feet weighing 50 pounds/cu ft, end on;

c. A compact car weighing 1800 pounds with an impact area of 20 square feet;

d. Pieces of concrete 6-1/2 inches x 12 inches x 2 inches thick, end on, as a result of the spalling effect from the concrete chimney during postulated failure of chimney from tornado winds; and

e. Aircraft warning beacon from chimney.

As an upper limit, each missile is assumed to be travelling 300 mph at impact. No credit is taken for the crushing effect of missiles. The depth to which these missiles will penetrate the exterior concrete walls of the Reactor Building is calculated by the modified Petry formula as given in a report by Amirikian. Maximum penetration is 9 inches (one half of wall thickness). According to Moore, spalling on the inside face will not occur for penetration less than two-thirds the wall thickness. The reactor building walls can therefore adequately resist the spectrum of postulated missiles.

SECTION 12.2.4

REINFORCED CONCRETE CHIMNEY (CLASS I)

SECTION 12.2.4.1

SHELL

The design cases are discussed below:

Case 1 - 100-mph Wind

This case is treated in accordance with the ACI Chimney code (ACI 307-69). This case does not control the design of any section in the chimney.

Case 2 - Tornado

This is the controlling design case for most of the height of the chimney. The tornado moments are those caused by a 300-mph wind. Since the wind pressure is proportional to the square of the velocity, these moments are nine times the moments for a 100-mph wind. It is not practical to design the chimney for these forces by the working stress method and to limit stresses in the extreme reinforcing bars to yield stress. The chimney is 600 feet high and is located south of the class I Off-Gas Treatment Building and approximately 365 feet from the class I Diesel Generator and Standby Gas Treatment Buildings. In order to ensure that the chimney does not fall on these buildings in the event of a tornado, the top 320 feet of the chimney are designed to fall well before the lower 280 feet reach their ultimate load capacity.

SECTION 12.2.7.1

PUMPING STATION STRUCTURES (CLASS I)

SECTION 12.2.7.1.1

CONCRETE STRUCTURE

The structure is investigated for a tornado consisting of an atmospheric pressure decrease of 3 psi in 5 seconds. The calculated pressure differential for the enclosed parts of the intake building, including its ventilation exhaust openings, is 126 psf. This

is less than the floor design live loads and is less than the roof deck dead load. The minimum reinforcing in the two-foot-thick walls is also more than adequate for this small pressure, using normal allowable design stresses.

SECTION 12.2.8

DIESEL GENERATOR BUILDING, UNITS 1 AND 2 (CLASS I)

SECTION 12.2.8.1

CONCRETE STRUCTURE

The tornado depressurization load is determined by an analysis that considers pressure versus time as a function of the vent area available. The result of this analysis is that the pressure differential is less than 40 psf. This is the internal pressure load used for load case involving tornado depressurization.

The static load of 230 lb/sq ft from tornado wind used for load case involving 300 mph tornado wind.

The concrete walls are capable of resisting the spectrum of postulated tornado generated missiles as described in paragraph 12.2.2.9.2 except for walls less than 18-inches thick above El. 583.5'. A probabilistic analysis of these walls show that the frequency of occurrence of tornado generated missile strike is less than or equal to 1.0 x 10^{-7} per year which meets the NUREG-0800 U.S. NRC Standard Review Plan acceptance criteria of 1.0 x 10^{-7} per year. The analysis demonstrated that tornado generated missile strikes are not credible events.

Several walls less than 18-inches thick are not included in the probability calculation. The east side of the units 1 and 2 and the west side of unit 3 cloistered air intake and exhaust plenum are not considered as probable targets of a missile hit because of protection provided by the taller reactor building. The north, south, and east walls of the unit 3 mechanical equipment room above El. 597.6' are not included because even if there is a loss of air conditioning equipment due to tornado missiles, acceptable temperature can be maintained in the 4KV shutdown board rooms 3EA, B, C, and D and the bus tie board room for up to 72 hours. Beyond that time , plant procedures will be followed.

Therefore, it is concluded that the integrity of system 82 (Diesel Generator) and its associated components can be maintained.

SECTION 12.2.8.2

ACCESS DOORS

SECTION 12.2.8.2.1

DESCRIPTION

The doors are closures for the four 8 feet high x 9 feet 6 inches wide openings and one 8 feet high x 11 feet 6 inches wide opening in the Diesel Generator Building. The four doors provide access to the diesel generator units, and one door provides access to the CO_2 room. Doors at the rear of rooms for the diesel generator units connect these rooms to the CO_2 room.

For all load combinations used in designing the doors, the resulting stresses do not exceed the allowable stresses except for load case involving 300 mph wind with impact from tornado missiles. For this load case, the doors will deform, but will stop a missile.

SECTION 12.2.8.2.2

SAFETY EVALUATION

These access doors provide adequate protection for the diesel-generator units and are designed to withstand tornado conditions and missiles generated by tornadoes, flood conditions, or earthquakes, with only one of these conditions occurring at any one time.

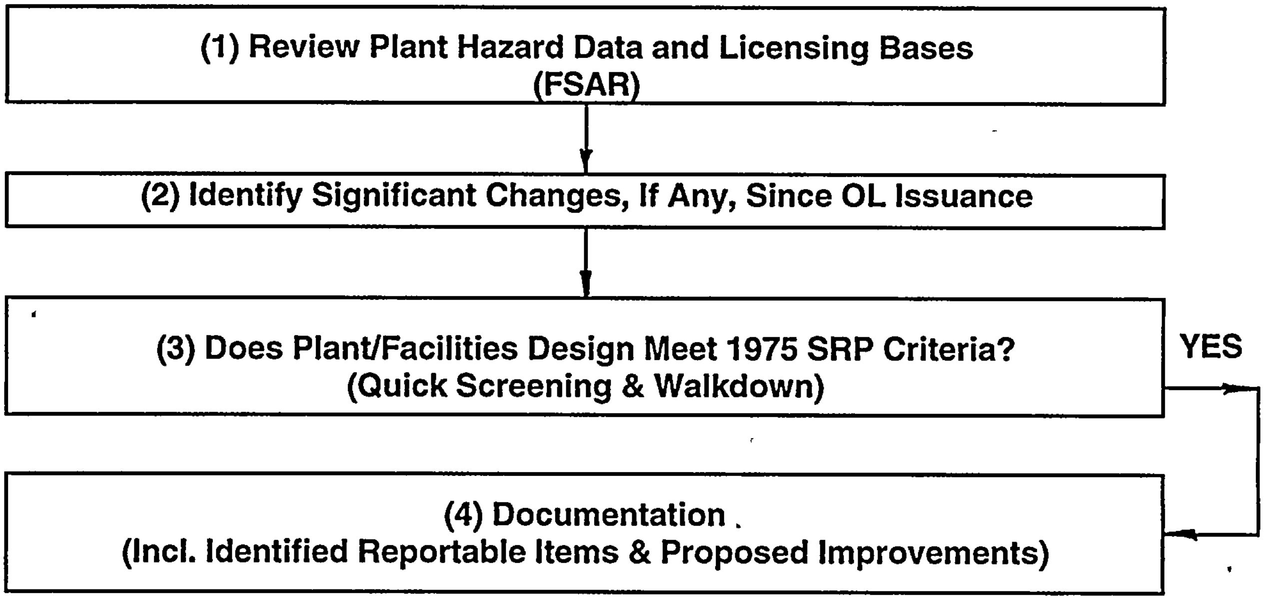
Tornado conditions consist of winds to 300 mph and missiles generated by the 300 mph wind.

5.2 FLOODS

5.2.1 METHODOLOGY

For the purposes of an IPEEE, TVA followed a progressive screening approach to identify potential vulnerabilities at Browns Ferry Nuclear Plant due to floods as outlined in NUREG-1407.

The steps shown in the figure below represent a series of steps followed in increasing level of detail, effort, and resolution.



Optional steps were bypassed because the 1975 Standard Review Plan (SRP) (NUREG-75/087) criteria were met or the potential vulnerabilities were either identified or demonstrated to be insignificant.

IPE for internal events have been completed. No plant-unique accident sequences different from those determined by the internal events IPE are predicted or identified; therefore, no additional containment performance assessment is needed.

5.2.2 REVIEW OF PLANT-SPECIFIC HAZARD DATA AND LICENSING BASES

The IPEEE team reviewed in detail the information on plant design hazard and the licensing bases for floods. The review was based on a detailed study of the latest version of the Browns Ferry Nuclear Plant Final Safety Analysis Report (FSAR). Some relevant FSAR information pertaining only to floods are documented in the Appendix B.

5.2.3 IDENTIFICATION OF SIGNIFICANT CHANGES SINCE OL ISSUANCE

The IPEEE team reviewed the Browns Ferry site for identifying any significant changes with respect to floods, since the operating license (OL) was issued and not reported per 10CFR 50.71(e). This was accomplished by a) comparing the FSAR issued at the time of issuance of the operating license to the latest version of the FSAR and b) by walkdown of the site area and noting the difference between original site layout plan and present as-built condition.

The following observations were made:

1) The Browns Ferry plant is on the right bank of Wheeler Lake at Tennessee river, 55 miles downstream from Guntersville Dam and 19 miles upstream from Wheeler Dam.

2) The natural phenomena which the plant must withstand include severe storms which may cause such events as upstream dam failure or locally intense storms which could cause flooding, equipment and system failure, building and structural damage.

3) There has been significant changes in the PMF design bases for floods. Initially it was determined that the Maximum Probable Flood at Browns Ferry would reach elevation 561. If failure of Guntersville Dam upstream from the plant site occurred at the most unfavorable time, the flood crest would reach no more than elevation 561.5. If winds were coincident with the flood crest and dam failure, the flood crest would reach elevation 563.5.

4) At present the PMF design level at the Browns Ferry site is 572.5 feet. The Probable Maximum Precipitation (PMP) has been taken as a 6-hour storm which would produce a total of 34.4 inches of rainfall with a maximum 1-hour amount of 16.7 inches.

5) After the site walkdown the following additional structures or components were noted:

1) Plant Administration Building 2) Sewage Lift Station 3) Plant Engineering Building 4) Gasoline Pump 5) Trash Compactor 6) Diesel HPFP House 7) Two Additional Condensate Water Storage Tanks 8) East Access Facility 9) Drywell Chillers and Transformer 10) Dewatering Facility 11) Acid Storage Area 12) Hypochlorite Building 13) Trash Compactor 14) Paint and Sandblasting Shop 15) Security Diesel Building 16) Liquid Nitrogen Storage Tank. 17) New Security Building 18) Maintenance Building 19) 480 Volt Motor Operated Control Center.

Other items or structures which are temporary or semi permanent in nature and located in owner controlled or protected area are:

1) A number of trailers being used as temporary worker's facility 2) Radwaste storage area 3) Portable toilets 4) Tool boxes 5) Loose construction materials 6) Garbage drums 7) Semi-permanent office buildings.

5.2.4 DETERMINATION IF THE PLANT/FACILITIES DESIGN MEETS 1975 SRP CRITERIA

The approach for determination was based on 1) Study of 1975 SRP Criteria relevant to floods, 2) Comparison of information obtained from the review discussed in Sections 5.2.2 and 5.2.3 for conformance to 1975 SRP criteria, and 3) Performing a confirmatory walkdown of the plant.

5.2.4.1 STUDY OF 1975 SRP CRITERIA

To perform a comprehensive study the following sections of the Standard Review Plan (SRP) NUREG-75/087 and relevant Regulatory Guides were reviewed and important design bases were noted:

a)	Section 2.4.2	Floods
b)	Section 2.4.3	Probable Maximum Flood (PMF) on Streams and Rivers
c)	Section 2.4.4	Potential Dam Failures
d)	Section 2.4.5	Probable Maximum Surge and Seiche Flooding
e)	Section 2.4.8	Cooling Water Canals And Reservoirs
f)	Section 2.4.10	Flooding Protection Requirements

g) Section 2.4.12 Groundwater

h) Section 3.4.1 Flood Protection

i) Section 3.4.2 Analysis Procedures

j) Regulatory Guide 1.59 "Design Basis Floods for Nuclear Power Plants".

k) Regulatory Guide 1.102 "Flood Protection for Nuclear Power Plants".

- General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," of Appendix A, "General Design Criteria for Nuclear Power Plants", to 10CFR Part 50, "Licensing of Production and Utilization Facilities," requires, in part, that structures, systems, and components important to safety be designed to withstand the effects of natural phenomena such as floods without loss of capability to perform their safety functions. Criterion 2 also requires that the design bases for these structures, systems, and components reflect (1) appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding region, with sufficient margin for the limited accuracy and quantity of the historical data and the period of time in which data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena, and (3) the importance of the safety functions to be performed.

- As per Regulatory Guide 1.59: a) "The conditions resulting from the worst site-related flood probable at a nuclear power plant (e.g., PMF, seismically induced flood, hurricane, seiche surge, heavy local precipitation) with attendant wind-generated wave activity constitute the design basis flood conditions that safety-related structures, systems, and components identified in Regulatory Guide 1.29 must be designed to withstand and retain capability for cold shutdown and maintenance thereof". b) "As an alternative to designing hardened protection for all safety-related structures, systems and components as specified regulatory position above, it is permissible not to provide hardened protection for some of these features if: i) sufficient warning time is shown available to shut the plant down and implement adequate emergency procedures; ii) All safety related structures, systems, and components are designed to withstand the flood conditions resulting from a Standard Project event with attendant wind-generated activity that may be produced by the worst winds of record and remain functional".

  The appendix of this guide gives a method for determination of design basis flooding at power reactor sites.

- The Regulatory Guide 1.102 describes three types of flood protection methods for nuclear power plants which are acceptable to NRC:

1) DRY SITE

The plant is built above the Design Basis Flooding Level (DBFL), and therefore safety-related structures, systems, and components are not affected by flooding.

2) EXTERIOR BARRIER

Safety-related structures, systems, and components protected from inundation and static and dynamic forces thereof by engineered features external to the immediate plant area. Such features (e.g.,Levee, Seawall or Floodwall, Bulkhead, Revetment, Breakwater) may, when properly designed and maintained, produce the equivalent of a dry site.

3) INCORPORATED BARRIER

Safety-related structures, systems, and components are protected from inundation and static and dynamic effects by engineered features in the structure/environment interface. Protection is provided by special design of walls and penetration closures. Walls are usually reinforced concrete designed to resist the static and dynamic forces of the DBFL and incorporate special waterstops at construction joints to prevent leakage. Penetrations include personnel access, equipment access, and through wall piping. Pipe penetrations are usually sealed with special rubber boots and flanges. Personnel access closures that have been found acceptable include submarine doors and hatches.

- As per section 2.4.3 of NUREG-75/087, the criteria for accepting the PMF-related design basis depend on one of the following three conditions.

  1. The elevation attained by the PMF (with coincident wind waves) establishes a required protection level to be used in the design of the facility.

  2. The elevation attained by the PMF (with coincident wind waves) is not controlling; the design basis flood protection level is established by another flood phenomena (e.g., the probable maximum hurricane).

  3. The site is "dry" that is, the site is well above the elevation attained by a PMF (with coincident wind waves).

5.2.4.2 COMPARISON OF INFORMATION OBTAINED FROM THE REVIEW DISCUSSED IN SECTION 5.2.2 AND 5.2.3 FOR CONFORMANCE TO 1975 SRP CRITERIA

- As required by the SRP all structures, systems, and components important to

safety have been designed to withstand the effects of flood without loss of capability to perform their safety functions.

- As required by the Regulatory Guide 1.59 BFNP safety related structures, systems, and components have been designed to withstand and retain capability for cold shutdown in the event of a PMF due to seismically induced flood or heavy local precipitation. Also proper plant procedures (Reference 14 and 15) are in place so that sufficient warning time is available to shut the plant down.

- As per description of Regulatory Guide 1.102, BFNP has been designed with combination of two types of flood protection methods: 1) Exterior Barrier 2) Incorporated Barrier.

- The lowest ground elevation in the site vicinity is about 560 feet while the average ground elevation is about 580 feet.

  Though recent studies completed in 1981 estimates probable maximum flood level as 570 feet, the PMF design value of 572.5 feet is used with the 2.5 feet difference as a design margin. The elevation attained by the PMF with coincident wind wave has been established to be 574 feet (Reference 5.2.16, 5.2.28).

  All safety-related structures are protected against all flood conditions and would not be endangered by the Probable Maximum Flood.

- PMF level for Browns Ferry has been determined based on the adequate and conservative watershed model. The unit hydrograph which is a basic component of watershed model, was verified by using it to reproduce two large floods of record (Reference 5.2.16, 5.2.21, 5.2.22, 5.2.23, 5.2.24, 5.2.25).

- All the doors for the exterior opening for Radwaste Building, Reactor Building, Diesel Generator Building, and Turbine Building etc. have been designed to withstand Probable Maximum Flood and proper seals to prevent any leakage in case of a flood.

- Probable Maximum Precipitation (PMP) for the plant drainage system and roofs of safety related structures was determined from Hydrometeorological Report No. 56 (US Weather Bureau).

5.2.4.3 WALKDOWN OF THE PLANT

The walkdown concentrated on outdoor facilities that could be affected by floods. Additional structures or components since issuance of the operating license have been noted in section 5.2.3 observation item 4. The temporary structures have been taken into consideration for calculating basis for the maximum flood level on the local plant

area from a local storm (Reference 5.2.21).

The results of the on-site walkdown confirm that there are no potential flooding vulnerabilities for the safety related facilities/structures due to local stormwater runoff.

5.2.5 CONCLUSION

Based on the comparison of Section 5.2.2 and 5.2.3 to Section 5.2.4, it is concluded that the Browns Ferry Plant/Facilities design is robust in relation to the 1975 SRP Criteria, and the walkdown reveals no potential vulnerabilities which could not be demonstrated as insignificant or not included in the original design basis analysis. Also TVA developed Abnormal Operating Instruction O-AOI-100-3 (Reference 5.2.14) and 0-AOI-100-4 (Reference 5.2.15) in order to provide specific actions to be taken for a cold shutdown of the plant in the event of an increase in Wheeler reservoir to above elevation 564, due to a break of Guntersville Dam, excessive rains, and breach of Wheeler Dam due to a design basis earthquake, other natural phenomena, or man-made disturbances. It is judged that the contribution from the flood hazard to core damage frequency is less than 10^{-6} per year, and the IPEEE screening criteria is met.

No other plant-unique external event is known that poses any significant threat of severe accident within the context of the screening approach for "Floods."

5.2.6 REFERENCES FOR FLOODS

5.2.1 General Design Criteria BFN-50-C-7101 Revision R1.

5.2.2 Design Criteria BFN-50-C-7100 Revision R9.

5.2.3 U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.59 "Design Basis Floods for Nuclear Power Plants".

5.2.4 U.S. NRC Regulatory Guide 1.102 "Flood Protection For Nuclear Power Plants".

5.2.5 U.S NRC Standard Review Plan (SRP) NUREG-75/087 Section 2.4.2 "Floods".

5.2.6 U.S NRC Standard Review Plan (SRP) NUREG-75/087 Section 2.4.3 "Probable Maximum Flood (PMF) On Streams And Rivers".

5.2.7 U.S NRC Standard Review Plan (SRP) NUREG-75/087 Section 2.4.4 "Potential Dam Failures".

5.2.8 U.S NRC Standard Review Plan (SRP) NUREG-75/087 Section 2.4.5 "Probable Maximum Surge and Seiche Flooding".

5.2.9 U.S NRC Standard Review Plan (SRP) NUREG-75/087 Section 2.4.8 "Cooling Water Canals and Reservoirs".

5.2.10 U.S NRC Standard Review Plan (SRP) NUREG-75/087 Section 2.4.10 "Flooding Protection Requirements".

5.2.11 U.S NRC Standard Review Plan (SRP) NUREG-75/087 Section 2.4.12 "Groundwater".

5.2.12 U.S NRC SRP NUREG-75/087 Section 3.4.1 "Flood Protection".

5.2.13 U.S NRC NUREG-1407 "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities".

5.2.14 Abnormal Operating Instruction 0-AOI-100-3 "Flood Above Elevation 565'" Revision 15.

5.2.15 Abnormal Operating Instruction 0-AOI-100-4 "Breach of Wheeler Dam" Revision 6.

5.2.16 Calculation CD-Q0000-890045 "PMF Calculations".

5.2.17 Calculation ND-Q0999-910033 "Safe Shutdown Analysis".

5.2.18 Calculation ND-Q0064-920113 "Flood Protection Effects of Leaving Radwaste Building Flood Doors and Reactor Building Flood Gate Normally Open".

5.2.19 Calculation "Flow Duration Data" RIMS B22900803106.

5.2.20 Calculation "Failure OF Wheeler Dam - Cooling Water Available" RIMS B22900803107.

5.2.21 Calculation "Local Drainage" RIMS B24911219300.

5.2.22 Calculation "Unit Hydrographs" RIMS B22900803109.

5.2.23 Calculation "Runoff Model" RIMS B22900803110.

5.2.24 Calculation "21,400 Square-mile Storm -- PMF Discharges" RIMS B22900803111.

5.2.25 Calculation "16,170 Square mile Storm -- PMF Discharges" RIMS B22900803112.

5.2.26 Calculation "Precipitation Excess" RIMS B22900803113.

5.2.27 Calculation "Headwater And Tailwater Ratings For Dams" RIMS B22900803114.

5.2.28 Calculation "Probable Maximum Flood (PMF) Elevations" RIMS B22900803115.

5.2.29 Calculation "Wind Waves" RIMS B22900803116.

APPENDIX - B

SECTION 1.2 DEFINITIONS

Probable Maximum Flood - The Probable Maximum Flood (PMF) is the hypothetical flood (peak discharge, volume, and hydrograph shape) that is considered to be the most severe reasonable possible based on comprehensive hydrometerological application of probable maximum precipitation, and other hydrologic factors favorable for maximum flood runoff, such as sequential storms and snowmelt. The PMF design level at the Browns Ferry site is 572.5 feet.

The term Maximum Possible Flood (MPF) has also been used in Browns Ferry design documents, however the preferred term for all Browns Ferry design is PMF.

SECTION 2.3.5.2 PRECIPITATIONS

The maximum 1-hour rainfall which may be representative for the Browns Ferry site area was 2.12 inches, recorded in ,Moulton, 20 miles southwest of the site, for the 11-year period, 1940-50. The maximum 1-hour rainfall for a 100-year frequency is 3.3 inches. The site underground storm drainage system is designed for a maximum rainfall of 4 inches per hour.

Precipitation statistics from Huntsville for the 13-year period, 1968-1980 shows a maximum rainfall in 24-hour period was 7.7 inches compared to 3.7 inches recorded by Browns Ferry meteorological facility for January 1, 1977 - December 31, 1979. Precipitation statistics from Huntsville are considered more representative of the normal than from the Browns Ferry meteorological facility.

SECTION 2.4.2.1 GROUND WATER

Ground water at Browns Ferry is derived from precipitation. Some of the precipitation evaporates, some runs off into streams, and some seeps into the soil. A portion of the water entering the soil is used by vegetation and some of it seeps downward to become ground water.

SECTION 2.4.2.1.2 SITE AREA

Ground water movement from the regional area into the Browns Ferry site area is controlled by topography and geologic structure. Recharge is also derived from local precipitation that has percolated through the residuum. Natural ground water movement in the area is from the plant site to the Tennessee river.

SECTION 2.4.2.2.2 STREAMFLOW

A flood equal to the maximum of record would produce average velocities up to 4 ft/sec in the channel and up to 2 ft/sec in the overbank area. Average velocities produced by a

maximum probable flood, regulated, would be about the same magnitude.

Failure of Wheeler Dam would require Browns Ferry plant to be shut down. The postulation of this event required an investigation into the effectiveness of the remaining reservoir to provide adequate cooling water for the plant. If Wheeler Dam were to fail, a pool of water approximately 1000 feet wide and seven miles long would be available at the Browns Ferry plant site. The water would be trapped by a 529-ft elevation at the SR22 station. The largest diffuser pipe reaches almost across the original river channel and extends above El. 529 for its full length. Thus, the trapped pool following a postulated failure of Wheeler Dam is essentially divided into two parts with about 33 percent downstream and 67 percent upstream of the diffusers.

The total flow of 100 cfs in the original river channel, in traversing the pool created at the Browns Ferry site by the failure of Wheeler Dam, would have an average residence time of about 11-1/2 days.

The intake channel to the pumping station is excavated to El. 523 and extends into the reservoir until it connects with the original channel where the aforementioned pool would be trapped. The pumping station floor elevation is at 518 which gives a minimum of 11 ft of water inside the structure. Eleven feet of water provides adequate submergence for the RHR service water pumps to deliver the shutdown cooling water requirements of 36,000 gpm (80 cfs) to the plant. This is sufficient flow to efficiently remove the decay heat from all three reactors plus the heat rejection from eight diesel generator sets operating at full load. All of the cooling water is discharged from the plant through either the RHRSW diffuser nozzles upstream of the CCW diffusers or the storm sewer downstream of the CCW diffusers.

In summary, if Wheeler Dam were to suddenly fail, the Browns Ferry plant would be shut down and maintained in a safe condition indefinitely.

SECTION 2.4.2.2.3 FLOODS

The Browns Ferry site is located on the right bank of Wheeler Lake at approximately Tennessee River mile 294. The lowest natural ground elevation in the site vicinity is about 560 feet above mean sea level and the average ground elevation is about 580.

The Probable Maximum Flood (PMF) at Browns Ferry is calculated to reach El. 572.5. This is the flood which defines the upper limit of potential flooding at the plant.

Results of completed investigation in 1981 using the information from the record-breaking flood of 1973 and flood studies for other nuclear plants upstream, give an estimated Probable Maximum Flood level of 570 at the Browns Ferry site, or 2.5 feet lower than that provided before. However, the PMF design value of 572.5 feet will continue to be used with the 2.5 feet difference as a design margin.

SECTION 2.4.7 CONCLUSIONS

Surface water runoff from the plant site is to the Tennessee River. Regulated by the TVA flood control system, the Probable Maximum Flood would result in increasing Wheeler Reservoir level to 572.5 feet above sea level at the site. Safety-related structures are protected against all flood conditions up to El. 578 and would not be endangered by the Probable Maximum Flood.

SECTION 12.2.5.1 CONCRETE STRUCTURE (RADWASTE BUILDING)

The Radwaste Building will not flood because all entrances are either above flood level or are protected by appropriately designed sealed doors. All piping penetrations below flood level are sealed to exclude the water and withstand the water pressure. Thus, the Radwaste Building is adequately protected from the Probable Maximum Flood.

SECTION 12.2.5.2 FLOOD PROTECTION DOORS

The seven doors which are closures for all personnel and equipment access openings into the Radwaste Building are identified in this section.

SECTION 12.2.7.1.1 CONCRETE STRUCTURE (PUMPING STN. STRUC)

The walls from the deck and grade El. 565 ± to El. 578 ±
are designed to protect the RHRSW pumps from water and wave forces resulting from the Probable Maximum Flood.

The deck was investigated for a flood to El. 578 creating maximum water pressure on the underside and no pressure on the top. The design method was used with the determination that the deck has a strength capability of 1.4 times that required to resist this flood condition.

SECTION 12.2.7.1.2 PERSONNEL ACCESS DOORS

The personnel access doors provide adequate flood protection against the Probable Maximum Flood for the Residual Heat Removal Service water pumps and are designed to withstand a wind velocity of 300 mph, static water pressure to El. 578.0 (13 feet) and the design basis earthquake.

SECTION 12.2.7.2 INTAKE CHANNEL

The intake channel that connects the pumping station to wheeler reservoir is designed to provide water to the pumps during both of the following conditions:

a. The Probable Maximum Flood condition of the reservoir; and

b. The minimum water level which could conceivably be caused by the breach of Wheeler Dam.

The gate guide cells were also investigated for the unlikely event of impact due to a runaway, fully-loaded coal barge at Maximum Probable Flood. Under these conditions the cells can absorb sufficient energy to stop a barge travelling at 5 mph without overturning. Even if a gate guide cells were to turn over so that it fell across the channel, flow to the intake pumping station would be through either or both remaining gate openings. The smallest opening with the gates closed provides an area of approximately 130 square feet to accommodate the 80 cfs flow needed for shutdown cooling of all three units.

If a barge were to sink in the channel, flow of water to the intake pumping station would not be blocked since normal minimum pool level is El. 550, bottom of the intake channel is El. 523, and a maximum depth of a coal barge is less than 15 feet.

SECTION 12.2.8.1 CONCRETE STRUCTURE (DIESEL GENERATOR BLDG.)

The analysis of the structure assumes the exterior walls fixed at El. 565.5, continuous across the El. 583.5 floor, and pinned at the roof El. 595.0. This frame is analyzed by the moment distribution method using the loading conditions which include Probable Maximum Flood (water level El.572.5) and wave forces.

SECTION 12.2.8.4.1 DESCRIPTION (PORTABLE BULKHEAD)

The portable bulkhead is part of the Diesel-Generator Building flood protection for the Probable Maximum Flood. It will be used to seal any of the four doorways between the Diesel-Generator Rooms and the pipe and electrical corridor. When an exterior door to a Diesel-Generator room is to remain open for an extended period, the rear door in the room is to be removed and the portable bulkhead bolted over the doorway.

SECTION 12.2.9.3.1 DESCRIPTION (FLOOD GATE)

The equipment access flood gate is located on the outside face of the equipment access lock and is part of the Reactor Building flood protection for the Probable Maximum Flood. The gate will normally be in the open position for access into or from the equipment access lock, but may be lowered in the event of impending high water.

SECTION 12.2.9.4.1 DESCRIPTION(WATERTIGHT PERSONNEL ACCESS DOOR)

The watertight personnel access door is located at the south (outside) end of the personnel corridor, which is on the east side of the equipment access lock. The door is a part of the flood protection for the Reactor Building from Probable Maximum Flood.

SECTION 12.2.14 OFFGAS TREATMENT BUILDING

The building is sealed against flood to the elevation of the stairway entrance at El. 568.0. PVC seals are provided in all construction joints in the exterior walls, roof, and base slabs to prevent leakage through the joint.

Appendix 2.4A Browns Ferry Nuclear Plant maximum possible flood

Failure of upstream dam during the adopted maximum possible flood would create maximum flood elevations at Browns Ferry Nuclear plant. For this study maximum headwater levels were determined at each major upstream dam for the project maximum possible flood. This analysis revealed that only the earth portions of the main river dam from Fort Loudon through Guntersville were subject to potential failure as a result of overtopping.

A 6-hour storm which would produce a total of 34.4 inches of rainfall with a maximum 1-hour amount of 16.7 inches was determined to be critical and was used to develop probable maximum flood inflows.

5.3 TRANSPORTATION AND NEARBY FACILITY ACCIDENTS

5.3.1 INTRODUCTION

This is a review in regard to the IPEEE at Browns Ferry Nuclear Plant. This report includes a review of the existing Browns Ferry Nuclear Plant Updated Final Safety Analysis Report (UFSAR).

5.3.2 LICENSING BASES

The Licensing Bases with regard to Nearby Industrial, Military and Transportation Facilities are identified in Section 2.2 of the BFNP UFSAR. Tables 2.2-10 and 2.2-11 include the updated information in regard to chemicals shipped on the river passing the BFNP. The river transportation information is obtained from the US Army Corps of Engineers. The information on test firing of missiles on the Redstone Arsenal was obtained from the Redstone Arsenal Testing and Evaluation Group.

5.3.2.1 REVIEW PLANT-SPECIFIC HAZARD DATA AND LICENSING BASES

The Licensing Bases as discussed in the Browns Ferry Nuclear Plant UFSAR with regard to nearby Facility Accidents and Transportation of Hazardous material close to the plant is discussed in Sections 2.2 including Tables 2.2.10 and 2.2.11 "Hazardous River Traffic that passes Browns Ferry Nuclear Plant." Section 10.12.5.3 discusses the Toxic Gas Protection in the control building as a result of potential gas releases.

The data in Tables 2.2.10 and 2.2.11 provides the information in regard as to the kind of chemicals and the yearly amounts that are barged along the river near the Browns Ferry Nuclear Plant. These tables have been updated since the original licensing basis and represent information through the year 1992.

Section 10.12.5.3 Control Building discusses the Toxic Gas Protection at Browns Ferry Nuclear Plant in regard to the Plant Control Room. The methods of analysis used are those outlined in NRC Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release".

5.3.3 SIGNIFICANT CHANGES SINCE OPERATING LICENSE ISSUANCE

Prior to restart of Unit 2 in May of 1991 NRC issued a SER (Reference 5.3.2) including a supplemental SER (Reference 5.3.3) where NRC concluded that even if all toxic gases transported by barges past Browns Ferry are considered, the probability that a toxic release would result in a severe accident condition exceeding 10 CFR 100 guidelines is sufficiently small and meets the staff's established regulatory position.

The information provided in the UFSAR has been updated to account for an aggregate of toxic gases and it also shows the yearly tonnage of the different chemicals. The calculated probability of a toxic gas invasion of the Browns Ferry Control Room incapacitating the control room operators considering all six toxic gases listed in section 10.12.5.3 of the UFSAR as an aggregate, is less than 1.8 X 10^{-6} events/year (Reference 5.3.1).

Other analyses have also determined there is no significant vulnerability to severe accidents from aircraft crashes or from pipelines, and rail and traffic accidents because the probabilities for significant damage from these events is vanishingly small. Since it has been shown that the aggregate probability (Reference 5.3.1) for toxic gas invasion of the Browns Ferry control room incapacitating the control room operators meets the criterion of SRP Section 2.2.3 probability limit for regulatory consideration, there is no further action required with regard to the IPEEE Section 2.5 review.

5.3.4 DETERMINE IF THE PLANT MEETS 1975 SRP CRITERIA

The following are the acceptance criteria in SRP Sections 2.2.1 and 2.2.2 on Identification of Potential Hazards in Site Vicinity:

1. Data in UFSAR adequately describes the locations and distances of industrial, military, and transportation facilities in the vicinity of the plant, and is in agreement with data obtained from other resources, when available.

2. Descriptions of the nature and extent of activities conducted at nearby facilities, including the products and materials likely to be processed, stored, used, or transported, are adequate to permit identification of possible hazards in Part 3 for evaluation of specific hazards.

3. Where potentially hazardous materials may be processed, stored, used, or transported in the vicinity of the plant, sufficient statistical data on such material are provided to establish a basis for evaluating the potential hazard to the plant.

SRP Section 2.2.3 Evaluation of Potential Accidents, stated the acceptance criteria as:

The identification of design basis events resulting from the presence of hazardous materials in the vicinity of the plant is acceptable if the design basis events include the postulated type of accident for which the expected rate of occurrence of potential exposures in excess of 10 Part 100 guidelines is estimated to exceed NRC objective of approximately 10^{-7} per year. Because of the difficulty of assigning the accurate numerical values to the expected rate of unprecedented potential hazards generally considered in this review plan, judgment must be used as to the acceptability of the overall risk presented.

The probability of occurrence of the initiating events leading to potential consequences in excess of 10 CFR Part 100 exposure guidelines should be estimated using assumptions that are as representative of the specific site as practicable. In addition, because of the low probabilities of the events under consideration, data are often not available to permit accurate calculation of probabilities. Accordingly, the expected rate of occurrence of potential exposures in excess of the 10 CFR Part 100 guidelines of approximately 10^{-6} per year is acceptable if, when combined with reasonable qualitative arguments, the realistic probability can be shown to be lower.

5.3.4.1 COMPLIANCE WITH 1975 SRP ACCEPTANCE CRITERIA

The acceptance criteria of SRP Sections 2.2.1 and 2.2.2 are met because the BFNP UFSAR provides adequate descriptions of the locations and distances of nearby (within five miles of the plant) industrial, military, and transportation facilities, the nature and extent of activities conducted at the identified facilities, the products and materials likely to be processed, stored, or used at the facilities, or transported to and from the facilities, and statistical data or worst case assumptions on the potential hazard from the materials.

5.3.5 SCREENING RESULTS

The evaluation of Nearby Industrial, Transportation and Military Facilities has not resulted in the identification of any vulnerabilities. BFNP conforms to the 1975 SRP Criteria and, therefore the original design basis analysis of potential hazards in the site vicinity is considered adequate and acceptable. Using the progressive screening approach outlined in NUREG 1407 and Supplement 4 to GL 88-20, Nearby Industrial, Transportation and Military Facilities can be screened out for the BFNP IPEEE, and no further analyses of these potential hazards are necessary.

5.3.6 REFERENCES FOR TRANSPORTATION AND NEARBY FACILITY ACCIDENTS

5.3.1 Calculation ND-Q0000-940024 R0, "Probability of Operator Incapacitation from Toxic Chemicals due to a Barge Accident,"

5.3.2 Letter T.M.Ross (NRC) to O.D.Kingsley (TVA), Safety Evaluation of the Effect of Accidental Releases of Hazardous Chemicals Transported by Barges on Control Room Habitability (TAC NOS. 00155, 00156,and 00157), with enclosure, Sept. 14, 1990.

5.3.3 Letter S.A.Varga (NRC) to O.D.Kingsley (TVA), "TVA Backfit Claim Regarding the NRC's Safety Evaluation of the Potential Impact of Hazardous Chemicals Transported by Barges upon Habitability of the Control Room at Browns Ferry," with enclosure, Nov. 20, 1990.

6. LICENSEE PARTICIPATION AND INTERNAL REVIEW TEAM

The maximum benefit from the performance of the IPEEE is obtained if the licensee staff is involved in all aspects of the examination. Such involvement typically provides a more accurate picture of the as-built, as designed facility and helps to integrate knowledge gained about plant equipment and procedures by allowing early ownership of the IPEEE process and results. This section describes the BFNP's involvement in the IPEEE and its review, including major review comments and resolutions.

6.1 IPEEE PROGRAM ORGANIZATION

The IPEEE project team is composed of individuals with diverse backgrounds and experience in PRA, seismic design and analysis, civil/structural engineering and fire analysis expertise chosen from BFNP Site Engineering group of TVA Nuclear organization. Portions of the information presented in the fire analysis report were developed as a result of Browns Ferry's participation in the EPRI Tailored Collaboration Project for the development of the FIVE software. Overall project management is provided by the Lead Civil Engineer of BFNP Site Engineering. Expertise of on-site engineers is utilized throughout the IPEEE study. The names and educational degrees of the BFNP individuals on the IPEEE team are identified in Table 6.1.

6.2 COMPOSITION OF INDEPENDENT REVIEW TEAM

Independent review of the IPEEE review is achieved in three ways. First, the calculation packages which form the bases of the examination and outlier resolutions undergo independent evaluation as part of the calculation process. Second, the IPEEE report is routed within the TVA Nuclear department and also to an outside consultant (Peer review of the fire analysis has been performed by Science Applications International Corporation (SAIC)) with expertise in risk and fire analysis prior to releasing report. This routing includes various engineering disciplines, operations and licensing. The comments received and their resolutions are incorporated as appropriate. Table 6.1 provides the list of preparers, reviewers and expert consultants of this IPEEE submittal.

6.3 AREAS OF REVIEW AND MAJOR COMMENTS

Comments were generated by reviewers of IPEEE report drafts. These comments improved both the technical accuracy and general readability of the report. The major comments received are discussed in this report section along with the comment responses. Although the standard table of contents call for separate sections for comments and responses, the two are grouped together for ease of review. The major comments concern the two areas of the IPEEE analysis, that is, the Internal Fires and High Winds.

Science Applications International Corporation (SAIC) provided peer review of the fire analysis. Comments related to the compartment ignition frequency calculations, heat release

rates, fire model / hot gas layer temperatures, etc. were provided and incorporated into the fire analysis report.

Peer review for High Winds and External Flood is performed by TVA Corporate Engineering. Comments related to the use of valid meteorological data, comparing BFNP criteria with 1975 SRP, address the effect of multiple missile and performing a bounding analysis to determine the probability of core damage due to tornado missile hazard, etc. are provided and incorporated into the High Winds analysis report.

Table 6.1 IPEEE Project Team Participants and Reviewers

Name / Organization	Professional Certification	College Degree	Years of Experience
John R. Glass TVA-BFN Site Engg-Civil	Registered Professional Engineer	B.S. - Mech Engg M.S. - Mech Engg	20
Rick D. Cutsinger TVA-Corporate Engg-Civil	Registered Professional Engineer	B.S. - Civil Engg M.S. - Struct Engg	20
Russell O. Jansen TVA-BFN Site Engg-Civil	Registered Professional Engineer	B.S. - Civil Engg M.S. - Struct Engg	20
Robert O. Enis TVA-Corporate Engg-Civil	------	B.S. - Civil Engg M.S. - Civil Engg	21
Braulio M. Pedroso Jr. TVA-BFN Site Engg-Civil	-------	B.S.-Civil Engg	25
Partha S. Ghosal TVA-BFN Site Engg-Civil	Registered Professional Engineer	B.S. - Civil Engg M.S. - Struct Engg	24
Henry L. Jones TVA-BFN Site Engg-Specialist	------	B.S. - Mech Engg	30
Rashid Abbas TVA-BFN Site Engg-Mech	Registered Professional Engineer	B.S. - Mech Engg	23
Bo G. Rosberg TVA-BFN Site Engg-Mech	Registered Professional Engineer	B.S. - Mech Engg M.S. - Naval Arch	30
J. D. McCamy TVA-BFN Site Engg-Mech	Senior Reactor Operator	B.S. - Physics	20
Anita H. Robeson TVA-BFN Site Engg-Mech	-------	B.S. - Chem Engg	13
Shawn Rodgers PLG - Consultant	-------	B.S. - Math M.S. - Math M.S. - Mech Engg	23
Mitchell A. Waller Delta Prime Inc. - Consultant	Registered Professional Engineer	B.S. - Indus Engg	19
Bijan Nijafi SAIC - Consultant	Registered Professional Engineer	B.S. - Elec Engg M.S. - Nuclear Engg	16
Bill Parkinson SAIC - Consultant	Registered Professional Engineer	B.S. - Nuclear Engg M.S. - Nuclear Engg	16

7. PLANT IMPROVEMENTS AND IMPORTANT SAFETY FEATURES

7.1 PLANT IMPROVEMENTS

No changes are identified as a result of the IPEEE study. There are no potential vulnerabilities to internal fires, high winds, external floods or facilities/transportation accidents identified which reduce the plant's safety margins.

7.2 IMPORTANT SAFETY FEATURES

7.2.1 INTERNAL FIRE

The elimination or limitation of fire sources is generally within the control of plant personnel. Fires affect only one or a limited set of locations (at least initially). For this reason, boundaries between fire zones are important. Although fire initiates at a single source, the "fire event" continues until fuel or oxygen is exhausted, or until suppression occurs. Thus, steps to limit fire spread are important. One aspect of defense to fire events is the existence of multiple divisions of equipment capable of plant shutdown and decay heat removal. The important safety features at BFNP are thus:

1. Good housekeeping to limit fire sources, including storage of combustibles away from safety related equipment and cable and especially in cable spreading rooms.

2. Fire rated compartmentation is provided in accordance with Appendix R requirements. Those separating the reactor, control, and turbine buildings are most safety significant.

3. Fire watches for hot work because most fires are initiated as a result of welding and grinding activities.

4. The current use of IEEE-383 rated fire resistant cable to limit fire related equipment loss and fire progression. Previously installed non-qualified cables are coated with flame retardant material.

5. Wide physical separation of different divisions of safety related equipment and cable. Physical separation is key to equipment survivability and provides assurance of "defense-in-depth" for fires.

7.2.2 HIGH WINDS, EXTERNAL FLOODS AND OTHERS

The important safety features for high winds, external floods, and nearby facility/ transportation accidents are:

1. Safety equipment is located only inside the structures which are designed to withstand or mitigate the above mentioned events.

2. Non-safety related structures are designed such that their failures does not affect the function of safety related structures or equipments.

3. Abnormal operating instructions are in place to take proper precaution against possible external flooding and tornadic events.

8. SUMMARY AND CONCLUSIONS

The major results of the IPEEE study for Internal Fires Analysis, High Winds, Floods, and Transportation/Nearby Facility Accidents are documented in section 4 and 5 of this submittal.

In conclusion, the IPEEE effort confirms that BFNP is well designed and capable of withstanding severe external challenges. After more than two decades of operations, physical condition of the plant, including cleanliness , is good. The findings of this report provide confidence that BFNP has significant safety margin in terms of design.

50-259 Superseded) Per Rev 59 To REP [illegible] 6/2/01 #ML011[illegible]

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TENNESSEE VALLEY AUTHORITY

NUCLEAR

RADIOLOGICAL EMERGENCY PLAN

TVA NUCLEAR

3330978412
CHAT REP
REP-GENERIC PART
020501 58

REVISION LEVEL: 58 REVISION DATE: 2/5/2001

APPROVED: [signature]
Vice President
Engineering & Technical Services

TENNESSEE VALLEY AUTHORITY

Nuclear Power - Radiological Emergency Plan

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TENNESSEE VALLEY AUTHORITY
RADIOLOGICAL EMERGENCY PLAN

REVISION LOG

Rev. & Date		Pages Revised	Reason for Change
0	4/22/88	All	General Revision to convert from individual site-REPs to Common REP with site-specific appendices. Also revises REP approval cycle.
1	12/7/88	A-9, A-14	Revised to add wind speeds to Browns Ferry's Emergency Action Levels (EALs) per memorandum from the NRC dated November 1, 1988.
2	6/5/89	Revised pages marked with an asterisk (*); all pages issued.	Revisions from annual review.
3	6/30/89	A-8, A-9, A-14	Revised Appendix A EALs regarding tornado warnings.
4	9/25/89	i, v, vii, 1, 12, 14, 15-19, 29, 38, 65, 74, A-25, thru A-29, B-30 thru B-33, B-47	Changed emergency action level for tornado at SQN; removed plant communication and CECC Communicator function; clarified training requirement.
5	4/6/90	i.ii, vii, viii, x, 11, 14, 15, 29-31, 35, 37, 44, 48, 52, 55, 56, 62, 63, 65, B-3 thru B-47	Revised to incorporate annual review comments which include: rewrite Section 6; title changes; word changes for clarification; minor changes in implementation; substantial rewording to SQN EALs.
6	5/4/90	v, vi, vii, viii, x, A-1 thru A-45 (A-46 thru A-115 added), B-1 thru B-47 (B-48 thru B-154 added).	Revised to incorporate new EAL format.
7	04/01/91	Revised pages marked with an asterisk (*); all pages issued.	Revisions from annual review and SQN and BFN EAL changes resulting from NRC comments.
8	10/25/91	A-65 and B-92	Revised for clarification of BFN EAL HU14 and SQN EAL HU15.
9	12/17/91	A-29, B-4, and B-5	Revised for clarification of BFN EAL SU7 and SQN EAL FU2.

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RADIOLOGICAL EMERGENCY PLAN

REVISION LOG

Rev.	Date	Pages Revised	Reason for Change
10	5/5/92	3, 11, 12, 15, 17, 37, 52, (App. A) A-6, A-61, A-62, A-75, A-94, A-97, A-100, A-103, A-104, A-115; (App.B) B-4, B-9, B-18, B-22, B-25, B-32, B-37, B-39, B-44, B-45, B-46, B-48, B-49, B-52, B-53, B-54, B-57, B-58, B-59, B-60, B-61, B-62, B-63, B-66, B-71, B-88, B-106, B-119, B-122, B-131, B-132, B-133, B-139, B-140, B-141, B-142, B-148, B-149, B-154	Annual Review.
11	11/25/92	B-4 thru B-8, B-14 thru B-22, B25, thru B-33, B-36, B-38 thru B-39, B-44, B-46, B-60 thru B-67, B-70, B-73, B-73A, B-74, B-76 thru B-77, B-87, B-90, B-92, B-96 thru B-97, B-113 thru B-114, B-128 thru B-129.	EAL Review.
12	04/09/93	18, 35, 38, 42, A-75, A-100, A-105, B-14, B-25, B-37, B-49, B-54, B-59, B-67, B-92, B-139, and B-149.	Annual Review
13	04/30/93	1-2; B-14; B-22.	Changes in response to comments received from NRC in a letter dated April 2, 1993, after review of Revision 12 of the REP.
14	10/04/93	13, 35, 38, A-100, B-139, B-143, B-147, and B-148	Updated in response to NRC comments.
15	01/01/94	x, 1, 3, 5, 48-49, 51, 53, 54, 55, A-43-45, A-47, A-50, A-51, B-60, B-61, B-63 thru B-67, and B-69 thru B-76	Incorporate 10CFR20 and EPA 400 changes.
16	03/21/94	A-94, A-98, A-100, A-101, A-103, and A-104	Annual Review.
17	06/30/94	1, 3, 5, 13, 15, 17-20, 25, 26, 30, 32-33, 35, 37,43, 46, 49-50, 58-60, 64, 67, 69, 74, B-70, B-73, B-73A, B-76, B-83, B-136, B-150 - B-152	Annual Review.

REVISION LOG

Rev. & Date		Pages Revised	Reason for Change
18	10/18/94	1-2, 5, 25, 59, 67, 69-73, A-48, A-50, A-51, B-26, B-34, B-60, B-61, B-63, B-65, B-66, B-67, B-68, B-70 thru B-73, B73A, B-74 thru B-76, B-76A, B-97	Annual Review.
19	1/12/95	A-11	Annual Review.
20	2/22/95	Page 63, B-10, B-61, B-65, B-67, B-70, B-73, B-73A, B-74, B-76	Annual Review.
21	4/12/95	C-1 thru C-218	Issue Appendix C (WBN).
22	7/12/95	C-11, C-15, C-17, C-26, C-27, C-28, C-49, C-53, C-75, C-77, C-95, C-98, C-102, C-107, C-113, C-119, C-124, C-132, C-133, C-134, C-136, C-138, C-151, C-159, C-162, C-163, C-165 thru C-181, C-187 thru C-190, C-194, C-197, C-200 thru C-206, C-211, C-212, C-216	Resolution of NRC Comments.
23	8/14/95	Appendix B, all pages	Issue revised EALs based on NUMARC criteria.
24	9/27/95	2, 11, 12, 14, 17, 18, 21, 22, 23, 24, 35, 42, 46, 49, 51, 57, 58, 59, 60, 61, A-100, A-114, C-7, C-13, C-14, C-15, C-18, C-20, C-24, C-28, C-62, C-64, C-66, C-88, C-115, C-116, C-128, C-129, C-132, C-162, C-163, C-165 - C-180, C-184 - C-190, C-200, C-201	Revised for clarification and organizational changes, add statement that the RAM is responsible for dose authorization for personnel under him, remove references for downgrading emergency classifications, revise PAR descriptions and add PAR diagram, revise BFN staffing chart, revise WBN EALs for accuracy.
25	11/01/95	51 Appendix A, all pages	Revise PAR Diagram. Issue revised EALs based on NUMARC criteria.
26	11/16/95	A-45, A-47, A-87, A-88, A-89	Incorporate BFN Unit 3 temperature limits and incorporate SSI-16 relating to Control Room abandonment.
27	5/31/96	Generic Part, pages 1-63 A145, A149, A150, A151, A154, A-155, B6 thru B8, B19, B23, B-25, B37, B39, B41, B47, B51, B53, B55, B64, B65, B66, B67, B72, B74, B79, B86, B94, B96,	Annual Review.

REVISION LOG

Rev. & Date		Pages Revised	Reason for Change
27 (Continued)		B99, B100, B103, B113, B121, B122, B124, B125, B130, B132, B133, B138, B139, B141, B145, B147, B159, B165, B166, B176, B177, B178, C190, C216	
28	6/18/96	Generic Part, Page 54	Section 14.2.1--Revised the section to reflect revised requirements in 10 CFR 50 Appendix E.
29	7/12/96	B-6, B-7, B-36, B-38, B-40, B-46, B-113, B-159, B-167, B-169, B-170, B-173, B-176, B-177, B-183	Changes made for clarification, routine updates, and correction of minor inaccuracies.
30	7/26/96	53, 54, 55	Clarify environmental monitoring drill requirements. Revise exercise requirements to meet new NRC regulations.
31	9/27/96	47, 57, C-115, C-116, C-135, C-136, C-137, C-190, C-202, C-203, E-2	Remove requirement for directions to REAC/TS be included in site EPIPs. Add instructions for approval at Appendix E revisions. Editorial changes to Appendix C EALs. Update staffing figures to reflect current plant organization and titles. Add 46m reading from met. tower. Replace reference to local and perimeter monitors with radiological monitoring survey points on the plant perimeter. Update references in Appendix E.
32	11/01/96	41, A-141, B-3, B-6, B-7, B-8, B-15, B-23, B-41, B-44, B-45, B-46, B-48, B-51, B-59, B-65, B-71, B-72, B-73, B-76, B-80, B-82, B-84, B-114, B-117, B-118, B-121, B-122, B-133, B-140, B-141, B-144, B-146, B-150, B-154, B-157, B-162, B-163, B-165, B-166, B-169 thru B-188	Revise PAR Diagram, revise BFN Emer. Org. Chart, revise SQN containment rad monitor thresholds for indication of fuel damage, add SQN emergency positions and duties. Editorial and organizational changes.
33	12/17/96	C-203 through C-223	Add listing and responsibilities of key WBN emergency responders, change Athens Community Hospital to Athens Regional Medical Center, renumber pages due to addition of several new pages.

REVISION LOG

Rev. & Date		Pages Revised	Reason for Change
34	2/25/97	19, 23, 24, 25, 26, 27, 37, 49, 50 51, 57, 58, 59, 63, A-135, A-136, A-137, A-138, A-139, A-141, A-142, A-144, A-145, A-148, A-149, A-151, B-159	Remove reference to equipment no longer used, update position and organizational changes, minor editorial changes, add requirement for on-shift dose assessment capability at the sites, revise duties of BFN TSC clerks, add BFN TSC and OSC as locations for emergency supplies, annual review, correct typographical error on pages B-159. All generic pages issued.
35	7/17/97	49, 50, 51, 57, 58, 59, 60, 61, 63, A-1 - A-155, B-113, B-167	Update approval authority and recovery organization. Update new ambulance service name. In Appendix A change references from NUMARC/NESP-007, Rev. 2 to Reg. Guide 1.101, Rev. 3 to refer to NRC document. Revise calculation reference. Revise tables 1.1-G2 and 3.1 per applicable revisions to EOI tables. Option to obtain site dose assessment when CECC not staffed added. Clarify wording of security EALs. General editorial and position updates. In Appendix B remove reference to Knoxville National Weather Service telephone number and correct value of 1.2-RM-90-2 in table 7-2.
36	8/25/97	49, A-45, A-47	Correct title for Manager of Nuclear Licensing. Correct value for U-3 "Main Steam Line Leak Detection High" max safe operating value °F in Table 3.1.
37	12/23/97	C-6, C-7, C-15, C-16, C-21, C-24, C-44, C-46, C-51, C-59, C-115, C-116, C-121, C-141, C-145, C-155, C-166, C-167, C-171, C-172, C-174, C-180, C-184, C-190, C-195, C-219	Change title "Shift Operations Supervision" to "Shift Manager", "SOS" to "SM", "ASOS" to "US." Update clad failure percentage range. Revise EAL 1.3, change "plant computer" to "P-2500 plant computer." Remove reference to the REP from references list on p C-51. Revise EAL 2.4, EAL 5.4, EAL 5.5, EAL 7.1. Change "1-RE-90-421 through 424(B)" to " 1-RE-90-421 through 424." Remove references to TI-30, minor editorial changes.
38	6/9/98 ~~6/4/98~~ *RR*	i, iv, 2, 9, 10, 11, 14, 15, 19, 22, 24, 25, 29, 30, 33, 34, 35, 36, 39, 46, 47, 50, 56, 58, 63	Annual review. Change title "Shift Operations Supervision" to "Shift Manager", "SOS" to "SM", "ASOS" to "US." Remove reference to EIC as the term is no longer used, title changes, organizational changes, minor changes in duties. Update CECC Figure. Update CECC-EPIP descriptions. Remove reference to American Medical Response. Clarify use of rev. log sheet. All generic section pages issued.

REVISION LOG

Rev. & Date		Pages Revised	Reason for Change
39	7-27-98	All Appendix A pages issued.	Change to mode nomenclature due to BFN Generic Tech. Spec. implementation.
40	8-6-98	All Appendix B pages issued.	EALs updated based on redundant information sources in CR. Procedure title updates, editorial and organization title updates. Update emergency center diagrams.
41	10-5-98	All Appendix A pages issued.	Update EAL heat capacity and pressure suppression curves, update PSIG values (page A-12) and update Table 1.1-G2.
42	10-28-98	All Generic Part pages issued.	Editorial and organizational changes. Add state that the Plan Effectiveness determination is done in accordance with 10 CFR 50.54q. Change name of North Park Hospital to Memorial North Park Hospital.
43	12-28-98	All Generic Part, Appendix A, and Appendix B pages issued.	In generic part a statement concerning SAMGs was added. Revisions to Appendix A resulted from a revision to the Emergency Operating Instructions Writer's Guide which required Operations to review and modify EOI and basis information. Revisions to Appendix B were made due to Technical Specification change 98-02.
44	2/22/99	All Generic Part and Appendix A pages issued.	In generic part organization titles revised, PAR diagram revised. In Appendix A, revise radiation monitor values, update references, remove references to RCI-1.1, editorial changes.
45	3/19/99	Appendix B pages issued.	Editorial corrections.
46	4/22/99	Appendix C pages issued.	Editorial and organizational title change. Remove reference to de-escalation, change Site Perimeter (SP) to Exclusion Area Boundary (EAB). Update equipment nomenclature.
47	5/1/99	All Generic Part and Appendix A pages issued.	In Generic Part PAR diagram revised. In Appendix A, revise radiation monitor readings, update references, changes due to outage modification, editorial changes.
48	5/20/99	All Generic Part and Appendix C pages issued.	In Generic Part CECC layout diagram revised. In Appendix C, Onshift Staffing diagram revised.

REVISION LOG

Rev. & Date		Pages Revised	Reason for Change
49	7/13/99	All Appendix B pages issued.	In Appendix B EAL 2.1 statements for NOUE, Alert, and SAE are being returned to the intent status they were in prior to Rev. 40 although the wording has been modified. Additionally, for the SAE EAL (page B-36), the statement that excludes consideration of annunciators that are out of service due to scheduled maintenance or testing activities has been deleted.
50	8/10/99	All Appendix B pages issued.	In Appendix B Radiological Effluent EALs revised to bring into agreement with Rev. 41 to the ODCM.
51	10/12/99	All Appendix C pages issued.	Editorial changes, update earthquake EALs based on equipment change out, change terminology from Lower Toxicity Limit (LTL) to Permissible Exposure Limit (PEL), update Figure C-1, remove Figure C-2.
52	11/17/99	All Appendix B pages issued.	Change terminology from Lower Toxicity Limit (LTL) to Permissible Exposure Limit (PEL). Add Condensate Storage Tank and Addition Equipment Bldg. to lists. Add multi-purpose building to Figure 4-A. Revise EAL 5.1 for Alert and Unusual Event to correspond to new seismic instruction.
53	11/18/99	All Appendix A pages issued.	Change terminology from Lower Toxicity Limit (LTL) to Permissible Exposure Limit (PEL).
54	3/21/00	All Appendix B pages issued.	Editorial changes for clarification on pages B-88 and B-110. Correct typographical error for liquid release alert trigger value in Table 7-1, page B-159.
55	4/28/00	All Appendix A pages issued.	Revise Table 3.1, Unit 3, Temp. Values for RWCU RECIRC PUMP A & B areas. Make clarification, editorial, and format changes due to annual review and self-assessment. Remove reference to Transmission Power System Engineer as this position is no longer used. Remove reference to STD-5.1.
56	6/30/00	All Generic, Appendix B and Appendix C pages issued.	Annual review and self-assessment identified items.
57	8/17/00	All Generic and Appendix C pages issued.	In Generic Part revise PAR diagram. In Appendix C revise EAL 1.1.5.

REVISION LOG

Rev. & Date		Pages Revised	Reason for Change
58	2/5/01	All Generic pages issued.	In Generic Part correct PAR diagram.

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1.0 DEFINITIONS AND ABBREVIATIONS

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

Exceptions:

1. Exercises, drills, emergency information for residents, media training, and offsite emergency response training is 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-EPIPs (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-EPIPs (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.

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 "Red Phone" 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 - Operations 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 (Federal Emergency Management Agency) - An agency of the U.S. Government.

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.

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

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 and 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.

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.

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.

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.

PED - Plan Effectiveness Determination.

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

PNS - Prompt Notification System.

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.

PSS - Public Safety Service.

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

RAA - Radiological Assessment Area of CECC.

RADCON - Radiological Control.

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 a part of 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.

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."

STA - Shift Technical Advisor.

SQN - Sequoyah Nuclear Plant.

SQN-EPIPs (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.

T&CS - Transmission and Customer Services.

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

TLD - Thermoluminescent Dosimeter.

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.

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-EPIPs (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 (NP). The Senior Vice President of NP has delegated the authority for overall program control of the NP-REP to the Manager, Emergency Preparedness.

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.

Site	Appendices
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 EPIPs. Site specific procedures for abnormal and emergency operation and control exist but are not included in the EPIPs. These plant operating procedures are designed to ensure the implementation of the EPIPs.

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 Federal Radiological Emergency Response Plan

The Federal Emergency Management Agency (FEMA) administers the Federal Radiological Emergency Response Plan (FRERP) which is the coordinated Federal Government response to a fixed nuclear power plant facility incident. This emergency plan is activated by either the affected State notifying the Federal Emergency Management Agency, or the utility notifying the NRC of a radiological emergency at a nuclear plant site. The FRERP 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 FEMA. 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 FRERP 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 FRERP 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 FRERP includes the Federal Radiological Monitoring and Assessment Plan for use by Federal agencies with radiological monitoring and assessment capabilities. The CECC Director is the TVA person authorized to request Federal assistance. Such a request from TVA will be made to NRC.

FIGURE 2-1

PRINCIPAL ORGANIZATIONAL RESPONSIBILITIES

	Local	State	TVA
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 and 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.

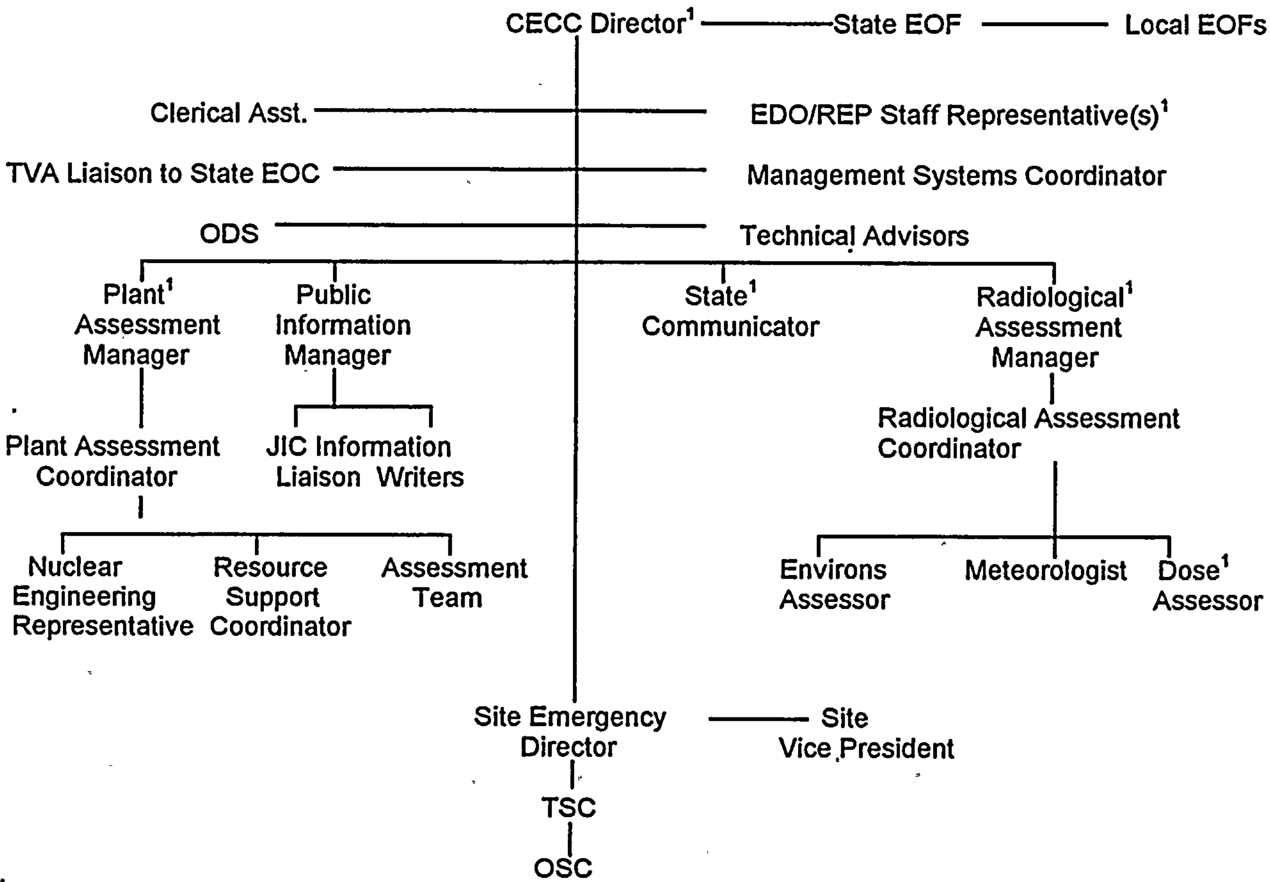
3.1 Onsite Organization

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. 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.

FIGURE 3-1

OFFSITE EMERGENCY ORGANIZATION



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

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-EPIPs.

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, FEMA, 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 the RadCon Superintendent recommends against the entry; and initiating long-term 24-hour per day accident mitigation operations.

The SED makes recommendations for protective actions (if necessary) to the State and local agencies through the ODS prior to the CECC being staffed (this responsibility can be transferred only to the CECC Director). The SED is also responsible for determining the emergency classification as well as the approval of 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 RADCON.

3.1.6 Radiological Control (RADCON) Manager

The RADCON Manager is responsible for assessing inplant and onsite radiological conditions; directing the onsite RADCON activities; coordinating additional RADCON 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 RADCON support to maintenance teams; and making final recommendation to the SED for personnel entry to radiologically hazardous environments.

3.1.7 Chemistry and Environmental Manager

Chemistry and Environmental 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 advise 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 and reporting recommended protective actions, determined by the site, to the appropriate State emergency organization. In addition, the ODS notifies appropriate TVA offsite emergency personnel. In the event of the initiation of the event as a General Emergency, he is required to notify the appropriate local response agencies.

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 develop 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 NP 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 and SEOC. He will ensure the TVA Chief 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.

He 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 RADCON 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 RADCON 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.

3.2.10 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 Systems

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 FRERP (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 EPIPs. 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 minor events which could develop into or be indicative of more serious conditions which are not yet fully realized.

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.

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 progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public.

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 progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity.

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-EPIPs, 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 EPIPs. These EALs are used as thresholds for determining the emergency classifications. EAL's are presented in the site-specific appendix. The EALs are reviewed annually by the appropriate State.

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 EPIPs.

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 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 carrying pagers. 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 unusual event by the SED. The ODS records the details of the event in accordance with the appropriate EPIP.

2. The ODS notifies and relays the information to the State within 15 minutes of declaration of the event. The ODS also notifies and relays the information to the EDO and CECC directors.

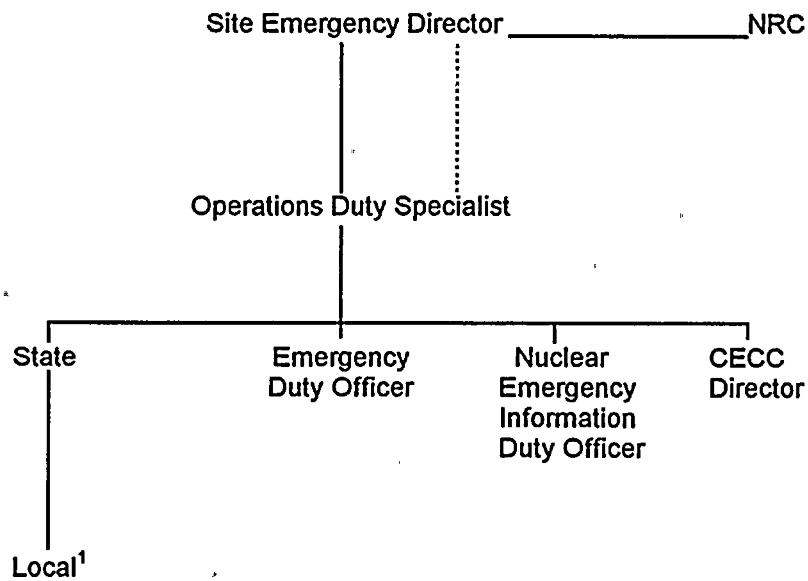
3. The EDO keeps the CECC Directors and the Nuclear Emergency Information Duty Officer informed of the situation as necessary.

4. The Nuclear Emergency Information Duty Officer notifies the Site Communications Consultant; General Manager Communications; and Media Relations.

5. The SED augments plant shift personnel as necessary to initiate corrective or protective actions.

FIGURE 5-1

CHAINS OF NOTIFICATION



[1]The ODS also notifies the local governments if the initial classification is a General Emergency

------ Verification

5.2.2 Alert

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

1. The ODS in Chattanooga is notified of the incident by the SED. The ODS records the details of the event in accordance with the appropriate EPIP.

2. The ODS makes the notifications described in section 5.2.1.

3. The CECC is staffed.

4. Environmental sampling teams may be dispatched.

5. The TSC and the OSC are activated.

6. The situation is analyzed and any appropriate corrective or preventive actions initiated.

7. 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 (airbourne, waterbourne, 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 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.

8. The JIC may be activated.

9. Periodic media releases are provided.

10. 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 actions 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 recommendation to the State are required upon declaration of General Emergency.

2. If this is the initial classification, the ODS 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/Plant Manager of the Affected Site

4. CECC Director

5. Radiological Assessment Manager

6. Plant 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 Radwaste 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 loudspeaker paging system.

6.3 Plant Loudspeaker Paging

This system may be accessed from the plant telephone system and is used for normal plant operations and to instruct and notify personnel during an emergency. Also, 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.

Other communications include those not provided by TVA, but that reside at TVA facilities. These are the ENS and HPN telephones (NRC FTS 2000 System) which 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 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 and to the following locations:

Radiological Control.

Technical Support Center.

Control Room at each plant.

CECC in Chattanooga.

6.7 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, nonemergency 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 TVA Chief Spokesperson

The TVA Chief 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 Vice President, Communications

Vice President, 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 Shared Resources Communications

Shared Resources 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) Shared Resources 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

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 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 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 recommended protective actions to the appropriate State officials.

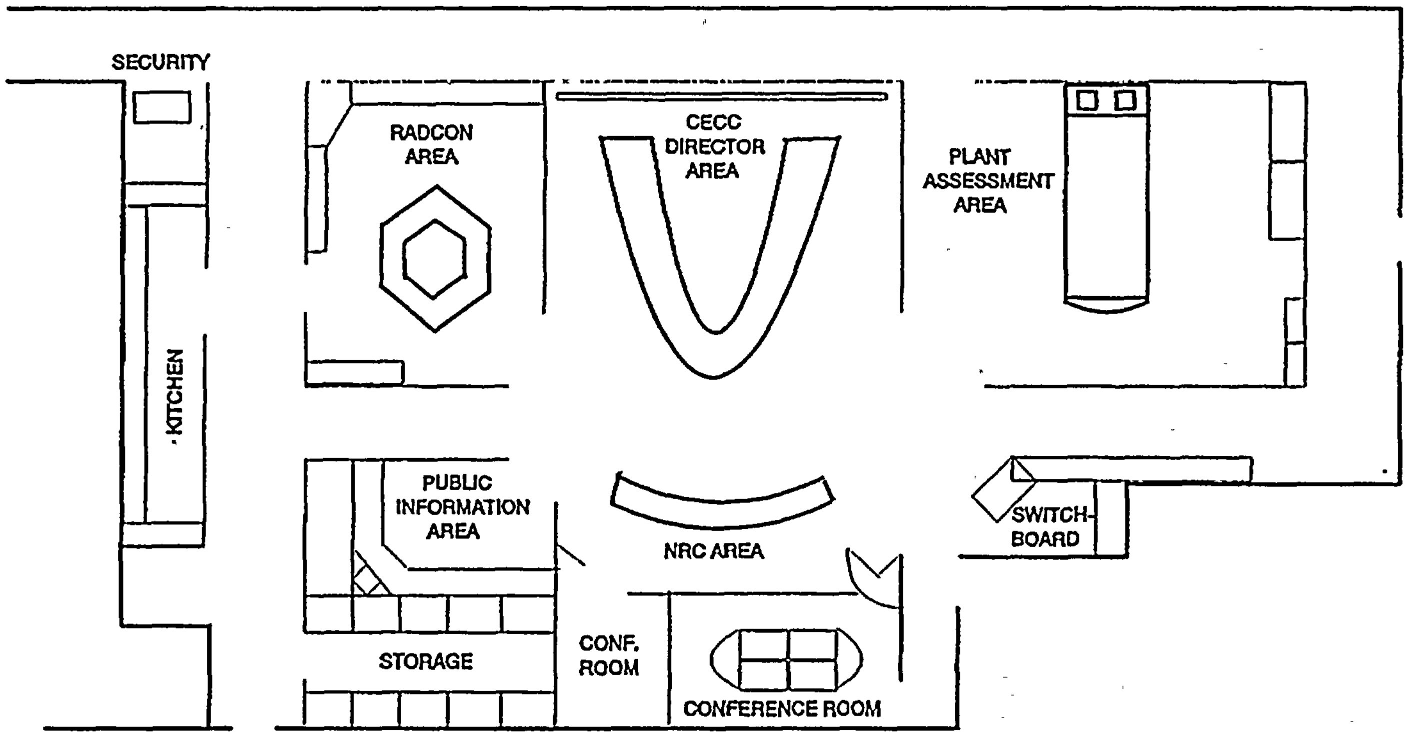
The CECC staff will coordinate with all other TVA emergency centers to ensure an effective TVA effort in response to an accident situation. The CECC staff will also provide an accurate description of the emergency situation for TVA management and public information. In addition, the CECC will coordinate 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.

FIGURE 8-1

CENTRAL EMERGENCY CONTROL CENTER



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.

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 specific predetermined monitoring points to collect data. This data is passed by radio to the RMCC and relayed to the CECC for integration and analysis with the plant data.

Facilities at the RMCC include radio and telephone communications, tie-in to the Hard Copy Transmitting System, and necessary desks, tables, and chairs. Maps of the 10-mile EPZ and the 50-Mile EPZ with preselected radiological sampling and monitoring points are located at the RMCC. The preselected mobile laboratory locations are also reflected on a map at the RMCC.

8.4 Joint Information Center (JIC)

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

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

8.5 Prompt Notification System (PNS)

Each site has a PNS capable of warning the public within the plume exposure EPZ of a serious event. Specific PNS information is provided in the site-specific appendix.

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-EPIPs. 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-EPIPs 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-EPIPs, 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-EPIPs. 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-EPIPs and are summarized below.

CECC-EPIP-1 - CENTRAL EMERGENCY CONTROL CENTER ALERT, SITE AREA EMERGENCY, AND GENERAL EMERGENCY

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

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

CECC-EPIP-4 - OPERATIONS DUTY SPECIALIST PROCEDURE FOR SITE AREA EMERGENCY

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

CECC-EPIP-5 - OPERATIONS DUTY SPECIALIST PROCEDURE FOR GENERAL EMERGENCY

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

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 - WATER MANAGEMENT RADIOLOGICAL EMERGENCY PROCEDURES

Cancelled - Pertinent parts moved to CECC-EPIP-8.

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 - ENVIRONMENTAL RESEARCH AND SERVICES RADIOLOGICAL EMERGENCY PROCEDURES

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-13- TERMINATION AND RECOVERY

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

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- JOINT INFORMATION CENTER ACTIVATION, SHIFT CHANGE, AND DEACTIVATION

Cancelled - Combined with CECC EPIP-14.

CECC-EPIP-16- CENTRAL EMERGENCY CONTROL CENTER INFORMATION STAFF ACTIVATION, SHIFT CHANGE, AND DEACTIVATION

* Cancelled - Combined with CECC EPIP-14.

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 CORE 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- CECC TRAINING REQUIREMENTS

Cancelled - replaced by TRN-30

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-EPIPs.

These vehicles are dispatched for environmental monitoring for Site Area Emergency and General Emergency classes. They may be deployed for the Notification of Unusual Event and Alert classes, 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 EPIPs.

For the Site Area Emergency, and General Emergency classes, 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 RadCon 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

A mobile radioanalytical laboratory can be dispatched to the site to be the central point for receipt of samples and for detailed field analysis. Samples obtained by the sampling teams may be returned to the WARL, which has the capability to perform further quantitative and qualitative analysis. The mobile radiological laboratory and the WARL are available at all times and can be operated 24 hours per day.

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 (from other TVA nuclear plants and the National Weather Service first-order stations). Each procedure has an accompanying statement of reliability.

9.2.3.3 Real Time and Forecast Meteorological Data

A meteorologist in the CECC has the responsibility for providing meteorological information to CECC Staff. The dose assessors use this meteorological information to project offsite doses. The meteorological support actions and projection of doses are discussed in detail in CECC-EPIPs. Plume positions are plotted on a site area map.

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 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-EPIPs. 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 resuspended 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

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 preassigned areas to be accounted for and await further instructions from the SED. The preassigned 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-EPIPs.

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 RADCON 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-EPIPs.

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 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-EPIPs 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 to recommend protective action 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 projected radiological conditions offsite. They will provide a coordinated recommendation for a specific protective action considering both plant and offsite conditions. 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-EPIPs 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 appropriate state plan).

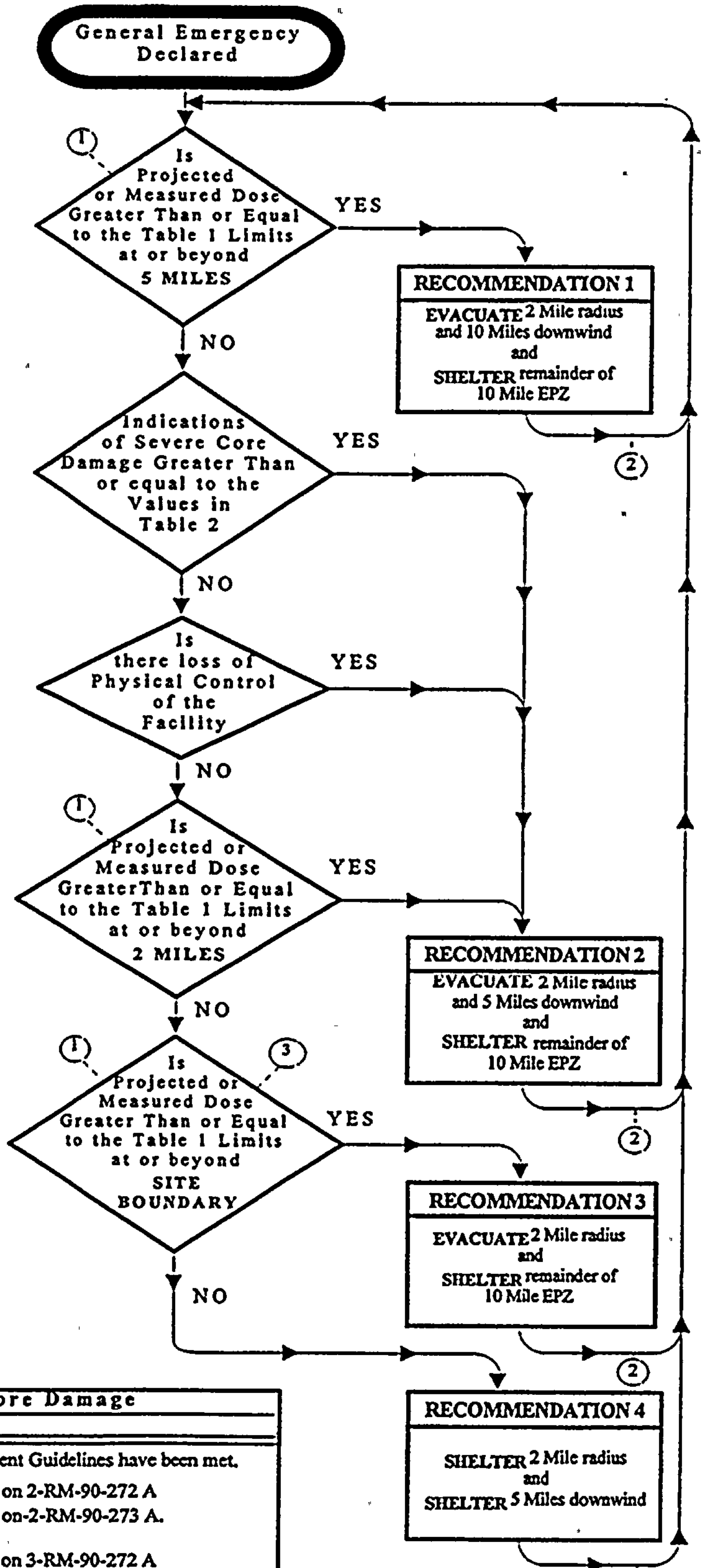
FIGURE 10-1

NOTES
(1) IF Conditions Are not known, Then Answer No.
(2) CONTINUE ASSESSMENT Modify protective actions based on available plant and field monitoring infomation. Locate and evacuate additional localized hot spots.
BFN ONLY
(3) When Dose Assessment Projections OR Actual Measured Exposures are not known, a stack release rate of ≥ 1.3 E+11 μCi/sec noble gas can be utilized to meet the condition of 1 REM/hr External Dose at the site boundary.

TABLE 1 RADIOACTIVITY RELEASE DOSE	
TYPE	LIMIT
Measured	3.9 E-6 μCi/cc of Iodine-131
	1 REM/hr External Dose
Projected	1REM TEDE
	5 REM Thyroid CDE

WBN TABLE 2 - Severe Core Damage INDICATIONS
1. Containment radiation monitor reading on 1-RE-90-271 and 272 equal to or greater than 7.4 E+1 R/hr (74 R/hr). or Containment radiation monitor reading on 1-RE-90-273 and 274 equal to or greater than 5.9 E+1 R/hr (59 R/hr).
2. Reactor Coolant Activity of ≥ 300 μCi/gm Dose Equivalent Iodine-131.
3. Inadequate core cooling as indicated by "red" path from core cooling status tree.
4. Core exit TCs greater than 1200 °F

SQN TABLE 2- Severe Core Damage INDICATIONS
1. Containment radiation monitor reading on RM-90-271 and 272 equal to or greater than 2.8 E+1 REM/hr (28 REM/hr). or Containment radiation monitor reading on RM-90-273 and 274 equal to or greater than 2.9 E+1 REM/hr (29 REM/hr).
2. Reactor Coolant Activity of ≥ 300 μCi/gm Dose Equivalent Iodine-131.
3. Inadequate core cooling as indicated by "red" or "orange" path from core cooling status tree.



BFN TABLE 2 - Severe Core Damage INDICATIONS
*1. Entry conditons for Severe Accident Management Guidelines have been met.
*2. Unit 2 - Drywell Radiation Exceeds 345 R/HR on 2-RM-90-272 A
* or 164 R/HR on-2-RM-90-273 A.
* Unit 3 - Drywell Radiation Exceeds 106 R/HR on 3-RM-90-272 A
* or 164 R/HR on 3-RM-90-273 A.
3. Equilibrium Reactor Coolant Activity of ≥ 300 μCi/gm Dose Equivalent Iodine.

NEWPAR 01/19/01

* Revision

11.0 RADIOLOGICAL PROTECTION

The RADCON Section at the site is responsible for all RADCON 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 TVAN SPP-5.1, "Radiation Protection Plan." 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 TVAN 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-EPIPs 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 TLD badges are issued and read in accordance with the site RCIs. The electronic dosimeters can be read at any time and the TLD badges can be read by the Central Dosimetry Processing section at SQN. Dose records are maintained on each monitored individual by a computer.

TVAN 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 EPIPs. Drinking water and eating controls are established in TVAN SPP-5.1.

FIGURE 11-1

EMERGENCY WORKER DOSE GUIDANCE

TEDE Dose	Condition
5 rem	All, maintain dose ALARA
10 rad	Protection of valuable property when lower dose not practicable.
Greater than 10 rad	Lifesaving or protection of large populations when lower dose not practicable.

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 unmeasureable 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 TVAN 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 RAD 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), October 15, 1991, page 2-18.

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-EPIPs. 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. RADCON 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 RADCON.

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

13.1.1 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.

13.1.2 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:

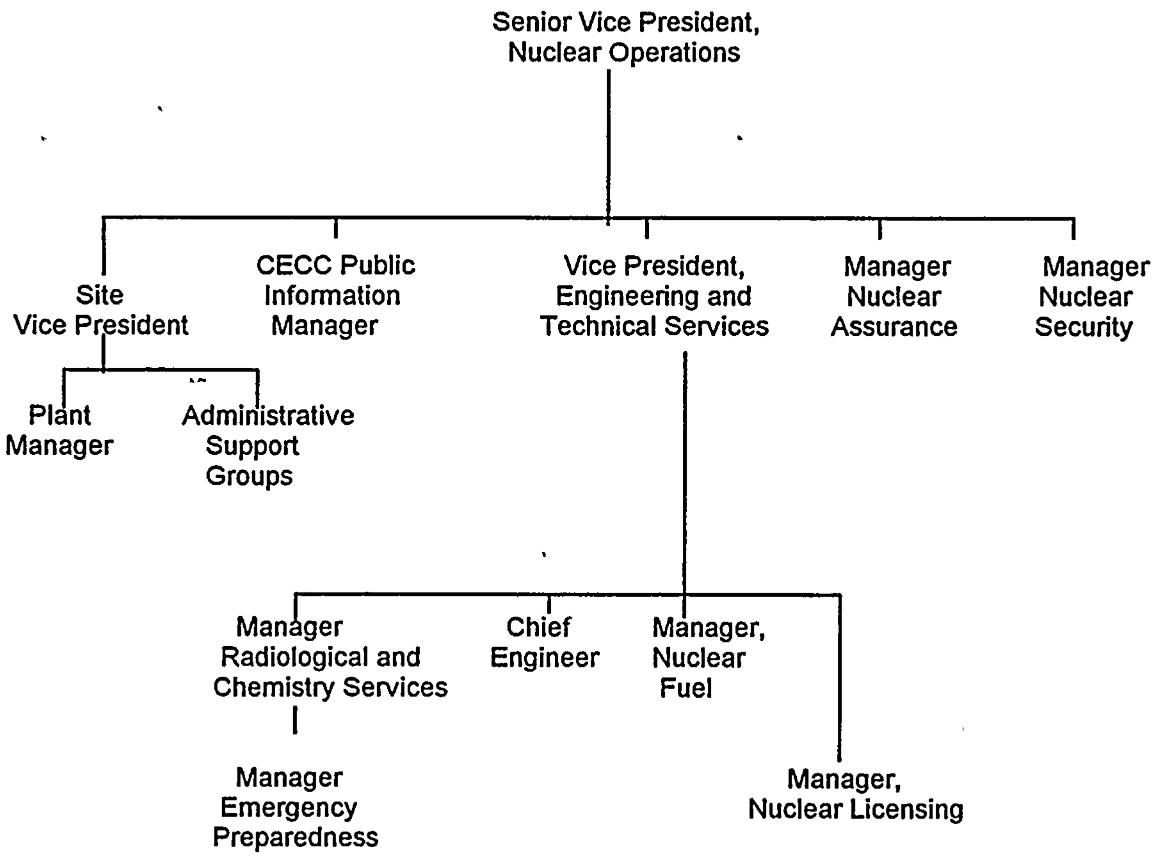
1. Stability of the reactor shutdown condition, i.e., successful progress toward a cold shutdown condition.
2. Integrity of the reactor containment building.
3. Operability of engineered safety systems and decontamination facilities.
4. The availability and operability of a heat sink.
5. The integrity 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 Senior Vice President, Nuclear Operations - 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 Senior Vice President Nuclear Operations.

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

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

13.2.4 Vice President, Engineering and Technical Services - Will manage needed services to the site in the areas of Engineering, Licensing activities, QA activities, Security, and Emergency Preparedness.

13.2.5 CECC Public Information Manager - Acts as an interface between TVA and the news media. They assist the Senior Vice President, Nuclear Operations, 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 Chief Engineer - Will provide needed technical and engineering services to the site.

13.2.7 Manager, Nuclear Fuels - Will provide needed technical services to the site. Technical services available include fuel management and core analysis, core performance, nuclear fuel control and accountability, and startup support.

13.2.8 Manager, Radiological and Chemistry Services - Will provide corporate level guidance and needed radiological services as requested. Services include technical support, dose assessment, and environmental monitoring. Will provide technical support and environs sampling assistance as requested by the State. Will provide technical assistance to the site in the areas of chemistry and environmental issues.

13.2.9 Manager, Emergency Preparedness - Will provide assistance in any aspects of emergency preparedness plans, procedures, coordination, and implementation.

13.2.10 Manager, Nuclear Security - Will provide technical assistance to the site in the area of security.

13.2.11 Manager of Nuclear Licensing - Will provide technical assistance in NRC licensing activities.

13.2.12 Manager of Nuclear Assurance - Will provide technical assistance in Quality Assurance activities.

13.2.13 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-EPIPs 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 Senior Vice President, Nuclear Operations, or his designee will service 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, 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 RadCon Drills

RadCon 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 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 pm 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 Nuclear Quality Assurance, INPO, NRC, FEMA, 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 followup 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

Nuclear Training/plant will provide training in emergency procedures to all permanent plant personnel and applicable nonplant 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.

Nickajack Fire Training Academy 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.

Engineering and Technical Services is responsible for providing agreement hospital and ambulance support training. The sites are responsible for providing fire support training, with assistance from Engineering and Technical Services 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, or, whenever an entire page has been added or substantially changed, the change will be denoted by a statement at the bottom of the page.

Formal site 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 responsible site management allowing time for site 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 President, Nuclear Power. All changes to the NP-REP will be approved by the Vice President, Engineering and Technical Services, or his designee.

16.1.4 Distribution

Each NP-REP, its additions, and revisions will be authorized by an approval form and distributed by Administrative Support and Procedures.

Administrative Support and Procedures 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.

Administrative Support and Procedures 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 EPIPs

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:

EPIPs	Approving Authority
CECC	Vice President, Engineering & Technical Services
BFN	Plant Manager, BFN
SQN	Plant Manager, SQN
WBN	Plant Manager, WBN

FIGURE 16-1

UPDATE PROCEDURE FOR NP-REP AND APPENDICES

Draft Proposed Change
to NP-REP or Appendix

Plan Effectiveness Determination (PED)
(In Accordance With 10 CFR 50.54q)

Site Review in Accordance With Plant Approved
Procedures
(Sites Are Not Required
To Review and Approve
Other Site's Appendix)

EP
Review/Coordination and Approval

Vice President, Engineering and Technical Services, or Representative

No — Significant Change
As Determined by
PED — Yes

NRC Review
and Approval

Administrative Support and Procedures
Issues Revision

16.2.2 Periodic Review

The EPIPs are reviewed annually for accuracy, completeness, operational readiness, and compliance with existing regulations by the responsible organization listed below. This review is initiated by Engineering and Technical Services and results are documented.

EPIPs	Organization
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 an 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.

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 by the word "All" under Revised Pages on the cover page.

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. Administrative Support and Procedures will issue the changes to other document holders in accordance with Administrative Support and Procedures document control practices.

FIGURE 16-2

UPDATE PROCEDURE FOR EPIPs

Proposed Change to CECC-EPIP

Review, Coordination, and Approval

Engineering and Technical Services Review and Approval

Administrative Support and Procedures Issues CECC-EPIP Revision to Onsite and Offsite Document Holders

Proposed Permanent Change to Site-EPIP

Review and Coordination

Site Review and Approval According to Approved Site Procedures

Site Issues EPIP Revision to Onsite Document Holders

Administrative Support and Procedures Issues Procedure To Offsite Document Holders

16.2.3.3 CECC-EPIP Changes

In addition to the change mechanism depicted in figure 16-2, in order to ensure that minor changes (e.g., personnel changes, phone numbers, etc.) are rapidly implemented, pen-and-ink changes may be made by the responsible organizations to their procedures in documents which they possess. Pen-and-ink changes will be authorized by the approving authority and documented. The initials of the individual making the pen-and ink change and the date of the change will be clearly marked in the margin adjacent to the change. Such changes will be immediately followed by a formal change request.

16.2.4 Distribution

Each CECC-EPIP or revision will be authorized by an approval form and distributed by Administrative Support and Procedures. 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 with in two weeks to Administrative Support and Procedures.

Each revision will be accompanied by a revised cover page for that procedure. Administrative Support and Procedures 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

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

Nuclear 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 Medical Service, Chattanooga, TN
Athens-Limestone Ambulance Service, Athens, AL
Rhea County 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 Medical Center, Chattanooga, TN
Memorial North Park Hospital, 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:

Dayton City Fire Department, TN
Rhea County Fire Department, TN
Soddy Daisy Fire Department, TN
Clements 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. TVA agrees to provide two-hours notice prior to any such use and to pay the college for facilities and services provided.

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.

50-259 Superseded Per Rev. 60 To REP Dtd 6/11/01
#011660462

TENNESSEE VALLEY AUTHORITY

NUCLEAR

RADIOLOGICAL EMERGENCY PLAN

TVA NUCLEAR

1238956555
CHAT REP
REP-GENERIC PART
033001 59

REVISION LEVEL: 59 REVISION DATE: 3/30/01

APPROVED: [signature]
Vice President
Engineering & Technical Services

TENNESSEE VALLEY AUTHORITY

Nuclear Power - Radiological Emergency Plan

List of Effective Pages

This List of Effective Pages must be retained with the Nuclear Power Radiological Emergency Plan.

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TENNESSEE VALLEY AUTHORITY
RADIOLOGICAL EMERGENCY PLAN

REVISION LOG

Rev. & Date		Pages Revised	Reason for Change
0	4/22/88	All	General Revision to convert from individual site-REPs to Common REP with site-specific appendices. Also revises REP approval cycle.
1	12/7/88	A-9, A-14	Revised to add wind speeds to Browns Ferry's Emergency Action Levels (EALs) per memorandum from the NRC dated November 1, 1988.
2	6/5/89	Revised pages marked with an asterisk (*); all pages issued.	Revisions from annual review.
3	6/30/89	A-8, A-9, A-14	Revised Appendix A EALs regarding tornado warnings.
4	9/25/89	i, v, vii, 1, 12, 14, 15-19, 29, 38, 65, 74, A-25, thru A-29, B-30 thru B-33, B-47	Changed emergency action level for tornado at SQN; removed plant communication and CECC Communicator function; clarified training requirement.
5	4/6/90	i.ii, vii, viii, x, 11, 14, 15, 29-31, 35, 37, 44, 48, 52, 55, 56, 62, 63, 65, B-3 thru B-47	Revised to incorporate annual review comments which include: rewrite Section 6; title changes; word changes for clarification; minor changes in implementation; substantial rewording to SQN EALs.
6	5/4/90	v, vi, vii, viii, x, A-1 thru A-45 (A-46 thru A-115 added), B-1 thru B-47 (B-48 thru B-154 added).	Revised to incorporate new EAL format.
7	04/01/91	Revised pages marked with an asterisk (*); all pages issued.	Revisions from annual review and SQN and BFN EAL changes resulting from NRC comments.
8	10/25/91	A-65 and B-92	Revised for clarification of BFN EAL HU14 and SQN EAL HU15.
9	12/17/91	A-29, B-4, and B-5	Revised for clarification of BFN EAL SU7 and SQN EAL FU2.

REVISION LOG

Rev. & Date		Pages Revised	Reason for Change
10	5/5/92	3, 11, 12, 15, 17, 37, 52, (App. A) A-6, A-61, A-62, A-75, A-94, A-97, A-100, A-103, A-104, A-115; (App.B) B-4, B-9, B-18, B-22, B-25, B-32, B-37, B-39, B-44, B-45, B-46, B-48, B-49, B-52, B-53, B-54, B-57, B-58, B-59, B-60, B-61, B-62, B-63, B-66, B-71, B-88, B-106, B-119, B-122, B-131, B-132, B-133, B-139, B-140, B-141, B-142, B-148, B-149, B-154	Annual Review.
11	11/25/92	B-4 thru B-8, B-14 thru B-22, B25, thru B-33, B-36, B-38 thru B-39, B-44, B-46, B-60 thru B-67, B-70, B-73, B-73A, B-74, B-76 thru B-77, B-87, B-90, B-92, B-96 thru B-97, B-113 thru B-114, B-128 thru B-129.	EAL Review.
12	04/09/93	18, 35, 38, 42, A-75, A-100, A-105, B-14, B-25, B-37, B-49, B-54, B-59, B-67, B-92, B-139, and B-149.	Annual Review
13	04/30/93	1-2; B-14; B-22.	Changes in response to comments received from NRC in a letter dated April 2, 1993, after review of Revision 12 of the REP.
14	10/04/93	13, 35, 38, A-100, B-139, B-143, B-147, and B-148	Updated in response to NRC comments.
15	01/01/94	x, 1, 3, 5, 48-49, 51, 53, 54, 55, A-43-45, A-47, A-50, A-51, B-60, B-61, B-63 thru B-67, and B-69 thru B-76	Incorporate 10CFR20 and EPA 400 changes.
16	03/21/94	A-94, A-98, A-100, A-101, A-103, and A-104	Annual Review.
17	06/30/94	1, 3, 5, 13, 15, 17-20, 25, 26, 30, 32-33, 35, 37,43, 46, 49-50, 58-60, 64, 67, 69, 74, B-70, B-73, B-73A, B-76, B-83, B-136, B-150 - B-152	Annual Review.

REVISION LOG

Rev. & Date		Pages Revised	Reason for Change
18	10/18/94	1-2, 5, 25, 59, 67, 69-73, A-48, A-50, A-51, B-26, B-34, B-60, B-61, B-63, B-65, B-66, B-67, B-68, B-70 thru B-73, B73A, B-74 thru B-76, B-76A, B-97	Annual Review.
19	1/12/95	A-11	Annual Review.
20	2/22/95	Page 63, B-10, B-61, B-65, B-67, B-70, B-73, B-73A, B-74, B-76	Annual Review.
21	4/12/95	C-1 thru C-218	Issue Appendix C (WBN).
22	7/12/95	C-11, C-15, C-17, C-26, C-27, C-28, C-49, C-53, C-75, C-77, C-95, C-98, C-102, C-107, C-113, C-119, C-124, C-132, C-133, C-134, C-136, C-138, C-151, C-159, C-162, C-163, C-165 thru C-181, C-187 thru C-190, C-194, C-197, C-200 thru C-206, C-211, C-212, C-216	Resolution of NRC Comments.
23	8/14/95	Appendix B, all pages	Issue revised EALs based on NUMARC criteria.
24	9/27/95	2, 11, 12, 14, 17, 18, 21, 22, 23, 24, 35, 42, 46, 49, 51, 57, 58, 59, 60, 61, A-100, A-114, C-7, C-13, C-14, C-15, C-18, C-20, C-24, C-28, C-62, C-64, C-66, C-88, C-115, C-116, C-128, C-129, C-132, C-162, C-163, C-165 - C-180, C-184 - C-190, C-200, C-201	Revised for clarification and organizational changes, add statement that the RAM is responsible for dose authorization for personnel under him, remove references for downgrading emergency classifications, revise PAR descriptions and add PAR diagram, revise BFN staffing chart, revise WBN EALs for accuracy.
25	11/01/95	51 Appendix A, all pages	Revise PAR Diagram. Issue revised EALs based on NUMARC criteria.
26	11/16/95	A-45, A-47, A-87, A-88, A-89	Incorporate BFN Unit 3 temperature limits and incorporate SSI-16 relating to Control Room abandonment.
27	5/31/96	Generic Part, pages 1-63 A145, A149, A150, A151, A154, A-155, B6 thru B8, B19, B23, B-25, B37, B39, B41, B47, B51, B53, B55, B64, B65, B66, B67, B72, B74, B79, B86, B94, B96,	Annual Review.

REVISION LOG

Rev. & Date		Pages Revised	Reason for Change
27 (Continued)		B99, B100, B103, B113, B121, B122, B124, B125, B130, B132, B133, B138, B139, B141, B145, B147, B159, B165, B166, B176, B177, B178, C190, C216	
28	6/18/96	Generic Part, Page 54	Section 14.2.1--Revised the section to reflect revised requirements in 10 CFR 50 Appendix E.
29	7/12/96	B-6, B-7, B-36, B-38, B-40, B-46, B-113, B-159, B-167, B-169, B-170, B-173, B-176, B-177, B-183	Changes made for clarification, routine updates, and correction of minor inaccuracies.
30	7/26/96	53, 54, 55	Clarify environmental monitoring drill requirements. Revise exercise requirements to meet new NRC regulations.
31	9/27/96	47, 57, C-115, C-116, C-135, C-136, C-137, C-190, C-202, C-203, E-2	Remove requirement for directions to REAC/TS be included in site EPIPs. Add instructions for approval at Appendix E revisions. Editorial changes to Appendix C EALs. Update staffing figures to reflect current plant organization and titles. Add 46m reading from met. tower. Replace reference to local and perimeter monitors with radiological monitoring survey points on the plant perimeter. Update references in Appendix E.
32	11/01/96	41, A-141, B-3, B-6, B-7, B-8, B-15, B-23, B-41, B-44, B-45, B-46, B-48, B-51, B-59, B-65, B-71, B-72, B-73, B-76, B-80, B-82, B-84, B-114, B-117, B-118, B-121, B-122, B-133, B-140, B-141, B-144, B-146, B-150, B-154, B-157, B-162, B-163, B-165, B-166, B-169 thru B-188	Revise PAR Diagram, revise BFN Emer. Org. Chart, revise SQN containment rad monitor thresholds for indication of fuel damage, add SQN emergency positions and duties. Editorial and organizational changes.
33	12/17/96	C-203 through C-223	Add listing and responsibilities of key WBN emergency responders, change Athens Community Hospital to Athens Regional Medical Center, renumber pages due to addition of several new pages.

REVISION LOG

Rev. & Date		Pages Revised	Reason for Change
34	2/25/97	19, 23, 24, 25, 26, 27, 37, 49, 50 51, 57, 58, 59, 63, A-135, A-136, A-137, A-138, A-139, A-141, A-142, A-144, A-145, A-148, A-149, A-151, B-159	Remove reference to equipment no longer used, update position and organizational changes, minor editorial changes, add requirement for on-shift dose assessment capability at the sites, revise duties of BFN TSC clerks, add BFN TSC and OSC as locations for emergency supplies, annual review, correct typographical error on pages B-159. All generic pages issued.
35	7/17/97	49, 50, 51, 57, 58, 59, 60, 61, 63, A-1 - A-155, B-113, B-167	Update approval authority and recovery organization. Update new ambulance service name. In Appendix A change references from NUMARC/NESP-007, Rev. 2 to Reg. Guide 1.101, Rev. 3 to refer to NRC document. Revise calculation reference. Revise tables 1.1-G2 and 3.1 per applicable revisions to EOI tables. Option to obtain site dose assessment when CECC not staffed added. Clarify wording of security EALs. General editorial and position updates. In Appendix B remove reference to Knoxville National Weather Service telephone number and correct value of 1.2-RM-90-2 in table 7-2.
36	8/25/97	49, A-45, A-47	Correct title for Manager of Nuclear Licensing. Correct value for U-3 "Main Steam Line Leak Detection High" max safe operating value °F in Table 3.1.
37	12/23/97	C-6, C-7, C-15, C-16, C-21, C-24, C-44, C-46, C-51, C-59, C-115, C-116, C-121, C-141, C-145, C-155, C-166, C-167, C-171, C-172, C-174, C-180, C-184, C-190, C-195, C-219	Change title "Shift Operations Supervision" to "Shift Manager", "SOS" to "SM", "ASOS" to "US." Update clad failure percentage range. Revise EAL 1.3, change "plant computer" to "P-2500 plant computer." Remove reference to the REP from references list on p C-51. Revise EAL 2.4, EAL 5.4, EAL 5.5, EAL 7.1. Change "1-RE-90-421 through 424(B)" to " 1-RE-90-421 through 424." Remove references to TI-30, minor editorial changes.
38	6/9/98 ~~6/4/98~~ *RR*	i, iv, 2, 9, 10, 11, 14, 15, 19, 22, 24, 25, 29, 30, 33, 34, 35, 36, 39, 46, 47, 50, 56, 58, 63	Annual review. Change title "Shift Operations Supervision" to "Shift Manager", "SOS" to "SM", "ASOS" to "US." Remove reference to EIC as the term is no longer used, title changes, organizational changes, minor changes in duties. Update CECC Figure. Update CECC-EPIP descriptions. Remove reference to American Medical Response. Clarify use of rev. log sheet. All generic section pages issued.

REVISION LOG

Rev. & Date		Pages Revised	Reason for Change
39	7-27-98	All Appendix A pages issued.	Change to mode nomenclature due to BFN Generic Tech. Spec. implementation.
40	8-6-98	All Appendix B pages issued.	EALs updated based on redundant information sources in CR. Procedure title updates, editorial and organization title updates. Update emergency center diagrams.
41	10-5-98	All Appendix A pages issued.	Update EAL heat capacity and pressure suppression curves, update PSIG values (page A-12) and update Table 1.1-G2.
42	10-28-98	All Generic Part pages issued.	Editorial and organizational changes. Add state that the Plan Effectiveness determination is done in accordance with 10 CFR 50.54q. Change name of North Park Hospital to Memorial North Park Hospital.
43	12-28-98	All Generic Part, Appendix A, and Appendix B pages issued.	In generic part a statement concerning SAMGs was added. Revisions to Appendix A result from a revision to the Emergency Operating Instructions Writer's Guide which required Operations to review and modify EOI and basis information. Revisions to Appendix B were made due to Technical Specification change 98-02.
44	2/22/99	All Generic Part and Appendix A pages issued.	In generic part organization titles revised, PAR diagram revised. In Appendix A, revise radiation monitor values, update references, remove references to RCI-1.1, editorial changes.
45	3/19/99	Appendix B pages issued.	Editorial corrections.
46	4/22/99	Appendix C pages issued.	Editorial and organizational title change. Remove reference to de-escalation, change Site Perimeter (SP) to Exclusion Area Boundary (EAB). Update equipment nomenclature.
47	5/1/99	All Generic Part and Appendix A pages issued.	In Generic Part PAR diagram revised. In Appendix A, revise radiation monitor readings, update references, changes due to outage modification, editorial changes.
48	5/20/99	All Generic Part and Appendix C pages issued.	In Generic Part CECC layout diagram revis In Appendix C, Onshift Staffing diagram revised.

REVISION LOG

Rev.	Date	Pages Revised	Reason for Change
49	7/13/99	All Appendix B pages issued.	In Appendix B EAL 2.1 statements for NOUE, Alert, and SAE are being returned to the intent status they were in prior to Rev. 40 although the wording has been modified. Additionally, for the SAE EAL (page B-36), the statement that excludes consideration of annunciators that are out of service due to scheduled maintenance or testing activities has been deleted.
50	8/10/99	All Appendix B pages issued.	In Appendix B Radiological Effluent EALs revised to bring into agreement with Rev. 41 to the ODCM.
51	10/12/99	All Appendix C pages issued.	Editorial changes, update earthquake EALs based on equipment change out, change terminology from Lower Toxicity Limit (LTL) to Permissible Exposure Limit (PEL), update Figure C-1, remove Figure C-2.
52	11/17/99	All Appendix B pages issued.	Change terminology from Lower Toxicity Limit (LTL) to Permissible Exposure Limit (PEL). Add Condensate Storage Tank and Addition Equipment Bldg. to lists. Add multi-purpose building to Figure 4-A. Revise EAL 5.1 for Alert and Unusual Event to correspond to new seismic instruction.
53	11/18/99	All Appendix A pages issued.	Change terminology from Lower Toxicity Limit (LTL) to Permissible Exposure Limit (PEL).
54	3/21/00	All Appendix B pages issued.	Editorial changes for clarification on pages B-88 and B-110. Correct typographical error for liquid release alert trigger value in Table 7-1, page B-159.
55	4/28/00	All Appendix A pages issued.	Revise Table 3.1, Unit 3, Temp. Values for RWCU RECIRC PUMP A & B areas. Make clarification, editorial, and format changes due to annual review and self-assessment. Remove reference to Transmission Power System Engineer as this position is no longer used. Remove reference to STD-5.1.
56	6/30/00	All Generic, Appendix B and Appendix C pages issued.	Annual review and self-assessment identified items.
57	8/17/00	All Generic and Appendix C pages issued.	In Generic Part revise PAR diagram. In Appendix C revise EAL 1.1.5.

REVISION LOG

Rev. & Date		Pages Revised	Reason for Change
58	2/5/01	All Generic pages issued.	In Generic Part correct PAR diagram.
59	3/30/01	All Generic and App. A pages issued.	In Generic Part issue new PAR diagram. In App. A update references, adjust EAL release rates to match new dose code mixes, remove reference to evacuation signal, minor updates.

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1.0 DEFINITIONS AND ABBREVIATIONS

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

Exceptions:

1. Exercises, drills, emergency information for residents, media training, and offsite emergency response training is 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-EPIPs (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-EPIPs (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.

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 "Red Phone" 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 - Operations 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 (Federal Emergency Management Agency) - An agency of the U.S. Government.

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.

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

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 and 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.

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.

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.

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.

PED - Plan Effectiveness Determination.

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

PNS - Prompt Notification System.

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.

PSS - Public Safety Service.

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

RAA - Radiological Assessment Area of CECC.

RADCON - Radiological Control.

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 a part of 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.

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."

STA - Shift Technical Advisor.

SQN - Sequoyah Nuclear Plant.

SQN-EPIPs (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.

T&CS - Transmission and Customer Services.

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

TLD - Thermoluminescent Dosimeter.

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.

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-EPIPs (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 (NP). The Senior Vice President of NP has delegated the authority for overall program control of the NP-REP to the Manager, Emergency Preparedness.

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.

Site	Appendices
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 EPIPs. Site specific procedures for abnormal and emergency operation and control exist but are not included in the EPIPs. These plant operating procedures are designed to ensure the implementation of the EPIPs.

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 Federal Radiological Emergency Response Plan

The Federal Emergency Management Agency (FEMA) administers the Federal Radiological Emergency Response Plan (FRERP) which is the coordinated Federal Government response to a fixed nuclear power plant facility incident. This emergency plan is activated by either the affected State notifying the Federal Emergency Management Agency, or the utility notifying the NRC of a radiological emergency at a nuclear plant site. The FRERP 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 FEMA. 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 FRERP 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 FRERP 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 FRERP includes the Federal Radiological Monitoring and Assessment Plan for use by Federal agencies with radiological monitoring and assessment capabilities. The CECC Director is the TVA person authorized to request Federal assistance. Such a request from TVA will be made to NRC.

FIGURE 2-1

PRINCIPAL ORGANIZATIONAL RESPONSIBILITIES

	Local	State	TVA
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 and 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.

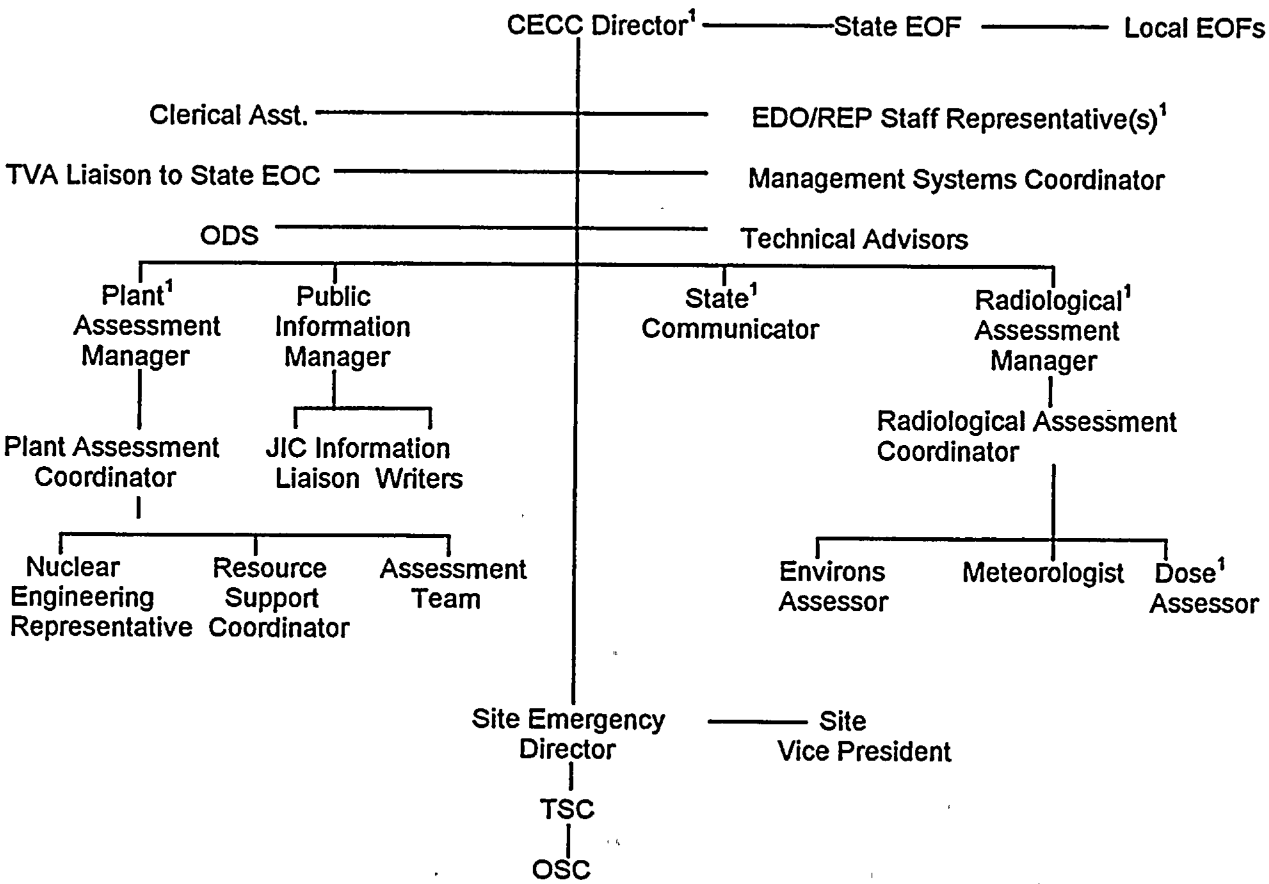
3.1 Onsite Organization

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. 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.

FIGURE 3-1

OFFSITE EMERGENCY ORGANIZATION



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

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-EPIPs.

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, FEMA, 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 the RadCon Superintendent recommends against the entry; and initiating long-term 24-hour per day accident mitigation operations.

The SED makes recommendations for protective actions (if necessary) to the State and local agencies through the ODS prior to the CECC being staffed (this responsibility can be transferred only to the CECC Director). The SED is also responsible for determining the emergency classification as well as the approval of 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 RADCON.

3.1.6 Radiological Control (RADCON) Manager

The RADCON Manager is responsible for assessing inplant and onsite radiological conditions; directing the onsite RADCON activities; coordinating additional RADCON 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 RADCON support to maintenance teams; and making final recommendation to the SED for personnel entry to radiologically hazardous environments.

3.1.7 Chemistry and Environmental Manager

Chemistry and Environmental 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 advise 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 and reporting recommended protective actions, determined by the site, to the appropriate State emergency organization. In addition, the ODS notifies appropriate TVA offsite emergency personnel. In the event of the initiation of the event as a General Emergency, he is required to notify the appropriate local response agencies.

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 develop 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 NP 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 and SEOC. He will ensure the TVA Chief 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.

He 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 RADCON 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 RADCON 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.

3.2.10 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 Systems

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 FRERP (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 EPIPs. 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 minor events which could develop into or be indicative of more serious conditions which are not yet fully realized.

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.

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 progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public.

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 progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity.

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-EPIPs, 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 EPIPs. These EALs are used as thresholds for determining the emergency classifications. EAL's are presented in the site-specific appendix. The EALs are reviewed annually by the appropriate State.

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 EPIPs.

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 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 carrying pagers. 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 unusual event by the SED. The ODS records the details of the event in accordance with the appropriate EPIP.

2. The ODS notifies and relays the information to the State within 15 minutes of declaration of the event. The ODS also notifies and relays the information to the EDO and CECC directors.

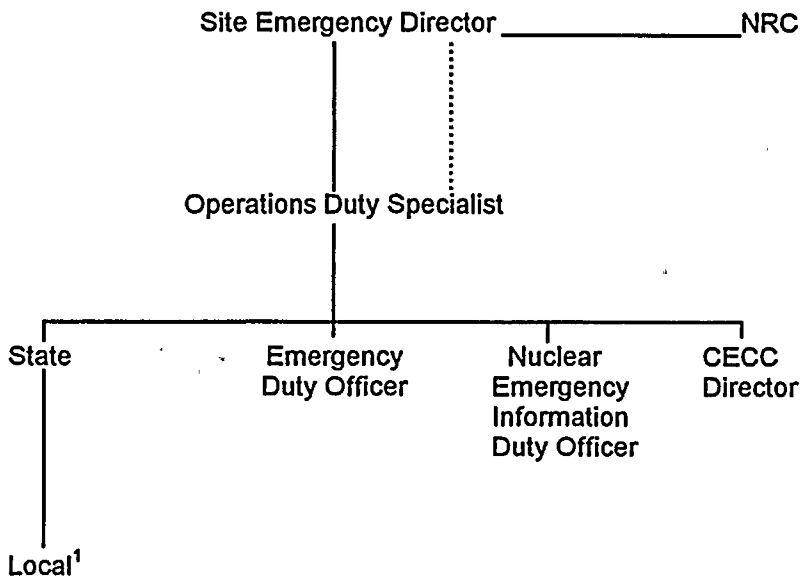
3. The EDO keeps the CECC Directors and the Nuclear Emergency Information Duty Officer informed of the situation as necessary.

4. The Nuclear Emergency Information Duty Officer notifies the Site Communications Consultant; General Manager Communications; and Media Relations.

5. The SED augments plant shift personnel as necessary to initiate corrective or protective actions.

FIGURE 5-1

CHAINS OF NOTIFICATION



[1]The ODS also notifies the local governments if the initial classification is a General Emergency

------ Verification

5.2.2 Alert

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

1. The ODS in Chattanooga is notified of the incident by the SED. The ODS records the details of the event in accordance with the appropriate EPIP.

2. The ODS makes the notifications described in section 5.2.1.

3. The CECC is staffed.

4. Environmental sampling teams may be dispatched.

5. The TSC and the OSC are activated.

6. The situation is analyzed and any appropriate corrective or preventive actions initiated.

7. 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 (airbourne, waterbourne, 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 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.

8. The JIC may be activated.

9. Periodic media releases are provided.

10. 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 actions 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 recommendation to the State are required upon declaration of General Emergency.

2. If this is the initial classification, the ODS 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/Plant Manager of the Affected Site

4. CECC Director

5. Radiological Assessment Manager

6. Plant 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 Radwaste 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 loudspeaker paging system.

6.3 Plant Loudspeaker Paging

This system may be accessed from the plant telephone system and is used for normal plant operations and to instruct and notify personnel during an emergency. Also, 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.

Other communications include those not provided by TVA, but that reside at TVA facilities. These are the ENS and HPN telephones (NRC FTS 2000 System) which 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 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 and to the following locations:

Radiological Control.

Technical Support Center.

Control Room at each plant.

CECC in Chattanooga.

6.7 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, nonemergency 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 TVA Chief Spokesperson

The TVA Chief 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 Vice President, Communications

Vice President, 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 Shared Resources Communications

Shared Resources 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) Shared Resources 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

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 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 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 recommended protective actions to the appropriate State officials.

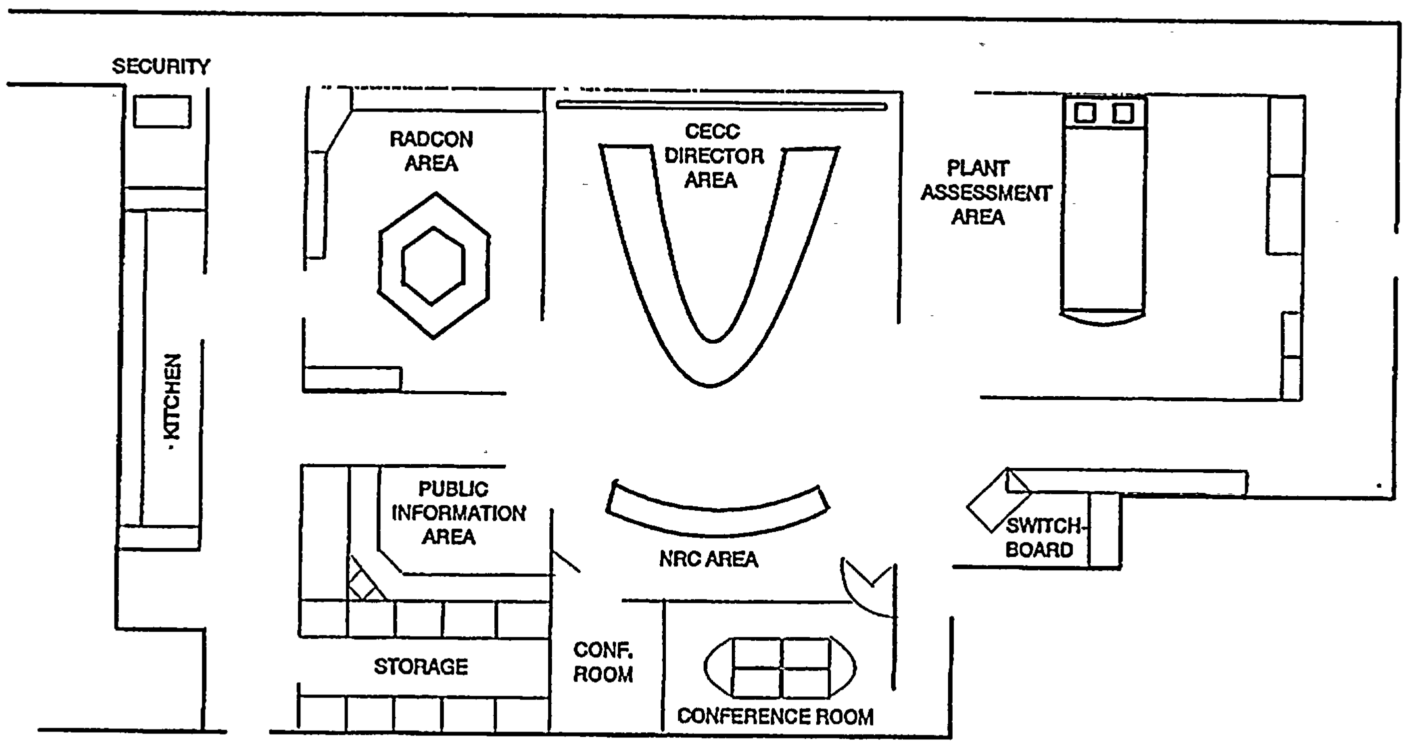
The CECC staff will coordinate with all other TVA emergency centers to ensure an effective TVA effort in response to an accident situation. The CECC staff will also provide an accurate description of the emergency situation for TVA management and public information. In addition, the CECC will coordinate 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.

FIGURE 8-1

CENTRAL EMERGENCY CONTROL CENTER



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.

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 specific predetermined monitoring points to collect data. This data is passed by radio to the RMCC and relayed to the CECC for integration and analysis with the plant data.

Facilities at the RMCC include radio and telephone communications, tie-in to the Hard Copy Transmitting System, and necessary desks, tables, and chairs. Maps of the 10-mile EPZ and the 50-Mile EPZ with preselected radiological sampling and monitoring points are located at the RMCC. The preselected mobile laboratory locations are also reflected on a map at the RMCC.

8.4 Joint Information Center (JIC)

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

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

8.5 Prompt Notification System (PNS)

Each site has a PNS capable of warning the public within the plume exposure EPZ of a serious event. Specific PNS information is provided in the site-specific appendix.

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-EPIPs. 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-EPIPs 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-EPIPs, 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-EPIPs. 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-EPIPs and are summarized below.

CECC-EPIP-1 - <u>CENTRAL EMERGENCY CONTROL CENTER ALERT, SITE AREA EMERGENCY, AND GENERAL EMERGENCY</u>

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 - <u>OPERATIONS DUTY SPECIALIST PROCEDURE FOR NOTIFICATION OF UNUSUAL EVENT</u>

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 - <u>OPERATIONS DUTY SPECIALIST PROCEDURE FOR ALERT</u>

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

CECC-EPIP-4 - <u>OPERATIONS DUTY SPECIALIST PROCEDURE FOR SITE AREA EMERGENCY</u>

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

CECC-EPIP-5 - <u>OPERATIONS DUTY SPECIALIST PROCEDURE FOR GENERAL EMERGENCY</u>

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

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

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 - <u>CECC RADIOLOGICAL ASSESSMENT STAFF PROCEDURE FOR ALERT, SITE AREA EMERGENCY, AND GENERAL EMERGENCY</u>

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 - <u>DOSE ASSESSMENT STAFF ACTIVITIES DURING NUCLEAR PLANT RADIOLOGICAL EMERGENCIES</u>

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 - <u>EMERGENCY ENVIRONMENTAL RADIOLOGICAL MONITORING PROCEDURES</u>

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 - <u>WATER MANAGEMENT RADIOLOGICAL EMERGENCY PROCEDURES</u>

Cancelled - Pertinent parts moved to CECC-EPIP-8.

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 - ENVIRONMENTAL RESEARCH AND SERVICES RADIOLOGICAL EMERGENCY PROCEDURES

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-13- TERMINATION AND RECOVERY

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

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- JOINT INFORMATION CENTER ACTIVATION, SHIFT CHANGE, AND DEACTIVATION

Cancelled - Combined with CECC EPIP-14.

CECC-EPIP-16- CENTRAL EMERGENCY CONTROL CENTER INFORMATION STAFF ACTIVATION, SHIFT CHANGE, AND DEACTIVATION

* Cancelled - Combined with CECC EPIP-14.

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 CORE 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- CECC TRAINING REQUIREMENTS

Cancelled - replaced by TRN-30

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-EPIPs.

These vehicles are dispatched for environmental monitoring for Site Area Emergency and General Emergency classes. They may be deployed for the Notification of Unusual Event and Alert classes, 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 EPIPs.

For the Site Area Emergency, and General Emergency classes, 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 RadCon 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

A mobile radioanalytical laboratory can be dispatched to the site to be the central point for receipt of samples and for detailed field analysis. Samples obtained by the sampling teams may be returned to the WARL, which has the capability to perform further quantitative and qualitative analysis. The mobile radiological laboratory and the WARL are available at all times and can be operated 24 hours per day.

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 (from other TVA nuclear plants and the National Weather Service first-order stations). Each procedure has an accompanying statement of reliability.

9.2.3.3 Real Time and Forecast Meteorological Data

A meteorologist in the CECC has the responsibility for providing meteorological information to CECC Staff. The dose assessors use this meteorological information to project offsite doses. The meteorological support actions and projection of doses are discussed in detail in CECC-EPIPs. Plume positions are plotted on a site area map.

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 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-EPIPs. 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 resuspended 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

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 preassigned areas to be accounted for and await further instructions from the SED. The preassigned 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-EPIPs.

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 RADCON 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-EPIPs.

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 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-EPIPs 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 to recommend protective action 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 projected radiological conditions offsite. They will provide a coordinated recommendation for a specific protective action considering both plant and offsite conditions. 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-EPIPs 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 appropriate state plan).

FIGURE 10-1
PROTECTIVE ACTION RECOMMENDATIONS

Note 1: If conditions are unknown utilizing the flowchart, then answer NO.

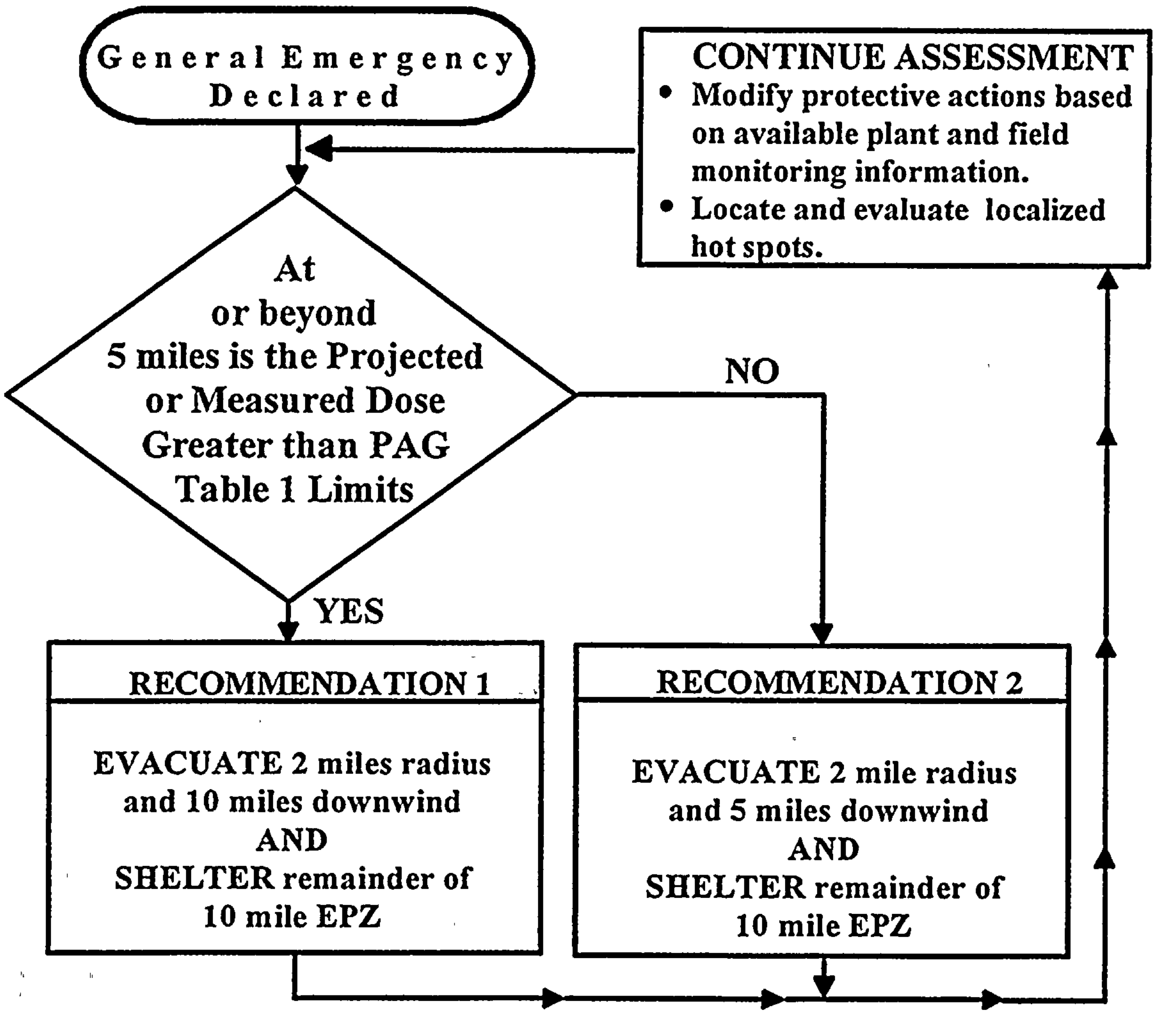


TABLE 1 Protective Action Guides	
TYPE	LIMIT
Measured	3.9E-6 microCi/cc of Iodine 131 or 1 REM/hr External Dose
Projected	1 REM TEDE or 5 REM Thyroid CDE

Entire Page Revised

11.0 RADIOLOGICAL PROTECTION

The RADCON Section at the site is responsible for all RADCON 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 TVAN SPP-5.1, "Radiation Protection Plan." 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 TVAN 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-EPIPs 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 TLD badges are issued and read in accordance with the site RCIs. The electronic dosimeters can be read at any time and the TLD badges can be read by the Central Dosimetry Processing section at SQN. Dose records are maintained on each monitored individual by a computer.

TVAN 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 EPIPs. Drinking water and eating controls are established in TVAN SPP-5.1.

FIGURE 11-1

EMERGENCY WORKER DOSE GUIDANCE

TEDE Dose	Condition
5 rem	All, maintain dose ALARA
10 rad	Protection of valuable property when lower dose not practicable.
Greater than 10 rad	Lifesaving or protection of large populations when lower dose not practicable.

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 unmeasureable 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 TVAN 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 RAD 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), October 15, 1991, page 2-18.

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-EPIPs. 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. RADCON 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 RADCON.

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

13.1.1 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.

13.1.2 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:

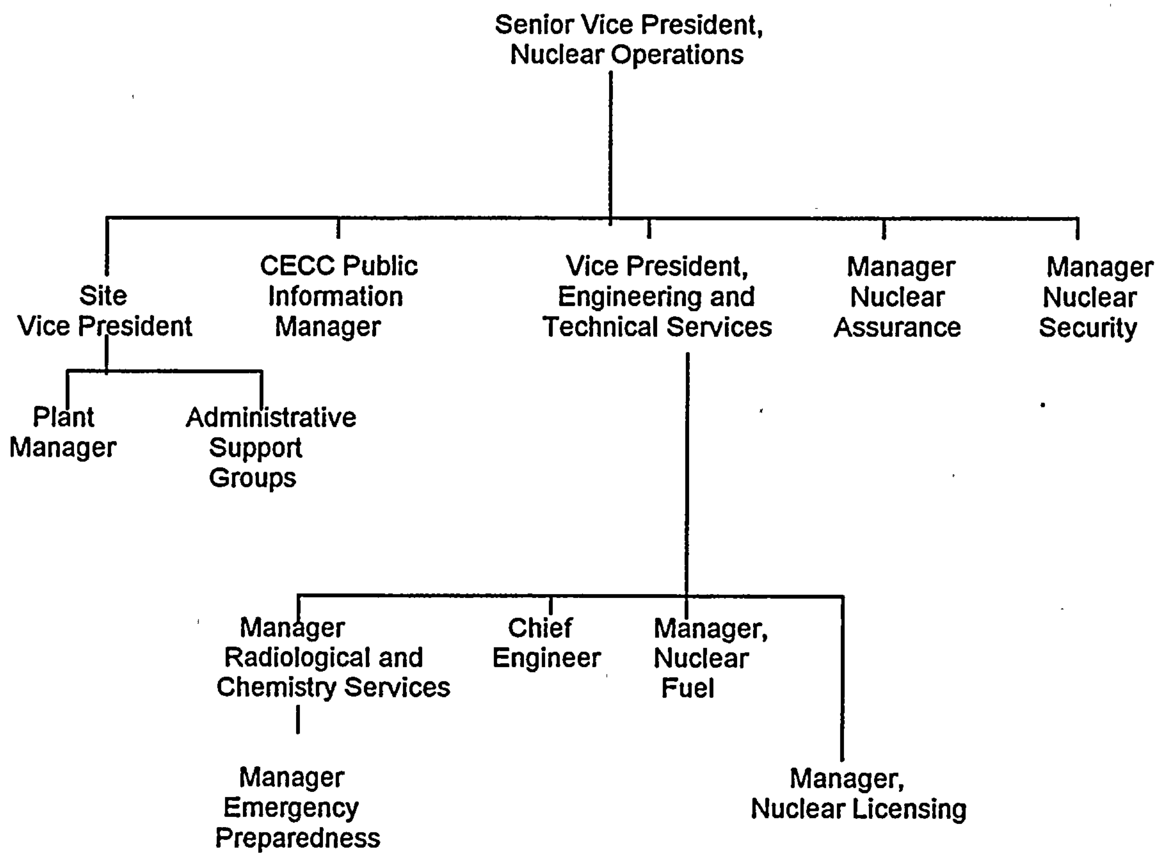
1. Stability of the reactor shutdown condition, i.e., successful progress toward a cold shutdown condition.
2. Integrity of the reactor containment building.
3. Operability of engineered safety systems and decontamination facilities.
4. The availability and operability of a heat sink.
5. The integrity 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 Senior Vice President, Nuclear Operations - 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 Senior Vice President Nuclear Operations.

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

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

13.2.4 Vice President, Engineering and Technical Services - Will manage needed services to the site in the areas of Engineering, Licensing activities, QA activities, Security, and Emergency Preparedness.

13.2.5 CECC Public Information Manager - Acts as an interface between TVA and the news media. They assist the Senior Vice President, Nuclear Operations, 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 Chief Engineer - Will provide needed technical and engineering services to the site.

13.2.7 Manager, Nuclear Fuels - Will provide needed technical services to the site. Technical services available include fuel management and core analysis, core performance, nuclear fuel control and accountability, and startup support.

13.2.8 Manager, Radiological and Chemistry Services - Will provide corporate level guidance and needed radiological services as requested. Services include technical support, dose assessment, and environmental monitoring. Will provide technical support and environs sampling assistance as requested by the State. Will provide technical assistance to the site in the areas of chemistry and environmental issues.

13.2.9 Manager, Emergency Preparedness - Will provide assistance in any aspects of emergency preparedness plans, procedures, coordination, and implementation.

13.2.10 Manager, Nuclear Security - Will provide technical assistance to the site in the area of security.

13.2.11 Manager of Nuclear Licensing - Will provide technical assistance in NRC licensing activities.

13.2.12 Manager of Nuclear Assurance - Will provide technical assistance in Quality Assurance activities.

13.2.13 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-EPIPs 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 Senior Vice President, Nuclear Operations, or his designee will service 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, 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 RadCon Drills

RadCon 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 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 pm 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 Nuclear Quality Assurance, INPO, NRC, FEMA, 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 followup 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

Nuclear Training/plant will provide training in emergency procedures to all permanent plant personnel and applicable nonplant 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.

Nickajack Fire Training Academy 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.

Engineering and Technical Services is responsible for providing agreement hospital and ambulance support training. The sites are responsible for providing fire support training, with assistance from Engineering and Technical Services 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 Page 1 -or- Rev. 1 Page A-1 Rev. 1	NP-REP Appendix A

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, or, whenever an entire page has been added or substantially changed, the change will be denoted by a statement at the bottom of the page.

Formal site 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 responsible site management allowing time for site 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 President, Nuclear Power. All changes to the NP-REP will be approved by the Vice President, Engineering and Technical Services, or his designee.

16.1.4 Distribution

Each NP-REP, its additions, and revisions will be authorized by an approval form and distributed by Administrative Support and Procedures.

Administrative Support and Procedures 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.

Administrative Support and Procedures 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 EPIPs

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:

EPIPs	Approving Authority
CECC	Vice President, Engineering & Technical Services
BFN	Plant Manager, BFN
SQN	Plant Manager, SQN
WBN	Plant Manager, WBN

FIGURE 16-1

UPDATE PROCEDURE FOR NP-REP AND APPENDICES

Draft Proposed Change
to NP-REP or Appendix

Plan Effectiveness Determination (PED)
(In Accordance With 10 CFR 50.54q)

Site Review in Accordance With Plant Approved
Procedures
(Sites Are Not Required
To Review and Approve
Other Site's Appendix)

EP
Review/Coordination and Approval

Vice President, Engineering and Technical Services, or Representative

No — Significant Change
As Determined by
PED — Yes

NRC Review
and Approval

Administrative Support and Procedures
Issues Revision

16.2.2 Periodic Review

The EPIPs are reviewed annually for accuracy, completeness, operational readiness, and compliance with existing regulations by the responsible organization listed below. This review is initiated by Engineering and Technical Services and results are documented.

EPIPs	Organization
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 an 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.

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 by the word "All" under Revised Pages on the cover page.

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. Administrative Support and Procedures will issue the changes to other document holders in accordance with Administrative Support and Procedures document control practices.

FIGURE 16-2

UPDATE PROCEDURE FOR EPIPs

Proposed Change to CECC-EPIP

Review, Coordination, and Approval

Engineering and Technical Services Review and Approval

Administrative Support and Procedures Issues CECC-EPIP Revision to Onsite and Offsite Document Holders

Proposed Permanent Change to Site-EPIP

Review and Coordination

Site Review and Approval According to Approved Site Procedures

Site Issues EPIP Revision to Onsite Document Holders

Administrative Support and Procedures Issues Procedure To Offsite Document Holders

16.2.3.3 CECC-EPIP Changes

In addition to the change mechanism depicted in figure 16-2, in order to ensure that minor changes (e.g., personnel changes, phone numbers, etc.) are rapidly implemented, pen-and-ink changes may be made by the responsible organizations to their procedures in documents which they possess. Pen-and-ink changes will be authorized by the approving authority and documented. The initials of the individual making the pen-and ink change and the date of the change will be clearly marked in the margin adjacent to the change. Such changes will be immediately followed by a formal change request.

16.2.4 Distribution

Each CECC-EPIP or revision will be authorized by an approval form and distributed by Administrative Support and Procedures. 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 with in two weeks to Administrative Support and Procedures.

Each revision will be accompanied by a revised cover page for that procedure. Administrative Support and Procedures 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

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

Nuclear 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 Medical Service, Chattanooga, TN
Athens-Limestone Ambulance Service, Athens, AL
Rhea County 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 Medical Center, Chattanooga, TN
Memorial North Park Hospital, 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:

Dayton City Fire Department, TN
Rhea County Fire Department, TN
Soddy Daisy Fire Department, TN
Clements 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. TVA agrees to provide two-hours notice prior to any such use and to pay the college for facilities and services provided.

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.

APPENDIX B

SEQUOYAH NUCLEAR PLANT

TABLE OF CONTENTS

*Revision

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 progress or have occurred, which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA Protection Action Guideline exposure levels offsite for more than the immediate site area.

For a Site Area Emergency to be declared, events should be in progress or have occurred that involve actual or likely major failures of plant functions needed for protection of the public. Any releases are not expected to exceed EPA protective action guides.

For an Alert to be declared, an event should be in process or have occurred that involves an actual or potential substantial degradation of the plant. Releases of radioactive material are expected to be limited to small fractions of the EPA protective action guidelines.

For an Unusual Event to be declared, unusual events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant.

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.

B.2 Emergency Event Methodology

The SQN methodology for event classification and declaration has 35 emergency events broken down into the following seven categories:

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

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, HS1, HG1
4.7 SED Judgement	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

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 follow-up investigation show 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.

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.

During an event when 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 and shall not be declared unless it is a noted exception. 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 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 the classification. 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.

The **ACCEPTABLE** time frame for notification to the (ODS) is five (5) minutes. This is the time between declaration of the emergency and notifying the ODS.

References:

10 CFR 50	Domestic Licensing of Production and Utilization Facilities
REG GUIDE-1.101	*Emergency Planning and Preparedness For Nuclear Power Reactors endorsing NUMARC NESP-007 Methodology for Development of Emergency Action Levels*

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.

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.

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 at 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.

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.

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 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-O 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.

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 ≥ 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 inhala[illegible]r 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 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).

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) by 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 equipment, systems, devices, or material which the failure, destruction, or release of, could directly or indirectly endanger the public health and s[illegible]y exposure to radiation.

•Revision

Sequoyah Nuclear Plant

Emergency Classification and Declaration Methodology

BASIS

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Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
Event 1.1	**FUEL CLAD BARRIER**
IC 1.1.1	**Critical Safety Function Status**
Mode	1,2,3,4
Description	**LOSS:** Core Cooling Red (FR-C.1). **Potential LOSS:** Core Cooling Orange (FR-C.2). OR Heat Sink Red (FR-H.1) (RHR shut down cooling not in service).
Basis	**LOSS:** The "Loss" IC addresses the condition of inadequate core cooling. 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. Core Cooling - Red indicates significant superheating and core uncovery and is considered to indicate a "Loss" of the fuel clad barrier. **Potential LOSS:** 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. 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.
Escalation	Not Applicable.
References	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

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
Event 1.1	**FUEL CLAD BARRIER**
IC 1.1.2	**Primary Coolant Activity Level**
Mode	1,2,3,4
Description	**LOSS:** RCS sample activity is >300 μCi/gm dose equivalent Iodine -131. **Potential LOSS:** Not Applicable.
Basis	**LOSS:** The "Loss" IC addresses the condition of high RCS activity. If the reading of RCS activity is > 300 μCi/gm it is well above expected iodine spikes and corresponds to about 2% to 5% fuel clad damage. This amount of clad damage indicates that significant clad heating has occurred. **Potential LOSS:** These is no "Potential Loss" IC associated with this item.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
Event 1.1	**FUEL CLAD BARRIER**
IC 1.1.3	**Incore TCs Hi Quad Average**
Mode	1,2,3,4
Description	**LOSS**: Greater than 1200°F on XI-94-101 or 102 (EXOSENSOR). **Potential LOSS:** Greater than or equal to 700°F on XI-94-101 or 102 (EXOSENSOR).
Basis	**LOSS**: 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. **Potential LOSS:** 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.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101 FR-C.1 Inadequate Core Cooling FR-C.2 Degraded Core Cooling

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
Event 1.1	**FUEL CLAD BARRIER**
IC 1.1.4	**Reactor Vessel Water Level**
Mode	1,2,3,4
Description	**LOSS:** Not Applicable. **Potential LOSS:** VALID RVLIS Level <40% On LI-68-368 or 371 (No RCP running).
Basis	**LOSS:** There is no "Loss" IC corresponding to this item because it is covered by the other fuel clad barrier "Loss". **Potential LOSS:** The "Potential Loss" IC is defined by an orange path on the core cooling status tree. The numeric value used is 40% level with no reactor coolant pumps running. This condition indicates that considerable clad heating and loss of RCS subcooling has occurred. A RVLIS reading of 40% 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.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101

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	<u>LOSS:</u> VALID reading of greater than: 2.8E+01 Rem/hr on RM-90-271 and 272. <u>OR</u> 2.9E+01 Rem/hr on RM-90-273 and 274. <u>Potential LOSS</u>: Not Applicable.
Basis	<u>LOSS:</u> The "Loss" IC is defined by a VALID reading of 2.8E+01 Rem/hr on the upper containment hi rad monitors or 2.9E+01 Rem/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. 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 clad failure. Thus, this IC indicates a loss of both the fuel clad barrier and a loss of the RCS barrier. <u>Potential LOSS:</u> There is no "Potential Loss" IC associated with this item.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101

Section 1.0	FISSION PRODUCT BARRIER MATRIX
Event 1.1	FUEL CLAD BARRIER
IC 1.1.6	Site Emergency Director Judgement
Mode	1,2,3,4
Description	*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.
Basis	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. 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.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101

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Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
Event 1.2	**RCS BARRIER**
IC 1.2.1	**Critical Safety Function Status**
Mode	1,2,3,4
Description	**<u>LOSS:</u>** Not Applicable. **<u>Potential LOSS</u>:** Pressurized Thermal Shock Red (FR-P.1). <u>OR</u> Heat Sink Red (FR-H.1) (RHR shutdown cooling not in service).
Basis	**<u>LOSS:</u>** There is no "Loss" IC associated with this item. **<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.
Escalation	Not Applicable.
References	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

Section 1.0	FISSION PRODUCT BARRIER MATRIX
Event 1.2	RCS BARRIER
IC 1.2.2	RCS Leakage/LOCA
Mode	1,2,3,4
Description	<u>LOSS:</u> *RCS leak results in subcooling <40°F as indicated on XI-94-101 or 102. <u>Potential LOSS</u>: Non-isolatable RCS leak exceeding the capacity of one charging pump in the normal charging alignment. <u>OR</u> RCS leakage results in entry into E-1.
Basis	<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. <u>Potential LOSS:</u> 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.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101 E-1 Loss of Reactor or Secondary Coolant

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Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
Event 1.2	**RCS BARRIER**
IC 1.2.3	**Steam Generator Tube Rupture**
Mode	1,2,3,4
Description	**<u>LOSS</u>:** SGTR that results in a safety injection actuation. <u>OR</u> Entry into E-3. **<u>Potential LOSS:</u>** Not Applicable.
Basis	**<u>LOSS</u>:** 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, 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. 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. This IC also addresses the entry into EOP, E-3, Steam Generator Tube Rupture, under any circumstance. 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. **<u>Potential LOSS:</u>** There is no "Potential Loss" IC associated with this item.
Escalation	Not Applicable.
References	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

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
Event 1.2	**RCS BARRIER**
IC 1.2.4	Reactor Vessel Water Level
Mode	1,2,3,4
Description	**LOSS:** VALID RVLIS level <40% on LI-68-368 or 371 with no RCP running. **Potential LOSS:** Not Applicable.
Basis	**LOSS:** The "Loss" IC is defined by an orange path on the core cooling status tree (CSF). The numeric value used is 40% 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. This "Loss" EAL in conjunction with the fuel clad barrier "Potential Loss" IC 1.1.4 requires an event classification of Site Area Emergency. A RVLIS reading of 40% 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. **Potential LOSS:** There is no "Potential Loss" IC associated with this item.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
Event 1.2	**RCS BARRIER**
IC 1.2.5	**Site Emergency Director Judgement**
Mode	1,2,3,4
Description	*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.
Basis	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. 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.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101

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Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
Event 1.3	**CNTMT BARRIER**
IC 1.3.1	**Critical Safety Function Status**
Mode	1,2,3,4
Description	**LOSS:** Not Applicable. **POTENTIAL LOSS:** Containment Red (FR-Z.1). OR Actions of FR-C.1 (Red Path) are INEFFECTIVE (ie: core TC's trending up).
Basis	**LOSS:** There is no "Loss" IC associated with this item. **Potential LOSS:** 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. 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. 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.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101 FR-Z.1 High Containment Pressure FR-C.1 Inadequate Core Cooling

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
Event 1.3	**CNTMT BARRIER**
IC 1.3.2	**Containment Pressure/Hydrogen**
Mode	1,2,3,4
Description	<u>LOSS</u>: Rapid unexplained pressure decrease following initial increase on PdI 30-44 or 45. <u>OR</u> Containment pressure or sump level not increasing on LI-63-178 or 179 with a LOCA in progress. <u>Potential LOSS:</u> Containment hydrogen increases to >4% by volume on H_2I-43-200 or 210. <u>OR</u> Pressure >2.81 PSID (Ø B) with no containment spray operating when required (FR-Z.1).
Basis	<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. 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. <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. 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 2.81 psid where the equipment should actuate. This condition is not applicable if the pumps are secured under other procedures. The condition of high containment pressure, >12 psid, is addressed by the CSF, containment red, "Potential Loss", IC #1.3.1. These "Potential Loss" ICs are primarily a discrimination between the Site Area Emergency and General Emergency representing a potential loss of the third barrier.
Escalation	Not Applicable.

Section 1.0	FISSION PRODUCT BARRIER MATRIX
Event 1.3	CNTMT BARRIER
IC 1.3.3	Containment Isolation Status
Mode	1,2,3,4
Description	**LOSS:** Containment isolation, when required, is incomplete and a release path to the environment exists. **Potential LOSS:** Not Applicable.
Basis	**LOSS:** 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. **Potential LOSS:** There is no "Potential Loss" IC associated with this item.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101

Section 1.0	FISSION PRODUCT BARRIER MATRIX
Event 1.3	CNTMT BARRIER
IC 1.3.4	Containment Bypass
Mode	1,2,3,4
Description	**LOSS:** Secondary side release outside CNTMT from a RUPTURED S/G that cannot be terminated in <15 Minutes (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). **Potential LOSS:** Unexpected VALID increase in area or ventilation RAD monitors adjacent to containment (with LOCA in progress).
Basis	**LOSS:** 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. 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. 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.

Section 1.0	FISSION PRODUCT BARRIER MATRIX
Event 1.3	CNTMT BARRIER
IC 1.3.4	Containment Bypass (continued)
Mode	1,2,3,4
Basis (continued)	**Potential LOSS:** 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. 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. 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. This IC in conjunction with the RCS barrier IC 1.2.2 results in an event classification of Site Area Emergency.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev. 2, 1/92 per REG GUIDE 1.101 E-2 Faulted Steam Generator Isolation

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	**LOSS:** Not Applicable. **Potential LOSS:** VALID reading of greater than 3.6E+02 Rem/hr on RM-90-271 and RM-90-272. OR 2.8E+02 Rem/hr on RM-90-273 and RM-90-274.
Basis	**LOSS:** There is no "Loss" IC associated with this item. **Potential LOSS:** The "Potential Loss" IC is defined by containment radiation readings of 3.6E+02 Rem/hr and 2.8E+02 Rem/hr. 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%.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev. 2, 1/92 per REG GUIDE 1.101 NUREG-1228, Source Estimates During Incident Response to Severe Nuclear Power Plant Accidents.

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
Event 1.3	**CNTMT BARRIER**
IC 1.3.6	Site Emergency Director Judgement
Mode	1,2,3,4
Description	*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.
Basis	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. 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.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev. 2, 1/92 per REG GUIDE 1.101

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FISSION PRODUCT BARRIER UTILIZATION

in

EMERGENCY EVENT CLASSIFICATION

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
Event	**Not Applicable.**
Classification	**GENERAL EMERGENCY**
Mode	1,2,3,4
Description	**LOSS** of any two barriers and **Potential LOSS** of third barrier.
Basis	Definition: Events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA Plume Protective Action Guidelines exposure levels outside the exclusion area boundary. (At SQN the Site Boundary and EAB are synonymous for emergency planning purposes). 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. 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. 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.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
Event	**Not Applicable**
Classification	**SITE AREA EMERGENCY**
Mode	1,2,3,4
Description	**LOSS** or **Potential LOSS** of any two barriers.
Basis	Definition: Events are in process or have occurred which involve actual or likely major failures of plant functions needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Plume Protective Action Guideline exposure levels outside the exclusion area boundary. (At SQN the Site Boundary and EAB are synonymous for emergency planning purposes). 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. This approach also simplifies the Site Area Emergency classification from the fission product barrier matrix. 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.
Escalation	Escalation will be based on actual or imminent substantial core degradation.
References	NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
Event	**Not Applicable**
Classification	**ALERT**
Mode	1,2,3,4
Description	Any **LOSS** or **Potential LOSS** of Fuel Clad Barrier. OR Any **LOSS** or **Potential LOSS** of **RCS** barrier.
Basis	Definition: Events are in process or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of the EPA Plume Protective Action Guideline exposure levels. The "Loss" or "Potential Loss" of either the fuel clad barrier or **RCS** barrier is considered to be an actual or potential substantial degradation of the level of safety of the plant. The Alert classification resulting from potential degradation of the fuel clad or **RCS** integrity also addresses the operation staff's need for help by staffing the Technical Support Center (TSC), independent of whether an actual decrease in plant safety is determined. This increased monitoring can then be used to better determine the actual plant safety state, whether escalation to a higher emergency class is warranted, or whether de-escalation or termination of the emergency class declaration is warranted. Dose consequences from these events are small fractions of the EPA PAG plume exposure levels, i.e., about 10 millirem to 100 millirem. 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.
Escalation	Escalation will be based on actual or likely major failures of plant functions needed to protect the public.
References	NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
Event	**Not Applicable.**
Classification	**UNUSUAL EVENT**
Mode	1,2,3,4
Description	**LOSS** or **Potential LOSS** of Containment Barrier.
Basis	Definition: Events are in process or have occurred which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite responses or monitoring are expected unless further degradation of safety systems occurs. 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). 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. 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.
Escalation	Escalation will be based on actual or potential substantial degradation of the level of safety of the plant.
References	NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101

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Section 2.0	**SYSTEM DEGRADATION**
Event 2.1	**LOSS OF INSTRUMENTATION**
Classification	**GENERAL EMERGENCY**
Mode	Not Applicable.
Description	Refer to the "Fission Product Barrier Matrix" (Section 1) and "Radiological Effluents" (Section 7).
Basis	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).
Escalation	Not applicable.
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 2.0	SYSTEM DEGRADATION
Event 2.1	LOSS OF INSTRUMENTATION
Classification	SITE AREA EMERGENCY
Mode	1,2,3,4
Description	**Inability to monitor a SIGNIFICANT TRANSIENT in progress on either unit (1 and 2 and 3 and 4):** 1. Loss of > 75% of MCR annunciators and annunciator printer or > 75% of safety system indications. 2. Loss of plant computer. 3. Inability to directly monitor any of the following CSFs: Subcriticality, PTS, Core Cooling Containment, Heat Sink, Inventory 4. SIGNIFICANT TRANSIENT in progress.
Basis	This IC is intended to recognize the inability of the control room staff to monitor the plant response to a transient. When the loss of safety system annunciators is complicated with an unplanned power change as well as loss of the plant computer 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. For the purposes of quantification, 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. 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%. 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.

Section 2.0	**SYSTEM DEGRADATION**
Event 2.1	**LOSS OF INSTRUMENTATION**
Classification	**SITE AREA EMERGENCY (continued)**
Mode	1,2,3,4
Escalation	Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
References	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

Section 2.0	SYSTEM DEGRADATION
Event 2.1	LOSS OF INSTRUMENTATION
Classification	ALERT
Mode	1,2,3,4
Description	**On either unit UNPLANNED loss of > 75% MCR annunciators and annunciator printer or safety system indications > 15 minutes with a SIGNIFICANT TRANSIENT in progress or plant computer unavailable (1 and 2 and 3):** 1. UNPLANNED loss of > 75% MCR annunciators and the annunciator printer 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) a. SIGNIFICANT TRANSIENT in progress. OR b. Loss of plant computer.
Basis	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. Fifteen minutes was selected as a threshold value to exclude momentary transients or power losses. For the purposes of quantification, 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. 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. 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 ≥ 10%.

Section 2.0	**SYSTEM DEGRADATION**
Event 2.1	**LOSS OF INSTRUMENTATION**
Classification	ALERT (continued)
Mode	1,2,3,4
Basis (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.
Escalation	Escalation will be based on the inability of the operating crew to monitor a transient in progress.
References	NUMARC/NESP-007, SA4, Rev. 2, 1/92 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

Section 2.0	SYSTEM DEGRADATION
Event 2.1	LOSS OF INSTRUMENTATION
Classification	UNUSUAL EVENT
Mode	1,2,3,4
Description	**On either unit UNPLANNED loss of > 75% MCR annunciators and annunciator printer or safety system indications for > 15 minutes and plant computer available (1 and 2 and 3):** 1. UNPLANNED loss of > 75% MCR annunciators and annunciator printer 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 plant computer is capable of displaying data requested.
Basis	For this IC, if annunciators are partially or completely lost it is still possible to use other systems to indicate plant conditions (e.g., plant computer). However, it is prudent to declare an Unusual Event since there is a greater risk that a degraded condition could go undetected. The loss of annunciators excludes scheduled maintenance and testing activities. Fifteen minutes was selected as a threshold value to exclude momentary power losses or transients. For the purposes of quantification, 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. The declaration will ensure that adequate resources are available to monitor and control plant systems. 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.

Section 2.0	**SYSTEM DEGRADATION**
Event 2.1	**LOSS OF INSTRUMENTATION**
Classification	**UNUSUAL EVENT (continued)**
Mode	1,2,3,4
Escalation	Escalation will be based on loss of annunciators complicated by the loss of the plant computer or a transient in progress.
References	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

Section 2.0	**SYSTEM DEGRADATION**
Event 2.2	**LOSS OF COMMUNICATION**
Classification	**GENERAL EMERGENCY**
Mode	Not Applicable.
Description	Not Applicable.
Basis	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.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
Event 2.2	**LOSS OF COMMUNICATION**
Classification	**SITE AREA EMERGENCY**
Mode	Not Applicable.
Description	Not Applicable.
Basis	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.
Escalation	Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 2.0	SYSTEM DEGRADATION
Event 2.2	LOSS OF COMMUNICATION
Classification	ALERT
Mode	Not Applicable.
Description	Not Applicable.
Basis	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.
Escalation	Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
References	NUMARC/NESP-007, Rev. 2, 1/92 Expanded

Section 2.0	SYSTEM DEGRADATION
Event 2.2	LOSS OF COMMUNICATION
Classification	UNUSUAL EVENT
Mode	All
Description	**A. UNPLANNED loss of all in-plant communication capability** (1 and 2 and 3): 1. UNPLANNED loss of EPABX phones. 2. UNPLANNED loss of all sound powered phones. 3. UNPLANNED loss of all radios. OR **B. UNPLANNED loss of all Offsite Communication capability** (1 and 2 and 3 and 4 and 5): 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 FTS 2000 (NRC) system.
Basis	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. The loss of offsite communications ability is expected to be significantly more comprehensive than those addressed by 10 CFR 50.72. Onsite communications loss must encompass the loss of all means of routine communications (i.e., phones, page party system and radio/walkie talkies). 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).
Escalation	Escalation of this event will involve the loss of other plant functions.
References	NUMARC/NESP-007, SU6, Rev. 2, 1/92 NP Radiological Emergency Plan (REP) 10 CFR 50.72 NUREG 0654

Section 2.0	SYSTEM DEGRADATION
Event 2.3	FAILURE OF RX PROTECTION
Classification	GENERAL EMERGENCY
Mode	1
Description	**Rx power >5% and not decreasing after VALID trip signals and loss of core cooling capability (1 and 2):** 1. FR-S.1 entered and immediate operator actions did not result in a reactor power of ≤5% and decreasing. 2. (a or b) a. CSF status tree indicates Core Cooling Red (FR-C.1). OR b. CSF status tree indicates Heat Sink Red (FR-H.1).
Basis	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. 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. 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. 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. In either situation, if these challenges exist at a time that the reactor has not been brought ≤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.
Escalation	Not Applicable.
References	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

Section 2.0	SYSTEM DEGRADATION
Event 2.3	FAILURE OF RX PROTECTION
Classification	SITE AREA EMERGENCY
Mode	1
Description	Reactor power >5% and not decreasing after VALID auto and manual trip signals: Note: Although a mode change may occur before classification this event will still be classified and declared as a SAE.
Basis	This IC indicates a failure of the automatic and main control room manual signals to scram the reactor. 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. 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 Emergenc 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. 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.
Escalation	Escalation will be based on the inability to trip the reactor and indications of heat sink red or core cooling red.
References	NUMARC/NESP-007, SS2 (Modified), 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

Section 2.0	SYSTEM DEGRADATION
Event 2.3	FAILURE OF RX PROTECTION
Classification	ALERT
Mode	1,2
Description	**Reactor power >5% and not decreasing after VALID auto trip signal but manual trip is successful (1 and 2):** 1. Reactor power >5% and not decreasing following auto. trip signal. 2. Manual trip from the MCR successfully reduces reactor power to ≤5%. *NOTE: Although a mode change will occur this event will still be classified as an ALERT.*
Basis	This IC indicates a failure of the automatic protection system to trip the reactor with a successful manual trip being initiated. 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. 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.
Escalation	Escalation will be based on the reactor power not being reduced to less than five percent by a successful manual trip.
References	NUMARC/NESP-007, SA2 (Modified), 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

Section 2.0	**SYSTEM DEGRADATION**
Event 2.3	**FAILURE OF RX PROTECTION**
Classification	**UNUSUAL EVENT**
Mode	Not Applicable.
Description	Refer to the "Fission Product Barrier Matrix" (Section 1).
Basis	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.
Escalation	Escalation will be based on a successful manual trip the reactor from the main control room.
References	NUMARC/NESP-007, SA2 (Modified), Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
Event 2.4	**FUEL CLAD DEGRADATION**
Classification	**GENERAL EMERGENCY**
Mode	Not Applicable.
Description	Refer to the "Fission Product Barrier Matrix" (Section 1).
Basis	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.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
Event 2.4	**FUEL CLAD DEGRADATION**
Classification	**SITE AREA EMERGENCY**
Mode	Not Applicable.
Description	Refer to the "Fission Product Barrier Matrix" (Section 1).
Basis	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.
Escalation	Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
References	NUMARC/NESP-007, Rev. 2, 1/924

Section 2.0	**SYSTEM DEGRADATION**
Event 2.4	**FUEL CLAD DEGRADATION**
Classification	**ALERT**
Mode	Not Applicable.
Description	Refer to the "Fission Product Barrier Matrix" (Section 1).
Basis	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.
Escalation	Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 2.0	SYSTEM DEGRADATION
Event 2.4	FUEL CLAD DEGRADATION
Classification	UNUSUAL EVENT
Mode	1,2,3
Description	**Reactor coolant system specific activity exceeds Tech. Spec. 3.4.8 LCO:** 1. Radiochemistry analysis indicates (a or b): a. Dose equivalent iodine (I-131) >0.35 µCi/gm for >48 hours or in excess of T/S figure 3.4-1 with Tave ≥ 500°F. <u>OR</u> b. Specific activity >100/ $\bar{E}$ µCi/gm with Tave ≥500°F.
Basis	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. The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) accident. The LCO contains specific activity limits for both dose equivalent I-131 and gross specific activity. The allowable levels are intended to limit the 2-hour dose at the exclusion area boundary to a small fraction of the 10 CFR 100 dose guideline values. The limits in the LCO are standardized and based on parametric evaluations of offsite radioactivity dose consequences for typical site locations. 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.
Escalation	Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
References	NUMARC/NESP-007, SU4, Rev. 2, 1/92 T.S. 3.4.8 RCS Specific Activity AOP R.06 High RCS Activity

Section 2.0	**SYSTEM DEGRADATION**
Event 2.5	**RCS UNIDENTIFIED LEAKAGE**
Classification	**GENERAL EMERGENCY**
Mode	Not Applicable.
Description	Refer to the "Fission Product Barrier Matrix" (Section 1).
Basis	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.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
Event 2.5	**RCS UNIDENTIFIED LEAKAGE**
Classification	**SITE AREA EMERGENCY**
Mode	Not Applicable.
Description	Refer to the "Fission Product Barrier Matrix" (Section 1).
Basis	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.
Escalation	Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
Event 2.5	**RCS UNIDENTIFIED LEAKAGE**
Classification	**ALERT**
Mode	Not Applicable.
Description	Refer to the"Fission Product Barrier Matrix" (Section 1).
Basis	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.
Escalation	Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
Event 2.5	**RCS UNIDENTIFIED LEAKAGE**
Classification	**UNUSUAL EVENT**
Mode	1,2,3,4
Description	**Unidentified or pressure boundary RCS leakage > 10 GPM:** 1. Unidentified or pressure boundary leakage (as defined by Tech. Specs.) > 10 GPM as indicated by (a or b): a. SI-OPS-068-137.0 results. <u>OR</u> 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. NOTE: Refer to "Shutdown System Degradation" (Section 6.3).
Basis	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. Only operating modes in which there is fuel in the reactor coolant system and the system is pressurized are specified.
Escalation	Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
References	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

Section 2.0	**SYSTEM DEGRADATION**
Event 2.6	**RCS IDENTIFIED LEAKAGE**
Classification	**GENERAL EMERGENCY**
Mode	Not Applicable.
Description	Refer to the "Fission Product Barrier Matrix" (Section 1).
Basis	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.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
Event 2.6	**RCS IDENTIFIED LEAKAGE**
Classification	**SITE AREA EMERGENCY**
Mode	Not Applicable.
Description	Refer to the "Fission Product Barrier Matrix" (Section 1).
Basis	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.
Escalation	Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
Event 2.6	**RCS IDENTIFIED LEAKAGE**
Classification	**ALERT**
Mode	Not Applicable.
Description	Refer to the "Fission Product Barrier Matrix" (Section 1).
Basis	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.
Escalation	Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
Event 2.6	**RCS IDENTIFIED LEAKAGE**
Classification	**UNUSUAL EVENT**
Mode	1,2,3,4
Description	**Identified RCS leakage >25 GPM:** 1. Identified RCS leakage (as defined by Tech. Specs.) >25 GPM as indicated by (a or b): a. SI-OPS-068-137.0 results. OR b. Level rise in excess of 25 GPM total into PRT, RCDT or CVCS holdup tank (Refer to TI-28). NOTE: Also refer to "Shutdown System Degradation" (Section 6.3).
Basis	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. Only operating modes in which there is fuel in the reactor coolant system and the system is pressurized are specified.
Escalation	Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
References	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

Section 2.0	**SYSTEM DEGRADATION**
Event 2.7	**UNCONTROLLED COOLDOWN**
Classification	**GENERAL EMERGENCY**
Mode	Not Applicable.
Description	Refer to the "Fission Product Barrier Matrix" (Section 1).
Basis	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.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
Event 2.7	**UNCONTROLLED COOLDOWN**
Classification	**SITE AREA EMERGENCY**
Mode	Not Applicable.
Description	Refer to the "Fission Product Barrier Matrix" (Section 1).
Basis	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.
Escalation	Escalation of this event will be based on "Fission Product Barrier Matrix" (Section 1).
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 2.0	SYSTEM DEGRADATION
Event 2.7	UNCONTROLLED COOLDOWN
Classification	ALERT
Mode	Not Applicable.
Description	Refer to the "Fission Product Barrier Matrix" (Section 1).
Basis	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.
Escalation	Escalation of this event will be based on "Fission Product Barrier Matrix" (Section 1).
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 2.0	SYSTEM DEGRADATION
Event 2.7	UNCONTROLLED COOLDOWN
Classification	UNUSUAL EVENT
Mode	1,2,3
Description	**UNPLANNED rapid depressurization of the main steam system resulting in a rapid RCS cooldown and safety injection initiation (1 and 2):** 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.
Basis	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. Unplanned is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities.
Escalation	Escalation of this event will be based on "Fission Product Barrier Matrix" (Section 1).
References	NUMARC/NESP-007, HU5, Rev. 2, 1/92 E-2 Faulted Steam Generator Isolation T.S. 3.3.2 Engineering Safety Features Actuation System Instrumentation (ESFAS)

Section 2.0	**SYSTEM DEGRADATION**
Event 2.8	**TURBINE FAILURE**
Classification	**GENERAL EMERGENCY**
Mode	Not Applicable.
Description	Refer to the "Fission Product Barrier Matrix" (Section 1).
Basis	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.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
Event 2.8	**TURBINE FAILURE**
Classification	**SITE AREA EMERGENCY**
Mode	Not Applicable.
Description	Refer to the "Fission Product Barrier Matrix" (Section 1).
Basis	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.
Escalation	Escalation of this event will be based on "Fission Product Barrier Matrix" (Section 1).
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 2.0	SYSTEM DEGRADATION
Event 2.8	TURBINE FAILURE
Classification	ALERT
Mode	1,2,3
Description	**Turbine failure has generated projectiles that cause visible damage to any area containing safety related equipment:** Turbine generated projectiles have resulted in visible damage in any of the following areas: 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 Condensate Storage Tanks Additional Equipment Buildings
Basis	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. PROJECTILE: An object ejected, thrown, or launched towards a plant structure resulting in damage sufficient to cause concem 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 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. 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.
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

Section 2.0	**SYSTEM DEGRADATION**
Event 2.8	**TURBINE FAILURE**
Classification	**UNUSUAL EVENT**
Mode	1,2,3
Description	**Turbine failure results in casing penetration or main generator seal damage:** Turbine failure which results in penetration of the turbine casing or damage to main generator seals. NOTE: Refer to "Hazards and SED Judgement" (Section 4.3).
Basis	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.
Escalation	Escalation will be based on potential damage done by turbine projectiles to safety related equipment.
References	NUMARC/NESP-007, HU1, Rev. 2, 1/92 FSAR 10.2.3 Turbine Missiles

Section 2.0	**SYSTEM DEGRADATION**
Event 2.9	**SAFETY LIMIT**
Classification	**GENERAL EMERGENCY**
Mode	Not Applicable.
Description	Not Applicable.
Basis	Safety limit is not applicable for a General Emergency.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
Event 2.9	**SAFETY LIMIT**
Classification	**SITE AREA EMERGENCY**
Mode	Not Applicable.
Description	Not Applicable.
Basis	Safety limit is not applicable for a Site Area Emergency.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
Event 2.9	**SAFETY LIMIT**
Classification	**ALERT**
Mode	Not Applicable.
Description	Not Applicable.
Basis	Safety limit is not applicable for an Alert.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
Event 2.9	**SAFETY LIMIT**
Classification	**UNUSUAL EVENT**
Mode	1,2,3,4,5
Description	**Safety limit has been exceeded (1 or 2):** 1. The combination of thermal power, RCS temperature and RCS pressure > safety limit as indicated by SQN TS Fig. 2.1-1 "Reactor Core Safety Limit". OR 2. RCS/ pressurizer pressure exceeds safety limit (>2735 psig).
Basis	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. 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. 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. 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. This EAL is consistent with the definition of an Unusual Event and warrants declaration.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev. 2, 1/92 T.S. 2.1.1 and B.2.1

Section 3.0	LOSS OF POWER
Event 3.1	LOSS OF AC (Power Ops)
Classification	GENERAL EMERGENCY
Mode	1,2,3,4
Description	**Prolonged loss of all offsite and all onsite AC power to EITHER UNIT (1 and 2):** 1. Both unit related 6.9KV shutdown boards de-energized for >15 minutes. 2. (a or b) a. Core cooling status tree red or orange path. OR b. Restoration of either a 6.9KV shutdown board or a 6.9KV unit board is not likely within 4 hours of the loss.
Basis	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. 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. The 15 minute time duration was selected to exclude transient or momentary power losses. 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: 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?

Section 3.0	**LOSS OF POWER**
Event 3.1	**LOSS OF AC (Power Ops)**
Classification	**GENERAL EMERGENCY (continued)**
Mode	**1,2,3,4**
Basis (continued)	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? 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.
Escalation	Not Applicable.
Reference	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

Section 3.0	**LOSS OF POWER**
Event 3.1	**LOSS OF AC (Power Ops)**
Classification	**SITE AREA EMERGENCY**
Mode	1,2,3,4
Description	**Loss of all offsite and all onsite AC power to EITHER UNIT for > 15 Minutes:** Both unit related 6.9KV shutdown boards de-energized for > 15 minutes.
Basis	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. The 15 minute time duration was selected to exclude transient or momentary power losses.
Escalation	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).
References	NUMARC/NESP-007, SS1, Rev. 2, 1/92 AOP P.01 Loss of Offsite Power FSAR 15.2.9 Loss of Offsite Power to the Station Auxiliaries General Design Criteria 17, App. A, 10 CFR 50 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

Section 3.0	**LOSS OF POWER**
Event 3.1	**LOSS OF AC (Power Ops)**
Classification	**ALERT**
Mode	1,2,3,4
Description	**Loss of offsite power to EITHER UNIT with degraded onsite AC power for >15 minutes (1a and b or 2):** 1a. All four (4) 6.9KV unit boards de-energized for >15 minutes. b. One (1) unit related 6.9KV shutdown board de-energized for >15 minutes. OR 2. Any AC power condition lasting >15 minutes where an additional single failure will result in a unit blackout.
Basis	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. The 15 minute time duration was selected to exclude transient or momentary power losses.
Escalation	Escalation will be based on prolonged loss of all offsite power and prolonged loss of all onsite power.
References	NUMARC/NESP-007,SA5, Rev. 2, 1/92 *AOP P.01 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.3.8.2 Onsite Power Distribution Systems, Operating

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Section 3.0	**LOSS OF POWER**
Event 3.1	**LOSS OF AC (Power Ops)**
Classification	**UNUSUAL EVENT**
Mode	1,2,3,4
Description	**Loss of offsite power to EITHER UNIT for > 15 minutes (1 and 2):** 1. All four (4) 6.9KV unit boards de-energized for > 15 minutes. 2. Both unit related 6.9KV shutdown boards are energized.
Basis	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 minutes was selected as a threshold to exclude transient or momentary power losses.
Escalation	Loss of one additional power supply to the shutdown board will escalate this event.
References	NUMARC/NESP-007, SU1, Rev. 2, 1/92 AOP P.01 Loss of Offsite Power 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 General Design Criterion 17 of App. A, 10 CFR 50 T.S. 3.8.1 AC Sources, Operating T.S. 3.8.2 Onsite Power Distribution Systems, Operating

Section 3.0	**LOSS OF POWER**
Event 3.2	**LOSS OF AC (Shutdown)**
Classification	**GENERAL EMERGENCY**
Mode	Not Applicable.
Description	Not Applicable.
Basis	Loss of AC power in Mode 5 and 6 will not cause a declaration of a General Emergency.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev 2, 1/92

Section 3.0	**LOSS OF POWER**
Event 3.2	**LOSS OF AC (Shutdown)**
Classification	**SITE AREA EMERGENCY**
Mode	Not Applicable.
Description	Not Applicable.
Basis	Loss of AC power in Mode 5 and 6 will not cause a declaration of a Site Area Emergency.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev 2, 1/92

Section 3.0	**LOSS OF POWER**
Event 3.2	**LOSS OF AC** (Shutdown)
Classification	**ALERT**
Mode	5,6, or Defuelled
Description	**UNPLANNED loss of all offsite and onsite AC power to EITHER UNIT for >15 minutes:** Both unit related 6.9KV shutdown boards de-energized >15 minutes. Also refer to "Loss of Shutdown Systems" (Section 6.1).
Basis	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. Fifteen minutes was selected as a threshold to exclude transient or momentary power losses. Unplanned is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities.
Escalation	Escalation is not applicable from this event.
References	NUMARC/NESP-007 SU1 (expanded), Rev 2, 1/92 T.S. 3.8.1 AC Sources Shutdown T.S. 3.8.2 Onsite Power Distribution Systems - Shutdown

Section 3.0	**LOSS OF POWER**
Event 3.2	**LOSS OF AC (Shutdown)**
Classification	**UNUSUAL EVENT**
Mode	5,6, Defuelled
Description	**UNPLANNED loss of all offsite power to EITHER UNIT for >15 minutes (1 and 2):** 1. All four 6.9KV unit boards de-energized for >15 minutes. 2. One unit related 6.9KV shutdown board de-energized for >15 minutes.
Basis	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 minutes was selected as a threshold to exclude transient or momentary power losses. Unplanned is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities.
Escalation	Escalation will be based on loss of one additional power supply to the shutdown boards.
References	NUMARC/NESP-007, SU1, Rev 2, 1/92 T.S. 3.8.1 AC Sources Shutdown T.S. 3.8.2 Onsite Power Distribution Systems Shutdown

Section 3.0	**LOSS OF POWER**
Event 3.3	**LOSS OF DC**
Classification	GENERAL EMERGENCY
Mode	Not Applicable.
Description	Refer to "Fission Product Barrier Matrix" (Section 1) and "Loss of Communication" (Section 2.2).
Basis	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.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev 2, 1/92

Section 3.0	**LOSS OF POWER**
Event 3.3	**LOSS OF DC**
Classification	**SITE AREA EMERGENCY**
Mode	1,2,3,4
Description	**Loss of all vital DC power for > 15 minutes:** Voltage < 105V DC on 125V DC vital battery board buses I and II and III and IV for > 15 minutes. Also refer to the "Fission Product Barrier Matrix" (Section 1), "Loss of Communication" (Section 2.2), and "Loss of Instrumentation" (Section 2.1).
Basis	Loss of all DC power compromises the ability to monitor and control plant safety functions. Prolonged loss of all DC power will cause core uncovering and loss of containment integrity when there is significant decay heat and sensible heat in the reactor system. Fifteen minutes was selected as a threshold to exclude transient or momentary power losses. 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.
Escalation	Escalation will be based on "Fission Product Barrier Matrix" (Section 1) or "Loss of Communication" (Section 2.2).
References	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

Section 3.0	LOSS OF POWER
Event 3.3	LOSS OF DC
Classification	ALERT
Mode	Not Applicable.
Description	Refer to "Fission Product Barrier Matrix" (Section 1), "Loss of Communication" (Section 2.2), and "Loss of Instrumentation" (Section 2.1).
Basis	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.
Escalation	Escalation will be based on loss of or the inability to monitor a significant transient in progress.
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 3.0	**LOSS OF POWER**
Event 3.3	**LOSS OF DC**
Classification	**UNUSUAL EVENT**
Mode	5,6
Description	**UNPLANNED loss of a required train of DC power for > 15 minutes: (1 or 2)** (1) Voltage < 105 VDC on 125V DC vital battery board buses I and III for > 15 Minutes. or (2) Voltage < 105 VDC on 125V DC vital battery board buses II and IV for > 15 Minutes.
Basis	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. Unplanned is included in the IC to preclude the declaration of an emergency as a resu[lt] of planned maintenance activities. 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. The fifteen minute threshold is utilized to exclude a transient or momentary power losses.
Escalation	The event will escalate if the DC loss results in an inability to maintain cold shutdown.
References	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

Section 4.0	HAZARDS AND SED JUDGEMENT
Event 4.1	FIRE
Classification	GENERAL EMERGENCY
Mode	Not Applicable.
Description	Refer to "Fission Product Barrier Matrix" (Section 1).
Basis	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.
Escalation	Not Applicable.
References	NUMARC/NESP-007 Rev. 2, 1/92

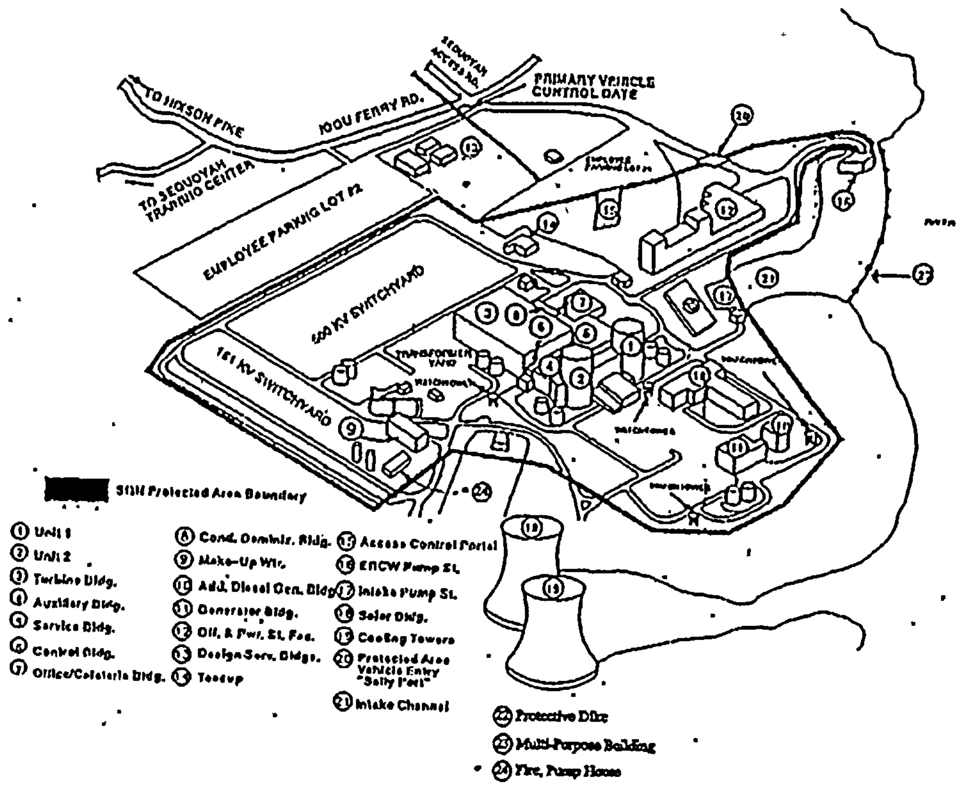
Section 4.0	HAZARDS AND SED JUDGEMENT
Event 4.1	FIRE
Classification	SITE AREA EMERGENCY
Mode	Not Applicable.
Description	Refer to "Control Room Evacuation," (Section 4.5) and "Fission Product Barrier Matrix" (Section 1).
Basis	The basis for a Site Area Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected. 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).
Escalation	Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
References	NUMARC/NESP-007 Rev. 2, 1/92

Section 4.0	HAZARDS AND SED JUDGEMENT
Event 4.1	FIRE
Classification	ALERT
Mode	All
Description	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): 1. FIRE in any of the areas listed in Table 4-1. 2. (a or b): a. Visible damage to permanent structure or safety related equipment in the specified area is observed due to the fire. OR b. Control room indication of degraded safety system or component response due to the fire.
Basis	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. Table 4-1 Plant structures associated with fire and explosion EALs: 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 Condensate Storage Tanks Additional Equipment Buildings 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.

Section 4.0	**HAZARDS AND SED JUDGEMENT**
Event 4.1	**FIRE**
Classification	ALERT (continued)
Mode	All
Basis (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.
Escalation	Escalation would be based on "Fission Product Barrier Matrix" (Section 1) or "Control Room Evacuation" (Section 4.5).
References	NUMARC/NESP-007, HA2, Rev. 2, 1/92

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	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. The 15 minute time limit has been established to exclude small fires that can be controlled by plant fire fighting resources. 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. Table 4-1 Plant structures associated with fire and explosion EALs: 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 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.

Figure 4-A
SEQUOYAH PROTECTED AREA



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Section 4.0	**HAZARDS AND SED JUDGEMENT**
Event 4.1	**FIRE**
Classification	**UNUSUAL EVENT (continued)**
Mode	All
Basis (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.
Escalation	Escalation will be based on the fire affecting plant safety related equipment required to establish or maintain safe shutdown.
References	NUMARC/NESP-007, HU2, Rev. 2, 1/92 SQN Protected Area Figure 4-A

Section 4.0	HAZARDS AND SED JUDGEMENT
Event 4.2	EXPLOSIONS
Classification	GENERAL EMERGENCY
Mode	Not Applicable.
Description	Refer to the "Fission Product Barrier Matrix" (Section 1).
Basis	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.
Escalation	Not Applicable.
References	NUMARC/NESP-007 Rev. 2, 1/92

Section 4.0	HAZARDS AND SED JUDGEMENT
Event 4.2	EXPLOSIONS
Classification	SITE AREA EMERGENCY
Mode	Not Applicable.
Description	Refer to the "Fission Product Barrier Matrix" (Section 1).
Basis	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.
Escalation	Escalation would be based on "Fission Product Barrier Matrix" (Section 1).
References	NUMARC/NESP-007, Rev. 2, 1/92

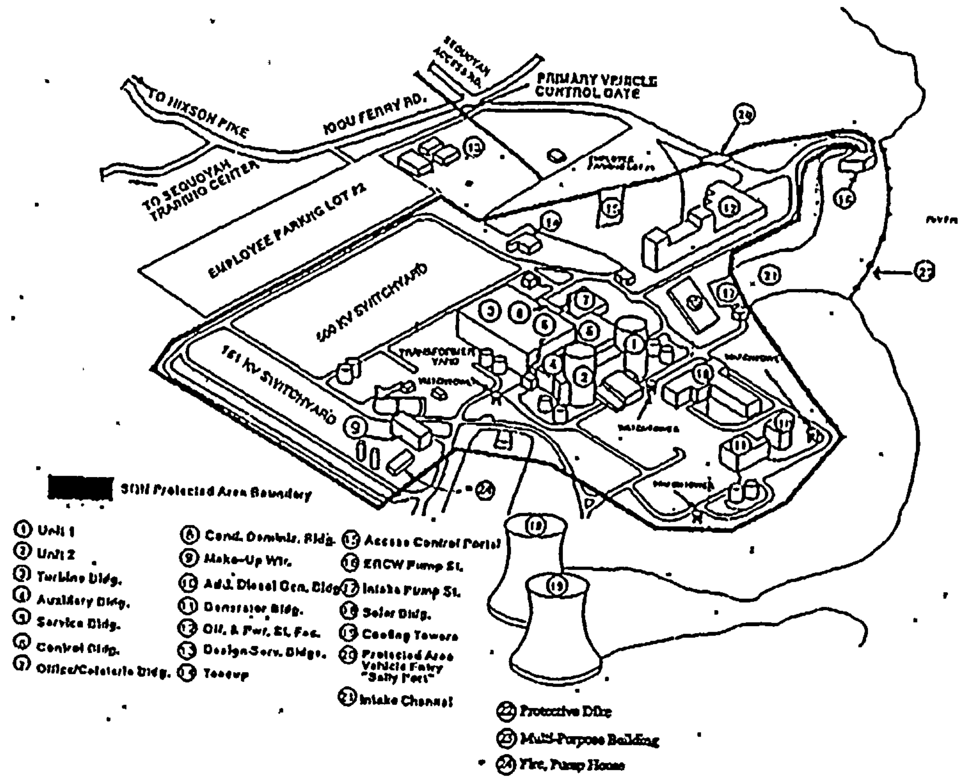
Section 4.0	HAZARDS AND SED JUDGEMENT
Event 4.2	EXPLOSIONS
Classification	ALERT
Mode	All
Description	Explosion in any of the areas listed in Table 4-1 that is affecting safety related equipment required to establish or maintain safe shutdown (1 and 2): 1. Explosion in any of the areas listed in Table 4-1. 2. (a or b): a. Visible damage to permanent structure or safety related equipment in the specified area due to the explosion. OR b. Control room indication of degraded safety system or component response due to the explosion. Refer to Security (Section 4.6).
Basis	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. 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. 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.

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 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 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

Section 4.0	HAZARDS AND SED JUDGEMENT
Event 4.2	EXPLOSIONS
Classification	UNUSUAL EVENT
Mode	All
Description	**UNPLANNED explosion within the protected area (Figure 4-A) resulting in VISIBLE DAMAGE to any permanent structure or equipment.** Refer to Security (Section 4.6).
Basis	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. 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. 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. 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. Unplanned is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities.

Section 4.0	**HAZARDS AND SED JUDGEMENT**
Event 4.2	**EXPLOSIONS**
Classification	**UNUSUAL EVENT (continued)**
Mode	All
Escalation	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.
References	NUMARC/NESP-007, HU1, Rev 2, 1/92 FSAR 2.2 Nearby Industrial, Transportation and Military Facilities

Figure 4-A
SEQUOYAH PROTECTED AREA



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Section 4.0	**HAZARDS AND SED JUDGEMENT**
Event 4.3	**FLAMMABLE GAS**
Classification	**GENERAL EMERGENCY**
Mode	Not Applicable.
Description	Refer to the "Fission Product Barrier Matrix" (Section 1).
Basis	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.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev 2, 1/92

Section 4.0	**HAZARDS AND SED JUDGEMENT**
Event 4.3	**FLAMMABLE GAS**
Classification	**SITE AREA EMERGENCY**
Mode	Not Applicable.
Description	Refer to the "Fission Product Barrier Matrix" (Section 1).
Basis	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.
Escalation	Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
References	NUMARC/NESP-007, Rev 2, 1/92

Section 4.0	HAZARDS AND SED JUDGEMENT
Event 4.3	FLAMMABLE GAS
Classification	ALERT
Mode	All
Description	**UNPLANNED release of flammable gas within a facility structure containing safety related equipment or associated with safe operation of the plant:** *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. Note: Refer to the Material Safety Data Sheet for the LEL.
Basis	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. Table 4-2 Plant structures associated with toxic or flammable gas EALs: 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 Unplanned is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities.
Escalation	Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
References	NUMARC/NESP-007, HA3, Rev 2, 1/92

*Revision

Section	4.0	HAZARDS AND SED JUDGEMENT
Event	4.3	FLAMMABLE GAS
Classification		UNUSUAL EVENT
Mode		All
Description		**A. UNPLANNED release of flammable gas within the EXCLUSION AREA BOUNDARY that may affect normal operations:** *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). OR **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.** Note: Refer to the Material Safety Data Sheet for the LEL.
Basis		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. 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. 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.

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Section 4.0	HAZARDS AND SED JUDGEMENT
Event 4.3	FLAMMABLE GAS
Classification	UNUSUAL EVENT (continued)
Mode	All
Basis (continued)	Unplanned is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities. EXCLUSION AREA BOUNDARY encompasses all areas in the immediate site environs as shown on Figure 4-B.
Escalation	Escalation of this will be based on flammable gases entering a plant area that jeopardizes life, safe operations or impacts cold shutdown capabilities.
References	NUMARC/NESP-007, HU3, Rev 2, 1/92 FSAR 2.2 Nearby Industrial, Transportation and Military Facilities DOT Emergency Response Guide for Hazardous Material

Figure 4-B

SEQUOYAH EXCLUSION AREA BOUNDARY

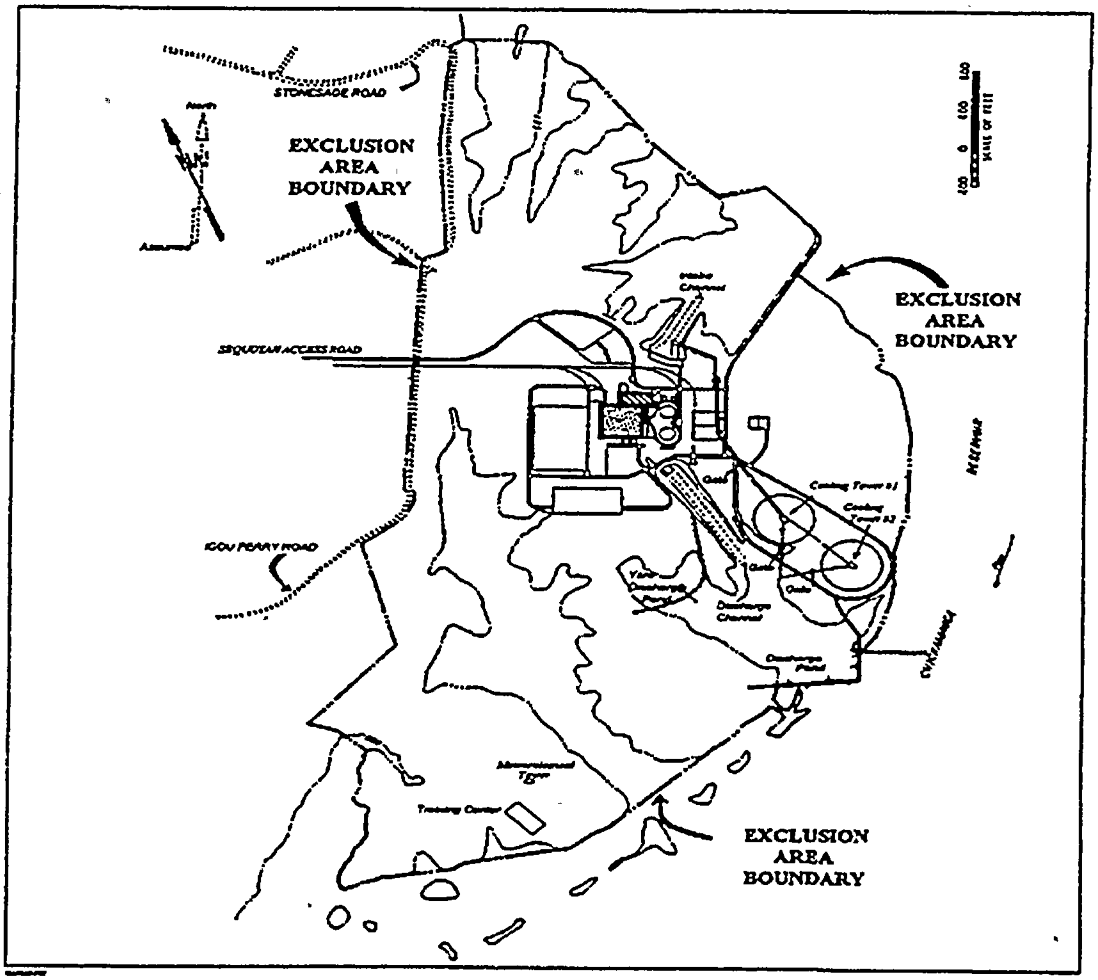
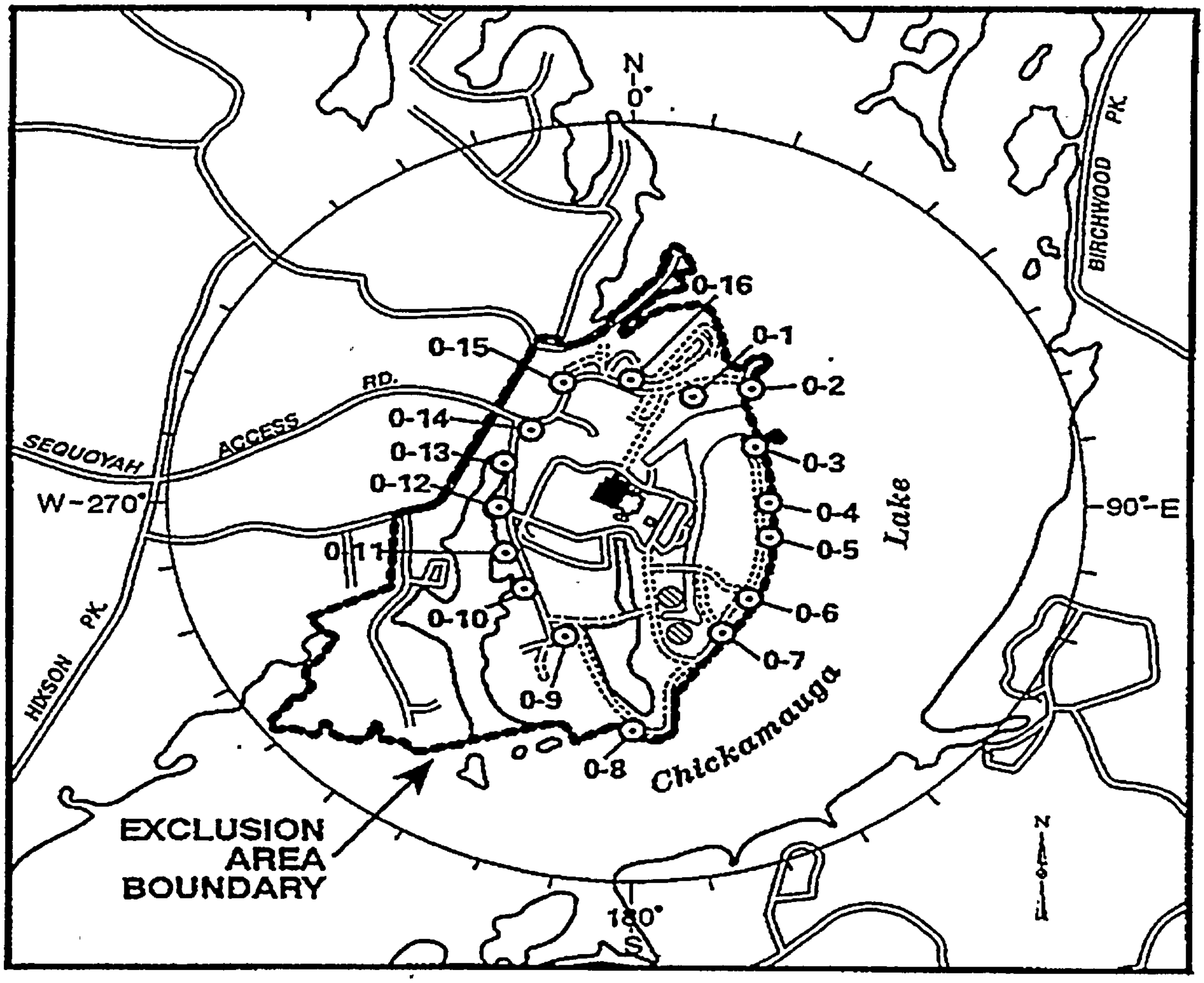


Figure 4-C

SEQUOYAH ONE MILE RADIUS



Section 4.0	HAZARDS AND SED JUDGEMENT
Event 4.4	TOXIC GAS OR SMOKE
Classification	GENERAL EMERGENCY
Mode	Not Applicable.
Description	Refer to the "Fission Product Barrier Matrix" (Section 1).
Basis	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.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev 2, 1/92

Section 4.0	HAZARDS AND SED JUDGEMENT
Event 4.4	TOXIC GAS OR SMOKE
Classification	SITE AREA EMERGENCY
Mode	Not Applicable.
Description	Refer to the "Fission Product Barrier Matrix" (Section 1).
Basis	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.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev 2, 1/92

Section 4.0	HAZARDS AND SED JUDGEMENT
Event 4.4	TOXIC GAS OR SMOKE
Classification	ALERT
Mode	All
Description	**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):** 1. Plant personnel report toxic gas or smoke within any building listed in Table 4-2. 2. (a of b): a. Plant personnel report severe adverse health reactions due to toxic gas or smoke (i.e., burning eyes, nose, throat, dizziness). OR b. Sampling indicates > Permissible Exposure Limit (PEL). 3. Plant personnel unable to perform actions necessary to establish and maintain co[ld] shutdown while utilizing appropriate personnel protection equipment. Note: Refer to the Material Safety Data Sheet for the PEL.
Basis	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. Table 4-2 Plant structures associated with toxic or flammable gas or smoke EALs: 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 TOXIC GAS or SMOKE is a gas that is dangerous to life or limb by reason of inhalation or skin contact (e.g., chlorine, CO_2, etc.).
Escalation	Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
References	NUMARC/NESP-007, HA2, Rev. 2, 1/92

Section 4.0	HAZARDS AND SED JUDGEMENT
Event 4.4	TOXIC GAS OR SMOKE
Classification	UNUSUAL EVENT
Mode	All
Description	A. Safe operations impeded due to access restrictions caused by TOXIC GAS or SMOKE concentrations within a facility structure listed in Table 4-2. OR B. Confirmed report by Local, County, or State officials that an offsite TOXIC GAS release has occurred within one mile of the site (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. Note: Refer to the Material Safety Data Sheet for the PEL.
Basis	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. 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. 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. TOXIC GAS or SMOKE is a gas that is dangerous to life or limb by reason of inhalation or skin contact (e.g., chlorine, CO_2, etc.). Exclusion Area Boundary encompasses all areas in the immediate site environs as shown on Figure 4-B.

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 — 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 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

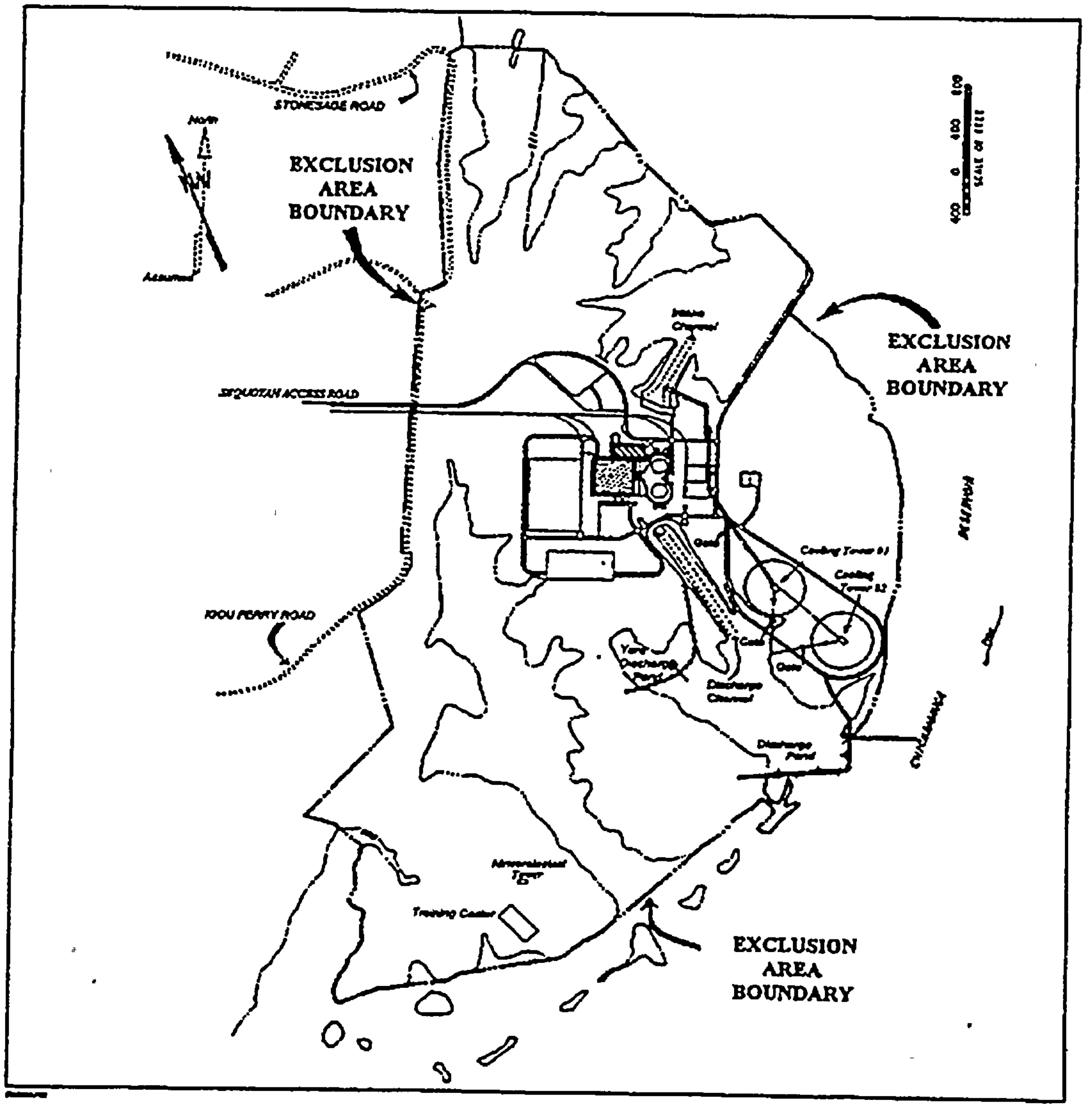
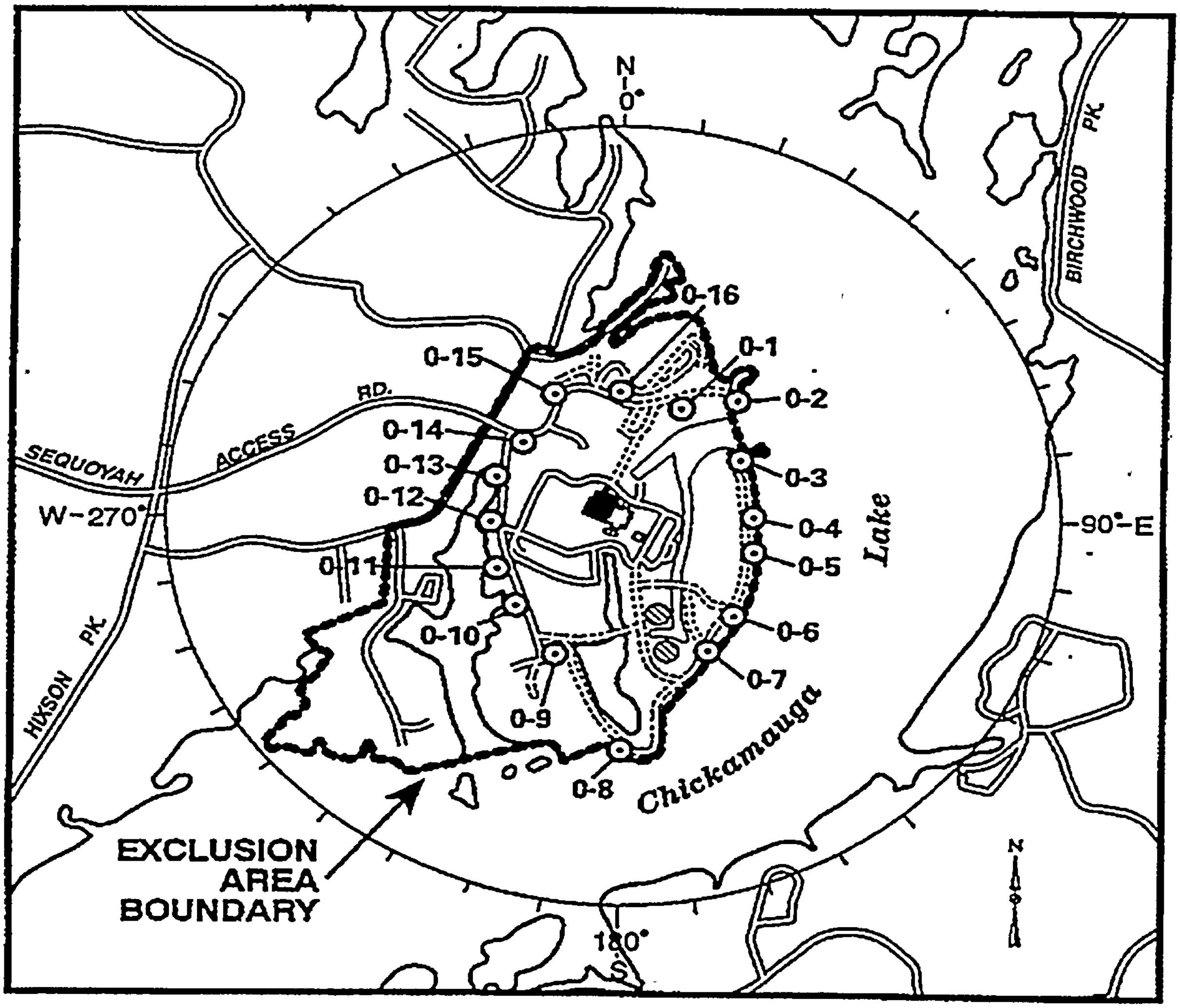


Figure 4-C

SEQUOYAH ONE MILE RADIUS



Section 4.0	**HAZARDS AND SED JUDGEMENT**
Event 4.5	**CONTROL ROOM EVACUATION**
Classification	**GENERAL EMERGENCY**
Mode	Not Applicable.
Description	Refer to the "Fission Product Barrier Matrix" (Section 1).
Basis	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.
Escalation	Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
References	NUMARC/NESP-007, Rev 2, 1/92

Section 4.0	**HAZARDS AND SED JUDGEMENT**
Event 4.5	**CONTROL ROOM EVACUATION**
Classification	**SITE AREA EMERGENCY**
Mode	All
Description	**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):** 1. AOP C.04 "Main Control Room Inaccessibility" entered. 2. Control has not been established within 15 minutes of staffing the auxiliary control room and completing transfer of switches, listed on Checklist AOP C.04-1, on Panels L11A and L11B to the Aux. position.
Basis	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. 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 uncovery and/or core damage per (AOP C.04) Control Room Inaccessibility.
Escalation	Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
References	NUMARC/NESP-007, HS2, Rev 2, 1/92 AOP C.04 Control Room Inaccessibility

Section 4.0	HAZARDS AND SED JUDGEMENT
Event 4.5	CONTROL ROOM EVACUATION
Classification	ALERT
Mode	All
Description	**Evacuation of the control room is required:** *1. AOP C.04 "Control Room Inaccessibility" has been entered.
Basis	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.
Escalation	Escalation will be based on the inability to establish plant control from outside the Main Control Room within 15 minutes.
References	NUMARC/NESP-007, HA5, Rev. 2, 1/92 AOP C.04 Control Room Inaccessibility

Section 4.0	HAZARDS AND SED JUDGEMENT
Event 4.5	CONTROL ROOM EVACUATIONS
Classification	UNUSUAL EVENT
Mode	Not Applicable.
Description	An Unusual Event for this event is "Not Applicable".
Basis	Not Applicable.
Escalation	Escalation will be based on evacuation of the main control room.
References	*NUMARC/NESP-007, Rev 2, 1/92

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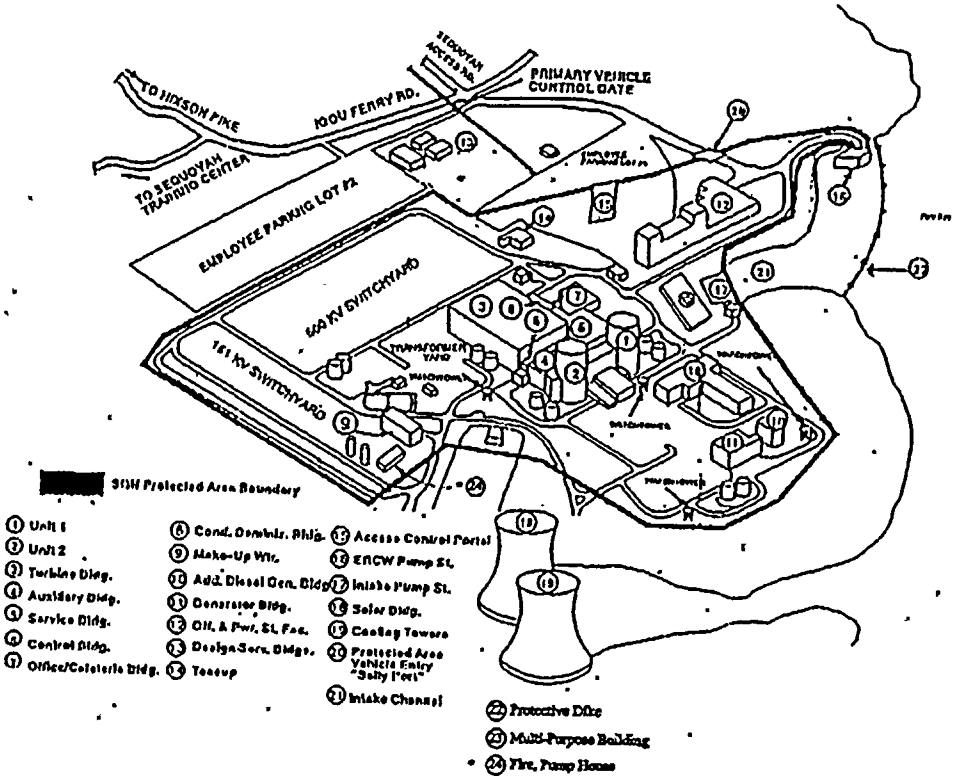
Section 4.0	HAZARDS AND SED JUDGEMENT
Event 4.6	SECURITY
Classification	GENERAL EMERGENCY
Mode	All
Description	**Security event resulting in loss of control of the plant:** Hostile armed force has taken control of the plant, or control room, or remote shutdown capability.
Basis	This event represents a condition where a hostile force has taken control of the main control room or vital areas within the plant that are required to reach and maintain a cold shutdown. This loss could be due to physical loss of control or by the damage of essential equipment. This situation leaves the plant in a very unstable condition with a high potential of multiple barrier failures.
Escalation	Not Applicable.
References	NUMARC/NESP-007, HG1, Rev 2, 1/92

Section 4.0	HAZARDS AND SED JUDGEMENT
Event 4.6	SECURITY
Classification	SITE AREA EMERGENCY
Mode	All
Description	**Security event has or is occurring which results in actual or likely failures of plant functions needed to protect the public:** VITAL AREA, other than the control room, has been penetrated by a hostile armed force.
Basis	This event represents a threat to the safety of the plant since there has been a hostile intrusion into the areas of the plant that contain equipment important to maintaining the plant in a safe condition. A confirmed security event is satisfied when physical evidence of a hostile intrusion exist. VITAL AREA is any area within the protected area (Figure 4-A) which contains equipment, systems, devices, or material, the failure, destruction, or release of which could directly or indirectly endanger the public health and safety by exposure to radiation.
Escalation	Escalation will be based on loss of plant control.
References	NUMARC/NESP-007, HS1, Rev 2, 1/92

Section 4.0	**HAZARDS AND SED JUDGEMENT**
Event 4.6	**SECURITY**
Classification	**ALERT**
Mode	All
Description	**Confirmed security event which indicates an actual or potential substantial degradation in the level of safety of the plant** (1 or 2 or 3): 1. BOMB discovered within a vital area. OR 2. CIVIL DISTURBANCE ongoing within the protected area (Figure 4-A). OR 3. PROTECTED AREA (Figure 4-A) has been penetrated by a hostile armed force.
Basis	These classes of security events represent a threat to the level of safety of the plant. A confirmed report is satisfied if physical evidence supporting the hostile intrusion or bomb is discovered in a vital area. BOMB refers to an explosive device. A CIVIL DISTURBANCE exists when there is a group of twenty (20) or more persons violently protesting station operations or activities at the site.
Escalation	Escalation will be based on hostile intrusion into plant vital areas.
References	NUMARC/NESP-007, HA4, Rev 2, 1/92 Figure 4-A SQN Protected Area

Figure 4-A
SEQUOYAH PROTECTED AREA



* Revision

Section 4.0	HAZARDS AND SED JUDGEMENT
Event 4.6	SECURITY
Classification	UNUSUAL EVENT
Mode	All
Description	**Confirmed security event which indicates a potential degradation in the level of safety of the plant (1 or 2):** 1. BOMB discovered within the protected area (Figure 4-A). <u>OR</u> 2. Security Shift Supervisor reports any of the events listed in Table 4-3.
Basis	A security threat that is identified as being directed towards SQN which represents a potential degradation in the level of safety of the plant warrants declaration of an Unusual Event. A confirmed report is satisfied if physical evidence supporting the threat exists, information independent from the actual threat message exists or a specific group claims responsibility for the threat. In addition, Sequoyah uses a trained security organization and an approved physical security plan and procedures. External events which may result in a security threat *would be reported to the duty Shift Manager (SM) by the Nuclear Security *Supervisor. If in the SM's judgment these events constitute an actual threat, they would be reported and a declaration made. BOMB refers to an explosive device. A HOSTAGE is a person(s) held as leverage against the station to ensure that demands will be met by the station. PROTECTED AREA encompasses all areas within the security protected area fence as shown on Figure 4-A. EXCLUSION AREA BOUNDARY is within the area shown on Figure 4-B.

*Revision

Section 4.0	HAZARDS AND SED JUDGEMENT
Event 4.6	SECURITY
Classification	UNUSUAL EVENT (continued)
Mode	All
Basis (continued)	SABOTAGE is deliberate damage, misalignment, or misoperation of plant equipment with the intent to render the equipment inoperable. A CIVIL DISTURBANCE exists when there is a group of twenty (20) or more persons violently protesting station operations or activities at the site. A STRIKE ACTION is a work stoppage within the protected area by a body of workers to enforce compliance with demands made on TVA. The strike action must threaten to interrupt normal plant operations. EXTORTION is an attempt to cause an action at the station by threat of force. An INTRUSION/INTRUDER is a suspected hostile individual(s) present in a protected area without authorization.
Escalation	Escalation of this event will be based on hostile intrusion into the plant protected area.
References	NUMARC/NESP-007, HU4, Rev. 2, 1/92

Table 4-3 Security Event Examples

a. SABOTAGE/INTRUSION has or is occurring within the PROTECTED AREA (Figure 4-A).

b. HOSTAGE/EXTORTION situation that threatens to interrupt plant operations.

c. CIVIL DISTURBANCE ongoing between the EXCLUSION AREA BOUNDARY (Figure 4-B) and the PROTECTED AREA (Figure 4-A).

d. Hostile STRIKE ACTION within the PROTECTED AREA which threatens to interrupt normal plant operations (judgement based on behavior of strikers and/or intelligence received).

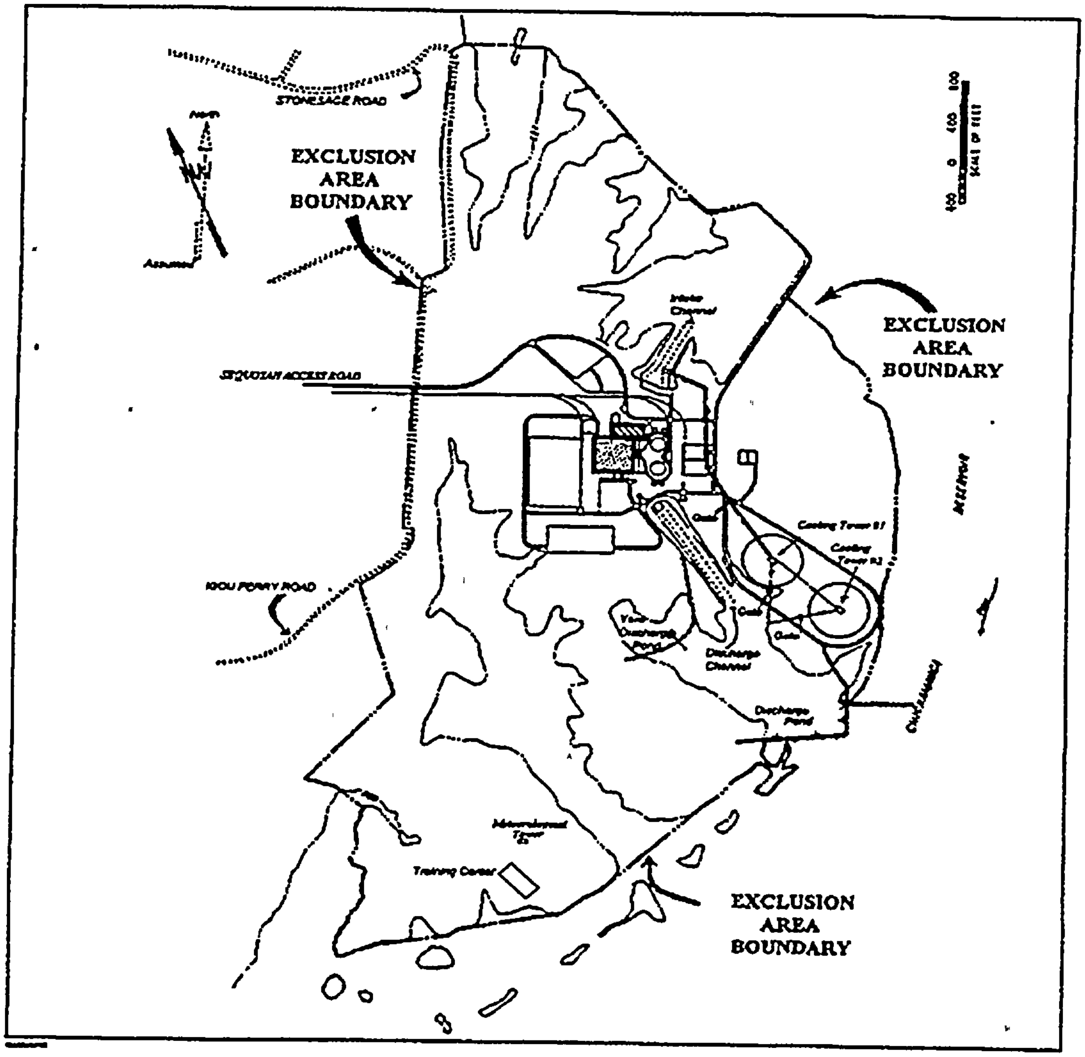
Section 4.0	HAZARDS AND SED JUDGEMENT
Event 4.7	EMERGENCY DIRECTOR JUDGEMENT
Classification	GENERAL EMERGENCY
Mode	All
Description	**Events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases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

Section 4.0	HAZARDS AND SED JUDGEMENT
Event 4.7	EMERGENCY DIRECTOR JUDGEMENT
Classification	SITE AREA EMERGENCY
Mode	All
Description	**Events are in process or have occurred which involve actual or likely major failures of plant functions needed for the protection of the public. Any releases are not expected to result in exposure levels which Exceed EPA Plume Protective Action Guidelines exposure levels outside the exclusion Area boundary (Figure 4-B).**
Basis	*This event classification provides the Shift Manager/Site Emergency Director, the flexibility to declare a Site Area Emergency if in his judgement unanticipated conditions not explicitly covered elsewhere warrant declaration. A Site Area Emergency indicates high probability of major failures of plant functions needed to protect the public.
Escalation	Escalation of this event will be based on actual or imminent substantial core degradation.
References	NUMARC/NESP-007, HS2, Rev. 2, 1/92

*Revision

Figure 4-B

SEQUOYAH EXCLUSION AREA BOUNDARY



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

Section 4.0	HAZARDS AND SED JUDGEMENT
Event 4.7	EMERGENCY DIRECTOR JUDGEMENT
Classification	UNUSUAL EVENT
Mode	All
Description	Events are in process or have occurred which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.
Basis	*This event classification provides the Shift Manager the flexibility to declare an Unusual Event if, in his judgement, unanticipated conditions not explicitly covered elsewhere warrant declaration of an emergency.
Escalation	Escalation will be based on actual degradation of plant safety systems.
References	NUMARC/NESP-007, HU5, Rev. 2, 1/92

*Revision

Section 5.0	**DESTRUCTIVE PHENOMENON**
Event 5.1	**EARTHQUAKE**
Classification	**GENERAL EMERGENCY**
Mode	Not Applicable.
Description	Refer to the "Fission Product Barrier Matrix" (Section 1).
Basis	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.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 5.0	**DESTRUCTIVE PHENOMENON**
Event 5.1	**EARTHQUAKE**
Classification	**SITE AREA EMERGENCY**
Mode	Not Applicable.
Description	Refer to the "Fission Product Barrier Matrix" (Section 1).
Basis	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.
Escalation	Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 5.0	DESTRUCTIVE PHENOMENON
Event 5.1	EARTHQUAKE
Classification	ALERT
Mode	All
Description	**Earthquake detected by site seismic instrumentation** (1 and 2): 1. Panel XA-55-15B alarm window 30 (E-2) plus window 22 (D-1) activated. 2. (a or b) a. Ground motion sensed by plant personnel. OR b. National Earthquake Information Center at (303) 273-8500 can confirm the event.
Basis	A seismic event of this level can cause damage to safety related systems. Plant seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix "A" of 10 CFR Part 100. All specified measurement ranges represent the minimum ranges of the instruments.
Escalation	Escalation of this event will be based on "Fission Product Barrier Matrix" (Section 1).
References	NUMARC/NESP-007, HA1, Rev. 2, 1/92 TRM 3.3.3.3 Seismic Monitoring Instrumentation NUREG 1.12 "Instrumentation for Earthquakes", April 1974 EPRI Report NP-6695 "Guidelines for Nuclear Plant Response to Earthquakes", December 1989

Section 5.0	DESTRUCTIVE PHENOMENON
Event 5.1	EARTHQUAKE
Classification	UNUSUAL EVENT
Mode	All
Description	**Earthquake detected by site seismic instrumentation** (1 and 2): 1. Panel XA-55-15B alarm window 22 (D-1) activated. 2. (a or b) a. Ground motion sensed by plant personnel. OR b. National Earthquake Information Center at (303) 273-8500 can confirm the event.
Basis	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. 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.
Escalation	Escalation of this event will be based on a safe shutdown earthquake (SSE).
References	NUMARC/NESP-007, HU1, Rev. 2, 1/92 TRM 3.3.3.3 Seismic Monitoring Instrumentation NUREG 1.12 "Instrumentation for Earthquakes", April 1974

Section 5.0	**DESTRUCTIVE PHENOMENON**
Event 5.2	**TORNADO**
Classification	**GENERAL EMERGENCY**
Mode	Not Applicable.
Description	Refer to the "Fission Product Barrier Matrix" (Section 1).
Basis	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.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 5.0	**DESTRUCTIVE PHENOMENON**
Event 5.2	**TORNADO**
Classification	**SITE AREA EMERGENCY**
Mode	Not Applicable.
Description	Refer to the "Fission Product Barrier Matrix" (Section 1).
Basis	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.
Escalation	Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 5.0	DESTRUCTIVE PHENOMENON
Event 5.2	TORNADO
Classification	ALERT
Mode	All
Description	**Tornado or high winds strikes any structure listed in Table 5-1 and results in VISIBLE DAMAGE (1 and 2):** 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) a. Confirmed report of any visible damage. OR b. Control room indications of degraded safety system or component response due to the event. **Note: National Weather Service Morristown 1-(423)-586-8400 can provide additional information.*
Basis	Tornados or high winds striking the structures listed in Table 5-1 can cause damage to plant structures or systems needed for safe shutdown of the plant. 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. 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. Table 5-1 Plant Structures Associated With Tornado/Hi Wind EALs 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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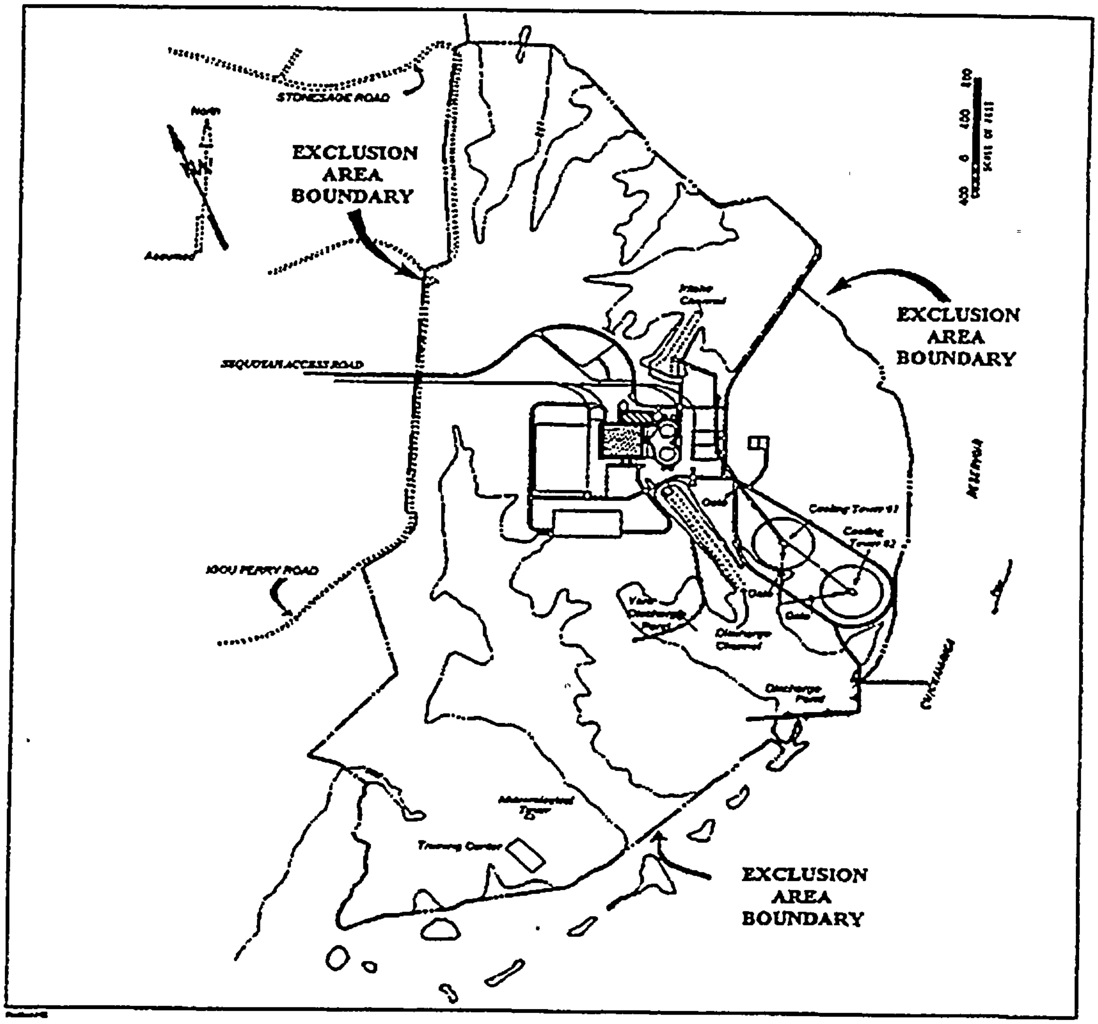
Section 5.0	DESTRUCTIVE PHENOMENON
Event 5.2	TORNADO
Classification	ALERT (continued)
Mode	All
Basis (continued)	VISIBLE DAMAGE is intended to be indicative of observed physical degradation. This damage has to affect plant safety systems or equipment required to establish or maintain cold shutdown.
Escalation	Escalation of this event will be based on" Fission Product Barrier Matrix" (Section 1).
References	NUMARC/NESP-007, HAI, Rev. 2, 1/92 FSAR 1.2 General Plant Description FSAR 2.3 Meteorology

Section 5.0	DESTRUCTIVE PHENOMENON
Event 5.2	TORNADO
Classification	UNUSUAL EVENT
Mode	All
Description	**Tornado within the Exclusion Area Boundary:** Plant personnel report a tornado has been sighted within the exclusion area boundary (Figure 5-A).
Basis	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.
Escalation	Escalation will be based on the tornado striking plant structures or high sustained winds within the protected area.
References	NUMARC/NESP-007, HA1, Rev. 2, 1/92 FSAR 1.2 General Plant Description

*Revision

Figure 5-A

SEQUOYAH EXCLUSION AREA BOUNDARY



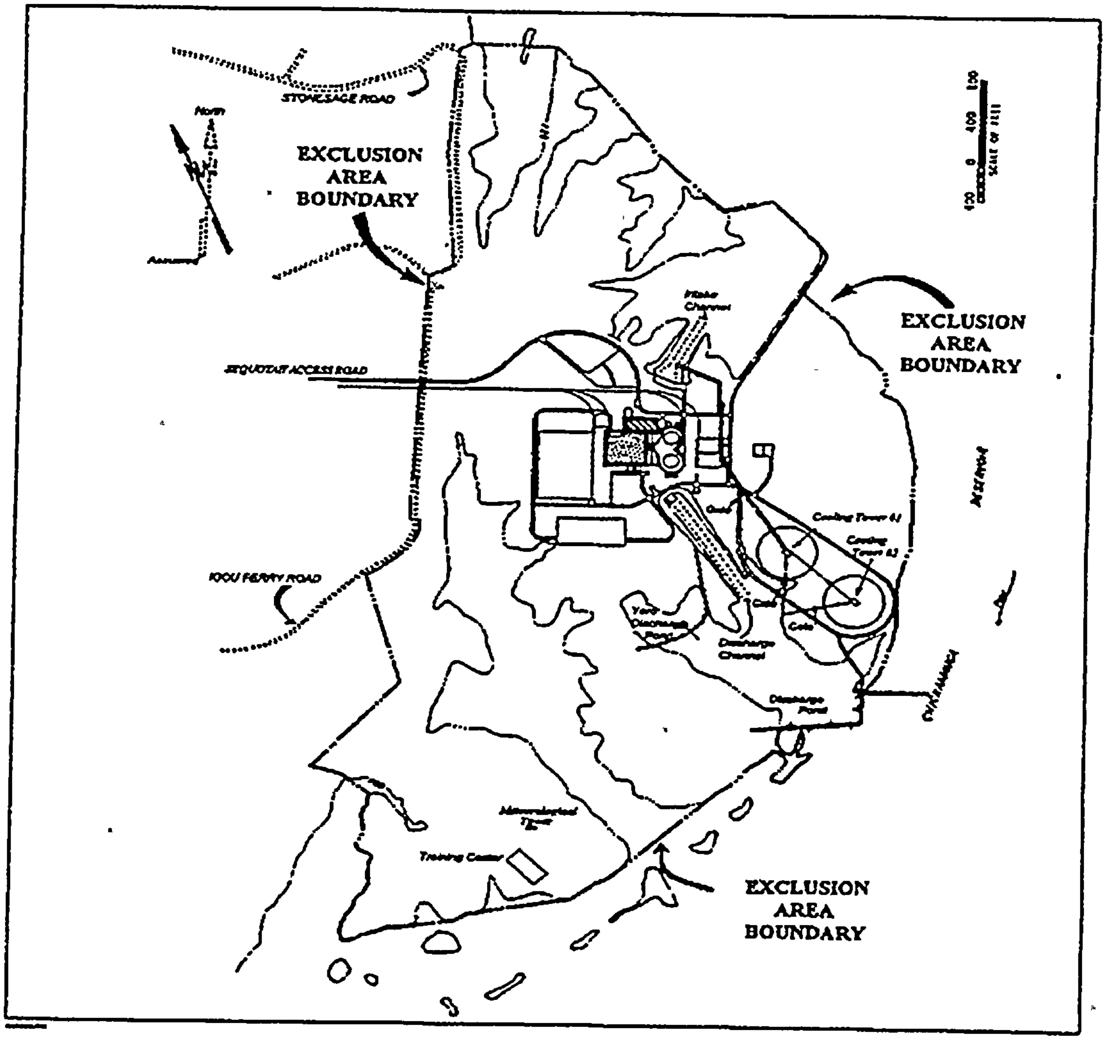
Section 5.0	**DESTRUCTIVE PHENOMENON**
Event 5.3	**AIRCRAFT/PROJECTILE IMPACT**
Classification	**GENERAL EMERGENCY**
Mode	Not Applicable.
Description	Refer to the "Fission Product Barrier Matrix" (Section 1).
Basis	The basis for a General Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 5.0	**DESTRUCTIVE PHENOMENON**
Event 5.3	**AIRCRAFT/PROJECTILE IMPACT**
Classification	**SITE AREA EMERGENCY**
Mode	Not Applicable.
Description	Refer to the "Fission Product Barrier Matrix" (Section 1).
Basis	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.
Escalation	Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 5.0	DESTRUCTIVE PHENOMENON
Event 5.3	AIRCRAFT/PROJECTILE IMPACT
Classification	ALERT
Mode	All
Description	**Aircraft or PROJECTILE impacts (strikes) any plant structure listed in Table 5-1 resulting in visible damage (1 and 2):** 1. Plant personnel report aircraft or projectile has impacted any structure listed in Table 5-1. 2. (a or b) a. Confirmed report of any visible damage. OR b. Control room indications of degraded safety system or component response due to the event within any structure listed in Table 5-1.
Basis	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 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. Table 5-1 Plant structures associated with Tornado/Hi Wind and Aircraft/Projectile EALs: 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

Section 5.0	DESTRUCTIVE PHENOMENON
Event 5.3	AIRCRAFT/PROJECTILE IMPACT
Classification	UNUSUAL EVENT
Mode	All
Description	**AIRCRAFT crash or PROJECTILE impacts (strikes) within the Exclusion Area Boundary:** Plant personnel report aircraft crash or projectile impact within the exclusion area boundary (Figure 5-A).
Basis	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. 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. EXCLUSION AREA BOUNDARY is the boundary shown on Figure 5-A. 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
Escalation	Escalation to this event will be based on an impact on plant structures or barriers.
References	NUMARC/NESP-007, HU1, Rev. 2, 1/92 FSAR 2.2 Nearby Industrial, Transportation And Military Facilities

Figure 5-A
SEQUOYAH EXCLUSION AREA BOUNDARY



Section 5.0	DESTRUCTIVE PHENOMENON
Event 5.4	RIVER LEVEL HIGH
Classification	GENERAL EMERGENCY
Mode	Not Applicable.
Description	Refer to the "Fission Product Barrier Matrix" (Section 1).
Basis	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.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 5.0	DESTRUCTIVE PHENOMENON
Event 5.4	RIVER LEVEL HIGH
Classification	SITE AREA EMERGENCY
Mode	Not Applicable.
Description	Refer to the "Fission Product Barrier Matrix" (Section 1).
Basis	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.
Escalation	Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 5.0	DESTRUCTIVE PHENOMENON
Event 5.4	RIVER LEVEL HIGH
Classification	ALERT
Mode	All
Description	***River reservoir level is at Stage II Flood Warning as reported by River *Operations.**
Basis	The requirements for flood protection ensures that facility protective actions will be *taken and operation will be terminated in the event of flood conditions. A 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. 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.
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 2.4 Hydrologic Engineering AOP N.03 Flooding

*Revision

Section 5.0	DESTRUCTIVE PHENOMENON
Event 5.4	RIVER LEVEL HIGH
Classification	UNUSUAL EVENT
Mode	All
Description	***River reservoir level is at Stage I Flood Warning as reported by River *Operations.**
Basis	The requirements for flood protection ensures that facility protective actions will be *taken and operation will be terminated in the event of flood conditions. A 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. 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. Because of its inland location, the Sequoyah plant is not endangered by tsunami flooding.
Escalation	Escalation of this event will be based on "Fission Product Barrier Matrix" (Section 1).
References	NUMARC/NESP-007, HU1, Rev. 2, 1/92 FSAR 2.4 Hydrologic Engineering AOP N.03 Flooding

*Revision

Section 5.0	DESTRUCTIVE PHENOMENON
Event 5.5	RIVER LEVEL LOW
Classification	GENERAL EMERGENCY
Mode	Not Applicable.
Description	Refer to the "Fission Product Barrier Matrix" (Section 1).
Basis	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.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 5.0	DESTRUCTIVE PHENOMENON
Event 5.5	RIVER LEVEL LOW
Classification	SITE AREA EMERGENCY
Mode	Not Applicable.
Description	Refer to the "Fission Product Barrier Matrix" (Section 1).
Basis	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.
Escalation	Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 5.0	DESTRUCTIVE PHENOMENON
Event 5.5	RIVER LEVEL LOW
Classification	ALERT
Mode	All
Description	***River reservoir level < 670 feet as reported by River Operations.**
Basis	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 670 feet. This level applies for ERCW supply temperature less than or equal to 83°F. The limitations on minimum water level are based on providing sufficient flow to the ERCW heat loads after a postulated event assuming a time-dependent drawdown of reservoir level. Flow to the major transient heat loads (CCS and CS heat exchangers) is balanced assuming a reservoir level or elevation 670 feet. The time-dependent heatloads (ESF room coolers, etc.) are balanced assuming a reservoir level of elevation 636 feet. 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. 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.
Escalation	Escalation to this event will be based on "Fission Product Barrier Matrix" (Section 1).
References	*NUMARC/NESP-007, HA1, 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

*Revision

Section 5.0	DESTRUCTIVE PHENOMENON
Event 5.5	RIVER LEVEL LOW
Classification	UNUSUAL EVENT
Mode	All
Description	*River reservoir level < 673 feet as reported by River Operations.
Basis	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 670 feet. This level applies for ERCW supply temperature less than or equal to 83°F. 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. Because of its location on 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.
Escalation	Escalation to this event will be based on reduced river levels.
References	*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

*Revision

Section 5.0	DESTRUCTIVE PHENOMENON
Event 5.6	WATERCRAFT CRASH
Classification	GENERAL EMERGENCY
Mode	Not Applicable.
Description	Refer to the "Fission Product Barrier Matrix" (Section 1).
Basis	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.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 5.0	DESTRUCTIVE PHENOMENON
Event 5.6	WATERCRAFT CRASH
Classification	SITE AREA EMERGENCY
Mode	Not Applicable.
Description	Refer to the "Fission Product Barrier Matrix" (Section 1).
Basis	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.
Escalation	Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 5.0	DESTRUCTIVE PHENOMENON
Event 5.6	WATERCRAFT CRASH
Classification	ALERT
Mode	Not Applicable.
Description	Refer to the "Fission Product Barrier Matrix" (Section 1).
Basis	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.
Escalation	Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 5.0	DESTRUCTIVE PHENOMENON
Event 5.6	WATERCRAFT CRASH
Classification	UNUSUAL EVENT
Mode	All
Description	**Watercraft strikes the ERCW pumping station resulting in a reduction of Essential Raw Cooling Water (ERCW) (1 and 2):** 1. Plant personnel report a watercraft has struck the ERCW pumping station. 2. (a or b) a. ERCW supply header pressure Train A 1(2)-PI-67-493A is <15 psig. OR b. ERCW supply header pressure train B 1(2)-PI-67-488A is <15 psig.
Basis	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.
Escalation	Escalation would be based on "Fission Product Barrier Matrix" (Section 1).
References	*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

*Revision

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Section 6.0	**SHUTDOWN DEGRADATION**
Event 6.1	**LOSS OF SHUTDOWN SYSTEMS**
Classification	**GENERAL EMERGENCY**
Mode	Not Applicable.
Description	Refer to "Gaseous Effluents" (Section 7.1).
Basis	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.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 6.0	**SHUTDOWN DEGRADATION**
Event 6.1	**LOSS OF SHUTDOWN SYSTEMS**
Classification	**SITE AREA EMERGENCY**
Mode	5,6
Description	**Loss of water level in the reactor vessel that has or will uncover active fuel in the reactor vessel with CNTMT closure established (1 and 2 and 3 and 4):** 1. Loss of RHR capability. *2. **VALID** indication that reactor vessel water level < 695'. 3. Incore TCs (if available) indicate **RCS** temperature > 200°F. 4. Containment closure is established. Note: If containment is open, refer to "Gaseous Effluents" (Section 7.1).
Basis	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 uncovery 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 uncovery 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. The reactor vessel level indication of elevation 695' represents the water level at the hot leg center line.
Escalation	Escalation to this event will be based on "Gaseous Effluent" (Section 7.1).
References	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

*Revision

Section 6.0	SHUTDOWN DEGRADATION
Event 6.1	LOSS OF SHUTDOWN SYSTEMS
Classification	ALERT
Mode	5,6
Description	**Inability to maintain unit in cold shutdown when required with containment closure established (1 and 2 and 3):** 1. Cold shutdown required by technical specifications. 2. Incore TCs (if available) indicate core exit temperature is >200° F. 3. Containment closure is established. NOTE: If containment is open, refer to "Gaseous Effluents" (Section 7.1).
Basis	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. 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. 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 uncovery can occur. NRC analyses show that these sequences can cause core uncovery 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 200^0 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

Section 6.0	**SHUTDOWN DEGRADATION**
Event 6.1	**LOSS OF SHUTDOWN SYSTEMS**
Classification	**ALERT (continued)**
Mode	5,6
Escalation	Loss of water level in the reactor vessel that has or will uncover fuel in the vessel will escalate this event.
References	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

Section 6.0	**SHUTDOWN DEGRADATION**
Event 6.1	**LOSS OF SHUTDOWN SYSTEMS**
Classification	**UNUSUAL EVENT**
Mode	Not Applicable.
Description	An Unusual Event for this event is "Not Applicable".
Basis	Not Applicable.
Escalation	Escalation will be based on inability to maintain cold shutdown.
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 6.0	**SHUTDOWN SYSTEM DEGRADATION**
Event 6.2	**LOSS OF SHUTDOWN CAPABILITY**
Classification	**GENERAL EMERGENCY**
Mode	Not Applicable.
Description	Refer to the "Fission Product Barrier Matrix" (Section 1).
Basis	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.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 6.0	**SHUTDOWN SYSTEM DEGRADATION**
Event 6.2	**LOSS OF SHUTDOWN CAPABILITY**
Classification	**SITE AREA EMERGENCY**
Mode	1,2,3,4
Description	**Complete loss of function needed to achieve or maintain hot shutdown (1 and 2)** 1. Hot shutdown required. 2. (a or b) a. CSF status tree indicates Core Cooling Red (FR-C.1). <u>OR</u> b. CSF status tree indicates Heat Sink Red (FR-H.1) (RHR shutdown cooling not in service). Note: Also refer to "Failure of Rx Protection" (Section 2.3).
Basis	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.
Escalation	Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
References	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

Section 6.0	**SHUTDOWN SYSTEM DEGRADATION**
Event 6.2	**LOSS OF SHUTDOWN CAPABILITY**
Classification	ALERT
Mode	1,2,3,4
Description	**Complete loss of function needed to achieve cold shutdown when shutdown required by Tech. Specs.(1 and 2 and 3):** 1. Shutdown is required by technical specifications. 2. Loss of RHR capability 3. Loss of secondary heat sink and condenser.
Basis	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. 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.
Escalation	Escalation will be based on complete loss of functions needed to achieve or maintain hot shutdown.
References	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

Section 6.0	**SHUTDOWN SYSTEM DEGRADATION**
Event 6.2	**LOSS OF SHUTDOWN CAPABILITY**
Classification	**UNUSUAL EVENT**
Mode	1,2,3,4
Description	**Inability to reach required shutdown within Tech. Spec. limits:** The unit has not been placed in the required mode within the time prescribed by the LCO action statement.
Basis	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.
Escalation	Escalation will be based on complete loss of functions needed to achieve cold shutdown.
References	NUMARC/NESP-007, SU2, Rev 2, 1/92

Section 6.0	SHUTDOWN SYSTEM DEGRADATION
Event 6.3	LOSS OF RCS INVENTORY
Classification	GENERAL EMERGENCY
Mode	Not Applicable.
Description	Refer to "Gaseous Effluents" (Section 7.1).
Basis	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.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 6.0	SHUTDOWN SYSTEM DEGRADATION
Event 6.3	LOSS OF RCS INVENTORY
Classification	SITE AREA EMERGENCY
Mode	Not Applicable.
Description	Refer to "Gaseous Effluents" (Section 7.1).
Basis	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.
Escalation	Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 6.0	**SHUTDOWN SYSTEM DEGRADATION**
Event 6.3	**LOSS OF RCS INVENTORY**
Classification	**ALERT**
Mode	Not Applicable.
Description	Refer to "Gaseous Effluents" (Section 7.1).
Basis	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.
Escalation	Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
References	NUMARC/NESP-007 Rev. 2, 1/92

Section 6.0	**SHUTDOWN SYSTEM DEGRADATION**
Event 6.3	**LOSS OF RCS INVENTORY**
Classification	**UNUSUAL EVENT**
Mode	5,6
Description	**Loss of reactor coolant system inventory with inadequate makeup (1 and 2 and 3):** 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.
Basis	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. Unplanned is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities.
Escalation	Escalation will be based on "Fission Product Barrier Matrix" (Section 1).
References	NUMARC/NESP-007, SU7, Rev. 2, 1/92

Section 7.0	**RADIOLOGICAL EFFLUENTS**
Event 7.1	**GASEOUS EFFLUENTS**
Classification	**GENERAL EMERGENCY**
Mode	All
Description	**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):** 1. A VALID rad monitor reading exceeds the values under General Emergency in Table 7-1 (page 162) for > 15 minutes, unless assessment within that 15 minutes confirms that the criterion is not exceeded. OR 2. Field surveys indicate > 1000 mrem/hr β-γ or an I-131 concentration of 3.9E-06 $\mu Ci/cm^3$ at exclusion area boundary (Figure 7-A). OR 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). NOTE: TEDE = Total Effective Dose Equivalent and CDE = Committed Does Equivalent
Basis	If the monitor values in Table 7-1 are met or exceeded the required assessment must be performed. The assessment method is SQN EPIP-14. If the assessment cannot be completed within 15 minutes the appropriate emergency classification will be made based on the valid reading. Calculation To calculate the General Emergency gaseous effluent monitor values for Table 7-1, 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. Total Site General Emergency EAL = 1.31E+09 μCi/s

Section 7.0	RADIOLOGICAL EFFLUENTS
Event 7.1	GASEOUS EFFLUENTS
Classification	GENERAL EMERGENCY (continued)
Mode	All
Basis (continued)	The monitor specific allowable release rates are then converted into actual monitor readings which are presented in the Table. The GE limits are a factor of 10 greater than the SAE values; therefore, the monitor values for a GE are:

Shield Bldg. 1-,2-RM-90-400	1.31E+09 µCi/s
Auxiliary Bldg. 0-RM-90-101B	2.76 E+08 cpm
Service Bldg. 0-RM-90-132B	7.00 E+09 cpm
Steam Generator Discharge 1-,2-RM-90-421 thru -424	3.98 E+02 µCi/cc
Condenser Vacuum Exhaust 1-,2-RM-90-255,-256	1.09 E+ 06 mR/h

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.

The EXCLUSION AREA BOUNDARY is shown in Figure 7-A.

Escalation	Not Applicable.
References	NUMARC/NESP-007, AGI, Rev. 2, 1/92 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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Section 7.0	RADIOLOGICAL EFFLUENTS
Event 7.1	GASEOUS EFFLUENTS
Classification	SITE AREA EMERGENCY
Mode	All
Description	**Exclusion Area Boundary β-γ 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):** 1. A VALID rad monitor reading exceeds the values under Site Area in Table 7-1 (page 162) 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 β-γ or an I-131 concentration of 3.9E-07 $\mu Ci/cm^3$ at exclusion area boundary (Figure 7-A). <u>OR</u> 3. 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).
Basis	If the monitor values in Table 7-1 are met or exceeded the required assessment must be performed. The assessment method is SQN EPIP-14. If the assessment cannot be completed within 15 minutes the appropriate emergency classification will be made based on the valid reading. <u>Calculation</u> To calculate the SAE gaseous effluent EAL values the release rates that are calculated are those required to deliver the EAL dose in one hour. To perform the calculation, the mix fractions from the releases in the SQN FSAR Table 11.3.6-2 (SAR change #15-79) and the dose factors from EPA-400 (Manual of Protective Action Guides and protective Actions for Nuclear Incidents) Tables 5.1 and 5.2 are used in conjunction with the annual average meteorology for the period 1977-1993. However, the annual frequency (0.19) that the wind blows into the critical sector was removed since this is a short-term application. EAL release rates are backcalculated 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. First, a EAL release rate is backcalculated from a 100 mrem TEDE dose as follows:

Section 7.0	RADIOLOGICAL EFFLUENTS
Event 7.1	GASEOUS EFFLUENTS
Classification	SITE AREA EMERGENCY(continued)
Mode	All

Basis (continued)

Table 1

Nuclide	EPA-400 EDE (rem/hr per uCi/cc)	SQN FSAR (Ci/yr/unit)	Mix Fraction	Mix x EDE (rem/hr per uCi/cc)
Kr-85m	93	41.5	3.88E-03	3.61E-01
Kr-85	1.3	1310	1.22E-01	1.59E-01
Kr-87	510	19.4	1.81E-03	9.24E-01
Kr-88	1300	54.9	5.13E-03	6.67E+00
Xe-131m	4.9	2320	2.17E-01	1.06E+00
Xe-133m	17	104	9.72E-03	1.65E-01
Xe-133	20	6480	6.05E-01	1.21E+01
Xe-135m	250	8.71	8.14E-04	2.03E-01
Xe-135	140	353	3.30E-02	4.62E+00
Xe-137	110	1.55	1.45E-04	1.59E-02
Xe-138	720	7.8	7.29E-04	5.25E-01
I-131	53000	0.166	1.55E-05	8.22E-01
I-132	4900	0.661	6.18E-05	3.03E-01
I-133	15000	0.454	4.24E-05	6.36E-01
I-134	3100	1.06	9.90E-05	3.07E-01
I-135	8100	0.824	7.70E-05	6.24E-01
Total		10704		29.50

EAL (µCi/s) = 0.1 rem x 0.19/(29.5 rem/h per µCi/cc x 4.90E-06 sec/m^3 x 1E-06 m^3/cc) = 1.31E+08

Where, X/Q = 4.90E-06 sec/m^3 for 1977-1993 meteorological data

A second calculation is performed to determine a noble gas release rate which will correspond to the associated iodine release rate resulting in 500 mrem CDE thyroid.

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Section 7.0	RADIOLOGICAL EFFLUENTS
Event 7.1	GASEOUS EFFLUENTS
Classification	SITE AREA EMERGENCY (continued)
Mode	All

Basis (continued)

The EAL release rate is backcalculated from a 500 mrem CDE (thyroid) as follows:

Table 2

Nuclide	EPA-400 Thyroid CDE (rem/hr per uCi/cc)	SQN FSAR (Ci/yr/unit)	Mix Fraction	Mix x EDE (rem/hr per uCi/cc)
I-131	1.3E+06	0.166	1.55E-05	2.02E+01
I-132	7.7E+03	0.661	6.18E-05	4.75E-01
I-133	2.2E+05	0.454	4.24E-05	9.33E+00
I-134	1.3E+03	1.06	9.90E-05	1.29E-01
I-135	3.8E+04	0.824	7.70E-05	2.93E+00
Total		3.17		33.02

EAL (μCi/s) = 0.5 rem x 0.19/(33.02 rem/h per μCi/cc x 4.90E-06 sec/m^3 x 1E-06 m^3/cc) = 5.87E+08

Therefore, the EAL based on limiting the TEDE is the most conservative and is used for the SAE determination; therefore, the SAE noble gas release rate value used will be 1.31E+08 μCi/s.

The monitor specific allowable release rates are then converted into actual monitor readings as follows:

<u>Shield Building Radiation Monitors</u> **(1-,2-RM-90-400)**

These monitors read in the Control Room in units of μCi/s ; therefore, no calculation is needed.

<u>Service Building Radiation Monitor</u> **(0-RM-90-132B)**

Using the Xe-133 efficiency for the monitor from TI-18, and assuming the maximum system design ventilation flow of 9000 cfm, the monitor reading is obtained as:

$$\text{cpm} = \frac{\text{Allowable Release Rate } (\mu\text{Ci/s}) \text{ Efficiency (cpm per } \mu\text{Ci/cc)}}{\text{Effluent Flow Rate (cfm) Conversion Factor (cc/sec per cfm)}}$$

= 1.31E+08 * 2.27E+07 / (9000 * 472) = 7.00E+08 cpm

<u>Auxiliary Building Radiation Monitor</u> **(0-RM-90-101B)**

Using the formula above, and assuming the maximum system design ventilation flow of 228,000 cfm, the monitor reading is obtained as:

= 1.31E+08 * 2.27E+07 / (228,000 * 472) = 2.76E+07 cpm

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Section 7.0	RADIOLOGICAL EFFLUENTS
Event 7.1	GASEOUS EFFLUENTS
Classification	SITE AREA EMERGENCY (continued)
Mode	All
Basis (continued)	<u>Condenser Vacuum Exhaust Radiation Monitors</u> (1-,2-RM-90-255, -256) Using the formula above and assuming the maximum system design ventilation flow of 45 cfm, with a monitor efficiency of 0.05625 µCi/cc per mR/h (from engineering calculation SQNAPS3-100), the monitor reading in mR/h is obtained as: = 1.31E+08 / (45 * 0.05625 * 472) = 1.09E+05 mR/h <u>Steam Generator Discharge Monitors</u> (1-,2-RM-90-421 thru -424) Using the same methodology as for the NOUE we have, 1.31E+08 µCi/s /(3292000 cc/s) = 3.98E+01 µCi/cc
Escalation	Escalation will be based on increased release rates by a factor of 10.
References	NUMARC/NESP-007, AGI, Rev. 2, 1/92 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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Section 7.0	RADIOLOGICAL EFFLUENTS
Event 7.1	GASEOUS EFFLUENTS
Classification	ALERT
Mode	All
Description	**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):** 1. A VALID rad monitor reading exceeds the values under Alert in Table 7-1 (page 162) for > 15 minutes, unless assessment within that 15 minutes confirms that the criterion is not exceeded. OR 2. Field surveys indicate > 10 mrem/hr β-γ at the exclusion area boundary (Figure 7-A) > 15 minutes. OR 3. Dose assessment results indicate exclusion area boundary (Figure 7-A) dose >10 mrem TEDE for the duration of the release. OR 4. Sample results exceed 200 times the ODCM limit value for an unmonitored release of gaseous radioactivity > 15 minutes in duration. NOTE: TEDE = Total Effective Dose Equivalent
Basis	If the monitor values in Table 7-1 are met or exceeded the required assessment must be performed. The assessment method is SQN EPIP-14. If the assessment cannot be completed within 15 minutes the appropriate emergency classification will be made based on the valid reading. Calculation To calculate the gaseous effluent monitor values for Table 7.1, (Effluent Radiation Monitor EALs), the release rates corresponding to the ODCM limit as determined for the Unusual Event are used. For gaseous releases, the EAL value can be determined by multiplying the Unusual Event noble gas monitor readings by a factor of 100. Total Site Alert EAL = 4.90E+07 µCi/s The NOUE monitor reading values are also multiplied by a factor of 100:

Monitor	Alert Monitor Reading
Shield Bldg. 1-,2-RM-90-400	4.90E+07 µCi/s
Auxiliary Bldg. 0-RM-90-101B	1.03E+07 cpm
Service Bldg. 0-RM-90-132B	2.62E+08 cpm
Steam Generator Discharge 1,2-RM-90-421 thru -424	1.49E+01 µCi/cc
Condenser Vacuum Exhaust 1-,2-RM-90-255, -256	4.10E+04 mR/h

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Section 7.0	RADIOLOGICAL EFFLUENTS
Event 7.1	GASEOUS EFFLUENTS
Classification	ALERT (continued)
Mode	All
Basis (continued)	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. 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).
Escalation	Escalation will be based on dose rates greater than 100 mrem TEDE or 500 mrem thyroid CDE.
References	NUMARC/NESP-007, AAI, Rev. 2, 1/92 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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Section 7.0	RADIOLOGICAL EFFLUENTS
Event 7.1	GASEOUS EFFLUENTS
Classification	UNUSUAL EVENT
Mode	All
Description	**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):** 1. A VALID rad monitor reading exceeds the values under Unusual Event in Table 7-1 (Page 162) for >60 minutes, unless assessment within that 60 minutes confirms that the criterion is not exceeded. OR 2. Field surveys indicate >0.1 mrem/hr β-γ at the exclusion area boundary (Figure 7-A) > 60 minutes. OR 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. NOTE: TEDE = Total Effective Dose Equivalent
Basis	If the monitor values in Table 7-1 are met or exceeded the required assessment must be performed. The assessment method is SQN EPIP-14. If the assessment cannot be completed within 60 minutes the appropriate emergency classification will be made based on the valid reading. Calculation To calculate the gaseous effluent monitor values for Table 7.1, the first step is to determine release rates which correspond to the ODCM limit. The SQN OCDM limit is: $S_{DEF} = \frac{A \cdot DR_{LIM}}{2.94E+02 \cdot X/Q}$ (equation 1) Where: A = dose rate allocation factor for the release point, dimension less. The dose rate allocation factors for release points are defined based on design flowrates. DR_{lim} = the dose rate limit, 500 mrem/year to the total body for noble gases. X/Q = 6.94E-06 s/m^3 2.94E+02 = Dose Factor for Xe-133 $mrem/y/\mu Ci/m^3$ Using Equation 1, the total site (allocation factor = 1.0) noble gas release rate corresponding to the ODCM limit is equal to: 2.45E+05 μCi/s The NOUE site gaseous release rate value is then equal to: 2 * 2.45E+05 μCi/s = 4.90E+05 μCi/s.

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Section 7.0	RADIOLOGICAL EFFLUENTS
Event 7.1	GASEOUS EFFLUENTS
Classification	UNUSUAL EVENT (continued)
Mode	All
Basis (continued)	

The monitor specific allowable release rates are then converted into actual monitor readings which are presented in Table 7.1.

Shield Building Radiation Monitors (1-,2-RM-90-400A)

These monitors read in the Control Room in units of µCi/s ; therefore, no calculation is needed.

Service Building Radiation Monitor (0-RM-90-132B)

Using the Xe-133 efficiency for the monitor and ventilation flow from TI-18 of 9000 cfm, the monitor reading is obtained as:

472 = Conversion Factor, (28317 $cc/ft^3 \cdot min/60\ s$).

$$\text{cpm} = \frac{\text{Allowable Release Rate } (\mu Ci/s) \times \text{Efficiency (cpm per } \mu Ci/s)}{\text{Effluent Flow Rate (cfm)} \times \text{Conversion Factor (cc/sec per cfm)}}$$

= 4.90E05 * 2.27E+07 / (9000 * 472) = 2.62+06 cpm

Auxiliary Building Radiation Monitor (0-RM-90-101B)

Using the formula above, and assuming the maximum system design ventilation flow of 228,000 cfm, the monitor reading is obtained as:

= 4.90E+05 * 2.27E+07 / (228,000 * 472) = 1.03E+05 cpm

Condenser Vacuum Exhaust Radiation Monitors (1-,2-RM-90-255, -256)

Using the formula above and assuming the maximum system design ventilation flow of 45 cfm, with a monitor efficiency of 0.05625 µCi/cc per mR/h (from engineering calculation SQNAP3-100), the monitor reading in cpm is obtained as:

= 4.90E+5 / (45 * 0.05625 * 472) = 4.10E+02 mR/h

Steam Generator Discharge Monitors (1-,2-RM-90-421 thru -424)

To determine the readings for the Steam Generator (SG) Discharge Monitors,1-RM-90-421 through 424 in mr/hr, for the four emergency classifications, the calculation utilized a flow rate of 890,000 lb/hr at 1085 psig and 600 F. This is considered the limit of one SG, PORV (ref. SQN-DC-V-4.4.1, Main Steam System).

Section 7.0	RADIOLOGICAL EFFLUENTS
Event 7.1	GASEOUS EFFLUENTS
Classification	UNUSUAL EVENT (continued)
Mode	All
Basis (continued)	The 890,000 lb/hr was recalculated to 3292000 cc/sec (using the ASME steam tables) so that this value could be utilized with the calibration factor in the calculation. The curie limit associated with the Unusual Event was 4.90E+05 µCi/sec. By dividing this limit by the steam flow rate, a limit of 1.49E-01 µCi/cc, was calculated for the NOUE. 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). 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. The significance of the time factor to this criterion is primarily related to loss of control of radioactive material that has allowed the release to continue unabated for 60 minutes. It is this aspect rather than the magnitude of the release that establishes "...a potential degradation in the level of safety of the plant..." the fundamental definition of an Unusual Event.
Escalation	Escalation would be based on increasing the magnitude of the release by a factor of 100.
References	NUMARC/NESP-007, AUI, Rev. 2, 1/92

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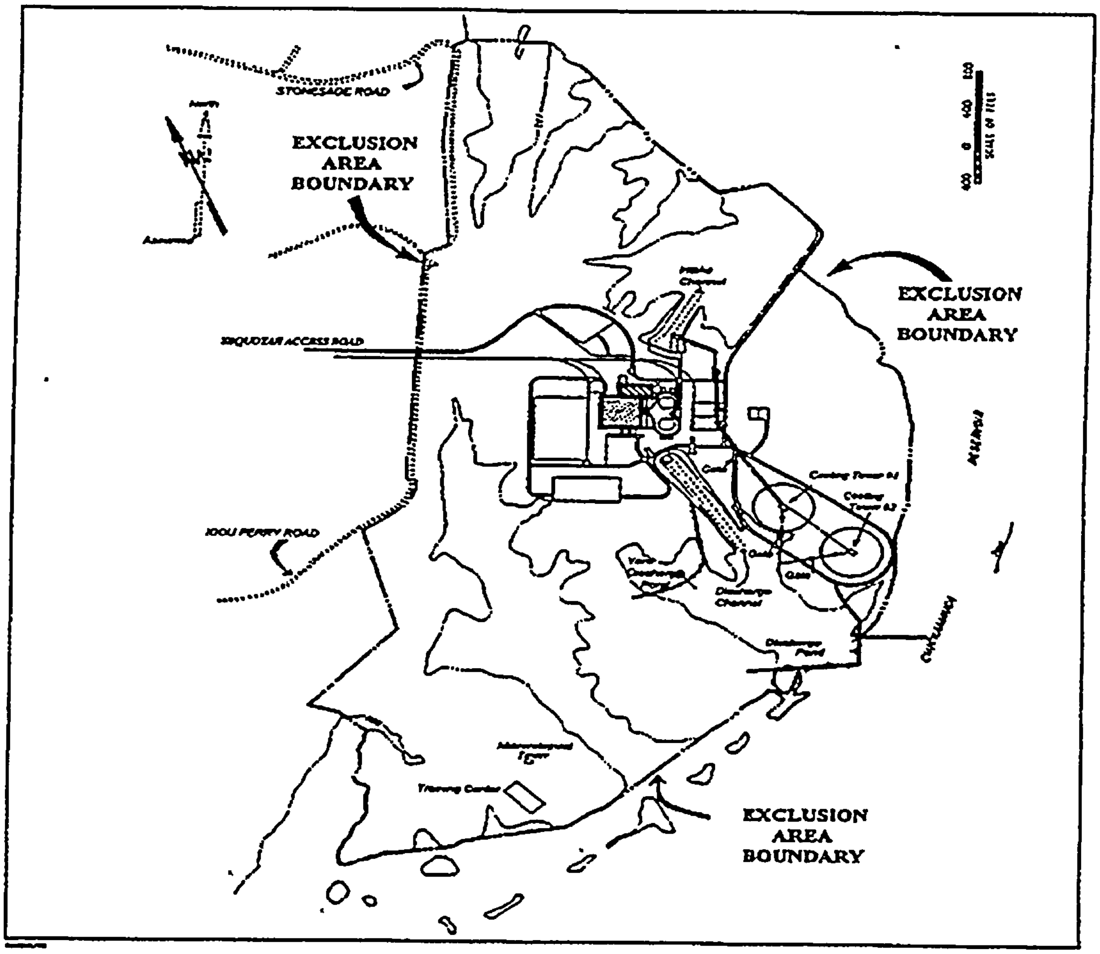
Figure 7-A

SEQUOYAH EXCLUSION AREA BOUNDARY



Section 7.0	RADIOLOGICAL EFFLUENTS
Event 7.2	LIQUID EFFLUENTS
Classification	GENERAL EMERGENCY
Mode	Not Applicable.
Description	Not Applicable.
Basis	Not Applicable.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 7.0	RADIOLOGICAL EFFLUENTS
Event 7.2	LIQUID EFFLUENTS
Classification	SITE AREA EMERGENCY
Mode	Not Applicable.
Description	Not Applicable.
Basis	Not Applicable.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 7.0	RADIOLOGICAL EFFLUENTS
Event 7.2	LIQUID EFFLUENTS
Classification	ALERT
Mode	All
Description	**Any UNPLANNED release of liquid radioactivity that exceeds 200 times the ODCM Section 1.2.1.1 limit for >15 minutes (1 or 2):** 1. A valid rad monitor reading exceeds the values under Alert in Table 7-1 (Page 162) for >15 minutes, unless assessment within this time period confirms that the criterion is not exceeded. OR 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.
Basis	Calculation To calculate the liquid effluent monitor values for Table 7-1, Effluent Radiation Monitor EAL's, the release rates corresponding to the ODCM limit as determined for the Unusual Event are used. For liquid releases, the EAL values are determined by multiplying the Unusual Event liquid monitor reading limit by a factor of 100.

0-RM-90-225	2.59E+07 cpm
1-,2-RM-90-120,-121	1.46E+07 cpm
1-,2-RM-90-122	1.95E+07 cpm
0-RM-90-212	4.10E+05 cpm

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Section 7.0	**RADIOLOGICAL EFFLUENTS**
Event 7.2	**LIQUID EFFLUENTS**
Classification	**ALERT (continued)**
Mode	All
Basis (continued)	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. 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). The significance of the time factor to this Criterion is primarily related to loss of control of radioactive material that has allowed the release to continue unabated for 15 minutes. It is this aspect rather than the magnitude of the release that establishes "... a potential substantial degradation in the level of safety of the plant..." the fundamental definition of an Alert.
Escalation	Not Applicable.
References	NUMARC/NESP-007, AA2, Rev. 2, 1/92 (ODCM) Offsite Dose Calculation Manual 10 CFR 20

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Section 7.0	RADIOLOGICAL EFFLUENTS
Event 7.2	LIQUID EFFLUENTS
Classification	UNUSUAL EVENT
Mode	All
Description	**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):** 1. A valid rad monitor reading exceeds the values under UE in Table 7-1 (Page 162) for >60 minutes, unless assessment within this time period confirms that the criterion is not exceeded. OR 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.
Basis	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. For this calculation, the liquid effluent nuclide mix from the SQN FSAR is used as follows:

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Section 7.0	RADIOLOGICAL EFFLUENTS
Event 7.2	LIQUID EFFLUENTS
Classification	UNUSUAL EVENT (continued)
Mode	All

Basis (Continued)

	Annual Release from Liquid Effluents (Ci/y) (SQN FSAR T 11.2.2-2)	Isotopic Mix	10 CFR 20 ECL (µCi/ml)	Mix / ECL
Na-24	3.79E-01	9.27E-05	5.0E-05	1.85E+00
Cr-51	1.16E-01	2.84E-05	5.0E-04	5.67E-02
Mn-54	1.12E-01	2.74E-05	3.0E-05	9.13E-01
Fe-59	1.15E-01	2.81E-05	1.0E-05	2.81E+00
Co-58	2.91E+00	7.12E-04	2.0E-05	3.56E+01
Co-60	1.13E-01	2.76E-05	3.0E-06	9.21E+00
Zn-65	4.74E-03	1.16E-06	5.0E-06	2.32E-01
Br-83	2.06E-03	5.04E-07	9.0E-04	5.60E-04
Br-84	1.72E-02	4.21E-06	4.0E-04	1.05E-02
Br-85	3.39E-06	8.29E-10	n/a	n/a
Rb-86	9.36E-03	2.29E-06	7.0E-06	3.27E-01
Rb-88	7.93E-01	1.94E-04	4.0E-04	4.85E-01
Sr-89	4.20E-02	1.03E-05	8.0E-06	1.28E+00
Sr-90	1.36E-03	3.33E-07	5.0E-07	6.65E-01
Y-90	1.65E-03	4.04E-07	7.0E-06	5.77E-02
Sr-91	1.47E-02	3.60E-06	2.0E-05	1.80E-01
Y-91	9.38E-03	2.29E-06	8.0E-06	2.87E-01
Y-91m	2.43E+00	5.94E-04	2.0E-03	2.97E-01
Y-93	3.33E-02	8.15E-06	2.0E-05	4.07E-01
Zr-95	1.36E-02	3.33E-06	2.0E-05	1.66E-01
Nb-95	1.42E-02	3.47E-06	3.0E-05	1.16E-01
Mo-99	1.78E+01	4.35E-03	2.0E-05	2.18E+02
Tc-99m	1.89E+01	4.62E-03	1.0E-03	4.62E+00
Ru-103	7.51E-02	1.84E-05	3.0E-05	6.12E-01
Rh-103m	7.48E-02	1.83E-05	6.0E-03	3.05E-03
Ru-106	8.38E-01	2.05E-04	1.0E-08	2.05E+04
Rh-106	8.34E-01	2.04E-04	n/a	n/a
Ag-110m	1.20E-02	2.94E-06	6.0E-06	4.89E-01
Te-125m	3.91E-03	9.56E-07	2.0E-05	4.78E-02
Te-127m	3.44E-02	8.41E-06	9.0E-06	9.35E-01
Te-127	4.09E-02	1.00E-05	1.0E-04	1.00E-01
Te-129m	1.66E-01	4.06E-05	7.0E-06	5.80E+00
Te-129	2.76E-01	6.75E-05	4.0E-04	1.69E-01
Te-131	3.00E-02	7.34E-06	8.0E-05	9.17E-02
Te-131m	7.07E-02	1.73E-05	8.0E-06	2.16E+00
Te-132	1.22E+00	2.98E-04	9.0E-06	3.32E+01

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Section 7.0	RADIOLOGICAL EFFLUENTS
Event 7.2	LIQUID EFFLUENTS
Classification	UNUSUAL EVENT (continued)
Mode	All

Basis (Continued)

	Annual Release from Liquid Effluents (Ci/y) (SQN FSAR T 11.2.2-2)	Isotopic Mix	10 CFR 20 ECL (µCi/ml)	Mix / ECL
I-130	2.90E-03	7.09E-07	2.0E-05	3.55E-02
I-131	2.42E+00	5.92E-04	1.0E-06	5.92E+02
I-132	6.37E-01	1.56E-04	1.0E-04	1.56E+00
I-133	1.30E+00	3.18E-04	7.0E-06	4.54E+01
I-134	4.49E-01	1.10E-04	4.0E-04	2.75E-01
I-135	1.04E+00	2.54E-04	3.0E-05	8.48E+00
Cs-134	8.34E-01	2.04E-04	9.0E-07	2.27E+02
Cs-135	2.85E-01	6.97E-05	1.0E-05	6.97E+00
Cs-137	3.30E+00	8.08E-04	1.0E-06	8.08E+02
Ba-140	1.39E-01	3.40E-05	8.0E-06	4.25E+00
La-140	2.39E-01	5.85E-05	9.0E-06	6.50E+00
Ce-141	9.43E-03	2.31E-06	3.0E-05	7.69E-02
Ce-143	2.50E-02	6.11E-06	2.0E-05	3.06E-01
Pr-143	2.90E-02	7.09E-06	2.0E-05	3.55E-01
Ce-144	5.05E-02	1.24E-05	3.0E-06	4.12E+00
Pr-144	4.07E-02	9.96E-06	6.0E-04	1.66E-02
W-187	2.12E-02	5.19E-06	1.0E-08	5.19E+02
Np-239	6.42E-02	1.57E-05	2.0E-05	7.85E-01
H-3	4.03E+03	9.86E-01	1.0E-03	9.86E+02
Total	**4.09E+03**	**1.00E+00**		**2.40E+04**

Total Site ECL (µCi/ml) = $1/\Sigma_i(\mathrm{Mix}_i/\mathrm{ECL}_i)$ = 1/2.40E+04 = 4.17E-05

The NOUE value is twice the ODCM limit:

4.17E-05 * 10 * 2 = 8.33E-04 µCi/ml.

The site NOUE liquid release EAL is equal to:

8.33E-04 µCi/ml

The total site ECL is then used to determine monitor readings corresponding to this EAL. A minimum cooling tower blowdown of 15,000 gpm, and the maximum undiluted effluent flow is assumed for conservatism. The monitor response for each release point monitor is determined for each nuclide. The calculation assumes the unplanned release is the only release in progress; any combination of simultaneous releases will require case by case assessment.

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Section 7.0	RADIOLOGICAL EFFLUENTS
Event 7.2	LIQUID EFFLUENTS
Classification	UNUSUAL EVENT (continued)
Mode	All
Basis (continued)	

Condenser Demineralizer Radiation Monitor 0-RM-90-225

Using the relationship defined in equations 6.1 and 6.2 from Section 6.1.2 of the SQN ODCM, the monitor response can be predicted.

$$f \times R / F \leq 10$$

Where, f is the effluent flow rate (180 gpm)
F is the dilution flow rate (15000 gpm)

Therefore,

$$R = 15000 \times 10/180$$

In addition, per SQN ODCM equation 6.1:

$$Rj = C_i / ECL_i$$

Where, C_i is the effluent concentration of nuclide (i), in μCi/cc.
ECL_i is the 10 CFR 20 effluent concentration limit, in μCi/cc.

To determine the maximum allowable total concentration (prior to dilution), C, we can substitute R for Rj and the total site ECL for ECL_i as follows:

$$C = (15000 \times 10/180) \times ECL$$

The SQN ODCM equation 6.3 expresses the monitor response as:

$$R = B + \Sigma E_i C_i$$

Since $C_i = f_i C$, then,

$$R = B + \Sigma E_i f_i ECL(15000 \times 10/180)$$

Where, f_i is the nuclide mix fraction for nuclide (i)
E_i is the monitor efficiency nuclide (i), cpm per μCi/cc

The values for $\Sigma E_i f_i ECL$ are computed as follows:

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Section 7.0	RADIOLOGICAL EFFLUENTS
Event 7.2	LIQUID EFFLUENTS
Classification	UNUSUAL EVENT (continued)
Mode	All
Basis (continued)	

		RM-90-120, -121, -212, -225		RM-90-122	
	FSAR Mix (f_i)	Monitor Efficiency E_i (cpm per uCi/cc) SQN TI-18	Monitor Response f_i x ECL x E_i (cpm)	Monitor Efficiency (cpm per uCi/cc) SQN TI-18	Monitor Response f_i x ECL x E_i (cpm)
Na-24	9.27E-05	5.20E+08	2.01E+00	1.17E+08	4.51E-01
Cr-51	2.84E-05	4.08E+07	4.82E-02	2.02E+07	2.39E-02
Mn-54	2.74E-05	3.80E+08	4.33E-01	1.28E+08	1.46E-01
Fe-59	2.81E-05	3.47E+08	4.06E-01	1.25E+08	1.46E-01
Co-58	7.12E-04	5.06E+08	1.50E+01	1.30E+08	3.85E+00
Co-60	2.76E-05	6.48E+08	7.45E-01	2.36E+08	2.71E-01
Zn-65	1.16E-06	1.88E+08	9.07E-03	6.60E+07	3.18E-03
Rb-86	2.29E-06	3.17E+07	3.02E-03	1.05E+07	1.00E-03
Sr-91	3.60E-06	2.74E+08	4.10E-02	9.05E+07	1.35E-02
Y-91	2.29E-06	1.02E+06	9.74E-05	3.57E+05	3.41E-05
Y-91m	5.94E-04	3.99E+08	9.87E+00	1.59E+08	3.93E+00
Y-93	8.15E-06	4.06E+07	1.38E-02	1.78E+07	6.03E-03
Zr-95	3.33E-06	3.89E+08	5.39E-02	1.34E+08	1.86E-02
Nb-95	3.47E-06	3.89E+08	5.62E-02	1.32E+08	1.91E-02
Mo-99	4.35E-03	1.19E+08	2.16E+01	1.64E+08	2.97E+01
Tc-99m	4.62E-03	3.09E+08	5.95E+01	1.26E+08	2.42E+01
Ru-103	1.84E-05	3.80E+08	2.91E-01	1.60E+08	1.22E-01
Te-129m	4.06E-05	1.35E+07	2.28E-02	4.59E+06	7.76E-03
Te-132	2.98E-04	3.78E+08	4.69E+00	1.72E+08	2.14E+00
I-131	5.92E-04	4.02E+08	9.90E+00	1.94E+08	4.78E+00
I-133	3.18E-04	4.06E+08	5.37E+00	1.21E+08	1.60E+00
I-135	2.54E-04	3.99E+08	4.22E+00	1.52E+08	1.61E+00
Cs-134	2.04E-04	8.79E+08	7.46E+00	3.22E+08	2.73E+00
Cs-137	8.08E-04	3.41E+08	1.15E+01	1.23E+08	4.14E+00
Ba-140	3.40E-05	1.34E+08	1.90E-01	7.31E+07	1.03E-01
La-140	5.85E-05	7.05E+08	1.72E+00	2.95E+08	7.18E-01
Ce-141	2.31E-06	1.79E+08	1.72E-02	7.55E+07	7.25E-03
Ce-144	1.24E-05	3.65E+07	1.88E-02	1.79E+07	9.20E-03
		Total	1.55E+02		8.08E+01

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Section 7.0	RADIOLOGICAL EFFLUENTS
Event 7.2	LIQUID EFFLUENTS
Classification	UNUSUAL EVENT (continued)
Mode	All

Basis (continued)

For a typical monitor background of 500 cpm, and a monitor response as calculated above gives the predicted monitor response:

R = 500 cpm + ((15000 x 10/180) x 155 cpm)
R = 1.30E+05 cpm

The NOUE value is 2 times the ODCM limit, so:

NOUE = 2 * 1.30E+05 = 2.59E+05 cpm

<u>Steam Generator Blowdown Radiation Monitors</u> **(1-,2-RM-90 120,-121)**

The same methodology is followed for these monitors as is given in the previous description, using a maximum undiluted effluent flow rate of 320 gpm. Thus the monitor response will be:

R = 500 cpm + ((15000 x 10/320) x 155 cpm)
R = 7.32E+04 cpm

The NOUE value is 2 * 7.32E+04 or 1.46E+05 cpm.

<u>Radwaste Radiation Monitor</u> **(1-,2-RM-90-122)**

The same methodology is followed for these monitors as is given in the previous description, using a maximum undiluted effluent flow rate of 125 gpm. Thus the monitor response will be:

R = 500 cpm + ((15000 x 10/125) x 80.8 cpm)
R = 9.75E+04 cpm

The NOUE value is 2 * 9.75E+04 cpm or 1.95E+05 cpm.

<u>Turbine Building Sump Radiation Monitor</u> **(0-RM-90-212)**

Since the flow from this release point is not diluted prior to being released, the undiluted I-131 concentration is inserted into ODCM equation 6.3:

R = 500 cpm + (10 x 155 cpm)
R = 2.05E+03 cpm

The NOUE value is then:

2 * 2.05E+03 cpm or 4.10E+03 cpm.

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Section 7.0	**RADIOLOGICAL EFFLUENTS**
Event 7.2	**LIQUID EFFLUENTS**
Classification	**UNUSUAL EVENT** (continued)
Mode	All
Basis (continued)	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). 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. The significance of the time factor to this criterion is primarily related to loss of control of radioactive material that has allowed the release to continue unabated for 60 minutes. It is this aspect rather than the magnitude of the release that establishes " . . . a potential degradation in the level of safety of the plant..." the fundamental definition of an unusual event.
Escalation	Escalation will be based on an unplanned release exceeding 200 times the ODCM limit for greater than 15 minutes.
References	NUMARC/NESP-007, AU2, Rev. 2, 1/92 (ODCM) Offsite Dose Calculation Manual 10 CFR 20

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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
Total Site (Gas)	*µCi/s*	*4.90E+05*	*4.90E+07*	*1.31E+08*	*1.31E+09*
Shield Building 1, 2-RI-90-400 (EFF Level)	µCi/s	4.90E+05	4.90E+07	1.31E+08	1.31E+09
Auxiliary Building 0-RM-90-101B	cpm	1.03E+05	Offscale(1)	Offscale(1)	Offscale(1)
Service Building 0-RM-90-132B	cpm	2.62E+06	Offscale(1)	Offscale(1)	Offscale(1)
Steam Generator Discharge(2) *(Main Steam Line Monitors)* 1,2-RI-90-421 thru -424	µCi/cc	1.49E-01	1.49E+01	3.98E+01	3.98E+02
Condenser Vacuum Exhaust 1, 2-RM-90-255, -256	mR/h	4.10E+02	4.10E+04	1.09E+05	1.09E+06
Total Site (Liquid)	*µCi/ml*	*8.33E-04*	*8.33E-02*	*N/A*	*N/A*
Radwaste Monitor RM-90-122	cpm	1.95E+05	Offscale(1)	N/A	N/A
SGBD RM-90-120,121	cpm	1.46E+05	Offscale(1)	N/A	N/A
Condensate Demin RM-90-225	cpm	2.59E+05	Offscale(1)	N/A	N/A
Turbine Bldg Sump RM-90-212	cpm	4.10E+03	4.10E+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 1085 psig 600 degrees F. 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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Section 7.0	RADIOLOGICAL EFFLUENTS
Event 7.3	RADIATION LEVELS
Classification	GENERAL EMERGENCY
Mode	Not Applicable.
Description	Refer to the "Fission Product Barrier Matrix" (Section 1) or "Gaseous Effluents"(Section 7.1).
Basis	Not Applicable.
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 7.0	RADIOLOGICAL EFFLUENTS
Event 7.3	RADIATION LEVELS
Classification	SITE AREA EMERGENCY
Mode	Not Applicable.
Description	Refer to the "Fission Product Barrier Matrix" (Section 1) or "Gaseous Effluents"(Section 7.1).
Basis	Not Applicable.
Escalation	Escalation may be based on "Fission Product Barrier Matrix" (Section 1) or "Gaseous Effluent Levels" (Section 7).
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 7.0	RADIOLOGICAL EFFLUENTS
Event 7.3	RADIATION LEVELS
Classification	ALERT
Mode	All
Description	**UNPLANNED increases in radiation levels within the facility that impedes safe operations or establishment or maintenance of cold shutdown (1 or 2):** 1. Valid area radiation monitor readings or survey results exceed 15 mrem/hr *in the control room or SAS. <u>OR</u> 2. (a and b) a. VALID area radiation monitor readings exceed values listed in Table 7-2 (Page 167). b. Access restrictions impede operation of systems necessary for safe operation or the ability to establish cold shutdown. 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).
Basis	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. 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. The control room, and the central alarm station (CAS) should be continuously manned. Thus, the 15 mrem/hr value applies to these facilities. 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.

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Section 7.0	RADIOLOGICAL EFFLUENTS
Event 7.3	RADIATION LEVELS
Classification	ALERT (continued)
Mode	All
Basis (continued)	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.). 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.
Escalation	Escalation will be based on "Fission Product Barrier Matrix" (Section 1) or "Gaseous Effluent" (Section 7.1).
References	NUMARC/NESP-007, AA3, Rev. 2, 1/92 (ODCM) Offsite Dose Calculation Manual

Section 7.0	RADIOLOGICAL EFFLUENTS
Event 7.3	RADIATION LEVELS
Classification	UNUSUAL EVENT
Mode	All
Description	**UNPLANNED increases in radiation levels within the facility:** 1. Valid area radiation monitor readings increase by a factor of 1000 mrem/hr over the highest reading in the past 24 hours excluding the current peak values. 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).
Basis	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. 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.
Escalation	Escalation will be based on the inability to access certain operating stations or equipment needed to establish or maintain cold shutdown.
References	NUMARC/NESP-007, AU2, Rev. 2, 1/92

Section 7.0	RADIOLOGICAL EFFLUENTS
Event 7.4	FUEL HANDLING
Classification	GENERAL EMERGENCY
Mode	Not Applicable.
Description	Refer to "Gaseous Effluents" (Section 7.1).
Basis	The basis for a General Emergency is primarily the extent and severity of "Gaseous Effluents" (Section 7.1).
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 7.0	RADIOLOGICAL EFFLUENTS
Event 7.4	FUEL HANDLING
Classification	SITE AREA EMERGENCY
Mode	Not Applicable.
Description	"Refer to Gaseous" Effluents (Section 7.1).
Basis	The basis for a Site Area Emergency is primarily the extent and severity of "Gaseous Effluents" (Section 7.1).
Escalation	Not Applicable.
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 7.0	RADIOLOGICAL EFFLUENTS
Event 7.4	FUEL HANDLING
Classification	ALERT
Mode	All
Description	**Major damage to irradiated fuel or loss of water level that has or will uncover irradiated fuel outside the reactor vessel (1 and 2):** 1. Valid alarm on RM-90-101 or RM-90-102 or RM-90-103 or RM-90/130/131 or RM-90-112. 2. (a or b) a. Plant personnel report damage to irradiated fuel sufficient to rupture fuel rods. <u>OR</u> 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.
Basis	The major concern of the EAL is a fuel handling accident or loss of water covering spe[illegible] 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. 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. 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.
Escalation	Escalation will occur by offsite dose rates. Refer to "Gaseous Effluents" (Section 7.1).
References	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

Section 7.0	**RADIOLOGICAL EFFLUENTS**
Event 7.4	**FUEL HANDLING**
Classification	**UNUSUAL EVENT**
Mode	All
Description	**UNPLANNED loss of water level in spent fuel pool or reactor cavity or transfer canal with fuel remaining covered (1 and 2 and 3):** 1. Plant personnel report water level drop in spent fuel pool or reactor cavity or transfer canal. 2. Valid alarm on RM-90-101 or RM-90-102 or RM-90-103. 3. Fuel remains covered with water.
Basis	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. Unplanned is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities. 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 exclusion area boundary, thus the impact to public health and safety is very low. Classification of an Unusual Event is warranted as a precursor to a more serious event. 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.
Escalation	Escalation will be based on uncovering an irradiated fuel assembly or indications of high radiation levels on the refueling floor.
References	NUMARC/NESP-007,AU2, Rev. 2, 1/92 AOP M.04 — Refueling Malfunctions NRC IEN No. 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

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 El. 734.0	1.5E + 03 mrem/hr
0-RM-90-3	Waste Packaging ARM El. 706.0	1.5E + 03 mrem/hr
0-RM-90-4	Decontamination Room ARM El. 690.0	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.

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 RadCon technician, as necessary.

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 Site 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-EPIPs.

B.5.1.8 Radiological Control Manager

1. Directs and/or performs assessment of inplant and onsite radiological conditions.
2. Directs onsite RadCon activities.
3. Coordinates additional RadCon 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 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.

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 Assistant OSC 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 RADCON Supervisor

1. Directs activities of the RadCon lab.
2. Ensure RadCon 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.

*Revision

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' RadCon Lab. Protective clothing, flashlights, other equipment and tools are available, as needed.

*Revision

B.6.3 RadCon Laboratory and Equipment

The RadCon laboratory is located in the service building, elevation 690'. The portable radiation monitoring and counting equipment normally used by the plant RadCon 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 RADCON
* procedures.

B.6.4 Onsite Monitoring Systems and Equipment

B.6.4.1 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.6.4.2 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.

B.6.4.2.1 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

*Revision

B.6.4.2.2 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.6.4.2.3 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.6.4.2.4 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.

B.6.4.3 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.

*Revision

B.6.4.4 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.

* Thermoluminescent dosimeters (TLDs) are placed at approximately 40 sites around the plant. These TLDs 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.6.5 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.

B.6.6 First Aid and Medical Facilities

B.6.6.1 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 RadCon lab. Equipment and materials for decontamination and first aid are available.

*Revision

B.6.6.2 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 RadCon. Specific information including authorization and dispersal of tablets is contained in the site EPIPs.

B.6.6.3 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.6.6.4 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.6.7 Additional Local Support

B.6.7.1 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.6.7.2 Law Enforcement

Agreements are maintained with local law enforcement agencies to support TVA when necessary. (See REP Section 16.5)

*Revision

B.6.8 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.6.9 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 all clear signal is a steady three-minute siren.

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.6.10 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)

B.7 SQN EMERGENCY PLAN IMPLEMENTING PROCEDURES

The following is a listing of the SQN-EPIPs:

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.

*Revision

B.7.6 SQN 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 SQN 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 SQN EPIP-8 Personnel Accountablity 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 SQN EPIP-9 Accountability and Evacuation of the Sequoyah Training Center

This procedure has been cancelled.

B.7.10 SQN-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 SQN EPIP-11 Security and Access Control

This procedure details responsibilities and requirements for access control and accountability during a radiological emergency.

B.7.12 SQN EPIP-12

This procedure has not been issued.

B.7.13 SQN EPIP-13 Call Lists

This procedure has not been issued.

B.7.14 SQN EQIP-14 Radiological Control Response

This procedure outlines the actions to be followed by health physics personnel during a plant emergency. It details responsibilities, RadCon 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 Procedure

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 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.7.18 EPIP-18

This procedure has not been issued.

B.7.19 EPIP-19 Radiological Emergency Preparedness Training and Drills

This procedure specifies the training provided to plant personnel who are required to respond and have specific duties as outlined in the NP-REP and describes the required EP drills or exercises.

B.8 PROMPT NOTIFICATION SYSTEM

The prompt notification system network consists of fixed sirens and tone-alert radios. The system is designed to provide warning within 15 minutes, to the population, within 10 miles of the plant.

B.8.1 Fixed Sirens

The fixed-siren component consists of electromechanical sirens. The sirens are activated by the Tennessee Emergency Management Agency (TEMA). A backup activation system is located in Hamilton County.

The siren system is activated on a monthly basis by TEMA 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 the silent tests. A growl
* test is conducted annually.

Preventive maintenance is performed by TVA on an annual basis commensurate with the manufacturer's recommendations. Unscheduled maintenance is performed on an as-needed basis.

B.8.2 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 an annual basis commensurate with the manufacturer's recommendations. Unscheduled maintenance is performed on as-needed basis.

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.

*Revision

Figure B-1
SITE EMERGENCY ORGANIZATION
(Including Minimum Staffing and Augmentation) (See Note 1)

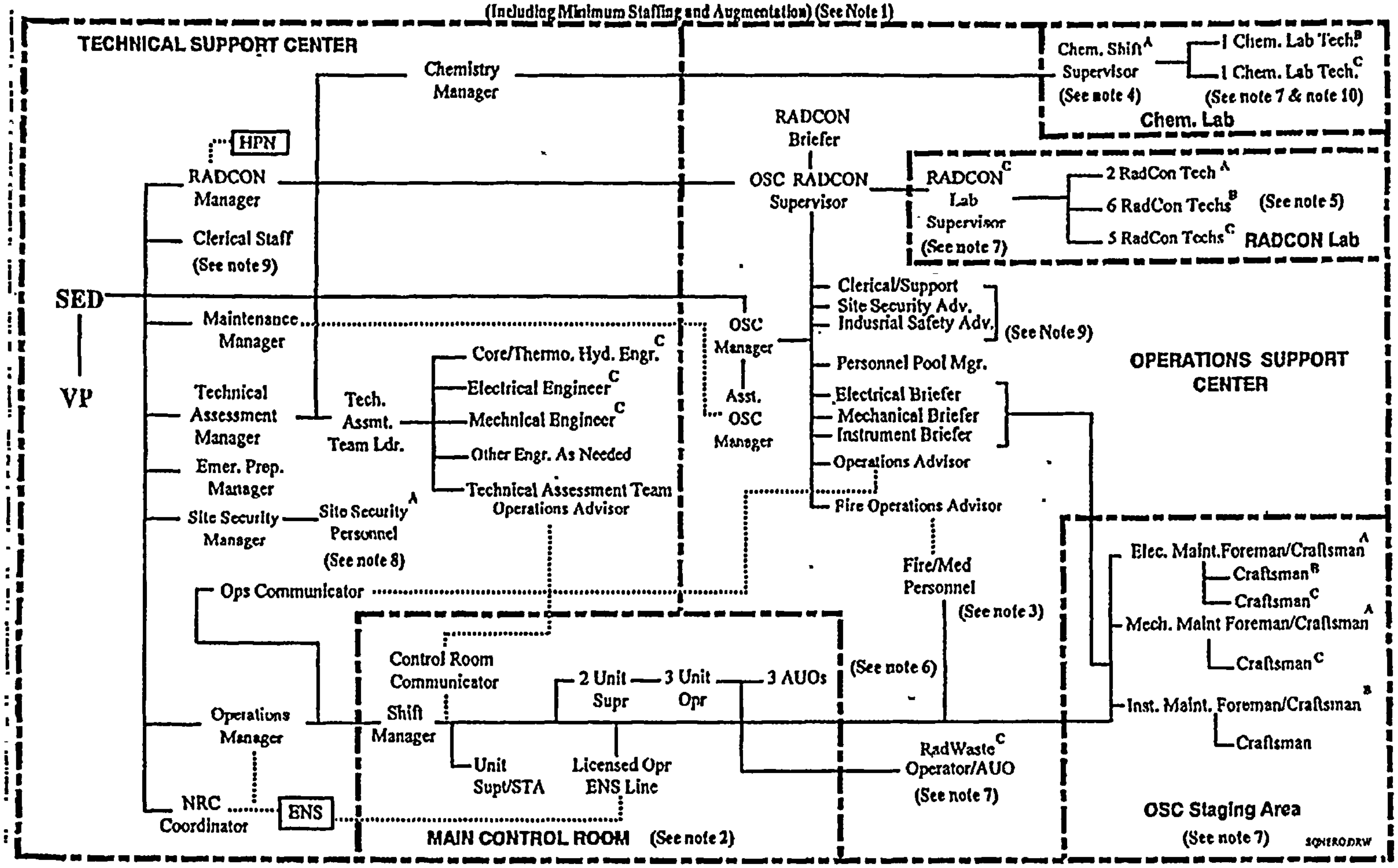

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Figure B-1 (Continued)

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.
  Minimum shift crew composition per Tech Specs Table 6.2-1 requires 1 Shift Manager, 1 SRO (Unit Supervisor), 3 ROs (Unit Operators), 1 STA, and 3 AUOS (assuming both units in mode 4 or above). NUREG 0654 Table B-1 requires as a minimum, 1 Shift Manager, 1 Unit Supervisor, 2 Unit Operators, 1 STA, and 2 AUOs. Additionlly, Table B-1 requires that each unaffected nuclear unit in operation maintain at least 1 Unit Supervisor, 1 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 Brigrade personnel on-shift will be in accordance with the Fire Protection Report. This group also provides medical and rescue funtions.

Note 4 - Offsite Dose Assessment task as shown in NUREG 0654 Table B-1 is provided by the Chemistry Shift Supervisor.

Note 5 - Two RADCON 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 RADCON Lab Supervisor may fill one of the six 60 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 - Onshift security personnel per the Security Plan.

Note 9 - Call in as needed.

Note 10 - May be filled by the Chemistry Lab. Supervisor (A position) so long as Dose Assessment functions have been assumed by other ERO members.

Note 11 - Chemistry Shift/RADCONLab supervisors, if holding proper qualifications, may fill the position of technician.

.... Shows communication networks.

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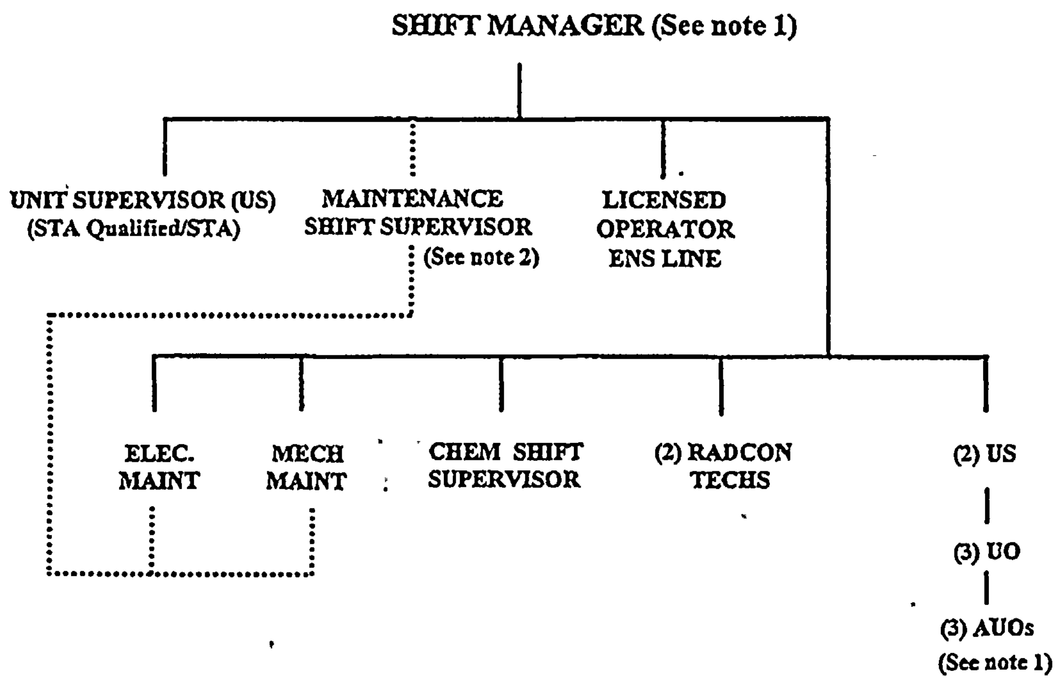
FIGURE B-2
MINIMUM ONSHIFT RESPONSE PERSONNEL



Note 1: See Note 2 on Figure B-1.

Note 2: The Maintenance Shift Supervisor provides the Shift Manager a single point of contact for maintenance groups.

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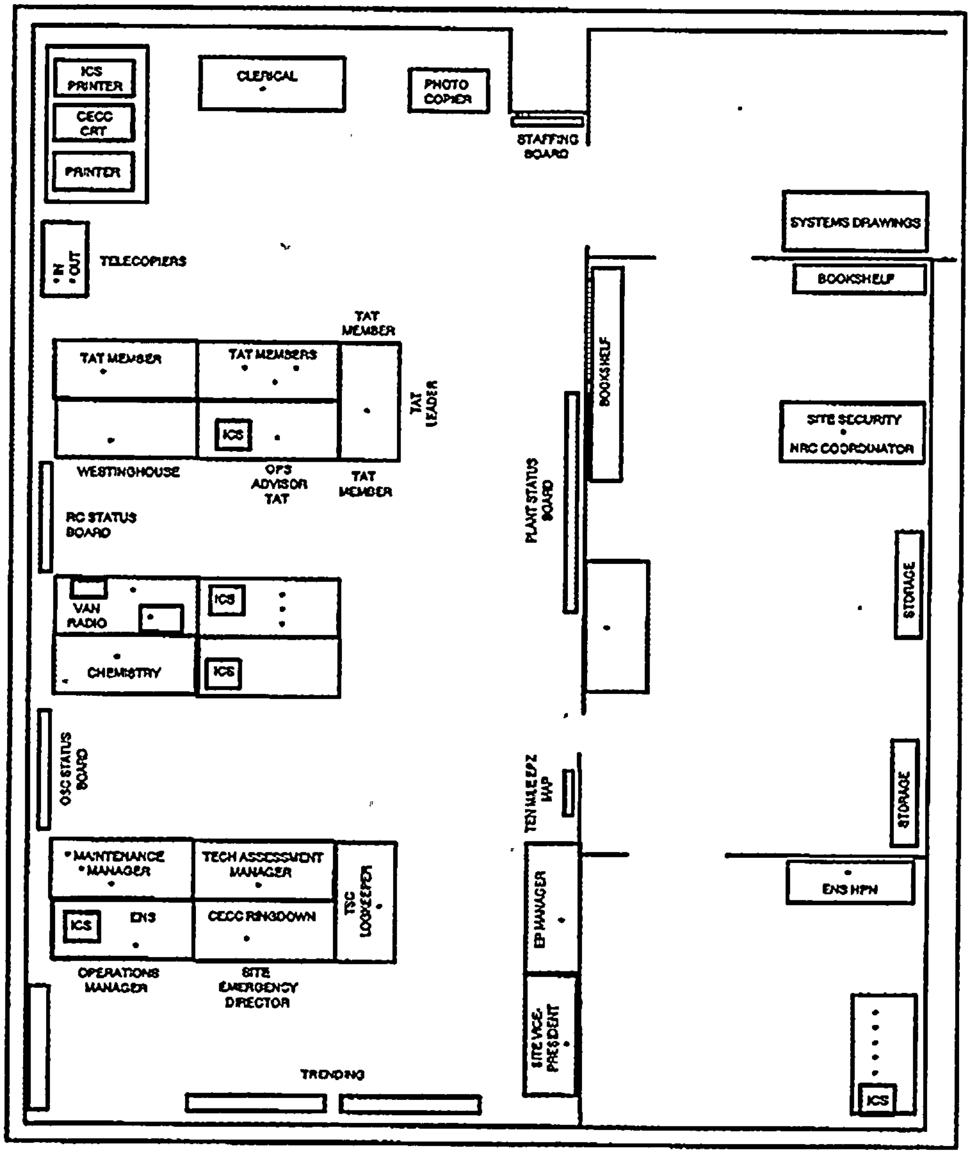
Figure B-3

TECHNICAL SUPPORT CENTER
CONTROL BUILDING
ELEVATION 732'

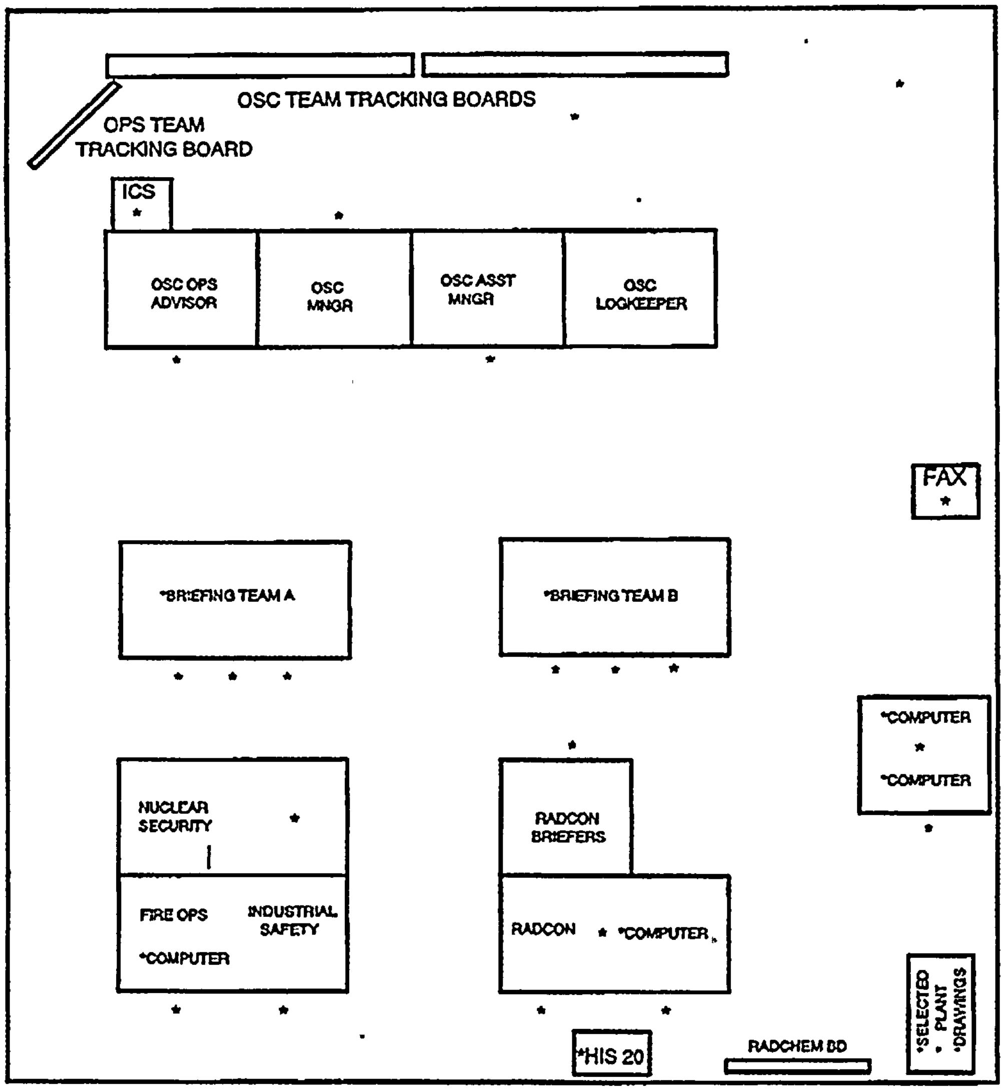


*Note : This is a general layout of the TSC.

* Revision

Figure B-4

OPERATIONS SUPPORT CENTER PLANT OFFICE BUILDING ELEVATION 706'



* Note: This is a general layout of the OSC

OSCLYOUT.DRW

* Revision

Figure B-5

EMERGENCY EQUIPMENT

Location	Description
1. RadCon 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 (RadCon, 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

*Revision

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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 progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA Protection Action Guideline exposure levels offsite for more than the immediate site area.

For a Site Area Emergency to be declared, events should be in progress or have occurred that involve actual or likely major failures of plant functions needed for protection of the public. Any releases are not expected to exceed EPA protective action guides.

For an Alert to be declared, an event should be in process or have occurred that involves an actual or potential substantial degradation of the plant. Releases of radioactive material are expected to be limited to small fractions of the EPA protective action guidelines.

For an Unusual Event to be declared, unusual events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant.

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.

C.2 Emergency Event Methodology

The WBN methodology for event classification and declaration has 37 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

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, HS1, HG1
4.7 SED Judgement	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
6.4 Fuel Handling	AU2, AA2

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.

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.

The acceptable time frame for notification to the Operations Duty Specialist (ODS) who makes notifications to offsite officials, is considered to be (5) five minutes. This is the time period between declaration of the emergency and notifying the ODS.

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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Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
Event 1.1	**FUEL CLAD BARRIER**
IC 1.1.1	**Critical Safety Function Status**
Mode	1,2,3,4
Description	<u>LOSS:</u> Core Cooling Red <u>Potential LOSS</u> Core Cooling Orange <u>OR</u> Heat Sink Red (RHR <u>Not</u> in Service)
Basis	<u>LOSS;</u> The "Loss" IC addresses the condition of inadequate Core Cooling. 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. Core Cooling - Red indicates significant superheating and core uncovery and is considered to indicate a "Loss" of the Fuel Clad Barrier. <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. 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. 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. Either of these two items indicate a "Potential Loss" of the Fuel Clad Barrier.
Escalation	Not Applicable
References	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

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
Event 1.1	**FUEL CLAD BARRIER**
IC 1.1.2	**Primary Coolant Activity Level**
Mode	1,2,3,4
Description	**LOSS:** RCS sample activity is Greater Than 300 μCi/gm dose equivalent Iodine - 131 **Potential LOSS** Not Applicable
Basis	**LOSS;** The "Loss" IC addresses the Condition of high RCS activity. If the reading of RCS activity is ≥ 300 μCi/gm it is well above expected iodine spikes and corresponds to about 2% to 5% fuel clad damage. This amount of clad damage indicates that significant clad heating has occurred. **Potential LOSS:** There is no "Potential Loss" IC associated with this item.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101

Section *1.0*	**FISSION PRODUCT BARRIER MATRIX**
Event *1.1*	**FUEL CLAD BARRIER**
IC *1.1.3*	**Incore Tcs Hi Quad Average**
Mode	1,2,3,4
Description	<u>**LOSS:**</u> Greater Than 1200°F <u>**Potential LOSS**</u> Greater Than 727°F
Basis	<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.) <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.
Escalation	Not Applicable
References	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)

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
Event 1.1	**FUEL CLAD BARRIER**
IC 1.1.4	**Reactor Vessel Water Level**
Mode	1,2,3,4
Description	<u>LOSS:</u> Not Applicable <u>Potential LOSS</u> VALID RVLIS Level <33% (No RCP running)
Basis	<u>LOSS;</u> There is no "Loss" IC corresponding to this item because it is covered by the other Fuel Clad Barrier "Loss". <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.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101 FR-C.2 Degraded Core Cooling WBN-OSG4-188, (K-01)

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	**LOSS:** VALID reading increase of Greater Than: * 74 R/hr On RM-90-271 and 272 or * 59 R/hr On RM-90-273 and 274 **Potential LOSS** Not Applicable
Basis	**LOSS;** * * The "Loss" IC is defined by a VALID reading of 74 R/hr on the upper Containment Hi Rad monitors or 59 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. 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 2 - 5% clad failure depending on core inventory and RCS volume). This value is higher than that specified for FCB barrier loss #5. Thus, this IC indicates a loss of both the fuel clad barrier and a loss of the RCS barrier. **Potential LOSS:** There is no "Potential Loss" IC associated with this item.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101 WBN, Q DCN 20764B - Radiation Monitor Readings for REP Response of the *Primary Containment High Range Radiation Monitors, TI-RPS-162 R6.

*Revision

Section *1.0*	**FISSION PRODUCT BARRIER MATRIX**
Event *1.1*	**FUEL CLAD BARRIER**
IC *1.1.6*	**Site Emergency Director Judgement**
Mode	1,2,3,4
Description	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.
Basis	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. 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.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
Event 1.2	**RCS BARRIER**
IC 1.2.1	**Critical Safety Function Status**
Mode	1,2,3,4
Description	**LOSS:** Not Applicable **Potential LOSS** Pressurized Thermal Shock Red OR Heat Sink Red (RHR Not in Service)
Basis	**LOSS:** There is no "Loss" IC associated with this item. **Potential LOSS:** 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). 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 breech 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.
Escalation	Not Applicable
References	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

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
Event 1.2	**RCS BARRIER**
IC 1.2.2	**RCS Leakage/LOCA**
Mode	1,2,3,4
Description	<u>LOSS:</u> RCS Leak results in Loss of subcooling (<65°F Indicated), [85°F ADV] <u>Potential LOSS</u> Non Isolatable RCS Leak Exceeding The Capacity Of <u>One</u> Charging Pump in the Normal Charging Alignment <u>OR</u> RCS Leakage Results in Entry into E-1
Basis	<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. <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.
Escalation	Not Applicable
References	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)

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
Event 1.2	**RCS BARRIER**
IC 1.2.3	**Steam Generator Tube Rupture**
Mode	1,2,3,4
Description	**LOSS:** SGTR that results in a safety injection actuation OR Entry into E-3 **Potential LOSS** Not Applicable
Basis	**LOSS;** 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. 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. This IC also addresses the entry into EOP, E-3, Steam Generator Tube Rupture, under any circumstance. 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. **Potential LOSS:** There is no "Potential Loss" IC associated with this item.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101 AOI-33 Steam Generator Tube Leak E-3 Steam Generator Tube Rupture

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
Event 1.2	**RCS BARRIER**
IC 1.2.4	Reactor Vessel Water Level
Mode	1,2,3,4
Description	<u>LOSS:</u> VALID RVLIS level < 33% (No RCP Running) <u>Potential LOSS</u> Not Applicable
Basis	<u>LOSS;</u> The "Loss" IC is defined by an Orange path on the Core cooling status tree (CSF). The numeric value used is 40% 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. This "Loss" EAL in conjunction with the Fuel Clad Barrier "Potential Loss" IC #4 requires an event classification of Site Area Emergency. <u>Potential LOSS:</u> There is no "Potential Loss" IC associated with this item.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101 FR-C.2 Degraded Core Cooling WBN-OSG4-188, (K-01)

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
Event 1.2	**RCS BARRIER**
IC 1.2.5	**Site Emergency Director Judgement**
Mode	1,2,3,4
Description	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.
Basis	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. 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 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.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101

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Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
Event 1.3	**CNTMT BARRIER**
IC 1.3.1	**Critical Safety Function Status**
Mode	1,2,3,4
Description	<u>LOSS:</u> Not Applicable <u>POTENTIAL LOSS:</u> Containment (FR-Z.1) <u>Red</u> <u>OR</u> Actions of FR-C.1 (Red Path) are **INEFFECTIVE**
Basis	<u>LOSS:</u> There is no "Loss" IC associated with this item. <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. 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. 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.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101 FR-Z.1 High Containment Pressure FR-C.1 Inadequate Core Cooling

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
Event 1.3	**CNTMT BARRIER**
IC 1.3.2	**Containment Pressure/Hydrogen**
Mode	1,2,3,4
Description	**LOSS:** Rapid unexplained pressure decrease following initial pressure increase OR Containment pressure or Sump Level Not increasing (with LOCA in progress) **Potential LOSS** Containment Hydrogen increases to >4% by volume OR Pressure >2.8 PSIG (Phase B) with < One full train of Containment spray
Basis	**LOSS;** 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. 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. **Potential LOSS:** The condition of high containment pressure, greater than 13.5 PSIG, is addressed by the CSF, Containment Red, "Potential Loss", IC #1.3.1 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. 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, ∅B (2.8 PSIG), at which the equipment was supposed to have actuated. These "Potential Loss" ICs are primarily a discrimination between the Site Area Emergency and General Emergency representing a potential loss of the third barrier.

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
Event 1.3	**CNTMT BARRIER**
IC 1.3.2	**Containment Pressure/Hydrogen** (continued)
Mode	1,2,3,4
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101 FR-Z.1 High Containment Pressure

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
Event 1.3	**CNTMT BARRIER**
IC 1.3.3	**Containment Isolation Status**
Mode	1,2,3,4
Description	<u>LOSS:</u> Containment Isolation is Incomplete <u>and</u> a Release Path to the Environment Exists <u>Potential LOSS</u> Not Applicable
Basis	<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. <u>Potential LOSS:</u> There is no "Potential Loss" IC associated with this item.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
Event 1.3	**CNTMT BARRIER**
IC 1.3.4	**Containment Bypass**
Mode	1,2,3,4
Description	**LOSS:** RUPTURED S/G is also FAULTED outside of CNTMT OR Prolonged (> 4 Hours) Secondary Side release outside CNTMT from a S/G with a SGTL > T/S Limits **POTENTIAL LOSS:** Unexplained VALID increase in area or Ventilation RAD monitors in areas adjacent to CNTMT (with LOCA in progress)
Basis	**LOSS:** 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. 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. 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,

Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
Event 1.3	**CNTMT BARRIER**
IC 1.3.4	**Containment Bypass (continued)**
Mode	1,2,3,4
Basis (continued)	<u>LOSS:</u> (continued) 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. RUPTURED: (Steam Generator) Existence of primary to secondary leakage of a magnitude greater than charging pump capacity. 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. <u>Potential LOSS:</u> 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.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101 E-2 Faulted Steam Generator Isolation

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	**LOSS:** Not Applicable **Potential LOSS** VALID Reading increase of Greater Than: $\underline{3.6 \times 10^2}$ R/hr on RM-90-271 and RM-90-272 or $\underline{2.8 \times 10^2}$ R/hr on RM-90-273 and RM-90-274
Basis	**LOSS:** There is no "Loss: IC associated with this item. **Potential LOSS:** The "Potential Loss" IC is defined by containment radiation readings of $\underline{3.6 \times 10^2}$ R/hr and $\underline{2.8 \times 10^2}$ R/hr. 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%.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101 NUREG-1228, Source Estimates During Incident Response to Severe Nuclear Power Plant Accidents WBN, Q DCN 20764B - Radiation Monitor Readings for REP Response of the Primary Containment High Range Radiation Monitors, TI-RPS-162 R1.

Section *1.0*	**FISSION PRODUCT BARRIER MATRIX**
Event *1.3*	**CNTMT BARRIER**
IC *1.3.6*	Site Emergency Director Judgement
Mode	1,2,3,4
Description	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.
Basis	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. 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.
Escalation	Not Applicable
References	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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Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
Event	**Not Applicable**
Classification	**GENERAL EMERGENCY**
Mode	1,2,3,4
Description	**LOSS** of any two barriers and **Potential LOSS** of third barrier
Basis	Definition: • Events are in process or have occurred which involve Actual or Imminent Substantial Core Degradation or Melting with Potential for Loss of Containment integrity. Releases can be reasonably expected to exceed EPA Plume Protective Action Guidelines Exposure Levels outside the EXCLUSION AREA BOUNDARY. The main differentiation between the Site Area and General Emergency classification is whether or not the EPA PAG plume exposure levels are expected to be exceeded outside the site boundary. This threshold, in addition to dynamic dose assessment considerations, addresses NRC and offsite emergency response agency concerns as to timely declaration of a General Emergency. The main objective of the General Emergency is to determine whether evacuation or sheltering of the general public is indicated based on EPA PAGs, and therefore should be interpreted to include radionuclide release regardless of cause. Consideration must be given to failures of systems and or structures that provide fission product barrier integrity which is the primary method of preventing uncontrolled radionuclide releases. In terms of fission product barriers, the loss of two barriers with potential loss of the third barrier constitutes a General Emergency. 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.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101

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Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
Event	**Not Applicable**
Classification	**SITE AREA EMERGENCY**
Mode	1,2,3,4
Description	**LOSS** or **Potential LOSS** of any two barriers
Basis	Definition: Events are in process or have occurred which involve Actual or Likely Major Failures of Plant Functions needed for the Protection of the Public. Any releases are not expected to result in Exposure Levels which Exceed EPA Plume Protective Action Guideline Exposure Levels outside the Exclusion Area Boundary. It is considered to be a challenge to plant functions necessary for the protection of the public if the integrity of any two of the three fission product barriers has or has the potential of being degraded. This approach is more conservative than 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. This approach also simplifies the Site Area Emergency classification from the fission product barrier matrix. 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.
Escalation	Escalation would be based on Actual or Imminent Substantial Core Degradation
References	NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101

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Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
Event	**Not Applicable**
Classification	**ALERT**
Mode	1,2,3,4
Description	Any **LOSS** or **Potential LOSS** of Fuel Clad Barrier OR Any **LOSS** or **Potential LOSS** of RCS barrier
Basis	Definition: Events are in process or have occurred which involve an Actual or Potential Substantial Degradation of the Level of Safety of the Plant. Any releases are expected to be limited to small fractions of the EPA Plume Protective Action Guideline Exposure Levels. The "Loss" or "Potential Loss" of either the Fuel Clad Barrier or RCS barrier is considered to be an actual or potential substantial degradation of the level of safety of the plant. The Alert classification resulting from potential degradation of the fuel clad or RCS integrity also addresses the operation staff's need for help by staffing the Technical Support Center (TSC), independent of whether an actual decrease in plant safety is determined. This increased monitoring can then be used to better determine the actual plant safety state, whether escalation to a higher emergency class is warranted, or 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). 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.
Escalation	Escalation would be based on Actual or Likely Major Failures of Plant Functions needed to Protect the Public.
References	NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101

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Section 1.0	**FISSION PRODUCT BARRIER MATRIX**
Event	**Not Applicable**
Classification	**UNUSUAL EVENT**
Mode	1,2,3,4
Description	**LOSS** or **Potential LOSS** of Containment Barrier
Basis	Definition: Unusual Events are in process or have occurred which indicate a Potential Degradation of the Level of Safety of the Plant. No releases of Radioactive Material requiring Offsite Responses or Monitoring are expected unless further degradation of Safety Systems occurs. 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). 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. 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.
Escalation	Escalation would be based on Actual or Potential Substantial Degradation of the Level of Safety of the Plant.
References	NUMARC/NESP-007, Rev 2, 1/92 per REG GUIDE 1.101

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Section *2.0*	**SYSTEM DEGRADATION**
Event *2.1*	**LOSS OF INSTRUMENTATION**
Classification	**GENERAL EMERGENCY**
Mode	All
Description	Refer to the "Fission Product Barrier Matrix" and "Radiological Effluents" (Section 7)
Basis	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.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
Event 2.1	**LOSS OF INSTRUMENTATION**
Classification	**SITE AREA EMERGENCY**
Mode	1,2,3,4
Description	**Inability to monitor a SIGNIFICANT TRANSIENT in progress (1 and 2 and 3 and 4)** 1. Loss of most (>75%) of MCR annunciators (and Annunciator Printer) or indications. 2. **SIGNIFICANT TRANSIENT** in progress 3. Loss of Integrated Computer System (ICS) and SPDS 4. Inability to directly monitor any of the following CSFs: Subcriticality PTS Core Cooling Containment Heat Sink Inventory
Basis	This IC is intended to recognize the inability of the control room staff to monitor the plant response to a transient. 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. The loss of annunciators excludes scheduled maintenance and testing activities. 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. 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. 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.

Section *2.0*	**SYSTEM DEGRADATION**
Event *2.1*	**LOSS OF INSTRUMENTATION**
Classification	**SITE AREA EMERGENCY** (continued)
Mode	1,2,3,4
Escalation	Escalation will be based on "Fission Product Barrier Challenges".
References	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

Section 2.0	**SYSTEM DEGRADATION**
Event 2.1	**LOSS OF INSTRUMENTATION**
Classification	**ALERT**
Mode	1,2,3,4
Description	**UNPLANNED loss of most (>75%) MCR annunciators (<u>and</u> Annunciator Printer) <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) 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) a. SIGNIFICANT TRANSIENT in progress b. Loss of ICS <u>and</u> SPDS
Basis	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. Fifteen minutes was selected as a threshold value to exclude momentary power losses or transients. 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. 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. Unplanned loss of annunciators excludes scheduled maintenance and testing activities.

Section 2.0	**SYSTEM DEGRADATION**
Event 2.1	**LOSS OF INSTRUMENTATION**
Classification	ALERT (continued)
Mode	1,2,3,4
Basis (continued)	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. Due to the limited number of safety systems in operation during cold shutdown and refueling modes, no initiating conditions are indicated during these modes of operation.
Escalation	Escalation of this event will be based on the inability of the operating crew to monitor a transient in progress.
References	NUMARC/NESP-007, SA4, Rev. 2, 1/92 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

Section 2.0	**SYSTEM DEGRADATION**
Event 2.1	**LOSS OF INSTRUMENTATION**
Classification	**UNUSUAL EVENT**
Mode	1,2,3,4
Description	**UNPLANNED loss of most or All Safety System annunciators or indications in the Control Room for >15 Minutes** (1 and 2 and 3) 1. Unplanned loss of most (>75%) of MCR annunciators (and Annunciator Printer) or 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) or SPDS is in service and capable of displaying data requested
Basis	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. Fifteen minutes was selected as a threshold value to exclude momentary power losses or transients. 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. Unplanned loss of annunciators excludes scheduled maintenance and testing activities. The declaration will ensure that adequate resources are available to monitor and control plant systems. 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.

Section 2.0	**SYSTEM DEGRADATION**
Event 2.1	**LOSS OF INSTRUMENTATION**
Classification	**UNUSUAL EVENT** (continued)
Mode	1,2,3,4
Escalation	Escalation of this event would be based on loss of annunciators complicated by the loss of SPDS and plant computer or a transient in progress.
References	NUMARC/NESP-007, SU3, Rev. 2, 1/92 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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Section 2.0	**SYSTEM DEGRADATION**
Event 2.2	**LOSS OF FUNCTION**
Classification	**GENERAL EMERGENCY**
Mode	Not Applicable
Description	Refer to the "Fission Product Barrier Matrix"
Basis	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.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
Event 2.2	**LOSS OF FUNCTION**
Classification	**SITE AREA EMERGENCY**
Mode	1,2,3,4
Description	**Complete loss of function needed to achieve or maintain Hot Shutdown (1 or 2)** 1. CSF status tree indicates Core Cooling Red 2. CSF status tree indicates Heat Sink Red (RHR not in service) Note: Also refer to "Failure of Rx Protection" (2.3)
Basis	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. 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. 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.
Escalation	Escalation will be based on "Fission Product Barrier Challenges".
References	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

Section 2.0	**SYSTEM DEGRADATION**
Event 2.2	**LOSS OF FUNCTION**
Classification	**ALERT**
Mode	4
Description	**Complete loss of function needed to achieve Cold Shutdown when Shutdown required by Tech Specs** (1 and 2 and 3) 1. Shutdown is required 2. Loss of RHR capability 3. Loss of secondary heat sink and condenser
Basis	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 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) 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.
Escalation	Escalation of this event would be based on complete loss of functions needed to achieve <u>or</u> maintain Hot Shutdown.
References	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

Section 2.0	**SYSTEM DEGRADATION**
Event 2.2	**LOSS OF FUNCTION**
Classification	**UNUSUAL EVENT**
Mode	All
Description	**A.** **UNPLANNED loss of all In-Plant Communication capability** (1 and 2 and 3) 1. **UNPLANNED** loss of EPABX (PAX) phones 2. **UNPLANNED** loss of all sound powered phones 3. **UNPLANNED** loss of all radios <u>or</u> **B.** **UNPLANNED loss of all Offsite Communication capability** (1 and 2 and 3 and 4 and 5) 1. **UNPLANNED** loss of all EPABX (PAX) 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 FTS 2000 (NRC) system
Basis	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. The loss of offsite communications ability is expected to be significantly more comprehensive than those addressed by 10 CFR 50.72. Onsite communications loss must encompass the loss of all means of routine communications (i.e., phones, page party system and radio/walkie talkies). 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).
Escalation	Escalation of this event will involve the loss of other plant functions.
References	NUMARC/NESP-007, SU6, Rev. 2, 1/92 10 CFR 50.72 NUREG 0654

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Section 2.0	**SYSTEM DEGRADATION**
Event 2.3	**FAILURE OF RX PROTECTION**
Classification	**GENERAL EMERGENCY**
Mode	1,2
Description	**Loss of Core Cooling capability and VALID Trip Signals did <u>not</u> result in a reduction of Rx power to <5% and decreasing (1 and 2)** 1. (a or b) 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
Basis	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. 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. 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. 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. 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.
Escalation	Not Applicable
References	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

Section 2.0	**SYSTEM DEGRADATION**
Event 2.3	**FAILURE OF RX PROTECTION**
Classification	**SITE AREA EMERGENCY**
Mode	1,2
Description	**RX power <u>Not</u> <5% and decreasing after VALID Auto and Manual Trip signals** (1 and 2 and 3) 1. VALID RX Auto Trip signal received or required. 2. Manual RX Trip from the MCR was <u>Not</u> successful. 3. FR-S.1 has been entered
Basis	This IC indicates a failure of the automatic and main control room manual signals to scram the reactor. Under these conditions, the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed. A Site Area Emergency is indicated because conditions exist that lead to imminent loss or potential loss of both fuel clad and RCS. Although this IC may be viewed as anticipatory to the Fission Product Barrier Degradation IC, its inclusion is necessary to better assure timely recognition and emergency response. FR-S.1 lists actions intended to shutdown the reactor. This includes actions in the 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. 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.
Escalation	Escalation of this event would be based on the inability to trip the Rx and indications of Heat Sink Red or Core Cooling Red.
References	NUMARC/NESP-007, SS2, Rev. 2, 1/92 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

Section 2.0	**SYSTEM DEGRADATION**
Event 2.3	**FAILURE OF RX PROTECTION**
Classification	**ALERT**
Mode	1,2
Description	**Automatic RX trip did <u>NOT</u> occur after VALID Trip signal and manual trip from MCR was successful. (1 and 2)** 1. VALID Rx Trip signal received or required 2. Manual RX Trip from the MCR <u>was</u> successful and power is <5% and decreasing.
Basis	This EAL indicates failure of the Reactor Protection System (RPS) to automatically trip the reactor. This condition is a potential degradation of a safety system in that a primary front line automatic protection system did not function in response to a plant transient or condition requiring system actuation. 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. 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. 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. FR-S.1 is not used in this EAL since reactor power is below 5% normally associated with the transition to FR-S.1. 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.
Escalation	Escalation of this event would be based on the reactor power not being reduced to less than five percent by actions of FR-S.1.
References	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

Section 2.0	**SYSTEM DEGRADATION**
Event 2.3	**FAILURE OF RX PROTECTION**
Classification	**UNUSUAL EVENT**
Mode	Not Applicable
Description	Not Applicable
Basis	Not Applicable
Escalation	Escalation of this event is based on a successful manual scram of the Rx from the main control room.
References	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

Section 2.0	**SYSTEM DEGRADATION**
Event 2.4	**FUEL CLAD DEGRADATION**
Classification	**GENERAL EMERGENCY**
Mode	Not Applicable
Description	Refer to the "Fission Product Barrier Matrix"
Basis	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.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
Event 2.4	**FUEL CLAD DEGRADATION**
Classification	**SITE AREA EMERGENCY**
Mode	Not Applicable
Description	Refer to the "Fission Product Barrier Matrix"
Basis	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.
Escalation	Escalation will be based on "Fission Product Barrier Challenges".
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
Event 2.4	**FUEL CLAD DEGRADATION**
Classification	**ALERT**
Mode	Not Applicable
Description	Refer to the "Fission Product Barrier Matrix"
Basis	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.
Escalation	Escalation will be based on "Fission Product Barrier Challenges".
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
Event 2.4	**FUEL CLAD DEGRADATION**
Classification	**UNUSUAL EVENT**
Mode	1,2,3,4,5
Description	**Reactor Coolant System specific activity exceeds LCO (Refer to WBN Tech. Spec. 3.4.16) (1)** 1. Radiochemistry analysis indicates (a or b) a. Dose equivalent Iodine (I-131) >1.0μCi/gm for >48 Hours <u>or</u> in excess of T/S Figure 3.4.16-1 b. Specific activity >100/ $\bar{E}$μCi/gm
Basis	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. The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) accident. The LCO contains specific activity limits for both Dose Equivalent I-131 and gross specific activity. The allowable levels are intended to limit the 2-hour dose at the site boundary to a small fraction of the 10 CFR 100 dose guideline values. The limits in the LCO are standardized and based on parametric evaluations of offsite radioactivity dose consequences for typical site locations. 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.
Escalation	Escalation will be based on "Fission Product Barrier Challenges".
References	NUMARC/NESP-007, SU4, Rev. 2, 1/92 T.S. 3.4.16 RCS Specific Activity AOI-28 High Activity in Reactor Coolant

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Section 2.0	**SYSTEM DEGRADATION**
Event 2.5	**RCS UNIDENTIFIED LEAKAGE**
Classification	**GENERAL EMERGENCY**
Mode	Not Applicable
Description	Refer to the "Fission Product Barrier Matrix"
Basis	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.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
Event 2.5	**RCS UNIDENTIFIED LEAKAGE**
Classification	**SITE AREA EMERGENCY**
Mode	Not Applicable
Description	Refer to the "Fission Product Barrier Matrix"
Basis	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.
Escalation	Escalation will be based on "Fission Product Barrier Challenges".
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
Event 2.5	**RCS UNIDENTIFIED LEAKAGE**
Classification	**ALERT**
Mode	Not Applicable
Description	Refer to the "Fission Product Barrier Matrix"
Basis	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.
Escalation	Escalation will be based on "Fission Product Barrier Challenges".
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
Event 2.5	**RCS UNIDENTIFIED LEAKAGE**
Classification	**UNUSUAL EVENT**
Mode	1,2,3,4,5*
Description	**Unidentified <u>or</u> pressure boundary RCS leakage >10 GPM** 1. Unidentified or pressure boundary leakage (as defined by Tech. Specs.) >10 GPM as indicated below (a or b) a. 1-SI-68-32 results b. With RCS Temperature <u>and</u> PZR Level Stable, VCT level Dropping at a Rate >10 GPM NOTE: Applies to Mode 5 if RCS Pressurized
Basis	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. Only operating modes in which there is fuel in the reactor coolant system and the system is pressurized are specified.
Escalation	Escalation will be based on "Fission Product Barrier Challenges".
References	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

Section *2.0*	**SYSTEM DEGRADATION**
Event *2.6*	**RCS IDENTIFIED LEAKAGE**
Classification	**GENERAL EMERGENCY**
Mode	Not Applicable
Description	Refer to the "Fission Product Barrier Matrix"
Basis	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.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev. 2, 1/92

Section *2.0*	**SYSTEM DEGRADATION**
Event *2.6*	**RCS IDENTIFIED LEAKAGE**
Classification	**SITE AREA EMERGENCY**
Mode	Not Applicable
Description	Refer to the "Fission Product Barrier Matrix"
Basis	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.
Escalation	Escalation will be based on "Fission Product Barrier Challengers".
References	NUMARC/NESP-007, Rev. 2, 1/92

Section *2.0*	**SYSTEM DEGRADATION**
Event *2.6*	**RCS IDENTIFIED LEAKAGE**
Classification	**ALERT**
Mode	Not Applicable
Description	Refer to the "Fission Product Barrier Matrix"
Basis	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.
Escalation	Escalation will be based on "Fission Product Barrier Challenges".
References	NUMARC/NESP-007, Rev. 2, 1/92

Section *2.0*	**SYSTEM DEGRADATION**
Event *2.6*	**RCS IDENTIFIED LEAKAGE**
Classification	**UNUSUAL EVENT**
Mode	1,2,3,4,5*
Description	**Identified RCS leakage >25 GPM** 1. Identified RCS leakage (as defined by Tech. Specs.) >25 GPM (a or b) a. I-SI-68-32 results b. Level rise in excess of 25 GPM total into PRT, RCDT or CVCS Holdup Tank *NOTE: *Applies to Mode 5 if RCS Pressurized*
Basis	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 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. Only operating modes in which there is fuel in the reactor coolant system and the system is pressurized are specified.
Escalation	Escalation will be based on "Fission Product Barrier Challenges".
References	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

Section 2.0	**SYSTEM DEGRADATION**
Event 2.7	**UNCONTROLLED COOLDOWN**
Classification	**GENERAL EMERGENCY**
Mode	Not Applicable
Description	Refer to the "Fission Product Barrier Matrix"
Basis	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.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
Event 2.7	**UNCONTROLLED COOLDOWN**
Classification	SITE AREA EMERGENCY
Mode	Not Applicable
Description	Refer to the "Fission Product Barrier Matrix"
Basis	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.
Escalation	Escalation of this event will be based on "Fission Product Barriers Challenges".
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
Event 2.7	**UNCONTROLLED COOLDOWN**
Classification	**ALERT**
Mode	Not Applicable
Description	Refer to the "Fission Product Barrier Matrix"
Basis	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.
Escalation	Escalation of this event will be based on "Fission Product Barriers Challenges".
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
Event 2.7	**UNCONTROLLED COOLDOWN**
Classification	**UNUSUAL EVENT**
Mode	1,2,3
Description	**UNPLANNED rapid depressurization of the Main Steam System resulting in a rapid RCS cooldown <u>and</u> Safety Injection initiation (1 and 2)** 1. Rapid depressurization of Main Steam System (<675 psig) 2. Safety injection has initiated <u>or is</u> required
Basis	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. Unplanned is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities.
Escalation	Escalation of this event will be based on "Fission Product Barrier Challenges".
References	NUMARC/NESP-007, HU5, Rev. 2, 1/92 E-2 Faulted Steam Generator Isolation T.S. 3.3.2 Engineering Safety Features Activation System Instrumentation (ESFAS) WBN-OSG4-188, (O-05)

Section	*2.0*	**SYSTEM DEGRADATION**
Event	*2.8*	**TURBINE FAILURE**
Classification		**GENERAL EMERGENCY**
Mode		Not Applicable
Description		Refer to the "Fission Product Barrier Matrix"
Basis		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.
Escalation		Not Applicable
References		NUMARC/NESP-007, Rev. 2, 1/92

Section	*2.0*	**SYSTEM DEGRADATION**
Event	*2.8*	**TURBINE FAILURE**
Classification		**SITE AREA EMERGENCY**
Mode		Not Applicable
Description		Refer to the "Fission Product Barrier Matrix"
Basis		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.
Escalation		Escalation of this event will be based on "Fission Product Barrier Challenges."
References		NUMARC/NESP-007, Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
Event 2.8	**TURBINE FAILURE**
Classification	**ALERT**
Mode	1,2,3
Description	**Turbine Failure has generated PROJECTILES that cause VISIBLE DAMAGE to any area containing Safety Related equipment** 1. Turbine **PROJECTILES** have resulted in **VISIBLE DAMAGE** in any of the following areas: Control Building — Diesel Generator Bldg Auxiliary Building — RWST Unit #1 Containment — Intake Pumping Station CST
Basis	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. 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. 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). 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. 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.
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

Section 2.0	**SYSTEM DEGRADATION**
Event 2.8	**TURBINE FAILURE**
Classification	**UNUSUAL EVENT**
Mode	1,2,3
Description	**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
Basis	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.
Escalation	Escalation of this event would be based on potential damage done by turbine PROJECTILES to safety related equipment.
References	NUMARC/NESP-007, HU1, Rev. 2, 1/92 FSAR 3.5.1.3 Turbine Missiles

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Section 2.0	**SYSTEM DEGRADATION**
Event 2.9	**TECHNICAL SPECIFICATION**
Classification	**GENERAL EMERGENCY**
Mode	Not Applicable
Description	Not Applicable
Basis	Technical Specifications is not applicable for a General Emergency
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
Event 2.9	**TECHNICAL SPECIFICATION**
Classification	**SITE AREA EMERGENCY**
Mode	Not Applicable
Description	Not Applicable
Basis	Technical Specifications is not applicable for a Site Area Emergency
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
Event 2.9	**TECHNICAL SPECIFICATION**
Classification	**ALERT**
Mode	Not Applicable
Description	Not Applicable
Basis	Technical Specifications is not applicable for an Alert
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
Event 2.9	**TECHNICAL SPECIFICATION**
Classification	**UNUSUAL EVENT**
Mode	1,2,3,4
Description	**Inability to reach required Shutdown within Tech. Spec. limits (1 and 2)** 1. Any Tech. Spec. LCO Statement, requiring a Mode reduction, has been entered 2. The Unit has not been placed in the required Mode within the time prescribed by the LCO Action Statement
Basis	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.
Escalation	Not Applicable
References	NUMARC/NESP-007, SU2, Rev. 2, 1/92 T.R. 3.0 Technical Requirements (TR) Applicability (T.R. 3.03)

Section 2.0	**SYSTEM DEGRADATION**
Event 2.10	**SAFETY LIMIT**
Classification	**GENERAL EMERGENCY**
Mode	Not Applicable
Description	Not Applicable
Basis	Safety limit is not applicable for a General Emergency
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
Event 2.10	**SAFETY LIMIT**
Classification	**SITE AREA EMERGENCY**
Mode	Not Applicable
Description	Not Applicable
Basis	Safety limit is not applicable for a Site Area Emergency
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
Event 2.10	**SAFETY LIMIT**
Classification	**ALERT**
Mode	Not Applicable
Description	Not Applicable
Basis	Safety limit is not applicable for an Alert
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 2.0	**SYSTEM DEGRADATION**
Event 2.10	**SAFETY LIMIT**
Classification	**UNUSUAL EVENT**
Mode	1,2,3,4,5
Description	**Safety Limits have been Exceeded** (1 or 2) 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)
Basis	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. 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. 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. 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. This EAL is consistent with the definition of an Unusual Event and warrants declaration.

Section 2.0	**SYSTEM DEGRADATION**
Event 2.10	**SAFETY LIMIT**
Classification	**UNUSUAL EVENT** (continued)
Mode	1,2,3,4,5
Escalation	Not Applicable
References	NUMARC/NESP-007, SU2, Rev. 2, 1/92 B.2.1.1 Reactor Core Safety Limits (SLs)

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Section 3.0	**LOSS OF POWER**
Event 3.1	**LOSS OF AC (Power Ops)**
Classification	**GENERAL EMERGENCY**
Mode	**1,2,3,4**
Description	**Prolonged loss of Offsite <u>and</u> Onsite AC Power (1 and 2)** 1. 1A <u>and</u> 1B 6.9KV Shutdown Bds de-energized for >15 minutes 2. (a or b) 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
Basis	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. 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. The (15 minute) time duration was selected to exclude transient or momentary power losses. 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: 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?

Section 3.0	**LOSS OF POWER**
Event 3.1	**LOSS OF AC (Power Ops)**
Classification	**GENERAL EMERGENCY (continued)**
Mode	**1,2,3,4**
Basis (continued)	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? 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.
Escalation	Not Applicable
Reference	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

Section 3.0	**LOSS OF POWER**
Event 3.1	**LOSS OF AC (Power Ops)**
Classification	**SITE AREA EMERGENCY**
Mode	**1,2,3,4**
Description	**Loss of Offsite <u>and</u> Onsite AC power for >15 Minutes** 1. 1A <u>and</u> 1B 6.9KV Shutdown Bds de-energized for >15 minutes
Basis	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. The (15 minute) time duration was selected to exclude transient or momentary power Losses.
Escalation	Prolonged loss of all offsite power and prolonged loss of all onsite power will, when combined with inadequate core cooling, result in an escalation of this event.
References	NUMARC/NESP-007 SS1, Rev. 2, 1/92 AOI-35 Loss of Offsite Power FSAR 15.2.9 Loss of Offsite Power to the Station Auxiliaries General Design Criteria 17, App. ,A 10 CFR 50 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

Section	*3.0*	**LOSS OF POWER**
Event	*3.1*	**LOSS OF AC (Power Ops)**
Classification		**ALERT**
Mode		**1,2,3,4**
Description		**Loss of Offsite Power and 1A or 1B Diesel Generator (1 and 2)** 1. C and D CSSTs not available for >15 minutes 2. 1A or 1B Diesel Generator not available
Basis		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. The (15 minute) time duration was selected to exclude transient or momentary power losses.
Escalation		Prolonged Loss of all offsite power and prolonged Loss of all onsite power will escalate this event.
References		NUMARC/NESP-007,SA5, Rev. 2, 1/92 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

Section 3.0	**LOSS OF POWER**
Event 3.1	**LOSS OF AC (Power Ops)**
Classification	**UNUSUAL EVENT**
Mode	**1,2,3,4**
Description	**Loss of Offsite Power for >15 Minutes (1 and 2)** 1. C <u>and</u> D CSSTs not available for > 15 minutes 2. Each Diesel Generator is supplying power to its respective Shutdown Bd
Basis	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.
Escalation	Loss of one additional power supply to the shutdown boards will escalate this event.
References	NUMARC/NESP-007 SU1, Rev. 2, 1/92 AOI-35 Loss of Offsite Power 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 General Design Criterion 17 of App. A, 10 CFR 50 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

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Section *3.0*	**LOSS OF POWER**
Event *3.2*	**LOSS OF AC (Shutdown)**
Classification	**GENERAL EMERGENCY**
Mode	Not Applicable
Description	Not Applicable
Basis	Loss of AC Power in Mode 5 and 6 will Not cause a declaration of a General Emergency
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev 2, 1/92

Section *3.0*	**LOSS OF POWER**
Event *3.2*	**LOSS OF AC (Shutdown)**
Classification	**SITE AREA EMERGENCY**
Mode	Not Applicable
Description	Not Applicable
Basis	Loss of AC Power in Mode 5 and 6 will Not cause a declaration of a Site Area Emergency.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev 2, 1/92

Section 3.0	**LOSS OF POWER**
Event 3.2	**LOSS OF AC** (Shutdown)
Classification	**ALERT**
Mode	5,6, or Defuel
Description	**UNPLANNED Loss of Offsite <u>And</u> Onsite AC power for >15 minutes** 1. 1A <u>and</u> 1B 6.9KV Shutdown Bds de-energized for >15 minutes Also Refer to "Loss of Shutdown Systems" (6.1)
Basis	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. Unplanned is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities.
Escalation	Escalation is not applicable from this event.
References	NUMARC/NESP-007 SU1 (expanded), Rev 2, 1/92 T.S. 3.8.2 AC Sources - Shutdown T.S. 3.8.10 Distribution Systems - Shutdown

Section 3.0	**LOSS OF POWER**
Event 3.2	**LOSS OF AC (Shutdown)**
Classification	**UNUSUAL EVENT**
Mode	5,6, Defuel
Description	**UNPLANNED Loss of Offsite Power for >15 minutes (1 and 2)** 1. C and D CSSTs not available for >15 minutes 2. Either Diesel Generator is supplying power to its respective Shutdown Board
Basis	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. Unplanned is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities.
Escalation	Loss of one additional power supply to the shutdown boards will escalate this event.
References	NUMARC/NESP-007, SU1, Rev 2, 1/92 T.S. 3.8.2 AC Sources - Shutdown T.S. 3.8.10 Distribution Systems - Shutdown

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Section *3.0*	**LOSS OF POWER**
Event *3.3*	**LOSS OF DC**
Classification	**GENERAL EMERGENCY**
Mode	Not Applicable
Description	Refer to "Fission Product Barrier Matrix" and "Loss of Function" (2.2)
Basis	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.
Escalation	Refer to the "Fission Product Barrier Matrix" or " Loss of Function" (2.2) for escalation considerations.
References	NUMARC/NESP-007, Rev 2, 1/92

Section 3.0	**LOSS OF POWER**
Event 3.3	**LOSS OF DC**
Classification	**SITE AREA EMERGENCY**
Mode	1,2,3,4
Description	**Loss of All Vital DC Power for >15 minutes** 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 Also Refer to the "Fission Product Battier Matrix", "Loss of Function" (2.2), and "Loss of Instrumentation" (2.1)
Basis	Loss of all DC power compromises the ability to monitor and control plant safety functions. Prolonged loss of all DC power will cause core uncovering and loss of containment integrity when there is significant decay heat and sensible heat in the reactor system. Fifteen minutes was selected as a threshold to exclude transient or momentary power losses. 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.
Escalation	Escalation would occur through the Fission Product Barrier Matrix Degradation or Loss or Function (2.2)
References	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. 3.8.4 DC Sources - Operating AOI - 21.1 - 21.8 Loss of 125V DC Vital Battery Boards

Section *3.0*	LOSS OF POWER
Event *3.3*	LOSS OF DC
Classification	ALERT
Mode	1,2,3,4
Description	Refer to "Fission Product Barrier Matrix", "Loss of Function" (2.2), and "Loss of Instrumentation" (2.1)
Basis	There is NO 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.
Escalation	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.
References	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-21.1-21.8 Loss of 125V DC Vital Battery Boards FSAR 8.3.2 DC Power Systems

Section 3.0	**LOSS OF POWER**
Event 3.3	**LOSS OF DC**
Classification	**UNUSUAL EVENT**
Mode	5,6 and Defuel
Description	**UNPLANNED Loss of the Required Train of DC power for >15 Minutes** (1 or 2) 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
Basis	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. Unplanned is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities. 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. The fifteen minute threshold is utilized to exclude a transient or momentary power losses.
Escalation	The event will escalate if the DC loss results in an inability to maintain cold shutdown.
References	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-21.1-21.8 Loss of 125V DC Vital Battery Boards

Section 4.0	**HAZARDS AND SED JUDGEMENT**
Event 4.1	**FIRE**
Classification	**GENERAL EMERGENCY**
Mode	Not Applicable
Description	Refer to "Fission Product Barrier Matrix"
Basis	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.
Escalation	Not Applicable
References	NUMARC/NESP-007 Rev. 2, 1/92

Section 4.0	**HAZARDS AND SED JUDGEMENT**
Event 4.1	**FIRE**
Classification	**SITE AREA EMERGENCY**
Mode	Not Applicable
Description	Refer to "Control Room Evacuation," (4.5) or "Fission Product Barrier Matrix"
Basis	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. 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"
Escalation	Escalation would be based on "Fission Product Barrier Challenges"
References	NUMARC/NESP-007 Rev. 2, 1/92

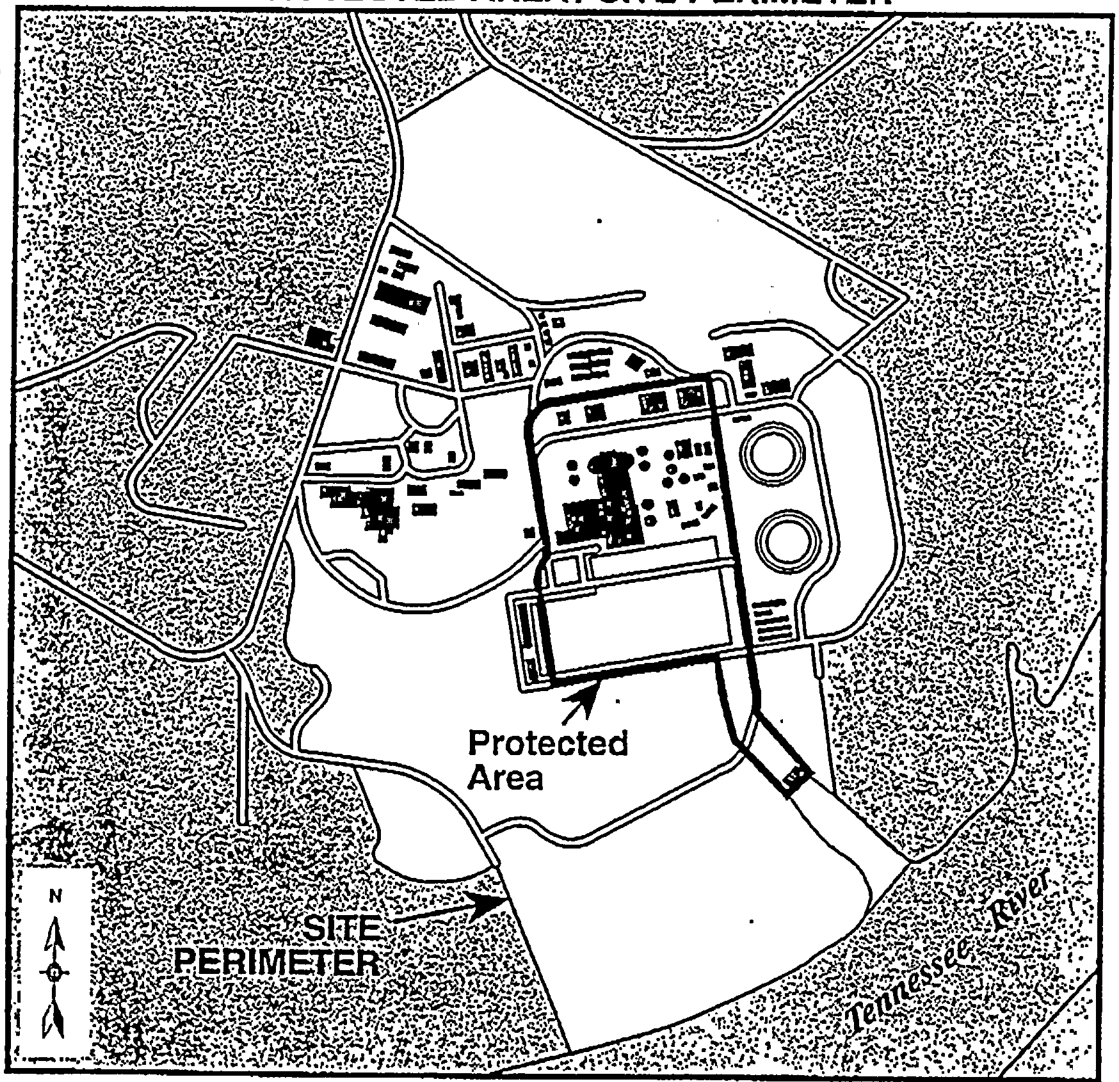
Section 4.0	**HAZARDS AND SED JUDGEMENT**
Event 4.1	**FIRE**
Classification	**ALERT**
Mode	All
Description	**FIRE in any of the areas listed in Table 4-1 That is Affecting Safety Related Equipment** (1 and 2) 1. **FIRE** in any of the areas listed in Table 4-1 2. (a or b) a. **VISIBLE DAMAGE** to permanent 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
Basis	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. Table 4-1 Plant Structures Associated with Fire and Explosion EALs 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 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.

Section *4.0*	**HAZARDS AND SED JUDGEMENT**
Event *4.1*	**FIRE**
Classification	**ALERT** (continued)
Mode	All
Basis (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.
Escalation	Escalation would be based on Fission Product Barrier challenges or Control Room Evacuation (4.5)
References	NUMARC/NESP-007, HA2, Rev. 2, 1/92 Figure 4-A Protected Area and Site Perimeter

Section 4.0	**HAZARDS AND SED JUDGEMENT**
Event 4.1	**FIRE**
Classification	**UNUSUAL EVENT**
Mode	All
Description	**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**
Basis	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. The 15 minute time limit has been established to exclude small fires that can be controlled by Plant Fire Fighting resources. 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. Table 4-1 Plant Structures Associated with Fire and Explosion EALs 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

Figure 4-A

PROTECTED AREA / SITE PERIMETER



Section 4.0	**HAZARDS AND SED JUDGEMENT**
Event 4.1	**FIRE**
Classification	**UNUSUAL EVENT** (continued)
Mode	All
Basis (continued)	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. PROTECTED AREA encompasses all owner controlled areas within the security protected area fence as shown on Figure 4-A.
Escalation	Escalation of this event is based on the Fire affecting plant safety related equipment required to establish or maintain safe shutdown.
References	NUMARC/NESP-007, HU2, Rev. 2, 1/92 Figure 4-A Protected Area/Site Perimeter

Section	*4.0*	**HAZARDS AND SED JUDGEMENT**
Event	*4.2*	**EXPLOSIONS**
Classification		**GENERAL EMERGENCY**
Mode		Not Applicable
Description		Refer to the "Fission Product Barrier Matrix"
Basis		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.
Escalation		Not Applicable
References		NUMARC/NESP-007 Rev. 2, 1/92

Section	*4.0*	**HAZARDS AND SED JUDGEMENT**
Event	*4.2*	**EXPLOSIONS**
Classification		**SITE AREA EMERGENCY**
Mode		Not Applicable
Description		Refer to the "Fission Product Barrier Matrix"
Basis		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.
Escalation		Escalation would be based on "Fission Product Barrier Challenges".
References		NUMARC/NESP-007, Rev. 2, 1/92

Section 4.0	**HAZARDS AND SED JUDGEMENT**
Event 4.2	**EXPLOSIONS**
Classification	**ALERT**
Mode	All
Description	**EXPLOSION in any of the areas listed in Table 4-1 that is affecting Safety Related equipment** (1 and 2) 1. **EXPLOSION** in any of the areas listed in Table 4-1 2. (a or b) a. 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 Refer to Security (4.6)
Basis	EXPLOSIONS include those that are of sufficient magnitude to damage permanent structures or equipment within the plant vital area. As used here, an EXPLOSION is a rapid, violent, unconfined combustion, or a catastrophic failure of pressurized equipment, that potentially imparts significant energy to near-by structures and material. VISIBLE DAMAGE is damage to equipment that is readily observable without measurements, testing, or analyses. Damage is sufficient enough to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering. Surface blemishes (e.g., paint chipping, scratches) should NOT be included. The "Report of VISIBLE DAMAGE" should not be interpreted as requiring a lengthy damage assessment prior to classification. The observation of damage to a structure is sufficient to make a declaration. The declaration of the Alert and the activation of the TSC is warranted and will provide the Site Emergency Director (SED) with resources necessary to perform damage assessment.

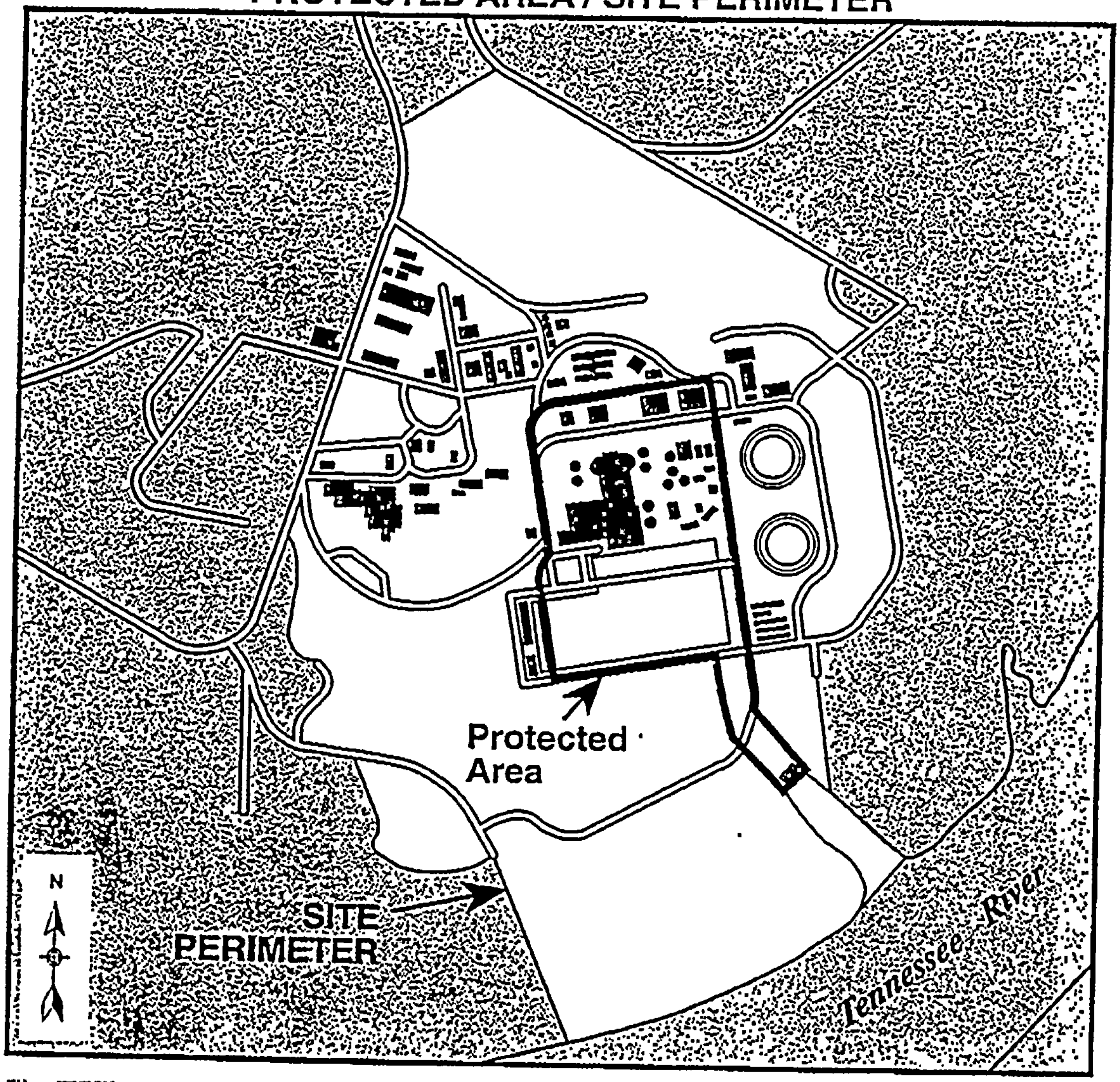
Section 4.0	**HAZARDS AND SED JUDGEMENT**
Event 4.2	**EXPLOSIONS**
Classification	**ALERT** (continued)
Mode	All
Basis (continued)	Table 4-1 Plant Structures Associated with Fire and Explosion EALs Unit #1 Reactor Building Auxiliary Building Control Building Diesel Generator Building CST Additional Diesel Generator Building Intake Pumping Station Additional Equipment Bldgs (Unit 1 & 2) RWST
Escalation	Escalation will be based on "Fission Product Barrier Challenges".
References	NUMARC/NESP-007, HA2, Rev 2, 1/92

Section 4.0	**HAZARDS AND SED JUDGEMENT**
Event 4.2	**EXPLOSIONS**
Classification	**UNUSUAL EVENT**
Mode	All
Description	**UNPLANNED EXPLOSION Within the PROTECTED AREA Resulting in VISIBLE DAMAGE to Any Permanent Structure <u>or</u> Equipment** (Figure 4-A) Refer to Security (4.6)
Basis	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. 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. 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. 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. Unplanned is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities. PROTECTED AREA encompasses all owner controlled areas within the security protected area fence as shown on Figure 4-A.

Section 4.0	**HAZARDS AND SED JUDGEMENT**
Event 4.2	**EXPLOSIONS**
Classification	**UNUSUAL EVENT** (continued)
Mode	All
Escalation	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.
References	NUMARC/NESP-007, HU2, Rev 2, 1/92 FSAR 2.2 Nearby Industrial, Transportation and Military Facilities

Figure 4-A

PROTECTED AREA / SITE PERIMETER



Section 4.0	**HAZARDS AND SED JUDGEMENT**
Event 4.3	**FLAMMABLE GAS**
Classification	**GENERAL EMERGENCY**
Mode	Not Applicable
Description	Refer to the "Fission Product Barrier Matrix"
Basis	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.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev 2, 1/92

Section 4.0	**HAZARDS AND SED JUDGEMENT**
Event 4.3	**FLAMMABLE GAS**
Classification	**SITE AREA EMERGENCY**
Mode	Not Applicable
Description	Refer to the "Fission Product Barrier Matrix"
Basis	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.
Escalation	Escalation will be based on "Fission Product Barrier Challenges".
References	NUMARC/NESP-007, Rev 2, 1/92

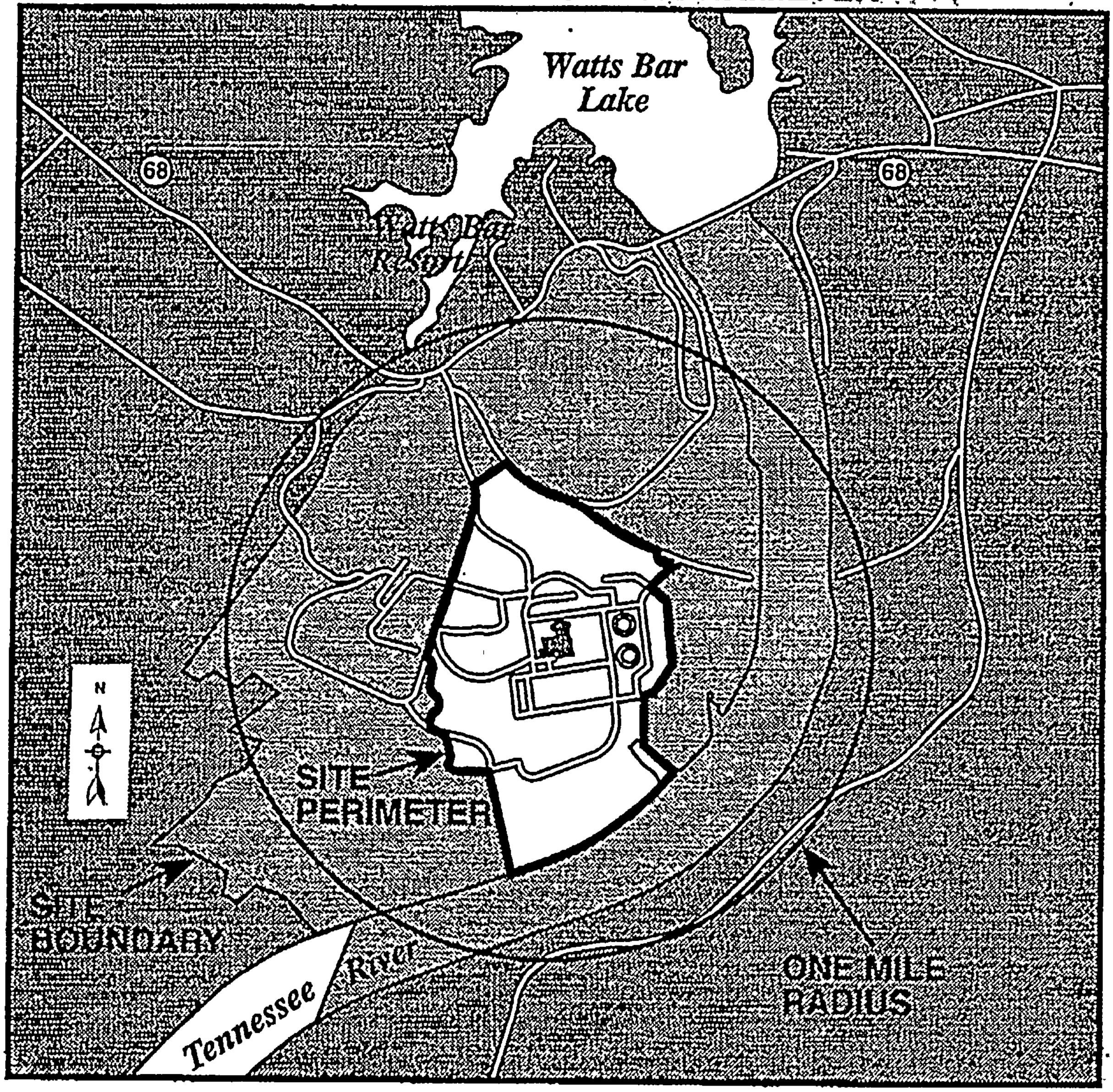
Section 4.0	**HAZARDS AND SED JUDGEMENT**
Event 4.3	**FLAMMABLE GAS**
Classification	**ALERT**
Mode	All
Description	**UNPLANNED release of Flammable Gas within a facility structure containing Safety Related equipment or associated with Power production.** 1. Plant personnel report the average of three readings taken in a ~10ft Triangular Area is > 25% (LEL) Lower Explosive Limit as indicated on the monitoring instrument within any building listed in Table 4-2
Basis	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. 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 Unplanned is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities.
Escalation	Escalation will be based on "Fission Product Barrier Challenges".
References	NUMARC/NESP-007, HA3, Rev 2, 1/92 Figure 4-B One Mile Radius/Site Perimeter

Section *4.0*	**HAZARDS AND SED JUDGEMENT**
Event *4.3*	**FLAMMABLE GAS**
Classification	**UNUSUAL EVENT**
Mode	All
Description	**A. UNPLANNED release of Flammable Gas within the SITE PERIMETER.** 1. Plant personnel report the average of three readings taken in a ~10ft Triangular Are is > 25% (LEL) Lower Explosive Limit as indicated on the monitoring instrument within the SITE PERIMETER (Refer to Figure 4-B) OR **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)**
Basis	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. 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. 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.

Section 4.0	**HAZARDS AND SED JUDGEMENT**
Event 4.3	**FLAMMABLE GAS**
Classification	**UNUSUAL EVENT** (continued)
Mode	All
Basis (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 Unplanned is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities. SITE PERIMETER encompasses all owner controlled areas in the immediate site environs as shown on Figure 4-B and 7-A.
Escalation	Escalation of this event would be based on flammable gases entering a plant area that jeopardizes life or impacts cold shutdown capabilities.
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-B

ONE MILE RADIUS / SITE PERIMETER



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Section *4.0*	**HAZARDS AND SED JUDGEMENT**
Event *4.4*	**TOXIC GAS**
Classification	**GENERAL EMERGENCY**
Mode	Not Applicable
Description	Refer to the "Fission Product Barrier Matrix"
Basis	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.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev 2, 1/92

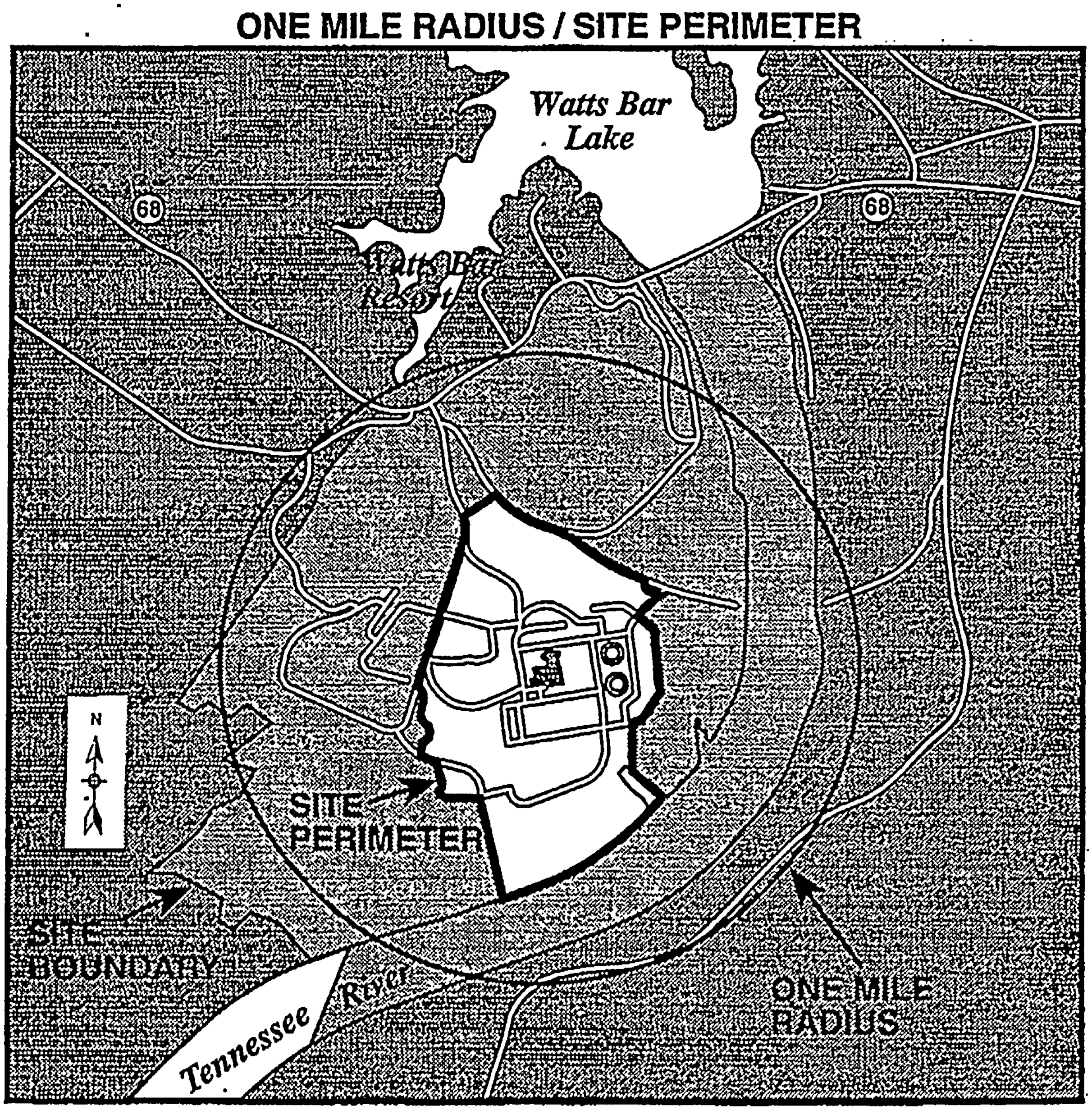
Section *4.0*	**HAZARDS AND SED JUDGEMENT**
Event *4.4*	**TOXIC GAS**
Classification	**SITE AREA EMERGENCY**
Mode	Not Applicable
Description	Refer to the "Fission Product Barrier Matrix"
Basis	The basis for a Site Area Emergency in this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev 2, 1/92

Section 4.0	**HAZARDS AND SED JUDGEMENT**
Event 4.4	**TOXIC GAS**
Classification	**ALERT**
Mode	All
Description	**Release of TOXIC GAS within a facility structure which Prohibits Safe Operation of systems required to establish or maintain Cold S/D** (1 and 2 and 3) 1. Plant personnel report **TOXIC GAS** within any building listed in Table 4-2 2. (a or b) 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
Basis	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. 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 TOXIC GAS is a gas that is dangerous to life or limb by reason of inhalation or skin contact (e.g., chlorine).
Escalation	Escalation will be based on "Fission Product Barrier Challenges".
References	NUMARC/NESP-007, HA2, Rev 2, 1/92 Figure 4-B One Mile Radius/Site Perimeter AOI-32 Chlorine Release

Section 4.0	**HAZARDS AND SED JUDGEMENT**
Event 4.4	**TOXIC GAS**
Classification	**UNUSUAL EVENT**
Mode	All
Description	**A. Normal Operations impeded due to access restrictions caused by TOXIC GAS concentrations within a Facility Structure Listed in Table 4-2** **OR** **B. 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 PERIMETER in Concentrations > than the Permissible Exposure Limit (PEL) thus causing an Evacuation (Figure 4-B)**
Basis	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. 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. 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. TOXIC GAS is a gas that is dangerous to life or limb by reason of inhalation or skin contact (e.g., chlorine). SITE PERIMETER encompasses all owner controlled areas in the immediate site environs as shown on Figure 4-A.

Section 4.0	**HAZARDS AND SED JUDGEMENT**
Event 4.4	**TOXIC GAS**
Classification	**UNUSUAL EVENT** (continued)
Mode	All
Basis (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
Escalation	Escalation to this event will be based on toxic gases entering a plant area that jeopardizes life or impacts cold shutdown capability
References	NUMARC/NESP-007, HU3, Rev 2, 1/92 FSAR 2.2 Nearby Industrial, Transportation and Military Facilities AOI-32 Chlorine Release DOT Emergency Response Guide for Hazardous Materials Figure 4-B One Mile Radius/Site Perimeter

Figure 4-B



ONE MILE RADIUS / SITE PERIMETER
Watts Bar Lake
68
68
Watts Bar Resort
N
SITE PERIMETER
SITE BOUNDARY
Tennessee River
ONE MILE RADIUS

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Section *4.0*	**HAZARDS AND SED JUDGEMENT**
Event *4.5*	**CONTROL ROOM EVACUATION**
Classification	**GENERAL EMERGENCY**
Mode	Not Applicable
Description	Refer to the "Fission Product Barrier Matrix"
Basis	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.
Escalation	Escalation will be based on "Fission Product Barrier Challenges".
References	NUMARC/NESP-007, Rev 2, 1/92

Section *4.0*	**HAZARDS AND SED JUDGEMENT**
Event *4.5*	**CONTROL ROOM EVACUATION**
Classification	**SITE AREA EMERGENCY**
Mode	All
Description	**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) 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 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.
Basis	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. 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 uncovery 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.
Escalation	Escalation will be based on "Fission Product Barrier Challenges".
References	NUMARC/NESP-007, HS2, Rev 2, 1/92 AOI-30.2 Fire Safe Shutdown AOI-27 "Main Control Room Inaccessibility

Section *4.0*	**HAZARDS AND SED JUDGEMENT**
Event *4.5*	**CONTROL ROOM EVACUATION**
Classification	**ALERT**
Mode	All
Description	**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
Basis	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.
Escalation	Escalation of this event would be based on the inability to establish plant control from outside the Main Control Room within 15 minutes.
References	NUMARC/NESP-007, HA5, Rev 2, 1/92 AOI-30.2 Fire Safe Shutdown AOI-27 "Main Control Room Inaccessibility"

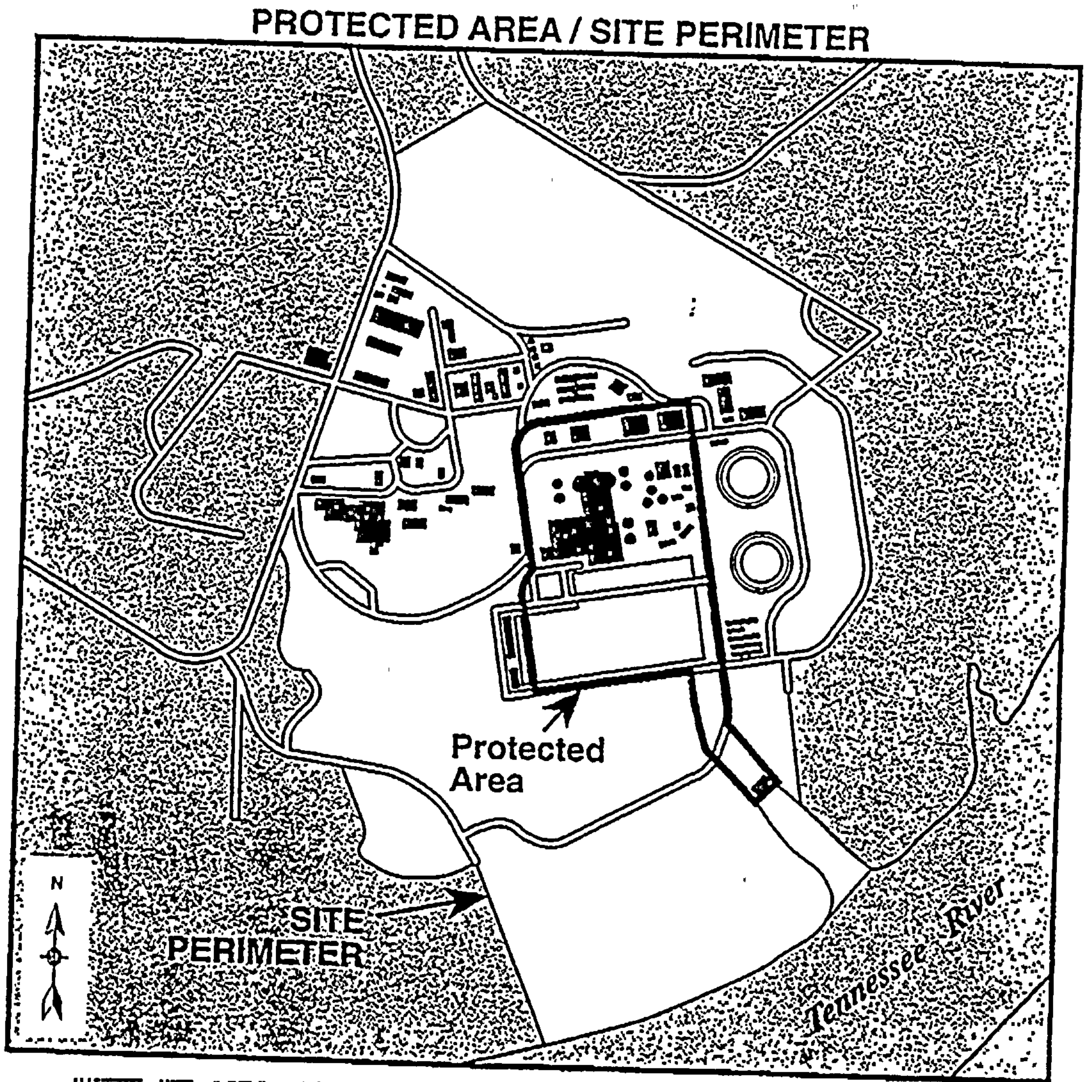
Section *4.0*	**HAZARDS AND SED JUDGEMENT**
Event *4.5*	**CONTROL ROOM EVACUATIONS**
Classification	**UNUSUAL EVENT**
Mode	Not Applicable
Description	An Unusual Event for this event is "Not Applicable"
Basis	Not Applicable
Escalation	Escalation of this event would be based on Evacuation of the Main Control Room.
References	NUMARC/NESP-007, Rev 2, 1/92

Section *4.0*	**HAZARDS AND SED JUDGEMENT**
Event *4.6*	**SECURITY**
Classification	**GENERAL EMERGENCY**
Mode	All
Description	**Security Event Resulting in loss of Control of the Plant** 1. Hostile Armed Force has taken Control of the Plant, Control Room, <u>or</u> Remote shutdown capability
Basis	This event represents a condition where a hostile force has taken control of the Main Control Room or vital areas within the plant that are required to reach and maintain a cold shutdown. This loss could be due to physical loss of control or by the damage of essential equipment. This situation leaves the plant in a very unstable condition with a high potential of multiple barrier failures.
Escalation	Not Applicable
References	NUMARC/NESP-007, HG1, Rev 2, 1/92

Section *4.0*	**HAZARDS AND SED JUDGEMENT**
Event *4.6*	**SECURITY**
Classification	**SITE AREA EMERGENCY**
Mode	All
Description	**Security Event has <u>or</u> is occurring which results in Actual <u>or</u> Likely Failures of Plant Functions needed to Protect the Public** 1. **VITAL AREA**, Other Than the Control Room, Has Been Penetrated By a Hostile Armed Force
Basis	This event represents a threat to the safety of the plant since there has been a hostile intrusion into the areas of the plant that contain equipment important to maintaining the plant in a safe condition. A confirmed security event is satisfied when physical evidence of a hostile intrusion exist. VITAL AREA is any area within the PROTECTED AREA which contains equipment, systems, devices, or material, the failure, destruction, or release of which could directly or indirectly endanger the public health and safety by exposure to radiation.
Escalation	Escalation of this event would be based on Loss of Plant Control.
References	NUMARC/NESP-007, HS1, Rev 2, 1/92

Section *4.0*	HAZARDS AND SED JUDGEMENT
Event *4.6*	SECURITY
Classification	ALERT
Mode	All
Description	**Confirmed Security Event which indicates an Actual <u>or</u> Potential Substantial Degradation in the level of Safety of the Plant** (1 or 2 or 3) 1. **BOMB** discovered within a **VITAL AREA** 2. **CIVIL DISTURBANCE** ongoing within the **PROTECTED AREA** 3. **PROTECTED AREA** has been penetrated By a Hostile Armed Force Refer to Figure 4-A For a Drawing of **Protected Area and Site Perimeter**
Basis	These class of Security, events represent a threat to the level of safety of the plant. A confirmed report is satisfied if physical evidence supporting the hostile intrusion or Bomb is discovered in the (Vital Area) Protected Area. BOMB refers to an explosive device. A CIVIL DISTURBANCE exists when there is a group of twenty (20) or more persons violently protesting station operations or activities at the site. PROTECTED AREA encompasses all owner controlled areas within the security protected area fence as shown on Figure 4-A.
Escalation	Escalation of this event would be based on hostile intrusion into plant vital areas.
References	NUMARC/NESP-007, HA4, Rev 2, 1/92 Figure 4-A Protected Area and Site Perimeter

Figure 4-A


PROTECTED AREA / SITE PERIMETER
Protected
Area
SITE
PERIMETER
N
Tennessee River

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Section 4.0	**HAZARDS AND SED JUDGEMENT**
Event 4.6	**SECURITY**
Classification	**UNUSUAL EVENT**
Mode	All
Description	**Confirmed Security Event which indicates a Potential Degradation in the level of Safety of the Plant** (1 or 2) 1. **BOMB** discovered within the **PROTECTED AREA** 2. Security Shift Supervisor reports one <u>or</u> more of the events listed in Table 4-3
Basis	A security threat that is identified as being directed towards the Station which represents a potential degradation in the level of safety of the plant warrants declaration of an Unusual Event. A confirmed report is satisfied if physical evidence supporting the threat exists, information independent from the actual threat message exists or a specific group claims responsibility for the threat. Examples of security events are provided in Table 4-3 Security Events a. **SABOTAGE/INTRUSION** has <u>or</u> is occurring within the **PROTECTED AREA** b. **HOSTAGE/EXTORTION** Situation that threatens to interrupt Plant Operations c. **CIVIL DISTURBANCE** ongoing between the **SITE PERIMETER** and **PROTECTED AREA** d. Hostile **STRIKE ACTION** within the **PROTECTED AREA** which threatens to interrupt Normal Plant Operations (judgement based on behavior of Strikers <u>and/or</u> intelligence received) In addition, Watts Bar uses a trained security organization and an approved physical security plan and procedures. External events which may result in a security threat would be reported to the duty Shift Manager (SM) by the Nuclear Security Supervisor. If in the SM's judgment these events constitute an actual threat, they would be reported and a declaration made. BOMB refers to an explosive device. A HOSTAGE is a person(s) held as leverage against the station to ensure that demands will be met by the station. PROTECTED AREA encompasses all owner controlled areas within the security protected area fence as shown on Figure 4-A.

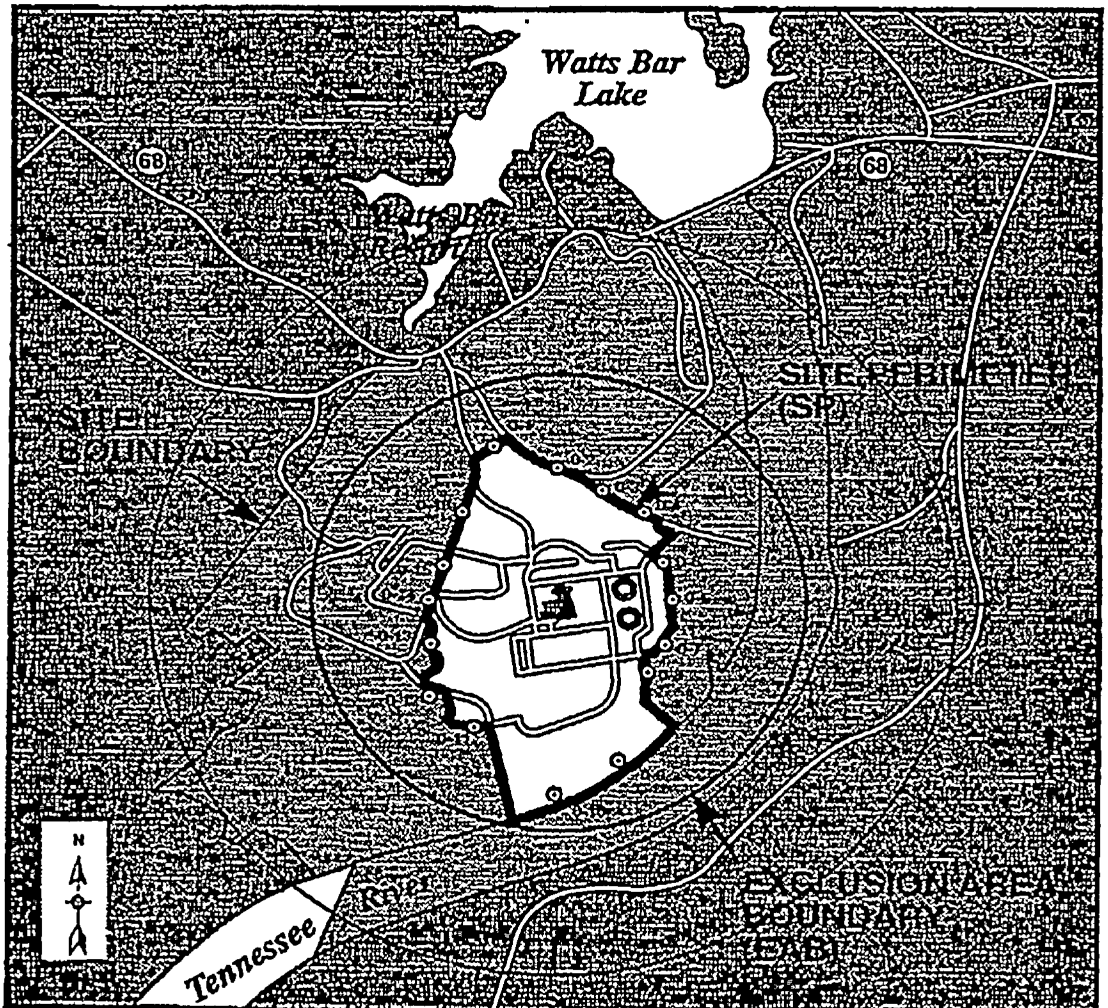
Section 4.0	**HAZARDS AND SED JUDGEMENT**
Event 4.6	**SECURITY**
Classification	**UNUSUAL EVENT** (continued)
Mode	All
Basis (continued)	SABOTAGE is deliberate damage, mis-alignment, or mis-operation of plant equipment with the intent to render the equipment inoperable. A CIVIL DISTURBANCE exists when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

Section 4.0	**HAZARDS AND SED JUDGEMENT**
Event 4.7	**EMERGENCY DIRECTOR JUDGEMENT**
Classification	**GENERAL EMERGENCY**
Mode	All
Description	**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. Releases can be reasonably expected to exceed EPA Plume Protective Action Guidelines Exposure Levels outside the EXCLUSION AREA BOUNDARY. Refer to Figure 7.A**
Basis	This event classification provides the Shift Supervisor/Site Emergency Director, the flexibility to declare a General Emergency if in their judgement unanticipated conditions not explicitly covered elsewhere warrant declaration of an emergency. The declaration of a General Emergency indicates that there is a very high probability that the fuel has been damaged and the loss of containment integrity is possible or other conditions exist that may result in a release to the environment that may be greater than the EPA Protective Action Guides.
Escalation	Not Applicable
References	NUMARC/NESP-007, HG2, Rev 2, 1/92

Section 4.0	**HAZARDS AND SED JUDGEMENT**
Event 4.7	**EMERGENCY DIRECTOR JUDGEMENT**
Classification	**SITE AREA EMERGENCY**
Mode	All
Description	**Events are in process <u>or</u> have occurred which involve Actual or Likely Major Failures of Plant Functions needed for the Protection of the Public. Any releases are not expected to result in Exposure Levels which Exceed EPA Plume Protective Action Guideline Exposure Levels Outside the EXCLUSION AREA BOUNDARY. Refer to Figure 7.A**
Basis	This event classification provides the Shift Supervisor/Site Emergency Director, the flexibility to declare a Site Area Emergency if in their judgement unanticipated conditions not explicitly covered elsewhere warrant declaration. The declaration of a Site Area Emergency indicates high probability of Major failures of plant functions needed to protect the public.
Escalation	Escalation of this event would be based on actual or imminent substantial core degradation.
References	NUMARC/NESP-007, HS2, Rev 2, 1/92

Figure 7-A

EXCLUSION AREA/SITE BOUNDARY



Section *4.0*	**HAZARDS AND SED JUDGEMENT**
Event *4.7*	**EMERGENCY DIRECTOR JUDGEMENT**
Classification	**ALERT**
Mode	All
Description	**Events are in process or have occurred which involve an Actual or Potential Substantial Degradation of the level of safety of the plant. Any releases are not expected to be limited to small fractions of the EPA Plume Protective Action Guideline Exposure Levels.**
Basis	This event classification provides the Shift Supervisor or the Site Emergency Director, the flexibility to declare an Alert if, in their judgement, unanticipated conditions not explicitly covered elsewhere warrant declaration of an emergency.
Escalation	Escalation of this event would be based on actual or likely failures in plant functions needed to protect the public.
References	NUMARC/NESP-007, HA6, Rev 2, 1/92

Section *4.0*	**HAZARDS AND SED JUDGEMENT**
Event *4.7*	**EMERGENCY DIRECTOR JUDGEMENT**
Classification	**UNUSUAL EVENT**
Mode	All
Description	**Unusual Events are in Process or have occurred which indicate a Potential Degradation of the level of Safety of the Plant. No Releases of Radioactive material requiring Offsite Response or Monitoring are expected unless further degradation of Safety Systems occurs.**
Basis	This event classification provides the Shift Supervisor the flexibility to declare an Unusual Event if, in his judgement, unanticipated conditions not explicitly covered elsewhere warrant declaration of an emergency.
Escalation	Escalation of this event would be based on actual degradation of plant safety systems.
References	NUMARC/NESP-007, HU5, Rev 2, 1/92

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Section *5.0*	**DESTRUCTIVE PHENOMENON**
Event *5.1*	**EARTHQUAKE**
Classification	**GENERAL EMERGENCY**
Mode	Not Applicable
Description	Refer to the "Fission Product Barrier Matrix"
Basis	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.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev. 2, 1/92

Section *5.0*	**DESTRUCTIVE PHENOMENON**
Event *5.1*	**EARTHQUAKE**
Classification	**SITE AREA EMERGENCY**
Mode	Not Applicable
Description	Refer to the "Fission Product Barrier Matrix"
Basis	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.
Escalation	Escalation will be based on "Fission Product Barrier Challenges".
References	NUMARC/NESP-007, Rev. 2, 1/92

Section *5.0*	DESTRUCTIVE PHENOMENON
Event *5.1*	EARTHQUAKE
Classification	ALERT
Mode	All
Description	**Earthquake detected by site seismic instrumentation (1 and 2)** 1. (a and b) a. Ann.166 D indicates "OBE Spectra Exceeded" b. Ann.166 E indicates "Seismic Recording Initiated" 2. (a or b) a. Ground motion sensed by Plant personnel b. National Earthquake Information Center at 1-(303) 273-8500 can confirm the event
Basis	A seismic event of this level can cause damage to safety related systems. Plant seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix "A" of 10 CFR Part 100. All specified measurement ranges represent the minimum ranges of the instruments.
Escalation	Escalation of this event will be based on " Fission Product Barrier Challenges".
References	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

Section 5.0	**DESTRUCTIVE PHENOMENON**
Event 5.1	**EARTHQUAKE**
Classification	**UNUSUAL EVENT**
Mode	All
Description	**Earthquake detected by site seismic instrumentation (1 and 2)** 1. Ann. 166E indicator "Seismic Recording initiated" 2. (a or b) a. Ground motion sensed by Plant personnel b. National Earthquake Information Center at 1-(303) 273-8500 can confirm the event
Basis	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. 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.
Escalation	Escalation of this event will be based on a Safe Shutdown Earthquake. (SSE)
References	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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Section *5.0*	**DESTRUCTIVE PHENOMENON**
Event *5.2*	**TORNADO**
Classification	**GENERAL EMERGENCY**
Mode	Not Applicable
Description	Refer to the "Fission Product Barrier Matrix"
Basis	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.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev. 2, 1/92

Section *5.0*	**DESTRUCTIVE PHENOMENON**
Event *5.2*	**TORNADO**
Classification	**SITE AREA EMERGENCY**
Mode	Not Applicable
Description	Refer to the "Fission Product Barrier Matrix"
Basis	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.
Escalation	Escalation will be based on "Fission Product Barrier Challenges".
References	NUMARC/NESP-007, Rev. 2, 1/92

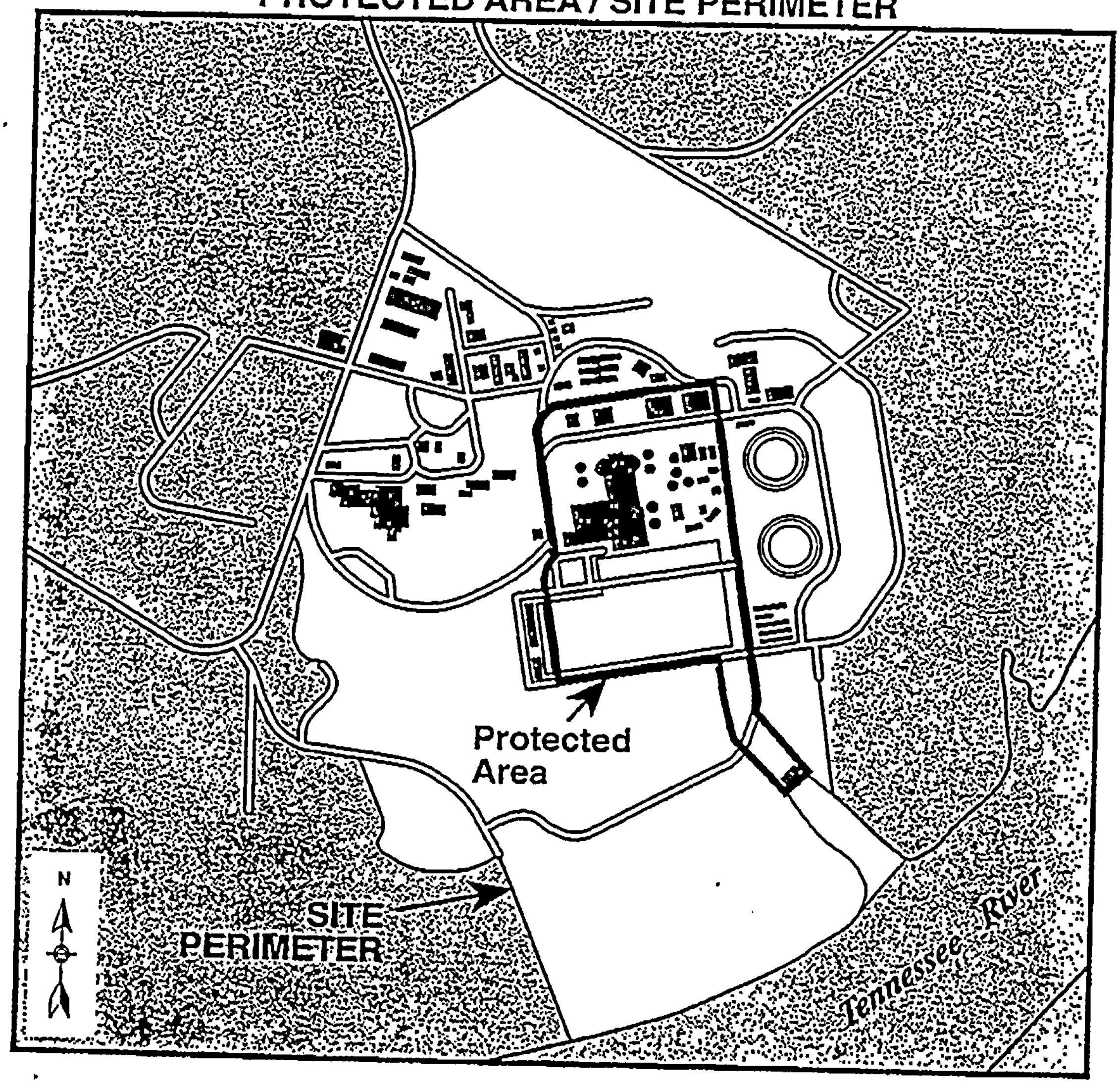
Section 5.0	**DESTRUCTIVE PHENOMENON**
Event 5.2	**TORNADO**
Classification	**ALERT**
Mode	All
Description	**Tornado or High Winds strikes any structure listed in Table 5-1 and results in VISIBLE DAMAGE** (1 and 2) 1. Tornado or High Winds (Sustained >80 mph > one minute) strikes any structure listed in Table 5-1. 2. (a or b) a. Confirmed report of any VISIBLE DAMAGE b. Control Room indications of degraded Safety System or component response due to event *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.*
Basis	Tornados or high winds striking the structures listed in Table 5-1 can cause damage to plant structures or systems needed for Safe Shutdown of the Plant. 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. 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. Table 5-1 Plant Structures Associated With Tornado/Hi Wind and Aircraft EALs 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

Section 5.0	**DESTRUCTIVE PHENOMENON**
Event 5.2	**TORNADO**
Classification	**ALERT** (continued)
Mode	All
Basis (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).
Escalation	Escalation of this event will be based on Fission Product Barriers Challenges.
References	NUMARC/NESP-007, HAI, Rev. 2, 1/92 FSAR 1.2 General Plant Description FSAR 2.3 Meteorology

Section 5.0	**DESTRUCTIVE PHENOMENON**
Event 5.2	**TORNADO**
Classification	**UNUSUAL EVENT**
Mode	All
Description	**Tornado Within The SITE PERIMETER** 1. Plant personnel report a Tornado has been Sighted within the **SITE PERIMETER** (Refer to Figure 5-A)
Basis	A tornado touchdown 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.
Escalation	Escalation of this event will be based on the tornado striking plant structures or high sustained winds within the protected area.
References	NUMARC/NESP-007, HUI, Rev. 2, 1/92 FSAR 1.2 General Plant Description FSAR 2.3 Meteorology Figure 5-A

Figure 5-A

PROTECTED AREA / SITE PERIMETER



Section 5.0	**DESTRUCTIVE PHENOMENON**
Event 5.3	**AIRCRAFT/PROJECTILE CRASH**
Classification	**GENERAL EMERGENCY**
Mode	Not Applicable
Description	Refer to the "Fission Product Barrier Matrix"
Basis	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.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev. 2, 1/92

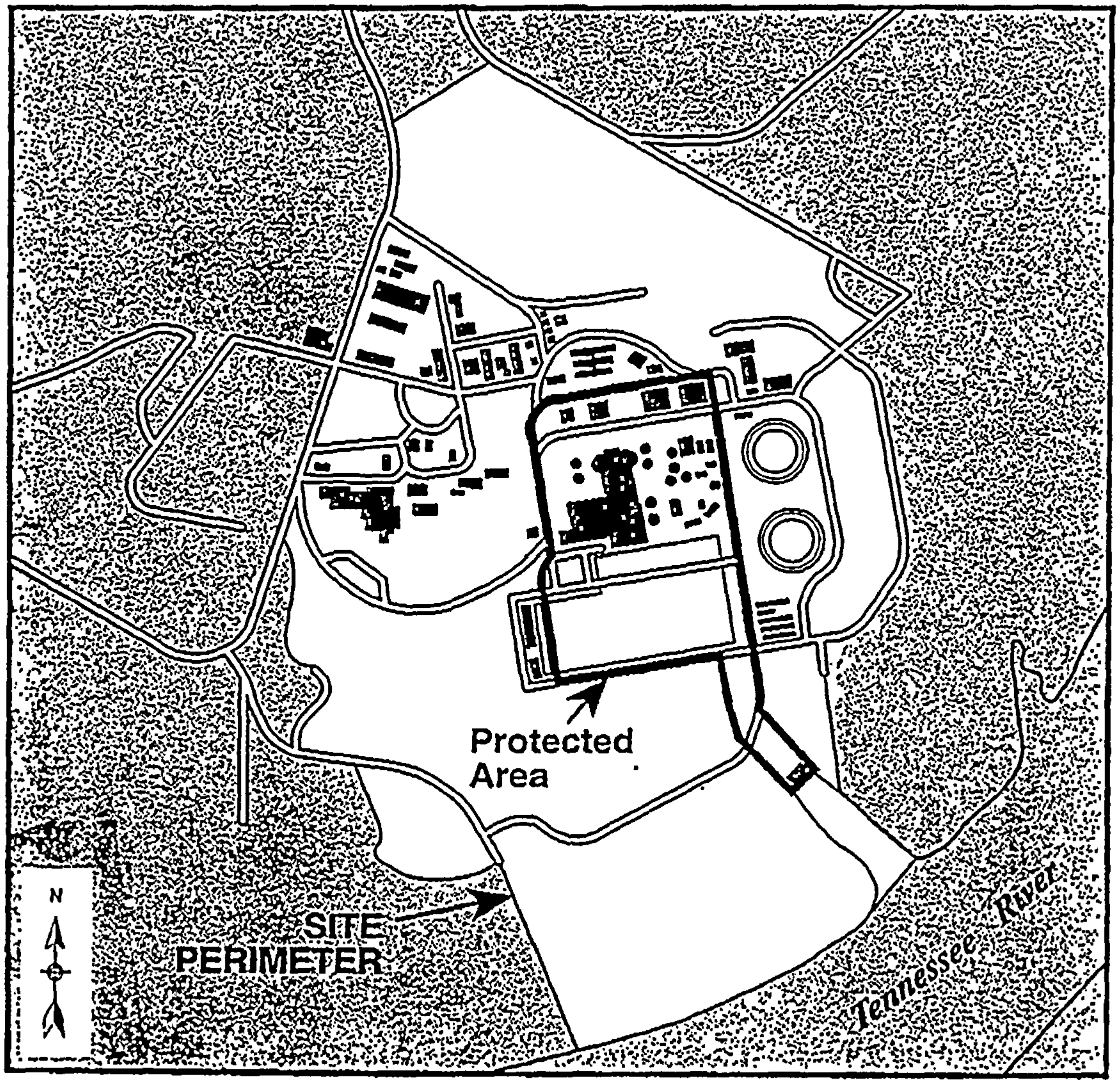
Section 5.0	**DESTRUCTIVE PHENOMENON**
Event 5.3	**AIRCRAFT/PROJECTILE CRASH**
Classification	**SITE AREA EMERGENCY**
Mode	Not Applicable
Description	Refer to the "Fission Product Barrier Matrix"
Basis	The basis for a Site Area Emergency for this event is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.
Escalation	Escalation will be based on "Fission Product Barrier Challenges".
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 5.0	**DESTRUCTIVE PHENOMENON**
Event 5.3	**AIRCRAFT/PROJECTILE CRASH**
Classification	ALERT
Mode	All
Description	**Aircraft <u>or</u> PROJECTILE impacts (Strikes) any Plant structure Listed in Table 5-1 resulting in VISIBLE DAMAGE (1 and 2)** 1. Plant personnel report aircraft <u>or</u> PROJECTILE has impacted any structure listed in Table 5-1 2. (a or b) 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.
Basis	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). 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. Table 5-1 Plant Structures Associated With Tornado/Hi Wind and Aircraft EALs 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
Escalation	Escalation to this event will be based on "Fission Product Barriers Challenges"..
References	NUMARC/NESP-007, HA1, HA2, Rev. 2, 1/92 FSAR 2.2 Nearby Industrial, Transportation And Military Facilities Table 5-1

Section *5.0*	DESTRUCTIVE PHENOMENON
Event *5.3*	AIRCRAFT/PROJECTILE CRASH
Classification	UNUSUAL EVENT
Mode	All
Description	**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)
Basis	Aircraft or PROJECTILE Impacts within the SITE PERIMETER are off normal events that can indicate a potential degradation of the level of safety of the plant. 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. SITE PERIMETER encompasses all owner controlled areas in the immediate site *environs as shown on Figure 5-A and 7-A. 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.
Escalation	Escalation to this event will be based on an Impact on plant structures or barriers.
References	NUMARC/NESP-007, HU1, Rev. 2, 1/92 FSAR 2.2 Nearby Industrial, Transportation And Military Facilities

Figure 5-A

PROTECTED AREA / SITE PERIMETER



Section 5.0	**DESTRUCTIVE PHENOMENON**
Event 5.4	**RIVER LEVEL HIGH**
Classification	**GENERAL EMERGENCY**
Mode	Not Applicable
Description	Refer to the "Fission Product Barrier Matrix"
Basis	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.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 5.0	**DESTRUCTIVE PHENOMENON**
Event 5.4	**RIVER LEVEL HIGH**
Classification	**SITE AREA EMERGENCY**
Mode	Not Applicable
Description	Refer to the "Fission Product Barrier Matrix"
Basis	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.
Escalation	Escalation will be based on "Fission Product Barrier Challenges".
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 5.0	DESTRUCTIVE PHENOMENON
Event 5.4	RIVER LEVEL HIGH
Classification	ALERT
Mode	All
Description	River Reservoir level is at Stage II Flood Warning (1 or 2) 1. River Reservoir level >727 Ft 2. Stage II Flood Warning (AOI-7) has been issued by River Operations
Basis	The requirements for flood protection ensures that facility protective actions will be taken and operation will be terminated in the event of flood conditions. A 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. 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.
Escalation	Escalation of this event will be based on "Fission Product Barriers Challenges".
References	NUMARC/NESP-007, HA1, Rev. 2, 1/92 FSAR 2.4 Hydrologic Engineering T.R. 3.7.2 Flood Protection Plan

Section *5.0*	DESTRUCTIVE PHENOMENON
Event *5.4*	RIVER LEVEL HIGH
Classification	UNUSUAL EVENT
Mode	All
Description	**River Reservoir level is at Stage I Flood Warning** (1 or 2 or 3) 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
Basis	The requirements for flood protection ensures that facility protective actions will be taken and operation will be terminated in the event of flood conditions. A 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 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. 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. Because of its inland location, the Watts Bar plant is not endangered by tsunami flooding.
Escalation	Escalation of this event will be based on river level being at Stage II Flood Warning.
References	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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Section *5.0*	**DESTRUCTIVE PHENOMENON**
Event *5.5*	**RIVER LEVEL LOW**
Classification	**GENERAL EMERGENCY**
Mode	Not Applicable
Description	Refer to the "Fission Product Barrier Matrix"
Basis	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.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev. 2, 1/92

Section *5.0*	**DESTRUCTIVE PHENOMENON**
Event *5.5*	**RIVER LEVEL LOW**
Classification	**SITE AREA EMERGENCY**
Mode	Not Applicable
Description	Refer to the "Fission Product Barrier Matrix"
Basis	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.
Escalation	Escalation will be based on "Fission Product Barrier Challenges".
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 5.0	**DESTRUCTIVE PHENOMENON**
Event 5.5	**RIVER LEVEL LOW**
Classification	**ALERT**
Mode	All
Description	**River Reservoir level is <668 Ft (AOI-22) as reported by River Operations**
Basis	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. 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 inland location on a relatively small, narrow lake, low water levels resulting from surges, seiches, or tsunamis are not a potential problem.
Escalation	Escalation to this event will be based on "Fission Product Barrier Challenges."
References	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

Section 5.0	**DESTRUCTIVE PHENOMENON**
Event 5.5	**RIVER LEVEL LOW**
Classification	**UNUSUAL EVENT**
Mode	All
Description	**River Reservoir level is ≤673 Ft (AOI-22) as reported by River Operations**
Basis	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. 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.
Escalation	Escalation to this event will be based on reduced river levels.
References	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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Section *5.0*	**DESTRUCTIVE PHENOMENON**
Event *5.6*	**WATERCRAFT CRASH**
Classification	**GENERAL EMERGENCY**
Mode	Not Applicable
Description	Refer to the "Fission Product Barrier Matrix"
Basis	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.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev. 2, 1/92

Section *5.0*	**DESTRUCTIVE PHENOMENON**
Event *5.6*	**WATERCRAFT CRASH**
Classification	**SITE AREA EMERGENCY**
Mode	Not Applicable
Description	Refer to the "Fission Product Barrier Matrix"
Basis	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.
Escalation	Escalation will be based on "Fission Product Barrier Challenges".
References	NUMARC/NESP-007, Rev. 2, 1/92

Section *5.0*	**DESTRUCTIVE PHENOMENON**
Event *5.6*	**WATERCRAFT CRASH**
Classification	**ALERT**
Mode	Not Applicable
Description	Refer to the "Fission Product Barrier Matrix"
Basis	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.
Escalation	Escalation will be based on "Fission Product Barrier Challenges".
References	NUMARC/NESP-007, Rev. 2, 1/92

Section *5.0*	**DESTRUCTIVE PHENOMENON**
Event *5.6*	**WATERCRAFT CRASH**
Classification	**UNUSUAL EVENT**
Mode	All
Description	**Watercraft Strikes the Intake Pumping Station resulting in a reduction of Essential Raw Cooling Water (ERCW) or Raw Cooling Water (RCW) (1 and 2)** 1. Plant personnel report a Watercraft has struck the Intake Pumping Station 2. (a or b or c) a. ERCW Supply Header Pressure Train A O-PI-67-18A is <15 psig b. ERCW Supply Header Pressure Train B O-PI-67-17A is <15 psig c. RCW Supply Header Pressure 0-PI-24-22 is <15 psig
Basis	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.
Escalation	Escalation would be based on "Fission Product Barrier Challenges".
References	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

Section *6.0*	**SHUTDOWN DEGRADATION**
Event *6.1*	**LOSS OF SHUTDOWN SYSTEMS**
Classification	**GENERAL EMERGENCY**
Mode	5,6
Description	**Note: Additional Information will be provided later pending NRC/NUMARC guidance on Shutdown EALs.** **Refer to Gaseous Effluents (7.1)**
Basis	Pending
Escalation	Not Applicable
References	NRC/NUMARC, future guidance

Section 6.0	**SHUTDOWN DEGRADATION**
Event 6.1	**LOSS OF SHUTDOWN SYSTEMS**
Classification	**SITE AREA EMERGENCY**
Mode	5,6
Description	**Loss of water level in the Rx vessel that has <u>or</u> will uncover fuel in the Rx vessel with CNTMT closure established. (1 and 2 and 3 and 4 and 5)** 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 5. CNTMT closure is established Note: If CNTMT, open, refer to Gaseous Effluents" (7.1)
Basis	For WBN, this IC is based on concems 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 uncovery 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 uncovery 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. The Rx vessel level indication of el 718' represents the water level at the hot leg center line.
Escalation	Escalation to this event will be based on Gaseous Effluent (7.2)
References	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

Section 6.0	**SHUTDOWN DEGRADATION**
Event 6.1	**LOSS OF SHUTDOWN SYSTEMS**
Classification	**ALERT**
Mode	**5,6**
Description	**Inability to maintain unit in Cold Shutdown** (1 and 2 and 3) 1. RHR capability is not available for RCS cooling 2. Incore TCs (if available) indicate RCS temp > 200° F. 3. CNTMT closure is established.
Basis	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. 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. 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 uncovery can occur. NRC analyses show that these sequences can cause core uncovery in 15 to 20 minutes and severe core damage within an hour after decay heat removal is lost. Under these conditions, RCS integrity is lost and fuel clad integrity is lost or potentially lost, which is consistent with 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. The inability to achieve this condition warrants declaration of an Alert.
Escalation	Loss of water level in the reactor vessel that has or will uncover fuel in the vessel will escalate this event.
References	NUMARC/NESP-007, SA3, Rev. 2, 1/92 AOI-14 Loss of RHR T.S. 3.1.2 Shutdown Margin Tavg. ≤ 200 F. Generic Letter 88-17 "Loss of Decay Heat Removal"

Section *6.0*	**SHUTDOWN DEGRADATION**
Event *6.1*	**LOSS OF SHUTDOWN SYSTEMS**
Classification	**UNUSUAL EVENT**
Mode	5,6
Description	**Note: Additional information will be provided later pending NRC/NUMARC Guidance on Shutdown EALs**
Basis	Pending
Escalation	Not Applicable
References	NRC/NUMARC, Future guidance

Section *6.0*	**SHUTDOWN SYSTEM DEGRADATION**
Event *6.2*	**LOSS OF AC (Shutdown)**
Classification	**GENERAL EMERGENCY**
Mode	Not Applicable
Description	Not Applicable
Basis	Loss of AC Power in Mode 5 and 6 will Not cause a declaration of a General Emergency
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev 2, 1/92

Section *6.0*	**SHUTDOWN SYSTEM DEGRADATION**
Event *6.2*	**LOSS OF AC (Shutdown)**
Classification	**SITE AREA EMERGENCY**
Mode	Not Applicable
Description	Not Applicable
Basis	Loss of AC Power in Mode 5 and 6 will Not cause a declaration of a Site Area Emergency.
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev 2, 1/92

Section *6.0*	**SHUTDOWN SYSTEM DEGRADATION**
Event *6.2*	**LOSS OF AC** (Shutdown)
Classification	**ALERT**
Mode	5,6, or Defuel
Description	**UNPLANNED loss of Offsite <u>And</u> Onsite AC Power for >15 Minutes** 1. 1A <u>and</u> 1B 6.9KV Shutdowns Bds de-energized for >15 minutes
Basis	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. Unplanned is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities. Fifteen (15) minutes was selected as threshold to exclude transient or momentary power losses.
Escalation	Escalation is not applicable from this event.
References	NUMARC/NESP-007, SU1 (expanded), Rev 2, 1/92 T.S. 3.8.2 AC Sources - Shutdown T.S. 3.8.10 Distribution Systems - Shutdown

Section 6.0	**SHUTDOWN SYSTEM DEGRADATION**
Event 6.2	**LOSS OF AC (Shutdown)**
Classification	**UNUSUAL EVENT**
Mode	5,6 or Defuel
Description	**UNPLANNED loss of Offsite Power for >15 minutes (1 and 2)** 1. C and D CSSTs not available for >15 minutes 2. Either Diesel Generator is supplying power to its respective Shutdown board.
Basis	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). Unplanned is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities. Fifteen (15) minutes was selected as a threshold to exclude transient or momentary power losses.
Escalation	Loss of one additional power supply to the shutdown boards will escalate this event.
References	NUMARC/NESP-007, SU1, Rev 2, 1/92 T.S. 3.8.2 AC Sources - Shutdown T.S. 3.8.10 Distribution Systems - Shutdown

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Section	*6.0*	**SHUTDOWN SYSTEM DEGRADATION**
Event	*6.3*	**LOSS OF DC (SHUTDOWN)**
Classification		**GENERAL EMERGENCY**
Mode		Not Applicable
Description		Not Applicable
Basis		Loss of DC Power (Shutdown) in Modes 5 and 6 will not cause a General Emergency.
Escalation		Not Applicable
References		NUMARC/NESP-007, Rev 2, 1/92

Section	*6.0*	**SHUTDOWN SYSTEM DEGRADATION**
Event	*6.3*	**LOSS OF DC (SHUTDOWN)**
Classification		**SITE AREA EMERGENCY**
Mode		Not Applicable
Description		Not Applicable
Basis		Loss of DC power (Shutdown) in Modes 5-6 will not cause a Site Area Emergency.
Escalation		Not Applicable
References		NUMARC/NESP-007, Rev. 2, 1/92

Section 6.0	**SHUTDOWN SYSTEM DEGRADATION**
Event 6.3	**LOSS OF DC (SHUTDOWN)**
Classification	**ALERT**
Mode	Not Applicable
Description	Not Applicable
Basis	Loss of DC power (Shutdown) in Modes 5-6 will not cause an Alert.
Escalation	Not Applicable
References	NUMARC/NESP-007 Rev. 2, 1/92

Section 6.0	**SHUTDOWN SYSTEM DEGRADATION**
Event 6.3	**LOSS OF DC (SHUTDOWN)**
Classification	**UNUSUAL EVENT**
Mode	5,6, defueled
Description	**UNPLANNED loss of the required train of DC power for >15 minutes (1 or 2)** 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
Basis	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. Unplanned is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities. 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. The fifteen minute threshold is utilized to exclude transient or momentary power losses.
Escalation	Not Applicable
References	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-21.1-21.8 Loss of 125V DC Vital Battery Boards

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Section 6.0	**SHUTDOWN SYSTEM DEGRADATION**
Event 6.4	**FUEL HANDLING**
Classification	**GENERAL EMERGENCY**
Mode	Not Applicable
Description	Refer to Gaseous Effluents (event 7.1)
Basis	The basis for a General Emergency is primarily the extent and severity of Gaseous Effluents (event 7.1)
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 6.0	**SHUTDOWN SYSTEM DEGRADATION**
Event 6.4	**FUEL HANDLING**
Classification	**SITE AREA EMERGENCY**
Mode	Not Applicable
Description	Refer to Gaseous Effluents (event 7.1)
Basis	The basis for a Site Area Emergency is primarily the extent and severity of Gaseous Effluents (event 7.1)
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev. 2, 1/92

Section *6.0*	**SHUTDOWN SYSTEM DEGRADATION**
Event *6.4*	**FUEL HANDLING**
Classification	**ALERT**
Mode	All
Description	**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) 1. **VALID** alarm on 0-RE-90-101 <u>or</u> 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) 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
Basis	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. 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. 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.
Escalation	Escalation would occur by offsite dose rates. See Gaseous Effluents (7.1)
References	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

Section 6.0	**SHUTDOWN SYSTEM DEGRADATION**
Event 6.4	**FUEL HANDLING**
Classification	**UNUSUAL EVENT**
Mode	All
Description	**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) 1. Plant personnel report water level drop in Spent Fuel Pool <u>or</u> Reactor Cavity, <u>or</u> Transfer Canal 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 3. Fuel remains covered with water
Basis	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. Unplanned is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities. 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. 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.
Escalation	Escalation of this event would be based on uncovering an irradiated fuel assembly or indications of high radiation levels on the refueling floor.
References	NUMARC/NESP-007, AU2, 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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Section 7.0	**RADIOLOGICAL**
Event 7.1	**GASEOUS EFFLUENTS**
Classification	**GENERAL EMERGENCY**
Mode	All
Description	**EAB 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) 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 Not exceeded 2. Field survey results indicate >1000 mrem/hr β-γ or an I-131 concentration of 3.9E-6 μCi/cc at SP 3. EP dose assessment results indicate EAB dose >1000 mrem TEDE or >5000 mrem Thyroid CDE for the actual or projected duration of the release (Figure 7-A)
Basis	Calculation To calculate the General Emergency gaseous effluent monitor values for Table 7-1, the release rates for the determination of General Emergency are calculated in the same manner as for the Site Area Emergency. The General Emergency release rate will be equal to 10 times the release rate used for the Site Area Emergency. To calculate the General Emergency gaseous effluent monitor values for Table 7-1, the values calculated for SAE are multiplied by a factor of 10.

Release Point	Monitor	Units	
U1 Shield Bldg	1-RE-90-400	μCi/s	1.08E+09
		μCi/cc	1.93E+03
U2 Shield Bldg	2-RE-90-400	μCi/s	2.50E+08
		μCi/cc	4.46E+02
Aux Bldg	0-RE-90-101	cpm	2.07E+08
Service Bldg	0-RE-90-132	cpm	9.84E+07
U1 CVE	1-RE-90-404	μCi/cc	8.83E+02

Section 7.0	**RADIOLOGICAL**
Event 7.1	**GASEOUS EFFLUENTS**
Classification	**GENERAL EMERGENCY** (continued)
Mode	All
Basis (continued)	<u>Steam Generator Discharge Monitors</u> 1-RE-90-421 through 424, 3.5 E+04 mr/Hr See Site Area Emergency gaseous effluent basis on page 171 for explanation. 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. The EXCLUSION AREA BOUNDARY (EAB) is the demarcation of the area surrounding the WBN units in which postulated FSAR accidents will not result in population doses exceeding the criteria of 10 CFR Part 100. Refer to Figure 7-A.
Escalation	Not Applicable
References	NUMARC/NESP-007, AGI, 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 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

Section 7.0	**RADIOLOGICAL**
Event 7.1	**GASEOUS EFFLUENTS**
Classification	**SITE AREA EMERGENCY**
Mode	All
Description	**EAB 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) 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 Not exceeded 2. Field survey results indicate >100 mrem/hr β-γ or 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 or >500 mrem Thyroid CDE for the actual or projected duration of the release. Refer to Figure 7-A.
Basis	Calculation To calculate the SAE gaseous effluent monitor values for Table 7-1, the release rates that are calculated to determine monitor readings are those required to deliver the EAL dose in one hour. To perform the calculation, the mix fraction in the WBN FSAR Chapter 11 and the dose factors from EPA-400 (Manual of Protective Action Guides and Protective Actions for Nuclear Incidents) Tables 5.1 and 5.2 are used in conjunction with the annual average meteorology found in the WBN ODCM. EAL release rates are backcalculated from both the 100 mrem TEDE and 500 mrem Thyroid CDE criteria separately. The most conservative of these release rates will be used in the determination of monitor readings.

Section 7.0	**RADIOLOGICAL**
Event 7.1	**GASEOUS EFFLUENTS**
Classification	**SITE AREA EMERGENCY** (continued)
Mode	All
Basis (continued)	Calculation (continued)

First, an EAL release rate is backcalculated from a 100 mrem TEDE dose as follows:

	EPA-400 TEDE DCF	WBN FSAR (Ci/yr)	Mix Fraction	Mix x TEDE
Kr-85m	9.30E+01	2.57E+01	4.74E-03	4.41E-01
Kr-85	1.30E+00	6.99E+02	1.29E-01	1.68E-01
Kr-87	5.10E+02	1.62E+01	2.99E-03	1.52E+00
Kr-88	1.30E+03	3.84E+01	7.08E-03	9.29E+00
Kr-89	1.20E+03		0.00E+00	0.00E+00
Xe-131m	4.90E+00	1.19E+03	2.19E-01	1.08E+00
Xe-133m	1.70E+01	4.88E+01	9.00E-03	1.53E-01
Xe-133	2.00E+01	3.20E+03	5.90E-01	1.18E+01
Xe-135m	2.50E+02	8.51E+00	1.57E-03	3.92E-01
Xe-135	1.40E+02	1.85E+02	3.41E-02	4.78E+00
Xe-137	1.10E+02	1.54E+00	2.84E-04	3.12E-02
Xe-138	7.20E+02	7.65E+00	1.41E-03	1.02E+00
I-131	5.30E+04	1.53E-01	2.82E-05	1.50E+00
I-132	4.90E+03	6.74E-01	1.24E-04	6.09E-01
I-133	1.50E+04	4.58E-01	8.44E-05	1.27E+00
I-134	3.10E+03	1.08E+00	1.99E-04	6.17E-01
I-135	8.10E+03	8.45E-01	1.56E-04	1.26E+00
Total		5.42E+03	1.00E+00	3.58E+01

EAL = 0.1/(35.8 rem/h per μCi/cc x 1.09E-05 sec/m^3 x 1E-06 m^3/cc)

EAL = 2.45E+08 μCi/s

Section 7.0	RADIOLOGICAL
Event 7.1	GASEOUS EFFLUENTS
Classification	SITE AREA EMERGENCY (continued)
Mode	All
Basis (continued)	Calculation (continued)

A second calculation is performed to determine a noble gas release which will correspond to the associated iodine release rate resulting in 500 mrem Thyroid CDE.

	Thyroid CDE DCF	WBN FSAR (Ci/yr)	Mix Fraction	Mix x CDE
I-131	5.30E+04	1.53E-01	4.77E-02	2.53E+03
I-132	4.90E+03	6.74E-01	2.10E-01	1.03E+03
I-133	1.50E+04	4.58FE-01	1.43E-01	2.14E+03
I-143	3.10E+03	1.08E+00	3.36E-01	1.04E+03
I-135	8.10E+03	8.45E-01	2.63E-01	2.13E+03
Total		3.21E+00	1.00E+00	8.91E+03

Iodine EAL = 0.5/(8910 rem/h per μCi/cc x 1.09E-05 sec/m^3 x 1E-06 m^3/cc)
Iodine EAL = 5.15E+06 μCi/s

To determine the associated noble gas release rate, the iodine release rate is divided by a factor of 0.001. This is the standard ratio of iodine to noble gas realease rate which is used in CECC procedures for dose assessment and is conservative with repsect to the WBN FSAR nuclide distribution. Thus, the corresponding noble gas release rate is:

NG EAL = 5.15E+06/.0.001 = 5.15E+09 μCi/s

The mose conservative of the calculated noble gas release rates is used for the SAE determination; therefore, the SAE noble gas release rate value will be 2.56E+08 μCi/s

Section 7.0	**RADIOLOGICAL**
Event 7.1	**GASEOUS EFFLUENTS**
Classification	**SITE AREA EMERGENCY** (continued)
Mode	All
Basis (continued)	Calculation (continued) Using the methodology presented in the NOUE section for the determination of monitor values associated with the site release limit:

Release Point	Monitor	Units	
U1 Shield Bldg	1-RE-90-400	µCi/s	1.08E+08
		µCi/cc	1.92E+02
U2 Shield Bldg	2-RE-90-400	µCi/s	2.50E+07
		µCi/cc	4.46E+01
Aux Bldg	0-RE-90-101	cpm	2.07E+07
Service Bldg	0-RE-90-132	cpm	9.84E+06
U1 CVE	1-RE-90-404	µCi/cc	8.83E+01

Section 7.0	**RADIOLOGICAL**
Event 7.1	**GASEOUS EFFLUENTS**
Classification	**SITE AREA EMERGENCY** (continued)
Mode	All

Basis (continued)

The EXCLUSION AREA BOUNDARY (EAB) is the demarcation of the area surrounding the WBN units in which postulated FSAR accidents will not result in population doses exceeding the criteria of 10 CFR Part 100. Refer to Figure 7-A.

Steam Generator Discharge Monitors
To determine the readings for the Steam Generator (SG) Discharge Monitors, 1-RE-90-421 through 424 in mr/Hr, for the four emergency classifications, the calculation utilized a flow rate of 970,000 lb/hr at 1185 psig and 600 F. This is considered the limit of one SG, PORV (ref. N3-1-4002).

The 970,000 lb/hr was recalculated to 3101889.6 cc/sec so that this value could be utilized with the calibration factor in the calculation. The curie limit associated with the Unusual Event was 2.9 E+05 µci/sec (ref. Table 7-1, Effluent Radiation Monitor EALs). By dividing this limit by the steam flow rate, a limit of 9.35 E-02 µci/cc, was calculated. The monitor calibration factor is

$$2.65\text{ E-02 } \frac{\mu ci/cc}{mr/Hr}$$

(ref. WBN APS3-047).

By dividing the limit of 9.35 E-02 µci/cc by the calibration factor, a monitor reading of 3.5 mr/Hr (N/A), was calculated for the Unusual Event.

By using the curie limit 2.9 E+07 and dividing this limit by the steam flow rate and by the calibration factor, a monitor reading of 3.5 E+02 mr/Hr, was calculated for the Alert.

By using the curie limit 3.0 E+08 and dividing this limit by the steam flow rate and by the calibration factor, a monitor reading of 3.5 E+03 mr/Hr, was calculated for the Site Area Emergency.

By using the curie limit 3.0 E+09 and dividing this limit by the steam flow rate and by the calibration factor, a monitor reading of 3.5 E+04 mr/Hr, was calculated for the General Emergency.

Section 7.0	RADIOLOGICAL
Event 7.1	GASEOUS EFFLUENTS
Classification	SITE AREA EMERGENCY (continued)
Mode	All
Basis (continued)	Service Generator Discharge Monitors (continued) 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.
Escalation	Escalation would be based on increased release rates by a factor of 10.
References	NUMRAC/NESP-007, AGI, Rev 2, 1/92 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

Section 7.0	**RADIOLOGICAL**
Event 7.1	**GASEOUS EFFLUENTS**
Classification	**ALERT**
Mode	All
Description	**Any UNPLANNED release of Gaseous Radioactivity that exceeds 200 times the ODCM Limit for >15 minutes (1 or 2 or 3)** 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 Not exceeded 2. Field survey results indicate >10 mrem/hr β-γ 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)
Basis	Calculation For an Alert, the gaseous effluent monitor values for Table 7-1 (Effluent Radiation Monitor EALs), are the monitor values which correspond to two hundred times the ODCM Dose Rate limit of 500 mrem/year. The monitor values calculated for the NOUE are tabulated below in the units appropriate for each monitor.

Release Point	Monitor	Units	
U1 Shield Bldg	1-RE-90-400	μCi/s	6.71E+04
		μCi/cc	2.78E-02
U2 Shield Bldg	2-RE-90-400	μCi/s	1/56E+04
		μCi/cc	3.67E-03
Aux Bldg	0-RE-90-101	cpm	1.29E+04
Service Bldg	0-RE-90-132	cpm	4.31E+03
U1 CVE	1-RE-90-404	μCi/cc	5.51E-02

The values corresponding to the Alert are one hundred (100) times the values in the above table:

Release Point	Monitor	Units	
U1 Shield Bldg	1-RE-90-400	μCi/s	6.71E+06
		μCi/cc	2.78E+00
U2 Shield Bldg	2-RE-90-400	μCi/s	1.56E+06
		μCi/cc	3.67E-01
Aux Bldg	0-RE-90-101	cpm	1.29E+06
Service Bldg	0-RE-90-132	cpm	4.31E+05
U1 CVE	1-RE-90-404	μCi/cc	5.51E+00

Section 7.0	RADIOLOGICAL
Event 7.1	GASEOUS EFFLUENTS
Classification	ALERT (continued)
Mode	All
Basis (continued)	<u>Steam Generator Dischage Monitors</u> 1-RE-90-421 through 424, 3.5 E + 02 mr/Hr See Site Area Emergency gaseous effluent basis on page 171 for explanation. 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. 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. The significance of the time factor to this CRITERION is primarily related to loss of control of radioactive material that has allowed the release to continue unabated for 15 minutes. It is this aspect rather than the magnitude of the release that establishes "...a potential substantial degradation in the level of safety of the plant..."--the fundamental definition of an ALERT.
Escalation	Escalation would be based on dose rates greater than 100 mrem Total Body or 500 mrem child thyroid.
References	NUMARC/NESP-007, AAI, 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 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

Section *7.0*	**RADIOLOGICAL**
Event *7.1*	**GASEOUS EFFLUENTS**
Classification	**UNUSUAL EVENT**
Mode	All
Description	**Any UNPLANNED release of Gaseous Radioactivity that exceeds 2 times the ODCM Limit for** >60 minutes (1 or 2 or 3) 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 β-γ 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)
Basis	<u>Calculation</u> For a NOUE, the gaseous effluent monitor values for Table 7-1 (Effluent Radiation Monitor EALs), are the monitor values which correspond to two times the ODCM Dose Rate limit of 500 mrem/year. These values are also two times the default setpoint values tabulated in TI-18. The TI-18 methodology is used below to determine the table values. The total site limits are divided among the plant discharge points using allocation factors. The TI-18 allocation factors are given below: CVE, U1 — 0.01; Service Building — 0:01 Shield Building, U1 — 0.32; Shield Building, U2 — 0.32 Auxiliary Building — 0.34 During the period of Unit 1 operation only, the U2 Shield Building Exhaust (SBE) allocation factor is reduced since there will be limited radioactive releases to be made through the U2 SBE. The U2 SBE allocation factor will be set to 0.1. The remaining 0.22 will be divided evenly between the U1 SBE and the Auxiliary Building Exhaust. The resulting allocation factors to be used during Unit 1 operation are given below: CVE, U1 — 0.01; Service Building — 0.01 Shield Building, U1 — 0.43; Shield Building, U2 — 0.1 Auxiliary Building — 0.45

Section 7.0	**RADIOLOGICAL**
Event 7.1	**GASEOUS EFFLUENTS**
Classification	**UNUSUAL EVENT** (continued)
Mode	All

Basis (continued)

A safety factor of 0.5 is assigned to each effluent monitor to account for the expected inaccuracy and to provide a margin of safety for the releases. This factor is conservative with respect to the accuracies specified in the Plant SSD or NE SSD for associated radiation monitors.

WBN ODCM Sections 7.1.1.3 and 7.1.1.4 provide guidance on the determination of setpoints for gaseous radiation monitors. The monitor values for the emergency classifications will be determined by calculating the maximum calculated setpoint described by ODCM Equation 7.3 using Xe-133 monitor efficiencies, design flow rates, and setting the ratio of the dose rate limit to the calculated dose rate equal to 1.0. ODCM Equation 7.3 is:

$$S_{max} = AF\ VCF\ SF \left\{ \left(\frac{DRlim}{DR} \right) (EB - B) \right\} + B$$

where:

- AF = dose rate allocation factor for the release point, dimensionless. The sum of all dose rate allocation factors must be ≤ 1.
- VCF = vacuum correction factor applied to noble gas monitors whose detector operates at a negative pressure.
- SF = safety factor for the monitor, dimensionless. Safety factors will be ≤ 1.
- DR_{Lim} = the dose rate limit, mrem/y.
- DR = the calculated dose rate for the release, mrem/y.
- ER = expected monitor response, cpm.
- B = the monitor background, cpm.

Section 7.0	**RADIOLOGICAL**
Event 7.1	**GASEOUS EFFLUENTS**
Classification	**UNUSUAL EVENT** (continued)
Mode	All

Basis (continued)

The vacuum correction factor used is defined (from Sorrento Electronics, "Calibration Report for RD-52 Offline Beta Detector, June 1986) as:

$$CF = \frac{29.92}{[(29.92-Pn)(1+PnAn)]}$$

where

P_n = negative pressure in the chamber (in inches of Hg vacuum)

A_n = 0.013 for Xe-133.

Using the lowest expected operating pressure for the detector (10 in Hg) into this equation yields:

$$CF = \frac{29.92}{[(29.92-10)(1+10x0.013)]} = 1.33$$

This correction factor is applied by dividing the detector response by the factor. Since this application uses VCF, which is multiplied by the detector response, the inverse of the CF is used. Thus, the VCF is equal to:

VCF = 0.75 for Xe-133

This correction factor is applied to all monitors considered in this calculation, with the exception of the Shield Building Exhaust monitor, which does not operate under negative pressure.

The expected monitor response (ER) is determined by ODCM Equation 7.1:

ER = B + E * C

where:

B = monitor background, cpm.

E = Xe-133 response factor for the monitor, cpm per μCi/cc, from TI-18, Appendix G.

C = measured concentration of Xe-133, μCi/cc.

The ODCM limit values, S_{lim}, are defined as a value above background, therefore the background will be set equal to zero for these calculations. Substituting the equation for ER into the equation, setting the dose rate ratio to 1.0, and dropping the background terms from the equation yields:

S_{lim} = AF SF VCF E C

Section 7.0	**RADIOLOGICAL**
Event 7.1	**GASEOUS EFFLUENTS**
Classification	**UNUSUAL EVENT** (continued)
Mode	All
Basis (continued)	The concentration of Xe-133 which corresponds to the site dose rate limit(s) must be calculated for each discharge point to insert into the equation. This concentration is determined by calculating the site release rate limit (in µCi/s) and then converting that limit to a concentration based on the maximum design flow rate for each discharge point. The site release rate limit is: $$Limit = \frac{DRlim}{(x/Q)\ DF}$$ where: DR_{lim} = the dose rate limit for the site, mrem/year = 500 for Total Body, 3000 for Skin X/Q = the unrestricted area boundary dispersion factor, s/m^3, from WBN ODCM Table 7.3. = 1.09E-05. DF = the noble gas submersion dose factor, mrem/year per $\mu Ci/m^3$, from ODCM Table 7.4, = 2.94E + 02 for Total Body, 3.06E + 02 for Skin Limit(total body) = 500 / (1.09E-05 x 2.94E + 02) = 1.56E + 05 µCi/s Limit(skin) = 3000 / (1.09E-05 x 3/06E + 02) = 8.99E + 05 µCi/s The ODCM limit value for the Shield Buildings' monitors (in units of µCi/s) is obtained by multiplying the site release rate limit by the applicable allocation factor and safety factor. For the remaining monitors, the most limiting concentration, that calculated for the total body dose, is then converted to the corresponding discharge point limiting concentration based on the discharge point's maximum flow rate: C_{lim} = 1.56E+05 / (Flow x 472) where: Flow = the maximum flow rate for the discharge point, cfm, from WBN FSAR 11.3.7.5, = 23,800 for the U1 Shield Building = 9,000 for the U2 Shield Building = 224,000 for the Auxiliary Building = 10,500 for the Service Building = 45 for the Condenser Vacuum Exhaust 472 = conversion factor from cfm to cc/s.

Section 7.0	**RADIOLOGICAL**
Event 7.1	**GASEOUS EFFLUENTS**
Classification	**UNUSUAL EVENT** (continued)
Mode	All

Basis (continued)

Thus,

C_{lim}(U1 Shield Bldg.) = 1.56E+05 µCi/s / (23,800 x 472) = 1.39E-02 µCi/cc
C_{lim}(U2 Shield Bldg.) = 1.56E+05 µCi/s / (9,000 x 472) = 3.67E-02 µCi/cc
C_{lim}(Auxiliary Bldg.) = 1.56E+05 µCi/s / (224,000 x 472) = 1.48E-03 µCi/cc
C_{lim}(Service Bldg.) = 1.56E+05 µCi/s / (14,950 x 472) = 2.21E-02 µCi/cc
C_{lim}(CVE) = 1.56E+05 µCi/s / (45 x 472) = 7.34E+00 µCi/cc

The limiting concentrations are then inserted into the equation, with the discharge point allocation factors, safety factors, vacuum correction factors, and Xe-133 monitor efficiencies to obtain the ODCM limit value in cpm for the Auxiliary and Service Building monitors.

The ODCM limit value for the Condenser Vacuum Exhaust (CVE) monitor (in units of µCi/cc) is obtained by multiplying the site release rate limit by the applicable allocation factor, safety factor, and vacuum correction factor.

The ODCM limit values for all the site discharge points which result from these calculations are tabulated below.

	U1 Shield Building	U2 Shield Building	Auxiliary Building	Service Building	CVE
Allocation Factor	0.43	0.1	0.43	0.01	0.01
Safety Factor	0.5	0.5	0.5	0.5	0.5
Vacuum Correction Factor	1.0	1.0	0.75	0.75	0.75
Xe-133 Efficiency (cpm per µCi/cc)	N/A	N/A	2.6E+07	2.6E+07	N/A
Allocated Release Rate (µCi/s)	6.71E+04	1.55E+04	7.02E+04	1.56E+03	1.56E+03
Limiting Release Rate (µCi/s)	3.35E+04	7.80E+03	N/A	N/A	N/A
Limiting Concent. (µCi/cc)	1.39E-02	1.84E-03	N/A	N/A	2.75E-02
Limiting (cpm)	N/A	N/A	6,500	2,100	N/A
ODCM Limit (monitor units)	6.71E+04 1.39E-02	7.80E+03 1.39E-02	6,500	2,100	2.75E-02

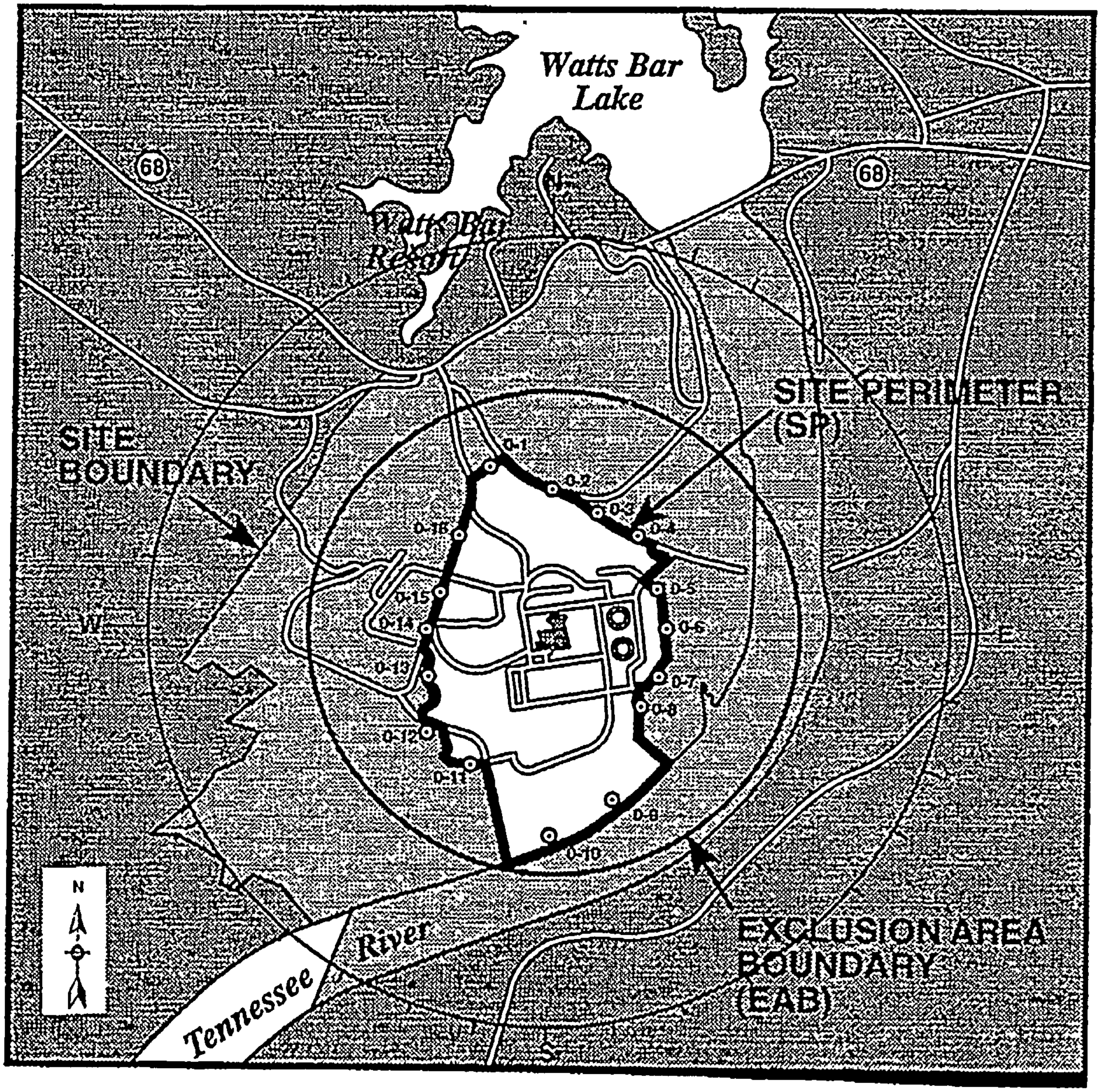
The values corresponding to the NOUE are twice the values in the above table:

Release Point	Monitor	Units	
U1 Shield Bldg	1-RE-90-400	µCi/s	6.71E+04
		µCi/cc	2.78E-02
U2 Shield Bldg	2-RE-90-400	µCi/s	1.56E+04
		µCi/cc	3.67E-03
Aux Bldg	0-RE-90-101	cpm	1.29E+04
Service Bldg	0-RE-90-132	cpm	4.31E+03
U1 CVE	1-RE-90-404	µCi/cc	5.51E-02

Section 7.0	RADIOLOGICAL
Event 7.1	GASEOUS EFFLUENTS
Classification	UNUSUAL EVENT (continued)
Mode	All
Basis (continued)	Steam Generator Discharge Monitors 1-RE-90-421 through 424 3.5 mr/Hr NOTE: It was decided not to reference this value, 3.5 mr/Hr, for these monitors at the Unusual Event classification. The value was considered too close to monitored background and an upscale reading on any of these monitors would not necessarily indicate an off-site release. For this reason, an N/A was placed in the effluent radiation monitor EAL table. See Site Area Emergency gaseous effluent basis for additional information. 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. 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. The significance of the time factor to this CRITERION is primarily related to loss of control of radioactive material that has allowed the release to continue unabated for 60 minutes. It is this aspect rather than the magnitude of the release that establishes "...a potential degradation in the level of safety of the plant..."--the fundamental definition of an UNUSUAL EVENT.
Escalation	Escalation would be based on increasing the magnitude of the release by a factor of 100.
References	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 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

Figure 7-A

EXCLUSION AREA/SITE BOUNDARY



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Section	*7.0*	**RADIOLOGICAL**
Event	*7.2*	**LIQUID EFFLUENTS**
Classification		**GENERAL EMERGENCY**
Mode		Not Applicable
Description		Not Applicable
Basis		Not Applicable
Escalation		Not Applicable
References		NUMARC/NESP-007, Rev 2, 1/92

Section	*7.0*	**RADIOLOGICAL**
Event	*7.2*	**LIQUID EFFLUENTS**
Classification		**SITE AREA EMERGENCY**
Mode		Not Applicable
Description		Not Applicable
Basis		Not Applicable
Escalation		Not Applicable
References		NUMARC/NESP-007, Rev 2, 1/92

Section 7.0	**RADIOLOGICAL**
Event 7.2	**LIQUID EFFLUENTS**
Classification	**ALERT**
Mode	All
Description	**Any UNPLANNED release of Liquid Radioactivity that exceeds 200 times the ODCM Limit for >15 minutes (1 or 2)** 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 2. Sample results exceed 200 times the ODCM limit value for an unmonitored release of liquid radioactivity >15 minutes in duration
Basis	For liquid releases, the ODCM limit is equal to 10 times the Effluent Concentration Limits (ECL) listed in 10 CFR 20 Appendix B, Table 2, Column 2. For this calculation, the most restrictive of the concentrations for nuclides which might be released from WBN is selected. The concentration used is that for Cs-134. ECL(Cs-134) = 9.0E-07 µCi/ml (NOTE: The values chosen for WBN are those found in the revised 10 CFR Part 20.) The Alert value is 200 times the ODCM limit: 10 * 9.0E-07 * 200 = 1.8E-03 µCi/ml This concentration must then be used to determine corresponding monitor readings. The monitor values for the Alert classification will be calculated in the same manner as the Unusual Event values, but will be a factor of 100 higher, thus the values are as given below: RE-90-225 1.80E+08 cpm (maximum reading for the monitor is 1.0E7) 1,2-RE-90-120,121 1.04E+08 cpm RE-90-122 1.11E+08 cpm (maximum reading for the monitor is 1.0E7) RE-90-212 1.56E+06 cpm 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.

Section 7.0	RADIOLOGICAL
Event 7.2	LIQUID EFFLUENTS
Classification	ALERT (continued)
Mode	All
Basis (continued)	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. The significance of the time factor to this Criterion is primarily related to loss of control of radioactive material that has allowed the release to continue unabated for 15 minutes. It is this aspect rather than the magnitude of the release that establishes "...a potential substantial degradation in the level of safety of the plant..."--the fundamental definition of an ALERT.
Escalation	Not Applicable
References	NUMARC/NESP-007, AA2, Rev 2, 1/92 (ODCM) Offsite Dose Calculation Manual 10 CFR 20 TI-18 Calculation Methods for Effluent Radiation Monitors

Section 7.0	RADIOLOGICAL
Event 7.2	LIQUID EFFLUENTS
Classification	UNUSUAL EVENT
Mode	All
Description	**Any UNPLANNED release of Liquid Radioactivity to the Environment that exceeds 2 times the ODCM Limit for** >60 minutes (1 or 2) 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 2. Sample results exceed 2 times the ODCM limit value for an unmonitored release of liquid radioactivity >60 minutes in duration
Basis	For the UNUSUAL EVENT 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. For this calculation, the most restrictive of the concentrations for nuclides which might be released from WBN is selected. The concentration used is that for Cs-134. ECL(Cs-134) = 9.0E-07 µCi/ml NOTE: The values chosen for WBN are those found in the revised 10 CFR Pa 20. The Unusual Event value is twice the ODCM limit: 10 * 9.0E-07 * 2 = 1.8E-05 µCi/ml The Unusual Event site liquid release concentration is equal to: 1.8E-05 µCi/ml

Section 7.0	**RADIOLOGICAL**
Event 7.2	**LIQUID EFFLUENTS**
Classification	**UNUSUAL EVENT (continued)**
Mode	All
Basis (continued)	This concentration must then be used to determine corresponding monitor readings. The monitor readings are determined using the most limiting conditions for conservatism. A minimum cooling tower blowdown of 20,000 gpm, and the maximum undiluted effluent flow is assumed. The monitor response for each release point monitor is determined for Cs-134. The calculation assumes the unplanned release is the only release in progress; any combination of simultaneous releases will require case by case assessment. <u>Condensate Demineralizer Radiation Monitor</u> RE-90-225 Using the relationship defined in equations 6.1 and 6.2 from Section 6.1.2 of the WBN ODCM, the monitor response can be predicted. $f_1 R_1 / F \leq 1$ 180 gpm R_1/20,000 gpm = 1.0 R_1 = 111.11 $R_j = C_i / ECL_i$ Substituting R_1 for R_j (10 times the ECL for Cs-134): 111.11 = C_i /9E-06 µCi/ml C_i = 1.0E-03 µCi/ml The monitor response is given by : $R = B + Eff_i\, C_i$

Section 7.0	RADIOLOGICAL
Event 7.2	LIQUID EFFLUENTS
Classification	UNUSUAL EVENT (continued)
Mode	All
Basis (continued)	Substituting values for the monitor efficiency (obtained from TI-18, Appendix H, a typical monitor background of 500 cpm, and C_i as determined previously, gives the predicted monitor response: R = 500 cpm + (8.79E+08 cpm/µCi/ml * 1E-03 µCi/ml) R = 8.8E+05 cpm The Unusual Event value is 2 times this, therefore the Unusual Event = 1.8E+06 cpm <u>Steam Generator Blowdown Radiation Monitors</u> (1,2-RE-90-120,-121) The same methodology is followed for these monitors as is given in the previous description, using a maximum undiluted effluent flow rate of 320 gpm. The allowed concentration C_1 is determined to be 5.63 E-04 µCi/ml. The monitor response will be: R = 500 cpm + (9.21E+08 * 5.63E-04) R = 5.2E+05 The Unusual Event value is 2 times 5.2E+05 cpm or 1.04E+06 cpm. <u>Radwaste Radiation Monitor</u> (O-RE-90-122) The same methodology is followed for this monitor as is given in the previous description, using a maximum undiluted effluent flow rate of 300 gpm. The allowed concentration C_1 is determined to be 6.0E-04 µCi/ml. The monitor response will be: R = 500 cpm + (9.21E+08 * 6.0E-04) R = 5.53E+05 The Unusual Event value is 2 * 5.53E+05 or 1.11E+06 cpm.

Section 7.0	**RADIOLOGICAL**
Event 7.2	**LIQUID EFFLUENTS**
Classification	**UNUSUAL EVENT** (continued)
Mode	All
Basis (continued)	<u>Turbine Building Sump Radiation Monitor</u> (RE-90-212) Since the flow from this release point is not diluted prior to being released, the undiluted Cs-134 concentration is inserted into the ODCM equation 6.3: R = 500 cpm + (9.21E+08 cpm/µCi/ml * 9E-06 µCi/ml) R = 7.79E+03 cpm The Unusual Event value is then: 2 * 7.79E+03 cpm = 1.56E+04 cpm 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. 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. The significance of the time factor to this CRITERION is primarily related to loss of control of radioactive material that has allowed the release to continue unabated for 60 minutes. It is this aspect rather than the magnitude of the release that establishes "...a potential degradation in the level of safety of the plant..."--the fundamental definition of an UNUSUAL EVENT.
Escalation	Escalation would be based on an UNPLANNED release exceeding 200 times the ODCM limit for greater than 15 minutes.
References	NUMARC/NESP-007, AU2, Rev 2, 1/92 (ODCM) Offsite Dose Calculation Manual 10 CFR 20 TI-18 Calculation Methods for Effluent Radiation Monitors

TABLE 7-1
EFFLUENT RADIATION MONITOR EALS

NOTE: The values below, if exceeded, indicate the need to perform the specified assessment. If the assessment can not be completed within 15 minutes (60 minutes for UE), the declaration shall be made based on the VALID reading. As used here, the radiation monitor indications as displayed on ICS are the primary indicators. If ICS is unavailable, utilize the radiation monitor readings in the control room or local indication as necessary.

Monitor	ICS Screen	Units	UE	Alert	Site	General
Total Site	EFF1	µCi/s (2)	1.5E+05	1.5E+07	2.5E+08	2.5E+09
U1 Shield Building 1-RE-90-400	EFF1	µCi/s	6.7E+04	6.7E+06	1.0E+08	1.0E+09
U2 Shield Building 2-RE-90-400	EFF1	µCi/s	1.5E+04	1.5E+06	2.5E+07	2.6E+08
Auxiliary Building 0-RE-90-101B	4RM1	cpm	1.2E+04	1.2E+06	*****(1)	*****(1)
Service Building 0-RE-90-132B	4RM1	cpm	4.3E+03	4.3E+05	9.8E+06	*****(1)
U1 Condenser Vacuum Exhaust						
1-RE-90-404A	3PAM	µCi/cc(3)	5.5E-02	5.5E+00	8.83E+01	8.83E+02
1-RE-90-404B	3PAM	µCi/cc	5.5E-02	5.5E+00	8.83E+01	8.83E+02
S/G Discharge Monitors						
1-RE-90-421 thru	4RM2					
424	4RM2	mR/hr (4)	NA	3.5E+02	3.5E+03	3.5E+04
Liquid Monitors	N/A	µCi/ml (2)	*1.8E-05	1.8E-03	N/A	N/A
0-RE-90-122	4RM2	cpm	1.1E+06	*****(1)	N/A	N/A
1-RE-90-120,121	4RM2	cpm	1.0E+06	*****(1)	N/A	N/A
0-RE-90-225	4RM2	cpm	9.2E+05	*****(1)	N/A	N/A
0-RE-90-212	4RM2	cpm	1.5E+04	1.5E+06	N/A	N/A
RELEASE DURATION		minutes	60	15	15	15
ASSESSMENT METHOD		ICS or radiation monitor (RM) readings in the MCR or local indication as necessary				

NOTE:

(1) Table values are calculated values. The ***** indicates the monitor is off scale.

(2) These release rate values in µCi/s and µCi/ml are provided on the gaseous and liquid release points for <u>Information Only</u>. Actual monitor readings are giveni n the table corresponding to the monitor for the four emergency classifications.

(3) This eberline channel reads out in cpm in the MCR. Indications of a radioactivity release via this pathway would be S/G blowdown monitors or other indications of primary-to-secondary leakage such as S/G level increase or pressurizer level decrease. ICS calculates µCi/cc and has a visual indication of an alarm condition when the indications exceeds 5.5E-02 µCi/cc. This channel was included in the table to provide a means to further assess a release detected by other indications and to provide a path for possible escalation.

(4) These unit values are based on flow rates through one [1] PORV of 970,000 lb/hr at 1,185 psig, 600 °F. Before using these values, ensure a release to the environment is ongoing, (e.g. PORV).

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Section 7.0	**RADIOLOGICAL**
Event 7.3	**RADIATION LEVELS**
Classification	**GENERAL EMERGENCY**
Mode	Not Applicable
Description	Refer to the "Fission Product Barrier Matrix" or "Gaseous Effluents" (7.1)
Basis	Not Applicable
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev 2, 1/92

Section 7.0	**RADIOLOGICAL**
Event 7.3	**RADIATION LEVELS**
Classification	**SITE AREA EMERGENCY**
Mode	Not Applicable
Description	Refer to the "Fission Product Barrier Matrix" or "Gaseous Effluents" (7.1)
Basis	Not Applicable
Escalation	Escalation may be based on "Fission Product Barrier Challenges" or Gaseous Effluent levels.
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 7.0	**RADIOLOGICAL**
Event 7.3	**RADIATION LEVELS**
Classification	**ALERT**
Mode	All
Description	**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) 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) 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 See UNUSUAL EVENT Note
Basis	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. EAL #1 applies to areas that are manned continuously. The value of 15 mrem/hr has been determined to 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. The Control Room, and the Central Alarm Station (CAS) should be continuously manned. Thus, the 15 mrem/hr value applies to these facilities. 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.

Section 7.0	**RADIOLOGICAL**
Event 7.3	**RADIATION LEVELS**
Classification	**ALERT** (continued)
Mode	All
Basis (continued)	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.). 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. 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.
Escalation	Escalation may be based on "Fission Product Barrier Challenges" or Gaseous Effluent levels.
References	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.

Section 7.0	**RADIOLOGICAL**
Event 7.3	**RADIATION LEVELS**
Classification	**UNUSUAL EVENT**
Mode	All
Description	**UNPLANNED increases in Radiation levels within the Facility** 1. VALID area Radiation Monitor readings increase by a factor of 1000 over normal levels. 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).
Basis	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. 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. 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.
Escalation	Escalation will be based on the inability to access certain operating stations or equipment needed to establish or maintain Cold Shutdown.
References	NUMARC/NESP-007, AU2, Rev 2, 1/92 WBN QDCN 20764 B - Radiation Monitor Readings for the REP. WBN TSR-044, R0 (B18 891227 254) Required Range and Accuracy for the WBN Area Radiation Monitors. WBN TSR-077, R2 (B18 920727 317) Radiation Zones.

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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	E1.757.0 (spent fuel pool)	2.5×10^3 mR/hr
1-RE-90-2	Auxiliary	E1.757.0 (personnel air lock)	2.5×10^0 R/hr
0-RE-90-3	Auxiliary	E1.729.0 (waste pac. area)	2.5×10^3 mR/hr
0-RE-90-4	Auxiliary	E1.713.0 (decon room)	1.5×10^3 mR/hr
0-RE-90-5	Auxiliary	E1.737.0 (spt. fuel pool pmp. ar.)	1.5×10^3 mR/hr
1&2-RE-90-6	Auxiliary	E1.737.0 (comp. cl. wtr. ht. ex. ar.)	1.5×10^3 mR/hr
1&2-RE-90-7	Auxiliary	E1.713.0 (sample room)	2×10^3 mR/hr
1&2-RE-90-8	Auxiliary	E1.713.0 (aux. feed pump area)	1.5×10^3 mR/hr
0-RE-90-9	Auxiliary	E1.692.0 (wst. cond. evap. tk. ar.)	1.5×10^3 mR/hr
1-RE-90-10	Auxiliary	E1.692.0 (cvcs area)	1.5×10^3 mR/hr
0-RE-90-11	Auxiliary	E1.676.0 (ctmt.spry. & rhr pmp ar.)	1.5×10^3 mR/hr
1-RE-90-61	Auxiliary	E1.736.0 (RB low. cmpt .inst. m.)	2.5×10^3 mR/hr
0-RE-90-230	Turbine	E1.685.0 (conden. demin.)	1.5×10^3 mR/hr
0-RE-90-231	Turbine	E1.685.0 (conden. demin.)	1.5×10^3 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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Section 7.0	**RADIOLOGICAL**
Event 7.4	**FUEL HANDLING**
Classification	**GENERAL EMERGENCY**
Mode	Not Applicable
Description	Refer to Gaseous Effluents (event 7.1)
Basis	The basis for a General Emergency is primarily the extent and severity of Gaseous Effluents (event 7.1)
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 7.0	**RADIOLOGICAL**
Event 7.4	**FUEL HANDLING**
Classification	**SITE AREA EMERGENCY**
Mode	Not Applicable
Description	Refer to Gaseous Effluents (event 7.1)
Basis	The basis for a Site Area Emergency is primarily the extent and severity of Gaseous Effluents (event 7.1)
Escalation	Not Applicable
References	NUMARC/NESP-007, Rev. 2, 1/92

Section 7.0	**RADIOLOGICAL**
Event 7.4	**FUEL HANDLING**
Classification	**ALERT**
Mode	All
Description	**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) 1. VALID Alarm on 0-RE-90-101 <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) 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
Basis	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. 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. 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.
Escalation	Escalation would occur by offsite dose rates. See Gaseous Effluents (7.1)
References	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

Section *7.0*	**RADIOLOGICAL**
Event *7.4*	**FUEL HANDLING**
Classification	**UNUSUAL EVENT**
Mode	All
Description	**UNPLANNED** loss of water level in Spent Fuel Pool or Reactor Cavity or Transfer Canal with fuel remaining covered (1 and 2 and 3) 1. Plant personnel report water level drop in Spent Fuel Pool or Reactor Cavity, or Transfer Canal 2. **VALID** alarm on 0-RE-90-102 or 0-RE-90-103 or 1-RE-90-59 or 1-RM-90-60 3. Fuel remains covered with water
Basis	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. Unplanned is included in the IC to preclude the declaration of an emergency as a result of planned maintenance activities. 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. 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.
Escalation	Escalation of this event would be based on uncovering an irradiated fuel assembly or indications of high radiation levels on the refueling floor.
References	NUMARC/NESP-007, AU2, 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

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.

It is noted that the WBN on-shift staffing requirements meet the NUREG-0654, Table B-1, recommendations for both on-shift and the 30 minute staffing requirements.

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 Radcon/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.

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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 hour) 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 3-C), the Maintenance onshift staffing falls under the control of the OSC.

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)
- Radcon Shift Staff (one technician, per Technical Specifications) (remaining staff, notified within two hour to arrive, as soon as possible)
- Chemistry Shift Staff (notified within two hour to arrive, as soon as possible)
- Maintenance Shift Staff (notified within two hour to arrive, as soon as possible)

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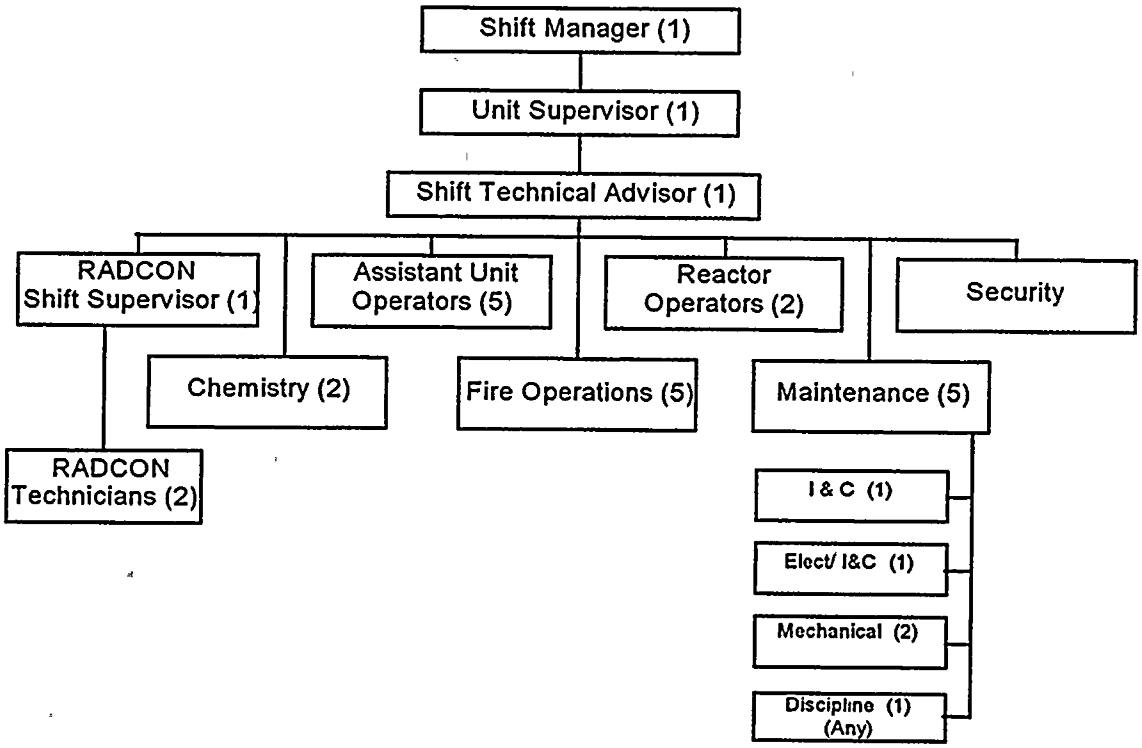
FIGURE 1-C
ON-SHIFT STAFFING



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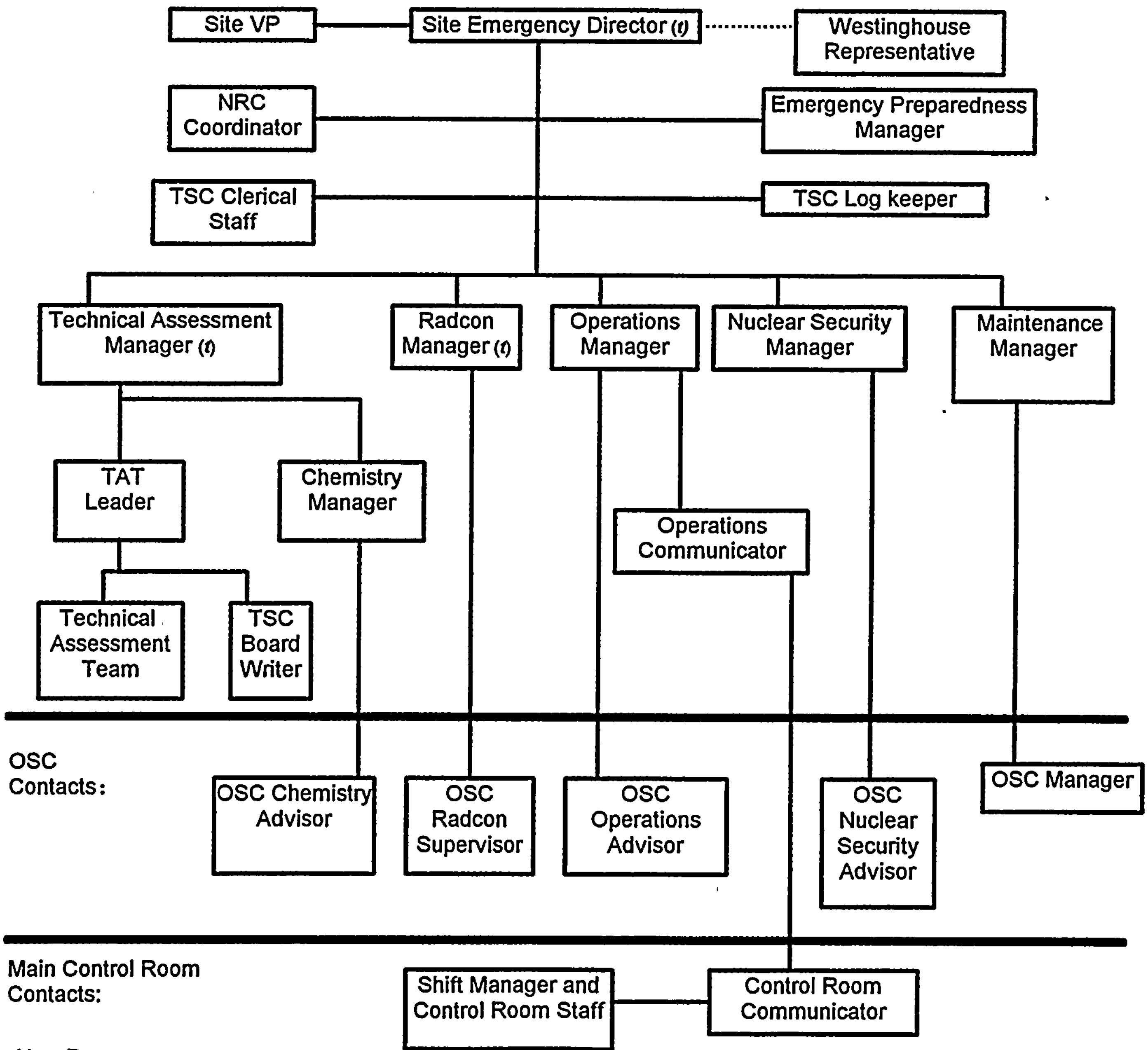
C.5.9 The TSC emergency response positions are described in WBN EPIP-6, "Activation and Operation of the Technical Support Center. Figure 2-C provides the typical staffing of the TSC. (*t*) denotes minimum staffing position(s) per NUREG 0654.

Figure 2-C
Technical Support Center (TSC)

WBN EMERGENCY RESPONSE ORGANIZATION



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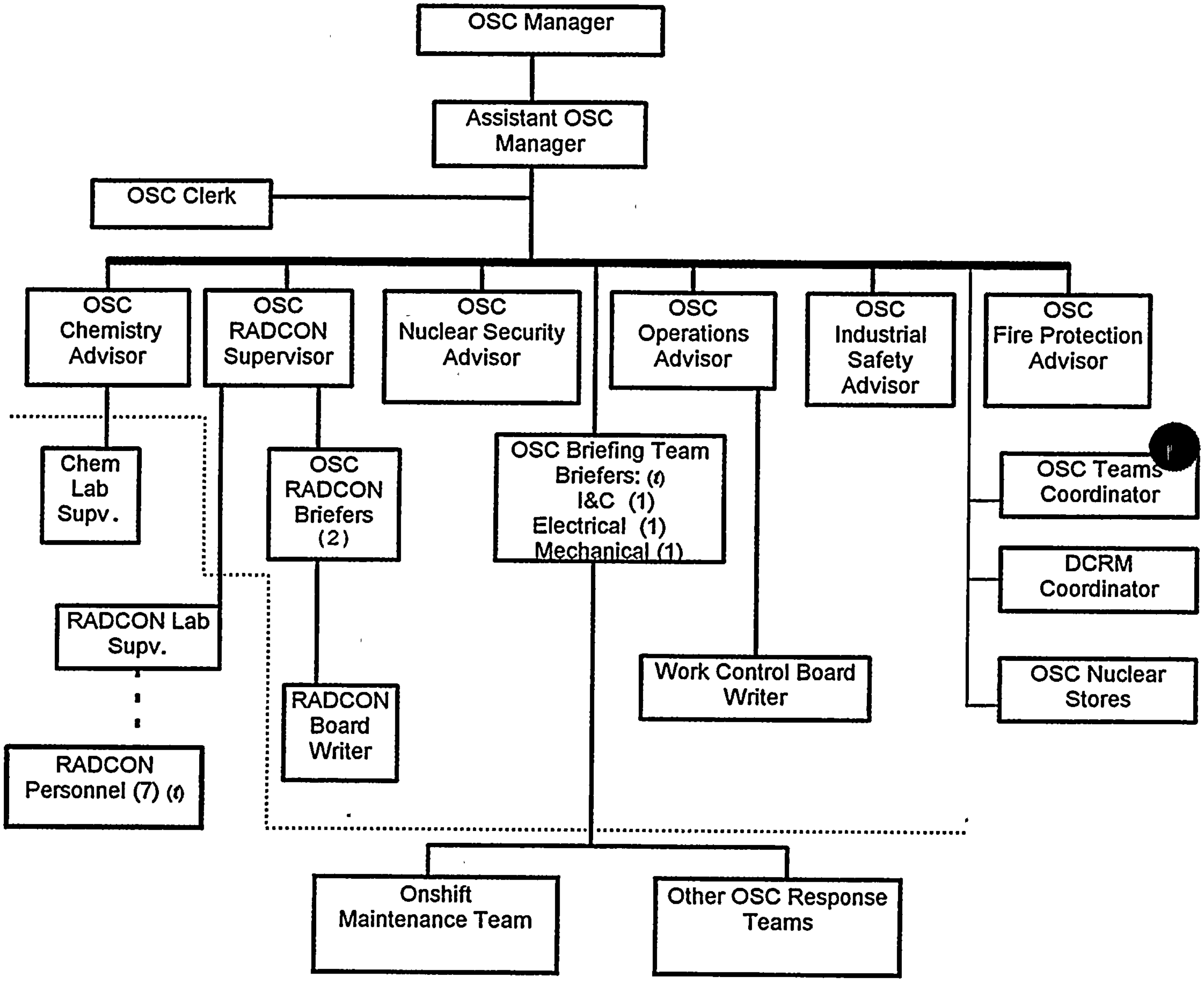
C.5.10 The Operations Support Center positions are described in WBN EPIP-7 "Activation and Operation of the Operations Support Center" Figure 3-C provides the typical staffing of the OSC. (*t*) denotes minimum staffing position(s) per NUREG 0654.

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.

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 Clerical Staff

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.

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-EPIPs.

C.5.17 Radiological Control Manager

1. Directs and/or performs assessment of inplant and onsite radiological conditions.

2. Directs onsite RadCon activities.

3. Coordinates additional RadCon 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 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.

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.

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 RadCon technician, as necessary.

C.5.24 Assistant OSC 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 RADCON Supervisor

1. Directs activities of the RadCon lab.
2. Ensure RadCon coverage of damage repair teams.
3. Verify habitability of the TSC, OSC, and Control Room.
4. Briefs the OSC Manager and TSC on RadCon 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.

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.

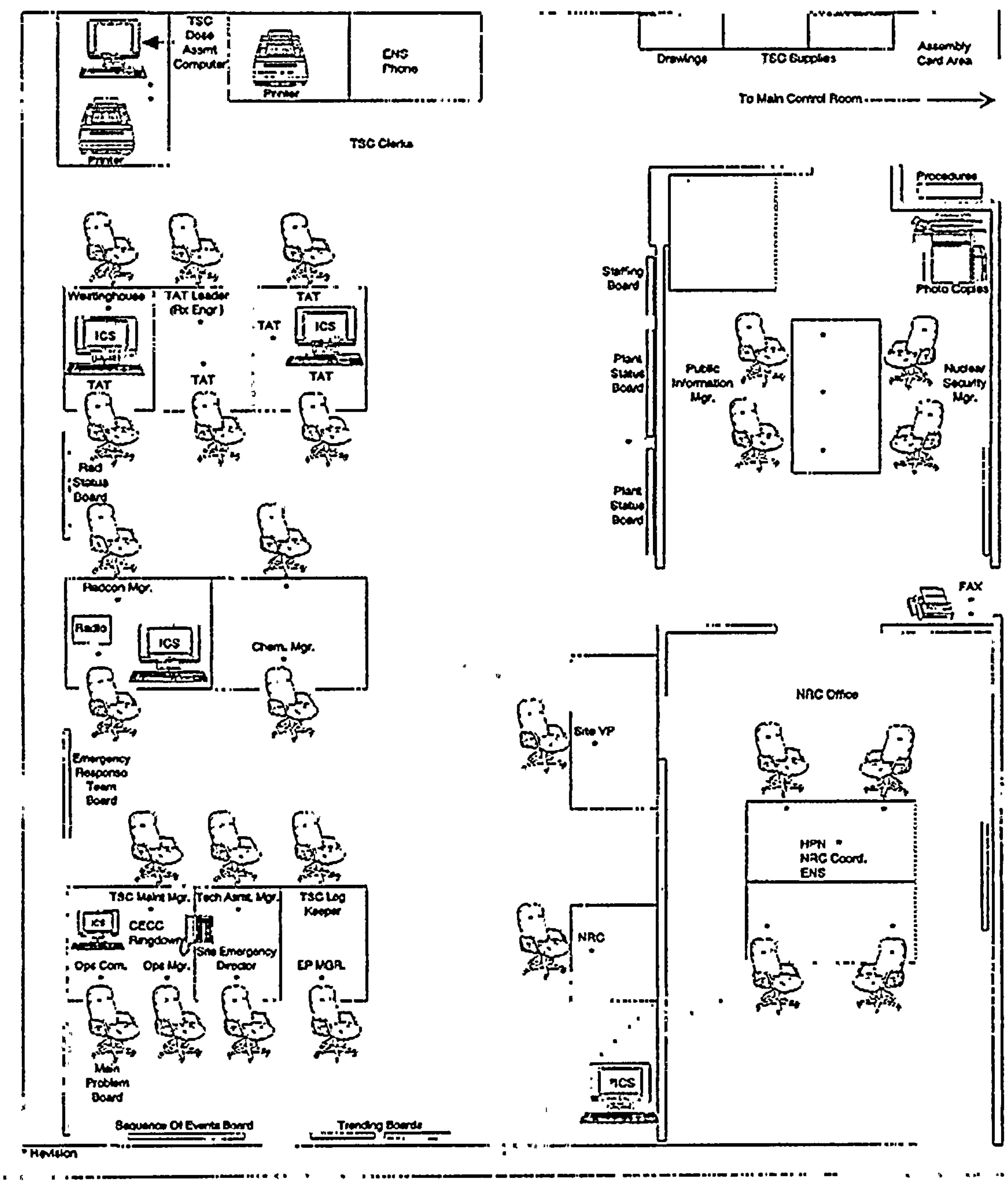
C.6.1 Technical Support Center (TSC)

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 2-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-16. Should the Met tower be unavailable, WBN EPIP-9 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.

Figure 2-C
TECHNICAL SUPPORT CENTER
CONTROL BUILDING EL. 755'
Facility Diagram



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 he OSC Manager or a designated alternate. The OSC is located on Elevation 713 adjacent to the RADCON 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 Plant Office Building Conference Room, with additional space provided in the adjacent Plant Assembly Room for staging, briefing and dispatching maintenance teams. The OSC is provided with telephone and radio communications. Figures 3-C and 4-C show the OSC areas. Respiratory protective devices, protective clothing, portable lighting, other protective equipment and tools are available, as needed.

FIGURE 3-C
OPERATIONS SUPPORT CENTER
SERVICE BUILDING
Elevation 713

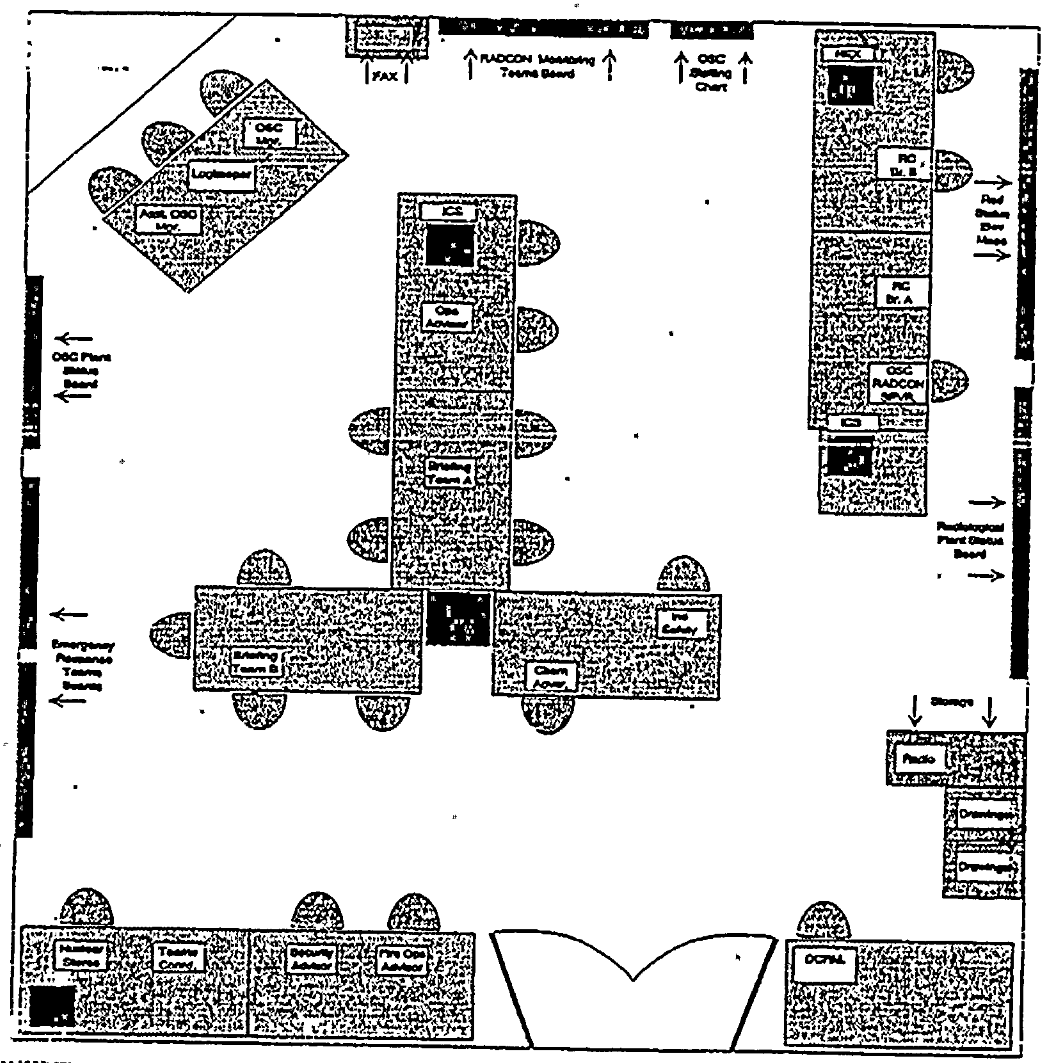
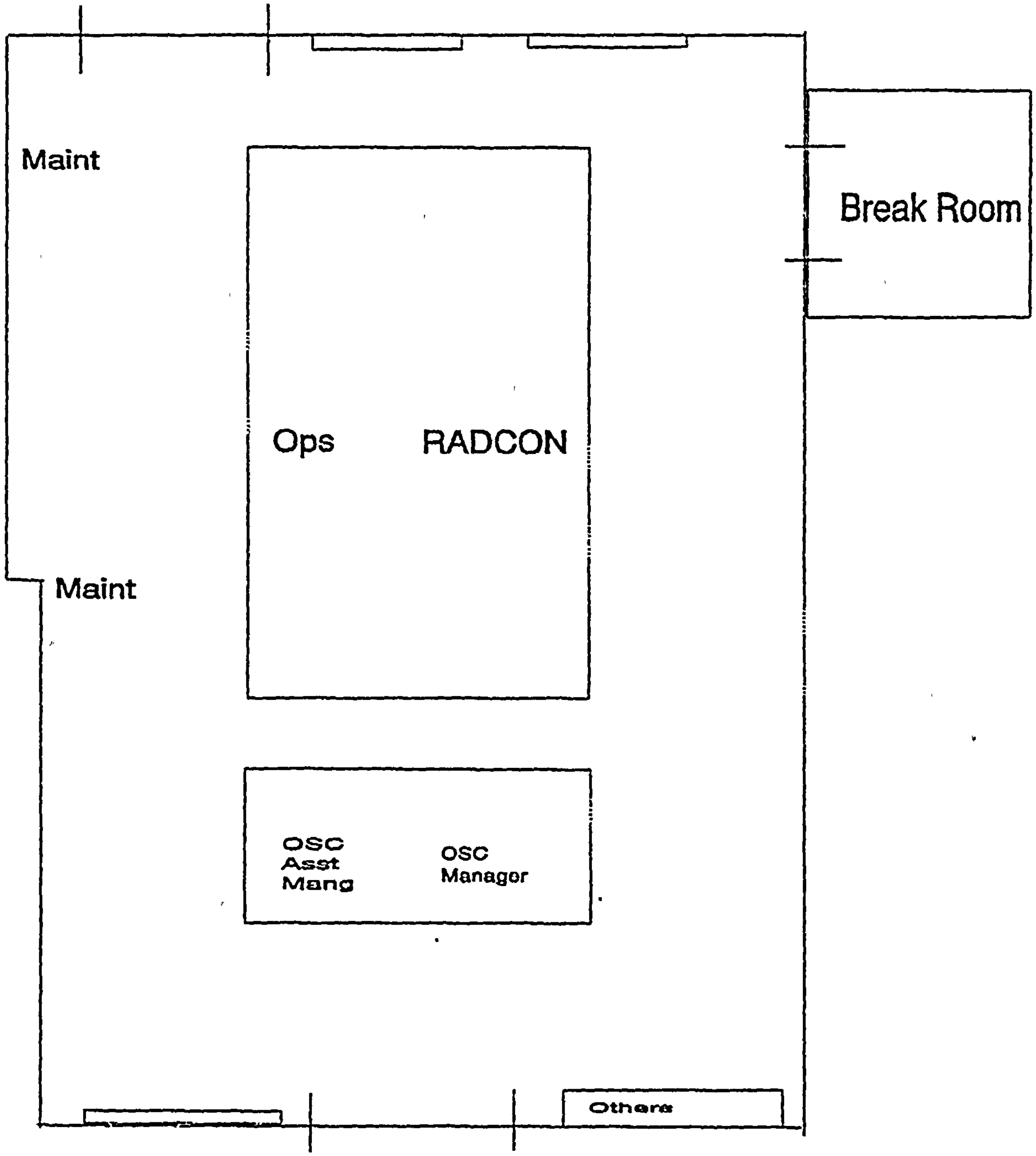


FIGURE 4-C
ALTERNATE OPERATIONS SUPPORT CENTER
MAIN OFFICE BUILDING
Elevation 729



C.6.3 RadCon Laboratory and Equipment

The RadCon laboratory is located in the Service Building, Elevation 713. The portable radiation monitoring and counting equipment normally used by the plant RadCon 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 Onsite Monitoring Systems and Equipment

C.6.4.1 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.6.4.2 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.

C.6.4.2.1 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.6.4.2.2 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.6.4.2.3 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.6.4.2.4 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.

C.6.4.2.5 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.6.4.2.6 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.

Thermoluminescent dosimeters (TLDs) are placed at approximately 40 sites around the plant. These TLDs 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.

C.6.5 Emergency Equipment

Figure 5-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.6.6 First Aid and Medical Facilities

C.6.6.1 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.

FIGURE 5-C

EMERGENCY EQUIPMENT

	Location	Description
1.	RadCon 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 (RadCon, 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

C.6.6.2 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 RadCon. Specific information including authorization and dispersal of tablets is contained in the site EPIPs.

C.6.6.3 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.6.6.4 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.6.7 Additional Local Support

C.6.7.1 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.

C.6.7 Additional Local Support (continued)

C.6.7.2 Law Enforcement

Agreements are maintained with local law enforcement agencies to support TVA when necessary.

C.6.8 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.6.9 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 all clear signal is a steady three-minute intermittent horn.

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.6.10 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 6-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.

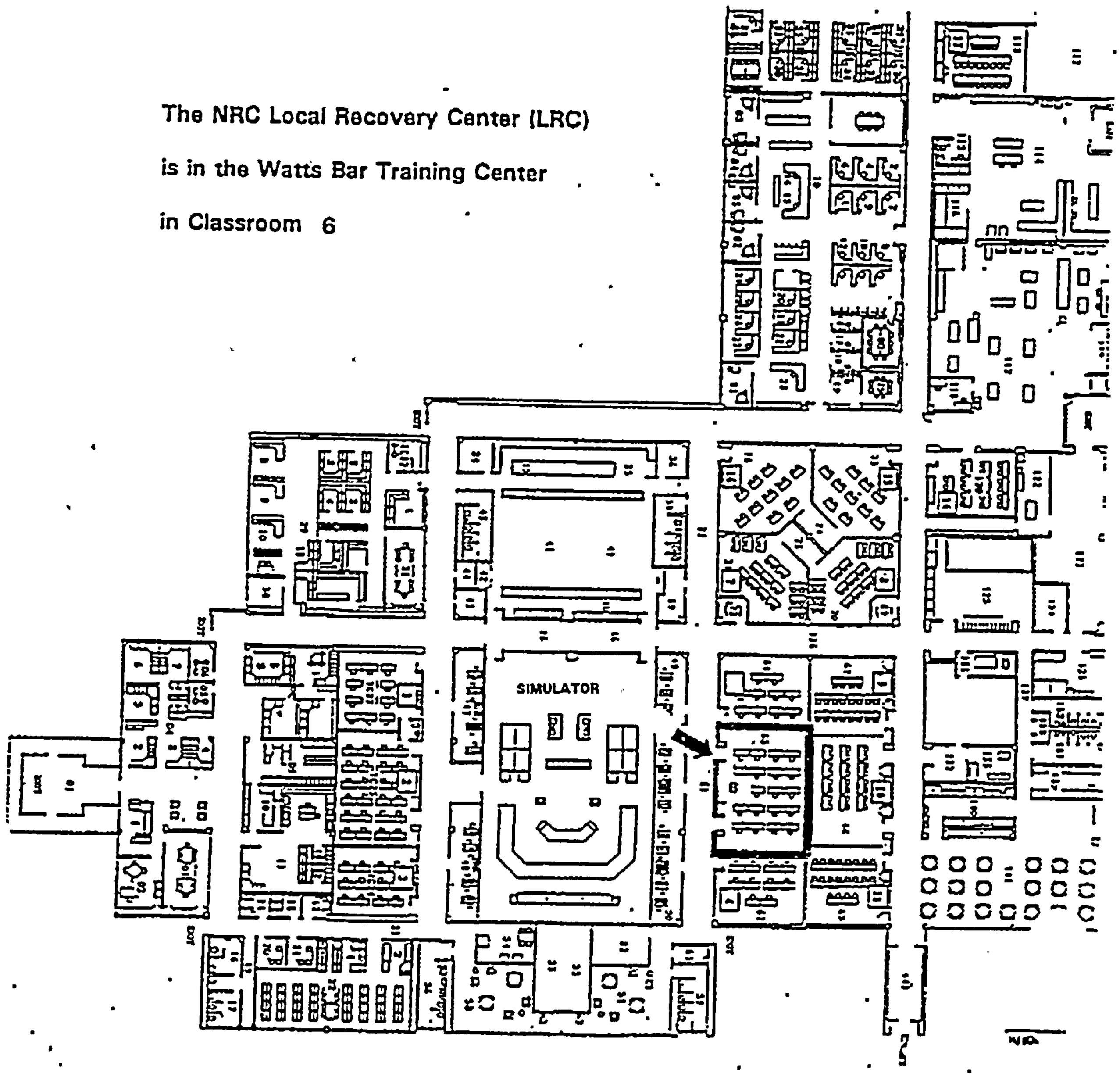
Other equipment in the WTC available for use by LRC personnel include:

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 6-C

LOCAL RECOVERY CENTER

The NRC Local Recovery Center (LRC) is in the Watts Bar Training Center in Classroom 6



C.7 WBN Emergency Plan Implementing Procedures

The following is a listing of the WBN-EPIPs:

C.7.1 WBN EPIP-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 EPIP-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 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.

C.7.3 WBN EPIP-3 Alert

This procedure provides for the timely notification of appropriate individuals when the Shift Manager SED or TSC SED has determined by WBN 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.

C.7.4 WBN EPIP-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 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.

C.7.5 WBN EPIP-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 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.

C.7.6 WBN 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.

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

This procedure directs operators on actions to take when meteorological indicators are lost in the main control room.

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 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.7.14 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, RadCon 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 Initial Dose Assessment for Radiological Emergencies

This procedure provides initial guidance to support site activities concerning dose assessment for an actual or exercise airbome release situation.

C.8 PROMPT NOTIFICATION SYSTEM

The prompt notification system network consists of fixed sirens and tone-alert radios. The system is designed to provide warning within 15 minutes to the population within 10 miles of the plant.

C.8.1 Fixed Sirens

The fixed-siren component consists of electromechanical sirens. The sirens are activated by the Tennessee Emergency Management Agency (TEMA). A backup activation system is located in Rhea County.

The siren system is activated on a monthly basis by TEMA 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 the silent tests. A growl test is conducted annually.

Preventive maintenance is performed by TVA on an annual basis commensurate with the manufacturer's recommendations. Unscheduled maintenance is performed on an as-needed basis.

C.8.2 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 an annual basis commensurate with the manufacturer's recommendations. Unscheduled maintenance is performed on an as-needed basis.

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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50-259 Superseded Per Rev 61 To REP Dtd 8/24/01
#ML012400400

TENNESSEE VALLEY AUTHORITY

NUCLEAR

RADIOLOGICAL EMERGENCY PLAN

TVA NUCLEAR

REVISION LEVEL: 60 REVISION DATE: 6/5/01

APPROVED: [signature]
Vice President
Engineering & Technical Services

TENNESSEE VALLEY AUTHORITY

Nuclear Power - Radiological Emergency Plan

List of Effective Pages

This List of Effective Pages must be retained with the Nuclear Power Radiological Emergency Plan.

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REVISION LOG

Rev. & Date		Pages Revised	Reason for Change
0	4/22/88	All	General Revision to convert from individual site-REPs to Common REP with site-specific appendices. Also revises REP approval cycle.
1	12/7/88	A-9, A-14	Revised to add wind speeds to Browns Ferry's Emergency Action Levels (EALs) per memorandum from the NRC dated November 1, 1988.
2	6/5/89	Revised pages marked with an asterisk (*); all pages issued.	Revisions from annual review.
3	6/30/89	A-8, A-9, A-14	Revised Appendix A EALs regarding tornado warnings.
4	9/25/89	i, v, vii, 1, 12, 14, 15-19, 29, 38, 65, 74, A-25, thru A-29, B-30 thru B-33, B-47	Changed emergency action level for tornado at SQN; removed plant communication and CECC Communicator function; clarified training requirement.
5	4/6/90	i.ii, vii, viii, x, 11, 14, 15, 29-31, 35, 37, 44, 48, 52, 55, 56, 62, 63, 65, B-3 thru B-47	Revised to incorporate annual review comments which include: rewrite Section 6; title changes; word changes for clarification; minor changes in implementation; substantial rewording to SQN EALs.
6	5/4/90	v, vi, vii, viii, x, A-1 thru A-45 (A-46 thru A-115 added), B-1 thru B-47 (B-48 thru B-154 added).	Revised to incorporate new EAL format.
7	04/01/91	Revised pages marked with an asterisk (*); all pages issued.	Revisions from annual review and SQN and BFN EAL changes resulting from NRC comments.
8	10/25/91	A-65 and B-92	Revised for clarification of BFN EAL HU14 and SQN EAL HU15.
9	12/17/91	A-29, B-4, and B-5	Revised for clarification of BFN EAL SU7 and SQN EAL FU2.

REVISION LOG

Rev. & Date		Pages Revised	Reason for Change
10	5/5/92	3, 11, 12, 15, 17, 37, 52, (App. A) A-6, A-61, A-62, A-75, A-94, A-97, A-100, A-103, A-104, A-115; (App.B) B-4, B-9, B-18, B-22, B-25, B-32, B-37, B-39, B-44, B-45, B-46, B-48, B-49, B-52, B-53, B-54, B-57, B-58, B-59, B-60, B-61, B-62, B-63, B-66, B-71, B-88, B-106, B-119, B-122, B-131, B-132, B-133, B-139, B-140, B-141, B-142, B-148, B-149, B-154	Annual Review.
11	11/25/92	B-4 thru B-8, B-14 thru B-22, B25, thru B-33, B-36, B-38 thru B-39, B-44, B-46, B-60 thru B-67, B-70, B-73, B-73A, B-74, B-76 thru B-77, B-87, B-90, B-92, B-96 thru B-97, B-113 thru B-114, B-128 thru B-129.	EAL Review.
12	04/09/93	18, 35, 38, 42, A-75, A-100, A-105, B-14, B-25, B-37, B-49, B-54, B-59, B-67, B-92, B-139, and B-149.	Annual Review
13	04/30/93	1-2; B-14; B-22.	Changes in response to comments received from NRC in a letter dated April 2, 1993, after review of Revision 12 of the REP.
14	10/04/93	13, 35, 38, A-100, B-139, B-143, B-147, and B-148	Updated in response to NRC comments.
15	01/01/94	x, 1, 3, 5, 48-49, 51, 53, 54, 55, A-43-45, A-47, A-50, A-51, B-60, B-61, B-63 thru B-67, and B-69 thru B-76	Incorporate 10CFR20 and EPA 400 changes.
16	03/21/94	A-94, A-98, A-100, A-101, A-103, and A-104	Annual Review.
17	06/30/94	1, 3, 5, 13, 15, 17-20, 25, 26, 30, 32-33, 35, 37,43, 46, 49-50, 58-60, 64, 67, 69, 74, B-70, B-73, B-73A, B-76, B-83, B-136, B-150 - B-152	Annual Review.

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Rev. & Date		Pages Revised	Reason for Change
18	10/18/94	1-2, 5, 25, 59, 67, 69-73, A-48, A-50, A-51, B-26, B-34, B-60, B-61, B-63, B-65, B-66, B-67, B-68, B-70 thru B-73, B73A, B-74 thru B-76, B-76A, B-97	Annual Review.
19	1/12/95	A-11	Annual Review.
20	2/22/95	Page 63, B-10, B-61, B-65, B-67, B-70, B-73, B-73A, B-74, B-76	Annual Review.
21	4/12/95	C-1 thru C-218	Issue Appendix C (WBN).
22	7/12/95	C-11, C-15, C-17, C-26, C-27, C-28, C-49, C-53, C-75, C-77, C-95, C-98, C-102, C-107, C-113, C-119, C-124, C-132, C-133, C-134, C-136, C-138, C-151, C-159, C-162, C-163, C-165 thru C-181, C-187 thru C-190, C-194, C-197, C-200 thru C-206, C-211, C-212, C-216	Resolution of NRC Comments.
23	8/14/95	Appendix B, all pages	Issue revised EALs based on NUMARC criteria.
24	9/27/95	2, 11, 12, 14, 17, 18, 21, 22, 23, 24, 35, 42, 46, 49, 51, 57, 58, 59, 60, 61, A-100, A-114, C-7, C-13, C-14, C-15, C-18; C-20, C-24, C-28, C-62, C-64, C-66, C-88, C-115, C-116, C-128, C-129, C-132, C-162, C-163, C-165 - C-180, C-184 - C-190, C-200, C-201	Revised for clarification and organizational changes, add statement that the RAM is responsible for dose authorization for personnel under him, remove references for downgrading emergency classifications, revise PAR descriptions and add PAR diagram, revise BFN staffing chart, revise WBN EALs for accuracy.
25	11/01/95	51 Appendix A, all pages	Revise PAR Diagram. Issue revised EALs based on NUMARC criteria.
26	11/16/95	A-45, A-47, A-87, A-88, A-89	Incorporate BFN Unit 3 temperature limits and incorporate SSI-16 relating to Control Room abandonment.
27	5/31/96	Generic Part, pages 1-63 A145, A149, A150, A151, A154, A-155, B6 thru B8, B19, B23, B-25, B37, B39, B41, B47, B51, B53, B55, B64, B65, B66, B67, B72, B74, B79, B86, B94, B96,	Annual Review.

REVISION LOG

Rev. & Date		Pages Revised	Reason for Change
27 (Continued)		B99, B100, B103, B113, B121, B122, B124, B125, B130, B132, B133, B138, B139, B141, B145, B147, B159, B165, B166, B176, B177, B178, C190, C216	
28	6/18/96	Generic Part, Page 54	Section 14.2.1--Revised the section to reflect revised requirements in 10 CFR 50 Appendix E.
29	7/12/96	B-6, B-7, B-36, B-38, B-40, B-46, B-113, B-159, B-167, B-169, B-170, B-173, B-176, B-177, B-183	Changes made for clarification, routine updates, and correction of minor inaccuracies.
30	7/26/96	53, 54, 55	Clarify environmental monitoring drill requirements. Revise exercise requirements to meet new NRC regulations.
31	9/27/96	47, 57, C-115, C-116, C-135, C-136, C-137, C-190, C-202, C-203, E-2	Remove requirement for directions to REAC/TS be included in site EPIPs. Add instructions for approval at Appendix E revisions. Editorial changes to Appendix C EALs. Update staffing figures to reflect current plant organization and titles. Add 46m reading from met. tower. Replace reference to local and perimeter monitors with radiological monitoring survey points on the plant perimeter. Update references in Appendix E.
32	11/01/96	41, A-141, B-3, B-6, B-7, B-8, B-15, B-23, B-41, B-44, B-45, B-46, B-48, B-51, B-59, B-65, B-71, B-72, B-73, B-76, B-80, B-82, B-84, B-114, B-117, B-118, B-121, B-122, B-133, B-140, B-141, B-144, B-146, B-150, B-154, B-157, B-162, B-163, B-165, B-166, B-169 thru B-188	Revise PAR Diagram, revise BFN Emer. Org. Chart, revise SQN containment rad monitor thresholds for indication of fuel damage, add SQN emergency positions and duties. Editorial and organizational changes.
33	12/17/96	C-203 through C-223	Add listing and responsibilities of key WBN emergency responders, change Athens Community Hospital to Athens Regional Medical Center, renumber pages due to addition of several new pages.

REVISION LOG

Rev. & Date		Pages Revised	Reason for Change
34	2/25/97	19, 23, 24, 25, 26, 27, 37, 49, 50 51, 57, 58, 59, 63, A-135, A-136, A-137, A-138, A-139, A-141, A-142, A-144, A-145, A-148, A-149, A-151, B-159	Remove reference to equipment no longer used, update position and organizational changes, minor editorial changes, add requirement for on-shift dose assessment capability at the sites, revise duties of BFN TSC clerks, add BFN TSC and OSC as locations for emergency supplies, annual review, correct typographical error on pages B-159. All generic pages issued.
35	7/17/97	49, 50, 51, 57, 58, 59, 60, 61, 63, A-1 - A-155, B-113, B-167	Update approval authority and recovery organization. Update new ambulance service name. In Appendix A change references from NUMARC/NESP-007, Rev. 2 to Reg. Guide 1.101, Rev. 3 to refer to NRC document. Revise calculation reference. Revise tables 1.1-G2 and 3.1 per applicable revisions to EOI tables. Option to obtain site dose assessment when CECC not staffed added. Clarify wording of security EALs. General editorial and position updates. In Appendix B remove reference to Knoxville National Weather Service telephone number and correct value of 1.2-RM-90-2 in table 7-2.
36	8/25/97	49, A-45, A-47	Correct title for Manager of Nuclear Licensing. Correct value for U-3 "Main Steam Line Leak Detection High" max safe operating value °F in Table 3.1.
37	12/23/97	C-6, C-7, C-15, C-16, C-21, C-24, C-44, C-46, C-51, C-59, C-115, C-116, C-121, C-141, C-145, C-155, C-166, C-167, C-171, C-172, C-174, C-180, C-184, C-190, C-195, C-219	Change title "Shift Operations Supervision" to "Shift Manager", "SOS" to "SM", "ASOS" to "US." Update clad failure percentage range. Revise EAL 1.3, change "plant computer" to "P-2500 plant computer." Remove reference to the REP from references list on p C-51. Revise EAL 2.4, EAL 5.4, EAL 5.5, EAL 7.1. Change "1-RE-90-421 through 424(B)" to " 1-RE-90-421 through 424." Remove references to TI-30, minor editorial changes.
38	6/9/98 ~~6/4/98~~ *RR*	i, iv, 2, 9, 10, 11, 14, 15, 19, 22, 24, 25, 29, 30, 33, 34, 35, 36, 39, 46, 47, 50, 56, 58, 63	Annual review. Change title "Shift Operations Supervision" to "Shift Manager", "SOS" to "SM", "ASOS" to "US." Remove reference to EIC as the term is no longer used, title changes, organizational changes, minor changes in duties. Update CECC Figure. Update CECC-EPIP descriptions. Remove reference to American Medical Response. Clarify use of rev. log sheet. All generic section pages issued.

REVISION LOG

Rev. & Date		Pages Revised	Reason for Change
39	7-27-98	All Appendix A pages issued.	Change to mode nomenclature due to BFN Generic Tech. Spec. implementation.
40	8-6-98	All Appendix B pages issued.	EALs updated based on redundant information sources in CR. Procedure title updates, editorial and organization title updates. Update emergency center diagrams.
41	10-5-98	All Appendix A pages issued.	Update EAL heat capacity and pressure suppression curves, update PSIG values (page A-12) and update Table 1.1-G2.
42	10-28-98	All Generic Part pages issued.	Editorial and organizational changes. Add state that the Plan Effectiveness determination is done in accordance with 10 CFR 50.54q. Change name of North Park Hospital to Memorial North Park Hospital.
43	12-28-98	All Generic Part, Appendix A, and Appendix B pages issued.	In generic part a statement concerning SAMGs was added. Revisions to Appendix A result from a revision to the Emergency Operating Instructions Writer's Guide which required Operations to review and modify EOI and basis information. Revisions to Appendix B were made due to Technical Specification change 98-02.
44	2/22/99	All Generic Part and Appendix A pages issued.	In generic part organization titles revised, PAR diagram revised. In Appendix A, revise radiation monitor values, update references, remove references to RCI-1.1, editorial changes.
45	3/19/99	Appendix B pages issued.	Editorial corrections.
46	4/22/99	Appendix C pages issued.	Editorial and organizational title change. Remove reference to de-escalation, change Site Perimeter (SP) to Exclusion Area Boundary (EAB). Update equipment nomenclature.
47	5/1/99	All Generic Part and Appendix A pages issued.	In Generic Part PAR diagram revised. In Appendix A, revise radiation monitor readings, update references, changes due to outage modification, editorial changes.
48	5/20/99	All Generic Part and Appendix C pages issued.	In Generic Part CECC layout diagram revis[ed]. In Appendix C, Onshift Staffing diagram revised.

REVISION LOG

Rev. & Date		Pages Revised	Reason for Change
49	7/13/99	All Appendix B pages issued.	In Appendix B EAL 2.1 statements for NOUE, Alert, and SAE are being returned to the intent status they were in prior to Rev. 40 although the wording has been modified. Additionally, for the SAE EAL (page B-36), the statement that excludes consideration of annunciators that are out of service due to scheduled maintenance or testing activities has been deleted..
50	8/10/99	All Appendix B pages issued.	In Appendix B Radiological Effluent EALs revised to bring into agreement with Rev. 41 to the ODCM.
51	10/12/99	All Appendix C pages issued.	Editorial changes, update earthquake EALs based on equipment change out, change terminology from Lower Toxicity Limit (LTL) to Permissible Exposure Limit (PEL), update Figure C-1, remove Figure C-2.
52	11/17/99	All Appendix B pages issued.	Change terminology from Lower Toxicity Limit (LTL) to Permissible Exposure Limit (PEL). Add Condensate Storage Tank and Addition Equipment Bldg. to lists. Add multi-purpose building to Figure 4-A. Revise EAL 5.1 for Alert and Unusual Event to correspond to new seismic instruction.
53	11/18/99	All Appendix A pages issued.	Change terminology from Lower Toxicity Limit (LTL) to Permissible Exposure Limit (PEL).
54	3/21/00	All Appendix B pages issued.	Editorial changes for clarification on pages B-88 and B-110. Correct typographical error for liquid release alert trigger value in Table 7-1, page B-159.
55	4/28/00	All Appendix A pages issued.	Revise Table 3.1, Unit 3, Temp. Values for RWCU RECIRC PUMP A & B areas. Make clarification, editorial, and format changes due to annual review and self-assessment. Remove reference to Transmission Power System Engineer as this position is no longer used. Remove reference to STD-5.1.
56	6/30/00	All Generic, Appendix B and Appendix C pages issued.	Annual review and self-assessment identified items.
57	8/17/00	All Generic and Appendix C pages issued.	In Generic Part revise PAR diagram. In Appendix C revise EAL 1.1.5.

REVISION LOG

Rev. & Date		Pages Revised	Reason for Change
58	2/5/01	All Generic pages issued.	In Generic Part correct PAR diagram.
59	3/30/01	All Generic and App. A pages issued.	In Generic Part issue new PAR diagram. In App. A update references, adjust EAL release rates to match new dose code mixes, remove reference to evacuation signal, minor updates.
60	6/5/01	All Generic, App. B, and App. C pages issued.	In Generic Part remove reference to Dayton City Fire Dept. In App. B clarify state reporting requirements for non-declared events, revise EALs, revise TSC and OSC layout diagrams, and remove reference to all clear signal. In App. C adjust EAL release rates to match new dose code mixes, in EALs change Annunciator Printer to Annunciator Monitor, modify Figures 1 and 3, replace drawing of alternate OSC, and remove reference to all clear signal.

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1.0 DEFINITIONS AND ABBREVIATIONS

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

Exceptions:

1. Exercises, drills, emergency information for residents, media training, and offsite emergency response training is 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-EPIPs (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-EPIPs (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.

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 "Red Phone" 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 - Operations 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 (Federal Emergency Management Agency) - An agency of the U.S. Government.

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.

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

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 and 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.

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.

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.

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.

PED - Plan Effectiveness Determination.

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

PNS - Prompt Notification System.

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.

PSS - Public Safety Service.

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

RAA - Radiological Assessment Area of CECC.

RADCON - Radiological Control.

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 a part of 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.

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."

STA - Shift Technical Advisor.

SQN - Sequoyah Nuclear Plant.

SQN-EPIPs (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.

T&CS - Transmission and Customer Services.

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

TLD - Thermoluminescent Dosimeter.

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.

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-EPIPs (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 (NP). The Senior Vice President of NP has delegated the authority for overall program control of the NP-REP to the Manager, Emergency Preparedness.

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.

Site	Appendices
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 EPIPs. Site specific procedures for abnormal and emergency operation and control exist but are not included in the EPIPs. These plant operating procedures are designed to ensure the implementation of the EPIPs.

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 Federal Radiological Emergency Response Plan

The Federal Emergency Management Agency (FEMA) administers the Federal Radiological Emergency Response Plan (FRERP) which is the coordinated Federal Government response to a fixed nuclear power plant facility incident. This emergency plan is activated by either the affected State notifying the Federal Emergency Management Agency, or the utility notifying the NRC of a radiological emergency at a nuclear plant site. The FRERP 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 FEMA. 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 FRERP 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 FRERP 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 FRERP includes the Federal Radiological Monitoring and Assessment Plan for use by Federal agencies with radiological monitoring and assessment capabilities. The CECC Director is the TVA person authorized to request Federal assistance. Such a request from TVA will be made to NRC.

FIGURE 2-1

PRINCIPAL ORGANIZATIONAL RESPONSIBILITIES

	Local	State	TVA
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 and 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.

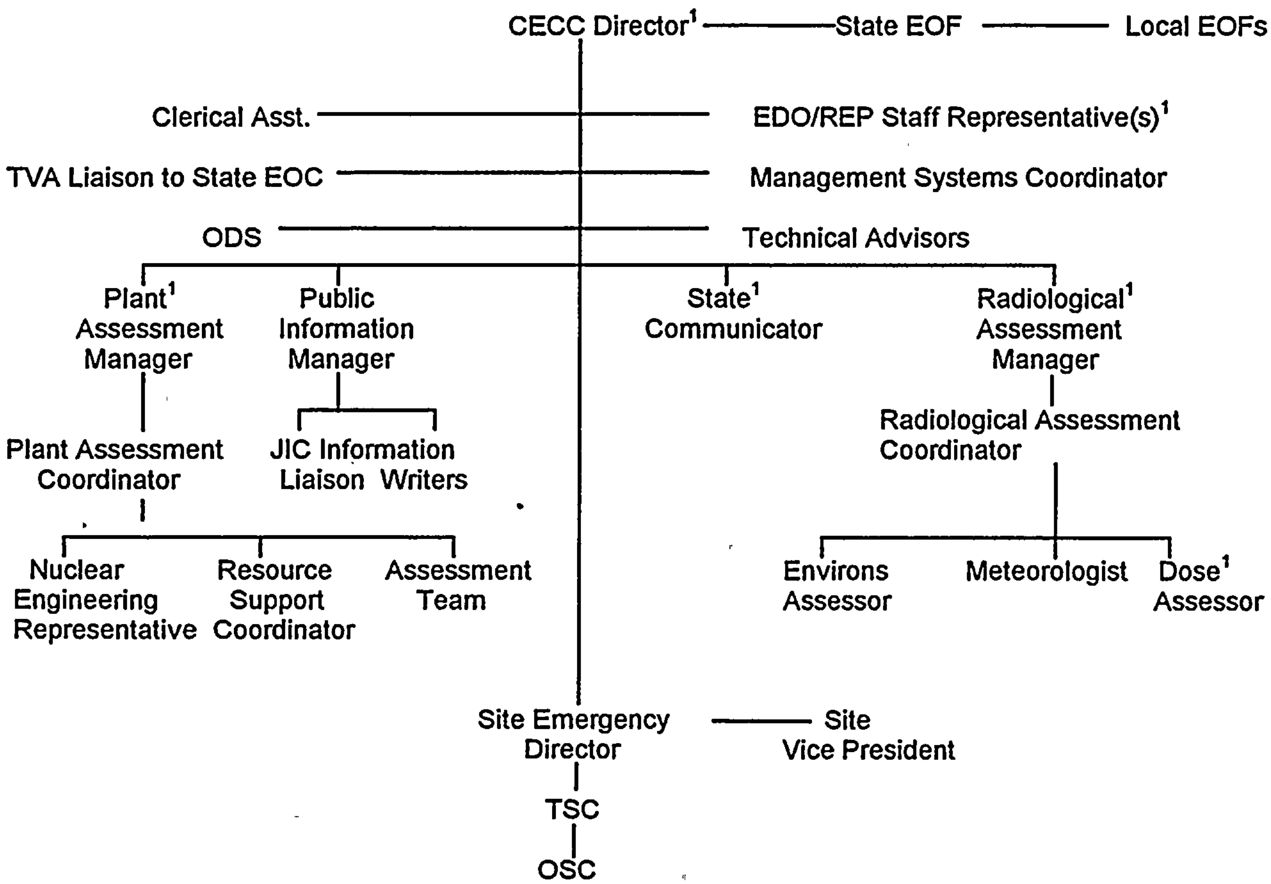
3.1 Onsite Organization

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. 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.

FIGURE 3-1

OFFSITE EMERGENCY ORGANIZATION



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

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-EPIPs.

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, FEMA, 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 the RadCon Superintendent recommends against the entry; and initiating long-term 24-hour per day accident mitigation operations.

The SED makes recommendations for protective actions (if necessary) to the State and local agencies through the ODS prior to the CECC being staffed (this responsibility can be transferred only to the CECC Director). The SED is also responsible for determining the emergency classification as well as the approval of 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 RADCON.

3.1.6 Radiological Control (RADCON) Manager

The RADCON Manager is responsible for assessing inplant and onsite radiological conditions; directing the onsite RADCON activities; coordinating additional RADCON 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 RADCON support to maintenance teams; and making final recommendation to the SED for personnel entry to radiologically hazardous environments.

3.1.7 Chemistry and Environmental Manager

Chemistry and Environmental 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 advise 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 and reporting recommended protective actions, determined by the site, to the appropriate State emergency organization. In addition, the ODS notifies appropriate TVA offsite emergency personnel. In the event of the initiation of the event as a General Emergency, he is required to notify the appropriate local response agencies.

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 develop 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 NP 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 and SEOC. He will ensure the TVA Chief 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.

He 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 RADCON 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 RADCON 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.

3.2.10 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 Systems

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 FRERP (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 EPIPs. 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 minor events which could develop into or be indicative of more serious conditions which are not yet fully realized.

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.

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 progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public.

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 progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity.

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-EPIPs, 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 EPIPs. These EALs are used as thresholds for determining the emergency classifications. EAL's are presented in the site-specific appendix. The EALs are reviewed annually by the appropriate State.

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 EPIPs.

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 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 carrying pagers. 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 unusual event by the SED. The ODS records the details of the event in accordance with the appropriate EPIP.

2. The ODS notifies and relays the information to the State within 15 minutes of declaration of the event. The ODS also notifies and relays the information to the EDO and CECC directors.

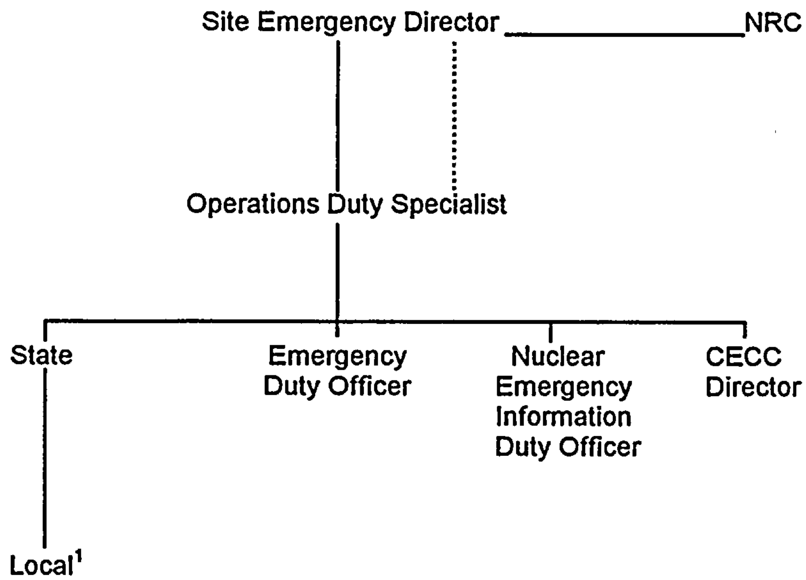
3. The EDO keeps the CECC Directors and the Nuclear Emergency Information Duty Officer informed of the situation as necessary.

4. The Nuclear Emergency Information Duty Officer notifies the Site Communications Consultant; General Manager Communications; and Media Relations.

5. The SED augments plant shift personnel as necessary to initiate corrective or protective actions.

FIGURE 5-1

CHAINS OF NOTIFICATION



[1]The ODS also notifies the local governments if the initial classification is a General Emergency

------ Verification

5.2.2 Alert

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

1. The ODS in Chattanooga is notified of the incident by the SED. The ODS records the details of the event in accordance with the appropriate EPIP.

2. The ODS makes the notifications described in section 5.2.1.

3. The CECC is staffed.

4. Environmental sampling teams may be dispatched.

5. The TSC and the OSC are activated.

6. The situation is analyzed and any appropriate corrective or preventive actions initiated.

7. 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 (airbourne, waterbourne, 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 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.

8. The JIC may be activated.

9. Periodic media releases are provided.

10. 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 actions 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 recommendation to the State are required upon declaration of General Emergency.

2. If this is the initial classification, the ODS 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/Plant Manager of the Affected Site

4. CECC Director

5. Radiological Assessment Manager

6. Plant 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 Radwaste 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 loudspeaker paging system.

6.3 Plant Loudspeaker Paging

This system may be accessed from the plant telephone system and is used for normal plant operations and to instruct and notify personnel during an emergency. Also, 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.

Other communications include those not provided by TVA, but that reside at TVA facilities. These are the ENS and HPN telephones (NRC FTS 2000 System) which 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 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 and to the following locations:

Radiological Control.

Technical Support Center.

Control Room at each plant.

CECC in Chattanooga.

6.7 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, nonemergency 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 TVA Chief Spokesperson

The TVA Chief 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 Vice President, Communications

Vice President, 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 Shared Resources Communications

Shared Resources 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) Shared Resources 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

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 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 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.

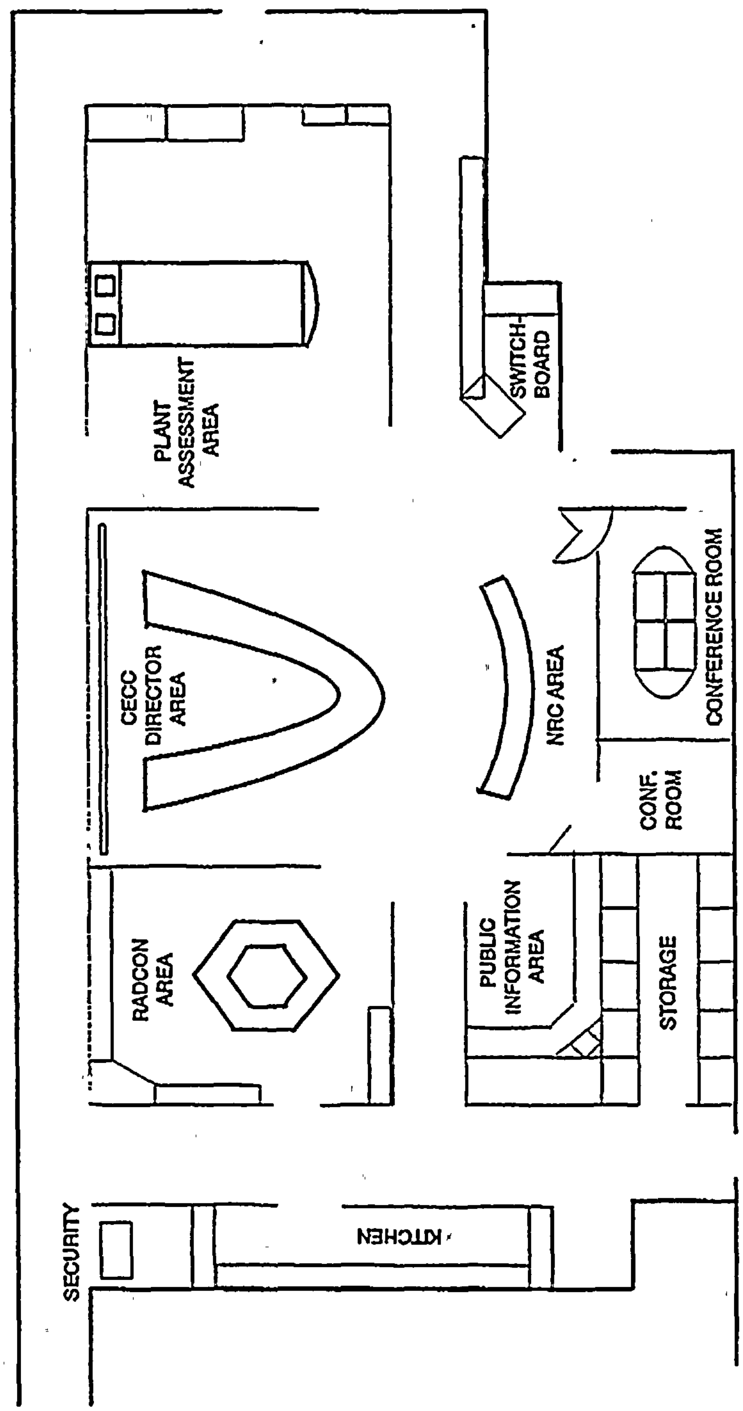
During an emergency, the CECC Director and his staff will review the response to the emergency by TVA and the appropriate State agencies to ensure that an effective and cooperative effort is being made. The CECC Director is responsible for providing TVA's recommended protective actions to the appropriate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

FIGURE 8-1
CENTRAL EMERGENCY CONTROL CENTER



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.

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 specific predetermined monitoring points to collect data. This data is passed by radio to the RMCC and relayed to the CECC for integration and analysis with the plant data.

Facilities at the RMCC include radio and telephone communications, tie-in to the Hard Copy Transmitting System, and necessary desks, tables, and chairs. Maps of the 10-mile EPZ and the 50-Mile EPZ with preselected radiological sampling and monitoring points are located at the RMCC. The preselected mobile laboratory locations are also reflected on a map at the RMCC.

8.4 Joint Information Center (JIC)

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

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

8.5 Prompt Notification System (PNS)

Each site has a PNS capable of warning the public within the plume exposure EPZ of a serious event. Specific PNS information is provided in the site-specific appendix.

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-EPIPs. 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-EPIPs 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-EPIPs, 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-EPIPs. 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-EPIPs and are summarized below.

CECC-EPIP-1 - CENTRAL EMERGENCY CONTROL CENTER ALERT, SITE AREA EMERGENCY, AND GENERAL EMERGENCY

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 - <u>OPERATIONS DUTY SPECIALIST PROCEDURE FOR NOTIFICATION OF UNUSUAL EVENT</u>

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 - <u>OPERATIONS DUTY SPECIALIST PROCEDURE FOR ALERT</u>

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

CECC-EPIP-4 - <u>OPERATIONS DUTY SPECIALIST PROCEDURE FOR SITE AREA EMERGENCY</u>

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

CECC-EPIP-5 - <u>OPERATIONS DUTY SPECIALIST PROCEDURE FOR GENERAL EMERGENCY</u>

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

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

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 - <u>CECC RADIOLOGICAL ASSESSMENT STAFF PROCEDURE FOR ALERT, SITE AREA EMERGENCY, AND GENERAL EMERGENCY</u>

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 - <u>DOSE ASSESSMENT STAFF ACTIVITIES DURING NUCLEAR PLANT RADIOLOGICAL EMERGENCIES</u>

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 - <u>EMERGENCY ENVIRONMENTAL RADIOLOGICAL MONITORING PROCEDURES</u>

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 - <u>WATER MANAGEMENT RADIOLOGICAL EMERGENCY PROCEDURES</u>

Cancelled - Pertinent parts moved to CECC-EPIP-8.

CECC-EPIP-11 - <u>SECURITY OF OFFSITE EMERGENCY FACILITIES</u>

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

CECC-EPIP-12 - <u>ENVIRONMENTAL RESEARCH AND SERVICES RADIOLOGICAL EMERGENCY PROCEDURES</u>

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-13- <u>TERMINATION AND RECOVERY</u>

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

CECC-EPIP-14- <u>NUCLEAR EMERGENCY PUBLIC INFORMATION ORGANIZATION AND OPERATIONS</u>

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- <u>JOINT INFORMATION CENTER ACTIVATION, SHIFT CHANGE, AND DEACTIVATION</u>

Cancelled - Combined with CECC EPIP-14.

CECC-EPIP-16- <u>CENTRAL EMERGENCY CONTROL CENTER INFORMATION STAFF ACTIVATION, SHIFT CHANGE, AND DEACTIVATION</u>

Cancelled - Combined with CECC EPIP-14.

CECC-EPIP-17- <u>CENTRAL EMERGENCY CONTROL CENTER METEOROLOGIST PROCEDURES</u>

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- <u>TRANSPORTATION AND STAFFING UNDER ABNORMAL CONDITIONS</u>

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- <u>POST ACCIDENT CORE DAMAGE ASSESSMENT</u>

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- CECC TRAINING REQUIREMENTS

Cancelled - replaced by TRN-30

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-EPIPs.

These vehicles are dispatched for environmental monitoring for Site Area Emergency and General Emergency classes. They may be deployed for the Notification of Unusual Event and Alert classes, 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 EPIPs.

For the Site Area Emergency, and General Emergency classes, 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 RadCon 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

A mobile radioanalytical laboratory can be dispatched to the site to be the central point for receipt of samples and for detailed field analysis. Samples obtained by the sampling teams may be returned to the WARL, which has the capability to perform further quantitative and qualitative analysis. The mobile radiological laboratory and the WARL are available at all times and can be operated 24 hours per day.

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 (from other TVA nuclear plants and the National Weather Service first-order stations). Each procedure has an accompanying statement of reliability.

9.2.3.3 Real Time and Forecast Meteorological Data

A meteorologist in the CECC has the responsibility for providing meteorological information to CECC Staff. The dose assessors use this meteorological information to project offsite doses. The meteorological support actions and projection of doses are discussed in detail in CECC-EPIPs. Plume positions are plotted on a site area map.

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 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-EPIPs. 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 resuspended 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

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 preassigned areas to be accounted for and await further instructions from the SED. The preassigned 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-EPIPs.

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 RADCON 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-EPIPs.

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 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-EPIPs 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 to recommend protective action 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 projected radiological conditions offsite. They will provide a coordinated recommendation for a specific protective action considering both plant and offsite conditions. 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-EPIPs 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 appropriate state plan).

FIGURE 10-1
PROTECTIVE ACTION RECOMMENDATIONS

Note 1: If conditions are unknown utilizing the flowchart, then answer NO.

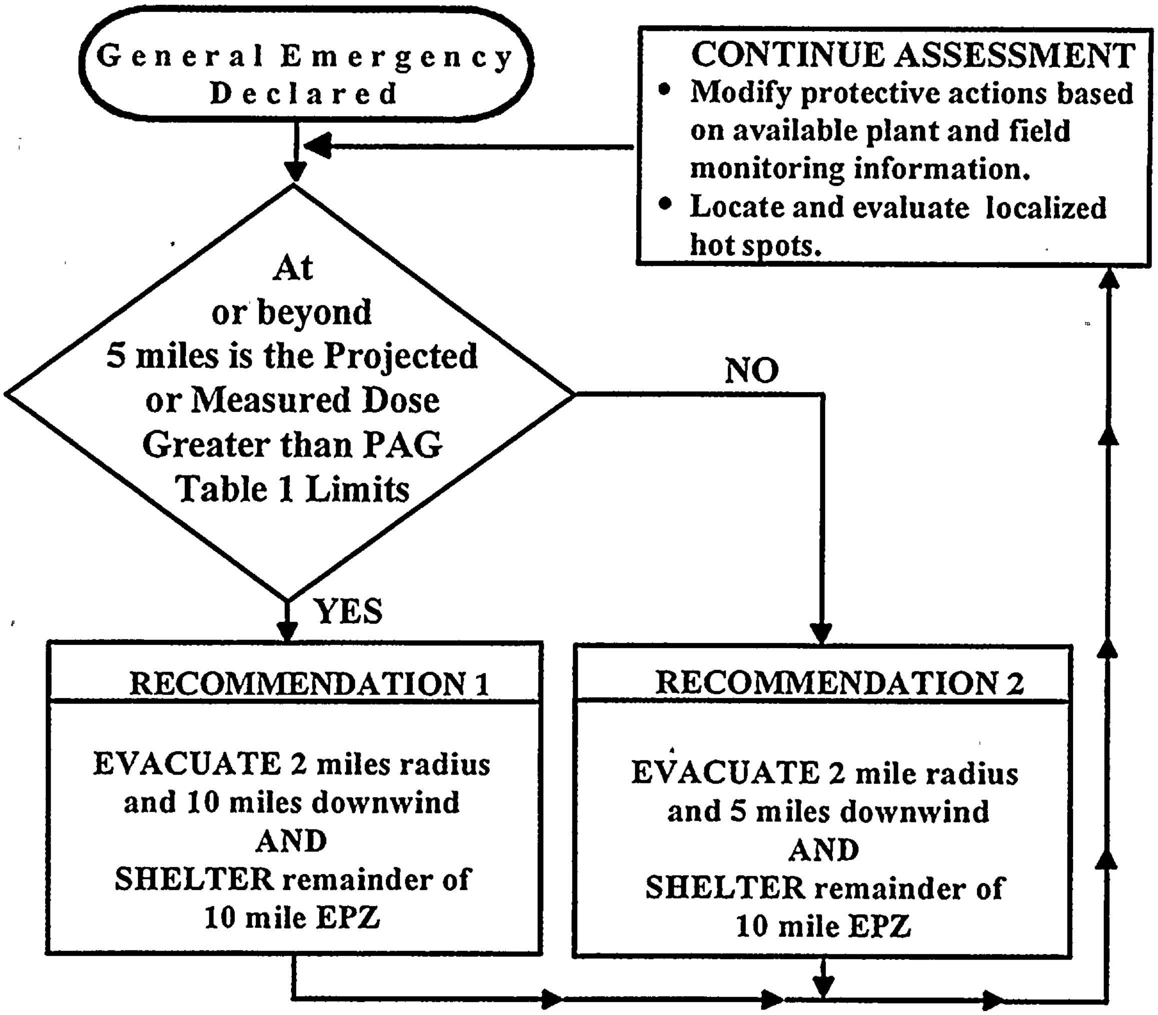


TABLE 1 Protective Action Guides	
TYPE	LIMIT
Measured	3.9E-6 microCi/cc of Iodine 131 or 1 REM/hr External Dose
Projected	1 REM TEDE or 5 REM Thyroid CDE

11.0 RADIOLOGICAL PROTECTION

The RADCON Section at the site is responsible for all RADCON 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 TVAN SPP-5.1, "Radiation Protection Plan." 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 TVAN 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-EPIPs 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 TLD badges are issued and read in accordance with the site RCIs. The electronic dosimeters can be read at any time and the TLD badges can be read by the Central Dosimetry Processing section at SQN. Dose records are maintained on each monitored individual by a computer.

TVAN 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 EPIPs. Drinking water and eating controls are established in TVAN SPP-5.1.

FIGURE 11-1

EMERGENCY WORKER DOSE GUIDANCE

TEDE Dose	Condition
5 rem	All, maintain dose ALARA
10 rad	Protection of valuable property when lower dose not practicable.
Greater than 10 rad	Lifesaving or protection of large populations when lower dose not practicable.

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 unmeasureable 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 TVAN 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 RAD 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), October 15, 1991, page 2-18.

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-EPIPs. 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. RADCON 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 RADCON.

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

13.1.1 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.

13.1.2 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:

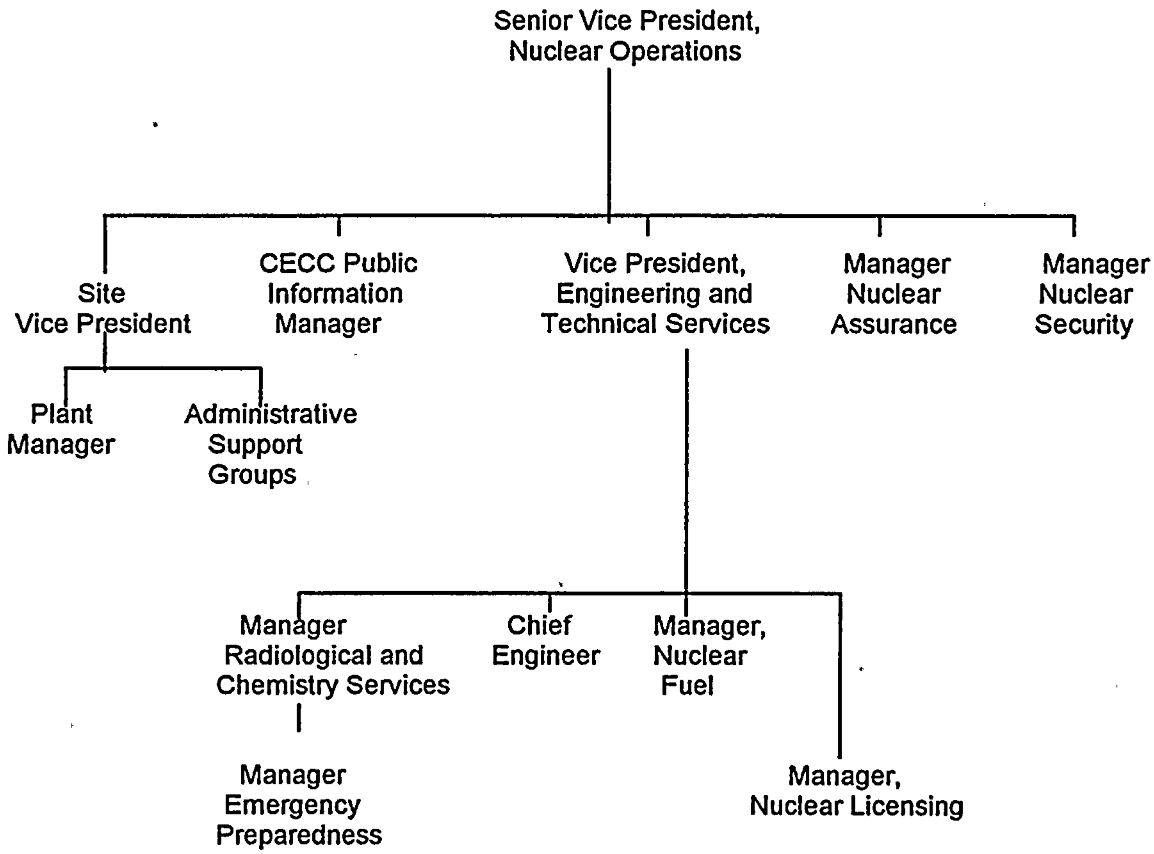
1. Stability of the reactor shutdown condition, i.e., successful progress toward a cold shutdown condition.
2. Integrity of the reactor containment building.
3. Operability of engineered safety systems and decontamination facilities.
4. The availability and operability of a heat sink.
5. The integrity 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 Senior Vice President, Nuclear Operations - 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 Senior Vice President Nuclear Operations.

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

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

13.2.4 Vice President, Engineering and Technical Services - Will manage needed services to the site in the areas of Engineering, Licensing activities, QA activities, Security, and Emergency Preparedness.

13.2.5 CECC Public Information Manager - Acts as an interface between TVA and the news media. They assist the Senior Vice President, Nuclear Operations, 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 Chief Engineer - Will provide needed technical and engineering services to the site.

13.2.7 Manager, Nuclear Fuels - Will provide needed technical services to the site. Technical services available include fuel management and core analysis, core performance, nuclear fuel control and accountability, and startup support.

13.2.8 Manager, Radiological and Chemistry Services - Will provide corporate level guidance and needed radiological services as requested. Services include technical support, dose assessment, and environmental monitoring. Will provide technical support and environs sampling assistance as requested by the State. Will provide technical assistance to the site in the areas of chemistry and environmental issues.

13.2.9 Manager, Emergency Preparedness - Will provide assistance in any aspects of emergency preparedness plans, procedures, coordination, and implementation.

13.2.10 Manager, Nuclear Security - Will provide technical assistance to the site in the area of security.

13.2.11 Manager of Nuclear Licensing - Will provide technical assistance in NRC licensing activities.

13.2.12 Manager of Nuclear Assurance - Will provide technical assistance in Quality Assurance activities.

13.2.13 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-EPIPs 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 Senior Vice President, Nuclear Operations, or his designee will service 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, 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 RadCon Drills

RadCon 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 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 pm 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 Nuclear Quality Assurance, INPO, NRC, FEMA, 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 followup 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

Nuclear Training/plant will provide training in emergency procedures to all permanent plant personnel and applicable nonplant 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.

Nickajack Fire Training Academy 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.

Engineering and Technical Services is responsible for providing agreement hospital and ambulance support training. The sites are responsible for providing fire support training, with assistance from Engineering and Technical Services 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 Page 1 -or- Rev. 1 Page A-1 Rev. 1	NP-REP Appendix A

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, or, whenever an entire page has been added or substantially changed, the change will be denoted by a statement at the bottom of the page.

Formal site 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 responsible site management allowing time for site 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 President, Nuclear Power. All changes to the NP-REP will be approved by the Vice President, Engineering and Technical Services, or his designee.

16.1.4 Distribution

Each NP-REP, its additions, and revisions will be authorized by an approval form and distributed by Administrative Support and Procedures.

Administrative Support and Procedures 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.

Administrative Support and Procedures 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 EPIPs

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:

EPIPs	Approving Authority
CECC	Vice President, Engineering & Technical Services
BFN	Plant Manager, BFN
SQN	Plant Manager, SQN
WBN	Plant Manager, WBN

FIGURE 16-1

UPDATE PROCEDURE FOR NP-REP AND APPENDICES

Draft Proposed Change
to NP-REP or Appendix

Plan Effectiveness Determination (PED)
(In Accordance With 10 CFR 50.54q)

Site Review in Accordance With Plant Approved
Procedures
(Sites Are Not Required
To Review and Approve
Other Site's Appendix)

EP
Review/Coordination and Approval

Vice President, Engineering and Technical Services, or Representative

No — Significant Change
As Determined by
PED — Yes

NRC Review
and Approval

Administrative Support and Procedures
Issues Revision

16.2.2 Periodic Review

The EPIPs are reviewed annually for accuracy, completeness, operational readiness, and compliance with existing regulations by the responsible organization listed below. This review is initiated by Engineering and Technical Services and results are documented.

EPIPs	Organization
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 an 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.

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 by the word "All" under Revised Pages on the cover page.

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. Administrative Support and Procedures will issue the changes to other document holders in accordance with Administrative Support and Procedures document control practices.

FIGURE 16-2

UPDATE PROCEDURE FOR EPIPs

Proposed Change to CECC-EPIP

Review, Coordination, and Approval

Engineering and Technical Services Review and Approval

Administrative Support and Procedures Issues CECC-EPIP Revision to Onsite and Offsite Document Holders

Proposed Permanent Change to Site-EPIP

Review and Coordination

Site Review and Approval According to Approved Site Procedures

Site Issues EPIP Revision to Onsite Document Holders

Administrative Support and Procedures Issues Procedure To Offsite Document Holders

16.2.3.3 CECC-EPIP Changes

In addition to the change mechanism depicted in figure 16-2, in order to ensure that minor changes (e.g., personnel changes, phone numbers, etc.) are rapidly implemented, pen-and-ink changes may be made by the responsible organizations to their procedures in documents which they possess. Pen-and-ink changes will be authorized by the approving authority and documented. The initials of the individual making the pen-and ink change and the date of the change will be clearly marked in the margin adjacent to the change. Such changes will be immediately followed by a formal change request.

16.2.4 Distribution

Each CECC-EPIP or revision will be authorized by an approval form and distributed by Administrative Support and Procedures. 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 with in two weeks to Administrative Support and Procedures.

Each revision will be accompanied by a revised cover page for that procedure. Administrative Support and Procedures 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

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

Nuclear 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 Medical Service, Chattanooga, TN
Athens-Limestone Ambulance Service, Athens, AL
Rhea County 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 Medical Center, Chattanooga, TN
Memorial North Park Hospital, 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 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. TVA agrees to provide two-hours notice prior to any such use and to pay the college for facilities and services provided.

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.

*Revision