ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN) UNIT 1

RESPONSE TO NRC GENERIC LETTER (GL) 88-20, SUPPLEMENT 4
INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE) FOR
SEVERE ACCIDENT VULNERABILITIES

BROWNS FERRY NUCLEAR PLANT UNIT 1 SEISMIC IPEEE REPORT

FACILITY RISK
CONSULTANTS, INC.

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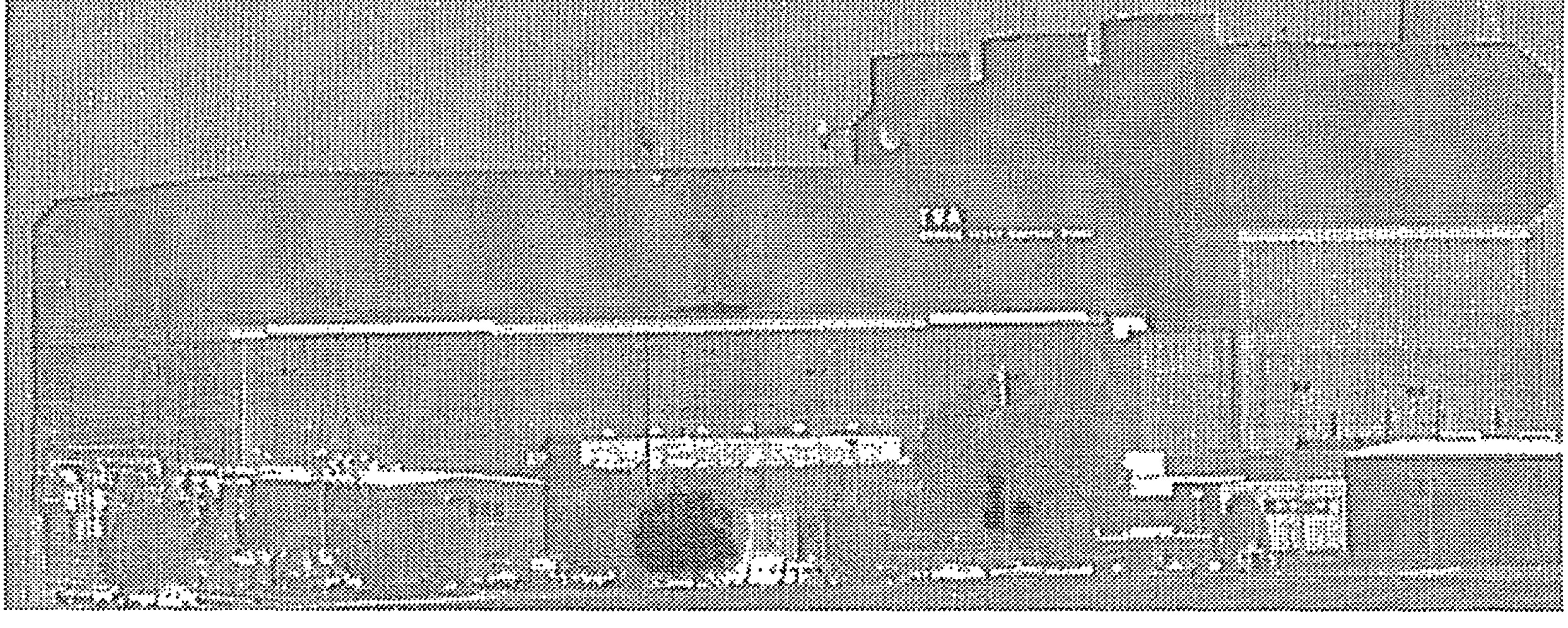
Prepared for:

BROWNS FERRY UNIT 1 RESTART PROJECT

October 2004



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Prepared for:

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Prepared by:

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APPROVAL COVER SHEET

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TABLE OF REVISIONS		
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0	ORIGINAL ISSUE This report contains 104 pages total	10/07/04

EXECUTIVE SUMMARY

The Browns Ferry Nuclear Plant Unit 1(BFN-1) seismic evaluation for Individual Plant Examination for External Events (IPEEE) was performed in accordance with Nuclear Regulatory Commission (NRC) NUREG 1407 guidelines using the seismic margins methodology developed by the Electric Power Research Institute (EPRI NP-6041). The evaluation was performed in a manner similar to that performed for the seismic IPEEE programs for BFN-2 and BFN-3.

The seismic margins assessment was performed in conjunction with evaluations for the resolution of Unresolved Safety Issue (USI) A-46. All seismic IPEEE components were added to the USI A-46 safe shutdown equipment list (SSEL), and evaluated using the screening evaluation criteria in the Seismic Qualification Utility Group (SQUG) Generic Implementation Procedure (GIP). Outliers identified in the USI A-46 screening evaluations have been or are being resolved by maintenance-related work orders and design changes. The seismic margins assessment took credit for these outlier resolutions. The seismic margins assessment also took credit for planned modifications stemming from the BFN-2 and BFN-3 seismic IPEEE programs.

The review level earthquake for the seismic margin assessment for the BFN plants is defined as an earthquake having a response spectrum that matches the median (50% Non Exceedance Probability – NEP) CR-0098 spectral shape anchored to a peak ground acceleration of 0.30g. After capacity screening based on EPRI NP-6041, high confidence low probability of failure (HCLFP) capacity evaluations were performed for a number of plant components. The seismic IPEEE evaluations conclude that the BFN-1 HCLPF capacity is at least as high as 0.30g.

One item was identified as requiring further strengthening during the seismic-induced-fire walkdown screening evaluation per the NUREG 1407 guidelines. Batteries on the emergency lighting battery rack in the cable spreading room lacked end restraints, side restraints, and spacers between batteries. Corrective action was initiated to add the necessary restraints to these batteries. No additional corrective actions were required as a result of the seismic IPEEE evaluations, encompassing the in-plant walkdowns, capacity screening evaluations, HCLPF capacity evaluations, seismic systems interaction reviews, and seismic-induced spray and flooding reviews.

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APPENDICES

1. INTRODUCTION AND METHODOLOGY SELECTION

In the Commission policy statement on severe accidents in nuclear power plants issued in 1985, the Commission concluded, based on available information, that existing plants pose no undue risk to the public health and safety and that there is no present basis for immediate action on any regulatory requirements for these plants. However, the Commission recognized that systematic examinations are beneficial in identifying plant-specific vulnerabilities to severe accidents that could be fixed with low-cost improvements. In 1988 the Commission requested that each licensee conduct an individual plant examination (IPE) for internally initiated events including internal flooding. Many Probabilistic Risk Assessments (PRAs) performed in support of the IPEs indicated that, in some instances, the risk from external events could contribute significantly to core damage.

In July 1990, following public comments and a workshop, the Commission issued Supplement 4 to Generic Letter 88-20 (Reference 1) requesting that each licensee conduct an individual plant examination of external events (IPEEE). The general objectives of the IPEEE are similar to that of the IPE; that is, for each licensee (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.

The staff has concluded that five external events need to be included in the IPEEE: seismic events, internal fires, high winds, floods, and transportation and nearby facility accidents. This report addresses seismic events.

Acceptable methodologies for performing the seismic IPEEE are summarized in NUREG-1407 (Reference 2). This evaluation may be conducted by performing a seismic PRA or a Seismic Margins Assessment (SMA). The SMA methodology was designed to demonstrate sufficient margin over the Safe Shutdown Earthquake (SSE) to ensure plant safety and to find any "weak links" that might limit the plant shutdown capacity to safely withstand a seismic event larger than the SSE or lead to seismically induced core damage. The SMA may in turn be performed using the methodology developed by Lawrence Livermore National Laboratories (LLNL) or by the Electric Power Research Institute (EPRI). TVA has opted to perform a SMA using the EPRI methodology (Reference 3).

Browns Ferry was placed in the focused-scope category for margin assessment. The basic information used was the 1989 Lawrence Livermore National Laboratory seismic hazard estimates for nuclear power plant locations in the eastern United States (Reference 4) and the EPRI hazard study (Reference 6).

New seismic hazard data were published in October, 1993, which demonstrate that the seismic hazard at existing eastern United States nuclear power plants is much less than what the NRC staff originally believed (Reference 5). Supplement 5 to Generic Letter 88-20 (Reference 1) would permit Browns Ferry to change to a modified focused scope classification.

TVA elected to complete the Browns Ferry SMA following NUREG 1407 and EPRI NP-6041 as a focused-scope plant without schedule delays or major scope changes. The new information and extensive seismic evaluation performed for the recent vintage plant were, however, considered when determining the quantity of components selected for high-confidence-of-low-probability of failure (HCLPF) evaluation and the level of evaluation for issues such as soils, structures, and NSSS components.

Detailed plant walkdowns are considered the most cost-effective and beneficial aspect of the SMA program. Combined USI A-46 and IPEEE walkdowns were performed in accordance with the Seismic Qualification Group (SQUG) Generic Implementation Procedure (GIP) (Reference 7), with enhancements based on EPRI NP-6041 (Reference 3). Seismic-induced-flooding evaluations were performed based on the results of a recently-completed seismic II/I spray program implemented as part of the BFN-1 Restart Project. Specific seismic-induced-fire walkdown evaluations were performed as part of the BFN-1 SMA.

The BFN-1 SMA was performed following the same approach as used for the BFN-2 and BFN-3 seismic IPEEE programs. Common system components and BFN-1 components that were included in the BFN-2 and BFN-3 Seismic IPEEE programs were re-evaluated only if the item of equipment is located inside of the BFN-1 Reactor Building. Other components previously addressed in the BFN-2 and BFN-3 programs (such as items in the Diesel Generator Buildings and the Intake Pumping Station) were not re-evaluated.

Two (2) items of equipment were determined to have HCLFP capacity values less than the 0.30g review level earthquake (RLE) in the BFN-2 and BFN-3 Seismic IPEEE program. These include the 4kV/480V transformers 0-OXF-219-TDA and 0-OXF-219-TDB at elevation 583'-6" of the diesel generator buildings for units 1 and 2. TVA has committed to replace these transformers, so the less than 0.30g HCLPF capacity is not considered in the BFN-1 Seismic IPEEE program.

The BFN-1 USI A-46 resolution program is documented in References 15 and 16 for the equipment and relay reviews, respectively. Certain items of equipment were identified as outliers in the USI A-46 program, and the outlier resolution activity for some of these outliers involves work orders and design change notices (DCNs). The BFN-1 Seismic IPEEE program takes credit for these outlier resolution modifications.

1.1 MAPPING OF REGULATORY REPORTING REQUIREMENTS TO CONTENTS OF THIS REPORT

The regulatory reporting requirements for the utility submittals for the seismic IPEEE program are described in NUREG-1407 (Reference 2), Section C.2.2 and Table C.1. The contents of this report include all of the requirements, as summarized in Tables 1-1 and 1-2.

Table 1-1: Submittal Requirements from NUREG-1407, Section C.2.2

Item No.	Description	BFN-1 Documentation in this Report
1.	Description of the methodology and list of important assumptions.	Chapter 1
2.	Summary of the walkdown results.	Chapter 5
	Concise description of the walkdown team and procedures used.	Appendix A
3.	Seismic event trees when NRC SMM is used.	N/A (1)
4.	Description of the success paths and procedures used for their selection and of each component in the controlling success path.	Chapter 3
5.	Any seismically induced containment failures and other containment performance insights.	Chapter 9
6.	Table of HCLPF's.	Chapters 5, 6
7.	Documentation with regard to other seismic issues. (USI A-46)	Chapter 1
8.	Non-seismic failures and human actions for the NRC SMM method.	N/A (1)

Notes: (1) Not applicable because the NRC SMM method is not used for BFN-1

Table 1-2: Submittal Requirements from NUREG-1407, Table C.1

Item No.	Description	BFN-1 Documentation in this Report
3.0	Methodology Selection	Chapter 1
3.1.1	Review of plant information, Screening, and Walkdown	Chapters 1 and 5
3.1.2	System Analysis	Chapter 3
3.1.3	Analysis of Structure Response	Chapter 4
3.1.4	Evaluation of Seismic Capacities of Components and Plant	Chapters 5, 6, and 11
3.1.5	Analysis of Containment Performance	Chapter 9

2. REVIEW OF PLANT INFORMATION

Brief descriptions of the general plant description, ground response spectra, structures, equipment, and distribution systems for Browns Ferry Nuclear Plant (BFN) are presented below. Detailed information and description are contained in existing plant licensing documents, including the Final Safety Analysis Report (FSAR, Reference 10).

2.1 GENERAL PLANT DESCRIPTION

The Browns Ferry site is located on the north shore of Wheeler Lake at river mile 294 in Limestone County in north Alabama. The site is approximately 10 miles southwest of Athens, Alabama, and 10 miles northwest of the center of Decatur, Alabama. The plant consists of three General Electric (GE) boiling water reactors with Mark I containments, each with an electrical output of about 1,100 megawatts. Commercial operation of each unit began on the following dates: Unit 1 on August 1, 1974, Unit 2 on March 1, 1975, and Unit 3 on March 1, 1977.

For the Browns Ferry project, TVA acts as its own engineer-constructor. GE designed, fabricated, and supplied the nuclear steam supply system (NSSS) and nuclear fuel for the plant, as well as the turbine-generators. GE also provided technical supervision for the installation and startup services of this equipment.

Detailed description of the BFN site hydrology, water quality and marine biology is contained in Section 2.4 of the FSAR (Reference 10). The geology and seismology of the general region as well as the plant site are discussed in detail in Section 2.5 of the FSAR (Reference 10).

2.2 GROUND RESPONSE SPECTRA

The BFN licensing-basis Design Basis Earthquake (DBE) ground motion acceleration response spectrum is defined in Sections 2.5.4 and 12.2 of the BFN Final Safety Analysis Report (Reference 10). Seismic requirements for Class I structures are defined in TVA General Design Criteria BFN-50-C-7102 (Reference 11). The horizontal peak ground acceleration (PGA) corresponding to the DBE is 0.20g, defined at the top of sound rock. Vertical ground motion is two-thirds of the horizontal ground motion as specified in the FSAR. The site DBE design ground spectrum is that of a Housner shaped spectrum, anchored to 0.2g PGA.

2.3 STRUCTURES

The design of all structures and facilities (Class I & II) conformed to the applicable general codes or specifications such as Uniform Building Code (UBC); American Institute of Steel Construction (AISC) "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings"; American Concrete Institute (ACI) "Building Code Requirements for Reinforced Concrete" (ACI 318) and "Requirements for Reinforced Concrete Chimneys" (ACI 307); and American Welding Society (AWS) "Structural Welding Code - Steel" (AWS-D.1.1), among others.

Seismic requirements for Class I structures, features, and systems are contained in TVA General Design Criteria BFN-50-C-7102 (Reference 11). Basically, the design of Class I structures were based on the following criteria:

- Operational basis earthquake (OBE) considered a horizontal ground acceleration of 0.10g.
- Design basis earthquake (DBE) considered a horizontal ground acceleration of 0.20g.
- Vertical ground accelerations associated with the OBE and DBE were defined as 2/3 of the corresponding horizontal response spectra.

Class I structures, equipment and safety related piping were designed such that stress and deformation behavior of structures, piping, and equipment were maintained within the allowable limits when subjected to loads such as dead, live, pressure, and thermal, under normal operating conditions combined with the seismic effects resulting from the response to the OBE. These allowable limits are defined in appropriate design standards such as the ASME Boiler and Pressure Vessel Code; American National Standards Institute (ANSI) Code for Pressure Piping ANSI B31.1.0, Power Piping; ACI 318 Building Code Requirements for Reinforced Concrete; and AISC Specification for the Design, Fabrication and Erection of Structural Steel for Buildings. In addition, the stresses that resulted from normal loads and design basis loss-of-coolant accident loads combined with the response to the DBE were limited so that no loss of function occurred and the capability of making a safe and orderly plant shutdown was maintained.

Class II structures were designed in accordance with procedures of the Uniform Building Code for Zone 1. The combined stresses from normal and earthquake loadings were limited to those permitted by the design criteria and applicable industry standards and codes.

2.4 EQUIPMENT SUPPLIED BY THE NSSS VENDOR

General Electric (GE) designed, fabricated, and supplied the nuclear steam supply system (NSSS), turbine-generators, as well as the nuclear fuel for the plant. GE also provided technical supervision for the installation and startup services of this equipment. In general, the modules were designed to withstand and perform their functions during an OBE and a DBE. This qualification was ascertained by either analytical techniques, vibration testing techniques, or a combination of the two. A seismic specification covering the following procedure was made a part of the purchase order.

2.5 EQUIPMENT SUPPLIED BY OTHER THAN NSSS VENDOR

All the Class I instrumentation and electrical equipment were designed and tested or analyzed to ensure their capability to perform their required functions during and after the Design Basis Earthquake (DBE). This includes equipment made by General Electric (GE) as well as that purchased by GE. Suppliers of Class I equipment were required to verify the adequacy of their equipment by submitting test, analytical, or operating experience data. Typically, equipment supplied as part of the original design are in compliance with IEEE-344-71 requirements.

2.6 SEISMIC CLASS I PIPING AND INSTRUMENT TUBING

Analytical and design methods used for seismic Class I piping, including buried piping, and instrument tubing are contained in TVA General Design Criteria BFN-50-C-7103 (Reference 12). All systems were re-evaluated and strengthened as required as part of the BFN-1 Restart Project.

2.7 SEISMIC CLASS I DISTRIBUTION SYSTEMS

Seismic Class I cable trays, conduit, and HVAC duct systems are analyzed and designed in accordance with TVA General Design Criteria BFN-50-C-7104 (Reference 14). All existing cable trays, conduit, and HVAC duct systems and supports have been or are being re-evaluated and strengthened as required as part of the BFN-1 Restart Project. The existing cable trays and conduit were evaluated as part of the USI A-46 resolution program at BFN-1 using the guidelines of the Seismic Qualification Utility Group (SQUG) Generic Implementation Procedure (GIP, Reference 7).

2.8 SEISMIC SPATIAL SYSTEM INTERACTIONS

Browns Ferry has a seismic categorization similar to Regulatory Guide 1.29, using the terminology of Class I and Class II. The term II/I is used to describe physical conditions where Class II components are located above or in proximity to Class I components. Seismic induced spray refers to the possible breach of a fluid pressure boundary due to its own seismic response or its seismic interaction with other plant features. Seismic induced spray is a hazard when there are target Class I components, vulnerable to fluid spray, in the vicinity of the source.

A comprehensive "II/I" seismic interaction verification program was implemented as part of the BFN-1 Restart Project. Seismic spatial interactions (failure, falling, and impact) were evaluated for all Safe Shutdown Equipment List (SSEL) items during the USI A-46 resolution program. Impact-related seismic interactions are further addressed by the TVA BFN Potential Clearance Discrepancy (PCD) evaluation program for piping clearance discrepancies of 3" and under. Seismic-induced spray evaluations were addressed by detailed walkdowns and bounding evaluations (Reference 19) in accordance with TVA Design Criteria BFN-50-C-7306 (Reference 13).

3. SYSTEM DESCRIPTION AND SUCCESS PATH SELECTION

The success path selection and identification of components for the BFN-1 Seismic IPEEE program were based on the previous BFN-2 and BFN-3 seismic IPEEE programs. The SQUG GIP was utilized as guidance in choosing the items and identifying boundary conditions and assumptions.

The seismic safe shutdown equipment list (SSEL) identifies the equipment necessary to maintain operability of those frontline systems required to safely shut down the plant and maintain it in hot shutdown for 72 hours. The relevant plant functions are as follows:

- Reactivity control
- Reactor coolant system inventory control
- Reactor coolant system pressure control
- Decay heat removal

The above functions are assured through evaluation of the systems, structures, and components included in the following frontline systems:

- Reactor protection system (RPS)
- Control rod drive/hydraulic control unit (CRD/HCU) system
- Safety relief valve (SRV) system
- Core spray (CS) system
- Residual heat removal (RHR) system in low pressure injection (LPCI) mode and suppression pool cooling (SPC) mode
- Primary containment isolation

The following support systems are required to ensure frontline system operation:

- AC power system, including the emergency diesel generators
- DC power system
- Residual heat removal service water (RHRSW) system
- Essential equipment cooling water (EECW) system

- Heating, ventilating, and air-conditioning (HVAC) systems for RHR and CS areas, emergency diesel generator rooms, and control room
- Containment atmosphere dilution (CAD) system

Success path logic diagrams (SPLDs) were constructed for the BFN-2 and BFN-3 seismic IPEEE programs based on an understanding of available plant equipment functions as well as the plant normal and emergency operating procedures. The SPLDs were reviewed and agreed upon by Browns Ferry Operations personnel. They were used as a basis for the identification of the equipment to be included on the BFN-2 and BFN-3 SSELs. Equipment selected for inclusion on the SSEL was evaluated in a manner similar to that described in the SQUG GIP (Reference 7). Guidance from EPRI NP-6041 (Reference 3) was also used in the evaluation as well as in preparing the format for the list of components.

The assessment of equipment necessary to maintain the identified functions is made under a set of boundary conditions. Offsite power is assumed to be lost, however, the potential is evaluated for adverse effects should the power were not lost or if it were to be restored. The success paths are capable of maintaining the plant in hot shutdown for a period of 72 hours. The success path development addresses seismically-induced transient events or a seismically-induced one-inch loss of coolant accident (LOCA). Non-seismic components of system availabilities are not addressed for multiple or redundant train systems, but are considered for single train systems.

In addition to the components of the systems discussed above, the structures housing the components included in the above systems are also reviewed. These include seismic Class I structures such as the Reactor Buildings, the Diesel Generator Buildings, and the Intake Pumping Station. All Seismic Class I structures are cast-in-place reinforced concrete structures. The floors are supported on beams and girders which are in turn supported on interior columns and/or exterior walls. Where interior shear walls are installed, the beams and girders are supported on the shear walls. All interior shielding walls and partitions, other than structural shear walls, are either reinforced concrete or concrete block and are not load bearing. The Reactor Buildings (RB) and the Intake Pumping Station (IPS) are founded on sound rock, while the Diesel Generator Buildings (DGB) are founded on 3 feet of compacted soil and 32 feet of crushed stone above the sound rock.

4. SEISMIC MARGIN EARTHQUAKE DEMAND

4.1 INTRODUCTION

In-structure response spectra (IRS) corresponding to the Review Level Earthquake (RLE) are required for the Seismic Margin Assessment (SMA). For Browns Ferry, the RLE is defined as an earthquake having a response spectrum that matches the median CR-0098 spectral shape (Reference 8) anchored to a peak ground acceleration of 0.30g.

The IRS for the Reactor Building (RB), Diesel Generator Building (DGB) and Intake Pumping Station (IPS) were obtained from the A-46 spectra using scaling procedures, following the recommendations given in Reference 3. The procedure used to generate the IRS is described briefly below. A more complete treatment of the subject can be found in Reference 9.

4.2 DESCRIPTION OF SCALING PROCEDURE

The dominant mode scaling procedure described in Reference 3 is used here since the input motion spectra for the A-46 and the SMA earthquakes have similar shapes over the relevant range of frequencies. This procedure uses a scale factor for the spectral amplitude change when the input motion is changed.

The factor for the spectral amplitude change in the response of the combined, soil-structure system is controlled by several parameters. It can be defined as the ratio between the spectral ordinates of the A-46 and the SMA acceleration input spectra at the predominant frequencies and damping ratios of the combined, soil-structure system. The scaling factor, R_{Sa}, is

$$R_{Sa} = \frac{S_a\,(f_{SMA}, \beta_{SMA})}{S_a\,(f_{A\text{-}46}, \beta_{A\text{-}46})}$$

where $S_a\,(f_{SMA}, \beta_{SMA})$ is the spectral ordinate of the input acceleration response spectrum for the SMA review level earthquake at the predominant frequency, f_{SMA}, and equivalent damping ratio, β_{SMA}, of the soil-structure system, and Sa $(f_{A\text{-}46}, \beta_{A\text{-}46})$ is the spectral ordinate of the input acceleration response spectra for the design basis earthquake at the predominant frequency, f_{A-46}, and equivalent damping ratio, β_{A-46}, of the soil-structure system.

The predominant frequency of the soil-structure system was estimated as the frequency corresponding to the peak spectral acceleration in the A-46 in-structure response spectra. The damping for the SMA RLE was taken as 7% for reinforced concrete structures at RB and DGB, and 5% for IPS. The level of damping was estimated as the sum of two parts: (1) damping of 5% for structures founded on rock such as RB and IPS, assuming the structure is not highly stressed at the RLE, and (2) 2% additional damping to reflect the material and radiation damping of the soil for soil-supported structures such as DGB. Note that a damping value of 7% was assumed for RB based on the estimated stress state of the structure at the RLE.

The vertical input ground motion specified for seismic IPEEE is defined, according to Reference 8, as two-thirds of the horizontal motion. Since the vertical A-46 IRS is also defined as two-thirds of the horizontal spectra, the scale factors used to obtain the vertical SMA IRS are the same as for the horizontal case.

5. SEISMIC MARGIN ASSESSMENT SCREENING AND WALKDOWN

5.1 SEISMIC REVIEW TEAM

The Seismic Review Team (SRT) was assembled following guidance provided in Reference 3, drawing on the experience and expertise of Facility Risk Consultants, Inc..

Each walkdown team included a minimum of two Seismic Capability Engineers (SCEs) members who had completed the Seismic Qualification Utility Group (SQUG) Walkdown Screening and Seismic Evaluation training course. In addition, some members also attended the seismic IPEEE training course (see Appendix A). The following persons participated in the SRT walkdowns and evaluations:

- John O. Dizon
- Stephen J. Eder
- Farid Elsabee
- Rickard Tiong
- Robert D. Hookway
- Jess O. Betlack (no participation in walkdowns)

Among the team members there is strong experience in each of the areas listed below:

- Knowledge of the failure modes and performance of structures, tanks, piping, process and control equipment, and active electrical and mechanical components during strong earthquakes.
- Knowledge of nuclear design standards, seismic design practices, and equipment qualification practices for nuclear power plants.
- Ability to perform fragility evaluations including structural/mechanical analysis of essential elements of nuclear power plants.
- General knowledge of the plant system functions and normal and emergency operating procedures.

The resumes for each of the seismic walkdown team members are presented in Appendix A.

5.2 WALKDOWN PREPARATION AND PRE-SCREENING

The purpose of pre-screening was to ensure efficiency in the walkdowns and subsequent evaluations by completing the maximum amount of data entry in advance of the walkdown. This was accomplished by incorporating existing data onto the seismic IPEEE and SQUG GIP Screening Evaluation Work Sheet (SEWS) documentation forms prior to the walkdowns. Data that was reviewed consisted of the Final Safety Analysis Report, design criteria, stress reports, equipment qualification reports (testing and analysis), structures and equipment support drawings, equipment location drawings, anchorage calculations, and records from other related programs previously performed at Browns Ferry (including the BFN-2 and BFN-3 USI A-46 and seismic IPEEE programs). An initial walkdown was performed by the SRT as part of the pre-screening task to review the SSEL and to group items according to the "Rule of the Box."

Pre-screening was performed with three purposes in mind:

- To identify critical failure modes to be specifically reviewed on the walkdown.
- Assemble qualification and installation data for use as a basis for screening in the margins review.
- To provide data to be utilized in HCLPF calculations.

A considerable amount of information was extracted from the existing documentation and was subsequently recorded on the Screening and Evaluation Work Sheets (SEWS) prior to commencing the detailed walkdowns. Information entered into SEWS during prescreening was intended to provide available data to the SRT to assist in equipment screening.

5.3 SCREENING CRITERIA

The Browns Ferry seismic IPEEE was completed following the EPRI seismic margins methodology recommended by NUREG-1407 (Reference 2) for a focused scope plant.

Civil structures, equipment and subsystems were screened following the methodology provided in EPRI NP-6041 (Reference 3) for a focused-scope plant. Screening criteria are provided in Tables 2-3 and 2-4 of Reference 3 for civil structures and equipment and subsystems, respectively. The criteria corresponding to 5 percent-damped peak spectral acceleration less than 0.8g were used for Browns Ferry based on the RLE. The guidelines are supplemented by Appendix A of the EPRI seismic margins methodology (Reference 3) which provides the basis for the seismic capacity screening guidelines. Walkdown data sheets from the SQUG GIP augmented to include additional review per EPRI NP-6041 were used during the SRT walkdowns.

5.4 SEISMIC MARGIN WALKDOWNS

The walkdowns were performed following the procedures of the SQUG GIP supplemented by EPRI NP-6041 and NUREG-1407. The walkdowns concentrated on the strength and load path of the equipment as well as function and integrity. The review of equipment anchorage was a prime objective for the walkdown teams. The anchorage evaluation addressed both physical attributes of the anchorage installation and the capacity relative to other success path items as well as the postulated demand at the RLE.

Interaction reviews were performed to identify falling, impact, spray and flood, and seismic-induced-fire issues that could affect success path items. Falling, impact, and spray and flood evaluations were specifically performed for every item on the SSEL. The seismic-induced-spray program was performed separately, on an area-by-area basis, as part of the BFN-1 Restart Project. After the equipment walkdowns were completed, an additional focused seismic-induced-fire walkdown was performed on an area-by-area basis.

Suspended systems including cable trays, conduit, and ductwork were walked down separately from the SSEL seismic margin walkdowns, as part of USI A-46 and the BFN-1 Restart Project. The ceiling above the control room was reviewed to verify if the light fixtures and ceiling grid were adequately supported, and to evaluate the potential for ceiling panels to fall especially at the expansion joint between BFN-1 and BFN-2.

Containment penetrations were reviewed on an area basis to identify anomalies that might affect containment performance. Concerns such as falling and differential building displacement were considered. Displacement concerns between the containment shell and internal structure were also reviewed. Containment isolation valves were added to the SSEL.

Following the completion of the plant A-46/IPEEE walkdowns, SRT members convened to complete the IPEEE ranking and screening task. SRT members reviewed the SEWS and categorized components into the following resolution categories:

- Screened out by the SRT based on Table 2-4 of EPRI NP-6041 or A-46 evaluations with factor of safety greater than 2
- Screened out pending resolution of A-46 outliers
- Candidate for HCLPF evaluation identified during walkdown

As a result of this screening process, items were selected for HCLPF evaluation. A summary of this screen is presented in Table 6-1. These items are considered to represent the most vulnerable issues observed by the SRT that have not been identified for repair. Other items may have comparable seismic capacity but are considered bounded by the selected items. These items identified for HCLPF evaluation were grouped into the following seven (7) categories based on similarity of the equipment and identified controlling failure mode. HCLPF evaluations are summarized in Section 6.

Group 1:	Anchorage of Motor Control Centers
Group 2:	Anchorage of Instrument racks
Group 3:	Anchorage of I&C Panels and Cabinets
Group 4:	Anchorage of Main Control Room Cabinets
Group 5:	Anchorage of RHR and CS Pumps
Group 6:	Anchorage of RHR Heat Exchangers
Group 7:	Anchorage of Remote Control Cabinets

The HCLPF capacity evaluations for other categories of equipment such as transformers, low voltage switchgear, battery racks, battery chargers, and the RHRSW pumps were performed under the BFN-2 and BFN-2 seismic IPEEE programs. The SRT reviewed these calculations and concurred with the conclusions (HCLPF > 0.30g), so no new HCLPF capacity evaluations were performed for these categories of equipment.

5.5 STRUCTURES

Table 5-1 lists civil structures following the format of EPRI NP-6041, Table 2-3, along with screening results for the Browns Ferry plant. All Browns Ferry Category I structures are screened from further review based on Reference 3, Table 2.3 and Section 12 of the FSAR. All of the buildings were screened out in the BFN-2 and BFN-3 Seismic IPEEE programs, and there have been no significant changes to these building structures since that time. A brief description of each of the buildings within seismic IPEEE success paths is provided in the following subsections.

5.5.1 Reactor Building

This plant uses a separate reactor building for each nuclear unit. The reactor building encloses its reactor and pressure suppression primary containment and provides secondary containment during power operation. The building also serves as the main containment during reactor refueling and maintenance operations when the primary containment is open. Browns Ferry does not have a separate control building; rather, the control room and associated electrical rooms are an integral part of the reactor building.

The reactor building is primarily of reinforced concrete shear wall and floor slab construction, with concrete beams and columns provided for vertical load support. The foundation bears on bedrock at approximately Elevation 519', with crushed rock and compacted soil backfill to Elevation 595'. The light bulb-shaped drywell is also constructed from reinforced concrete that is cast integrally with the rest of the reactor building. The internal structures within the drywell include the reactor pedestal and sacrificial shield wall. Structural steel framing above the refueling floor at Elevation 664' supports the roof and the crane girder. Lateral load resistance is provided by moment frames in the N-S direction and braced frames in the E-W direction.

The dynamic seismic response of the reactor building in the original design analysis was determined using 2-D lumped mass stick models. Mass was lumped at the five floors, the roof, the suppression chamber support, and the crane rail. Structural stiffness was represented by equivalent beam properties between each of the masses. Dynamic seismic response of the drywell internals was determined by a lumped mass mathematical model coupling the internals to the building. Included in this model were the reactor pressure vessel, the reactor pedestal and the sacrificial shield wall, as well as the building.

5.5.2 Diesel Generator Buildings

The diesel generator building for Units 1 and 2 is located on the west side of the reactor building. It is isolated from other structures by a two-inch expansion joint. The diesel generator building is of reinforced concrete construction with concrete floor slabs. The foundation bears on three feet of compacted soil backfill. Beneath the soil backfill to bedrock is a crushed rock backfill. The structure is partially embedded, the south wall facing soil for its entire height.

Dynamic seismic response of the diesel generator building in the original design analysis was determined using 2-D lumped mass stick models including translational and rotational soil springs. Soil spring stiffnesses were obtained using finite element analyses of the foundation conditions. The soil-crushed rock backfill was assumed to amplify bedrock ground motions by a factor of 1.6.

The diesel generator building for Unit 3 is located on the east side of the reactor building. In other respects, it is similar to the diesel generator building for Units 1 and 2. The dynamic seismic response of the Unit 1/2 building was applied to the Unit 3 building.

5.5.3 RHR Service Water Intake Structure (Intake Pumping Station)

The residual heat removal service water intake structure is a single structure serving all three units. It is constructed from reinforced concrete walls and slabs. The structure is founded on bedrock with soil backfilled on three sides to the roof at Elevation 565'. Discontinuous subfloors occur at Elevations 540', 542' and 550'. The structure is symmetrical in the transverse (N-S) direction with several walls resisting loads. In the longitudinal (E-W) direction, two walls on the north side resist seismic forces. The south wall is discontinuous below Elevation 537' to permit cooling water intake. The intake structure was designed for 0.2g design basis earthquake.

5.5.4 Reinforced Concrete Chimney

The reinforced concrete chimney stands 600 feet high and varies in diameter from 62 feet at the base to 6 feet at the top. Internal structures housed within the chimney bear on the same foundation and are seismically separated by expansion joints at the floor slab. The chimney is reinforced by vertical and hoop steel. The foundation is anchored by steel reinforcement grouted in holes drilled 23 feet into bedrock. Seismic shear and moment envelopes were developed in the original design analysis by subjecting a dynamic model of the chimney to the 1940 El Centro and seven other earthquake records. Seismic loads governed design of the chimney from 460 feet above the base to the top.

5.5.5 Turbine Building

The turbine building is located north of the reactor building. These structures are separated by a two-inch expansion joint. The turbine building was designed as a Class II structure. Below the operating floor at Elevation 617', the turbine building is constructed of reinforced concrete moment frames, shear walls and floor slabs. The turbine-generators are supported by pedestals that are isolated from the floor slabs. Above the operating floor, structural steel framing is used. Resistance to lateral loads is provided by braced frames in the N-S direction and moment frames in the E-W direction. Horizontal roof bracing transfers in-plane roof forces to the vertical elements. The turbine building could suffer damage to the moment resisting frames under the review level earthquake; however, the SRT judged that total collapse sufficient to damage equipment or systems inside the reactor building was not credible.

5.6 SOILS EVALUATION

The structures housing safe shutdown components are either founded directly on rock or on crushed rock backfill over rock. Soil failure is deemed not a significant issue based on a review of the FSAR and is screened per Revision 5 of Generic Letter 88-20 (Reference 1).

5.7 NSSS REVIEW

Each nuclear unit includes a single cycle, forced circulation, boiling water reactor supplied by General Electric. The reactor and primary coolant system components were designed for 0.2g design basis earthquake. The NSSS system and supports are screened from further review per Revision 5 of Generic Letter 88-20 (Reference 1).

The control rod drive (CRD) mechanisms are cantilevered vertically from the bottom of the reactor shell. The CRD housing ends are supported by rod-hung restraints. The primary purpose of the restraints is to support the CRD housings vertically in the event of a CRD housing failure. The restraints and CRD housing ends are joined with a bolted clamp and plate mechanism which results in an interconnected grid work, and also allows for disassembly in the event repair is required. This grid work is captured laterally by a GE-designed restraint beam assembly, which is attached to the reactor pedestal interior. Based on the demonstration of lateral support for the CRD housing ends, this issue is screened from further review.

5.8 DISTRIBUTION SYSTEMS

The following sections address the distribution systems; cable tray and conduit, HVAC duct and piping.

5.8.1 Cable Tray and Conduit

Cable trays and conduit were reviewed on an area-by-area basis as part of the USI A-46 program to identify any anomalies that could lead to failure. A few items were identified as potential outliers and will be dispositioned by analysis and/or modification as appropriate under the A-46 program. Cable trays and conduit are screened from further review based on Appendix A of Reference 3, and SRT walkdowns.

5.8.2 HVAC Duct

All Class I HVAC ducting and supports were specifically reviewed under the BFN-1 Restart Project in accordance with TVA General Design Criteria BFN-50-C-7104 (Reference 14) by members of the SRT. This included design of upgrades as required, in order to achieve compliance with the criteria. Based on this review and the Design Change Notices (DCNs) in progress, the BFN-1 HVAC ducts are screened from further following the guidance in Appendix A of Reference 3.

5.8.3 Piping

Piping systems were reviewed on an area basis during SRT equipment and subsystem walkdowns. The SRT looked for any anomalies related to potential displacement induced failure modes. No such anomalies were observed.

Additionally, the SRT looked for potential failure modes of piping system appurtenances such as instrument tubing and associated instruments, vent valves and drain valves. Seismic interaction and seismic anchor motion were considered potential failure modes for small bore lines attached to larger piping systems. No anomalies noted that could lead to the loss of a pressure boundary of a success path list system were observed.

Containment penetrations were also reviewed on an area basis to identify any anomalies that may affect containment performance. Anomalies such as seismic interaction (falling) and differential building displacement were considered. No anomalies that could affect containment performance were observed.

Browns Ferry piping was screened from further review based on Appendix A of Reference 3, and SRT walkdowns. In addition, all BFN-1 Class 1 piping and instrument tubing is being re-evaluated and upgraded as required, in accordance with TVA General Design Criteria BFN-50-C-7103, as part of the BFN-1 Restart Project.

5.9 OTHER COMPONENTS

5.9.1 Masonry Walls

Masonry walls were inspected and evaluated in response to IE Bulletin 80-11 during the 1980's. Details of construction were confirmed and the walls were evaluated. As a result of this evaluation, some masonry walls were modified by the addition of bracing.

Evaluations performed for the IE Bulletin 80-11 response formed the basis for the IPEEE review. These evaluations documented the as-built conditions including:

- Rebar details
- Anchorage to other structural members
- Attachment details
- Additional loading such as electrical system components
- Bracing details

All masonry walls near equipment on the SSEL were reviewed for IPEEE. All of the walls had been previously reviewed for IE Bulletin 80-11 program. Evaluations were performed in accordance with EPRI NP-6041. Wall HCLPF capacities were determined by modifying the IE Bulletin 80-11 calculations to reflect the scaled IPEEE seismic demand and to remove conservatisms as applicable.

Most of the walls were reinforced. All of the walls except for one had HCLPF capacities greater than 0.3g using the evaluation guidelines of EPRI NP-6041. The one wall had a (preliminary) HCLPF capacity of 0.27g. It was determined that the failure mode of this wall would be forming a plastic hinge at mid-height leading to a vertical collapse rather than lateral tipping. The distance between the wall and the nearby SSEL equipment was judged sufficient that there would be no impact. Therefore, this wall was also screened out. This is the same methodology used for the BFN-2 and BFN-3 Seismic IPEEE Program.

The unreinforced walls were very low (three courses high) walls set in spaces between the top of concrete walls and concrete slabs. The HCLPF capacities for these walls were well above 0.3g.

A biased sample of the most critical walls was generated for the IPEEE review by reviewing analyses performed under both the initial and subsequent programs.

5.9.2 Control Room Ceiling

A seismic upgrade program was implemented for the Main Control Room (MCR) ceiling as part of the BFN-1 Restart Project in consideration of control room habitability improvements. This seismic upgrade included modification of the expansion joint gap between BFN-1 and BFN-2, and elimination of the rattle space around the perimeter walls. A HCLPF capacity evaluation was performed to evaluate the modified gap. The evaluation concluded that the HCLPF capacity is greater than 0.30g.

Table 5-1

SUMMARY OF CIVIL STRUCTURES SEISMIC MARGIN EVALUATION (FORMAT FOLLOWS EPRI NP-6041, TABLE 2-3)

Type of Structure	IPEEE HCLPF Evaluation
Concrete containment	Screened based on EPRI NP-6041, Table 2-3
Containment internal structure	Screened based on EPRI NP-6041, Table 2-3. The structure was designed for greater than 0.1g.
Shear walls, footing and containment shield walls	Screened based on EPRI NP-6041, Table 2-3. The walls were designed for greater than 0.1g.
Diaphragms	Screened based on EPRI NP-6041, Table 2-3. Diaphragms were designed for greater than 0.1g.
Category I concrete frame structures	Screened based on EPRI NP-6041, Table 2-3. Concrete frame structures were designed for greater than 0.1g.
Masonry walls	Masonry walls are reviewed based on past upgrade programs.
Control room ceilings	Control room ceiling is reviewed based on seismic upgrade program.
Impact between structures	Screened based on EPRI NP-6041, Table 2-3.
Category II structures with safety-related equipment or with potential to fail Category I structures	Screened based on no SSEL items located within Seismic Category II structures.
Dams, levees, dikes	Not required based on Supplement 5 to Generic Letter 88-20.
Soil failure modes	Not required based on Supplement 5 to Generic Letter 88-20.

6. ASSESSMENT OF ELEMENTS NOT SCREENED OUT

Seven (7) groups of equipment items were selected for HCLPF capacity evaluation by the SRT. A total of ninety (90) equipment components are addressed in the HCLPF capacity evaluation calculations. The selected equipment items are discussed below along with the results of the HCLPF evaluations. The evaluation results are summarized in Table 6-1.

6.1 MOTOR CONTROL CENTERS

Six (6) bounding HCLPF calculations were performed on MCCs in order to adequately envelope the various configurations, elevations, etc., found at Browns Ferry without introducing undo conservatism. HCLPF capacities were calculated based on anchorage demand versus capacity. The evaluation determined a HCLPF anchorage capacity in excess of 0.3g for the MCCs. In two (2) cases, the evaluations are based on the improved anchorage (top bracing) implemented as a result of USI A-46 outlier resolution modifications.

6.2 INSTRUMENT RACKS

Fifteen (15) bounding HCLPF capacities were calculated for instrument racks. The HCLPF capacities were calculated based on anchorage demand versus capacity. The evaluation determined a HCLPF anchorage capacity in excess of 0.3g for the instrument racks.

6.3 I&C PANELS AND CABINETS

Thirty-four (34) bounding HCLPF capacities were calculated for instrumentation and control (I&C) panels and cabinets. The HCLPF capacities were calculated based on anchorage demand versus capacity. The evaluation determined a HCLPF anchorage capacity in excess of 0.3g for the I&C panels and cabinets.

6.4 MAIN CONTROL ROOM CABINETS

Twenty-one (21) bounding HCLPF capacities were calculated for Main Control Room (MCR) cabinets. The HCLPF capacities were calculated based on anchorage demand versus capacity. The evaluation determined a HCLPF anchorage capacity in excess of 0.3g for the MCR cabinets.

6.5 RHR & CS PUMPS

HCLPF capacities were calculated for each of the eight (8) pumps. The HCLPF capacities were calculated based on anchorage demand versus capacity. The evaluation determined a HCLPF anchorage capacity in excess of 0.3g for the RHR & CS pumps.

6.6 RHR HEAT EXCHANGERS

HCLPF capacities were calculated for each of the four (4) RHR heat exchangers. The HCLPF capacities were calculated based on anchorage demand versus capacity. The evaluation determined a HCLPF anchorage capacity in excess of 0.3g for RHR heat exchangers.

6.7 REMOTE CONTROL CABINETS

HCLPF capacities were calculated for both of the remote control cabinets. The HCLPF capacities were calculated based on anchorage demand versus capacity. The evaluation determined a HCLPF anchorage capacity in excess of 0.3g for the remote control cabinets.

TABLE 6-1
HCLPF EVALUATION RESULTS

Group	Identification Number	Component	HCLPF Capacity
1. Motor Control Centers	1-BDBB-281-0001A	250V DC RMOV BOARD 1A	Anchorage modified. HCLPF > 0.3g
	1-BDBB-281-0001B	250V DC RMOV BOARD 1B	HCLPF > 0.3g
	1-BDBB-281-0001C	250V DC RMOV BOARD 1C	HCLPF > 0.3g
	1-BDBB-265-0001B	480V RB VENT BD 1B	HCLPF > 0.3g
	1-BDBB-268-0001A	480V RMOV BD 1A	Anchorage modified. HCLPF > 0.3g
	1-BDBB-268-0001B	480V RMOV BD 1B	HCLPF > 0.3g
2. Instrument Racks	1-LPNL-925-005A	Local Panel 25-5A	HCLPF > 0.3g
	1-LPNL-925-005B	Local Panel 25-5B	HCLPF > 0.3g
	1-LPNL-925-005D	Local Panel 25-5-001	HCLPF > 0.3g
	1-LPNL-925-006A	Local Panel 25-6A	HCLPF > 0.3g
	1-LPNL-925-006B	Local Panel 25-6B	HCLPF > 0.3g
	1-LPNL-925-006D	Local Panel 25-6-001	HCLPF > 0.3g
	1-LPNL-925-0059	Local Panel 25-59	HCLPF > 0.3g
	1-LPNL-925-0062	Local Panel 25-62	HCLPF > 0.3g
	1-LPNL-925-0001	Local Panel 25-1	HCLPF > 0.3g
	1-LPNL-925-0060	Local Panel 25-60	HCLPF > 0.3g
	1-LPNL-925-247A	Local Panel 1-25-247A	HCLPF > 0.3g
	1-LPNL-925-247B	Local Panel 1-25-247B	HCLPF > 0.3g
	1-LPNL-925-0007A	Local Panel 1-25-7A	HCLPF > 0.3g
	1-LPNL-925-0007B	Local Panel 1-25-7B	HCLPF > 0.3g
	1-LPNL-925-0223	Local Panel 1-25-223	HCLPF > 0.3g
3. I&C Panels	1-LPNL-925-044A/11	COMMON BD LOGIC RELAY PANEL 25-44-A11	HCLPF > 0.3g
	1-LPNL-925-044A/12	COMMON BD LOGIC RELAY PANEL 25-44-A12	HCLPF > 0.3g
	1-LPNL-925-044B/11	COMMON BD LOGIC RELAY PANEL 25-44-B11	HCLPF > 0.3g
	1-LPNL-925-044B/12	COMMON BD LOGIC RELAY PANEL 25-44-B12	HCLPF > 0.3g
	1-PNLA-009-0015	RPS CH A (DIV I)	HCLPF > 0.3g
	1-PNLA-009-0016	RPS CH A, B, C, D	HCLPF > 0.3g
	1-PNLA-009-0017	RPS CH B (DIV II)	HCLPF > 0.3g
	1-PNLA-009-0018	FW & RECIRC PNL	HCLPF > 0.3g
	1-PNLA-009-0019	PROCESS INSTR PNL	HCLPF > 0.3g
	1-PNLA-009-0028	CRD SELECT RELAY AUX PNL	HCLPF > 0.3g
	1-PNLA-009-0030	AUTO BLOWNDOWN AUX PNL	HCLPF > 0.3g
	1-PNLA-009-0032	RHR, CS, & HPCI (CH A) PNL	HCLPF > 0.3g
	1-PNLA-009-0033	RHR, CS, & HPCI (CH B) PNL	HCLPF > 0.3g
	1-PNLA-009-0039	HPCI RELAY AUX PNL	HCLPF > 0.3g
	1-PNLA-009-0042	MSIV (INBOARD) DIV II PNL	HCLPF > 0.3g
	1-PNLA-009-0043	MSIV (OUTBOARD) DIV II PNL	HCLPF > 0.3g
	1-PNLA-009-0081	DIV I ECCS ATU CABINET	HCLPF > 0.3g

TABLE 6-1: HCLPF EVALUATION RESULTS, CONTINUED

Group	Identification Number	Component	HCLPF Capacity
3. I&C Panels, Continued	1-PNLA-009-0082	DIV II ECCS ATU CABINET	HCLPF > 0.3g
	1-PNLA-009-0083	RPS ATU CAB	HCLPF > 0.3g
	1-PNLA-009-0084	RPS ATU CAB	HCLPF > 0.3g
	1-PNLA-009-0085	RPS ATU CAB	HCLPF > 0.3g
	1-PNLA-009-0086	RPS ATU CAB	HCLPF > 0.3g
	1-PNLA-009-0087	DIV I TORUS TEMP MONITORING	HCLPF > 0.3g
	1-PNLA-009-0088	DIV II TORUS TEMP MONITORING	HCLPF > 0.3g
	1-PNLA-009-0093	NEW PNL (INSTALLED BY DCN W19433)	HCLPF > 0.3g
	1-PNLA-009-0036A	PANEL 1-9-36A	HCLPF > 0.3g
	1-PROT-099-0001A1	RPS CIRCUIT PROTECTOR CABINET 1A1	HCLPF > 0.3g
	1-PROT-099-0001A2	RPS CIRCUIT PROTECTOR CABINET 1A2	HCLPF > 0.3g
	1-PROT-099-0001B1	RPS CIRCUIT PROTECTOR CABINET 1B1	HCLPF > 0.3g
	1-PROT-099-0001B2	RPS CIRCUIT PROTECTOR CABINET 1B2	HCLPF > 0.3g
	1-PROT-099-0001C1	RPS CIRCUIT PROTECTOR CABINET 1C1	HCLPF > 0.3g
	1-PROT-099-0001C2	RPS CIRCUIT PROTECTOR CABINET 1C2	HCLPF > 0.3g
	0-LPNL-925-0045A	PANEL 25-45A	HCLPF > 0.3g
	0-LPNL-925-0045B	PANEL 25-45B	HCLPF > 0.3g
4. MCR Cabinets	1-PNLA-009-0023/1	ELECTRICAL CONTROL PANEL 1-9-23-1	HCLPF > 0.3g
	1-PNLA-009-0023/2	ELECTRICAL CONTROL PANEL 1-9-23-2	HCLPF > 0.3g
	1-PNLA-009-0023/3	ELECTRICAL CONTROL PANEL 1-9-23-3	HCLPF > 0.3g
	1-PNLA-009-0023/4	ELECTRICAL CONTROL PANEL 1-9-23-4	HCLPF > 0.3g
	1-PNLA-009-0023/5	ELECTRICAL CONTROL PANEL 1-9-23-5	HCLPF > 0.3g
	1-PNLA-009-0023/6	ELECTRICAL CONTROL PANEL 1-9-23-6	HCLPF > 0.3g
	1-PNLA-009-0023/7	ELECTRICAL CONTROL PANEL 1-9-23-7	HCLPF > 0.3g
	1-PNLA-009-0023/8	ELECTRICAL CONTROL PANEL 1-9-23-8	HCLPF > 0.3g
	1-PNLA-009-0009	I&C BUS 1A (CAB 2 OF PNL 1-9-9)	HCLPF > 0.3g
	1-PNLA-009-0009	I&C BUS 1B (CAB 3 OF PNL 1-9-9)	HCLPF > 0.3g
	1-PNLA-009-0003A	REACTOR SD & CONT. COOLING PNL	HCLPF > 0.3g

TABLE 6-1: HCLPF EVALUATION RESULTS, CONTINUED

Group	Identification Number	Component	HCLPF Capacity
4. MCR Cabinets, Continued	1-PNLA-009-0003B	REACTOR SD & CONT. COOLING PNL	HCLPF > 0.3g
	1-PNLA-009-0004	CLEANUP & RECIRC PNL	HCLPF > 0.3g
	1-PNLA-009-0005	REACTOR CONTROL PNL	HCLPF > 0.3g
	1-PNLA-009-0006	FW & COND. PNL	HCLPF > 0.3g
	1-PNLA-009-0021	TEMP RECORDING PNL	HCLPF > 0.3g
	1-PNLA-009-0054	CONTAINMENT ATM. DILUTION PNL	HCLPF > 0.3g
	1-PNLA-009-0055	CONTAINMENT ATM. DILUTION PNL	HCLPF > 0.3g
	1-PNLA-009-012	PANEL 1-9-12	HCLPF > 0.3g
	1-PNLA-009-0020	PANEL 1-9-20	HCLPF > 0.3g
	1-PNLA-009-0008	PANEL 1-9-8	HCLPF > 0.3g
5. RHR & CS Pumps	1-PMP-75-5	CS PUMP 1A	HCLPF > 0.3g
	1-PMP-75-33	CS PUMP 1B	HCLPF > 0.3g
	1-PMP-75-14	CS PUMP 1C	HCLPF > 0.3g
	1-PMP-75-42	CS PUMP 1D	HCLPF > 0.3g
	1-PMP-74-5	RHR PUMP 1A	HCLPF > 0.3g
	1-PMP-74-28	RHR PUMP 1B	HCLPF > 0.3g
	1-PMP-74-16	RHR PUMP 1C	HCLPF > 0.3g
	1-PMP-74-39	RHR PUMP 1D	HCLPF > 0.3g
6. RHR HXs	1-HEX-74-900A	RHR/HEAT EXCHANGER 1A	HCLPF > 0.3g
	1-HEX-74-900B	RHR/HEAT EXCHANGER 1B	HCLPF > 0.3g
	1-HEX-74-900C	RHR/HEAT EXCHANGER 1C	HCLPF > 0.3g
	1-HEX-74-900D	RHR/HEAT EXCHANGER 1D	HCLPF > 0.3g
7. Remote Control Cabinets	1-PNLA-925-0031	LOCAL PANEL 25-31	HCLPF > 0.3g
	1-PNLA-925-0032	LOCAL PANEL 25-32	HCLPF > 0.3g

7. RELAY EVALUATION

This section describes the relay evaluation process and results for BFN-1. Browns Ferry is identified as a focused scope plant for the 0.3g earthquake by NRC Generic Letter 88-20, Supplement 4. NUREG-1407 requests that focused scope plants which are also included as an USI A-46 plant should follow the USI A-46 procedures for the relay review of A-46 equipment. If low ruggedness relays are identified during the A-46 review, then an additional low ruggedness relay review should also be performed for IPEEE-only equipment. The A-46 criteria for relay functionality review are contained in GL 87-02, which endorses the review procedure established in the SQUG Generic Implementation Procedure (GIP).

7.1 RELAY REVIEW APPROACH

The EPRI NP-7148-SL (Reference 17) methodology was used in performing the Browns Ferry Nuclear Plant Unit 1 (BFN-1) relay evaluation (Reference 16). In general the methodology consists of the following steps:

1. Examine the control circuits for the safe shutdown system components.
2. Screen out non-essential relays using systems and circuit evaluation techniques. Also screen out contact devices such as large switches, which are considered not vulnerable to seismic motion and relays considered inherently rugged such as solid state relays.
3. Assess the seismic adequacy of the remaining, essential relays.
4. Provide a traceable documentation of the evaluation.
5. For IPEEE additional components, perform a "bad actors" review.

Certain additional special evaluation methods were utilized for BFN-1 since it is essentially identical to BFN-2 and BFN-3 which have already been reviewed. This commonality is documented in the BFN Final Safety Analysis Report (FSAR, Reference 10), which applies to all three units, and based on safe shutdown equipment identification and seismic verification walkdowns of the equipment. These special case evaluation methods included:

1. Maintain compatibility with BFN-2 and BFN-3 by utilizing the BFN-2 and BFN-3 USI A-46/IPEEE review results, resolutions, calculations, Request for Additional Information (RAI) responses and conclusions of the NRC issued SER for that review.

2. Do not reevaluate BFN-1 equipment common to BFN-2 or BFN-3 which was evaluated in the prior review for those units.

3. Utilize qualification data for new replacement switchgear, MCC buckets and relays which are being purchased as seismically qualified equipment. Note that this planned evaluation method was not used because during the evaluation, replacement switchgear, MCC buckets and relays had not yet been installed. Accordingly, the existing equipment was evaluated. Note that TVA Design Criteria BFN-50-C-7105 (Reference 18) specifies that new equipment shall be qualified by current or USI A-46 methods. As such, the seismic margins of the new equipment will be at least as high as those of the existing safe shutdown equipment.

For each SSEL component, the control circuit drawings which identify the contact devices affecting the operation of that device were identified. These drawings were reviewed and the contacts of those devices which affect component operation were identified. EPRI NP- 7148 screening and evaluation methods were then applied to arrive at a resolution.

Often more than one screening method was used. Usually one method is simpler or more efficient. Two of the screening methods were chosen whenever possible. Chatter acceptable screening was used for cases in which contact chatter leads to an acceptable system or component safe shutdown state. This screening method is useful for contact devices where the seismic demand is high or when seismic capacity data is not available. An additional benefit of chatter acceptable screening is that it reduces the number of essential relays, which are those that must pass a seismic capacity versus demand screen, and it reduces the number of panels and cabinets having essential relays. This in turn reduces walkdown efforts. Appendices B and C of EPRI NP-7148 provide numerous examples of chatter acceptable screening. These examples were reviewed and approved for chatter acceptable screening guidance by the NRC staff and a four member NRC staff relay review group.

For safety systems such as reactor protection, ECCS and containment isolation, relay chatter in the control logic may cause actuation of the system, just as a valid initiation signal would. Chatter in failsafe systems such as reactor protection and containment isolation, however, will not prevent or reverse the actuation of the system. These design features provide the basis for the use of relay chatter acceptability for the initiation logic of these systems. Although USI A-46 does not assume a Loss of Cooling Accident (LOCA), initiation of the ECCS is acceptable, and may be desirable for some situations. These fluid systems have protective features to prevent damage during unneeded operation and operators in the control room can shut down any of the ECCS not needed.

The second screening method chosen whenever possible was the Level 1 screen. This seismic acceptability screen was developed as a simple screen for cases in which high capacity relays are located low in the plant. Specific criteria for this screening method are discussed in EPRI NP-7148, Section 3.6. In general, relay contacts with a seismic capacity of 8g or more located less than about 40 feet above grade satisfy the Level 1 seismic adequacy screen. The 8g capacity screen can be used for most panels and cabinets when the other Level 1 criteria are met. For moderate capacity relays, a 5g screen can be applied when the relay is in a low amplification cabinet or panel and the other Level 1 criteria are met. Appendix I of EPRI NP-7148 provides guidance in determining cabinet and panel amplification categories.

The Lead Relay Reviewer for BFN-1 was Mr. Jess Betlack. Mr. Betlack was the primary developer of the relay review guidelines for both USI A-46 resolution and seismic IPEEE. His qualifications are included in Appendix A as a member of the seismic review team.

7.2 RELAY REVIEW RESULTS

The Browns Ferry Nuclear Plant Unit 1(BFN-1) relay evaluation for Unresolved Safety Issue (USI) A-46 and the seismic portion of the Individual Plant Examination for External Events (IPEEE) was performed in accordance with the appropriate industry guidance documents developed by the Seismic Qualification Utility Group (SQUG) and the Electric Power Research Institute (EPRI), and approved by the Nuclear Regulatory Commission (NRC). The relay evaluation also utilized results of the similar relay evaluations for BFN-2 and BFN-3.

In summary, the relay evaluation findings are as follows:

- Inherent ruggedness of contact devices, chatter acceptability and seismic adequacy were sufficient to satisfactorily resolve the seismic acceptability of contact devices affecting the USI A-46 Safe Shutdown Equipment List (SSEL) components.
- No outliers were identified in the evaluation.
- No low ruggedness (bad actor) relays were found to be essential relays.
- No operator actions were identified in the evaluation as necessary to correct relay-chatter-caused malfunctions.

Essential relays and the cabinets housing those essential relays were identified for the seismic capability engineers performing the seismic verification walkdowns and evaluations. The SRT determined in-cabinet amplification factors for use in the relay capacity versus demand screening. The SRT also took appropriate cautions and factors of safety into consideration when evaluating the cabinets housing the essential relays. The cabinets were determined to be acceptable and no modifications were required.

Based on the result of the relay review, the BFN-1 relays can be assigned a HCLPF capacity exceeding 0.30g.

8. SEISMIC INDUCED FIRE AND FLOOD EVALUATION

IPEEE seismic-induced fire and flood issues relative to BFN-1 were addressed during the combined USI A-46/IPEEE equipment inspections and other evaluations. The concerns have been addressed as necessary in accordance with the guidelines established in NUREG-1407.

It is the conclusion of this review that the Browns Ferry plant is not at risk from fire or flood resulting from a seismic event at least as great as the 0.30g RLE. Combustible sources are controlled within critical areas of the plant such that there is no threat that a fire could eliminate the safe shutdown capability of the plant. The risk due to flooding is mitigated by systems separation, backup systems, and maintaining the seismic Category I (L) pressure retention criteria for non-Class I piping and fluid systems in Category I structures at BFN.

8.1 SEISMIC II/I SPRAY PROGRAM

A seismic-induced II/I spray evaluation program was implemented as part of the BFN-1 Restart Program (Reference 19). Key engineering attributes of the seismic II/I evaluation program consisted of the following:

- In-plant screening walkdown evaluations and identification of potential outliers;
- Further evaluations and resolution of potential outliers;
- Engineering design of plant modifications to resolve outliers;
- Work order requests to address general maintenance and housekeeping items.

In-plant screening walkdown evaluations of seismic II/I spray hazards were performed on an area-by-area basis. A total of 27 designated plant areas were included. The areas encompassed all of the BFN-1 Reactor Building. Other BFN plant areas were addressed in previous seismic II/I spray programs for BFN-2 and BFN-3.

Screening evaluations focused on certain key attributes of the non-seismic Class I (Class II) piping and fluid pressure boundary systems that may potentially pose as spray hazards to surrounding seismic Class I systems and components in the event of an earthquake. Screening tools such as seismic deflection estimates and charts for various plant features, pipe flexibility and seismic anchor movement evaluation charts, support and anchorage capacity screening charts, and others, were developed for use in the in-plant screening walkdown evaluations. Certain configurations identified during the in-plant screening walkdowns as not meeting the screening criteria were documented in the Potential Outlier Sheet (POS) as potential outliers and for further evaluation and disposition. Walkdown results, including a total of 179 potential outliers identified, were documented in the Walkdown Data Packages (WDP's) for the respective plant areas.

Potential outliers identified during the in-plant screening walkdowns were further evaluated to the acceptance criteria of TVA Design Criteria BFN-50-C-7306 (Reference 13). Further evaluations and bounding analyses of these potential outliers consisted of hand calculations using basic engineering mechanics techniques for simple configurations, and rigorous piping analyses (TPIPE computer program) for more complex piping configurations. A total of 19 outliers were found to have not met the acceptance criteria. Plant modifications were designed and Design Change Notice (DCN) issued to implement the changes so that all of these concerns were resolved. Furthermore, 13 maintenance and/or housekeeping items were also identified for corrective actions. Maintenance work order requests were issued to address these items.

Based on the results of the seismic II/I spray program, the potential for seismic-induced spray for BFN-1 is assigned a HCLPF capacity greater than 0.30g.

8.2 SEISMIC-INDUCED FLOOD HCLPF CAPACITY EVALUATION

The general approach used to eliminate any unacceptable seismic induced II/I spray interactions at BFN Unit 1 is by ensuring that the source has the proper seismic capacity to resist the earthquake and not to result in any possible leakage or spray. In general, this method eliminates the need to evaluate the interaction all together, i.e. the path of the spray and the acceptance of the interaction.

Thus, at the conclusion of the Seismic Induced II/I Spray program, and after implementation of the required modifications and maintenance work, BFN-1 is adequately protected against any possible flood or spray effects affecting the safe shutdown of the plant. The program however was implemented for the Safe Shutdown Earthquake (SSE), as required by the FSAR. To meet the requirements of the SMA, the conclusions of the Seismic Induced II/I Spray program must also be made applicable to the RLE used for the SMA. The items which have been screened out by the walkdown, evaluated as acceptable or modified by either a design change or a maintenance order are judged by the SRT to have significant seismic margin such that the HCLPF capacity is greater than the RLE of 0.3g.

During the walkdowns, the SRT determined that the controlling component which provides a good indication of the seismic capacity of the non seismic components which could result in a seismic induced flood interaction is the Gland Seal Storage Tank located in the Reactor Building at Elevation 639'. Therefore, a HCLPF capacity evaluation was performed for the Gland Seal Storage Tank. It was determined that this tank has a HCLPF capacity greater than 0.3g. Based on this bounding evaluation, all non seismic components which could pose a seismic induced flood hazard also have a HCLPF capacity greater than 0.3g.

8.3 SEISMIC-INDUCED FIRE EVALUATION

A separate, focused seismic-induced-fire walkdown was performed by the SRT on an area-by-area basis after the SSEL walkdown screening evaluations were completed. The walkdown screening evaluation followed the guidance in NUREG-1407, searching for flammable materials or fluids that could lose their containment during an earthquake, and sources of spark or ignition that could be triggered by an earthquake. One potential hazard was identified during the walkdowns. This consisted of unrestrained batteries on the emergency lighting system battery rack in the BFN-1 cable spreading room. The batteries lacked end restraints, side restraints, and spacers between the batteries. The concern was that the batteries could fall from the rack and cause sparks. The situation was corrected using the corrected by the TVA Problem Evaluation Report process. PER No. 64143 was issued to address and correct the condition.

9. CONTAINMENT INTEGRITY

The main objective of the containment analysis is to identify vulnerabilities that involve early failure of containment functions. This includes consideration of containment integrity, containment isolation, and other containment functions.

The guidance provided in NUREG-1407, Reference 2, states that "generally containment penetrations are seismically rugged; a rigorous fragility analysis is needed only at review levels greater than 0.30g, but a walkdown to evaluate for unusual conditions (e.g., spatial interactions, unique penetration configurations) is recommended." With regard to containment systems, the guidance provided is that "seismic failures of actuation and control systems are more likely to cause isolation system failures and should be included in the examination." The major concern deals with relay chatter, which is addressed in Section 7 of this report.

The BFN-1 containment structure is screened for further seismic review based on NP-6041, Reference 3. In addition to the containment structure, NUREG-1407 suggests that certain considerations could require additional study. Hatches that employ inflated seals is one potential area for concern. BFN hatches do not use inflated seals.

Another concern is the post-accident operation of penetration cooling systems. BFN makes combined use of insulation and penetration cooling for hot piping penetrations. The penetration cooling subsystem is non-safety-related. The portion of the piping inside primary containment has been designed to Class I (L) standards in order to minimize possible damage to Class I equipment inside the drywell from pipe break and flooding. Analysis shows that under a condition of total loss of coolant, and under the most adverse conditions, the concrete temperature adjacent to any penetration does not exceed 350°F. This analysis is based upon heat conduction and does not take into account dissipation into surrounding structures or atmosphere. Penetration coolers were added as a result of good engineering practice and design; however, as seen from the above, they are not considered necessary to safe operation of the plant or to maintain containment integrity.

All other containment issues relate to the seismic relay review and walkdown results. The relay review is addressed in Section 7. The containment walkdowns consisted of inspecting and evaluating unusual conditions or configurations in the drywell and torus.

The following is a representative listing of unusual conditions or configurations specifically searched for during the walkdown process:

- Spatial Interactions
- Unique Penetrations
- Piping hard spots
- Items or components bridging the seismic gap between the containment liner and interior structure

No unusual conditions or configurations were identified. As stated previously, the main objective of the containment analysis is to identify vulnerabilities that involve early failure of containment functions. The SRT reviews and walkdowns performed on the containment did not reveal any significant vulnerabilities. Therefore, the HCLPF for the containment is greater than 0.3g, based on SRT reviews, walkdowns, and Appendix A of NP-6041 (Reference 3).

10. SUMMARY AND CONCLUSIONS

The BFN-1 seismic IPEEE was completed in accordance with NUREG 1407 guidelines using the EPRI seismic margins methodology provided in EPRI NP-6041-SL.

The most important aspect of the program was the plant walkdowns. Detailed SRT walkdowns were performed in conjunction with USI A-46 walkdowns using the methodology, criteria, and SEWS provided in EPRI NP-6041 and the GIP.

The SRT identified issues related to anchorage design, maintenance, housekeeping, and seismic interaction that required design change notices (DCNs) or work orders to satisfy SRT field issues. These items will be resolved as part of the USI A-46 program. Several components were identified for subsequent HCLPF evaluation. None of the items had HCLPF capacities below 0.30g.

One item was observed to be a potential seismic-induced-fire hazard. A PER was initiated to correct the situation.

Relay evaluation for BFN-1 followed the methodology recommended in the GIP and resulted in no low ruggedness relays identified and no outliers.

The seismic IPEEE evaluation concluded that the BFN-1 HCLPF is at least as great as the 0.30g review level earthquake defined as an earthquake having a response spectrum that matches the median (50% Non Exceedance Probability) CR-0098 spectral shape anchored to a peak ground acceleration of 0.30g.

11. REFERENCES

1. USNRC, Generic Letter 88-20, Supplement No. 4, "Individual Plant Examination of External Events (IPEEE) for severe accident vulnerabilities," Final Report, April 1991.

2. USNRC, NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," Final Report, June 1991.

3. Electric Power Research Institute. Report No. EPRI NP-6041-SL, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin", Revision 1, August 1991.

4. USNRC, NUREG/CR-5250, "Seismic Hazard Characterization of 69 Nuclear Power Plant Sites East of the Rocky Mountains," January 1989.

5. USNRC, NUREG-1488, "Revised Livermore Seismic Hazard Estimates for 69 Nuclear Power Plant Sites East of the Rocky Mountains," October 1993.

6. Electric Power Research Institute. Report No. EPRI NP-6395-D, "Probabilistic Seismic Hazard Evaluation at Nuclear Plant Sites in the Central and Eastern United States: Resolution of the Charleston Issue," April 1989.

7. Seismic Qualification Utility Group (SQUG), "Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment," Revision 2A, March 1993.

8. Newmark, N.M. and Hall, W.J. 1978, "Development of Criteria for Seismic Review of Selected Nuclear Power Plants," US Nuclear Regulatory Commission, NUREG/CR-0098.

9. TVA Calculation CD-Q0000-940339, "Calculation of Basic Parameters for A-46 and Individual Plant Examination of External Events (IPEEE) Seismic Program." Revision 0.

10. Tennessee Valley Authority. "Browns Ferry Nuclear Plant Final Safety Analysis Report."

11. Tennessee Valley Authority. General Design Criteria No. BFN-50-C-7102. "Seismic Design." Revision 3.

12. Tennessee Valley Authority. General Design Criteria No. BFN-50-C-7103. "Seismic Analysis and Qualification of Mechanical and Electrical Systems (Piping and Instrument Tubing)." Revision 5.

13. Tennessee Valley Authority. General Design Criteria No. BFN-50-C-7306. "Qualification Criteria for Seismic Class II Piping, Pipe Supports, and Components." Revision 1.

14. Tennessee Valley Authority. General Design Criteria No. BFN-50-C-7104. "Design of Structural Supports." Revision 12.

15. Facility Risk Consultants, Inc. Report No. TVA.BFN-01-R-004. "Browns Ferry Nuclear Plant Unit 1 USI A-46 Seismic Evaluation Report." Revision 0, 23 September 2004.

16. Facility Risk Consultants, Inc. Report No. TVA.BFN-01-R-001. "USI A-46 / Seismic IPEEE Relay Evaluation Browns Ferry Nuclear Plant Unit 1." Revision 0, 23 January 2004.

17. Electric Power Research Institute. Report No. EPRI NP-7148-SL, "Seismic Ruggedness of Relays," and Addendums. August 1991.

18. Tennessee Valley Authority. General Design Criteria No. BFN-50-C-7105. "Pipe Rupture, Internal Missiles, Internal Flooding, Seismic Equipment Qualification, and Vibration Qualification of Piping." Revision 7.

19. Facility Risk Consultants, Inc. Report No. TVA.BFN-01-R-002. "Seismic-Induced II/I Spray Evaluations at Browns Ferry Nuclear Plant Unit 1." Revision 0, 26 March 2004.

APPENDIX A: SEISMIC REVIEW TEAM QUALIFICATIONS

RESUME – Lead Relay Reviewer

Performed Relay Functionality Review

Tennessee Valley Authority
Browns Ferry Nuclear Plant Unit 1

Name:	Jess O. Betlack	
Bachelors Degree:	B.S. Electrical Engineering, 1966	
Institution:	University of Kansas	
Advanced Degree:	M.S. Electrical Engineering, 1967 Graduate Studies in Electrical Engineering and Computer Science	
Institution:	University of Kansas and University of New Mexico	
Date and location of Seismic Adequacy Verification Training Courses:		
USI A-46	Course:	SQUG Relay Seismic Functionality Evaluation
	Date:	1988 – 1996
	Location:	SQUG Subject Matter Expert and Course Instructor
Seismic IPEEEE	Course:	
	Date:	
	Location:	
Earthquake engineering experience applicable to nuclear power plants and in structural or mechanical engineering:		
30 years (see attached resume)		
Licensed Professional Engineer: Yes ☑ No ☐	State of Maryland	

RESUME - Seismic Capability Engineer

Member of a Seismic Review Team

Tennessee Valley Authority
Browns Ferry Nuclear Plant Unit 1

<table>
<tr><td colspan="2">Name:</td><td colspan="2">John O. Dizon</td></tr>
<tr><td colspan="2">Bachelors Degree:</td><td colspan="2">B.S. Civil Engineering, 1973</td></tr>
<tr><td colspan="2">Institution:</td><td colspan="2">Mapua Institute of Technology</td></tr>
<tr><td colspan="2">Advanced Degree:</td><td colspan="2">M.S. Structural Engineering, 1975
Engineer Degree, 1977</td></tr>
<tr><td colspan="2">Institution:</td><td colspan="2">Stanford University</td></tr>
<tr><td colspan="4">Date and location of Seismic Adequacy Verification Training Courses:</td></tr>
<tr><td rowspan="3">USI A-46</td><td>Course:</td><td colspan="2">SQUG Walkdown Screening & Seismic Evaluation Training Course</td></tr>
<tr><td>Date:</td><td colspan="2">January 13 – 19, 1993</td></tr>
<tr><td>Location:</td><td colspan="2">San Francisco, CA (EQE)</td></tr>
<tr><td rowspan="3">Seismic IPEEEE</td><td>Course:</td><td colspan="2">Seismic IPE Add-on Training Course</td></tr>
<tr><td>Date:</td><td colspan="2">October 13 – 15, 1992</td></tr>
<tr><td>Location:</td><td colspan="2">Chicago, IL</td></tr>
<tr><td colspan="4">Earthquake engineering experience applicable to nuclear power plants and in structural or mechanical engineering:</td></tr>
<tr><td colspan="4">27 years (see attached resume)</td></tr>
<tr><td colspan="4"></td></tr>
<tr><td colspan="4"></td></tr>
<tr><td colspan="3">Licensed Professional Engineer: Yes ☑ No ☐</td><td>State of California</td></tr>
</table>

RESUME - Seismic Capability Engineer

Member of a Seismic Review Team

Tennessee Valley Authority
Browns Ferry Nuclear Plant Unit 1

Name:	Stephen J. Eder	
Bachelors Degree:	B.S. Civil and Environmental Engineering, 1980	
Institution:	Clarkson College of Technology	
Advanced Degree:	M.Eng. Structural Engineering & Structural Mechanics, 1982	
Institution:	University of California, Berkeley	
Date and location of Seismic Adequacy Verification Training Courses:		
USI A-46	Course:	SQUG Walkdown Screening & Seismic Evaluation Training Course
	Date:	1988 – 1994
	Location:	SQUG Subject Matter Expert and Course Instructor
Seismic IPEEEE	Course:	Seismic IPE Ad-on training
	Date:	1992 – 1994
	Location:	Reviewer of EPRI seismic margins training course as SQUG Subject Matter Expert
Earthquake engineering experience applicable to nuclear power plants and in structural or mechanical engineering:		
22 years (see attached resume)		
Licensed Professional Engineer: Yes ☑ No ☐	States of Alabama & California	

RESUME - Seismic Capability Engineer

Member of a Seismic Review Team

Tennessee Valley Authority
Browns Ferry Nuclear Plant Unit 1

<table>
<tr><td colspan="2">Name:</td><td colspan="2">Farid Elsabee</td></tr>
<tr><td colspan="2">Bachelors Degree:</td><td colspan="2">B.S. Engineering, 1973</td></tr>
<tr><td colspan="2">Institution:</td><td colspan="2">State University of New York at Stony Brook</td></tr>
<tr><td colspan="2">Advanced Degree:</td><td colspan="2">M.S. Civil Engineering (Structures), 1975</td></tr>
<tr><td colspan="2">Institution:</td><td colspan="2">Massachusetts Institute of Technology</td></tr>
<tr><td colspan="4">Date and location of Seismic Adequacy Verification Training Courses:</td></tr>
<tr><td rowspan="3">USI A-46</td><td>Course:</td><td colspan="2">SQUG Walkdown Screening & Seismic Evaluation Training Course</td></tr>
<tr><td>Date:</td><td colspan="2">August 10 – 14, 1992</td></tr>
<tr><td>Location:</td><td colspan="2">Millstone Nuclear Station</td></tr>
<tr><td rowspan="3">Seismic IPEEEE</td><td>Course:</td><td colspan="2">Seismic IPE Ad-on training</td></tr>
<tr><td>Date:</td><td colspan="2">November 2 – 4, 1992</td></tr>
<tr><td>Location:</td><td colspan="2">Millstone Nuclear Station</td></tr>
<tr><td colspan="4">Earthquake engineering experience applicable to nuclear power plants and in structural or mechanical engineering:</td></tr>
<tr><td colspan="4">27 years (see attached resume)</td></tr>
<tr><td colspan="4"></td></tr>
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<tr><td colspan="3">Licensed Professional Engineer: Yes ☐ No ☑</td><td></td></tr>
</table>

RESUME - Seismic Capability Engineer

Member of a Seismic Review Team

Tennessee Valley Authority
Browns Ferry Nuclear Plant Unit 1

<table>
<tr><td>Name:</td><td colspan="2">Robert D. Hookway</td></tr>
<tr><td>Bachelors Degree:</td><td colspan="2">B.S. Mechanical Engineering, 1963</td></tr>
<tr><td>Institution:</td><td colspan="2">Lowell Technological Institute</td></tr>
<tr><td>Advanced Degree:</td><td colspan="2">M.S. Mechanical Engineering, 1970</td></tr>
<tr><td>Institution:</td><td colspan="2">Northeastern University</td></tr>
<tr><td colspan="3">Date and location of Seismic Adequacy Verification Training Courses:</td></tr>
<tr><td rowspan="3">USI A-46</td><td>Course:</td><td>SQUG Walkdown Screening & Seismic Evaluation Training Course</td></tr>
<tr><td>Date:</td><td>January 13 – 19, 1993</td></tr>
<tr><td>Location:</td><td>San Francisco, CA (EQE)</td></tr>
<tr><td rowspan="3">Seismic IPEEEE</td><td>Course:</td><td></td></tr>
<tr><td>Date:</td><td></td></tr>
<tr><td>Location:</td><td></td></tr>
<tr><td colspan="3">Earthquake engineering experience applicable to nuclear power plants and in structural or mechanical engineering:</td></tr>
<tr><td colspan="3">30 years (see attached resume)</td></tr>
<tr><td colspan="3"></td></tr>
<tr><td colspan="3"></td></tr>
<tr><td colspan="2">Licensed Professional Engineer: Yes ☑ No ☐</td><td>States of Massachusetts & Virginia</td></tr>
</table>

RESUME - Seismic Capability Engineer

Member of a Seismic Review Team

Tennessee Valley Authority
Browns Ferry Nuclear Plant Unit 1

Name:	Richard L. Tiong	
Bachelors Degree:	B.S. Civil Engineering, 1978	
Institution:	University of London, England	
Advanced Degree:	M.S. Structural Engineering and Structural Mechanics, 1981	
Institution:	University of California, Berkeley	
Date and location of Seismic Adequacy Verification Training Courses:		
USI A-46	Course:	SQUG Walkdown Screening & Seismic Evaluation Training Course
	Date:	January 15 – 20, 1993
	Location:	Irvine, CA (EQE)
Seismic IPEEEE	Course:	
	Date:	
	Location:	
Earthquake engineering experience applicable to nuclear power plants and in structural or mechanical engineering:		
22 years (see attached resume)		
Licensed Professional Engineer: Yes ☑ No ☐	State of California	

JESS BETLACK, P.E.

EDUCATION

University of Kansas and University of New Mexico, Graduate Studies in Electrical Engineering and Computer Science

University of Kansas, M.S. Electrical Engineering, 1967

University of Kansas, B.S. Electrical Engineering, 1966 (With Distinction)

PROFESSIONAL HISTORY

1964 - 1968	*University of Kansas*
1968 - 1973	*Sandia Laboratories*
1973 - present	*MPR Associates, Inc.* (currently as a special assignment employee)

EXPERIENCE

Mr. Betlack has worked in the fields of electrical engineering and computer science since 1964. Specific activities have included the design, analysis, development and testing of computer systems and components (both hardware and software), electrical and electro-mechanical systems and components, and instrumentation and controls. Projects have involved data acquisition, processing, monitoring, simulation and control computer systems, database systems, test facility and power plant instrumentation and controls, and modeling and simulation of power plants and power plant equipment including steam generators, turbines, pumps, instrumentation, controls and electrical equipment. The design and analysis activities, including troubleshooting and plant life extension evaluations, have involved extensive in-plant experience. Projects have also included seismic functionality evaluations of relays and other electrical equipment, and vibration monitoring and testing of such equipment. Specific examples of Mr. Betlack's experience include:

Design, Development, and Evaluation of Computer Systems

Designed, developed, implemented and tested on-line and real-time computer systems used in support of flight, environmental, and full-scale testing. Specific activities have included feasibility studies, systems analysis and design, software development, preparation of functional specifications, component and system specifications, component and system procurement, data acquisition and reduction, acceptance and benchmark tests, and system and component evaluations.

Participated as a committee member and writer of several Electric Power Research Institute (EPRI) guidelines for design, development, licensing, and dedication of digital I&C system upgrades for the nuclear industry.

Evaluated the design and standards compliance of several power plant instrumentation, control, and protection digital systems.

Design and development of a networked environmental monitoring system.

Design and Development of Instrumentation and Controls (I&C)

Designed and coordinated the development and check-out of power plant I&C equipment including a reactor coolant pump seal leakage flowmeter, an ultrasonic level sensing system controller, and a nuclear plant steam generator cleaning system controller.

Specified, reviewed and provided oversight of the design, development, installation and testing of a five-instrument two-phase flowmeter and other advanced instruments used in international reactor safety test facilities.

Developed instrumentation and monitoring capabilities for the evaluation of equipment problems including fan vibration and pump seal rubbing.

Technical lead for implementation of U.S. Nuclear Regulatory Commission (USNRC) Regulatory Guide (R.G.) 1.97 design requirements at a nuclear power station.

Life Extension Evaluations

Performed life extension evaluations of electrical equipment, instrumentation and controls in over 15 fossil and nuclear plants.

Relay/Control Seismic Functionality Evaluations

Developed relay seismic functionality evaluation procedure for application in resolving USNRC Unresolved Safety Issue (USI A-46) and Individual Plant Examination of External Events (IPEEE). Coordinated and evaluated seismic testing of over 150 relays in support of the nuclear power industry resolution of USI A-46.

Performed and reviewed relay seismic functionality evaluations for several nuclear plants.

Training and Teaching

Taught over 20 industry training courses on USI A-46 relay seismic functionality evaluation and courses on computer-based I&C systems. Teaching assistant in logic design and laboratory instructor for computer science and programming at the University of Kansas.

MEMBERSHIPS

Tau Beta Pi - National Engineering

Eta Kappa Nu - National Electrical Engineering

Instrument Society of America (ISA)-Senior Member

Registered Professional Engineer – Maryland

PUBLICATIONS

Publications have consisted of many reports, software documents and internal design and evaluation documents at MPR, Sandia Laboratories, and CRES (University of Kansas Research Center). These have included:

- Master's Thesis -- "A Preprocessor for Multi-Spectral Images."
- SC-DR-21-0419 -- "Area III Automated Data Processing System."
- SC-TM-69-509 -- "Operations and Maintenance Documents for Program FIXCAM."
- EPRI NP-7148 -- "Procedure for Evaluating Nuclear Power Plant Relay Seismic Functionality." (Primary author).
- EPRI TR-107980, "I&C Upgrades for Nuclear Plants -- Desk Reference 1997," December 1997.
- EPRI TR-107339, "Evaluating Commercial Digital Equipment for High Integrity Applications," December 1997 (Co-writer).
- EPRI TR-106436, "Guidance on Evaluation and Acceptance of Commercial Grade Digital Equipment for Nuclear Safety Applications," October 1996 (Co-writer).
- EPRI TR-102348, "Guideline on Licensing Digital Upgrades," December 1993 (Co-writer).

Presentations on digital I&C, database systems, and relay seismic functionality evaluation and testing have been made at national and international conferences including:

- EPRI International I&C Conference (December, 1997).
- ANS 1994 Winter Meeting
- EPRI Workshop on Licensing Issues Concerning Digital I&C Upgrades for Nuclear Power Plants (March, 1992).
- 10th International Conference on Structural Mechanics in Reactor Technology (SMIRT).
- 3rd Symposium on Current Issues Related to Nuclear Power Plant Structures, Equipment and Piping.
- 6th Water Reactor Safety Information Meeting of the U.S. Nuclear Regulatory Commission.

JOHN O. DIZON, P.E.

PROFESSIONAL HISTORY

Facility Risk Consultants, Inc., Fremont, California & Huntsville, Alabama, President, 2002-present

ABS Consulting (formerly EQE International), Oakland, California, Director and VP of Facility Risk, 2000-2002

EQE International, Oakland, California, Vice President, 1998-2000; Associate, 1991-1998; Senior Engineer, 1986-1991

Engineering Decision Analysis Company, Cupertino, California, Senior Engineer, 1984-1986

General Electric Company, San Jose, California, Senior Engineer, 1984

URS/John A. Blume & Associates, San Francisco, California, Senior Engineer, 1982-1984; Associate Engineer, 1977-1980

Structural Systems Engineering, Inc., Lafayette, California, Senior Engineer, 1980-1982

Stanford University, John A. Blume Earthquake Engineering Center, Palo Alto, California, Teaching and Research Assistant, 1975-1977

PROFESSIONAL EXPERIENCE

Mr. Dizon has over 25 years of experience in the field of civil and structural engineering, earthquake engineering, risk assessment and project management. He has extensive knowledge in the areas of seismic analyses and design assessments of primary structures and piping systems, seismic upgrade and retrofit design, seismic qualification of mechanical and electrical systems and components, and technical development of seismic evaluation criteria and programs for various industries, including power, oil and gas, petrochemical, and high tech process and manufacturing facilities. Mr. Dizon has undertaken and managed a wide variety of seismic projects, ranging from traditional structural engineering design and seismic retrofits to complex nuclear power plant and DOE facilities' seismic verification projects.

As President of Facility Risk Consultants, Mr. Dizon is currently managing all associated tasks under a subcontract with Bechtel Power Corporation for all seismic-related issues associated with Browns Ferry Unit 1 Restart Project for Tennessee Valley Authority. The seismic works include USI A-46/IPEEE implementation programs, seismic II/I spray hazard evaluations, new cable routing utilizing the SQUG/GIP methodology, MSIV seismic ruggedness verification, among others. Currently, he is also actively involved in the development of seismic II/I design criteria for distribution systems and equipment for DOE's PDCF project, under a subcontract with the Washington Group, Inc. In addition, Mr. Dizon is participating as a subject matter expert witness in a litigation project for a large foreign company in the area of seismic performance of structures, piping systems and associated equipment associated with earthquake damges in a coal-fired power plant located in South America.

As EQE Project Manager for various seismic programs associated with the restart of Browns Ferry Units 2 and 3, Mr. Dizon was responsible for all engineering activities associated with USI

A-46 resolution and seismic IPEEE implementation; seismic proximity and II/I spray interaction evaluations; MSIV seismic ruggedness verification; cable tray and conduit raceway and supports; and HVAC support evaluation programs. These activities consisted of seismic criteria development, seismic walkdown assessments and mitigation of findings, including retrofit designs and plant upgrades. He was also responsible for the A-46 seismic evaluation program for major equipment items at Davis-Besse, Duane Arnold and H.B. Robinson power plants. Mr. Dizon also served as Project Manager for the HVAC seismic verification program at Salem Nuclear Plant, MSIV seismic projects at Hope Creek and Brunswick plants, and participated in a number of related seismic evaluation projects at Sequoyah, Watts Bar, Bellefonte, Pickering A, Bruce A, Forsmark, Liebstadt, among others.

As Managing Director of EQE's Hsinchu, Taiwan project office following the 1999 Chi-Chi earthquake, he was in charge of the region's business development and project management. Mr. Dizon managed a number of seismic risk assessment and structural upgrade projects for the high tech industry, including seismic consultation on a number of projects for Taiwan Semiconductor Manufacturing Co., seismic strengthening projects for United Microelectronics, Applied Materials, Winbond Electronics and Macronix International in Taiwan. In addition, he also managed the seismic upgrades for the Cypress Semiconductor and Amkor facilities and seismic design review project for IBM in the Philippines, seismic risk assessment for AMP facilities in Japan, and seismic assessment of structural and non-structural components of several Intel fab plants in the Northwest region in U.S., among others.

As Group Manager for EQE at the US Department of Energy Savannah River Site, Mr. Dizon was responsible for the seismic verification program of safety-related mechanical and electrical systems and components. His tasks included developing seismic evaluation criteria and procedures for restart and long-term seismic programs; managing the seismic walkdown and evaluation efforts; providing technical support in resolving seismic issues; and serving as an interface with the client. Mr. Dizon was also responsible for the seismic walkdown and evaluation of various distribution systems at the Pantex Facilities, including developing the walkdown screening criteria and evaluation acceptance criteria. Mr. Dizon has participated in the seismic evaluation of the High Flux Isotope Reactor at Oak Ridge National Laboratory. This project involved performing seismic analyses and upgrades for the primary coolant piping system and related equipment, and the reactor and control buildings. Other DOE facilities he has involvement with included Los Alamos, Livermore and Hanford sites. Mr. Dizon has also been involved in a number of risk assessment programs for petrochemical plants and refineries, including seismic walkdowns at the Imperial West Chemical plants in Pittsburg and Antioch, CA; Tosco Refinery in Avon, CA; and Dupont Chemical plant in Antioch, CA, among others.

At EDAC, Mr. Dizon was responsible for the development and verification of a pipe support optimization program (OPTPIPE) and was involved in a number of snubber reduction pilot projects. Other areas of his involvement consisted of finite element analyses of the MX-missile launch tube components and systems for thermal and pressure loads, equipment qualification of major mechanical and electrical components, and seismic evaluation of cooling towers.

With General Electric Company, Mr. Dizon was responsible for stress analysis and code conformation of main steam and recirculation piping systems for generic BWR plants. He was also involved in the developmental phase of an in-house pipe support optimization program.

At URS/Blume & Associates, Mr. Dizon was responsible for the development and maintenance of in-house computer programs for both linear and nonlinear analyses of structural and piping systems. He was also involved in the linear and nonlinear dynamic analyses, finite element modeling, and generation of floor response spectra for several nuclear power plants. He helped develop a soil-structure interaction computer program using a three-dimensional finite element technique to evaluate the dynamic response of structures due to arbitrary plane body and surface wave excitations. He performed a research study involving soil-structure interaction analysis using the finite element FLUSH program to investigate the dynamic response of typical containment structures due to underground blast excitations.

Mr. Dizon worked as a consultant to Bechtel Power Corporation with Structural Systems Engineering, Inc. He performed structural analyses and design assessments of the primary containment structure and the reactor/control buildings of several BWR plants for the various types of hydrodynamic loads. He was involved in a BWR in-plant test procedures, data reduction and correlation study to determine the dynamic response, including soil-structure interaction of the reactor/control buildings during GE Mark II reactor hydrodynamic load actuation in the primary containment.

At Stanford University, Mr. Dizon performed statistical analyses of earthquake accelerograms and various response parameters, as part of his research work under Professor Haresh Shah. He also conducted seismic risk analyses and formulated seismic design criteria for Nicaragua. In addition, he was involved in the dynamic testing of structural models and equipment.

EDUCATION

STANFORD UNIVERSITY, Palo Alto, California: Engineer Degree, 1977

STANFORD UNIVERSITY, Palo Alto, California: M.S. Structural Engineering, 1975

MAPUA INSTITUTE OF TECHNOLOGY, Manila, Philippines: B.S. Civil Engineering, 1973

AFFILIATIONS AND AWARDS

Earthquake Engineering Research Institute, Member

Philippine Board Examination for Civil Engineers, Fifth Place, 1973

Philippine Association of Civil Engineers, Certificate of Merit, 1974

REGISTRATION

California: Civil Engineer

Philippines: Civil Engineer

SELECTED PUBLICATIONS

With S. J. Eder, and R. D. Cutsinger. 2003. " Browns Ferry Cable Tray Evaluations." Presented to the SQUG/SEQUAL Annual Meeting, San Antonio, TX, December 10-12, 2003.

With S. J. Eder. 2003. " Technical Position Paper for Seismic II/I Design of Cable Tray Raceway Systems at PDCF." Presented to Washington Group, Inc., December 2003.

With S. J. Eder, W. H. Tong, and E. H. Wong, 1999. "Chichi, Taiwan Earthquake of September 21, 1999 (M7.6). An EQE Briefing. Oakland, CA. October, 1999.

With S. J. Eder. 1998. "Risk Management for Power and Industrial Facilities -- Focus on Business Interruption". Second Biennial Federation of Asian Pacific & African Risk Management Organization. Manila, Philippines. October, 1998.

With F. R. Beigi. 1995. "Application of Seismic Experience Based Criteria for Safety Related HVAC Duct System Evaluation." Fifth DOE Natural Phenomena Hazards Mitigation Symposium, Denver, Colorado, November 13-14, 1995.

With S. J. Eder, J. F. Glova, and R. L. Koch. 1994. "Seismic Adequacy Verification of HVAC Duct Systems and Supports for an USI A-46 Nuclear Power Plant." Fifth Symposium on Current Issues Related to Nuclear Power Plant Structures, Equipment and Piping, Orlando, Florida, December 14-16, 1994.

With E. J. Frevold and P. D. Osborne. 1993. "Seismic Qualification of Safety-related HVAC Duct Systems and Supports." ASME Pressure Vessel and Piping Division Conference, Denver, Colorado, July 1993.

With S. J. Eder. 1991. "Advancement in Design Standards for Raceway Supports and Its Applicability to Piping Systems." ASME Pressure Vessel and Piping Division Conference, San Diego, California, June 1991.

With R. D. Campbell and L. W. Tiong. 1990. "Response Predictions for Piping Systems Which Have Experienced Strong Motion Earthquakes." ASME Pressure Vessel and Piping Conference, Nashville, Tennessee, June 17-21, 1990.

With S. P. Harris, R. S. Hashimoto, and R. L. Stover. 1989. "Seismic, High Wind, and Probabilistic Risk Assessments of the High Flux Isotope Reactor." Second DOE Natural Phenomena Hazards Mitigation Conference.

With D. Ray and A. Kabir. 1979. "A 3-D Seismic Analysis for Arbitrary Plane Body and Surface Wave Excitations." American Society of Civil Engineers Nuclear Specialty Conference, Boston, Massachusetts.

With D. Ray and A. Zebarjadian. 1978. "Dynamic Response of Surface and Embedded Disk Foundations for SH, SV, P and Rayleigh Wave Excitations." Sixth Indian Symposium on Earthquake Engineering, Roorkee, India.

"A Statistical Analysis of Earthquake Acclerograms and Response Parameters." 1977. Thesis, Stanford University, Palo Alto, California,

With H. Shah, T. Zsutty, H. Krawinkler, and L. Padilla. 1977. "A Seismic Design Procedure for Nicaragua." Paper presented at the Sixth World Conference on Earthquake Engineering, New Delhi, India.

With H. Shah, T. Zsutty, H. Krawinkler, C. P. Mortgat, and A. Kiremidjian. 1976. "A Study of Seismic Risk for Nicaragua, Part II, Summary and Commentary." John A. Blume Earthquake Engineering Center, Report No. 12A and 12B. Stanford University, Palo Alto, California.

STEPHEN J. EDER, P.E.

PROFESSIONAL HISTORY

Facility Risk Consultants, Fremont, California, Chief Executive Officer, 2003-present

ABS Consulting, Houston, Texas, Vice President, North Asia Pacific Region, 2001-2003

EQE International, San Francisco, California, Senior Vice President, 1985-2001 (ABS Purchased EQE in 2000).

URS/John A. Blume & Associates, Engineers, San Francisco, California, 1982-1985

J. G. Bouwkamp, Inc., Structural Engineers, Berkeley, California, 1981-1982

PROFESSIONAL EXPERIENCE

Mr. Stephen J. Eder provides senior engineering and management consultant services, licensing support, and expert testimony in the fields of natural hazards risk assessment, seismic analysis, structural performance evaluation, and retrofit design. His background includes project management, engineering, risk management, and planning for domestic and multinational corporations, insurance and financial institutions, construction companies, utilities, and the government. Mr. Eder is based in Madison, Alabama.

Prior to Facility Risk Consultants, Mr. Eder was stationed in Tokyo, Japan for 8 years and led all operations for ABS Consulting Inc. (formerly EQE International, Inc.) in Japan, China, Korea and Taiwan -- including risk consulting, structural engineering and design, probabilistic financial loss estimation, and the development and maintenance of management systems.

Mr. Eder has performed many post-earthquake reconnaissance studies -- most notably he led investigations of the M8.4 earthquake in Arequipa, Peru of June 2001; the M7.6 earthquake in Chichi, Taiwan of September 1999; and he was lead investigator of the M8.1 earthquake in Mexico of September 1985, for the US Electrical Power Research Institute (EPRI).

Prior to his assignment in Japan, Mr. Eder focused primarily in the seismic risk evaluation and seismic retrofit design of critical equipment and systems. Mr. Eder pioneered the development of many seismic risk evaluation procedures and criteria for the US and European nuclear power industry, the Seismic Qualification Utilities Group (SQUG), and the US Department of Energy (DOE). This included conducting a series of week-long seismic evaluation training courses for a total of about 500 engineers, and serving as subject matter expert and technical liaison for industry groups including the DOE Tiger Team.

Mr. Eder served as project manager or project consultant for the seismic risk surveys of critical equipment and systems at about 60 nuclear power plants in the US and Europe, and many DOE facilities. He has developed unique, cost-effective structural designs for new installations and seismic strengthening of structures, equipment, and distribution systems including raceways, piping, and HVAC ducting. He performed research for and supported many U.S. industry and professional groups, to advance the state-of-the-art of seismic risk assessment techniques and seismic design guidelines.

EDUCATION

UNIVERSITY OF CALIFORNIA, Berkeley: M.Eng., Structural Engineering and Structural Mechanics, 1982

CLARKSON COLLEGE OF TECHNOLOGY, Potsdam, New York: B.S., Magna Cum Laude, Civil and Environmental Engineering, 1980

REGISTRATION

California: Civil Engineer, 1985

Alabama: Civil Engineer, 2003

PROFESSIONAL AND BUSINESS AFFILIATIONS

American Society of Civil Engineers

Earthquake Engineering Research Institute

Structural Engineers Association of Northern California

Applied Technology Council

Tau Beta Pi National Engineering Honor Society

Phi Kappa Phi National Honor Society

American and British Chambers of Commerce in Japan

COMMITTEES -- PAST EXPERIENCE

- *Electric Power Research Institute* - Post Earthquake Investigation Team - Leader
- *U.S. Department of Energy* – Tiger Team Member – Natural Hazards Risk Analysis
- *U.S. Department of Energy* - Steering Committee on Natural Hazards – Technical Liason – Mechanical and Eletrical Equipment Evaluation and Design
- *Seismic Qualification Utility Group* – Equipment Seismic Evaluation Training – Lead Instructor and Subject Matter Expert
- Joint *American Society of Mechanical Engineers* and *Institute of Electrical and Electronics Engineers* - Special Seismic Qualification Working Group - CoChairman
- *National Center for Earthquake Engineering Research* – Critical Equipment Seismic Risk Analysis – Chief Researcher
- *National Fire Protection Association* (NFPA) – Seismic Technical Committee Member, NFPA-13.
- *Building Seismic Safety Council* - Seismic Rehabilitation Advisory Panel Member – Mechanical Equipment. NEHRP, FEMA 273.
- *American Society of Civil Engineers* – Electrical Raceway and HVAC Duct Seismic Design - Working Groups
- *Structural Engineers Association of California* - Seismology Subcommittee - Non-Building Structures and Equipment

SELECTED PUBLICATIONS & PRESENTATIONS

"Analysis of Ilo2 Plant Components Affected by the June 23, 2001 Mw 8.4 Arequipa, Peru Earthquake". Prepared for Hitachi Corporation. December 2002. Presented in London, U.K.

"The Use of Modeling and Natural Risk Analysis for Power Plants". Presented at Second International Conference on Mitigating Your Risks in Energy. February 2002. Singapore.

"Using Risk Based Inspection Techniques to Assess Maintenance of Power Plants". 2002. Presented at Second International Conference on Mitigating Your Risks in Energy. February 2002. Singapore.

"Preparing Your Properties for Major Earthquakes". 2001. Prepared for Architecture, Construction, and Engineering Subcomittee, American Chamber of Commerce in Japan. December 2001. Tokyo.

"Earthquake Hazards and Earthquake Risks in Tokyo". 2001. TELS-Setagaya, Earthquake Disaster Information and Preparedness Seminar. October 2001. Tokyo.

"Geographic Information Systems". 2000. Prepared for Non-Life Insurance Institute, ISJ Advanced Course 2000 Program, Natural Hazards and Underwriting Capacity. November 2000. Tokyo.

With J. O. Dizon, W. H. Tong, and E. R. Wong, 1999. "Chichi, Taiwan Earthquake of September 21, 1999 (M7.6). An EQE Briefing. Oakland, CA. October, 1999.

With G.S. Johnson, R.E. Sheppard, M.D. Quilici, and C.R. Scawthorn, 1999. "Seismic Reliability Assessment of Critical Facilities: A Handbook, Supporting Documentation, and Model Code Provisions." Technical Report MCEER-99-0008. Multidisciplinary Center for Earthquake Engineering Research, Buffalo, NY.

"Earthquake Risk of Independent Power Producer Stations", 1999. Prepared for Lloyd's Japan Power Seminar. June 1999. Tokyo.

With J. O. Dizon. "Risk Management for Power and Industrial Facilities -- Focus on Business Interruption". Second Biennial Federation of Asian Pacific & African Risk Management Organization. Manilla, Philippines. October, 1998.

"3 Years After the Hanshin-Kobe Earthquake, Earthquake Risk Management, Damage Assessment and Mitigation". 1998. High Pressure Gase Safety Association of Japan. Vol. 35, No. 2 (1998). Tokyo.

With G. S. Johnson, R.E. Sheppard, and S.P. Harris. 1998. "A Method to Assess and Improve the Operational Reliability of Critical Systems Following Earthquakes." Presented at the 6th U.S. National Conference on Earthquake Engineering, Seattle, WA, June 1998.

With G. S. Johnson, R.E. Sheppard, and S.P. Harris. 1998. "The Development of Model Code Provisions to Address System Reliability Following Earthquakes." Presented at the ATC-29-1 Seminar on Seismic Design, Retrofit, and Performance of Nonstructural Components, San Francisco, CA, January 1998.

With D. W. Jones, M. K. Ravindra, C. R. Scawthorn, and K. Iida. 1996. "Earthquake Risk Management for Process Industries". High Pressure Gas Safety Institute of Japan. Vol. 35, No. 5 (1996). Tokyo.

With G. A. Antaki. 1994. "Recommended Provisions for Equipment Seismic Qualification Consistent with IEEE and ASME Criteria for Use of Experience." ASME 1994, PVP-Vol. 275-2, Seismic Engineering, Volume 2.

With P. J. Butler and R. P. Kassawara. 1994. "Application of the Generic Implementation Procedure Methodology to Demonstrate Seismic Adequacy of New and Replacement Equipment and Parts in USI A-46 Plants." ASME 1994, PVP-Vol. 275-2, Seismic Engineering - Volume 2. Proceedings American Power Conference, Illinois Institute of Technology, April 1994, Chicago, Illinois.

With N. P. Smith and R. P. Kassawara. 1994. "Future Direction for the Use of Earthquake Experience Data." Proceedings American Power Conference, Illinois Institute of Technology, April 1994, Chicago, Illinois.

With M. W. Eli and M. W. Salmon. November 1993. "Walkthrough Screening Evaluation Field Guide, Natural Phenomena Hazards at Department of Energy Facilities." UCRL-ID-115714, Revision 2. Lawrence Livermore National Laboratory.

"Seismic Design of Important Systems and Components--Functionality Considerations." 1993. Structural Engineers Association of Northern California, 1993 Fall Seminar, Nonstructural Components: Design and Detailing. San Francisco, California.

With C. Scawthorn, M. Zadeh, and G. Johnson. 1993. "Economic Impacts of Earthquake Damage to Nonstructural Components." 40th North American Meetings of the Regional Sciences Association International, Houston, Texas.

With M. W. Barlow, R. J. Budnitz, and M. W. Eli. 1993. "Use of Experience Data for DOE Seismic Evaluations." 4th DOE Natural Phenomena Hazards Mitigation Conference, Atlanta, Georgia.

With K. Porter, G. S. Johnson, M. M. Zadeh, and C. Scawthorn. 1993. "Seismic Vulnerability of Equipment in Critical Facilities: Life-safety and Operational Consequences." Technical; Report NCEER-93-0022. National Center for Earthquake Engineering Research.

With J. K. Arros. 1993. "Applications of Experience-based Methods for Seismic Qualification of Distribution Systems." Prepared for Advanced Reactor Corporation FOAKE ALWR Seismic Qualification Project.

With MPR Associates and Winston and Strawn. 1993. "Verifying the Seismic Adequacy of New and Replacement Equipment and Parts." Prepared for the SQUG Management Guidelines Document.

With Lawrence Livermore National Laboratory. 1992. "Program Plan for the Evaluation of Systems and Components in Existing DOE Facilities Subject to Nataral Phenonema Hazards." Prepared for the U.S. Department of Energy.

With J. O. Dizon, P. D. Baughman, and G. S. Johnson. 1992. "Peer Review of the Watts Bar Nuclear Plant Integrated Interaction Program Suspended Systems Proximity Task." Prepared for Tennessee Valley Authority.

With G. S. Hardy, G. S. Johnson, and R. W. Cushing of EQE; MPR; S&A; and URS. 1992. "Walkdown Screening and Seismic Evaluation Training Course." Prepared for Seismic Qualification Utility Group.

With M. W. Salmon. 1992. "Technical Safety Appraisal of the Idaho Chemical Processing Plant, NPH Discipline." Prepared for the U.S. Department of Energy.

With M. W. Eli. 1992. "NPH Walkdown Evaluation Summary Report - Paducah Gaseous Diffusion Plant." Prepared for the U.S. Department of Energy.

With G. S. Johnson, R. H. Kincaid, and G. S. Hardy. 1992. "High-rise Building Critical Equipment Study." Prepared for National Center for Earthquake Engineering Research.

With K. E. Smith. 1992. "Seismic Performance of Standby and Emergency Power Engine Generator Systems." Prepared for National Center for Earthquake Engineering Research.

With M. W. Eli. 1991. "Use of Earthquake Experience Data." Prepared for the Third DOE Natural Phenomena Hazards Mitigation Conference, St. Louis, Missouri.

With J. O. Dizon. 1991. "Advancement in Design Standards for Raceway Supports and Its Applicability to Piping systems." PVP-Volume 210-1, Codes and Standards and Applications for Design and Analysis of Pressure Vessel and Piping Components. ASME 1991.

"Cable Tray and Conduit System Seismic Evaluation Guidelines." March 1991. EPRI Report NP-7151. Prepared for the Electric Power Research Institute. San Francisco, CA: EQE International.

With G. S. Johnson. March 1991. "The Performance of Raceway Systems in Strong-motion Earthquakes." EPRI Report NP-7150. Prepared for the Electric Power Research Institute. San Francisco, CA: EQE International.

With G. S. Johnson. March 1991. "Longitudinal Load Resistance in Seismic Experience Data Base Raceway Systems." EPRI Report NP-7153. Prepared for the Electric Power Research Institute. San Francisco, CA: EQE International.

With J. P. Conoscente and B. N. Sumodobila. March 1991. "Seismic Evaluation of Rod Hanger Supports for Electrical Raceway Systems." EPRI Report NP-7152. Prepared for the Electric Power Research Institute. San Francisco, CA: EQE International.

With Winston & Strawn, MPR Associates, Inc., etal. June 1991. "Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment." Revision 2. Prepared for the Seismic Qualification Utility Group.

With M. W. Eli and L. J. Bragagnolo. 1991. "Walkthrough Screening Evaluation Field Guide, Natural Phenomena Hazards at Department of Energy Facilities." Special Release for 3rd DOE Natural Phenomena Hazard Mitigation Conference, October 1991, St. Louis, Missouri.

With L. J. Bragagnolo and J. P. Conoscente. 1990. "A Proposed Methodology for the Seismic Design of Rectangular Duct Systems." Applied Technology Center (ATC) Seminar on Seismic Design and Performance of Equipment and Nonstructural Elements in Building and Industrial Structures, Irvine, California. ATC-29.

With J. J. Johnson and N. P. Smith. 1990. "Developments of the Seismic Qualification Utility Group." Applied Technology Center (ATC) Seminar on Seismic Design and Performance of Equipment and Nonstructural Elements in Building and Industrial Structures, Irvine, California. ATC-29.

With W. Djordjevic, J. Eidinger, and F. Hettinger. 1990. "American Society of Civil Engineers Activities on Seismic Design of Electrical Raceways." Current Issues Related of Nuclear Power Plant Structures, Equipment, and Piping. Proceedings of the Third Symposium, Orlando, Florida, December 1990.

With H. L. Williams. 1990. "Qualification of Cable Tray Supports by Earthquake Experience Data: Application at H. B. Robinson Plant" Current Issues Related of Nuclear Power Plant Structures, Equipment, and Piping. Proceedings of the Third Symposium, Orlando, Florida, December 1990.

With R. P. Kennedy, J. D. Stevenson, J. J. Johnson, W. R. Schmidt, and K. Collins. June 1990. "Watts Bar Civil Program Review." Prepared for Tennessee Valley Authority.

With J. P. Conoscente, B. N. Sumodobila, and S. P. Harris. 1989. "Seismic Fatigue Evaluation of Rod Hung Systems." Prepared for the *Tenth Conference on Structural Mechanics in Reactor Technology*, (SMiRT).

With P. D. Smith and J. P. Conoscente. December 1988. "SQUG Cable Tray and Conduit Evaluation Procedure." Paper presented at the Second Symposium on Current Issues Related to Nuclear Power Plant Structures, Equipment and Piping, Orlando, FL.

With P. I. Yanev. 1988. "Evaluation of Cable Tray and Conduit Systems Using the Seismic Experience Data Base." *Nuclear Engineering and Design* (North-Holland, Amsterdam) 107: 149-153.

With S. P. Harris, P. D. Smith, and J. E. Hoekendijk. October 1988. "Performance of Condensers and Main Steam Piping in Past Earthquakes." Report prepared for General Electric Nuclear Energy Boiling Water Reactor Owners Group. San Francisco: EQE Engineering.

With J. J. Johnson, G. S. Hardy, N. G. Horstman, G. Rigamonti, M. R. Reyne, and D. R. Ketcham. August 1988. "Technical Basis, Procedures and Guidelines for Seismic Characterization of Savannah River Plant Reactors." E. I. Dupont De Nemours & Co, Aiken, South Carolina.

With S. P. Harris, P. S. Hashimoto, J. O. Dizon, B. Sumodobila, G. M. Zaharoff, and L. J. Bragagnolo. March 1988. "Seismic Evaluation of the High Flux Isotope Reactor Primary Containment System." Report prepared for Martin Marietta Energy Systems, Inc. San Francisco: EQE Engineering.

With S. W. Swan, "Summary of the Effects of the 1985 Mexico Earthquake to Power and Industrial Facilities." Proceedings of the American Society of Civil Engineers International Conference on the 1985 Mexico Earthquake, Factors Involved and Lessons Learned, Mexico City, Mexico, September 1986.

With A. F. Kabir and S. Bolourchi, "Seismic Response of Pipes Supported on Complex Framing Systems." Proceedings of the American Society of Civil Engineers Structures Congress, New Orleans, Louisiana, September 1986.

With S. W. Swan, "The Mexico Earthquake of September 19, 1985; Performance of Power and Industrial Facilities," Proceedings of the Third U. S. National Conference on Earthquake Engineering, Charleston, South Carolina, August 1986.

"Performance of Industrial Facilities in the Mexican Earthquake of September 19, 1985," Electric Power Research Institute Report No. NP-4605, Project 1707-30 Final Report, Palo Alto, California, June 1986, also presented at the IEEE Power Engineering Society Summer Meeting, Mexico City, Mexico, July 1986.

"Earthquake Response Analysis of a Braced Offshore Platform," University of California, Berkeley (June 1982), also American Petroleum Institute, October 1982, San Francisco, California.

FARID ELSABEE

SUMMARY Engineering: Analysis, Design, Inspection / Audit, and Management

Over twenty five years of extensive experience in the nuclear power industry undertaking and managing retrofit and analysis projects. Experience includes linear and non-linear analysis of static and dynamic structural and mechanical systems where compliance with codes, standards, and specifications is required. The methods used include detailed finite element analyses. Expertise was acquired at both Architect / Engineer and consulting firms using and developing state-of-the-art analysis techniques for nuclear power plants. Experience includes plant walkdowns at DOE facilities (Rocky Flats and Hanford) and numerous nuclear power plants (including SEP and USI A-46 related walkdowns) for review of seismic adequacy of various types of equipment. Attended EPRI sponsored training programs and certification for:

- CHEC Family of Codes, for Erosion Corrosion
- Walkdown Screening and Seismic Evaluation for USI A-46 (as a Seismic Capability Engineer)
- Add-on Seismic IPE, Seismic PRA and SMA for Seismic IPE Reviews

EDUCATION

MS Civil Engineering (Structures), Massachusetts Institute of Technology, 1975

BS Engineering, SUNY at Stony Brook, 1973

EXPERIENCE

APPLIED ENGINEERING ASSOCIATES (1990 - present)

Principal with responsibilities in client development, applied mechanics projects, technical reviews and development and verification of CAE/CAD/FEM and data base management applications. Consulting projects consisted of:

Providing Analysis and Field Support at TVA Browns Ferry Unit 1 (Through Facility Risk Consultants and Bèchtel Corporation) for the following projects:

- Seismic evaluation and field walkdowns of plant Equipment in support of the resolution of USI A-46 requirements as a Seismic Capability Engineer. The work included plant mechanical and electrical equipment required for safe shutdown, including all plant cable tray and conduit systems. Evaluation of seismic II over I interactions, of piping and other support systems, was also performed to insure adequate non seismic supports in the vicinity of the safety related equipment. All work was performed per the requirements of the SQUG GIP. Recommendations for resolution of outliers were also prepared.
- Development of a generic design for the installation of new cable tray and conduit systems following the guidelines used in the USI A-46 evaluations. The designs were prepared and presented in a procedure format which is extremely easy to follow and implement by field electricians and engineers.

Providing Stress analysis support at ABS Consulting for the following projects:

- Seismic analysis of the refueling machine at the Bruce Nuclear Power Plant using a finite element model of the machine and the entire supporting structure. The dynamic analysis was performed using a linear elastic model of the machine head and suspension assemblies as well as the supporting frames, columns, elevators, bridge and carriage. The analysis of the detailed model, using the SAP 2000 software, resulted in the elimination of numerous modifications planned for implementation prior to the restart of the unit.
- A three dimensional non-linear analysis of various Main Steam pipe clamps at snubber locations at the Pilgrim Nuclear power station. The clamp stresses were determined using static analyses of the non-linear models, which accounted for friction, gaps and interface loads at the pipe to clamp interface, using the ANSYS finite element program. The stress evaluations, performed for the mechanical loads applied by the snubbers to the pipe, were per the ASME Section III requirements.

Providing Project Engineering support at Altran Corporation for several projects including:

- Analysis, testing, project engineering and coordination for the evaluation and analysis of Steam Generator internals at Indian Point Unit 2. The project was in support of the restart effort associated with the tube leak resulting from Primary Water Stress Corrosion Cracking at a Row 2 U-bend tube. The project consisted of FEA of tube support plates and U-bend tubes; crack initiation and growth studies; experimental testing of U-bends; development of degradation assessments, condition monitoring and operational assessment of the secondary side components.
- Analysis, design and implementation of a large bore snubber reduction project for the steam generator support structures at Indian Point Unit 2.
- Inspection of roof structures and supporting decks, including development of repair specifications, for Con Edison's 59th Street fossil station.
- Assessment, evaluation and closure of condition reports and preparation of structural, mechanical and piping calculation revisions for Indian Point Unit 2.

Supporting Millstone Unit 2 (MP2) Design Engineering staff in day to day activities for a duration of six years, including their major recovery effort. Responsibilities included:

- Responsible engineer for seismic qualification and heavy load drop issues;
- Supporting the Configuration Management Project and Independent Corrective Action Verification Project (ICAVP) in investigating and addressing design basis issues and discrepancies in the areas of seismic equipment qualification, seismic structural analysis and design, heavy load drop analyses, and fuel rack design issues. This work was in support of the 50.54(f) effort and included interfacing with the NRC and the ICAVP contractor;
- Design Basis reconstitution of the reactor coolant loop system and the major NSSS components.
- Investigations of noted adverse conditions (Condition Reports) for design basis reviews of plant configurations;
- Review and preparation of FSAR changes and license amendments;
- Preparation of reportability and operability determinations, corrective actions and Licensee Event Reports;

- Preparation and review of plant modification packages for structural and equipment modifications;
- Preparation and review of calculations, DCN's, Technical Evaluations, Specifications and Procedures;
- Providing support for the reconstitution of the High Energy Line Break (HELB) program.
- Review and close out of administrative department items such as Assignment Requests, Engineering Work Requests, Project Files, etc; and
- Providing vendor interface.

USI A-46 and IPEEE related consulting services at Northeast Utilities for three plants (CY, MP1 and MP2). Work included development of a detailed project description, outline and plan, a program manual/instruction for the implementation phase, and a specification for new and replacement parts/components based on the EPRI/ STERI process to be used for plant end of life.

The project also included specific support for MP2 associated with plant walkdowns per the GIP (for USI A-46) and EPRI NP6041 (for IPEEE); interface with and managing other consulting companies; coordination of efforts between the appropriate Engineering and plant operations departments within the utility; preparing modification packages to correct identified outliers; providing response to NRC requests for additional information and justification of methods used (MP2's response on GIP Method A issues was identified by NRC as a good model to be used in other utilities' responses); providing interface with the NRC and SQUG steering group on generic responses to NRC questions; and preparing and updating final reports.

Responsible for the structural aspects of an Appendix R fire protection modification which improved on the existing oil collection system to the RCP motors of a PWR unit.

Seismic equipment qualification consulting services related to the MOV Generic Letter 89-10 (CY, MP1, MP2, MP3), USI A-46 & IPEEE (MP2) and numerous plant design changes (CY, MP1, MP2); including test plan development, seismic qualification (in accordance with IEEE 344-1975) and development of replacement part requirements for a complete hydraulic power unit and it's controls by shake table testing (CY). The GL 89-10 work included review and update of MOV seismic qualification (weak link) reports, for Northeast Utilities and New York Power Authority, covering six nuclear plants, to maximize the valve structural thrust capacity by eliminating conservatisms found in existing qualification reports and previously used criteria. Prepared, managed and implemented two associated modification packages.

Development of a Seismic Equipment Qualification Manual, for Northeast Utilities, which is applicable to four nuclear units spanning in design basis history from the SEP plants to a recently designed plant which is in full conformance with the SRP.

Development of a specification for the seismic evaluation of existing rod hung electrical raceway systems to allow addition of new cables as well as for the design of new systems.

Evaluation of an Erosion / Corrosion inspection program for MP3 using EPRI's recommended practices. Supported the overall E/C program and the structural evaluation of the measured data at two plants (MP1 & MP2).

Technical audits of the design and installation of piping, cable tray and conduit supports on the Philippines Nuclear Power Plant as part of an independent assessment of the design of the plant.

CYGNA ENERGY SERVICES (1985 - 1990)

Senior Project Manager

Technical Specialist

Division Manager

Responsibilities included development and verification of CAE/CAD/FEM applications, analysis and design, bid preparation, client development, project management and technical review for a number of diversified projects related to structural, equipment qualification, configuration management, licensing reviews, and Safety System Functional Inspections. Major specific projects included:

Specialized evaluation of non-conformance's associated with wind, tornado and seismic events for mechanical and electrical components using finite element analysis and simplified calculations.

Development and implementation of an asbestos survey program which included the identification, gathering and assimilation of the data into a computerized asbestos tracking system.

Gathering, assimilation and compilation of data into a Data Base Management System (DBMS) for several inspection programs which are part of a major fossil plant life extension program.

Independent review of a major drawing discrepancy resolution and update project. This effort which is part of the utility's configuration management program includes instrumentation, electrical, mechanical, and piping systems.

Participated in the Safety System Functional Inspections (SSFI) of the pneumatic systems at a nuclear power plant. Responsibilities included preparations, inspections and reviews for structural and seismic related topics.

Seismic and tornado evaluation of two existing structures against possible collapse using realistic structural characteristics and inelastic responses.

Supervised and coordinated projects in the area of erosion/corrosion of carbon steel piping and microbe induced corrosion in buried lined pipes.

STONE & WEBSTER ENGINEERING CORPORATION (1973 - 1977 & 1984 - 1985)

Engineer

Project Management

Responsible for several specialized projects in the area of seismic equipment qualification and large bore snubber elimination. Also participated in the development of the state-of-the-art procedures for soil structure interaction analysis. A few of the more noteworthy projects include:

Responsible for the analysis, design, and plant change packages for the elimination of large bore snubbers supporting steam generators and reactor coolant pumps on the primary coolant loop piping of three PWR's.

Development of generic seismic acceptance criteria of non-safety related equipment and review of numerous seismic equipment qualification vendor reports; including performing a seismic re-qualification of a 125 ton crane while accounting for its flexible girder support system.

Evaluation and resolution of safety related hazards resulting from postulated internally generated missiles.

Performed seismic analyses of various structures, including both detailed and simplified soil-structure interaction procedures and developed structural design forces and in-structure response spectra.

Performed dynamic analyses of a BWR Mark II containment structure for hydro-dynamic loads.

URS/JOHN A BLUME AND ASSOCIATES (1979 - 1984)

Equipment Qualification

Project Manager

Developed specialized expertise in the area of seismic equipment qualification. Experience includes the development and implementation of full-scale seismic shaking-table testing programs; in-situ low-level excitation modal testing programs; innovative, state of the art techniques in seismic qualification by combined testing and analysis procedures; and field investigations / plant walkdowns consisting of "Expert Earthquake Engineers" to evaluate the seismic ruggedness of existing equipment.

Attended and participated in early SQUG meetings where development of seismic equipment evaluation methods using experience gained from performance of equipment during strong motion earthquakes was being formulated. Some of the projects undertaken include:

Development of realistic anchorage guidelines for EPRI and SQUG to be used in the SQUG-GIP for the resolution of generic issue A-46.

Seismic evaluation of various types of mechanical and electrical equipment at nuclear power plants using finite element and simplified analyses, in-situ testing, shake-table testing and combined techniques. Some of the more significant equipment have included existing laterally flexible cable trays and conduits at the SEP plants (this work was subsequently used by the SQUG in the development of the GIP guidelines), control room panels and boards, motor control centers, switchgears and diesel generators.

Evaluation of more than 60 items of mechanical and electrical equipment for seismic and tornado loads at the DOE Rocky Flats Facility. Structural integrity and operability evaluations for the seismic effects were performed using analytical techniques, in-situ modal testing, and full-scale shaking-table testing.

Evaluation of the seismic ruggedness of existing equipment at the DOE Hanford facility and the Nine Mile Point Unit 1 nuclear power plant as a member of a team of "Expert Earthquake Engineers".

IMPELL CORPORATION (EDS Nuclear) (1977 - 1979)

Supervisor

Senior Engineer

Responsible for the technical and administrative management of specialized projects such as pipe ruptures, finite element analysis and soil structure interaction. Analysis and development of pipe restraint design loads due to pipe whip and jet impingement effects resulting from postulated pipe breaks at several nuclear power plants. Analysis procedures included both simplified energy balance techniques and detailed non-linear analysis.

Responsible for the structural design evaluation of miscellaneous structural steel frames for pipe support and pipe rupture loads.

Performed the seismic analysis of all Category I structures at a BWR plant including soil structure interaction and generation of design forces as well as amplified floor response spectra.

Performed static and dynamic analysis of a BWR Mark I suppression pool torus for hydrodynamic loads.

Responsible for finite element stress analysis of various piping components such as sweepolets, reducers, and anchoring devices.

PUBLICATIONS

"Seismic Investigation of Electrical Raceway Components," with L. Serdar, Jr. and D. Williams, ASME paper 84-PVP-43, PVP Conference and Exhibition, San Antonio, Texas (June 1984)

"A Seismic Evaluation Study of Mechanical and Electrical Equipment at an Existing Nuclear Handling Facility," with L. Serdar, Jr., et al., 7th International Conference on Structural Mechanics In Reactor Technology, Chicago, Illinois (August 1983)

"Seismic Evaluation of Electrical Raceway Systems," with S. Anagnostis and W. Djordjevic, ASME paper 83-PVP-18, 4th National Congress on Pressure Vessel and Piping Technology, Portland, Oregon (June 1983)

"A Survey and Assessment of Major Mechanical Equipment Capability During Seismic Events," with W. Djordjevic, ASME Pressure Vessel and Piping Conference and Exhibit, Orlando, Florida (June 1982)

"The Spring Method For Embedded Foundations," with E. Kausel, et al., Nuclear Engineering and Design Journal, Volume 48 (August 1978)

"Dynamic Analysis of Embedded Structures," with E. Kausel, et al., 4th international Conference on Structural Mechanics In Reactor Technology, San Francisco, California (August 1977)

"Dynamic Stiffness of Embedded Foundations," with E. Kausel, and J. M. Roesset, ASCE 2nd Annual Engineering Mechanics Division Speciality Conference, North Carolina (May 1977)

"Static Stiffness Coefficients For Circular Foundations Embedded In An Elastic Medium," M. S. Thesis, Massachusetts Institute of Technology, Cambridge, Massachusetts (June 1975)

ROBERT D. HOOKWAY, P.E.

PROFESSIONAL HISTORY

Hookway Engineering, Consulting Engineer, 1996-present

EQE International, Stratham, New Hampshire, Technical Manager, 1990-1996

Teledyne Engineering Services, Waltham, Massachusetts, Manager, 1967-1990

PROFESSIONAL EXPERIENCE

Mr. Hookway has over 35 years of professional engineering and project management experience. Specific background experience includes project management for design, analysis and evaluation of piping system and pressure vessels, supports and structures in power generation stations (fossil and Nuclear), petrochemical facilities, and Navy nuclear installations considering weight, thermal, seismic and dynamic loadings.

As a member of the ASME Section III Working Group on Piping Design, ASME B31.3 Main Piping Committee, and a past member of the ASME Section III Special Working Group on Seismic Design Rules, he is familiar with the requirements of and is involved in the development of these design codes for piping. Past projects include: litigation support for fossil plant piping failure, seismic evaluation of piping, mechanical, electrical and I & C equipment at the Paks Nuclear Plant in Hungary, seismic upgrade of piping at various U.S. Nuclear facilities, anchor bolt study to investigate the dynamic capacities of concrete expansion bolts, seismic margins evaluation of critical piping at the Haddam Neck Nuclear Plant, various piping design and evaluation projects at nuclear and fossil generation stations, petrochemical and cryogenic plants.

Mr. Hookway has completed the Seismic Qualification Utility Group (SQUG), Seismic Capacity Engineer (SCE) training required by the USNRC for A-46 evaluations. He has been involved with A-46 and other similar projects at numerous nuclear facilities in the U.S. and Internationally. Project management responsibilities include a variety of piping, mechanical, and civil/structural consulting engineering projects. Tasks include project planning, technical direction, manpower assignment, cost and schedule control, client interface, and writing of technical procedures.

Some of Mr. Hookway's specific projects include the following:

- Design Review support for General Electric HRSG' projects. The scope includes a detailed review of the design process for all Balance of Plant high energy piping systems and pipe supports provided by the G.E selected Architect Engineer. Services also include general consulting to GE on an as-needed basis regarding ASME code compliance and piping design.

- Design Analysis & Evaluation for numerous LNG facility upgrade projects.

- Expert witness and engineering support for litigation of feedwater piping <u>flow accelerated corrosion</u> (erosion/corrosion) failure at a midwest fossil powered electric generation plant. Engineering support for this project included complete documentation search and reviews, plant operations and maintenance reviews, detailed system flow evaluations, white paper preparation for various associated issues, depositions, deposition reviews and technical guidance for the legal staff. Discipline expertise for this project included pipe stress and design, failure evaluation, metallurgical, fluid mechanics, non-destructive examination and water chemistry.

- Millstone II Nuclear Power Plant 9/97-present: Design Engineering support for resolution of piping design issues considering weight, thermal expansion, seismic and other dynamic loadings to satisfy US Nuclear Regulatory Commission concerns relative to configuration control of the plant. In addition, provided support for the corrective action department performing Condition Report investigations, Root Cause analyses, Corrective action prescription, Operability Determinations, and Licensee Evaluation Reports (LER) preparation.

- Millstone III Nuclear Power Plant - 11/96-8/97 50.54f Restart Oversight Group - participated in the assessment of numerous systems (SSFI/vertical slice) in addition to the assessment of selected engineering programs such as erosion/corrosion, stress data packages, Seismic Design, Active Valves and Components, P&ID Upgrades, and Component Labeling.

- Connecticut Yankee Nuclear Plant - 6/96-11/97 Configuration Management Program (CMP) responsible for all structural tasks within the CMP. Tasks included Graded System Reviews and Topical Reports for selected Engineering Programs (e.g. Seismic Design, Piping Design and Structural Design)

- Task manager for the recently completed seismic qualification task for the Category I(L) piping at Tennessee Valley Authority's Watts Bar Nuclear Plant. This task included the walkthrough and bounding case evaluations for all piping interaction issues (proximity, shakespace flexibility and II/I).

- Lead engineer for the seismic upgrade of equipment at the Paks Nuclear Power Plant in Hungary. The plant is a four unit VVER type 213 Russian design plant. The project included walkdown screening evaluation for all equipment, seismic capacity evaluations and preparation of modification designs.

- Project manager for the expansion anchor bolt study to remove conservatism in design criteria for nuclear power plants, including dynamic anchor bolt testing as part of the EPRI study of Improved Guidelines and Criteria for Nuclear Piping and System Evaluation and Design. Scope of work included test plan preparation, supervision of anchor bolt testing, interpretation of test data, final report preparation, and EPRI interface.

- Project manager for anchor bolt testing in cracked concrete. The purpose of this project was to develop capacities for epoxy anchor bolts and to assist in the development of epoxy anchor bolts use for the power industry.

- Design, analysis, and evaluation of nuclear Class 1, 2, and 3 piping systems. Plants include Nine Mile 1, Vermont Yankee, Millstone 1, Hatch 1, Hatch 2, North Anna I, North Anna II, Davis Besse, DC Cook, Millstone 2, Millstone 3, Pilgrim, Fitzpatrick Hope Creek, and Monticello.

- Preparation of design specifications for Class 1 piping systems.

- Project management for various failure evaluation, plant modification, and new system design projects.
- Lead engineer for the evaluation of various pressure vessels, heat exchangers and valves for navy nuclear, commercial nuclear and process plant facilities.
- Direct interface with the NRC and provided technical management services. As project manager for "The Study to Determine the Effects of Hydrodynamic Loads on the Control Rod Drive Piping System for BWR Plant Designs," he managed the technical investigation and provided a communication link between the BWR Owner's Group (which included 11 utilit4ies and 1 plants) and the NRC.
- Participated in design reviews for the Millstone I, Vermont Yankee, Nine Mile Point I, LaSalle, and Comanche Peak (which was under contract to the NRC).
- A study to "Determine the Effects of Postulated Events Devices on Normal Operation of Piping systems in Nuclear Power Plants" the NRC. The results of this work contributed to the refinement of code criteria and NRC regulatory guides for seismic pipe whip restraint design. Plants included in the study included Farley Unit 2, Diablo Canyon and Zimmer.
- A study to upgrade the design criteria for submarine sea connected piping using high strength thin-walled piping material. This work was performed for the David Taylor Naval Research Center.

EDUCATION

NORTHEASTERN UNIVERSITY, Boston, MA: M.S. Mechanical Engineering, 1970

LOWELL TECHNOLOGICAL INSTITUTE: B.S. Mechanical Engineering, 1963

AFFILIATIONS

Member, ASME Section III Working Group Piping Design

Member, ASME B31.3 Chemical Plant and Petroleum Piping Committee

Member, Past Vice Chair, Pressure Vessel and Piping Subcommittee of the ASME NED

Past Member ASME Section III Special Working Group Seismic Design Rules

American Society of Mechanical Engineers

REGISTRATION

Professional Engineer: Massachusetts

Professional Engineer: Virginia

PUBLICATIONS

"Effects of Postulated Events Devices on Normal Operation of Piping Systems in Nuclear Power Plants." 1981. NUREG/CR-2136. U.S. Nuclear Regulatory Study.

With S. J. Eder and T. R. Kipp. 1991. "Commodity Clearance Requirements." Engineering Specification N3C-941. Prepared for Tennessee Valley Authority.

With S. J. Eder and T. R. Kipp. 1991. "Seismic Qualification of Category I(L) Fluid System Components and Electrical or Mechanical Equipment." Design Criteria WB-DC-40-31.13. Prepared for Tennessee Valley Authority.

With S. J. Eder and T. R. Kipp. 1991. "Seismic Design Specification for Category I (L) Piping, Pipe Supports, and In-line Components." Engineering Specification N3C-943. Prepared for Tennessee Valley Authority.

With R.D. Campbell, T.R. Roche, P.D. Baughman, S.J. Eder 1995 "Use of Seismic Experience Data for Seismic Verification of VVER Reactors" International Atomic Energy Agency Coordinated Research Program.

RICHARD L. W. TIONG, P.E.

PROFESSIONAL HISTORY

Independent Consultant, 2000 – present

EQE International (S) Pte. Ltd, Singapore, Senior Consultant, 1995 - 2000

EQE International, Irvine, California, Project Engineer to Technical Manager and Associate, 1986 - 1995

Structural Mechanics Associates, San Ramon, California, Staff Engineer, 1982 - 1985

PROFESSIONAL EXPERIENCE

Mr. Tiong has over twenty years of experience in the technical execution of projects requiring the design, construction, and integrity evaluation of civil structures. He started his consulting practice in California in 1982 after graduation from the Master's program at UC Berkerley, California, where he specialized in structural engineering and structural mechanics with emphasis on the seismic aspects. He has been involved in a wide variety of projects such as seismic risk assessment, structural and equipment fragilities quantification, soil-structure interaction analyses of nuclear facilities, earthquake protection of structures and critical equipment against seismic-induced damage, risk-based inspection (RBI) of offshore platforms, among others. He has performed numerous post-earthquake investigations in California, Japan, Taiwan and Indonesia. Some salient aspects of his previous work experiences are summarized in the following.

Mr. Tiong was exposed to detailed finite element analysis work in the early 80's while working for Structural Mechanics Associates, California. He is also well-versed in performing seismic response analyses including the effects of dynamic soil-structure interaction. He has also participated in research projects aimed at calibrating analytical methods for estimating structural response using actual recorded data obtained during strong motion earthquakes. These projects involved the Lotung scale model containment structures located in Taiwan, and the Pacific Bell telephone building at Watsonville. Mr. Tiong is conversant with Seismic Probabilistic Risk Assessment (PRA) and Seismic Margins methodology as practiced in the US for nuclear facilities, having direct involvement in over 10 such studies in the U.S., Japan and Korea. This type of study focuses on realistic failure modes of structures and components, to determine realistic factors of safety under seismic conditions.

Since joining EQE in 1986, Mr. Tiong participated in numerous seismic projects utilizing both the conventional structural dynamics as well as experience-based methodologies. These included the Seabrook cable tray seismic evaluation project, Browns Ferry Unit 2 cable tray and conduit seismic verification, and Comanche Peak II/I evaluation for non-safety, non-seismic large and small bore piping. He had contributed significantly in the early development work on the limited analytical review guidelines for SQUG conduit and cable tray raceway supports. Mr. Tiong completed the SQUG-sponsored walkdown screening and seismic evaluation training course as Seismic Capability Engineer in January 1993, and was actively involved in numerous seismic verification walkdowns of nuclear power plant equipment to support USI A-46 resolution for many U.S. plants.

In late 1995, Mr. Tiong transferred to EQE's Singapore office where his interests expanded to areas such as infrastructure design, structural integrity/damage assessment and rehabilitation activities. The type of structures included commercial buildings, manufacturing facilities, bridges, marine terminals and offshore structures. In December 1998, he was engaged by insurance interests to perform earthquake damage assessment at a large timber processing facility in Indonesia. He formulated and directed the field investigation program and retrofit design for a building owned by Unocal in Indonesia. The building suffered from under-strength concrete and required retrofit strengthening to meet ACI code. He has also performed integrity assessment of a jetty, including load test to address short-term operability concerns. He has also performed seismic evaluation walkdowns of mechanical and electrical equipment on an offshore platform located in Baku, Azerbaijan.

Practicing as an independent consultant since 2000, Mr. Tiong has been actively involved in the seismic strengthening of semiconductor manufacturing equipment and distribution systems in several wafer fabrication plants, and seismic consultation and design review for new fab construction in Hsin-Chu and Tainan Science-Based Parks in Taiwan. He is also involved in the seismic risk assessment of several oil refineries for Mitsubishi in Japan, RBI projects for BP/Amoco in Indonesia and Thailand. Currently, Mr. Tiong is actively involved in the Browns Ferry Unit 1 restart project

EDUCATION

UNIVERSITY OF CALIFORNIA, Berkeley: M.S. Structural Engineering and Structural Mechanics, 1981

UNIVERSITY OF LONDON, England: B.Sc (Engineering) 1st Class Honors, Civil Engineering, 1978

REGISTRATION

California: Civil Engineer since 1984

PUBLICATIONS

A selection of technical reports and journal articles for which Mr. Tiong is a principal contributor are lised below:

With A.P. Asfura, et. al., "Seismic Analysis of Multiple Supported Bridges," *Conference Proceeding, Bridge into the 21st Century,* Hong Kong, 2-5 October, 1995.

With R.D. Campbell and J.O. Dizon, "Response Predictions for Piping Systems Which Have Experienced Strong Motion Earthquakes," *ASME Pressure Vessel and Piping Conference,* Nashville, Tennessee, June 17-21, 1990.

With M.K. Ravindra, "Comparison of Methods for Seismic Risk Quantification." *Proceedings of 10th SMiRT Conference,* Anaheim, California, August 14-18, 1989.

With O.R. Maslenikov, J.J. Johnson, and M.J. Mraz, "Seismic Analysis of the MFTF Facility." In *Proceeding of 8th SMiRT Conference,* Brussels, Belgium, August 19-23, 1985.

With O. R. Maslenikov, J.J. Johnson, M. J. Mraz, S. Bumpus, and M.A. Gerhard. "SMACS - A System of Computer Programs for Probabilistic Seismic Analysis of Structures and Subsystems,

Volume I User's Manual, Volume II Example Problem." SMA 12211.31.01/12211.31.02. Prepared for *Lawrence Livermore National Laboratory*, 1984

With J.J. Johnson and B.J. Benda. "Stress Analysis of the Neutral Beam Pivot Point Bellows for Tokamak Fusion Test Reactor." In *Proceeding of the 10th Symposium on Fusion Engineering*, Philadelphia, PA, 1983.

APPENDIX B:
SUMMARY OF EQUIPMENT AND SUBSYSTEMS FOR SEISMIC MARGIN EVALUATION (SEISMIC REVIEW SSEL)

SSEL NUMBER	CLASS	EQUIPMENT I.D.	DESCRIPTION	BUILDING	ELEV.	ROOM	ISSUE	TRAIN
10001	00	1-HCU-85,1-185	CRD/HYDRAULIC CONTROL UNIT	U1 RB	565	R2 & R6/P-S	AI	1,2
10002	07	1-FCV-85-82A	CRD/WEST SDV VENT VALVE	U1 RB	565	R2/S	AI	1
10003	07	1-FCV-85-82	CRD/WEST SDV VENT VALVE	U1 RB	565	R2/S	AI	2
10004	07	1-FCV-85-37C	CRD/WEST SDV DRAIN VALVE	U1 RB	565	R2/P	AI	1
10005	07	1-FCV-85-37D	CRD/WEST SDV DRAIN VALVE	U1 RB	565	R2/P	AI	2
10006	07	1-FCV-85-83A	CRD/EAST SDV VENT VALVE	U1 RB	565	R6/S	AI	1
10007	07	1-FCV-85-83	CRD/EAST SDV VENT VALVE	U1 RB	565	R6/S	AI	2
10008	07	1-FCV-85-37E	CRD/EAST SDV DRAIN VALVE	U1 RB	565	R6/P	AI	1
10009	07	1-FCV-85-37F	CRD/EAST SDV DRAIN VALVE	U1 RB	565	R6/P	AI	2
10010	21	1-TNK-85-901	CRD/WEST SCRAM INSTRUMENT VOLUME	U1 RB	565	R2/P	AI	1,2
10011	21	1-TNK-85-902	CRD/EAST SCRAM INSTRUMENT VOLUME	U1 RB	565	R6/P	AI	1,2
10012	08B	1-FSV-85-37A	CRD/SCRAM DUMP VALVE	U1 RB	565	R5/N	AI	1
10013	08B	1-FSV-85-37B	CRD/SCRAM DUMP VALVE	U1 RB	565	R5/N	AI	1
10014	08B	1-FSV-85-35A	CRD/BACKUP SCRAM VALVE	U1 RB	565	R5/N	AI	2
10015	08B	1-FSV-85-35B	CRD/BACKUP SCRAM VALVE	U1 RB	565	R5/N	AI	2
10016	20	1-HS-99-5A/S1A	RPS/REACTOR MANUAL SCRAM CHANNEL A1	U1 CB	617	U1 MCR	AI	1
10017	20	1-HS-99-5A/S1B	RPS/REACTOR MANUAL SCRAM CHANNEL B1	U1 CB	617	U1 MCR	AI	1
10018	20	1-HS-99-5A-S1	RPS/REACTOR MODE SWITCH	U1 CB	617	U1 MCR	AI	2
10019	8B	1-FSV-85-70A	CRD/BACKUP PILOT SCRAM VALVE 'A'	U1 RB	565	R5/N	AI	
10020	8B	1-FSV-85-70B	CRD/ BACKUP PILOT SCRAM VALVE 'B'	U1 RB	565	R5/N	AI	
10023	8B	1-FSV-85-39A	CRD/ISOLATION VALVE	U1 RB	565	CRD RACKS	AI	
10024	8B	1-FSV-85-39B	CRD/ISOLATION VALVE	U1 RB	565	CRD RACKS	AI	
10025	18	1-PI-85-88	PRESSURE INDICATOR	U1 RB	565	R6/S	AI	
10026	18	1-PI-85-89	PRESSURE INDICATOR	U1 RB	565	R2/P	AI	
10027	18	1-PI-85-90	PRESSURE INDICATOR	U1 RB	565	R2/P	AI	
11001	08A	1-FCV-74-1	RHR/PUMP 1A SUCTION VALVE FROM SUPRESSION POOL	U1 RB	519	SW CORNER	AI	1

SSEL NUMBER	CLASS	EQUIPMENT I.D.	DESCRIPTION	BUILDING	ELEV.	ROOM	ISSUE	TRAIN
11002	08A	1-FCV-74-2	RHR/PUMP 1A SUCTION VALVE FROM SHUTDOWN COOLING	U1 RB	541	SW CORNER	AI	1
11004	06	1-PMP-74-5	RHR/PUMP 1A	U1 RB	519	SW CORNER	AI	1
11006	08A	1-FCV-74-7	RHR/PUMP 1A & 1C MINIMUM FLOW VALVE	U1 RB	541	SW CORNER	AI	1
11009	21	1-HEX-74-900A	RHR/HEAT EXCHANGER 1A	U1 RB	565	SW HX	AI	1
11011	08A	1-FCV-74-12	RHR/PUMP 1C SUCTION VALVE FROM SUPRESSION POOL	U1 RB	519	SW CORNER	AI	1A
11012	08A	1-FCV-74-13	RHR/PUMP 1C SUCTION VALVE FROM SHUTDOWN COOLING	U1 RB	541	SW CORNER	AI	1A
11014	06	1-PMP-74-16	RHR/PUMP 1C	U1 RB	519	SW CORNER	AI	1A
11017	21	1-HEX-74-900C	RHR/HEAT EXCHANGER 1C	U1 RB	565	SW HX	AI	1A
11018	20	1-FI-74-50	RHR/LOOP I FLOW INDICATOR	U1 CB	617	U1 MCR	AI	1
11019	20	1-FI-74-56	RHR/LOOP I FLOW INDICATOR	U1 CB	617	U1 MCR	AI	1
11020	08A	1-FCV-74-57	RHR/LOOP I TORUS CONTAINMENT COOLING/SPRAY VALVE	U1 RB	551	TORUS	AI	1
11021	08A	1-FCV-74-59	RHR/LOOP I SUPRESSION POOL COOLING VALVE	U1 RB	551	TORUS	AI	1
11022	08A	1-FCV-74-58	RHR/LOOP I SUPRESSION POOL SPRAY VALVE	U1 RB	551	TORUS	I	1
11023	08A	1-FCV-74-52	RHR/LOOP I OUTBOARD INJECTION VALVE	U1 RB	565	R4/T	AI	1
11024	08A	1-FCV-74-53	RHR/LOOP I INBOARD INJECTION VALVE	U1 RB	565	R4/T	AI	1
11026	08A	1-FCV-78-61	FPC/SPENT FUEL POOL COOLING X-TIE TO RHR	U1 RB	621	R5/S	AI	1
11027	08A	1-FCV-74-60	RHR/LOOP I OUTBOARD DRYWELL SPRAY VALVE	U1 RB	593	R3/S	AI	1
11028	08A	1-FCV-74-61	RHR/LOOP I INBOARD DRYWELL SPRAY VALVE	U1 RB	593	R3/S	I	1
11029	08A	1-FCV-74-24	RHR/PUMP 1B SUCTION VALVE FROM SUPRESSION POOL	U1 RB	519	SE CORNER	AI	2
11030	08A	1-FCV-74-25	RHR/PUMP 1B SUCTION VALVE FROM SHUTDOWN COOLING	U1 RB	541	SE CORNER	AI	2
11031	06	1-PMP-74-28	RHR/PUMP 1B	U1 RB	519	SE CORNER	AI	2
11033	08A	1-FCV-74-30	RHR/PUMP 1B & 1D MINIMUM FLOW VALVE	U1 RB	541	SE CORNER	AI	2
11036	21	1-HEX-74-900B	RHR/HEAT EXCHANGER 1B	U1 RB	565	SE HX	AI	2
11037	08A	1-FCV-74-35	RHR/PUMP 1D SUCTION VALVE FROM SUPRESSION POOL	U1 RB	519	SE CORNER	AI	2A
11038	08A	1-FCV-74-36	RHR/PUMP 1D SUCTION VALVE FROM SHUTDOWN COOLING	U1 RB	541	SE CORNER	AI	2A
11039	06	1-PMP-74-39	RHR/PUMP 1D	U1 RB	519	SE CORNER	AI	2A

SSEL NUMBER	CLASS	EQUIPMENT I.D.	DESCRIPTION	BUILDING	ELEV.	ROOM	ISSUE	TRAIN
11042	21	1-HEX-74-900D	RHR/HEAT EXCHANGER 1D	U1 RB	565	SE HX	AI	2A
11043	20	1-FI-74-64	RHR/LOOP II FLOW INDICATOR	U1 CB	617	U1 MCR	AI	2
11044	20	1-FI-74-70	RHR/LOOP II FLOW INDICATOR	U1 CB	617	U1 MCR	AI	2
11045	08A	1-FCV-74-71	RHR/LOOP II TORUS CONTAINMENT COOLING/SPRAY VALVE	U1 RB	551	TORUS	AI	2
11046	08A	1-FCV-74-73	RHR/LOOP II SUPRESSION POOL COOLING VALVE	U1 RB	551	TORUS	AI	2
11047	08A	1-FCV-74-72	RHR/LOOP II SUPRESSION POOL SPRAY VALVE	U1 RB	551	TORUS	I	2
11048	08A	1-FCV-74-66	RHR/LOOP II OUTBOARD INJECTION VALVE	U1 RB	565	R5/T	AI	2
11049	08A	1-FCV-74-67	RHR/LOOP II INBOARD INJECTION VALVE	U1 RB	565	R5/T	AI	2
11051	08A	1-FCV-74-74	RHR/LOOP II OUTBOARD DRYWELL SPRAY VALVE	U1 RB	565	R5/S	AI	2
11052	08A	1-FCV-74-75	RHR/LOOP II INBOARD DRYWELL SPRAY VALVE	U1 RB	565	R6/S	I	2
11053	08A	1-FCV-74-101	RHR/U2 TO U1 RHR DISCHARGE X-TIE ISO. VALVE (B,D)	U1 RB	565	R6/T	AI	1
12001	07	1-PCV-1-4	MS/MAIN STEAM SAFETY RELIEF VALVE	U1 DW	584	DW	AI	1
12003	07	1-PCV-1-5	MS/MAIN STEAM SAFETY RELIEF VALVE	U1 DW	584	DW	AI	1
12006	07	1-PCV-1-18	MS/MAIN STEAM SAFETY RELIEF VALVE	U1 DW	584	DW	AI	1
12009	07	1-PCV-1-19	MS/MAIN STEAM SAFETY RELIEF VALVE	U1 DW	584	DW	AI	1
12012	07	1-PCV-1-22	MS/MAIN STEAM SAFETY RELIEF VALVE	U1 DW	584	DW	AI	1
12015	07	1-PCV-1-23	MS/MAIN STEAM SAFETY RELIEF VALVE	U1 DW	584	DW	AI	1
12018	07	1-PCV-1-179	MS/MAIN STEAM SAFETY RELIEF VALVE	U1 DW	584	DW	AI	1
12021	07	1-PCV-1-30	MS/MAIN STEAM SAFETY RELIEF VALVE	U1 DW	584	DW	AI	2
12024	07	1-PCV-1-31	MS/MAIN STEAM SAFETY RELIEF VALVE	U1 DW	584	DW	AI	2
12027	07	1-PCV-1-34	MS/MAIN STEAM SAFETY RELIEF VALVE	U1 DW	584	DW	AI	2
12030	07	1-PCV-1-41	MS/MAIN STEAM SAFETY RELIEF VALVE	U1 DW	584	DW	AI	2
12033	07	1-PCV-1-42	MS/MAIN STEAM SAFETY RELIEF VALVE	U1 DW	584	DW	AI	2
12036	07	1-PCV-1-180	MS/MAIN STEAM SAFETY RELIEF VALVE	U1 DW	584	DW	AI	2
13001	07	1-FCV-1-14	MSIV "A" INBOARD ISOLATION VALVE	U1 DW	563	DW	AI	1
13002	07	1-FCV-1-15	MSIV "A" OUTBOARD ISOLATION VALVE	U1 RB	565	MSIV VAULT	AI	2

SSEL NUMBER	CLASS	EQUIPMENT I.D.	DESCRIPTION	BUILDING	ELEV.	ROOM	ISSUE	TRAIN
13003	07	1-FCV-1-26	MSIV "B" INBOARD ISOLATION VALVE	U1 DW	563	DW	AI	1
13004	07	1-FCV-1-27	MSIV "B" OUTBOARD ISOLATION VALVE	U1 RB	565	MSIV VAULT	AI	2
13005	07	1-FCV-1-37	MSIV "C" INBOARD ISOLATION VALVE	U1 DW	563	DW	AI	1
13006	07	1-FCV-1-38	MSIV "C" OUTBOARD ISOLATION VALVE	U1 RB	565	MSIV VAULT	AI	2
13007	07	1-FCV-1-51	MSIV "D" INBOARD ISOLATION VALVE	U1 DW	563	DW	AI	1
13008	07	1-FCV-1-52	MSIV "D" OUTBOARD ISOLATION VALVE	U1 RB	565	MSIV VAULT	AI	2
13009	08A	1-FCV-1-55	MAIN STEAM LINE DRAIN ISOLATION VALVE	U1 DW	563	DW	AI	1
13015	07	1-FCV-64-17	CONTAINMENT VENTILATION ISOLATION VALVE	U1 RB	565	R3/U	I	1,2
13016	07	1-FCV-64-30	CONTAINMENT VENTILATION ISOLATION VALVE	U1 RB	621	R3/Q	I	1,2
13017	07	1-FCV-64-33	CONTAINMENT VENTILATION ISOLATION VALVE	U1 RB	565	R2/P	I	1,2
13018	07	1-FCV-64-139	CONTAINMENT DW DP ISOLATION VALVE	U1 RB	565	R2/P	I	1,2
13019	07	1-FCV-64-140	CONTAINMENT DW DP ISOLATION VALVE	U1 RB	565	R2/P	I	1,2
13020	07	1-FCV-64-28A	SUPPRESSION CHAMBER/DRYWELL VACUUM BREAKERS	U1 DW	<550	IN TORUS	I	1,2
13021	07	1-FCV-64-28B	SUPPRESSION CHAMBER/DRYWELL VACUUM BREAKERS	U1 DW	<550	IN TORUS	I	1,2
13022	07	1-FCV-64-28C	SUPPRESSION CHAMBER/DRYWELL VACUUM BREAKERS	U1 DW	<550	IN TORUS	I	1,2
13023	07	1-FCV-64-28D	SUPPRESSION CHAMBER/DRYWELL VACUUM BREAKERS	U1 DW	<550	IN TORUS	I	1,2
13024	07	1-FCV-64-28E	SUPPRESSION CHAMBER/DRYWELL VACUUM BREAKERS	U1 DW	<550	IN TORUS	I	1,2
13025	07	1-FCV-64-28F	SUPPRESSION CHAMBER/DRYWELL VACUUM BREAKERS	U1 DW	<550	IN TORUS	I	1,2
13026	07	1-FCV-64-28G	SUPPRESSION CHAMBER/DRYWELL VACUUM BREAKERS	U1 DW	<550	IN TORUS	I	1,2
13027	07	1-FCV-64-28H	SUPPRESSION CHAMBER/DRYWELL VACUUM BREAKERS	U1 DW	<550	IN TORUS	I	1,2
13028	07	1-FCV-64-28J	SUPPRESSION CHAMBER/DRYWELL VACUUM BREAKERS	U1 DW	<550	IN TORUS	I	1,2
13029	07	1-FCV-64-28K	SUPPRESSION CHAMBER/DRYWELL VACUUM BREAKERS	U1 DW	<550	IN TORUS	I	1,2
13030	07	1-FCV-64-28L	SUPPRESSION CHAMBER/DRYWELL VACUUM BREAKERS	U1 DW	<550	IN TORUS	I	1,2
13031	07	1-FCV-64-28M	SUPPRESSION CHAMBER/DRYWELL VACUUM BREAKERS	U1 DW	<550	IN TORUS	I	1,2
13032	08A	1-FCV-69-1	RWCU INBOARD ISOLATION VALVE	U1 DW	584	DW	AI	1
13033	08A	1-FCV-69-2	RWCU OUTBOARD ISOLATION VALVE	U1 RB	593	R5/S	AI	2

SSEL NUMBER	CLASS	EQUIPMENT I.D.	DESCRIPTION	BUILDING	ELEV.	ROOM	ISSUE	TRAIN
13035	08A	1-FCV-70-47	RBCCW DRYWELL RETURN VALVE	U1 RB	551	TORUS	I	1,2
13037	08A	1-FCV-71-2	RCIC INBOARD ISOLATION VALVE	U1 DW	584	DW	AI	1
13038	08A	1-FCV-71-3	RCIC OUTBOARD ISOLATION VALVE	U1 RB	565	MSIV VAULT	AI	2
13039	08A	1-FCV-71-18	RCIC OUTBOARD SUCTION VALVE	U1 RB	519	NW CORNER	I	1,2
13040	08A	1-FCV-73-2	HPCI STEAM SUPPLY ISOLATION VALVE	U1 DW	563	DW	AI	1
13041	08A	1-FCV-73-3	HPCI STEAM SUPPLY ISOLATION VALVE	U1 RB	551	TORUS	AI	2
13042	08A	1-FCV-73-81	HPCI STEAM SUPPLY ISOLATION BYPASS VALVE	U1 RB	551	TORUS	AI	2
13043	08A	1-FCV-73-27	HPCI OUTBOARD SUCTION VALVE	U1 RB	519	HPCI ROOM	I	1,2
13044	07	1-FCV-75-57	PSC PUMP SUCTION ISOLATION VALVE	U1 RB	519	NW CORNER	AI	1
13045	07	1-FCV-75-58	PSC PUMP SUCTION ISOLATION VALVE	U1 RB	519	NW CORNER	AI	2
13046	07	1-FCV-76-24	PRIMARY CONTAINMENT ISOLATION VALVE	U1 RB	565	R3/U	I	1,2
13047	07	1-FCV-77-2B	DRYWELL FLOOR DRAIN SUMP DISCHARGE	U1 RB	551	TORUS	I	1,2
13048	07	1-FCV-77-15B	DRYWELL EQUIPMENT DRAIN SUMP DISCHARGE	U1 RB	551	TORUS	I	1,2
13049	07	1-FCV-84-19	CAD ISOLATION VALVE	U1 RB	621	R3/Q	I	1,2
13050	07	1-FCV-84-20	CAD ISOLATION VALVE	U1 RB	621	R3/Q	I	1,2
13051	20	1-LI-3-58AA	RPV LEVEL INSTRUMENTATION	U1 RB	617	U1 MCR	AI	1
13052	20	1-LI-3-58BB	RPV LEVEL INSTRUMENTATION	U1 RB	617	U1 MCR	AI	2
13053	20	1-PI-3-74A	RPV PRESSURE INSTRUMENT	U1 CB	617	U1 MCR	AI	1
13054	20	1-PI-3-74B	RPV PRESSURE INSTRUMENT	U1 CB	617	U1 MCR	AI	2
13055	20	1-XR-64-159	TORUS LEVEL AND DRYWELL PRESSURE INSTRUMENT	U1 CB	617	U1 MCR	AI	1
13056	20	1-LI-64-159A	TORUS LEVEL INSTRUMENT	U1 CB	617	U1 MCR	AI	2
13057	20	1-TI-64-161	TORUS TEMPERATURE INSTRUMENT	U1 CB	617	U1 MCR	AI	1
13058	20	1-TI-64-162	TORUS TEMPERATURE INSTRUMENT	U1 CB	617	U1 MCR	AI	2
13059	20	1-PI-64-67	DRYWELL PRESSURE INSTRUMENT	U1 CB	617	U1 MCR	I	1
13060	20	1-PI-64-160A	DRYWELL PRESSURE INSTRUMENT	U1 CB	617	U1 MCR	I	2
13061	20	1-TI-64-52A	DRYWELL TEMPERATURE INSTRUMENT	U1 CB	617	U1 MCR	I	1

SSEL NUMBER	CLASS	EQUIPMENT I.D.	DESCRIPTION	BUILDING	ELEV.	ROOM	ISSUE	TRAIN
13062	20	1-XR-64-50	DRYWELL TEMPERATURE AND PRESSURE INSTRUMENT	U1 CB	617	U1 MCR	I	2
13063	07	1-FCV-76-17	CONTAINMENT INERTING N2 MAKEUP	U1 RB	565	R5/T	I	1,2
13064	07	1-FCV-64-222	HARDENED WETWELL VENT	U1 RB	565	R3/T	I	1,2
13069	08A	1-FCV-71-40	PRIMARY CONTAINMENT ISOLATION VALVE	U1 RB	565	R4/P	AI	
13074	08A	1-FCV-71-17	RCIC INBOARD SUCTION VALVE	U1 RB	519	NW CORNER	I	1,2
13075	08A	1-FCV-1-56	MAIN STEAM LINE DRAIN ISOLATION VALVE	U1 RB	565	MSIV VAULT	AI	
13076	08A	1-FCV-73-26	HPCI INBOARD SUCTION VALVE	U1 RB	519	SW CORNER	I	1,2
13079	07	1-FCV-32-62	DRYWELL CONTROL AIR SUCTION VALVE	U1 RB	565	CLEAN RM	I	1,2
13080	07	1-FCV-77-2A	DRYWELL FLOOR DRAIN SUMP DISCHARGE	U1 RB	551	TORUS	I	1,2
13081	07	1-FCV-77-15A	DRYWELL EQUIPMENT DRAIN SUMP DISCHARGE	U1 RB	551	TORUS	I	1,2
13082	07	1-FCV-64-18	COOLING/PURGE AIR TO DRYWELL	U1 RB	565	R5/T	I	1,2
13083	07	1-FCV-64-19	COOLING/PURGE AIR TO SUPPRESSION CHAMBER	U1 RB	565	R3/T	I	1,2
13084	07	1-FCV-76-18	CONTAINMENT INERTING DRYWELL N2 MAKEUP VALVE	U1 RB	565	R5/T	I	1,2
13085	07	1-FCV-76-19	CONTAINMENT INERTING - PSC N2 MAKEUP VALVE	U1 RB	565	R3/T	I	1,2
13101	18	1-LT-64-159A	TORUS LEVEL TRANSMITTER	U1 RB	519	TORUS		
13102	18	1-LT-64-159B	TORUS LEVEL TRANSMITTER	U1 RB	519	TORUS		
13111	19	1-TE-64-161A	TORUS TEMPERATURE ELEMENT	U1 RB	519	TORUS		
13112	19	1-TE-64-161B	TORUS TEMPERATURE ELEMENT	U1 RB	519	TORUS		
13113	19	1-TE-64-161C	TORUS TEMPERATURE ELEMENT	U1 RB	519	TORUS		
13114	19	1-TE-64-161D	TORUS TEMPERATURE ELEMENT	U1 RB	519	TORUS		
13115	19	1-TE-64-161E	TORUS TEMPERATURE ELEMENT	U1 RB	519	TORUS		
13116	19	1-TE-64-161F	TORUS TEMPERATURE ELEMENT	U1 RB	519	TORUS		
13117	19	1-TE-64-161G	TORUS TEMPERATURE ELEMENT	U1 RB	519	TORUS		
13118	19	1-TE-64-161H	TORUS TEMPERATURE ELEMENT	U1 RB	519	TORUS		
13211	19	1-TE-64-162A	TORUS TEMPERATURE ELEMENT	U1 RB	519	TORUS		
13212	19	1-TE-64-162B	TORUS TEMPERATURE ELEMENT	U1 RB	519	TORUS		

SSEL NUMBER	CLASS	EQUIPMENT I.D.	DESCRIPTION	BUILDING	ELEV.	ROOM	ISSUE	TRAIN
13213	19	1-TE-64-162C	TORUS TEMPERATURE ELEMENT	U1 RB	519	TORUS		
13214	19	1-TE-64-162D	TORUS TEMPERATURE ELEMENT	U1 RB	519	TORUS		
13215	19	1-TE-64-162E	TORUS TEMPERATURE ELEMENT	U1 RB	519	TORUS		
13216	19	1-TE-64-162F	TORUS TEMPERATURE ELEMENT	U1 RB	519	TORUS		
13217	19	1-TE-64-162G	TORUS TEMPERATURE ELEMENT	U1 RB	519	TORUS		
13218	19	1-TE-64-162H	TORUS TEMPERATURE ELEMENT	U1 RB	519	TORUS		
14001	10	1-CLR-67-917	EECW/RHR PUMP 1A ROOM COOLER	U1 RB	519	SW CORNER	AI	1
14002	10	1-CLR-67-919	EECW/CS PUMP 1A ROOM COOLER	U1 RB	519	NW CORNER	AI	1
14003	10	1-CLR-67-921	EECW/RHR PUMP 1C ROOM COOLER	U1 RB	519	SW CORNER	AI	1
14004	21	1-HEX-67-915	EECW/RHR SEAL HX 1A	U1 RB	519	SW CORNER	AI	1
14013	10	1-CLR-67-918	EECW/RHR PUMP 1B ROOM COOLER	U1 RB	519	SE CORNER	AI	2
14014	10	1-CLR-67-920	EECW/CS PUMP 1B ROOM COOLER	U1 RB	519	NE CORNER	AI	2
14015	10	1-CLR-67-922	EECW/RHR PUMP 1D ROOM COOLER	U1 RB	519	SE CORNER	AI	2
14016	21	1-HEX-67-923	EECW/RHR SEAL HX 1B	U1 RB	519	SE CORNER	AI	2
14025	21	1-HEX-67-916	EECW/RHR SEAL HX 1C	U1 RB	519	SW CORNER	AI	1
14026	21	1-HEX-67-924	EECW/RHR SEAL HX 1D	U1 RB	519	SE CORNER	AI	2
14046	07	1-FCV-67-50	EECW NORTH HEADER BACKUP TO RBCCW	U1 RB	593	R3/P	AI	1
14049	07	0-FCV-67-53	EECW NORTH HEADER BACKUP TO THE AIR COMPRESSORS	U1 RB	565	R3/N	AI	1
14089	07	1-FCV-67-51	EECW SYSTEM SOUTH HEADER BACKUP TO RBCCW	U1 RB	565	R3/T	AI	2
15001	08A	1-FCV-75-2	CS/PUMP 1A SUCTION ISOLATION VALVE	U1 RB	519	NW CORNER	AI	1
15002	06	1-PMP-75-5	CS/PUMP 1A	U1 RB	519	NW CORNER	AI	1
15005	08A	1-FCV-75-9	CS/PUMPS 1A & 1C MINI-FLOW VALVE	U1 RB	541	NW CORNER	AI	1
15006	08A	1-FCV-75-11	CS/PUMP 1C SUCTION ISOLATION VALVE	U1 RB	519	NW CORNER	AI	1
15007	06	1-PMP-75-14	CS/PUMP 1C	U1 RB	519	NW CORNER	AI	1
15010	08A	1-FCV-75-22	CS/PUMPS 1A & 1C TEST ISOLATION VALVE	U1 RB	541	NW CORNER	AI	1
15011	20	1-FI-75-21	CS/PUMPS 1A & 1C FLOW INDICATOR	U1 CB	617	MCR	AI	1

SSEL NUMBER	CLASS	EQUIPMENT I.D.	DESCRIPTION	BUILDING	ELEV.	ROOM	ISSUE	TRAIN
15012	08A	1-FCV-75-23	CS/DIV I OUTBOARD INJECTION VALVE	U1 RB	593	R4/P	AI	1
15013	08A	1-FCV-75-25	CS/DIV I INBOARD INJECTION VALVE	U1 RB	593	R4/P	AI	1
15015	08A	1-FCV-75-30	CS/PUMP 1B SUCTION ISOLATION VALVE	U1 RB	519	NE CORNER	AI	2
15016	06	1-PMP-75-33	CS/PUMP 1B	U1 RB	519	NE CORNER	AI	2
15019	08A	1-FCV-75-37	CS/PUMPS 1B & 1D MINI-FLOW VALVE	U1 RB	541	NE CORNER	AI	2
15020	08A	1-FCV-75-39	CS/PUMP 1D SUCTION ISOLATION VALVE	U1 RB	519	NE CORNER	AI	2
15021	06	1-PMP-75-42	CS/PUMP 1D	U1 RB	519	NE CORNER	AI	2
15024	08A	1-FCV-75-50	CS/PUMPS 1B & 1D TEST ISOLATION VALVE	U1 RB	541	NE CORNER	AI	2
15025	20	1-FI-75-49	CS/PUMPS 1B & 1D FLOW INDICATOR	U1 CB	617	MCR	AI	2
15026	08A	1-FCV-75-51	CS/DIV II OUTBOARD DISCHARGE VALVE	U1 RB	593	R4/P	AI	2
15027	08A	1-FCV-75-53	CS/DIV II INBOARD DISCHARGE VALVE	U1 RB	593	R4/P	AI	2
16001	08B	1-FSV-84-8A	CAD/CAD TO DW (1-FCV-64-18) SOLENOID VALVE	U1 RB	565	R5/T	AI	1
16002	08B	1-FSV-84-8B	CAD/CAD TO DW (1-FCV-64-19) SOLENOID VALVE	U1 RB	565	R3/T	AI	1
16003	07	1-PREG-84-52	CAD/CAD SYSTEM "A" TO UNIT 1 DRYWELL CONTROL AIR	U1 RB	565	R4/U	AI	1
16004	08B	1-FSV-84-48	CAD/CAD SYSTEM "A" TO UNIT 1 DRYWELL CONTROL AIR	U1 RB	565	R3/T	AI	1
16009	21	1-ACC-32-6105	CA/ACCUMULATOR FOR PSV-1-19	U1 DW	584	DW	AI	1
16010	08B	1-PSV-1-19	MS/SOLENOID VALVE FOR PCV-1-19	U1 DW	584	DW	AI	1
16012	21	1-ACC-32-6107	CA/ACCUMULATOR FOR PSV-1-22	U1 DW	584	DW	AI	1
16013	08B	1-PSV-1-22	MS/SOLENOID VALVE FOR PCV-1-22	U1 DW	584	DW	AI	1
16015	21	1-ACC-32-6106	CA/ACCUMULATOR FOR PSV-1-5	U1 DW	584	DW	AI	1
16016	08B	1-PSV-1-5	MS/SOLENOID VALVE FOR PCV-1-5	U1 DW	584	DW	AI	1
16017	08B	1-PSV-1-23	MS/SOLENOID VALVE FOR PCV-1-23	U1 DW	584	DW	AI	1
16018	08B	1-PSV-1-179	MS/SOLENOID VALVE FOR PCV-1-179	U1 DW	584	DW	AI	1
16019	08B	1-PSV-1-4	MS/SOLENOID VALVE FOR PCV-1-4	U1 DW	584	DW	AI	1
16021	08B	1-PSV-1-18	MS/SOLENOID VALVE FOR PCV-1-18	U1 DW	584	DW	AI	2
16023	08B	1-FSV-84-8C	CAD/CAD TO DW (1-FCV-64-19) SOLENOID VALVE	U1 RB	565	R3/T	AI	2

SSEL NUMBER	CLASS	EQUIPMENT I.D.	DESCRIPTION	BUILDING	ELEV.	ROOM	ISSUE	TRAIN
16024	08B	1-FSV-84-8D	CAD/CAD TO DW (1-FCV-64-18) SOLENOID VALVE	U1 RB	565	R5/T	AI	2
16025	07	1-PREG-84-54	CAD/CAD SYSTEM "B" TO UNIT 1 DRYWELL CONTROL AIR	U1 RB	565	R4/U	AI	2
16026	08B	1-FSV-84-49	CAD/CAD SYSTEM "B" TO UNIT 1 DRYWELL CONTROL AIR	U1 RB	565	R3/T	AI	2
16029	21	1-ACC-32-6111	CA/ACCUMULATOR FOR PSV-1-30	U1 DW	584	DW	AI	2
16030	08B	1-PSV-1-30	MS/SOLENOID VALVE FOR PCV-1-30	U1 DW	584	DW	AI	2
16032	21	1-ACC-32-6108	CA/ACCUMULATOR FOR PSV-1-31	U1 DW	584	DW	AI	2
16033	08B	1-PSV-1-31	MS/SOLENOID VALVE FOR PCV-1-31	U1 DW	584	DW	AI	2
16035	21	1-ACC-32-6109	CA/ACCUMULATOR FOR PSV-1-34	U1 DW	584	DW	AI	2
16036	08B	1-PSV-1-34	MS/SOLENOID VALVE FOR PCV-1-34	U1 DW	584	DW	AI	2
16038	08B	1-PSV-1-41	MS/SOLENOID VALVE FOR PCV-1-41	U1 DW	584	DW	AI	2
16040	08B	1-PSV-1-42	MS/SOLENOID VALVE FOR PCV-1-42	U1 DW	584	DW	AI	2
16041	08B	1-PSV-1-180	MS/SOLENOID VALVE FOR PCV-1-180	U1 DW	584	DW	AI	2
16043	07	1-FCV-64-20	CONTAINMENT VENTILATION ISOLATION VALVE	U1 RB	565	R3/T	AI	
16044	07	1-FCV-64-21	CONTAINMENT VENTILATION ISOLATION VALVE	U1 RB	565	R3/T	AI	
18002	08A	1-FCV-23-034	RHR/RHRSW HX A OUTLET VALVE	U1 RB	565	R2/U	AI	1
18004	08A	1-FCV-23-040	RHR/RHRSW HX C OUTLET VALVE	U1 RB	565	R2/U	AI	1
18006	08A	1-FCV-23-046	RHR/RHRSW HX B OUTLET VALVE	U1 RB	565	R5/T	AI	2
18008	08A	1-FCV-23-052	RHR/RHRSW HX D OUTLET VALVE	U1 RB	565	R5/T	AI	2
18009	20	1-FI-23-36	RHRSW HX A FLOW INDICATOR	U1 CB	617	MCR	AI	1
18010	20	1-FI-23-42	RHRSW HX C FLOW INDICATOR	U1 CB	617	MCR	AI	1
18011	20	1-FI-23-48	RHRSW HX B FLOW INDICATOR	U1 CB	617	MCR	AI	2
18012	20	1-FI-23-54	RHRSW HX D FLOW INDICATOR	U1 CB	617	MCR	AI	2
18029	08A	1-FCV-23-046	RHR/RHRSW HX B OUTLET VALVE	U1 RB	565	R5/T	AI	2
18032	08A	1-FCV-23-052	RHR/RHRSW HX D OUTLET VALVE	U1 RB	565	R5/T	AI	2
18033	08A	1-FCV-23-57	RHR/RHRSW CROSS CONNECT VALVE	U1 RB	565	R6/S	AI	2
19001	20	1-PNLA-009-0023/1	ELECTRICAL CONTROL PANEL 1-9-23-1	U1 CB	617	U1 MCR	AI	

SSEL NUMBER	CLASS	EQUIPMENT I.D.	DESCRIPTION	BUILDING	ELEV.	ROOM	ISSUE	TRAIN
19002	20	1-PNLA-009-0023/2	ELECTRICAL CONTROL PANEL 1-9-23-2	U1 CB	617	U1 MCR	AI	
19003	20	1-PNLA-009-0023/3	ELECTRICAL CONTROL PANEL 1-9-23-3	U1 CB	617	U1 MCR	AI	
19004	20	1-PNLA-009-0023/4	ELECTRICAL CONTROL PANEL 1-9-23-4	U1 CB	617	U1 MCR	AI	
19005	20	1-PNLA-009-0023/5	ELECTRICAL CONTROL PANEL 1-9-23-5	U1 CB	617	U1 MCR	AI	
19006	20	1-PNLA-009-0023/6	ELECTRICAL CONTROL PANEL 1-9-23-6	U1 CB	617	U1 MCR	AI	
19007	20	1-PNLA-009-0023/7	ELECTRICAL CONTROL PANEL 1-9-23-7	U1 CB	617	U1 MCR	AI	
19008	20	1-PNLA-009-0023/8	ELECTRICAL CONTROL PANEL 1-9-23-8	U1 CB	617	U1 MCR	AI	
19030	01	1-BDBB-281-0001A	250V DC RMOV BOARD 1A	U1 RB	621	Q/R1	AI	II
19031	01	1-BDBB-281-0001B	250V DC RMOV BOARD 1B	U1 RB	593	Q/R1	AI	I
19033	01	1-BDBB-281-0001C	250V DC RMOV BOARD 1C	U1 RB	565	Q/R1	AI	I
19039	14	1-JBOX-253-6455	I&C BUS 1A DISC SWITCH 1A1	U1 RB	621	R/R1	AI	I
19040	04	1-XFA-253-0001A1	I&C BUS 1A 480/208-120V TRANSFORMER	U1 RB	621	R/R1	AI	I
19041	04	1-XFA-253-0001A	I&C BUS 1A REGULATING TRANSFORMER	U1 RB	621	R/R1	AI	I
19042	14	1-JBOX-253-6457	I&C BUS 1A DISC SW 1A2	U1 CB	593	BATT BD 1	AI	I
19043	14	1-JBOX-253-6456	I&C BUS 1A DISC SWITCHES 1A3, 1A4, 1A5, 1A6	U1 RB	621	R/R1	AI	I
19044	14	1-JBOX-253-8862	I&C BUS 1A DISC SWITCH	U1 RB	621	R/R1	AI	I
19045	20	1-PNLA-009-0009	I&C BUS 1A (CAB 2 OF PNL 1-9-9)	U1 CB	617	U1 MCR	AI	I
19046	20	1-PX-64-160B	POWER SUPPLY (PNL 1-9-19: 1-LI-64-159B,160B)	U1 CB	593	U1 AIR	AI	I
19047	20	1-PXMC-23-114	POWER SUPPLY (PNL 1-9-18: FI-23-36,42 : FI-74-50)	U1 CB	593	U1 AIR	AI	I
19048	20	1-PXMC-23-115 A&B	POWER SUPPLY (PNL 1-9-19: FI-23-48,54; FI-74-64)	U1 CB	593	U1 AIR	AI	II
19049	04	1-XFA-253-0001B1	I&C BUS 1B 480/208-120V TRANSFORMER	U1 RB	593	R/R1	AI	II
19050	04	1-XFA-253-0001B2	I&C BUS 1B REGULATING TRANSFORMER	U1 RB	593	R/R1	AI	II
19051	14	1-JBOX-253-6460	I&C BUS 1B DISC SW 1B2	U1 CB	593	BATT BD 1	AI	II
19052	14	1-JBOX-253-8865	I&C BUS 1B DISC SWITCH	U1 RB	593	R/R1	AI	II
19053	14	1-JBOX-253-6459	I&C BUS 1B DISC SWITCHES 1B3, 1B4, 1B5, 1B6	U1 RB	593	R/R1	AI	II
19054	20	1-PNLA-009-0009	I&C BUS 1B (CAB 3 OF PNL 1-9-9)	U1 CB	617	U1 MCR	AI	II

SSEL NUMBER	CLASS	EQUIPMENT I.D.	DESCRIPTION	BUILDING	ELEV.	ROOM	ISSUE	TRAIN
19055	20	1-PX-64-159B	POWER SUPPLY (PNL 1-9-19)	U1 CB	593	U1 AIR	AI	II
19068	14	1-JBOX-253-7160	I&C BUS 1B DISC SWITCH 1B1	U1 RB	593	R/R1	AI	II
19070	16	1-INVT-256-0001	DIV I ECCS ATU INVERTER	U1 RB	593	Q/R1	AI	I
19071	20	1-PX-71-60-1	ECCS ATU CAB 1-9-81 POWER SUPPLY	U1 CB	593	U1 AIR	AI	I
19072	20	1-PX-71-60-1A	ECCS ATU CAB 1-9-81 POWER SUPPLY	U1 CB	593	U1 AIR	AI	I
19073	20	1-PX-64-50	POWER SUPPLY (PNL 1-25-31: XR-64-50 [DEV BA TERM 11/12])	U1 RB	621	Q/R2	AI	I
19074	20	1-PX-74-56	POWER SUPPLY (PNL 1-9-18: FI-74-56)	U1 CB	593	U1 AIR	AI	I
19075	16	1-INVT-256-0002	DIV II ECCS ATU INVERTER	U1 RB	621	P/R1	AI	II
19076	20	1-PX-71-60-2	ECCS ATU CAB 1-9-82 POWER SUPPLY	U1 CB	593	U1 AIR	AI	II
19077	20	1-PX-71-60-2A	ECCS ATU CAB 1-9-82 POWER SUPPLY	U1 CB	593	U1 AIR	AI	II
19078	20	1-PX-74-70	POWER SUPPLY (PNL 1-9-19: FI-74-70)	U1 CB	593	U1 AIR	AI	II
19079	20	1-PX-64-159A	POWER SUPPLY (1-9-18)	U1 CB	593	U1 AIR	AI	I
19080	20	1-PX-64-160A	POWER SUPPLY (1-9-18)	U1 CB	593	U1 AIR	AI	I
19081	20	1-PX-64-67B	POWER SUPPLY (1-9-19)	U1 CB	593	U1 AIR	AI	II
19082	20	1-PX-64-161	POWER SUPPLY (PNL 9-87)	U1 CB	593	U1 AIR	AI	I
19083	20	1-PX-64-162	POWER SUPPLY (PNL 9-88)	U1 CB	593	U1 AIR	AI	II
19084	18	1-PS-67-50	PRESSURE SWITCH FOR 1-FCV-67-50 (14046)	U1 RB	593	P/R3	AI	
19085	18	1-PS-67-51	PRESSURE SWITCH FOR 1-FCV-67-51 (14047)	U1 RB	565	T/R3	AI	
19088	20	1-LPNL-925-044A/11	COMMON BD LOGIC RELAY PANEL 25-44-A11	U1 RB	621	S/R1	AI	
19089	20	1-LPNL-925-044A/12	COMMON BD LOGIC RELAY PANEL 25-44-A12	U1 RB	621	S/R2	AI	
19090	20	1-LPNL-925-044B/11	COMMON BD LOGIC RELAY PANEL 25-44-B11	U1 RB	621	S/R1	AI	
19091	20	1-LPNL-925-044B/12	COMMON BD LOGIC RELAY PANEL 25-44-B12	U1 RB	621	S/R2	AI	
19114	20	1-PNLA-009-0003A	REACTOR SD & CONT. COOLING PNL	U1 CB	617	U1 MCR	AI	
19115	20	1-PNLA-009-0003B	REACTOR SD & CONT. COOLING PNL	U1 CB	617	U1 MCR	AI	
19116	20	1-PNLA-009-0004	CLEANUP & RECIRC PNL	U1 CB	617	U1 MCR	AI	
19117	20	1-PNLA-009-0005	REACTOR CONTROL PNL	U1 CB	617	U1 MCR	AI	

SSEL NUMBER	CLASS	EQUIPMENT I.D.	DESCRIPTION	BUILDING	ELEV.	ROOM	ISSUE	TRAIN
19118	20	1-PNLA-009-0006	FW & COND. PNL	U1 CB	617	U1 MCR	AI	
19120	20	1-PNLA-009-0015	RPS CH A (DIV I)	U1 CB	593	U1 AIR	AI	I
19121	20	1-PNLA-009-0016	RPS CH A, B, C, D	U1 CB	593	U1 AIR	AI	
19122	20	1-PNLA-009-0017	RPS CH B (DIV II)	U1 CB	593	U1 AIR	AI	II
19123	20	1-PNLA-009-0018	FW & RECIRC PNL	U1 CB	593	U1 AIR	AI	
19124	20	1-PNLA-009-0019	PROCESS INSTR PNL	U1 CB	593	U1 AIR	AI	
19125	20	1-PNLA-009-0021	TEMP RECORDING PNL	U1 CB	617	U1 MCR	AI	
19126	20	1-PNLA-009-0028	CRD SELECT RELAY AUX PNL	U1 CB	593	U1 AIR	AI	
19127	20	1-PNLA-009-0030	AUTO BLOWNDOWN AUX PNL	U1 CB	593	U1 AIR	AI	
19128	20	1-PNLA-009-0032	RHR, CS, & HPCI (CH A) PNL	U1 RB	593	U1 AIR	AI	
19129	20	1-PNLA-009-0033	RHR, CS, & HPCI (CH B) PNL	U1 CB	593	U1 AIR	AI	
19130	20	1-PNLA-009-0039	HPCI RELAY AUX PNL	U1 CB	593	U1 AIR	AI	
19131	20	1-PNLA-009-0042	MSIV (INBOARD) DIV II PNL	U1 CB	593	U1 AIR	AI	
19132	20	1-PNLA-009-0043	MSIV (OUTBOARD) DIV II PNL	U1 CB	593	U1 AIR	AI	
19133	20	1-PNLA-009-0054	CONTAINMENT ATM. DILUTION PNL	U1 CB	617	U1 MCR	AI	
19134	20	1-PNLA-009-0055	CONTAINMENT ATM. DILUTION PNL	U1 CB	617	U1 MCR	AI	
19135	20	1-PNLA-009-0081	DIV I ECCS ATU CABINET	U1 CB	593	U1 AIR	AI	
19136	20	1-PNLA-009-0082	DIV II ECCS ATU CABINET	U1 CB	593	U1 AIR	AI	II
19137	20	1-PNLA-009-0083	RPS ATU CAB	U1 CB	593	U1 AIR	AI	I
19138	20	1-PNLA-009-0084	RPS ATU CAB	U1 CB	593	U1 AIR	AI	I
19139	20	1-PNLA-009-0085	RPS ATU CAB	U1 CB	593	U1 AIR	AI	II
19140	20	1-PNLA-009-0086	RPS ATU CAB	U1 CB	593	U1 AIR	AI	II
19141	20	1-PNLA-009-0087	DIV I TORUS TEMP MONITORING	U1 CB	593	U1 AIR	AI	I
19142	20	1-PNLA-009-0088	DIV II TORUS TEMP MONITORING	U1 CB	593	U1 AIR	AI	II
19145	20	1-PNLA-009-0093	NEW PNL (INSTALLED BY DCN W19433)	U1 CB	593	U1 AIR	AI	
19146	18	1-HS-74-7B	LOCAL HS STATION	U1 RB	541	SW CORNER	AI	I

SSEL NUMBER	CLASS	EQUIPMENT I.D.	DESCRIPTION	BUILDING	ELEV.	ROOM	ISSUE	TRAIN
19147	14	1-JB-668	JUNCTION BOX (TERM BLOCK) - SEALED BOX	U1 RB	541	SW CORNER	AI	I
19148	18	1-HS-74-57B	LOCAL HS STATION	U1 RB	551	TORUS	AI	I
19149	14	1-JB-654	JUNCTION BOX (TERM BLOCK) - SEALED BOX	U1 RB	551	TORUS	AI	I
19150	18	1-HS-74-59B	LOCAL HS STATION	U1 RB	551	TORUS	AI	I
19151	18	1-HS-74-58B	LOCAL HS STATION	U1 RB	551	TORUS	I	I
19153	18	1-HS-74-52B	LOCAL HS STATION	U1 RB	565	T/R3	AI	II
19154	14	1-JB-1079	JUNCTION BOX (TERM BLOCK) - SEALED BOX	U1 RB	583	T/R3	AI	II
19155	18	1-HS-74-53B	LOCAL HS STATION	U1 RB	565	T/R3	AI	I
19156	18	1-HS-74-60B	LOCAL HS STATION	U1 RB	593	S/R3	AI	I
19158	18	1-HS-74-61B	LOCAL HS STATION	U1 RB	593	S/R3	I	I
19160	18	1-HS-74-30B	LOCAL HS STATION	U1 RB	541	SE CORNER	AI	II
19162	18	1-HS-74-71B	LOCAL HS STATION	U1 RB	551	TORUS	AI	II
19163	14	1-JB-665	JUNCTION BOX (TERM BLOCK) - SEALED BOX	U1 RB	551	TORUS	AI	II
19164	18	1-HS-74-72B	LOCAL HS STATION	U1 RB	551	TORUS	I	II
19165	18	1-HS-74-66B	LOCAL HS STATION	U1 RB	583	T/R4	AI	II
19166	14	1-JB-1080	JUNCTION BOX (TERM BLOCK) - SEALED BOX	U1 RB	583	T/R4	AI	II
19167	18	1-HS-74-67B	LOCAL HS STATION	U1 RB	583	T/R4	AI	II
19170	18	1-HS-74-75B	LOCAL HS STATION	U1 RB	565	S/R6	I	II
19171	18	1-HS-70-47B	LOCAL HS STATION	U1 RB	551	TORUS	I	II
19172	14	1-JB-1204	JUNCTION BOX (TERM BLOCK) - SEALED BOX	U1 RB	551	TORUS	AI	II
19173	18	1-HS-75-09B	LOCAL HS STATION	U1 RB	519	NW CORNER	AI	I
19175	18	1-HS-75-25B	LOCAL HS STATION	U1 RB	593	P/R4	AI	I
19176	14	1-JBOX-1064	JUNCTION BOX (TERM BLOCK) - SEALED BOX	U1 RB	593	P/R4	AI	I
19177	18	1-HS-75-37B	LOCAL HS STATION	U1 RB	519	NE CORNER	AI	II
19179	18	1-HS-75-53B	LOCAL HS STATION (TERM BLOCK) - SEALED BOX	U1 RB	593	P/R4	AI	II
19180	14	1-JBOX-1067	LOCAL HS STATION (TERM BLOCK) - SEALED BOX	U1 RB	593	P/R4	AI	II

SSEL NUMBER	CLASS	EQUIPMENT I.D.	DESCRIPTION	BUILDING	ELEV.	ROOM	ISSUE	TRAIN
19181	18	1-HS-23-34B	LOCAL HS STATION (TERM BLOCK) - SEALED BOX	U1 RB	565	U/R2	AI	I
19182	14	1-JBOX-1077	JUNCTION BOX (TERM BLOCK) - SEALED BOX	U1 RB	565	U/R2	AI	I
19183	18	1-HS-23-40B	LOCAL HS STATION (TERM BLOCK) - SEALED BOX	U1 RB	565	U/R2	AI	I
19184	18	1-HS-23-46B	LOCAL HS STATION	U1 RB	565	T/R4	AI	II
19185	14	1-JB-1087	JUNCTION BOX (TERM BLOCK) - SEALED BOX	U1 RB	565	T/R4	AI	II
19186	18	1-HS-23-52B	LOCAL HS STATION	U1 RB	565	T/R4	AI	II
19187	18	1-HS-74-0005B	LOCAL HS STATION - RHR PUMP 1A	U1 RB	519	SW CORNER	AI	I
19188	18	1-HS-74-0028B	LOCAL HS STATION - RHR PUMP 1B	U1 RB	519	SE CORNER	AI	II
19189	18	1-HS-74-0016B	LOCAL HS STATION - RHR PUMP 1C	U1 RB	519	SW CORNER	AI	I
19190	18	1-HS-74-0039B	LOCAL HS STATION - RHR PUMP 1D	U1 RB	519	SE CORNER	AI	II
19191	18	1-HS-75-0005B	LOCAL HS STATION - CS PUMP 1A	U1 RB	519	NW CORNER	AI	I
19192	18	1-HS-75-0033B	LOCAL HS STATION - CS PUMP 1B	U1 RB	519	NE CORNER	AI	II
19193	18	1-HS-75-0014B	LOCAL HS STATION - CS PUMP 1C	U1 RB	519	NW CORNER	AI	I
19194	18	1-HS-75-0042B	LOCAL HS STATION - CS PUMP 1D	U1 RB	519	NE CORNER	AI	II
19195	18	1-LPNL-925-005A	LOCAL PANEL 25-5A	U1 RB	593	S/R3	AI	
19196	18	1-LPNL-925-005B	LOCAL PANEL 25-5B	U1 RB	593	S/R3	AI	
19197	18	1-LPNL-925-005D	LOCAL PANEL 25-5-001	U1 RB	593	S/R3	AI	
19198	18	1-LPNL-925-006A	LOCAL PANEL 25-6A	U1 RB	593	P/R5	AI	
19199	18	1-LPNL-925-006D	LOCAL PANEL 25-6-001	U1 RB	593	Q/R5	AI	
19200	18	1-LPNL-925-0059	LOCAL PANEL 25-59	U1 RB	519	SW CORNER	AI	
19201	18	1-LPNL-925-0062	LOCAL PANEL 25-62	U1 RB	519	SE CORNER	AI	
19204	20	1-PNLA-925-0031	LOCAL PANEL 25-31	U1 RB	621	Q/R2	AI	
19205	20	1-PNLA-925-0032	LOCAL PANEL 25-32	U1 RB	621	Q/R2	AI	
19206	18	1-LPNL-925-0001	LOCAL PANEL 25-1	U1 RB	519	NW CORNER	AI	
19207	18	1-LPNL-925-0060	LOCAL PANEL 25-60	U1 RB	519	N/E CORNER	AI	
19220	20	1-PROT-099-0001A1	RPS CIRCUIT PROTECTOR CABINET 1A1	U1 RB	593	BATT BD 1	AI	

SSEL NUMBER	CLASS	EQUIPMENT I.D.	DESCRIPTION	BUILDING	ELEV.	ROOM	ISSUE	TRAIN
19221	20	1-PROT-099-0001A2	RPS CIRCUIT PROTECTOR CABINET 1A2	U1 RB	593	BATT BD 1	AI	
19222	20	1-PROT-099-0001B1	RPS CIRCUIT PROTECTOR CABINET 1B1	U1 RB	593	BATT BD 1	AI	
19223	20	1-PROT-099-0001B2	RPS CIRCUIT PROTECTOR CABINET 1B2	U1 RB	593	BATT BD 1	AI	
19224	20	1-PROT-099-0001C1	RPS CIRCUIT PROTECTOR CABINET 1C1	U1 RB	593	BATT BD 1	AI	
19225	20	1-PROT-099-0001C2	RPS CIRCUIT PROTECTOR CABINET 1C2	U1 RB	593	BATT BD 1	AI	
19226	18	1-LPNL-925-247A	LOCAL PANEL 1-25-247A (CAD DRYWELL & SUPP. CHAMB. V.)	U1 RB	621	Q/R4	AI	
19227	01	1-BDBB-265-0001B	480V RB VENT BD 1B	U1 RB	565	U/R4	AI	
19228	20	1-PNLA-009-0036A	PANEL 1-9-36A	U1 CB	593	U1 AIR	AI	
19229	18	1-LPNL-925-0247B	LOCAL PANEL 1-25-247B (CAD N2 SUPPLY PANEL B)	U1 RB	621	Q/R4	AI	
19230	18	1-LPNL-925-0007A	LOCAL PANEL 1-25-7A	U1 RB	541	SW CORNER	AI	
19231	18	1-LPNL-925-0007B	LOCAL PANEL 1-25-7B	U1 RB	541	SW CORNER	AI	
19232	14	1-HS-74-101B	HANDSWITCH FOR 1-FCV-74-101 (11055)	U1 RB	565	T/R6	AI	
19239	18	1-TS-64-68	HANDSWITCH FOR 1-CLR-67-917 (14001)	U1 RB	541	SW CORNER	AI	
19240	18	1-HS-64-69	HANDSWITCH FOR 1-CLR-67-918 (14013)	U1 RB	541	SE CORNER	AI	
19241	18	1-TS-64-70	HANDSWITCH FOR 1-CLR-67-921 (14003)	U1 RB	541	SW CORNER	AI	
19242	18	1-HS-64-71	HANDSWITCH FOR 1-CLR-67-922 (14015)	U1 RB	541	SE CORNER	AI	
19243	14	1-HS-69-2B	HANDSWITCH FOR 1-FCV-69-2 (13033)	U1 RB	593	R5/S	AI	
19244	14	1-HS-71-18B	HANDSWITCH FOR 1-FCV-71-18 (13039)	U1 RB	519	NW CORNER	I	
19245	01	1-HS-71-2B	HANDSWITCH FOR 1-FCV-71-2 (13037)	U1 RB	593	R/R1	AI	
19246	14	1-HS-73-27	HANDSWITCH FOR 1-FCV-73-27 (13043)	U1 RB	519	HPCI	I	
19247	18	1-HS-73-3B	HANDSWITCH FOR 1-FCV-73-3 (13041)	U1 RB	551	TORUS	AI	
19248	18	1-HS-73-81B	HANDSWITCH FOR 1-FCV-73-81 (13042)	U1 RB	551	TORUS	AI	
19250	14	1-HS-74-12B	HANDSWITCH FOR 1-FCV-74-12 (11011)	U1 RB	519	SW CORNER	AI	
19251	14	1-HS-74-13B	HANDSWITCH FOR 1-FCV-74-13 (11012)	U1 RB	541	SW CORNER	AI	
19252	14	1-HS-74-1B	HANDSWITCH FOR 1-FCV-74-1 (11001)	U1 RB	519	SW CORNER	AI	
19253	14	1-HS-74-24B	HANDSWITCH FOR 1-FCV-74-24 (11029)	U1 RB	519	SE CORNER	AI	

SSEL NUMBER	CLASS	EQUIPMENT I.D.	DESCRIPTION	BUILDING	ELEV.	ROOM	ISSUE	TRAIN
19254	14	1-HS-74-25B	HANDSWITCH FOR 1-FCV-74-25 (11030)	U1 RB	541	SE CORNER	AI	
19255	14	1-HS-74-2B	HANDSWITCH FOR 1-FCV-74-2 (11002)	U1 RB	541	SW CORNER	AI	
19256	14	1-HS-74-35B	HANDSWITCH FOR 1-FCV-74-35 (11037)	U1 RB	519	SE CORNER	AI	
19257	14	1-HS-74-36B	HANDSWITCH FOR 1-FCV-74-36 (11038)	U1 RB	541	SE CORNER	AI	
19258	18	1-HS-74-73B	HANDSWITCH FOR 1-FCV-74-73 (11046)	U1 RB	551	TORUS	AI	
19260	18	1-HS-75-11B	HANDSWITCH FOR 1-FCV-75-11 (15006)	U1 RB	519	NW CORNER	AI	
19261	14	1-HS-75-22B	HANDSWITCH FOR 1-FCV-75-22 (15010)	U1 RB	541	NW CORNER	AI	
19262	18	1-HS-75-23B	HANDSWITCH FOR 1-FCV-75-23 (15012)	U1 RB	593	P/R4	AI	
19263	18	1-HS-75-2B	HANDSWITCH FOR 1-FCV-75-2 (15001)	U1 RB	519	NW CORNER	AI	
19264	18	1-HS-75-30B	HANDSWITCH FOR 1-FCV-75-30 (15015)	U1 RB	519	NE CORNER	AI	
19265	18	1-HS-75-39B	HANDSWITCH FOR 1-FCV-75-39 (15020)	U1 RB	519	NE CORNER	AI	
19266	14	1-HS-75-50B	HANDSWITCH FOR 1-FCV-75-50 (15024)	U1 RB	541	NE CORNER	AI	
19267	18	1-HS-75-51B	HANDSWITCH FOR 1-FCV-75-51 (15026)	U1 RB	593	P/R4	AI	
19268	14	1-HS-78-61B	HANDSWITCH FOR 1-FCV-78-61 (11026)	U1 RB	621	R5/S	AI	
19269	14	1-HS-64-72	HANDSWITCH FOR 1-CLR-67-919 (14002)	U1 RB	541	NW CORNER	AI	
19270	14	1-HS-64-73	HANDSWITCH FOR 1-CLR-67-920 (14014)	U1 RB	541	NE CORNER	AI	
19271	18	1-TS-64-68	TEMPERATURE SWITCH FOR 1-CLR-67-917 (14001)	U1 RB	519	SW CORNER	AI	
19272	18	1-TS-64-69	TEMPERATURE SWITCH FOR 1-CLR-67-918 (14013)	U1 RB	519	SE CORNER	AI	
19273	18	1-TS-64-70	TEMPERATURE SWITCH FOR 1-CLR-67-921 (14003)	U1 RB	519	SW CORNER	AI	
19274	18	1-TS-64-71	TEMPERATURE SWITCH FOR 1-CLR-67-922 (14015)	U1 RB	519	SE CORNER	AI	
19275	18	1-TS-1-17A	MAIN STEAM VAULT TEMPERATURE SWITCH	U1 RB	565	MSVLT N/T3	AI	
19276	18	1-TS-1-17B	MAIN STEAM VAULT TEMPERATURE SWITCH	U1 RB	565	MSVLT N/T3	AI	
19277	18	1-TS-1-17C	MAIN STEAM VAULT TEMPERATURE SWITCH	U1 RB	565	MSVLT N/T3	AI	
19278	18	1-TS-1-17D	MAIN STEAM VAULT TEMPERATURE SWITCH	U1 RB	565	MSVLT N/T3	AI	
19279	18	1-TS-1-29A	MAIN STEAM TUNNEL TEMPERATURE SWITCH	U1 TB	565	MSTNL K/T3	AI	
19280	18	1-TS-1-29B	MAIN STEAM TUNNEL TEMPERATURE SWITCH	U1 TB	565	MSTNL K/T3	AI	

SSEL NUMBER	CLASS	EQUIPMENT I.D.	DESCRIPTION	BUILDING	ELEV.	ROOM	ISSUE	TRAIN
19281	18	1-TS-1-29C	MAIN STEAM TUNNEL TEMPERATURE SWITCH	U1 TB	565	MSTNL K/T3	AI	
19282	18	1-TS-1-29D	MAIN STEAM TUNNEL TEMPERATURE SWITCH	U1 TB	565	MSTNL K/T3	AI	
19283	18	1-TS-1-40A	MAIN STEAM TUNNEL TEMPERATURE SWITCH	U1 TB	586	MSTNL K/T3	AI	
19284	18	1-TS-1-40B	MAIN STEAM TUNNEL TEMPERATURE SWITCH	U1 TB	586	MSTNL K/T3	AI	
19285	18	1-TS-1-40C	MAIN STEAM TUNNEL TEMPERATURE SWITCH	U1 TB	586	MSTNL K/T3	AI	
19286	18	1-TS-1-40D	MAIN STEAM TUNNEL TEMPERATURE SWITCH	U1 TB	586	MSTNL K/T3	AI	
19287	18	1-TS-1-54A	MAIN STEAM TUNNEL TEMPERATURE SWITCH	U1 TB	586	MSTNL K/T3	AI	
19288	18	1-TS-1-54B	MAIN STEAM TUNNEL TEMPERATURE SWITCH	U1 TB	586	MSTNL K/T3	AI	
19289	18	1-TS-1-54C	MAIN STEAM TUNNEL TEMPERATURE SWITCH	U1 TB	586	MSTNL K/T3	AI	
19290	18	1-TS-1-54D	MAIN STEAM TUNNEL TEMPERATURE SWITCH	U1 TB	586	MSTNL K/T3	AI	
19291	18	1-TS-64-72	TEMPERATURE SWITCH FOR 1-CLR-67-919 (14002)	U1 RB	519	NW CORNER	AI	
19292	18	1-TS-64-73	TEMPERATURE SWITCH FOR 1-CLR-67-920 (14014)	U1 RB	519	NE CORNER	AI	
19294	00	1-AMP-092-0007/41A	IRM CH. "A" VOLTAGE PREAMPLIFIER 7-34A	RB	565	S/R3	AI	
19295	00	1-AMP-092-0007/41B	IRM CH. "B" VOLTAGE PREAMPLIFIER 7-34B	RB	565	S/R3	AI	
19296	14	1-LPNL-925-0027	PANEL 1-25-27 IRM PREAMP. RPS I	RB	565	S/R3	AI	
19297	00	1-AMP-092-0007/41C	IRM CH. "C" VOLTAGE PREAMPLIFIER 7-34C	RB	577	Q/R5	AI	
19298	00	1-AMP-092-0007/41D	IRM CH. "D" VOLTAGE PREAMPLIFIER 7-34D	RB	577	Q/R5	AI	
19299	14	1-LPNL-925-0061	PANEL 1-25-61 IRM PREAMP. RPS II	RB	577	Q/R5	AI	
19300	20	1-NM-92-7/41A	CHANNEL "A" IRM INDICATOR	CB	617	U1 MCR	AI	
19301	20	1-NM-92-7/41B	CHANNEL "B" IRM INDICATOR	CB	617	U1 MCR	AI	
19302	20	1-NM-92-7/41C	CHANNEL "C" IRM INDICATOR	CB	617	U1 MCR	AI	
19303	20	1-NM-92-7/41D	CHANNEL "D" IRM INDICATOR	CB	617	U1 MCR	AI	
19304	20	1-PNLA-009-012	PANEL 1-9-12	CB	617	U1 MCR	AI	
19305	15	1-BATD-283-000A1	24V NEUTRON MONITORING BATTERY, U1 CHANNEL A	CB	593	BAT RM 1	AI	
19306	15	1-BATD-283-000B1	24V NEUTRON MONITORING BATTERY, U1 CHANNEL B	CB	593	BAT RM 1	AI	
19307	16	1-CHGD-283-A1-1	24V NEUTRON BATTERY CHARGERS A1-1	CB	593	BAT BD RM 1	AI	

SSEL NUMBER	CLASS	EQUIPMENT I.D.	DESCRIPTION	BUILDING	ELEV.	ROOM	ISSUE	TRAIN
19308	16	1-CHGD-283-A2-1	24V NEUTRON BATTERY CHARGERS A2-1	CB	593	BAT BD RM 1	AI	
19309	16	1-CHGD-283-B1-1	24V NEUTRON BATTERY CHARGERS B1-1	CB	593	BAT BD RM 1	AI	
19310	16	1-CHGD-283-B2-1	24V NEUTRON BATTERY CHARGERS B2-1	CB	593	BAT BD RM 1	AI	
19312	14	1-HS-71-17B	HANDSWITCH FOR 1-FCV-71-17 (13074)	U1 RB	519	NW CORNER	AI	
19313	20	1-HS-1-56A	HANDSWITCH FOR 1-FCV-1-56 (13075)	U1 CB	617	U1 MCR	AI	
19314	14	1-HS-73-26B	HANDSWITCH FOR 1-FCV-73-26 (13076)	U1 RB	519	SW CORNER	AI	
19316	20	1-HS-77-2A	HANDSWITCH FOR 1-FCV-77-2A (13080)	U1 CB	617	U1 MCR	AI	
19317	20	1-HS-77-15A	HANDSWITCH FOR 1-FCV-77-15A (13081)	U1 CB	617	U1 MCR	AI	
19318	20	1-HS-64-18	HANDSWITCH FOR 1-FCV-64-18 (13082)	U1 CB	617	U1 MCR	AI	
19319	20	1-HS-64-19	HANDSWITCH FOR 1-FCV-64-19 (13083)	U1 CB	617	U1 MCR	AI	
19320	20	1-HS-76-18	HANDSWITCH FOR 1-FCV-76-18 (13084)	U1 CB	617	U1 MCR	AI	
19321	20	1-HS-76-19	HANDSWITCH FOR 1-FCV-76-19 (13085)	U1 CB	617	U1 MCR	AI	
19323	14	1-JB-0375	JUNCTION BOX (TERM BLOCK) - SEALED BOX	U1 RB	519	SE CORNER	AI	
19324	14	1-JB-0662	JUNCTION BOX (TERM BLOCK) - SEALED BOX	U1 RB	541	SE CORNER	AI	
19325	14	1-JB-0658	JUNCTION BOX (TERM BLOCK) - SEALED BOX	U1 RB	541	SE CORNER	AI	
19326	14	1-JB-1032	JUNCTION BOX (TERM BLOCK) - SEALED BOX	U1 RB	593	S/R3	AI	
19327	14	1-JB-1095	JUNCTION BOX (TERM BLOCK) - SEALED BOX	U1 RB	565	S/R6	AI	
19328	14	1-JB-1559	JUNCTION BOX (TERM BLOCK) - SEALED BOX	U1 RB	565	T/R6	AI	
19329	14	1-JB-0670	JUNCTION BOX (TERM BLOCK) - SEALED BOX	U1 RB	541	NE CORNER	AI	
19330	14	1-JB-0791	JUNCTION BOX (TERM BLOCK) - SEALED BOX	U1 RB	519	NW CORNER	AI	
19331	14	1-JB-0681	JUNCTION BOX (TERM BLOCK) - SEALED BOX	U1 RB	541	NW CORNER	AI	
19332	14	1-CS-75-9B	CONTROL STATION FOR 1-HS-75-9B	U1 RB	519	NW CORNER	AI	
19333	14	1-CS-75-37B	CONTROL STATION FOR 1-HS-75-37B	U1 RB	519	NE CORNER	AI	
19334	14	1-JB-1231	JUNCTION BOX (TERM BLOCK) - SEALED BOX	U1 RB	565	S/R6	AI	
19351	14	1-JB-3828	JUNCTION BOX FOR 1-TS-1-29A	U1 RB	565	MSTNL K/T3	AI	
19352	14	1-JB-3829	JUNCTION BOX FOR 1-TS-1-29B	U1 RB	565	MSTNL K/T3	AI	

SSEL NUMBER	CLASS	EQUIPMENT I.D.	DESCRIPTION	BUILDING	ELEV.	ROOM	ISSUE	TRAIN
19353	14	1-JB-3830	JUNCTION BOX FOR 1-TS-1-29C	U1 RB	565	MSTNL K/T3	AI	
19354	14	1-JB-3831	JUNCTION BOX FOR 1-TS-1-29D	U1 RB	565	MSTNL K/T3	AI	
19355	14	1-JB-3801	JUNCTION BOX FOR 1-TS-1-40A	U1 RB	586	MSTNL K/T3	AI	
19356	14	1-JB-3802	JUNCTION BOX FOR 1-TS-1-40B	U1 RB	586	MSTNL K/T3	AI	
19357	14	1-JB-3803	JUNCTION BOX FOR 1-TS-1-40C	U1 RB	586	MSTNL K/T3	AI	
19358	14	1-JB-3804	JUNCTION BOX FOR 1-TS-1-40D	U1 RB	586	MSTNL K/T3	AI	
19359	14	1-JB-3813	JUNCTION BOX FOR 1-TS-1-54A	U1 RB	586	MSTNL K/T3	AI	
19360	14	1-JB-3814	JUNCTION BOX FOR 1-TS-1-54B	U1 RB	586	MSTNL K/T3	AI	
19361	14	1-JB-3815	JUNCTION BOX FOR 1-TS-1-54C	U1 RB	586	MSTNL K/T3	AI	
19362	14	1-JB-3816	JUNCTION BOX FOR 1-TS-1-54D	U1 RB	586	MSTNL K/T3	AI	
19412	03	0-BDAA-211-0000A	4KV SHDN BD A	U1 RB	621	R/R2	AI	IA
19413	03	0-BDAA-211-0000B	4KV SHDN BD B	U1 RB	593	Q/R2	AI	IB
19418	02	1-BDBB-231-0001A	480V SHDN BD 1A	U1 RB	621	S/R1	AI	I
19419	02	1-BDBB-231-0001B	480V SHDN BD 1B	U1 RB	621	S/R2	AI	II
19423	01	1-BDBB-268-0001A	480V RMOV BD 1A	U1 RB	621	R/R1	AI	I
19424	01	1-BDBB-268-0001B	480V RMOV BD 1B	U1 RB	593	R/R1	AI	II
19437	14	0-BDDD-280-0001	250V BATTERY BD 1	U1 RB	593	P/R4	AI	
19492	20	0-LPNL-925-0045A	PANEL 25-45A	U1 RB	621	R/R2	AI	
19493	20	0-LPNL-925-0045B	PANEL 25-45B	U1 RB	593	R/R2	AI	
19516	14	0-XSW-248-0001	250V MAIN BATT CHGR OUTPUT XFR SW 1	U1 RB	593	P/R4	AI	
19519	15	0-BATA-248-0000A	250V BATTERY SB-A	U1 RB	621	S/R2	AI	IA
19520	14	0-PNLA-248-A	250V DISTRIBUTION PANEL SB-A	U1 RB	621	S/R2	AI	IA
19521	16	0-CHGA-248-0000A	250V BATTERY CHARGER SB-A	U1 RB	621	S/R2	AI	IA
19522	15	0-BATA-248-0000B	250V BATTERY SB-B	U1 RB	621	R/R2	AI	IB
19523	14	0-PNLA-248-B	250V DISTRIBUTION PANEL SB-B	U1 RB	621	R/R2	AI	IB
19524	16	0-CHGA-248-0000B	250V BATTERY CHARGER SB-B	U1 RB	621	R/R2	AI	IB

SSEL NUMBER	CLASS	EQUIPMENT I.D.	DESCRIPTION	BUILDING	ELEV.	ROOM	ISSUE	TRAIN
19534	16	0-CHGA-248-0001	250V BATTERY CHARGER 1	U1 RB	593	P/R4	AI	
19537	15	0-BATA-248-0001	250V MAIN BATTERY 1	U1 RB	593	P/R4	AI	
19594	04	1-XFA-231-TS1A	4KV/480V TRANSFORMER TS1A	U1 RB	621	T/R1	AI	I
19595	04	1-XFA-231-TS1B	4KV/480V TRANSFORMER TS1B	U1 RB	621	S/R7	AI	II
19709	20	1-PNLA-009-0020	PANEL 1-9-20	U1 CB	617	U1 MCR	AI	
19716	18	1-LPNL-925-0223	LOCAL PANEL 1-25-223 – RAW COOLING WATER PANEL	U1 RB	593	Q/R2	AI	
19729	14	1-HS-23-57B	HANDSWITCH FOR 1-FCV-23-57	U1 RB	565	R6/S	AI	
19791	20	1-PNLA-009-0008	PANEL 1-9-8	U1 CB	617	U1 MCR	AI	
19792	18	1-LPNL-925-006B	LOCAL PANEL 25-6B	U1 RB	593	R5/P	AI	
19794	04	1-XFA-099-0010	RPS REGULATING TRANSFORMER TRP-1	U1 CB	593	BATT BD 1	AI	
19795	14	1-FUDS-099-0001CA	RPS REG XFMR DISC SW FROM 480 V RMOV BD 1B	U1 CB	593	BATT BD 1	AI	
19796	04	TUP1	UNIT PREFERRED XFMR	U1 CB	593	BATT BD 1	AI	
19797	14	1-FUDS-099-0001CB	RPS BUS XFMR DISC SW	U1 CB	593	BATT BD 1	AI	
19801	14	1-JB-6439	JUNCTION BOX SERVING 1-TE-161A	U1 RB	519	TORUS		
19802	14	1-JB-6440	JUNCTION BOX SERVING 1-TE-161B	U1 RB	519	TORUS		
19803	14	1-JB-6441	JUNCTION BOX SERVING 1-TE-161C	U1 RB	519	TORUS		
19804	14	1-JB-6442	JUNCTION BOX SERVING 1-TE-161D	U1 RB	519	TORUS		
19805	14	1-JB-6443	JUNCTION BOX SERVING 1-TE-161E	U1 RB	519	TORUS		
19806	14	1-JB-6444	JUNCTION BOX SERVING 1-TE-161F	U1 RB	519	TORUS		
19807	14	1-JB-6445	JUNCTION BOX SERVING 1-TE-161G	U1 RB	519	TORUS		
19808	14	1-JB-6446	JUNCTION BOX SERVING 1-TE-161H	U1 RB	519	TORUS		
19809	14	1-JB-6453	JUNCTION BOX SERVING 1-TE-162A	U1 RB	519	TORUS		
19810	14	1-JB-6454	JUNCTION BOX SERVING 1-TE-162B	U1 RB	519	TORUS		
19811	14	1-JB-6447	JUNCTION BOX SERVING 1-TE-162C	U1 RB	519	TORUS		
19812	14	1-JB-6448	JUNCTION BOX SERVING 1-TE-162D	U1 RB	519	TORUS		
19813	14	1-JB-6449	JUNCTION BOX SERVING 1-TE-162E	U1 RB	519	TORUS		

SSEL NUMBER	CLASS	EQUIPMENT I.D.	DESCRIPTION	BUILDING	ELEV.	ROOM	ISSUE	TRAIN
19814	14	1-JB-6450	JUNCTION BOX SERVING 1-TE-162F	U1 RB	519	TORUS		
19815	14	1-JB-6451	JUNCTION BOX SERVING 1-TE-162G	U1 RB	519	TORUS		
19816	14	1-JB-6452	JUNCTION BOX SERVING 1-TE-162H	U1 RB	519	TORUS		

ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN) UNIT 1

RESPONSE TO NRC GENERIC LETTER (GL) 88-20, SUPPLEMENT 4
INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE) FOR
SEVERE ACCIDENT VULNERABILITIES

UNIT 1 IPEEE FIRE INDUCED VULNERABILITY EVALUATION

ORIGINAL

TVAN CALCULATION COVERSHEET/CCRIS UPDATE

QA Record

REV 0 EDMS/RIMS NO.	EDMS TYPE:	EDMS ACCESSION NO (N/A for REV. 0)
W78041207004	calculations(nuclear)	W78 050114 003

Calc Title: **Unit 1 IPEEE Fire Induced Vulnerability Evaluation**

CALC ID	TYPE	ORG	PLANT	BRANCH	NUMBER	CUR REV	NEW REV	REVISION APPLICABILITY
CURRENT	CN	NUC	BFN	NTB	NDN1-999-2004-0010	0	1	Entire calc ☐
NEW	CN	NUC						Selected pages ☒

ACTION	NEW ☐ REVISION ☒	DELETE ☐ RENAME ☐	SUPERSEDE ☐ DUPLICATE ☐	CCRIS UPDATE ONLY ☐ (Verifier Approval Signatures Not Required)	No CCRIS Changes ☐ (For calc revision, CCRIS been reviewed and no CCRIS changes required)

UNITS	SYSTEMS	UNIDS
1	999	N/A

DCN,EDC,N/A	APPLICABLE DESIGN DOCUMENT(S)	CLASSIFICATION
N/A		D

QUALITY RELATED?	SAFETY RELATED? (If yes, QR = yes)	UNVERIFIED ASSUMPTION	SPECIAL REQUIREMENTS AND/OR LIMITING CONDITIONS?	DESIGN OUTPUT ATTACHMENT?	SAR/TS AFFECTED
Yes ☐ No ☒	Yes ☐ No ☒	Yes ☐ No ☒	Yes ☐ No ☒	Yes ☐ No ☒	Yes ☐ No ☒

PREPARER ID	PREPARER PHONE NO	PREPARING ORG (BRANCH)	VERIFICATION METHOD	NEW METHOD OF ANALYSIS
R. Abbas	256-729-7886	NTB	N/A	☐ Yes ☒ No

PREPARER SIGNATURE	DATE	CHECKER SIGNATURE	DATE
Rashid Abbas *Rashid Abbas*	1-14-05	*Mark A. Richardson*	1-14-05
VERIFIER SIGNATURE	**DATE**	**APPROVAL SIGNATURE**	**DATE**
N/A		*J. Valente for W.D. Crouch*	14 Jan 05

STATEMENT OF PROBLEM/ABSTRACT

Problem: *Perform an evaluation of BFN Unit 1 in response to Generic Letter 88-20, Supplement 4 to determine the plant vulnerability to internal fire events.*

Abstract:
This evaluation is primarily based on the Fire Induced Vulnerability Evaluation (FIVE) methodology developed by the Electric Power Research Institute (EPRI) (Reference 1). It provides a combination of probabilistic and deterministic techniques for examining BFN's fire probability and protection characteristics. The FIVE methodology consists of a progressive screening evaluation, in which plant fire areas are screened from consideration based on qualitative information or by quantitative analysis. The availability of plant equipment is based on a combination of events that lead to fire damage and loss of safe shutdown function. If at any point in the process, the frequency of loosing a safe shutdown function is less than 1E-6/reactor year, the vulnerability to the plant from a fire at that location will not be considered significant and can be screened out from further evaluation. Implicit in this statement is that core damage from that particular fire-initiated event in that fire compartment is negligible. If necessary, fire PRA methods utilizing fire severity, probability of non-suppression factors, etc., have also been used to analyze the affects of fire. This evaluation concluded that there are no significant fire-induced vulnerabilities in any area of the plant affecting operation of BFN Unit 1.

MICROFICHE/EFICHE Yes ☐ No ☒ FICHE NUMBER(S)

☒ LOAD INTO EDMS AND RETURN CALCULATION TO CALCULATION LIBRARY. ADDRESS:

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NDN1-999-2004-0010
Unit 1 IPEEE Fire Induced Vulnerability Evaluation

TVAN CALCULATION COVERSHEET/CCRIS UPDATE

Page ___________

CALC ID	TYPE	ORG	PLANT	BRANCH	NUMBER	REV
	CN	NUC	BFN	NTB	NDN1-999-2004-0010	1

ALTERNATE CALCULATION IDENTIFICATION

BLDG	ROOM	ELEV	COORD/AZIM	FIRM	Print Report Yes ☐
NA	NA	NA	NA		

CATEGORIES N/A

KEY NOUNS (A-add, D-delete)

ACTION (A/D)	KEY NOUN	A/D	KEY NOUN
A	PROBABILITY	A	FIRE
A	PROBABALISTIC		
A	PSA		
A	RISK		
A	APPENDIX R		

CROSS-REFERENCES (A-add, C-change, D-delete)

ACTION (A/C/D)	XREF CODE	XREF TYPE	XREF PLANT	XREF BRANCH	XREF NUMBER	XREF REV
	P	SO	BFN	NTB	RISKMAN® WINDOWS 7.10	

CCRIS ONLY UPDATES:
Following are required only when making keyword/cross reference CCRIS updates and page 1 of form NEDP-2-1 is not included:

PREPARER SIGNATURE	DATE	CHECKER SIGNATURE	DATE
PREPARER PHONE NO.	EDMS ACCESSION NO. W78 050114 003		

NDN1-999-2004-0010
Unit 1 IPEEE Fire Induced Vulnerability Evaluation

SITE ENGINEERING CALCULATION

TITLE:	**Unit 1 IPEEE Fire Induced Vulnerability Evaluation**
NO.	DESCRIPTION
0	INITIAL ISSUE The UFSAR and Fire Protection Report have been reviewed by R. Abbas and this calculation does not affect these documents. Total Pages 250
1	Revised Table of Contents to correct page numbers. Revision bars eliminated for clarity. The UFSAR and Fire Protection Report have been reviewed by R. Abbas and this calculation does not affect these documents. Total Pages 250

TVAN COMPUTER INPUT FILE STORAGE INFORMATION SHEET

Document	NDN1-999-2004-0010	Rev. 1	Plant: BFN	

Subject: This calculation performs IPEEE fire induced vulnerability evaluation.

☐ Electronic storage of the input files for this calculation is not required. Comments:

☒ Input files for this calculation have been stored electronically and sufficient identifying information is provided below for each input file. (Any retrieved file requires re-verification of its contents before use.)

FILE NUMBER	FILE NAME (ext. zip)	DESCRIPTION
	U1F41022.zip	RISKMAN® (Windows 7.10) for BFN U1 IPEEE Fire Analysis

NDN1-999-2004-0010
Unit 1 IPEEE Fire Induced Vulnerability Evaluation

TABLE OF CONTENTS

Tables

1.0 INTRODUCTION

This calculation evaluates the fire induced hazards at Browns Ferry Nuclear Plant Unit 1 and determines the risk of core damage due to various fire scenarios.

1.1 Purpose

This calculation performs an evaluation of BFN Unit 1 in response to Generic Letter 88-20, Supplement 4 to determine the plant vulnerability to internal fire events. This evaluation is primarily based on the Fire Induced Vulnerability Evaluation (FIVE) methodology developed by the Electric Power Research Institute (EPRI) (Reference 1). It provides a combination of probabilistic and deterministic techniques for examining BFN Unit 1 fire probability and protection characteristics. The FIVE methodology consists of a progressive screening evaluation, in which plant fire areas are screened from consideration based on qualitative information or by quantitative analysis. The availability of plant equipment is based on a combination of events that lead to fire damage and loss of safe shutdown function. If at any point in the process, the frequency of loosing a safe shutdown function is less than 1E-6/reactor year, the vulnerability to the plant from a fire at that location will not be considered significant and can be screened out from further evaluation. Implicit in this statement is that core damage from that particular fire-initiated event in that fire compartment is negligible.

1.2 Overview of Fire Induced Vulnerability Evaluation and Implementation Process

The BFN evaluation of fire induced hazards and for screening fire areas from further consideration is primarily based on EPRI FIVE documentation (Reference 1), Fire PRA Implementation Guide (Reference 5) and Supplement to the Implementation Guide (Reference 6). Additional PSA techniques involving fire severity factors and Fire Modeling (using Zone Models e.g., CFAST) to determine the consequences of postulated fires in terms of detection, growth, propagation and suppression were used to refine the initial conservatism. The screening criteria of less than 1E-06 core damage frequency due to fire related initiating events was used. The FIVE documentation describes the fire evaluation process in three phases. The steps involved in each of these phases and their implementation is described below.

<u>Phase I - Qualitative screening and fire compartment interaction analysis.</u>
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.

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 - 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)
This phase identifies individual and generic plant fire hazards and their associated fire ignition frequencies for the unscreened plant fire areas and zones. 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.

The ignition frequency 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.

Phase II (Step 2)
The purpose of this step is to evaluate the likelihood of redundant/alternate shutdown paths being unavailable at the same time a fire occurs within a fire compartment. The PSA model impacts caused by the fires of concern is evaluated generating a "conditional core damage probability," (CCDP) or "P2" value, as it is identified in the EPRI FIVE documentation. 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.

During this step, all fires were assumed to engulf the affected area and result in a plant trip or shutdown for Unit 1. The probability for redundant/alternate system unavailability, or "conditional core damage probability" (i.e., "P2" value) was calculated using the PSA 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) below the screening criteria of 1E-06 were then screened from further consideration.

Phase II (Step 3)
Detailed fire damage assessment involving fixed and transient combustibles is performed by using deterministic/fire modeling methods. Credit for fire protection features to limit fire damage is considered. The PSA model is further refined by identifying specific plant impacts due to fires in the various areas based on the detailed fire damage assessment.

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 or zone type fire models 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 6.0.

Due to the specific nature of the Control Room, guidance for the evaluation of this area was taken directly from Fire PRA Supplement (Reference 6) and from BFN's response to RAI (Reference 12). 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 based on fire severity factors, was used to segment the area fire frequency into individual cases for evaluation. This step included 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 and 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. In response to RAI, cable spreading room was further evaluated for potential of fire growth and propagation and is included in this calculation.

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, Reference 10) and the evaluation of containment performance.

2.0 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 1, but includes potential fire ignition sources that are located in the Unit 2 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.

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 1, the Unit 1 Reactor Building, was further subdivided into 6 separate fire zones. 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).
>
> 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).

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 lists the fire areas, fire zones, and compartments used in this study.

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

Area	Description
1-1	Unit 1 Reactor Building, 519' through 565' Elevation (West side of Reactor Building)
1-2	Unit 1 Reactor Building, 519' through 565' Elevations (East side of RX Bldg.) and Elevator/Stairway at El 593', 621', and 639'.
1-3	Unit 1 Reactor Building, 593' Elevation, North Side
1-4	Unit 1 Reactor Building, 593' Elevation, South Side and RHR Heat Exchanger Rooms
1-5	Unit 1 Reactor Building, 621' Elevation and North Side of 639' Elevations
1-6	Unit 1 Reactor Building, South Side of 639' Elevation
2	Unit 2 Reactor Building
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)
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

Table 2-2 Browns Ferry Fire Areas, Fire Zones and Compartments	
Area	Description
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)
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 6.2.10.

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	23
Battery Chargers	34
Fire Protection Panels	25
Non-Qualified Junction Boxes (Allocated by Millions of BTU of Cable)	13,388
Non-Qualified Cable (In Millions of BTU)	13,388
Offgas/Hydrogen Recombiners	3
Motor Generator Sets	19
Transformers (Indoor)	48
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	~8 to 12 feet
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)

3.0 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 (References 33 and 34). 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 1, 16 and 25 were further subdivided into separate fire zones and compartments.*

** *Subdivision of fire areas 2 & 3 are addressed in References 35 and 36.*

The results of the Fire Compartment Interaction Analysis for remaining plant fire areas, including the fire zones and compartments within fire areas 1, 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
1-1	1-2	Yes	Yes	Yes	(Note 7)	5, 8, 9
	1-3	Yes	Yes	No	3, 6	1, 5, 8, 9
	1-4	Yes	Yes	No	3, 6	1, 5, 8, 9
1-2	1-3	Yes	Yes	No	3, 6	1, 5, 8, 9
	1-1	Yes	Yes	Yes	(Note 7)	5, 8, 9
	1-4	Yes	Yes	No	3, 6	1, 5, 8, 9
1-3	1-1	Yes	Yes	No	3, 6	1, 5, 8, 9
	1-2	Yes	Yes	No	3, 6	1, 5, 8, 9
	1-4	Yes	Yes	Yes	(Note 7)	5, 8, 9
	1-5	Yes	Yes	No	3, 6	1, 8, 9
1-4	1-1	Yes	Yes	No	3, 6	1, 8, 9
	1-2	Yes	Yes	No	3, 6	1, 8, 9
	1-3	Yes	Yes	Yes	(Note 7)	8, 9
	1-5	Yes	Yes	No	3, 6	1, 8, 9
1-5	1-3	Yes	Yes	No	3, 6	1, 8, 9
	1-4	Yes	Yes	No	3, 6	1, 8, 9
	1-6	Yes	Yes	Yes	(Note 7)	1, 8, 9
1-6	1-5	Yes	Yes	Yes	(Note 7)	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
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

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
16-2	16-1	Yes	Yes	No	(Note 5) 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
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 6)
	25-3	Yes	Yes	No	2	(Note 6)
25-2	25-1	Yes	Yes	No	2	(Note 6)
	25-3	Yes	Yes	No	2	(Note 6)
25-3	25-1	Yes	Yes	No	2	(Note 6)
	25-2	Yes	Yes	No	2	(Note 6)

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 or shutdown

(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 6.2.8.1.

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

(5) The potential for fire spread from compartment 16-2 to 16-3 is discussed in Section 3.3.1. The detailed evaluation of this potential multiple area fire is presented in Section 6.2.8.2.

(6) Separation between Turbine Building compartments is described in Section 3.3.2.

(7) Unit 1 Reactor Building fire zones 1-1/1-2, 1-3/1-4 and 1-5/1-6 are separated by 20 foot boundary areas, 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 6.2.1.

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.

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). 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 and meet the location and placement requirements of NFPA 72. These rooms have a relatively low ceiling height (12'); ceiling is beamed construction type (will trap smoke and heat); and detectors are located within beam pockets. All these features help early fire detection. 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

These areas primarily house low voltage electrical cabinets. Fire events experience with low voltage (250V or less) electrical fires indicates that these fires are slowly developing. Electrical cabinets are separated from each other by double walls and have some air gap. Fire is not expected to spread to an adjacent cabinet for at least 15 minutes (EPRI TR-105928, Appendix H). 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 a fire developing in compartment 16-1 and assessment of smoke detection and manual suppression is further discussed in Section 6.2.8.1.

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.

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.

Potential for fire spread from compartment 16-2 to 16-3. Fire spread between these two compartments can be screened out based on the EPRI screening criteria 6 (automatic suppression in the area). However, due to the significant amount of combustibles and the history of fire in this area, the CSR was specifically evaluated.

Fire compartment 16-2, Cable Spreading Room (CSR) is located in the control building. It interfaces with the control rooms (16-3) above and below with series of rooms including auxiliary instrument rooms, computer rooms, etc. (16-1) and fire areas 17, 18 and 19. CSR is not separated from control rooms by fire rated barriers. However, 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 and provide protection against spread of fire and smoke. Note that appropriate pressure seals are provided to maintain control room habitability. Cable spreading room presents a deep seated fire hazard scenario and therefore, a quick response and high density sprinkler system was designed for the area. The CSR ceiling is of obstructed construction, i.e., construction where beams, trusses, or other members impede heat flow or water distribution in a manner that materially affects the ability of sprinklers to control or suppress a fire. Beams are approximately 30" deep and spaced approximately 8 ft. apart and therefore treated as separate spaces. Cross members provide additional obstruction forming deep pockets. The detection and suppression design considered all of these aspects. Smoke detectors and sprinkler are placed within the beam pockets to provide prompt detection capability and adequate spray pattern. Additionally, intermediate level sprinklers are installed in the flue space between stacks of cable trays (similar to protection of rack storage occupancies). Due to congestion of cable trays, two smoke detectors are placed within each beam pocket (37' x 8'). Sprinkler are spaced approximately 10' apart. Sprinkler and smoke detectors design meets the NFPA Code requirements.

Based on the above described ceiling construction and detector/sprinkler placement, smoke detector response and sprinkler activation time is calculated in Section 6.2.8.2. The Heat Release Rate is assumed to be a slow growth fire.

The evaluation shows that for slow growing fires, time to reach 300 Btu/sec fire size (design objective to limit the fire to one or two trays) is 219 seconds, whereas time to detect and activate sprinklers is no more than 50 seconds. Therefore, it can be concluded that fires in the CSR can be detected and suppressed well before critical conditions are reached. Even for medium and fast developing fires the time to reach 300 Btu/sec is 164 seconds and 82 seconds respectively and the sprinkler system is expected to control such fires and prevent fires from propagating to the control rooms located above.

3.3.2 Potential for Fire Spread between Turbine Building Compartments

The Turbine Building fire area is segmented for this analysis into 3 compartments. 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. 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.

<u>Potential for fire spread between compartments 25-2 and 25-1 or 25-3.</u> 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.

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.

4.0 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 and later updated in EPRI FEDB 2001 (Reference 3).

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

Fire area/Zone	Description	Ignition Frequency
1-1	Unit 1 Reactor Building, 519' through 565' Elevations (West side of Rx Bldg.)	6.08E-02
1-2	Unit 1 Reactor Building, 519' through 565' Elevations (East side of Rx Bldg.)	3.52E-02
1-3	Unit 1 Reactor Building, 593' Elevation, North Side	2.32E-02
1-4	Unit 1 Reactor Building, 593' Elevation, South Side and RHR HX Rooms	2.12E-02
1-5	Unit 1 Reactor Building, 621' Elevation and North Side of 639' Elevations	4.88E-02
1-6	Unit 1 Reactor Building, South Side of 639' Elevation	3.08E-02
2	Unit 2 Reactor Building	1.27E-01
3	Unit 3 Reactor Building	1.26E-01
4	4kV Shutdown Board Room B (Unit 1 Reactor Building, 593' Elevation)	1.94E-02
5	4kV Shutdown Board Room A and 250V Battery Room (Unit 1 RB, EL 621')	2.36E-02
6	480V Shutdown Board Room 1A (Unit 1 Reactor Building, 621' Elevation)	1.92E-02
7	480V Shutdown Board Room 1B (Unit 1 Reactor Building, 621' Elevation)	1.92E-02
8	4kV Shutdown Board Room D (Unit 2 Reactor Building, 593' Elevation)	1.92E-02
9	4kV Shutdown Board Room C and 250V Battery Room (Unit 2 RB, EL 621')	2.26E-02
10	480V Shutdown Board Room 2A (Unit 2 Reactor Building, 621' Elevation)	1.92E-02
11	480V Shutdown Board Room 2B (Unit 2 Reactor Building, 621' Elevation)	1.92E-02
12	Shutdown Board Room F (Unit 3 Reactor Building, 593' Elevation)	2.03E-02
13	Shutdown Board Room E (Unit 3 Reactor Building, 621' Elevation)	2.01E-02
14	480V Shutdown Board Room 3A (Unit 3 Reactor Building, 621' Elevation)	1.92E-02
15	480V Shutdown Board Room 3B (Unit 3 Reactor Building, 621' Elevation)	1.92E-02
16-1	Control Building - 593' Elevation	3.78E-02
16-2	Control Building - 606' (Cable Spreading Room)	1.20E-02
16-3	Control Building - 617' (Control Room)	6.92E-02
17	Unit 1 Battery and Battery Board Room, Control Building 593' Elevation	5.08E-02
18	Unit 2 Battery and Battery Board Room, Control Building 593' Elevation	4.91E-02

Table 4-1 Fire Area Ignition Frequencies		
Fire area/Zone	**Description**	**Ignition Frequency**
19	Unit 3 Battery and Battery Board Room, Control Building 593' Elevation	4.91E-02
20	Unit 1 and 2 Diesel Generator Building	2.14E-01
21	Unit 3 Diesel Generator Building	2.13E-01
22	4kV Shutdown Board Room 3EA and 3EB, 583' Elevation, Unit 3 DGB	1.93E-02
23	4kV Shutdown Board Room 3EC and 3ED, 583' Elevation, Unit 3 DGB	1.93E-02
24	4kV Bus Tie Board Room, 565' Elevation, Unit 3 Diesel Generator Building	1.92E-02
25-1	Intake Pump Station	7.77E-02
25-2	Pipe Tunnel	1.09E-05
25-3	Turbine Building	5.59E-01
Total Plant Fire Frequency for 3 Units		**2.10E+00**

5.0 PHASE II.2 - QUANTITATIVE SCREENING

Following the generation of the fire ignition frequencies for each fire, each plant area must be evaluated for the probability of core damage, given an assumed engulfing fire in the area and the resulting damage to safe shutdown components. Components remaining free of fire damage are assumed to be available (except for probabilistic failures as modeled in the PSA). 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 documented in References 24, 25, and 26.

Once the conditional core damage probability (CCDP), 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 fire induced 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 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 (References 2 and 3) 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.

This analysis is performed to enable screening of less-significant areas from further consideration and identifying those areas for which 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 6.0.

5.1 IPE Level 1 Shutdown Sequence and Unavailability (P2)

Given the plant impacts for an assumed engulfing fire in a given area, the PSA model 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 totaled and normalized to reflect an initiating event frequency of 1.0. This gives the conditional core damage probability (CCDP) 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 ignition frequency to generate a fire-induced 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 or 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.

If the cause of plant trip (i.e., loss of offsite power, reactor trip with 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, turbine trip is conservatively assumed to occur.

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 probabilities, 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.

Table 5-1
Conditional Core Damage Probabilities (CCDP) for Selected Initiating Events

Description of Plant Trip	Initiating Event Designation	Initiating Event Frequency (A)	Core Damage Frequency (B)	CCDP (P2 = B/A)
Loss of all condensate	TLCF	9.09E-03	3.76E-08	4.14E-06
MSIV closure	IMSIV	5.70E-02	9.67E-08	1.70E-06
Total loss of feedwater	TLFW	2.58E-02	4.59E-08	1.78E-06
Loss of plant air	LOPA	1.20E-02	6.58E-08	5.48E-06
Total loss of offsite power	LOSP	6.43E-03	2.17E-07	3.37E-05
Turbine trip	TT	5.09E-01	1.86E-07	3.65E-07

The initiating event frequencies and core damage frequencies shown in Table 5-1 were taken directly from RISKMAN® model. These values were then "normalized" to reflect an initiating event frequency of 1.0 to obtain the conditional core damage probability (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-induced core damage frequency.

For example, a transient resulting in turbine trip 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 TT initiating event, shown above. Table 5-1 shows the core damage frequency that would be reported by the quantification program for the TT initiating event (1.86E-07). 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 TT initiating event, if it had an initiating event frequency of 1.0 (1.86E-7/5.09E-1 = 3.65E-7). This value would then be multiplied by the new initiating event frequency (1E-02) to calculate a fire-related core damage frequency of 3.65E-09.

The above example applies to the situation where a base line initiating event (such as TT) can subsume the impact of a fire. If there are additional damages to the PRA equipment, the impact should be added to the event sequence model by changing the split fraction rules.

For the Unit 1 fire analysis in this report, a single RISKMAN model was developed for all the fire scenarios. Since fire in multiple areas can cause the same initiating event (e.g., turbine trip), each individual fire scenario was assigned a unique initiator designator, such as F5, which specifies the fire initiator for Fire Area 5. The event tree logic rules were developed for a fire initiator to first mimic all the logic rules of its associated base line initiating event (such as turbine trip), and then fail or degrade the appropriate top events based on the fire impact. These fire initiators were assigned an initiator frequency of 1.0, hence the core damage frequency calculated can be used directly as CCDP (i.e., the P2 value).

Tables 5-1.1 through 5-1.25 perform the risk evaluation of a postulated engulfing fire in a given area based on the above described methodology.

Unit 1 and Unit 2 PSA Model Comparisons - The BFN Unit 2 PSA model was the starting point for the ~~Unit 1~~ model along with the Extended Power Uprate (EPU) considerations. The CDF results for Unit 2 were dominated by sequences with the following characteristics:

1. An initiating event that causes a loss of injection from the power conversion system.
2. A common cause failure of HPCI and RCIC.
3. A failure to depressurize to allow low-pressure injection. This was comprised by both hardware and human action failures, but the human action failure was more important.

These sequences were reviewed in detail. Particular attention was paid to HPCI/RCIC common cause failures and the human action failure to depressurize during the development of the Unit 1 model. The Unit 1 model incorporated improvements to both elements. Although still dominant, the CDF resulting from these type sequences is much lower in the Unit 1 model when compared to the Unit 2 model. A secondary effect is that the data updates for the diesel generators resulted in lower probabilities of failure and reduces the CCDP for LOSP events. The Unit 1 base model CDF and LERF values are as follows:

CDF — 1.86E-06
LERF — 1.87E-07

Table 5-1.1 Quantitative Screening					
Fire Compartments	1-1, 1-2, 1-3, 1-4, 1-5, 1-6	Unit 1 Reactor Building			
	PSA MODEL IMPACTS DUE TO FIRE DAMAGE				
Fire damaged Components (Direct Impacts)	Mitigating Systems Impact (Top Events)	Event Tree Impact	Fire Damaged Components (Indirect Impacts)	Mitigating Systems Impact (Top Events)	Event Tree Impact
The Unit 1 Reactor Building consists of six fire zones, which are analyzed as individual fire areas in volume 1 of the Browns Ferry Fire Protection Report. 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 6.2.1.					
Risk Evaluation					
Initiating Event	Initiating Event Frequency (IE)	CDF_{MS}	$CCDP_{MS}$ (CDF_{MS}/IE)	Ignition Frequency (F1)	CDF_{FIRE} ($CCDP_{MS}$*F1)
			1.00E+00	6.08E-02	6.08E-02
			1.00E+00	3.52E-02	3.52E-02
			1.00E+00	2.32E-02	2.32E-02
			1.00E+00	2.12E-02	2.12E-02
			1.00E+00	4.88E-02	4.88E-02
			1.00E+00	3.08E-02	3.08E-02
				2.20E-01	

Table 5-1.2
Quantitative Screening

Fire Compartments	2	Unit 2 Reactor Building			
	PSA MODEL IMPACTS DUE TO FIRE DAMAGE				
Fire damaged Components (Direct Impacts)	Mitigating Systems Impact (Top Events)	Event Tree Impact	Fire Damaged Components (Indirect Impacts)	Mitigating Systems Impact (Top Events)	Event Tree Impact
Damage to "unit-specific" (i.e., Unit 2) components due to postulated fire scenarios in the Unit 2 Reactor Building would not be expected to require shutdown of Unit 1 (i.e., a plant trip, or initiating event, would not be expected to occur for Unit 1 due to fire in the Unit 2 Reactor Building). Damage to "unit-common" components however, such as power cables, that transit through the Unit 2 Reactor Building, may require Unit 1 to be shut down or tripped. It is expected that fires in unit 1 reactor building will bound core damage frequency due to fire-related initiating events in the Unit 2 Reactor Building. Refer to Section 6.2.2 for specific evaluation of fire damage to "unit-common" components and impact on Unit 1 core damage frequency.					
	Risk Evaluation				
Initiating Event	Initiating Event Frequency (IE)	CDF_{MS}	$CCDP_{MS}$ (CDF_{MS}/IE)	Ignition Frequency (F1)	CDF_{FIRE} ($CCDP_{MS}$*F1)
			1.00E+00	1.27E-01	1.27E-01
			(assumed)		
Conclusion: For a fire in the Unit 2 Reactor Building, it is unlikely that the operator would initiate a reactor trip on Unit 1. For purposes of this screening analysis, though, all Unit 2 Reactor Building fires are conservatively assumed to result in a precautionary trip of Unit 1. 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 6.2.2.					

Table 5-1.3 Quantitative Screening					
Fire Compartments	**3**	**Unit 3 Reactor Building**			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
Damage to "unit-specific" (i.e., Unit 3) components due to postulated fire scenarios in the Unit 3 Reactor Building would not be expected to require shutdown of Unit 1 (i.e., a plant trip, or initiating event, would not be expected to occur for Unit 1 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, may require Unit 1 to be shut down or tripped. It is expected that fires in unit 1 reactor building will bound core damage frequency due to fire-related initiating events in the Unit 3 Reactor Building. Refer to Section 6.2.3 for specific evaluation of fire damage to "unit-common" components and impact on Unit 1 core damage frequency.					
Risk Evaluation					
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
			1.00E+00	1.26E-01	1.26E-01
			(assumed)		
Conclusion: *For a fire in the Unit 3 Reactor Building, it is unlikely that the operator would initiate a reactor trip on Unit 1. For purposes of this screening* analysis, though, all Unit 3 Reactor Building fires are conservatively assumed to result in a precautionary trip of Unit 1. 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 6.2.3.					

Table 5-1.4 Quantitative Screening					
Fire Area	**4**	**4kV Shutdown Board Room B (Unit 1 Reactor Building, 593' Elevation)**			
	PSA MODEL IMPACTS DUE TO FIRE DAMAGE				
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
4kV Shutdown Board B	AB	ELECT12	Diesel Generator B	GB	ELECT12
480V RMOV Board 1B	RF	ELECT12	Shutdown Bus 1	SHUT1	ELECT12
250V RMOV Board 1B	RC	ELECT3	Shutdown Bus 2	SHUT2	ELECT12
1-LPNL-925-0541 (ACU 1B)			480V Shutdown Board 2A	RS	ELECT12
1-TS-031-7205D (SDBR ACU 1A)			480V RMOV Board 2A	RH	ELECT12
1-TS-031-7206C (SDBR ACU 1B)			RCIC Pump (New Cable Tray Impact)	RCI	HPGTET
Panel 0-PNL-25-45B (4kV Shutdown Board B Logic Relays)	AB	ELECT12	Core Spray Loop II (New Cable Tray Impact)	CS	LPGTET
Division I ECCS Analog Trip Unit Inverters (Unit 1 only)	PX1	SIGL	RHR Loop II (New Cable Tray Impact)	RPB, RPD	RHRGT
I&C Bus 1B Equipment	DO	ELECT3	Drywell High Pressure Signal (New Cable Tray Impact)	DW	SIGL
			Diesel Generator D (New Cable Tray Impact)	GD	ELECT12
			Crosstie from Unit 2 (New Cable Tray Impact)	U2X	LPGTET
			Recovery Action Impact:	SDREC	ELECT3
				RFRHW	RHRGT
				RCOK (macro)	ELECT3
Risk Evaluation					
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
IMSIV	5.70E-02	9.22E-04	1.62E-02	1.94E-02	3.14E-04

Table 5-1.4 Quantitative Screening
Comments: The potential fire-related failure of 4kV shutdown board B could also impact the operation of additional plant components listed under indirect impacts. 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 BFN Unit 1 (Table 2-4 of Unit 1 Initiator Notebook) documented possible MSIV isolation and reactor scram due to 480V RMOV Board 1B failure (initiator IMSIV). This scenario is modeled as fire initiator F4 in the RISKMAN model. New cable trays routed in this room would impact the following systems: RCIC pump, CS Loop II, RHR Loop II, Drywell high pressure signal, Diesel Generators B and D (with Diesel Generator B already modeled with existing cable trays), and RHR crosstie (Unit 2 Loop I to support Unit 1 Loop II). Recovery of 480V RMOV Board 1B (RFRHW top event), and crosstie from 4kV Shutdown Board D to B (SDREC top event) were also impacted by the new cable trays. In addition, loss of recirculation flow would result due to the new cable tray impact, causing a reactor trip due to power/flow mismatch; but this is enveloped by the more conservative initiator, IMSIV. The CCDP for this fire area were compared with the Unit 2 results for 4kV SD BD C. The Unit 1 CCDP is lower by roughly a factor of 5. The Unit 2 evaluation was replicated using the Unit 2 EPU model. The CDF for this area was numerically close (1.07E-2 versus 9.87E-03). The highest frequency sequences from both models were then compared. Several items were identified that make the comparison difficult. On Unit 2, two top events are important to the results. One is top event HS, which models the recovery of the heat sink. Failure of this top event occurred in all the sequences examined. The bases for these values are not documented, but the values are from the original BFN Unit 1 PRA. This recovery was not included in the Unit 1 PSA because a higher standard for data on Unit 1. The second top event that is important in Unit 2 is the OLP (Operators control low pressure injection). Top event OLP is guaranteed failed even though core spray is functional. These two top events have opposite affects on the results. **Conclusion:** 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 reactor trip (with MSIV closure) 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 6.2.4.

Table 5-1.5 Quantitative Screening					
Fire Area	5	4kV Shutdown Board Room A and 250V Battery Room (Unit 1 RB, EL 621')			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	Mitigating Systems Impact (Top Events)	Event Tree Impact	Fire Damaged Components (Indirect Impacts)	Mitigating Systems Impact (Top Events)	Event Tree Impact
4kV Shutdown Board A	AA	ELECT12	Diesel Generator A	GA	ELECT12
480V RMOV Board 1A	RE	ELECT12	Shutdown Bus 1	SHUT1	ELECT12
250V RMOV Board 1A	RB	ELECT12	Shutdown Bus 2	SHUT2	ELECT12
Panel 25-32 (backup control panel)			Diesel Generator B	GB	ELECT12
Panel 25-45A (4kV SDBD A Relay)	AA	ELECT12	RHR Loop I (New Cable Tray Impact)	RPA, RPC	RHRGT
I&C Bus 1B Equipment	DO	ELECT3	RHR SW Pumps C1, C2 (New Cable Tray Impact)	SW1C, SW2C	MESUPT
250V DC Battery, Battery Chargers, Dist. panel SB-A & B	DA, DC	ELECT12	EECW Pump A3 (New Cable Tray Impact)	EA	MESUPT
SDBR Emerg. Cooling Unit					
I&C Bus 1A Equipment	DN	ELECT3	Recovery Action Impact:	SDREC	ELECT3
ATU Inverters - Div II (Unit 1)	PX2, HPISUP	SIGL, HPGTET		RERHW	RHRGT
				RBOK (macro)	ELECT3
Risk Evaluation					
Initiating Event	Initiating Event Frequency (IE)	CDF_{MS}	$CCDP_{MS}$ (CDF_{MS}/IE)	Ignition Frequency (F1)	CDF_{FIRE} ($CCDP_{MS}$*F1)
IMSIV	5.70E-02	4.82E-04	8.46E-03	2.36E-02	1.99E-04

Comments: The walkdowns also evaluated cable routing through this area. During this review, it was observed 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. Top Event GB is conservatively assumed failed. Also, the fire-related failure of 4kV shutdown board A could potentially impact the operation of additional plant components listed under indirect impacts. The failure modes and effects analysis of BFN Unit 1 (Table 2-4 of Unit 1 Initiator Notebook) documented possible MSIV isolation and reactor scram due to 480V RMOV Board 1A failure (initiator IMSIV). This scenario is modeled as fire initiator F5 in the RISKMAN model. New cable trays routed in this room would impact the following systems: RHR loop I, RHRSW pumps C1 and C2, EECW pump A3, Diesel Generator A (DG A is also impacted by existing cable trays). In addition, loss of recirculation flow would result due to the new cable tray impact, causing a reactor trip due to power/flow mismatch; but this is enveloped by the more conservative initiator, IMSIV.

The CCDP developed in the Unit 1 model (for Shutdown Board A room) is lower than in the Unit 2 model (for Shutdown Board D room). The dominant sequences in the Unit 2 model contain failures of suppression pool cooling as the critical failure. Reviews have determined that the logic rules used to guarantee the failure of suppression pool cooling are conservative. It is believed that these conservatisms have an insignificant effect on the baseline PSA. However, given the extreme failures postulated in the fire analysis, the CCDP can be greatly overstated.

Conclusion: 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 reactor trip (with MSIV closure) 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

Table 5-1.5 Quantitative Screening
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.

Table 5-1.6 Quantitative Screening					
Fire Area	**6**	**480V Shutdown Board Room 1A (Unit 1 Reactor Building, 621' Elevation)**			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
480V Shutdown Board 1A	RQ	ELECT12	Drywell High Pressure Signal (New Cable Tray Impact)	DW	SIGL
Panel 1-25-44A-11	GA	ELECT12			
Panel 1-25-44B-11	GB	ELECT12			
			Recovery Actions Impact:		
			Shutdown Board Recovery	RQOK (macro)	ELECT3
Risk Evaluation					
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
IMSIV	5.70E-02	3.29E-07	5.76E-06	1.92E-02	1.11E-07

Comments: The walkdowns confirmed that no cables traverse this area, other than those associated with 480V shutdown board 1A and the 480V load shed panels. 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 division I diesel generators A and B at top events GA and GB, in addition to failing shutdown board recovery at macro RQOK. This treatment is conservative in that it fails 4160V switchgear following a loss of offsite power, in addition to the supplied 480V loads. The Failure Modes and Effects Analysis of Browns Ferry Unit 1 Key Support Systems (Table 2-4 of the initiating events notebook) states that failure of 480V Shutdown Board 1A can result in MSIV closure and reactor scram (initiator IMSIV). This scenario is modeled as fire initiator F6 in RISKMAN. New cable trays routed in this room would impact the following systems: Diesel Generator A (also impacted by existing cable trays), and drywell high pressure signal. In addition, loss of recirculation flow would result due to the new cable tray impact, causing a reactor trip due to power/flow mismatch; but this is enveloped by the more conservative initiator, IMSIV.

Conclusion: 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 reactor trip (with MSIV closure) and cause the loss of all plant equipment located in this fire area, regardless of fire severity or manual fire suppression.

Table 5-1.7 Quantitative Screening					
Fire Area	7	**480V Shutdown Board Room 1B (Unit 1 Reactor Building, 621' Elevation)**			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
480V Shutdown Board 1B	RR	ELECT12	480V RMOV Board 1B (New Cable Tray Impact)	RF	ELECT12
Panel 1-25-44A-12	GC	ELECT12	High Drywell Pressure Signal (New Cable Tray Impact)	DW	SIGL
Panel 1-25-44B-12	GD	ELECT12	Control Air Compressor A (New Cable Tray Impact)	PCAAS (macro)	MESUPT
			Recovery Actions Impact:		
			Shutdown Board Recovery	RROK (macro)	ELECT3
Risk Evaluation					
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
IMSIV	5.70E-02	5.96E-06	1.05E-04	1.92E-02	2.01E-06

Comments: The walkdowns confirmed that no cables traverse this area, other than those associated with 480V shutdown board 1B and the 480V load shed panels. The potential failure of the 480V load sequencing logic circuits in panels 1-PNL-25-44A-12 and 1-PNL-25-44B-12 was conservatively modeled by failing division II diesel generators C and D at top events GC and GD, in addition to failing shutdown board recovery at macro RROK. This treatment is conservative in that it fails 4160V switchgear following a loss of offsite power, in addition to the supplied 480V loads. The Failure Modes and Effects Analysis of Browns Ferry Unit 1 Key Support Systems (Table 2-4 of the initiating events notebook) states that failure of 480V Shutdown Board 1B can result in MSIV closure and reactor scram (initiator IMSIV). This scenario is modeled as fire initiator F7 in RISKMAN. Note F7 has a higher CCDP than F6 because failure of 480V Shutdown Board 1B (RR) causes failure of 480V RMOV Board 1B (RF), which fails to crosstie of Unit 2 RHR loop I to Unit 1 RHR loop II (Top Event U2X). New cable trays routed in this room would impact the following systems: Diesel Generator C (also impacted by existing cable trays), Control Air Compressor A degradation, high drywell pressure signal, 480V RMOV Board 1B (which will fail because 480V Shutdown Board 1B failure due to existing cable tray impact), and 480V RMOV Board 1C, which was not in the Unit 1 model.

Conclusion: Since the upper bound core damage frequency for this evaluation is greater than 1E-06, fires in this area will be retained for detailed analysis, see Section 6.2.6.

Table 5-1.8
Quantitative Screening

Fire Area	8	4kV Shutdown Board Room D (Unit 2 Reactor Building, 593' Elevation)			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
4kV Shutdown Board D	AD	ELECT12	Diesel Generator D	GD	ELECT12
480V RMOV Board 2B	RI	ELECT12	Shutdown Bus 1	SHUT1	ELECT12
250V RMOV Board 2B			Shutdown Bus 2	SHUT2	ELECT12
Panel 25-45D			480V Shutdown Board 2B	RT	ELECT12
I&C Bus 2B Equipment			480V RMOV Board 2C	RJ	ELECT12
ATU Inverters - Div 1 (Unit 2)			Diesel Generator B (New Conduits Impact)	GB	ELECT12
			Core Spray Loop II (New Conduits Impact)	CS	LPGTET
			RHR Pump 1D (New Conduits Impact)	RPD	RHRGT
			Recovery Actions Impact:		
			Top event RI Recovery	RIOK (macro)	RHRGT
Risk Evaluation					
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
TT	5.09E-01	2.70E-06	5.31E-06	1.92E-02	1.02E-07

Comments: Cable routing through this area was evaluated to ensure that no other risk-significant components could be impacted by fires in this area. The fire-related failure of 4kV shutdown board D could potentially impact the operation of additional plant components described under indirect affects. While a plant trip would not be expected for Unit 1 due to fires in this area, a transient involving turbine trip (Initiator TT) has been conservatively assumed for this analysis. This fire scenario is designated fire initiator F8 in the RISKMAN model. New conduits installed in this room would impact the following systems: Diesel Generator B, Core Spray Loop II, and RHR pump 1D. Crosstie to 4kV Shutdown Board D would also be failed, but this crosstie is not in the Unit 1 model.

A detailed comparison of Unit 1 CCDP for 4kV Shutdown Board D vs. Unit 2 CCDP for 4kV Shutdown Board A was not made. However, it is expected that the "failure to depressurize" sequences account for the differences, as discussed at the end of Section 5.1 text.

Conclusion: 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 turbine trip and cause the loss of all plant equipment located in this fire area, regardless of fire severity or manual fire suppression.

Table 5-1.9 Quantitative Screening					
Fire Area	**9**	**4kV Shutdown Board Room C and 250V Battery Room (Unit 2 RB, EL 621')**			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
4kV Shutdown Board C	AC	ELECT12	Diesel Generator C	GC	ELECT12
480V RMOV Board 2A	RH	ELECT12	Shutdown Bus 2	SHUT2	ELECT12
250V RMOV Board 2A	RB	ELECT12	480V Shutdown Board 2A	RS	ELECT12
Panel 25-32 (backup control panel)			480V RMOV Board 2C	RJ	ELECT12
Panel 25-45C (4kV SDBD C Relay)			Diesel Generator A (New Conduits Impact)	GA	ELECT12
I&C Bus 2A Equipment			Diesel Generator B (New Conduits Impact)	GB	ELECT12
250V DC Battery, Battery Chargers, Dist. panel SB-C & D	DB,DD	ELECT12	Diesel Generator D (New conduits Impact)	GD	ELECT12
SDBR Emerg. Cooling Unit & 250V DC Battery supply & Exhaust fan			CS Loop II (New Conduits Impact)	CS	ELECT12
I&C Bus 1A Equipment			RHR Pump 1B (New Conduits Impact)	RPB	LPGTET
ATU Inverters - Div II			**Recovery Actions Impact:**		RHRGT
Panels 25-42A-1 & B-1(common logic relays)			Recovery of top event RH	RHOK (Macro)	RHRGT
Panels 25-42A-2 & B-2(common logic relays)					
Risk Evaluation					
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
TT	5.70E-02	3.49E-05	6.12E-04	2.26E-02	1.38E-05

Table 5-1.9 Quantitative Screening
Comments: Cable routing through this area was evaluated to ensure that no other risk-significant components could be impacted by fires in this area. The fire-related failure of 4kV shutdown board C could potentially impact the operation of additional plant components described under indirect affects. While a plant trip would not be expected for Unit 1 due to fires in this area, a transient involving turbine trip (Initiator TT) has been conservatively assumed for this analysis. This fire scenario is designated fire initiator F9 in the RISKMAN model. New conduits installed in this room would impact the following systems: Diesel Generators A, B, C, and D control circuits; we conservatively assume all four diesel generators failed (note existing cable trays already damaged Diesel Generator C). Core Spray Loop II, and RHR pump 1B were also damaged by a engulfing fire in this area. Shutdown Bus 2 is conservatively assumed damaged in this area. But Shutdown Bus 1 is still available as well as its normal power supply. This will provide power to the Unit 1 equipment that's not damaged by this engulfing fire, limiting the fire impact in this area. A detailed comparison of Unit 1 CCDP for 4kV Shutdown Board C vs. Unit 2 CCDP for 4kV Shutdown Board B was not made. The differences can be explained similarly as discussed in Table 5-1.4 for Unit 1 CCDP for Board B vs. Unit 2 CCDP for Board C. **Conclusion:** Since the upper bound core damage frequency for this evaluation is more than 1E-06, fires in this area will be retained for detailed analysis, see Section 6.2.7. 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.

Table 5-1.10 Quantitative Screening					
Fire Area	10	480V Shutdown Board Room 2A (Unit 2 Reactor Building, 621' Elevation)			
	PSA MODEL IMPACTS DUE TO FIRE DAMAGE				
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
480V Shutdown Board 2A	RS	ELECT12			
2-PNL-25-44A-11, 480V Load sequencing logic panel.	GA	ELECT12			
2-PNL-25-44B-11, 480V Load sequencing logic panel.	GB	ELECT12			
	Risk Evaluation				
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
TT	5.09E-01	3.05E-07	5.99E-07	1.92E-02	1.15E-08

Comments: The walkdowns confirmed 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 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. This treatment is conservative in that it fails 4160V switchgear following a loss of offsite power, in addition to the supplied 480V loads. While a plant trip would not be expected for Unit 1 due to fires in this area, a transient involving turbine trip (Initiator TT) has been conservatively assumed for this analysis. This fire scenario is designated fire initiator F10 in the RISKMAN model.

Conclusion: Since core damage frequency for this evaluation is below 1E-06, fires in this area can be screened from further consideration.

Table 5-1.11 Quantitative Screening					
Fire Area	**11**	**480V Shutdown Board Room 2B (Unit 2 Reactor Building, 621' Elevation)**			
	PSA MODEL IMPACTS DUE TO FIRE DAMAGE				
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
480V Shutdown Board 2B	RT	ELECT12			
2-PNL-25-44A-12, 480V Load sequencing logic panel.	GC	ELECT12			
2-PNL-25-44B-12, 480V Load sequencing logic panel.	GD	ELECT12			
RBCCW Sectionalizing Valve FCV-70-48					
HPCI Test Valve FCV-73-35					
		Risk Evaluation			
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
TT	5.09E-01	1.23E-07	2.41E-07	1.92E-02	4.63E-09

Comments: The walkdowns confirmed that no cables traverse this area, other than those associated with 480V shutdown board 2B and the 480V load shed panels. 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 II 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. While a plant trip would not be expected for Unit 1 due to fires in this area, a transient involving turbine trip (Initiator TT) has been conservatively assumed for this analysis. This fire scenario is designated fire initiator F11 in the RISKMAN model.

Conclusion: Since core damage frequency for this evaluation is below 1E-06, fires in this area can be screened from further consideration.

Table 5-1.12 Quantitative Screening					
Fire Area	**12**	**Shutdown Board Room F (Unit 3 Reactor Building, 593' Elevation)**			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
480V RMOV Board 3B			Unit 3 Shutdown Bd A3EC	A3EC	ELECT3
250V RMOV Board 3B	NO UNIT 1 IMPACTS		Unit 3 Shutdown Bd A3ED	A3ED	ELECT3
480V HVAC Board B					
120V I&C Bus 3B Equipment					
ATU Inverters Division 1 (Unit 3)					
Panel 25-654B					
Risk Evaluation					
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**		**CDF_{FIRE} ($CCDP_{MS}$*F1)**
TT	5.09E-01	2.89E-07	5.68E-07	2.03E-02	1.15E-08

Comments: The walkdowns revealed that the cables supplying 250VDC control power for 4kV shutdown boards 3EA and 3EC are routed through this area. No other Unit 1 related plant components were identified. While a plant trip would not be expected due to fires in this area, a transient involving turbine trip (Initiator TT) has been conservatively assumed for this analysis. This fire scenario is designator initiator F12 in the RISKMAN model.

Conclusion: Therefore, 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 turbine trip and cause the loss of all plant equipment located in this fire area, regardless of fire severity or manual fire suppression.

Table 5-1.13 Quantitative Screening					
Fire Area	13	Shutdown Board Room E (Unit 3 Reactor Building, 621' Elevation)			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	Mitigating Systems Impact (Top Events)	Event Tree Impact	Fire Damaged Components (Indirect Impacts)	Mitigating Systems Impact (Top Events)	Event Tree Impact
480V RMOV Board 3A			480V RMOV Board 3B	Not Modeled	
250V RMOV Board 3A	NO UNIT 1 IMPACTS		480V Diesel Aux. Bd 3EB	RP	ELECT3
Unit 3 Panel 25-32			480V Shutdown Board 3A	RX	ELECT3
120V I&C Bus 3A Equipment			480V Shutdown Board 3B	RY	ELECT3
ATU Inverters Division II (Unit 3)					
Risk Evaluation					
Initiating Event	Initiating Event Frequency (IE)	CDF_{MS}	$CCDP_{MS}$ (CDF_{MS}/IE)	Ignition Frequency (F1)	CDF_{FIRE} ($CCDP_{MS}$*F1)
TT	5.09E-01	1.45E-07	2.84E-07	2.01E-02	5.71E-09

Comments: The walkdowns revealed that the control cables associated with the equipment listed under "indirect Impacts" could be impacted due to fires in this area. While a plant trip would not be expected due to fires in this area, a transient involving turbine trip (Initiator TT) has been conservatively assumed for the Unit 1 analysis. This fire scenario is designator initiator F13 in the RISKMAN model.

Conclusion: Therefore, 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 turbine trip and cause the loss of all plant equipment located in this fire area, regardless of fire severity or manual fire suppression.

Table 5-1.14 Quantitative Screening					
Fire Area	14	480V Shutdown Board Room 3A (Unit 3 Reactor Building, 621' Elevation)			
	PSA MODEL IMPACTS DUE TO FIRE DAMAGE				
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
480V Shutdown Board 3A	RX	ELECT3			
Risk Evaluation					
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
TT	5.09E-01	1.35E-07	2.66E-07	1.92E-02	5.11E-09

Comments: The walkdowns confirmed that there is no additional Unit 1 related equipment located in this area. Also, it was confirmed during these walkdowns that no additional Unit 1 related support cables traverse through this area. While a plant trip would not be expected due to fires in this area, a transient involving turbine trip with bypass available (Initiator TT) has been conservatively assumed for this analysis. This fire scenario is designator initiator F14 in the RISKMAN model.

Conclusion: Therefore, 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 turbine trip and cause the loss of all plant equipment located in this fire area, regardless of fire severity or manual fire suppression.

Table 5-1.15 Quantitative Screening					
Fire Area	**15**	**480V Shutdown Board Room 3B (Unit 3 Reactor Building, 621' Elevation)**			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
480V Shutdown Board 3B	RY	ELECT3			
Risk Evaluation					
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
TT	5.09E-01	1.44E-07	2.84E-07	1.92E-02	5.45E-09

Comments: The walkdowns confirmed that there is no additional Unit 1 related equipment located in this area. Also, it was confirmed during these walkdowns that no additional Unit 1 related support cables traverse through this area. While a plant trip would not be expected due to fires in this area, a transient involving turbine trip (Initiator TT) has been conservatively assumed for this analysis. his fire scenario is designator initiator F15 in the RISKMAN model.

Conclusion: Therefore, 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 turbine trip and cause the loss of all plant equipment located in this fire area, regardless of fire severity or manual fire suppression.

Table 5-1.16 Quantitative Screening					
Fire Compartments	16-1, 16-2, 16-3	Control Building EL 593, Cable Spreading Rooms EL 606, Unit 1/2/3 MCR EL 617			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	Mitigating Systems Impact (Top Events)	Event Tree Impact	Fire Damaged Components (Indirect Impacts)	Mitigating Systems Impact (Top Events)	Event Tree Impact
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.					
Risk Evaluation					
Initiating Event	Initiating Event Frequency (IE)	CDF_{MS}	$CCDP_{MS}$ (CDF_{MS}/IE)	Ignition Frequency (F1)	CDF_{FIRE} ($CCDP_{MS}$*F1)
			1.00E+00	3.78E-02	3.78E-02
			1.00E+00	1.20E-02	1.20E-02
			1.00E+00	6.92E-02	6.92E-02

Table 5-1.17 Quantitative Screening					
Fire Area	**17**	**Unit 1 Battery and Battery Board Room, Control Building 593' Elevation**			
	PSA MODEL IMPACTS DUE TO FIRE DAMAGE				
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
250VDC Battery 1 (located in the battery room)	DE	ELECT12			
Battery Board 1	DE	ELECT12			
250VDC Battery Charger 1	DE	ELECT12			
Unit Preferred MMG Set 1 and Associated Equipment	NO IMPACT ON UNIT 1 MODEL				
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	PX1, PX2	SIGL			
Unit 1 Panel 9-81 (Division 1 only) (FW Inverters)	PX1	SIGL			
Risk Evaluation					
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
TT	5.09E-01	2.41E-06	4.73E-06	5.08E-02	2.40E-07

Table 5-1.17 Quantitative Screening
Comments: 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. However, for initial screening purposes 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. While a plant trip would not be expected due to fires in this area, a transient involving turbine trip (Initiator TT) has been conservatively assumed for this analysis. It should be noted that a reactor trip will occur on de-energization of RPS circuit protectors. Use of the turbine trip (required) logic is conservative in that reactor trip failure (i.e., ATWS) is considered. The potential failure of the RPS circuit protectors would not prevent manual reactor trip if required. This scenario is designated fire initiator F17 in the RISKMAN Model. **Conclusion:** 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 turbine trip and cause the loss of all plant equipment located in this fire area, regardless of fire severity or manual fire suppression.

Table 5-1.18 Quantitative Screening					
Fire Area	**18**	**Unit 2 Battery and Battery Board Room, Control Building 593' Elevation**			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
250VDC Battery 2	DH (added DHF to MFF)	ELECT12			
Battery Board 2	Subsumed in DH	ELECT12			
250VDC Battery Charger 2A & 2B	Subsumed in DH	ELECT12			
4kV Shutdown bus 3ED control power	A3ED	ELECT3			
Unit Preferred MMG Set 2 and Associated Equipment	NO IMPACT ON UNIT 1 MODEL		**Recovery Actions Impact:**		
Division 2 instrument power (Panels 9-82, 9-88)			250V DC Division II control power recovery	CPREC	ELECT3
RCIC steam flow indication					
HPCI steam flow indication					
CS/RHR interlock logic II					
RPS circuit protectors					
Manual relief valves 2-PCV-1-4, -18, 23, 41, 42	Not Modeled				
Risk Evaluation					
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
TT	5.09E-01	1.59E-07	3.13E-07	4.91E-02	1.54E-08

Table 5-1.18 Quantitative Screening
Comments: 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. However, for initial screening purposes 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. While a plant trip would not be expected due to fires in this area, a transient involving turbine trip with bypass available (Initiator TT) has been conservatively assumed for Unit 1 in this analysis. Note some Unit 2 equipment (RCIC/HPCI/CS/RHR) does not impact Unit 1 CDF. This fire scenario is designated F18 in the RISKMAN model. **Conclusion:** 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 turbine trip in Unit 1 and cause the loss of all plant equipment located in this fire area, regardless of fire severity or manual fire suppression.

Table 5-1.19
Quantitative Screening

Fire Area	19	Unit 3 Battery and Battery Board Room, Control Building 593' Elevation			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
250VDC Battery 3	DG	ELECT3			
Battery Board 3	Subsumed in DG	ELECT3			
250VDC Battery Charger 3	Subsumed in DG	ELECT3			
Unit Preferred MMG Set 3 and Associated Equipment	NO IMPACT ON UNIT 1 MODEL				
24V Neutron Monitoring Batteries and Chargers			**Recovery Actions Impact:**		
48V Annunciator Battery Charger B			250V DC Division II control power recovery	CPREC	ELECT3
Unit 3 RPS Circuit Protectors					
Risk Evaluation					
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$* F1)**
TT	5.09E-01	2.85E-07	5.59E-07	4.91E-02	2.74E-08

Comments: 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. However, for initial screening purposes 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. While a plant trip would not be expected due to fires in this area, a transient involving turbine trip (Initiator TT) has been conservatively assumed for Unit 1 in this analysis. This fire scenario is designated F19 in the RISKMAN model.

Conclusion: 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 turbine trip and cause the loss of all plant equipment located in this fire area, regardless of fire severity or manual fire suppression.

Table 5-1.20
Quantitative Screening

Fire Area	20	Unit 1 and 2 Diesel Generator Building			
	PSA MODEL IMPACTS DUE TO FIRE DAMAGE				
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
Diesel Generator A	GA	ELECT12			
Diesel Generator B	GB	ELECT12			
Diesel Generator C	GC	ELECT12			
Diesel Generator D	GD	ELECT12			
			Recovery Actions Impact:		
			Shutdown Bus Recovery	SDREC	ELECT3
		Risk Evaluation			
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
TT	5.09E-01	1.14E-07	2.23E-07	2.14E-01	4.77E-08

Comments: 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. For a fire in the Unit 1 and 2 Diesel Generator Building, it is unlikely that the operator would initiate plant trip on Unit 1, except in the case of severe, unsuppressed fires. While a plant trip would not be expected due to fires in this area, a transient involving turbine trip (Initiator TT) has been conservatively assumed for this analysis. It should be noted that diesels are only required following a consequential loss of offsite power, following an assumed reactor trip for Unit 1 following a fire in the Unit 1 and 2 Diesel Generator Building. This scenario is designated fire initiator F20 in the RISKMAN model.

Conclusion: 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 turbine trip and cause the loss of all plant equipment located in this fire area, regardless of fire severity or availability of automatic or manual fire suppression.

Table 5-1.21 Quantitative Screening					
Fire Area	**21**	**Unit 3 Diesel Generator Building**			
	PSA MODEL IMPACTS DUE TO FIRE DAMAGE				
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
Diesel Generator 3A	GE	ELECT3			
Diesel Generator 3B	GF	ELECT3			
Diesel Generator 3C	GG	ELECT3			
Diesel Generator 3D	GH	ELECT3	**Recovery Actions Impact:**		
4KV Shutdown Board 3EB control batteries, charger (SB-3EB)	DF	ELECT3	250V DC DIV II Control Power Recovery	CPREC	ELECT3
			Shutdown Bus Recovery	SDREC	ELECT3
Risk Evaluation					
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
TT	5.09E-01	2.73E-07	5.37E-07	2.13E-01	1.15E-07

Comments: 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). For a fire in the Unit 3 Diesel Generator Building, it is unlikely that the operator would initiate plant trip on Unit 1, except in the case of severe, unsuppressed fires. While a plant trip would not be expected due to fires in this area, a transient involving turbine trip (Initiator TT) has been conservatively assumed for this analysis. It should be noted that diesels are only required following a consequential loss of offsite power, following an assumed reactor trip for Unit 1 following a fire in the Unit 3 Diesel Generator Building. This scenario is designated F21 in the RISKMAN model.

Conclusion: *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 turbine trip and cause the loss of all plant equipment located in this fire area, regardless of fire severity or availability of automatic or manual fire suppression.

Table 5-1.22 Quantitative Screening					
Fire Area	22	4kV Shutdown Board Room 3EA and 3EB, 583' Elevation, Unit 3 DGB			
	PSA MODEL IMPACTS DUE TO FIRE DAMAGE				
Fire damaged Components (Direct Impacts)	Mitigating Systems Impact (Top Events)	Event Tree Impact	Fire Damaged Components (Indirect Impacts)	Mitigating Systems Impact (Top Events)	Event Tree Impact
4kV Shutdown Board 3EA	A3EA	ELECT3			
4kV Shutdown Board 3EB	A3EB	ELECT3			
Control cables for RHR service water pump A1	SW1A	MEUSPT			
Control cables for RHR service water pump A3	EA	MEUSPT			
Control cables for RHR service water pump C1	SW1C	MEUSPT			
Control cables for RHR service water pump C3	EC	MEUSPT			
Risk Evaluation					
Initiating Event	Initiating Event Frequency (IE)	CDF_{MS}	$CCDP_{MS}$ (CDF_{MS}/IE)	Ignition Frequency (F1)	CDF_{FIRE} ($CCDP_{MS}$*F1)
TT	5.09E-01	1.94E-07	3.82E-07	1.93E-02	7.35E-09

Comments: Plant walkdowns confirmed that there is no additional Unit 1 related support cables traverse through this area. The Failure Modes and Effects Analysis of Browns Ferry Unit 1 Key Support Systems (Table 2-4 Initiating Event notebook) states that a plant trip would not be expected to occur following loss of 4kV shutdown boards 3EA and 3EB individually. While a plant trip would not be expected due to fires in this area, a transient involving turbine trip (Initiator TT) has been conservatively assumed for this analysis. This fire scenario is designated initiator F22 in the RISKMAN model.

Conclusion: 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 turbine trip and cause the loss of all plant equipment located in this fire area, regardless of fire severity or availability of manual fire suppression.

Table 5-1.23
Quantitative Screening

Fire Area	23	4kV Shutdown Board Room 3EC and 3ED, 583' Elevation, Unit 3 DGB			
	PSA MODEL IMPACTS DUE TO FIRE DAMAGE				
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
4kV Shutdown Board 3EC	A3EC	ELECT3			
4kV Shutdown Board 3ED	A3ED	ELECT3			
Control cables for 4KV shutdown board 3EA	A3EA	ELECT3			
Risk Evaluation					
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
TT	5.09E-01	3.46E-07	5.62E-07	1.93E-02	1.08E-08

Comments: Plant walkdowns confirmed that there is no additional Unit 1 related support cables traverse through this area. The Failure Modes and Effects Analysis of Browns Ferry Unit 1 Key Support Systems (Table 2-4 Initiating Event notebook) states that a plant trip would not be expected to occur following loss of 4kV shutdown boards 3EC and 3ED individually. While a plant trip would not be expected due to fires in this area, a transient involving turbine trip (Initiator TT) has been conservatively assumed for this analysis. This scenario is designated F23 in the RISKMAN model.

Conclusion: 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 turbine trip and cause the loss of all plant equipment located in this fire area, regardless of fire severity or availability of manual fire suppression.

Table 5-1.24 Quantitative Screening					
Fire Area	24	4kV Bus Tie Board Room, 565' Elevation, Unit 3 Diesel Generator Building			
	PSA MODEL IMPACTS DUE TO FIRE DAMAGE				
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
4KV Bus Tie Board	Not Modeled				
		Risk Evaluation			
Initiating Event	**Initiating Event Frequency (IE)**	**CDF**	**CCDP (CDF/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} (CCDP*F1)**
LOSP	6.43E-03	2.17E-07	3.37E-05	1.92E-02	6.49E-07
		(From Plant Model)			

Comments: Plant walkdowns confirmed that there is no additional Unit 1 related equipment located in this area. Also, it was confirmed during these walkdowns that no additional Unit 1 related support cables traverse through 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 1. 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).

Conclusion: 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 loss of offsite power, regardless of fire severity or availability of manual fire suppression.

Table 5-1.25 Quantitative Screening					
Fire Compartments	25-1, 25-2, 25-3	Turbine Building, Pipe Tunnel, Intake Pump Station			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	Mitigating Systems Impact (Top Events)	Event Tree Impact	Fire Damaged Components (Indirect Impacts)	Mitigating Systems Impact (Top Events)	Event Tree Impact
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 probability of 1.0 for this level of evaluation. The compartments will be analyzed in more detail in Section 6.2.					
Risk Evaluation					
Initiating Event	Initiating Event Frequency (IE)	CDF_{MS}	$CCDP_{MS}$ (CDF_{MS}/IE)	Ignition Frequency (F1)	CDF_{FIRE} ($CCDP_{MS}$*F1)
			1.00E+00	7.77E-02	7.77E-02
			1.00E+00	1.09E-05	1.09E-05
			1.00E+00	5.59E-01	5.59E-01

5.2 Summary of Quantitative Screening

Table 5-2 summarizes the results for quantitative screening.

Table 5-2
Fire Induced CDF Summary

Fire Area/Zone	Description	Fire Area CDF
1-1	Unit 1 Reactor Building, 519' through 565' Elevations (West side of Torus Area and Main Floor)	6.08E-02
1-2	Unit 1 Reactor Building, 519' through 565' Elevations (East side of Torus Area and Main Floor)	3.52E-02
1-3	Unit 1 Reactor Building, 593' Elevation, North Side	2.32E-02
1-4	Unit 1 Reactor Building, 593' Elevation, South Side and RHR Heat Exchanger Rooms	2.12E-02
1-5	Unit 1 Reactor Building, 621' Elevation and North Side of 639' Elevations	4.88E-02
1-6	Unit 1 Reactor Building, South Side of 639' Elevation	3.08E-02
2	Unit 2 Reactor Building	1.27E-01
3	Unit 3 Reactor Building	1.26E-01
4	4kV Shutdown Board Room B (Unit 1 Reactor Building, 593' Elevation)	3.14E-04
5	4kV Shutdown Board Room A and 250V Battery Room (Unit 1 RB, EL 621')	1.99E-04
6	480V Shutdown Board Room 1A (Unit 1 Reactor Building, 621' Elevation)	1.11E-07
7	480V Shutdown Board Room 1B (Unit 1 Reactor Building, 621' Elevation)	2.01E-06
8	4kV Shutdown Board Room D (Unit 2 Reactor Building; 593' Elevation)	1.02E-07
9	4kV Shutdown Board Room C and 250V Battery Room (Unit 2 RB, EL 621')	1.38E-05
10	480V Shutdown Board Room 2A (Unit 2 Reactor Building, 621' Elevation)	1.15E-08
11	480V Shutdown Board Room 2B (Unit 2 Reactor Building, 621' Elevation)	4.63E-09
12	Shutdown Board Room F (Unit 3 Reactor Building, 593' Elevation)	1.15E-08
13	Shutdown Board Room E (Unit 3 Reactor Building, 621' Elevation)	5.71E-09
14	480V Shutdown Board Room 3A (Unit 3 Reactor Building, 621' Elevation)	5.11E-09
15	480V Shutdown Board Room 3B (Unit 3 Reactor Building, 621' Elevation)	5.45E-09
16-1	Control Building - 593' Elevation	3.78E-02

Table 5-2
Fire Induced CDF Summary

Fire Area/Zone	Description	Fire Area CDF
16-2	Control Building - 606' (Cable Spreading Room)	1.20E-02
16-3	Control Building - 617' (Control Room)	6.92E-02
17	Unit 1 Battery and Battery Board Room, Control Building 593' Elevation	2.40E-07
18	Unit 2 Battery and Battery Board Room, Control Building 593' Elevation	1.54E-08
19	Unit 3 Battery and Battery Board Room, Control Building 593' Elevation	2.74E-08
20	Unit 1 and 2 Diesel Generator Building	4.77E-08
21	Unit 3 Diesel Generator Building	1.15E-07
22	4kV Shutdown Board Room 3EA and 3EB, 583' Elevation, Unit 3 Diesel Generator Building	7.35E-09
23	4kV Shutdown Board Room 3EC and 3ED, 583' Elevation, Unit 3 Diesel Generator Building	1.08E-08
24	4kV Bus Tie Board Room, 565' Elevation, Unit 3 Diesel Generator Building	6.49E-07
25-1	Intake Pump Station	7.77E-02
25-2	Pipe Tunnel	1.09E-05
25-3	Turbine Building	5.59E-01
	Total	**1.23E+00**

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 5, Section 4, Steps 4.2 and 9.1) 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. Of the areas screened from further consideration, this condition potentially applies to five areas listed in Table 5-2 with fire-related core damage frequencies between 1E-06 and 1E-07 (i.e., fire area 6, 8, 17, 21 and 24).

The following areas were screened with a CDF above 1E-7:

6	480V Shutdown Board Room 1A	1.11E-7
8	4kV Shutdown Board Room D	1.02E-7
17	Unit 1 Battery and Battery Board Room	2.40E-7
21	Unit 3 Diesel Generator Building	1.15E-7
24	4kV Bus Tie Board Room	6.49E-7

The fire induced LERF values for these evaluations are calculated in Table 5-3:

Table 5-3 Containment Bypass Scenarios						
Fire Area	**Initiating Event**	**IE Frequency (a)**	**LERF (b)**	**CLERP (c = b/a)**	**Ignition Frequency (d)**	**Fire-Related LERF (c x d)**
6	IMSIV	5.70E-02	5.50E-09	9.65E-08	1.92E-02	1.86E-09
8	TT	5.09E-01	8.19E-10	1.61E-09	1.92E-02	3.09E-11
17	TT	5.09E-01	9.31E-08	1.83E-07	5.08E-02	9.29E-09
21	TT	5.09E-01	7.85E-11	1.54E-10	2.13E-01	3.29E-11
24	LOSP	6.43E-03	9.32E-10	1.45E-07	1.92E-02	2.79E-09

Since each of these areas has a fire-related LERF that is below the cutoff of 1E-7, it can be concluded that these fires do not result in or cause containment breach concerns beyond those already addressed in the plant risk model. Therefore, these areas can continue to be screened from further consideration, as shown in Table 5-2.

6.0 PHASE II.3 - DETAILED AREA ANALYSIS

The areas that were not screened out in Section 5 will be further evaluated in detail in this section. Table 6-1 lists all the unscreened areas.

Table 6-1
Unscreened Areas

Fire Area / Zone	Description	Fire Area CDF (From Initial Screening)
1-1	Unit 1 Reactor Building, 519' through 565' Elevations (West side of Torus Area and Main Floor)	6.08E-02
1-2	Unit 1 Reactor Building, 519' through 565' Elevations (East side of Torus Area and Main Floor)	3.52E-02
1-3	Unit 1 Reactor Building, 593' Elevation, North Side	2.32E-02
1-4	Unit 1 Reactor Building, 593' Elevation, South Side and RHR Heat Exchanger Rooms	2.12E-02
1-5	Unit 1 Reactor Building, 621' Elevation and North Side of 639' Elevations	4.88E-02
1-6	Unit 1 Reactor Building, South Side of 639' Elevation	3.08E-02
2	Unit 2 Reactor Building	1.27E-01
3	Unit 3 Reactor Building	1.26E-01
4	4kV Shutdown Board Room B (Unit 1 Reactor Building, 593' Elevation)	3.14E-04
5	4kV Shutdown Board Room A and 250V Battery Room (Unit 1 RB, EL 621')	1.99E-04
7	480V Shutdown Board Room 1B (Unit 1 Reactor Building, 621' Elevation)	2.01E-06
9	4kV Shutdown Board Room C and 250V Battery Room (Unit 2 RB, EL 621')	1.38E-05
16-1	Control Building - 593' Elevation	3.78E-02
16-2	Control Building - 606' (Cable Spreading Room)	1.20E-02
16-3	Control Building - 617' (Control Room)	6.92E-02
25-1	Intake Pump Station	7.77E-02
25-2	Pipe Tunnel	1.09E-05
25-3	Turbine Building	5.59E-01

6.1 Review of the EPRI Fire Events Database (FEDB-2001) - Fire Severity Factors

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 review of the fire events database was performed. This evaluation segments the fire ignition frequency into "minor" and "severe" cases based on severity factors. A fire severity factor is defined as the fraction of the incipient fires that result in a fully developed fire. The data source used to generate this information is the fire events database developed by EPRI FEDB-2001 (Reference 3).

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 1003111 (This is an update of the earlier NSAC/178L data base). This version of the EPRI database contains 1885 fire events that occurred in PWRs and BWRs between 1968 and Dec. 2000. It contains fire events from NSAC-178L, NEIL, Sandia National lab, NRC SECY-93-143, INPO and direct solicitation from utilities. The ignition frequency model (Attachment B) used a total of 1430 fire events out of 1885. This represents approximately 2369 reactor years of operation in the U.S. commercial industry.

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 or portable extinguishers, etc.). 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 1885 fire events are documented in FEDB-2001. Of these database entries, 903 have specific entries describing the equipment that was used to suppress the fire. This information appears in the "EQUIP_USED" data field. These 903 fire events can be divided by fire ignition source and separated into categories (i.e., hose stream or installed system) as shown in Table 6-1 (a). The ignition source grouping similar to the IPEEE FIVE methodology was taken from "INIT_TAB12". The remaining 982 fire entries had no information regarding means of suppression. The time taken to suppress the fire once suppression personnel or equipment responded (SUPP_TIME), agent used (AGENT_USED) and the dollar value loss incurred due to the fire (DIRECTLOSS) information was reviewed to determine the severity of fire. A total of 66 fires (from the list of 982 fires) had suppression times 15 minutes or greater and had a dollar loss value of $20,000 or greater. These fires will be considered significant.

As shown in Table 6-1 (a), a total of 158 fires were suppressed by automatic sprinkler/deluge/gas/dry systems or hose streams. These fires along with the 66 fires described above will be considered "severe". In other words, (224/1885 =) 11.9% of fires occurred in nuclear plants are judged to be severe.

Ignition sources present in Fire Area 2 include electric cabinets, pumps and miscellaneous plant wide components. This represents a total of 187 fires (includes reactor and aux. buildings) per Table 6-1 (b). Of these, 7 fires will be considered severe. Additionally assume (66/982 =) 6.7% of the fires under the "information unavailable" category will be severe, or (0.067 x 102 =) 7 fires will be severe. In other words, (14/187 =) 7.5% of fires occurred in this area are judged to be severe.

Ignition sources present in Fire Areas 5 and 9 are primarily electric cabinets* and batteries*. This represents a total of 219 fires. Of these, 10 fires will be considered severe per Table 6-1 (a). Additionally assume (29/322 =) 9% of the fires under the "information unavailable" category will be severe, or (0.09 x 69 =) 6 fires will be severe. In other words, (16/219 =) 7.3% of fires occurred in these areas are judged to be severe.

Ignition sources present in Fire Areas 4 and 7 are primarily electric cabinets*. This represents a total of 290 fires. Of these, 9 fires will be considered severe per Table 6-1 (a). Additionally assume 6.7% of the fires under the "information unavailable" category will be severe, or (0.067 x 157 =) 11 fires will be severe. In other words, (20/290 =) 6.9% of fires occurred in these areas are judged to be severe.

Ignition frequency in fire compartments 16-1 and 16-3 (control building including control rooms) is primarily due to electric cabinets. The appropriate category from Table 6-1 (a) is the Electrical Cabinet (Panel/RPS). This represents a total of 123 fires. Of these, 2 fires will be considered severe per Table 6-1 (a). Additionally assume 6.7% of the fires under the "information unavailable" category will be severe, or (0.067 x 13 =) 4 fire will be severe. In other words, (6/123 =) 4.9% of fires occurred in this area are judged to be severe.

Ignition frequency in fire compartment 16-2 (cable spreading room) is primarily due to cables and non-qualified junction boxes (See Attachment B). This represents a total of 39 fires. Of these, 3 severe fires were listed in this category of ignition sources. Additionally assume 6.7% of the fires under the "information unavailable" category will be severe, or (0.067 x 23 =) 2 will be severe. In other words, (5/39 =) 12.8% of fires occurred in this area are judged to be severe. It should be noted that the FEDB-2001 lists Browns Ferry CSR fire in the transient category (entry 634, INIT_TAB12). Therefore, considering this category, a total of 242 fires are listed. Of these, 20 fires will be considered severe per Table 6-1 (a). Additionally assume 6.7% of the fires under the "information unavailable" category will be severe, or (0.067 x 147=) 10 fires will be severe. In other words, (30/242 =) 12% of fires occurred in this area are judged to be severe. Therefore, as calculated above, 12.8% of fires occurred in this area are judged to be severe.

Ignition sources present in fire compartment 25-1 (intake pump station) include electric cabinets, fire pumps, cables, non-qualified junction boxes, transformers and transients (See Appendix B). For electrical cabinets, using the electrical cabinet/MCC category a total of (86 + 10 + 36 + 3 + 40 + 242 =) 417 fires. Of these, 33 fires will be considered severe per Table 6-1 (a). Additionally assume 6.7% of the fires under the "information unavailable" category will be severe, or (0.067 x 239 =) 16 fires will be severe. In other words, (49/417 =) 11.8% of fires occurred in this area are judged to be severe.

No significant fires were noted for fire compartment 25-2 (Pipe Tunnel).

Fire compartment 25-3 (turbine building) includes most of the ignition sources listed in Table 6-1 (a). Therefore, 11.9% of the fires in this area (as calculated above) will be considered severe.

Based on the above calculated fire severity factors, the ignition frequency assigned to minor and severe fire cases is give in Table 6-1 (c).

**Transient fires involving welding were not considered in fire severity determination. Most of these fires were extinguished by portable extinguishers and were therefore, considered minor.*

Table 6-1 (a)
Means of Suppression

Ignition Source Category	No. of Fires	Automatic Deluge/ Wet pipe/ Gas/Dry pipe Systems	Hose Stream	Other Means (Portable Extinguisher, Fuel Source Removed, Self Extinguish, etc.)	Information Unavailable	None
Air Compressors	15	1		9	4	1
Batteries	9			5	2	2
Battery Chargers	13	1		5	6	1
Boiler	3				2	1
Cable Fire	36	1	2	6	23	4
Diesel Generators	93	7	5	24	41	16
Dryers	13		3	7	2	1
Electrical Cabinets (Panel / RPS)	123	2		34	63	24
Electrical Cabinets (Switchgear)	33		1	6	14	12
Electrical Cabinets / (Transformer)	48			11	30	7
Electrical Cabinets / MCC	86	4	2	18	50	12
Elevator Motor	10		1	3	3	3
Fire Protection Panels	3			1	1	1
Fire Pump	10		1	2	6	1
Gas Turbine	4	1		2		1
Hydrogen Tank	5		1		2	2
Junction Box (Non-Qualified Cable)	3			1		2
Junction Box (Qualified Cable)	7			4	3	

Table 6-1 (a)
Means of Suppression

Ignition Source Category	No. of Fires	Automatic Deluge/ Wet pipe/ Gas/Dry pipe Systems	Hose Stream	Other Means (Portable Extinguisher, Fuel Source Removed, Self Extinguish, etc.)	Information Unavailable	None
Main Feedwater Pump	21	1	5	8	8	
Miscellaneous Component	294	3	27	72	166	26
None Identified	2				2	
Off-Gas / H2 Recombiner (BWR)	43	3		1	11	28
Off-gas/H2 Recombiner (BWR)	1				1	
Other (Hydrogen Fire)	12			2	9	1
Other Pumps / RCP	116	2	6	26	63	20
RPS MG Set	17	1		5	7	4
T/G Exciter	9			3	4	1
T/G Hydrogen	15	6	1	2	5	1
T/G Oil	22		4	9	9	
Transformer	40	3	1	11	19	6
Transient fires caused by welding and cutting	18			1	17	
Transients	242	3	17	64	147	11
Ventilation Subsystem	44	1	1	13	25	4
Welding During Construction	403	2	17	163	197	24
Yard transformers	72	14	7	7	40	6
Total	**1885**	**56**	**102**	**525**	**982**	**223**

Table 6-1 (b) PWR-Auxiliary Building						
Ignition Source	**Number of Fires**	**Auto Deluge**	**Hose Stream**	**Other**	**Information Unavailable**	**None**
Air Compressors	1			1		
Battery Chargers	2			1		1
Cable Fire	8	1	1		6	
Dryers	1			1		
Electrical Cabinets (Panel / RPS)	11			6	5	
Electrical Cabinets (Switchgear)	4			2	1	1
Electrical Cabinets / (Transformer)	3				3	
Electrical Cabinets / MCC	7			1	4	2
Fire Protection Panels	2				1	1
Miscellaneous Component	5			1	2	2
Other (Hydrogen Fire)	4			2	1	1
Other Pumps / RCP	10		1	1	4	4
RPS MG Set	4			2	1	1
Transformer	4			1	1	2
Transient fires caused by welding and cutting	1				1	
Transients	20		1	4	15	
Ventilation Subsystem	4			2	2	
Welding During Construction	37		2	9	24	2
TOTAL	**128**	**1**	**5**	**34**	**71**	**17**
Cable Fire	1				1	
Electrical Cabinets (Panel / RPS)	3				3	
Electrical Cabinets / (Transformer)	4			1	2	1
Electrical Cabinets / MCC	7			3	2	2
Miscellaneous Component	1				1	
Off-Gas / H2 Recombiner (BWR)	4				1	3
Other Pumps / RCP	8			1	5	2
RPS MG Set	3				2	1
Transformer	1					1
Transients	8	1		2	5	
Ventilation Subsystem	1			1		
Welding During Construction	18			8	9	1
Total	**59**	**1**	**0**	**16**	**31**	**11**

Table 6-1 (c)
Fire Area Frequency Based on Fire Severity

Fire Area / Compartment	Description	Ignition Frequency (a)	Severe fire factor (b)	Minor Fire Factor (c)	Severe Fire Case Frequency (a*b)	Minor Fire Case Frequency (a*c)
2	Unit 2 Reactor Building	1.27E-01	0.075	0.925	9.54E-03	1.18E-01
3	Unit 3 Reactor Building	1.26E-01	N/A	N/A	N/A	N/A
4	4kV Shutdown Board Room B (Unit 1 Reactor Building, 593' Elevation)	1.94E-02	0.069	0.931	1.34E-03	1.81E-02
5	4kV Shutdown Board Room A and 250V Battery Room (Unit 1 RB, EL 621')	2.36E-02	0.073	0.927	1.72E-03	2.19E-02
7	480V Shutdown Board Room 1B (Unit 1 Reactor Building, 621' Elevation)	1.92E-02	0.069	0.931	1.33E-03	1.79E-02
9	4kV Shutdown Board Room C and 250V Battery Room (Unit 2 RB, EL 621')	2.26E-02	0.073	0.927	1.65E-03	2.09E-02
16-1	Control Building - 593' Elevation (8 Rooms)	3.78E-02	0.049	0.951	1.85E-03	3.59E-02
	Two Rooms on El 593 (Unit 1 Aux Inst. Room and Unit 1/2 Computer Room)	9.44E-03	0.077	0.923	7.27E-04	8.71E-03
	Six Rooms on El 593 (Unit 2 and 3 Aux Inst Rm, Unit 3 Comp. Rm, Mech. equip room, Process computer room and Communication room)	2.83E-02	N/A	N/A	N/A	N/A
16-2	Control Building - 606' (Cable Spreading Room)	1.20E-02	0.128	0.872	1.54E-03	1.05E-02
16-3	Control Building - 617' (Control Room)	6.92E-02	0.049	0.951	3.39E-03	6.58E-02
25-1	Intake Pump Station	7.77E-02	0.118	0.882	9.17E-03	6.86E-02
25-2	Pipe Tunnel	1.09E-05	0	1	0.00E+00	1.09E-05
25-3	Turbine Building	5.59E-01	0.119	0.881	6.66E-02	4.93E-01

6.2 Detailed fire Area/Compartment Evaluation

Due to the wide variance in fire sources, potentially damaged targets and area geometry, the methods used to evaluate each area will also vary. Fire is generally assumed to start in the most significant component in the area (i.e., electrical cabinet, transformer, pump, etc.). For minor fires, the fire is limited to that component. However, the component is considered disabled and no recovery is allowed. Credit for alternate power source to a component located outside the fire area may be taken. Severe fire cases were generally considered engulfing, unless the area is very large or involves multiple rooms (i.e., reactor building, turbine building, control building, etc.). The engulfing fire case treatment is considered similar to the evaluation in Section 5.

6.2.1 Unit 1 Reactor Building (Fire Area 1)

The Unit 1 Reactor Building consists of an extremely large volume, with individual fire zones of about 10,000 ft^2 on each of 5 major elevations. A detailed fire hazards analysis of all significant fire sources will be performed for this area. This may involve determination of estimated heat release rates, fire severity and propagation, potential target damage, effectiveness of suppression and detection, etc. 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 5. These fire sources were analyzed in Attachment C and are summarized in Table 6-2.1 (a).

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 loading 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 1 operator would trip the reactor due to a fire in the primary containment Hydrogen/Oxygen analyzer.

It may be noted that several of the fire ignition sources identified in Attachment B are not listed in Table 6-2.1 (a) 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.

Table 6-2.1 (b) lists all of the identified potential fire sources and assigns the ignition frequency for each source. The fire source is first associated with the corresponding component category from Attachment B and the ignition frequency is determined. For example, 250V RMOV Board 1C corresponds to electric cabinets category in fire zone 1-1 with an ignition frequency of 5.18E-2. This ignition frequency is divided by the number of similar components in the area that are analyzed as potentially significant fire sources.

Electrical components and raceways within the zone of influence (from Attachment C) of potential fire sources have been evaluated, based on the walkdown information (Reference 26). If this evaluation confirms that potential damage to the specific component and any other components within the zone of influence would not result in an automatic plant trip and that these components are not required for the safe shutdown of the plant and are not included in the PRA equipment list, then these components (or fire sources) will be screened from further consideration.

New cable trays and conduits impacts are included for the Unit 1 Reactor Building fire analysis. New cable trays were installed at the time of walkdown, hence all new cable trays in the zone of influence were recorded and modeled. Some new conduits have not been installed at the time of walkdown, these were identified by reviewing conduit and grounding drawings that were marked up for changes (References 37 through 41), Each fire ignition source equipment was superimposed to the conduit drawings based on its column and row information, and it is conservatively assumed that all the conduits that are adjacent to the equipment is damaged (regardless of the height of conduits). These impacted conduits were documented in a walkdown summary spreadsheet file (Reference 42). In general, these newly identified conduits have insignificant PRA impact.

Tables 6-2.1(c)(1) through 6-2.1(c)(22) provide detailed discussion of each of the potential fixed fire sources in the Unit 1 Reactor Building (these tables are placed after the main text of Section 6). 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. Fire induced CDF due to each of the identified fire sources is summarized in Table 6-2.1 (c).

Table 6-2.1 (a)
Unit 1 Reactor Building Fixed Fire Sources

Fire Zone	Fire Source
1-1	480 RMOV Board 1C
	480 RB Vent Board 1B
	250V RMOV Board 1C
	Core Spray Pumps 1A and 1C
	RHR Pumps 1A and 1C
	RCIC Pump
	HPCI Pump
	1-LPNL-25-340 ES Div I and II Panel
1-2	Core Spray Pumps 1B and 1D
	RHR Pumps 1B and 1D
1-3	RCW Pump 1A
1-4	All Fixed Fire Sources Equipment Screened (See Attachment C)
1-5	240V Lighting Board 1A
	240V Lighting Transformer TL1A
	4kV-480V Transformer TS1A
	4kV-80V Transformer TS1B
	4160V RPT Board 1-1 (Panel 1 and Panel 2)
	4160V RPT Board 1-2 (Panel 1 and Panel 2)
	RCIC Backup Control Panel 1-25-31
	Panel 25-3 (Filter Demin)
1-6	4kV-480V Shutdown Boards Emergency Transformer TS1E (Oil)
	VFD 1A (Panel)
	VFD 1B (Panel)

Table 6-2.1 (b) Ignition Frequency Assignment of Individual Fire Sources				
Fire Source Number	Description	Ignition Frequency Due to Similar Fire Sources (App. B) (a)	Similar Significant Fire Sources in Zone (b)	Ignition Frequency per Fire Source (a/b) * # of affected sources
1-1-1	480V RMOV Board 1C	2.07E-02	4	5.18E-03
1-1-2	480V RB Vent Board 1B	2.07E-02	4	5.18E-03
1-1-3	250 RMOV Board 1C	2.07E-02	4	5.18E-03
1-1-4	Core Spray Pumps 1A and 1C	1.91E-02	6	6.37E-03
1-1-5	RHR Pumps 1A and 1C	1.91E-02	6	6.37E-03
1-1-6	RCIC Pump	1.91E-02	6	3.18E-03
1-1-7	HPCI Pump	1.91E-02	6	3.18E-03
1-1-8	1-LPNL-925-0340 ES Div I and II Panel	2.07E-02	4	5.18E-03
1-2-1	Core Spray Pumps 1B and 1D	1.27E-02	4	6.35E-03
1-2-2	RHR Pumps 1B and 1D	1.27E-02	4	6.35E-03
1-3-1	RCW Pump 1A	3.18E-03	1	3.18E-03
1-5-1	240V Lighting Board 1A	2.59E-02	5	5.18E-03
1-5-2	240V Lighting Transformer TL1A	2.63E-03	3	8.77E-04
1-5-3	4kV-480V Transformer TS1A	2.63E-03	3	8.77E-04
1-5-4	4kV-480V Transformer TS1B	2.63E-03	3	8.77E-04
1-5-5	4160V RPT Board 1-1 (Panel 1 and Panel 2)	2.59E-02	5	5.18E-03
1-5-6	4160V RPT Board 1-2 (Panel 1 and Panel 2)	2.59E-02	5	5.18E-03
1-5-7	RCIC Control Panel 1-25-31	2.59E-02	5	5.18E-03
1-5-8	Panel 25-3 (Filter Demin)	2.59E-02	5	5.18E-03
1-6-1	4kV-480V Emergency Transformer TS1E (Oil)	8.75E-04	1	8.75E-04
1-6-2	VFD 1A (Panel)	1.04E-02	2	5.20E-03
1-6-3	VFD 1B (Panel)	1.04E-02	2	5.20E-03

Table 6-2.1 (c) Unit 1 Reactor Building Fire Sources		
Fire Source Number	Description	Fire Induced CDF
1-1-1	480V RMOV Board 1C	3.55E-08
1-1-2	480V RB Vent Board 1B	1.42E-09
1-1-3	250 RMOV Board 1C	1.45E-09
1-1-4	Core Spray Pumps 1A and 1C	1.05E-08
1-1-5	RHR Pumps 1A and 1C	1.29E-07
1-1-6	RCIC Pump	5.27E-09
1-1-7	HPCI Pump	2.05E-08
1-1-8	1-LPNL-925-0340 ES Div I Panel	7.03E-08
1-2-1	Core Spray Pumps 1B and 1D	1.75E-09
1-2-2	RHR Pumps 1B and 1D	3.58E-08
1-3-1	RVW Pump 1A	9.15E-10
1-5-1	240V Lighting Board 1A	4.92E-09
1-5-2	240V Lighting Transformer TL1A	5.76E-07
1-5-3	4kV to 480V Transformer TS1A	2.58E-09
1-5-4	4kV to 480V Transformer TS1B	5.71E-10
1-5-5	4160V RPT Board 1-1 (Panel 1 and Panel 2)	7.59E-08
1-5-6	4160V RPT Board 1-2 (Panel 1 and Panel 2)	1.89E-09
1-5-7	RCIC Control Panel 1-25-31	4.92E-09
1-5-8	Panel 25-3 (Filter Demin)	0.00E+00
1-6-1	4kV to 480V Emergency Transformer TS1E (Oil)	2.88E-08
1-6-2	VFD 1A (Panel)	1.70E-07
1-6-3	VFD 1B (Panel)	1.70E-07
	Transient Sources	8.27E-09
	Unqualified Cable	1.50E-07

Consideration of Non-Qualified Cables as Independent Fire Sources

The above discussion of fixed combustible sources did not include Non-Qualified cables. The 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. References 24 and 27 (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 3 million BTU. On a frequency basis, the potential for ignition of a fire in a section of cable of this size is

  F1 = 4.4E-03 x 3/2,328 = **5.7E-06**

  Where 4.4E-03 represents the total ignition frequency for non-qualified cables in the Unit 1 Reactor Building (see Attachment B); 3/2,328 Represents the ratio of a given cable tray segment heat capacity to the total BTU loading due to unqualified cables for the Unit 1 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).

- 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 = 4.4E-03 x 3.37E-05 = **1.5E-07**

  Where

  4.4E-03 represents the total fire frequency for unqualified cables in the Unit 1 Reactor Building

  3.37E-05 is the conditional core damage probability (CCDP) 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.

6.2.2 Unit 2 Reactor Building (Fire Area 2)

A bounding case evaluation of fires in Unit 2 and its impact on Unit 1 core damage frequency will be considered (see Table 6-2.2 and Table 6-2.2(a)).

6.2.3 Unit 3 Reactor Building (Fire Area 3)

The Unit 3 Reactor Building is the adjoining operational unit. Potential of fire in this area and its impact on the Unit 3 core damage frequency is evaluated separately. However, a bounding case evaluation of fires in Unit 3 and its impact on Unit 1 core damage frequency will be considered. The evaluation of this area is described in Table 6-2.3

6.2.4 Fire Area 4

Fires in 4kV Shutdown Board Room B was evaluated by first using fire severity cases (severe and minor fires) to identify fire impacts and subsequently by taking credit for manual suppression. 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 this area is described in Table 6-2.4 and Table 6-2.4 (a).

6.2.5 Fire Area 5

Fires in 4kV Shutdown Board Room A was evaluated by first using fire severity cases (severe and minor fires) to identify fire impacts and subsequently by taking credit for manual suppression. 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 this area is described in Tables 6-2.5 and 6-2.5 (a).

6.2.6 480V Shutdown Board Room 1B (Fire Area 7)

Fire Area 7 is screened based on the fact that administrative control would make transient fire in this area non-credible. The total cabinet fire frequency was used with the engulfing fire scenario (i.e., all cables and equipment in this area were assumed damaged) to estimate the fire induced CDF. Table 6-2.6 is referred to for more details.

6.2.7 4kV Shutdown Board Room C (Fire Area 9)

Fires in 4kV Shutdown Board Room C was evaluated by first using fire severity cases (severe and minor fires) to identify fire impacts and subsequently by taking credit for manual suppression. This was required due to the conservative nature of the initial evaluation, which assumed that all fires result in loss of 4kV Shutdown Bus 2. The evaluation of this area is described in Tables 6-2.7 and 6-2.7 (a).

6.2.8 Control Building (Fire Area 16)

Detailed evaluation of fire propagation scenarios was performed for each of the three compartments in the Control Building. This involved probabilistic as well as deterministic evaluations with consideration for fire severity and time for detection and suppression prior to damage. Control rooms were evaluated using bounding case evaluations as well as specific scenario involving critical cabinets. The evaluation of each of these areas is described as follows:

6.2.8.1 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.

El 593 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 within 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. This area has dimensions of approximately 1,000 sq. ft. The following risk-significant panels are located in this room:

1-9-18	Feedwater
1-9-29	Feedwater
1-9-30	Safety Relief Valves
1-9-32	Division I (A and C) RHR, Core Spray and HPCI
1-9-33	Division II (B and D) RHR, Core Spray and HPCI
1-9-39	HPCI Relay Auxiliary Panel
1-9-42	MSIV
1-9-43	MSIV Closure
1-9-48	Feedwater
1-9-49	Feedwater

1-9-50 Feedwater

- 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 1 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. There are no Unit 1 safe shutdown components in this room.

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 1 safe shutdown components in these areas and Unit 1 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 following fire scenarios have been evaluated for this compartment:

Case 1: A minor fire in Unit 1 Auxiliary Instrument Room or Unit 1 Computer Room resulting in loss of all Feedwater.

Case 2: A severe fire in Unit 1 Auxiliary Instrument Room or Unit 1 Computer Room resulting in loss of all Feedwater, MSIV Closure and HPCI failure.

Case 3: A fire in any other area of Compartment 16-1 does not result in plant trip.

The fire scenario is graphically shown below in an event tree format.

IF=3.78E-02	CASE 1	Minor Fire (TLFW)	Aux. Inst, Rm. or Computer Rm. (=8.71E-03)
	CASE 2	Severe Fire (IMSIV)	Aux. Inst, Rm. or Computer Rm. (=7.27E-04)
	CASE 3	Screened	All Other Areas (=2.83E-02)
		Total =	3.78E-02

See Table 6-2.8.1 for detailed evaluation of the above cases.

Fire Hazards Evaluation

A fire scenario involving an electrical cabinet in the Auxiliary Instrument Room was postulated (Computer room will be similar). Smoke detector response time was calculated based on peak heat release rate (HRR) of 190 Btu/sec. However, calculation was also done at lower HRR to conservatively determine the smoke detector response time and also make sure that there will be no smoke stratification. The calculations are shown in Table 6-2.8.1.

The following table depicts the results of the calculation. Note that in most cases the fire brigade will be at the location well before fire spread to an adjacent cabinet (based on review of fire drills).

Heat Release Rate Btu/sec	Smoke Detector Activation Time Sec	Fire Spread to Adjacent Cabinet Min	Fire Brigade Manual Response Min
50	24	15	5-10
100	8	15	5-10
190	3	15	5-10

The above analysis shows that the time to detection and the time taken for manual response will limit the fire damage to the cabinet of origin.

6.2.8.2 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 (FEDB-1999) shows that there have been six Cable Spreading Room fires in the commercial nuclear industry in addition to the BFN fire. Five of these fires occurred inside electrical cabinets, and one in the RPS MG set. Two of the fire appear to be "severe" in nature as automatic extinguishing systems were used. Due to the sparse nature of this data (7 entries in approximately 3238 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.

The following two cases have been evaluated:

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.

See Table 6-2.8.2 for details of risk quantifications.

The fire scenario is graphically shown below in an event tree format.

IF=1.20E-02	CASE 1	Minor Fire (TLFW)	Single cable tray, Loss of feedwater (=1.05E-02)
	CASE 2	Severe Fire (IMSIV)	Two cable trays, MSIV closure, HPCI/RCIC/CS failure (=1.54E-03)
		Total =	1.20E-02

6.2.8.3 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 1 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 1-9-8, to the left of the main core map area. To the right of the core map area is panel 1-9-3, 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 3.3.1

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

The evaluation of fires in the Control Room area is based on the guidance given in Appendix M of the EPRI Fire Risk Analysis Implementation Guide (Reference 5), Guidance for Development of Response to Generic Request for Additional Information for Fire IPEEE, TR-105928 March 2000 (Reference 6) and Technical Review of Risk-Informed Performance Based Methods for Nuclear Plants Fire Protection Analyses NUREG-1521-1998 (Reference 29).

As noted above, the most significant Control Room panel, from the aspect of potential impact on plant operation, is panel 1-9-3. 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 1-9-3. 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.

Various control room fire scenarios will be postulated and analyzed in this section. The input data and factors used in these evaluations and associated references are provided as follows:

Evaluation of Control Room Fires			
Factor/Term	**Description**	**Value**	**Remarks/Reference**
$IF_{CABINET}$	Frequency of fire in control room electric cabinets.	4.8E-2	Attachment B. Electric Cabinet fire frequency for one control room.
IF_{CR}	Frequency of fire in unit 1 control room	0.016	Attachment B. Total fire frequency divided by 3.
P_{NS}	Probability of non-suppression	0.0034	EPRI TR-105928, Appendix M figure M-1 (Lognormal distribution, mean value)
P_{SMOKE}	Probability that smoke will force abandonment of the control room given a fire	0.1	NUREG 1521, Table B-4
F_{RSP}	Failure of remote shutdown capability	0.064	NUREG 1521, Table B-4 Human error rate for successfully performing the control room abandonment procedure.
A_{RATIO}	Area ratio of panel 2-9-3 to total cabinet area within the unit 1 control room	0.15	From drawing 0-47600-263, approximated the area of panel 1-9-3 to the total cabinet area in the unit 1 control room.

Control Room Fire Scenario 1: The first scenario postulates a bounding case, where any fire originating in the unit 1 control room area that is unsuppressed will result in Control Room abandonment even for fires in non-critical panels.

$CDF_{Fire\ Induced}$ =	**IF_{CR} x P_{NS} x F_{RSP}** (Reference NUREG 1521)
	= 0.016 x 0.0034 x 0.064 **= 3.4E-6**

Control Room Fire Scenario 2: This scenario postulates a fire starting in critical panel 1-9-3 and subsequent smoke release forcing abandonment of the control room.

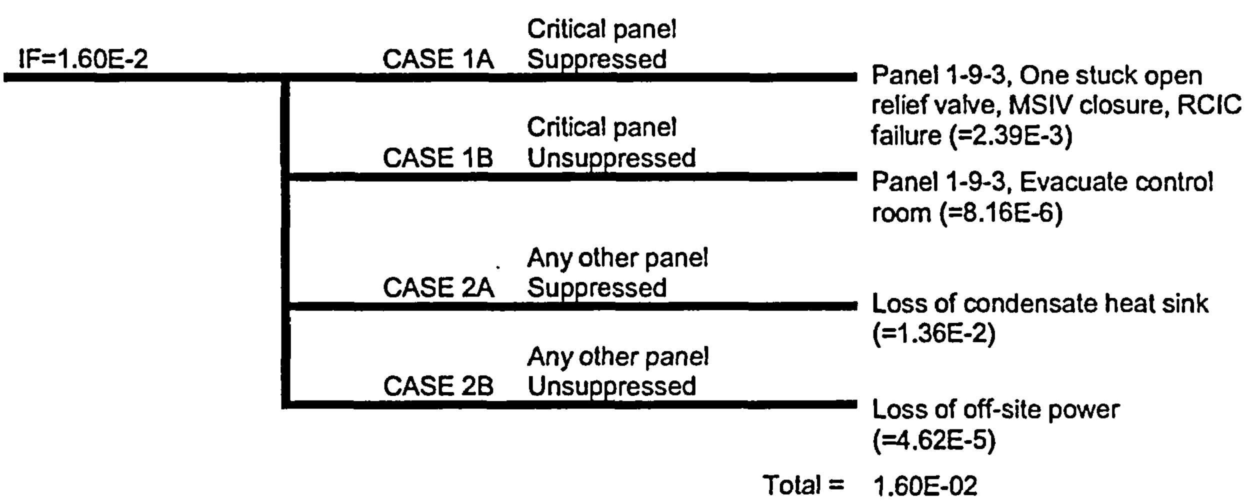
$CDF_{Fire\ Induced}$ =	**IF_{CR} x A_{RATIO} x F_{RSP} x P_{SMOKE}** (Reference NUREG 1521)
	= 0.016 x 0.15 x 0.064 x 0.1 **= 1.53E-5**

Control Room Fire Scenario 3: This scenario postulates a fire starting in any cabinet other than panel 1-9-3 and subsequent smoke release forcing abandonment of the control room. Credit is given for the RCIC system automatically cycling to control reactor level. Therefore, the RCIC system must randomly fail which adds the Q_{RCIC} term in the equation. For Browns Ferry RCIC failure has a nominal value of 0.059 (split fraction RCI1 in the plant model).

$CDF_{Fire\ Induced}$ =	$IF_{CR} \times (1-A_{RATIO}) \times F_{RSP} \times Q_{RCIC} \times P_{SMOKE}$ (Reference NUREG 1521)
	= 0.016 x (1-0.15) x 0.064 x 0.059 x 0.1 = 5.2E-6

Control Room Fire Scenario 4: This fire scenario evaluates fires in critical and non-critical panels. Even suppressed fires in critical panels can lead to significant damage. Suppressed fire in panel 1-9-3 is conservatively assumed to result in one stuck open relief valve, MSIV closure and RCIC failure (CCDP=4.95E-5). Unsuppressed fire in 1-9-3 leads to evacuation of the control room.

Suppressed fire in other (non-critical) panels is conservatively assumed to result in MSIV closure, Turbine trip and loss of condensate heat sink. However, unsuppressed fire leads to loss of off-site power. The fire scenario is graphically shown below in an event tree format and evaluated in Table 6-2.8.3.



Note that Case 1A assumes one stuck open relief valve. This form of failure would require a sustained "hot short" condition, such that the SRV remains energized to remain open. The likelihood of a single, sustained hot short failure, such that the circuit is not isolated by the installed fuses and circuit breakers, is judged to be extremely unlikely. Regarding the possibility more than one SRV opening and remaining open due to hot short failure conditions, this would require two such failures to occur, which is judged to be even more unlikely than the case of a single valve failing due to inadvertent energization of a single valve.

6.2.9 Turbine Building (Fire Area 25)

The Turbine Building consists of widely separated areas, i.e., the intake pump station and pipe tunnel and turbine building areas. The turbine building itself consists of 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, this area was analyzed using specific fire scenarios involving significant fire sources and corresponding damage to mitigating systems. The evaluation of each of these areas is described as follows:

6.2.9.1 Fire Compartment 25-1, Intake Pump Station

Fire scenarios involving the condenser circulating water pumps and fires simultaneously affecting RHRSW and EECW power cables were evaluated in Table 6-2.9.1 and Table 6-2.9.1 (a).

6.2.9.2 Fire Compartment 25-2, Pipe Tunnel

This area contains insignificant combustibles amount of combustibles and lacks plant components. However, manual shutdown was conservatively assumed in the evaluation in Table 6-2.9.2.

6.2.9.3 Fire Compartment 25-3, Turbine Building

This area includes Units 1, 2 and 3 Turbine Buildings and has the highest fire frequency of all areas, at 5.59E-1 fires per year. Various fire scenarios were postulated in different areas of the Turbine Building involving significant fire sources and components. In all 8 cases were postulated as described in Table 6-2.9.3.

A recombiner fire may require 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 2 or Unit 3 recombiners (Case 1A) would not be expected to impact operation of Unit 1, a plant trip is assumed for 1 out of every 3 Hydrogen recombiner fires (Case 1A). Recombiners have a fire frequency of 7.40E-2 per unit per year (fire frequency for 2 unit recombiners is 7.40E-2 * 2 = 1.48E-1).

Turbine generator lube oil fires have a total fire ignition frequency of 1.20E-02 and could be expected to lead to a plant trip for all fires in Unit 2 and unit 3 (Case 2A). Fires at Unit 1 are expected to result in plant trip (Case 2B). These areas are supplied with deluge water spray systems. Turbine generator oil fires contribute a total of 3.60E-02 to fire ignition frequency (1.20E-2 per unit).

Fires at the turbine deck have a frequency of (5.70E-03 + 7.70E-03 =) 1.34E-02 per unit per year, from turbine generator exciter and hydrogen sources, respectively. Minor fires on the turbine operating deck of Units 2 and 3 are not expected to require Unit 1 plant trip (Case 3A 1). The fire frequency of turbine deck fires for unit 2 and 3 is (1.34E-2*2)

= 2.68E-2. Based on EPRI FEDB-2001, 58% of the fires in this area are minor or the minor fire frequency is (2.68E-2*0.58) = 1.55E-2. Severe fires on the turbine deck for units 2 and 3 are assumed to have no impact on Unit 1 (Case 3A-2). 42% of the fires on unit 2 and 3 turbine deck are severe or the severe fire frequency is (2.68E-2*0.42) = 1.13E-2. The fire frequency of 1.34E-2 is assigned to fires on Unit 1 turbine deck (Case 3B). These fires are expected to result in Unit 1 plant trip.

The remaining fire frequency for turbine building (2.61E-01) is assigned to cover all other areas (Case 4). Plant trip is not expected for fires in these areas.

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

IF=5.59E-01	CASE 1A	Recombiners Unit 2 and 3	Screened (=1.48E-1)
	CASE 1B	Recombiners Unit 1	Turbine Trip (=7.40E-2)
	CASE 2A	Unit 2 and 3 Lube oil	Screened (=2.4E-2)
	CASE 2B	Unit 1 Lube oil	Turbine Trip (=1.2E-2)
	CASE 3A-1	Turbine Deck-Unit 2 & 3 Minor Fire	Screened (=1.55E-2)
	CASE 3A-2	Turbine Deck-Unit 2 & 3 Severe Fire	Screened (=1.13E-3)
	CASE 3B	Turbine Deck - Unit 1	Loss of Offsite Power (=1.34E-2)
	CASE 4	Other Areas	Screened (=2.61E-1)
		Total =	5.59E-01

6.2.10 Yard Areas

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:

Case 1. Yard fire propagating to the Turbine Building. The fire ignition frequency for a single unit plant is given as 2.60E-03. This fire is modeled as a reactor trip with MSIV closure.

Case 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 5.10E-03. As indicated, this is modeled as a total loss of all offsite power.

Case 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 or MSIV closure. The fire ignition frequency given for this type of fire at a single station is 2.60E-02. Since no material degradation beyond main transformer failure is indicated, this is modeled as a turbine trip.

The evaluation of this area is described in Table 6-2.10.

6.3 Evaluation of Transient Combustibles

The quantitative phase of the analysis (Section 5) considers engulfing fire in all compartments and components located within the compartment are assumed to be damaged. Therefore, fixed or transient combustible analysis was not necessary. The compartments which were not screened out initially were then evaluated as part of detailed evaluation process (Section 6). The unscreened fire areas/compartments (i.e., 4, 5, 16-1, 16-2, 16-3, 24, 25-1, 25-2 and 25-3) are mostly electrical/switchgear rooms with the exception of turbine building and some areas in the control building. In the electrical rooms, the source and targets will generally be electrical cabinets and cables associated with the same cabinet. Therefore, a source/target evaluation involving fixed or transient combustibles is not practical. These areas were evaluated based on the fire severity factors (derived from Reference 3) and their impact on mitigating systems. The fire frequency for these areas is segmented into a range of cases, depicting the consequences of minor and severe fires. The transient combustibles are inherently included in the derivation of fire severity factors (Section 6.1). Note that the total ignition frequency used in the detailed analysis also includes contribution of transient combustibles. Therefore, only unscreened Unit 1 reactor Building fire area will be evaluated for transient combustible impacts.

Unit 2 Reactor Building - Evaluation of Transient Combustibles

Transient combustibles are analyzed by applying the total generic fire frequency per unit to Unit 1 Reactor Building. These fire ignition sources consist of transients (3.60E-02), cable fires due to welding (1.30E-03) and transient fires due to welding (3.40E-02), with a total fire ignition frequency of **7.13E-2**.

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 Table 6-3.1. Since the 32 gallon trash bag represents the more restrictive case (i.e., larger range of plume and radiant exposure damage), this case is used to evaluate all transient combustibles.

Review of the Unit 1 Reactor Building shows that the majority of electrical raceways are located well above the plume damage height of 12 feet.

The 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.

The frequency of target damage due to plume or radiant exposure 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) determines 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 1 Reactor Building, the total floor area is listed as 69,277 ft^2. Assume that 20% of the Reactor Building floor area is occupied up by plant hardware or a net floor area of 56000 ft^2. 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. For radiant effects, the effective 5 ft. radius surface area is rounded up to 100 ft^2. The total exposed surface area of components located within the 5 ft. radius is estimated to be 1500 ft^2.

3. Calculation of a probability that the critical amount of transients will be present between inspections (w). For the Browns Ferry plant, fire hazard inspections are conducted on no less than a weekly basis (F_W = 52). Based on EPRI guidance, no less than one noncompliance is conservatively assumed to occur per year (F_{CCL}), even if none have been recorded. w = (X/2) x (ln(1/x))

No credit is being taken for suppression. Therefore, probability of suppression unavailability factor (P_{fst}) is equal to 1.

Table 6-3.1 provides the details of calculation for probability of transient combustible fire exposure.

Table 6-3.1
Probability of Transient Combustible Fire Exposure

	Plume Region		Radiant Exposure
Probability of combustibles being exposed (P)	0.1		0.1
Surface area of targets facing floor/facing transient (FT^2) (A_S)	1000		1500
Radiant exposure surface area (FT^2) (A_{sr})	0		100
Net floor area (FT^2) (net area)	56000		56000
Probability of transient combustibles being located in range of target (u) ($A_s + A_{sr}$)/net area	0.018		0.029
Frequency of critical combustible present per year (F_{CCL})	1		1
Frequency of inspections per year (F_W)	52		52
$x = F_{CCL}/F_W$	0.019		0.019
Probability of critical amount of transients being present between inspections (w) (x/2*ln 1/x)	0.038		0.038
probability of target exposure $P_{tc} = (P_{fst})^* \times (u) \times (p) \times (w)$	6.78E-05		1.09E-04
Transient combustible ignition frequency (total generic transient fire frequency applied to reactor building)	7.13E-02		7.13E-02
Frequency of target damage = (Pfst)* x (u) x (p) x (w) x ignition freq.	4.84E-06		7.74E-06
TOTAL FREQUENCY OF TARGET DAMAGE		**1.26E-05**	
Highest fixed combustible CCDP (for Transformer TL1A)		6.58E-04	
CORE DAMAGE FREQUENCY DUE TO TRANSIENT COMBUSTIBLES		**8.27E-09**	
CDF (radiant and plume exposure surface area = 2000 ft^2)		6.36E-09	
CDF (radiant and plume exposure surface area = 3000 ft^2)		9.54E-09	
CDF (radiant and plume exposure surface area = 4000 ft^2)		1.27E-08	

* *P_{fst} = Probability of fire suppression unavailability. No credit is taken for suppression.*

Due to the uncertainties involved in the surface area of target calculations, a sensitivity analysis has been performed to assess the impact on CDF estimates. The above table shows the surface area of targets for radiant exposure was changed to 2000, 3000 and 4000 square feet. The change in the CDF values were then computed. It can be seen that the transient combustible scenarios still remain screened. Therefore, the transient combustibles in Unit 1 Reactor Building can be screened from further consideration.

6.4 Consideration of Potential Fire-Induced Containment Bypass Scenarios

Similar to the review performed in Section 5.3, fire areas (including fire sources within Reactor Building) can be screened from further consideration only if fire induced 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. Table 6-4 evaluates all areas for potential containment bypass scenarios where fire induced core damage frequency is more than 1E-07. Fire induced LERF is calculated as follows:

Table 6-4
Containment Bypass Scenarios

Fire Area	Fire Induced CDF (Section 6)	Initiating Event	Initiating Event Frequency (IE) (a)	LERF (b)	CLERP (b/a)	Ignition Frequency (severe fires) (c)	Fire Induced LERF (b/a)*c
1-1-5	1.29E-07	TT	5.09E-01	1.51E-07	2.97E-07	6.37E-03	1.89E-09
1-5-2	5.76E-07	TT	5.09E-01	3.51E-05	6.90E-05	8.77E-04	6.05E-08
1-6-2	1.70E-07	TT	5.09E-01	3.43E-07	6.74E-07	5.20E-03	3.50E-09
1-6-3	1.70E-07	TT	5.09E-01	3.43E-07	6.74E-07	5.20E-03	3.50E-09
2 (Severe Fire Outside Fire Zone 2-3)	2.23E-07	TT	5.09E-01	2.04E-08	4.01E-08	9.47E-03	3.80E-10
4 (Unsuppressed Severe Fire)	7.59E-07	IMSIV	5.70E-02	1.69E-06	2.96E-05	4.69E-05	1.39E-09
5 (Unsuppressed Severe Fire, Excluding Transient)	7.48E-07	IMSIV	5.70E-02	1.87E-06	3.28E-05	8.84E-05	2.90E-09
7	3.56E-07	IMSIV	5.70E-02	1.79E-09	3.14E-08	3.40E-03	1.07E-10
9 (Minor Fire)	1.26E-07	TT	5.09E-01	1.32E-09	2.60E-09	2.09E-02	5.43E-11
9 (Unsuppressed Severe Fire)	7.97E-09	TT	5.09E-01	1.22E-06	2.40E-06	3.29E-04	7.91E-10
16-2 (Severe Fire)	5.16E-07	IMSIV	5.70E-02	2.80E-07	4.92E-06	1.54E-03	7.59E-09
16-3 (Case 1A)	1.18E-07	IOOV	4.36E-02	1.69E-08	3.88E-07	2.39E-03	9.27E-10
16-3 (Case 1B)*	5.22E-07	TT	5.09E-01	3.62E-03	7.10E-03	8.16E-06	5.80E-08
25-1 (Transients)	1.81E-07	TT	5.09E-01	8.92E-07	1.75E-06	7.43E-05	1.30E-10
25-1 (Other)	3.29E-07	IMSIV	5.70E-02	1.04E-09	1.82E-08	7.77E-02	1.42E-09
25-3 (U1 Turbine Deck, Case 3B)	4.52E-07	LOSP	6.43E-03	9.32E-10	1.45E-07	1.34E-02	1.94E-09
Yard (LOSP)	6.43E-03	LOSP	6.43E-03	9.32E-10	1.45E-07	5.10E-03	7.39E-10

Note: * For Unit 1 base model, CDF=1.68E-6, and LERF=1.87E-7, the ratio is 0.111, use 0.111 x 6.4E-2 = 7.1E-3 for CLERF for Fire Zone 16-3 Case 1B.

Since each of these areas has a fire-related LERF that is below the cutoff of 1E-7, it can be concluded that these fires do not result in or cause containment breach concerns beyond those already addressed in the plant risk model. Therefore, these areas can continue to be screened from further consideration.

Tables - Detailed Analysis

The detailed fire source/fire area analysis are provided at the end of Section 6, starting on the next page.

Table 6-2.1 (c) (1) Detailed Analysis					
Fire Source	**1-1-1**	**480V RMOV Board 1C**			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
480V RMOV Board 1C	NONE	NONE	Loop I RHR	RPA, RPC	RHRGT
			Loop I Core Spray	CS	HPGTET
			RCIC Suppress. Pool Valves	SP	LPGTET
			250V MOV Board 1C	RD	ELECT12
			RHRSW Pump A2	SW2A	MESUPT
Risk Evaluation					
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
TT	5.09E-01	3.49E-06	6.86E-06	5.18E-03	3.55E-08

Comments: Electrical components and raceways within the zone of influence of potential fires in 250V RMOV board 1C, as shown in Attachment C.2, have been evaluated, based on the walkdown information described in References 24 & 25. 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). Manual reactor shutdown of Unit 1 is assumed (RPS is guaranteed successful).

Conclusion: Since the fire induced CDF is less than 1E-06, fires attributed to 250V RMOV board 1C can be screened from further consideration.

Table 6-2.1 (c) (2) Detailed Analysis					
Fire Source	**1-1-2**	**480V RB Vent Board 1B**			
	PSA MODEL IMPACTS DUE TO FIRE DAMAGE				
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
480V Vent Board 1B	NONE	NONE	Containment Atmospheric Dilution	CAD	MESUPT
			Core Spray Pump Loop II	CS2BS, CS2DS CSIISUP CSIISUP	LPGTET MLOCA LLOCA
			RCIC	RCI	HPGTET
		Risk Evaluation			
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
TT	5.09E-01	1.51E-07	2.75E-07	5.18E-03	1.42E-9

Comments: However, this is a backup to the gear-driven oil supply and has a very minor affect on HPCI performance. Hence, it is ignored. Electrical components and raceways within the zone of influence of potential fires in the 480V Vent Board 1B, as shown in Attachment C.2, have been evaluated, based on the walkdown information described in References 24 & 25. This review confirmed that a fire in the 480V Vent Board 1B could impact conduits that contain cables associated with the Containment Atmospheric system and The HPCI Auxiliary Oil system. However, this is a backup to the gear-driven oil supply and has a very minor affect on HPCI performance. Hence, it is ignored. Manual reactor shutdown of Unit 1 is assumed (RPS is guaranteed successful).

Conclusion: Since the fire induced CDF is less than 1E-06, fires attributed to Drywell/Torus compressor can be screened from further consideration.

Table 6-2.1 (c) (3) Detailed Analysis					
Fire Source	**1-1-3**	**250 RMOV Board 1C**			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
250V RMOV board 1C	RD	ELCCT12	DG A	GA	ELECT12
			RHRSW Pump A3	EA	MESUPT
			RCIC SP valves	SP	LPGTET
Risk Evaluation					
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
TT	5.09E-01	1.43E-07	2.80E-07	5.18E-03	1.45E-9

Comments: The HPCI Pump aux oil system is also affected by this fire. However, this is a backup to the gear-driven oil supply and has a very minor affect on HPCI performance. Hence, it is ignored. The RHRSW pump A3 is an EECW pump; the fire disables the autosrat. It is conservatively failed here.

Conclusion: Since the fire induced CDF is less than 1E-06, fires attributed to Drywell/Torus compressor can be screened from further consideration.

Table 6-2.1 (c) (4) Detailed Analysis					
Fire Source	**1-1-4**	**Core Spray Pumps 1A and 1C**			
	PSA MODEL IMPACTS DUE TO FIRE DAMAGE				
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
Core Spray Pump Loop I	CS	LPGTET	RCIC	RCI	HPGTET
Risk Evaluation					
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
TT	5.09E-01	8.43E-07	1.66E-06	6.37E-03	1.05E-08

Comments: The Loop I core spray pumps are located in separate quadrants on EL 519 (torus area) of the reactor building. This quadrant is widely separated from other quadrants. The amount of combustibles in the torus area is negligible and does not provide continuity for fire propagation. A fire in this quadrant will thus be confined to this area and will not affect the redundant divisions. The oil contained in these pumps is in sealed housing. The HVAC equipment for these pumps are also located in the same quadrant up close to the ceiling. The fan motors are located outside the housing and are relatively small HP. The fire hazards due to these motors are negligible. A walkdown of this area indicates showed that fires in the core spray pump room 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 (RPS is guaranteed successful). The failure of core spray pumps 1A and 1C is modeled by conservatively failing both trains of core spray in top event CS, in addition to failing RCIC top event RCI.

Conclusion: Since the fire induced CDF is less than 1E-06, fires attributed to core spray pumps 2A and 2C can be screened from further consideration.

Table 6-2.1 (c) (5) Detailed Analysis					
Fire Source	**1-1-5**	**RHR Pumps 1A and 1C**			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
RHR Pumps 1A	RPA	LPGTET	HPCI	HPI	HPGTET
RHR Pumps 1C	RPC	LPGTET			
Risk Evaluation					
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
TT	5.09E-01	1.03E-05	2.02E-05	6.37E-03	1.29E-07

Comments: The RHR pumps are located in separate quadrants on EL 519 (torus area) of the reactor building. This quadrant is widely separated from other quadrants. The amount of combustibles in the torus area is negligible and does not provide continuity for fire propagation. A fire in this quadrant will thus be confined to this area and will not affect the redundant divisions. The oil contained in these pumps is in sealed housing. There is no indication of oil leakage in the area. The HVAC equipment for these pumps is also located in the same quadrant up close to the ceiling. The fan motors are located outside the housing and are relatively small HP. The fire hazards due to these motors are negligible. A walkdown of this area indicates that all equipment located in these quadrants are associated with the respective pumps and HPCI. Therefore, a fire in this area will only affect these two systems. Due to the potential loss of safety related equipment, manual reactor trip is assumed to occur. The failure of RHR pumps 1A and 1C is incorporated by failing top events RPA and RPC and adjusting the RPB and RPD split reaction in the plant model.

Conclusion: Since the fire induced CDF is less than 1E-06, fires attributed to RHR pumps 2A and 2C can be screened from further consideration.

Table 6-2.1 (c) (6) Detailed Analysis					
Fire Source	**1-1-6**	**RCIC Pump**			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
RCIC	RCI	HPGTET	Core Spray Pump Loop I	CS	LPGTET
Risk Evaluation					
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
TT	5.09E-01	8.43E-07	1.66E-06	3.18E-03	5.27E-09

Comments: The RCIC pump is located in the same area as the Loop I core spray pumps. This case is similar to that of Fire Area 1-1-4 except the direct impacts and the indirect impacts are reversed.

Conclusion: Since the fire induced CDF is less than 1E-06, fires attributed to RCIC pumps can be screened from further consideration.

Table 6-2.1 (c) (7) Detailed Analysis					
Fire Source	**1-1-7**	**HPCI Pump**			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
HPCI	HPI	HPGTET	RHR Pumps 1A	RPA	LPGTET
			RHR Pumps 1C	RPC	LPGTET
Risk Evaluation					
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
TT	5.09E-01	3.28E-06	6.44E-06	3.18E-03	2.05E-08

Comments: The HPCI pump is located in the same area as the Loop 1 RHR the direct impacts and the indirect impacts are reversed. The implementation in the model is identical to that of Fire Area 1-1-5.

Conclusion: Since the fire induced CDF is less than 1E-06, fires attributed to fires in this area are eliminated from further consideration.

Table 6-2.1 (c) (8) Detailed Analysis					
Fire Source	**1-1-8**	**1-LPNL-925-0340 ES Div I & II Panel**			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
Engineered Safeguards Division I & II	*	N/A	Core Spray Loop I.	CS	LPGTET
			RHR Loop I	RPA, RPB	LPGTET
			RCIC	RCI	HPGTET
			ADS Valves	RVD	HPGTET
			Recirculation Pump A Vibration and Speed	RPT	HPATWS
Risk Evaluation					
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
TT	5.09E-01	6.91E-06	1.36E-05	5.18E-03	7.03E-08

* Turbine trip is assumed.

Comments: The loss of a recirculation pump will cause a reduction in reactor power. It is assumed that feedwater control adequately compensates and then a manual shutdown is initiated due to loss of EECS equipment. The loss of electrical support to the ADS valves does not fail the valves but eliminates a diverse mean of actuation.

Conclusion: Since the fire induced CDF is less than 1E-06, this area is screened from further consideration.

Table 6-2.1 (c) (9) Detailed Analysis					
Fire Source	**1-2-1**	**Core Spray Pumps 1B and 1D**			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
Core Spray Pumps 1B and 1D	CS	LPGTET			
Risk Evaluation					
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
TT	5.09E-01	1.40E-07	2.75E-07	6.35E-03	1.75E-09

Comments: Electrical components and raceways within the zone of influence of potential fires in the area of Loop II Core Spray pumps, as shown in Attachment C.2, have been evaluated, based on the walkdown information described in References 24 & 25. It is assumed that the Loss of the pumps resulted in a controlled shutdown.

Conclusion: Since the fire induced CDF is less than 1E-06, fires attributed to Core Spray Pumps 1B and 1D can be screened from further consideration.

Table 6-2.1 (c) (10) Detailed Analysis					
Fire Source	**1-2-2**	**RHR Pumps 1B and 1D**			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
Loop II RHR	RPB, RPD	RHRGT			
Risk Evaluation					
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
TT	5.09E-01	2.87E-06	5.64E-06	6.35E-03	3.58E-08

Comments: Electrical components and raceways within the zone of influence of potential fires in the area of Loop II RHR pumps, as shown in Attachment C.2, have been evaluated, based on the walkdown information described in References 24 & 25. It is assumed that the Loss of the pumps resulted in a controlled shutdown.

Conclusion: Since the fire induced CDF is less than 1E-06, fires attributed to LOOP II RHR can be screened from further consideration.

Table 6-2.1 (c) (11) Detailed Analysis					
Fire Source	**1-3-1**	RCW Pump 1A			
	PSA MODEL IMPACTS DUE TO FIRE DAMAGE				
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
RCW Pump 1A	RCW	MESUPT	LT-3-208C/D	L8F, L8H	HPGTET
(Panels 25-6-1 and 25-6A)			LT-3-56C/D	LM (degraded)	SIGL
			LT-3-58C/D	LV (degraded)	SIGL
			LT-3-203C/D	LVP (degraded)	SIGL
			PT-3-204C/D	NH2	SIGL
Risk Evaluation					
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
TT	5.09E-01	1.46E-07	2.88E-07	3.18E-03	9.15E-10

Comments: Electrical components and raceways within the zone of influence of potential fires in the area of RCW Pump 1A, as shown in Attachment C.2, have been evaluated, based on the walkdown information described in References 24 & 25. It is assumed that the fire resulted in a plant trip. Indirect failures due to impact on Panels 25-6-1 and 25-6A include loss of instrumentation, level and pressure transmitters, as indicated above.

Conclusion: Since the fire induced CDF is less than 1E-06, fires attributed to RCW Pump 1A can be screened from further consideration.

Table 6-2.1 (c) (12) Detailed Analysis					
Fire Source	**1-5-1**	**240V Lighting Board 1A**			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
240V LIGHTING BOARD 1A	NA	NA	HPCI	HPI	HPGTET
			LOOP II DWS	DWS	LPGTET
			CAD	CAD	MESUPT
Risk Evaluation					
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
TT	5.09E-01	4.84E-07	9.50E-07	5.18E-03	4.92E-09

Comments: Electrical components and raceways within the zone of influence of potential fires in the area of Loop II Core Spray pumps, as shown in Attachment C.2, have been evaluated, based on the walkdown information described in References 24 & 25.This scenario is bounded by Fire Source 1-1-8.

Conclusion: Since the fire induced CDF is less than 1E-06, this area is screened from further consideration.

Table 6-2.1 (c) (13) Detailed Analysis					
Fire Source	**1-5-2**	**240V Lighting Transformer TL1A**			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
240V Lighting Transformer 1A	NA	NA	HPCI	HPI	HPGTET
			RCIC	RCI	HPGTET
			Loop II Drywell Spray	DWS	LPGTET
			CAD	CAD	MSUPT
			480V SD Board 1A	RQ	ELECT12
			480V SD Board 2A	RS	ELECT12
Risk Evaluation					
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
TT	5.09E-01	3.35E-04	6.58E-04	8.77E-04	5.76E-07

Comments: Table C.2-1 shows that the damage threshold elevation (Zcrit) of 14.0 feet. This is higher than the ceiling height in this fire zone. Hence hot gas layer would form. It is conservatively assumed that Transformer TL1A fire would damage all major equipment and cables in this fire zone.

Conclusion: Since the fire induced CDF is less than 1E-06, this area is screened from further consideration.

Table 6-2.1 (c) (14) Detailed Analysis					
Fire Source	**1-5-3**	**4kV-480V Transformer TS1A**			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
4KV TO 480V TRANSFORMER TS1A			480V SD BD 1A	RQ	ELECT12
Risk Evaluation					
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
TT	5.09E-01	1.50E-06	2.95E-06	8.77E-04	2.58E-09
Comments: Electrical components and raceways within the zone of influence of potential fires in the area of transformer TS1A, as shown in Attachment C.2, have been evaluated, based on the walkdown information described in References 24 & 25. There are no other systems indirectly affected by the fire. It is assumed however, that fire disables 480V Shutdown Board 1A. **Conclusion:** Since the fire induced CDF is less than 1E-06, this area is screened from further consideration.					

Table 6-2.1 (c) (15) Detailed Analysis					
Fire Source	**1-5-4**	**4kV-480V Transformer TS1B**			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
4KV TO 480V TRANSFORMER TS1B			480V SD BD 2A	RS	ELECT12
Risk Evaluation					
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
TT	5.09E-01	3.31E-07	6.51E-07	8.77E-04	5.71E-10

Comments: Electrical components and raceways within the zone of influence of potential fires in the area of transformer TS1B, as shown in Attachment C.2, have been evaluated, based on the walkdown information described in References 24 & 25. There are no other systems indirectly affected by the fire. It is assumed however, that fire disables 480V Shutdown Board 2A. Panel 1PL460 is for the normal supply breaker to 480V RMOV Board 1C. A fire in this area will only have a minor increase in CDF.

Conclusion: This area is screened from further consideration.

Table 6-2.1 (c) (16) Detailed Analysis					
Fire Source	**1-5-5**	**4160V RPT Board 1-1 (Panel 1 and Panel 2)**			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
4160V RPT BOARD 1-1, PANEL 1 AND PANEL 2			HPCI	HPI	HPGTET
			RCIC	RCI	HPGTET
Risk Evaluation					
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
TT	5.09E-01	7.46E-06	1.47E-05	5.18E-03	7.59E-08

Comments: Electrical components and raceways within the zone of influence of potential fires in the area of 4160V RPT Board 1-1 (Panel 1 and Panel 2), as shown in Attachment C.2, have been evaluated, based on the walkdown information described in References 24 & 25. The other systems indirectly affected by the fire are HPCI and RCIC. The loss of the Recirculation Pumps will cause a power reduction. It is assumed the Feedwater Control system does not adequately respond in time and that a feedwater ramp-up occurs. Reactor trip expected due to power/flow mismatch.

Conclusion: Since the fire induced CDF is less than 1E-06, this area is screened from further consideration.

Table 6-2.1 (c) (17) Detailed Analysis					
Fire Source	**1-5-6**	**4160V RPT Board 1-2 (Panel 1 and Panel 2)**			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
4160V RPT BOARD 1-2, PANEL 1 AND PANEL 2	Subsumed with TT				
Risk Evaluation					
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
TT	5.09E-01	1.86E-07	3.65E-07	5.18E-03	1.89E-09

Comments: Electrical components and raceways within the zone of influence of potential fires in the area of 4160V RPT Board 1-2 (Panel 1 and Panel 2), as shown in Attachment C.2, have been evaluated, based on the walkdown information described in References 24 & 25. Conduits 1A-1AT17 and AT19 are connected to the ATWS trip breaker. Failure of these breakers will put them in the tripped position. The loss of the Recirculation Pumps (due to RPT board failure) will cause a power reduction. It is assumed the Feedwater Control system does not adequately respond in time and that a feedwater ramp-up occurs. Reactor trip expected due to power/flow mismatch. The CCDP uses the value for the TT initiator in the base model.

Conclusion: Since the fire induced CDF is less than 1E-06, this area is screened from further consideration.

Table 6-2.1 (c) (18) Detailed Analysis					
Fire Source	**1-5-7**	**RCIC Control Panel 1-25-31**			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
RCIC Control Panel 1-25-31	NA	NA	HPCI	HPI	HPGTET
			LOOP II DWS	DWS	LPGTET
			CAD	CAD	MESUPT
Risk Evaluation					
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
TT	5.09E-01	4.84E-07	9.50E-07	5.18E-03	4.92E-09

Comments: Same impact as 240V Lighting Board 1A (this panel is on the right side of Board 1A). Electrical components and raceways within the zone of influence of potential fires in the area of Loop II Core Spray pumps, as shown in Attachment C.2, have been evaluated, based on the walkdown information described in References 24 & 25.This scenario is bounded by Fire Source 1-1-8.

Conclusion: Since the fire induced CDF is less than 1E-06, this area is screened from further consideration.

Table 6-2.1 (c) (19) Detailed Analysis					
Fire Source	**1-5-8**	**Panel 25-3 (Filter Demin)**			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
Panel 25-3 (Filter Demin)	NA	NA			
Risk Evaluation					
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
TT	5.09E-01	0.00E+00	0.00E+00	5.18E-03	0.00E+00

Comments: Panel 25-3 fire will not cause a reactor trip. Also, cable trays above the panel in ZOI have no PRA impact.

Conclusion: Since the fire induced CDF is less than 1E-06, this area is screened from further consideration.

Table 6-2.1 (c) (20) Detailed Analysis					
Fire Source	**1-6-1**	**4kV-480V Emergency Transformer TS1E (Oil)**			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
4kV-480V Transformer TS1E	NA	NA	RCIC	RCI	HPGTET
			250V RMOV Board 1A	RB	ELECT12
			480V RMOV Board 1A	RE	ELECT12
Risk Evaluation					
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
TT	5.09E-01	1.67E-05	3.27E-05	8.75E-04	2.86E-08

Comments: Electrical components and raceways within the zone of influence of potential fires in the area of Emergency Transformer TS1E, as shown in Attachment C.2, have been evaluated, based on the walkdown information described in References 24 & 25. There are no direct failures due to loss of Transformer TS1E. Indirect failures impact systems RCIC, 250V RMOV Board 1A, 480V RMOV Board 1A, ATWS Initiation Channels A & B (not required), Signal Cable for VFD (no adverse PRA impact), Narrow Range Torus Level (no PRA impact). The loss of the Recirculation Pumps will cause a power reduction. Reactor trip expected due to power/flow mismatch. It is assumed that the loss of the pumps resulted in a controlled shutdown.

Conclusion: Since the fire induced CDF is less than 1E-06, this area is screened from further consideration.

Table 6-2.1 (c) (21) Detailed Analysis					
Fire Source	**1-6-2**	**VFD 1A (Panel)**			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
Variable Frequency Drive 1A (Panel)	NA	NA	RCIC	RCI	HPGTET
			250V RMOV Board 1A	RB	ELECT12
			480V RMOV Board 1A	RE	ELECT12
Risk Evaluation					
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
TT	5.09E-01	1.67E-05	3.27E-05	5.20E-03	1.70E-07

Comments: Electrical components and raceways within the zone of influence of potential fires in the area of Recirculation Pump VFD 1A Panel, as shown in Attachment C.2, have been evaluated, based on the walkdown information described in References 24 & 25. Impact of this fire is bounded by Fire 1-6-1. Indirect failures impact systems RCIC, 250V RMOV Board 1A, 480V RMOV Board 1A, ATWS Initiation Channels A & B (not required), Signal Cable for VFD (no adverse PRA impact), Narrow Range Torus Level (no PRA impact). The loss of the Recirculation Pumps will cause a power reduction. Reactor trip expected due to power/flow mismatch. It is assumed that the loss of the pumps resulted in a controlled shutdown.

Conclusion: Since the fire induced CDF is less than 1E-06, this area is screened from further consideration.

Table 6-2.1 (c) (22) Detailed Analysis					
Fire Source	**1-6-3**	**VFD 1B (Panel)**			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
Variable Frequency Drive 1B (Panel)	NA	NA	RCIC	RCI	HPGTET
			250V RMOV Board 1A	RB	ELECT12
			480V RMOV Board 1A	RE	ELECT12
Risk Evaluation					
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
TT	5.09E-01	1.67E-05	3.27E-05	5.20E-03	1.70E-07

Comments: Electrical components and raceways within the zone of influence of potential fires in the area of Recirculation Pump VFD 1B Panel, as shown in Attachment C.2, have been evaluated, based on the walkdown information described in References 24 & 25. Impact of this fire is bounded by Fire 1-6-1. Indirect failures impact systems RCIC, 250V RMOV Board 1A, 480V RMOV Board 1A, ATWS Initiation Channels A & B (not required), Signal Cable for VFD (no adverse PRA impact), Narrow Range Torus Level (no PRA impact). The loss of the Recirculation Pumps will cause a power reduction. Reactor trip expected due to power/flow mismatch. It is assumed that the loss of the pumps resulted in a controlled shutdown.

Conclusion: Since the fire induced CDF is less than 1E-06, this area is screened from further consideration.

Table 6-2.2 Detailed Analysis					
Fire Compartments	**2**	**Unit 2 Reactor Building**			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
480V RMOV Board 2C	RJ	ELECT12	DG B Breaker to SDB 4kV B	GB	ELECT12
480V RMOV Board 2D	RK	ELECT12	DG C Breaker to SDB 4kV C	GC	ELECT12
480V RMOV Board 2E	RL	ELECT12	DG D Breaker to SDB 4kV D	GD	ELECT12
250V RMOV Board 2C	RD	ELECT12	Battery Charger 2A	DH	ELECT12
Risk Evaluation					
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
TT	5.09E-01	6.54E-05	1.29E-04	1.27E-01	1.64E-05

Comments: A bounding case run was evaluated for potential fires in the Unit 2 Reactor Building. Turbine Trip of Unit 1 is assumed. This fire scenario is modeled as initiator F2 in RISKMAN.

Conclusion: Since the fire induced CDF is more than 1E-06, fires in the Unit 2 Reactor Building will be further evaluated in Table 6-2.2 (a). This evaluation is conservative in that all fires damage all significant cables that are assumed to transit through the area regardless of their separation from each other, severity of fires, availability of manual or automatic suppression, etc.

<table>
<tr><th colspan="6">Table 6-2.2 (a)
Detailed Analysis</th></tr>
<tr><td>Fire Compartments</td><td>2</td><td colspan="4">Unit 2 Reactor Building</td></tr>
<tr><td colspan="6">PSA MODEL IMPACTS DUE TO FIRE DAMAGE</td></tr>
<tr><th>Fire damaged Components (Direct Impacts)</th><th>Mitigating Systems Impact (Top Events)</th><th>Event Tree Impact</th><th>Fire Damaged Components (Indirect Impacts)</th><th>Mitigating Systems Impact (Top Events)</th><th>Event Tree Impact</th></tr>
<tr><td colspan="3" rowspan="4">CASE 1: Fire in Fire Zone 2-3
U2 Preferred AC Transformer in 2-3 catches fire (Fails GB, GC, GD, and DH Top Events)</td><td>DG B Breaker to SDB 4kV B</td><td>GB</td><td>ELECT12</td></tr>
<tr><td>DG C Breaker to SDB 4kV C</td><td>GC</td><td>ELECT12</td></tr>
<tr><td>DG D Breaker to SDB 4kV D</td><td>GD</td><td>ELECT12</td></tr>
<tr><td>Battery Charger 2A</td><td>DH</td><td>ELECT12</td></tr>
<tr><td colspan="3"></td><td colspan="3">CASE 2a: Minor Fires in U2 RB (Other than 2-3)
No plant trip. Screened from further consideration.</td></tr>
<tr><td>480V RMOV Board 2C</td><td>RJ</td><td>ELECT12</td><td colspan="3" rowspan="4">Case 2b: Severe Fires in U2 RB (Other than 2-3)
U1 Reactor trip is assumed. Fire damages RJ, RK, RL, and RD top events.</td></tr>
<tr><td>480V RMOV Board 2D</td><td>RK</td><td>ELECT12</td></tr>
<tr><td>480V RMOV Board 2E</td><td>RL</td><td>ELECT12</td></tr>
<tr><td>250V RMOV Board 2C</td><td>RD</td><td>ELECT12</td></tr>
<tr><td colspan="6">Risk Evaluation</td></tr>
<tr><th>Initiating Event
CASE 1
CASE 2a
CASE2b</th><th>Initiating Event Frequency (IE)</th><th>CDF_{MS}</th><th>$CCDP_{MS}$ (CDF_{MS}/IE)</th><th>Ignition Frequency (F1)</th><th>CDF_{FIRE} ($CCDP_{MS}$*F1)</th></tr>
<tr><td>TT</td><td>5.09E-01</td><td>1.42E-07</td><td>2.80E-07</td><td>8.94E-04</td><td>2.50E-10</td></tr>
<tr><td>N/A</td><td>N/A</td><td>N/A</td><td>N/A</td><td>1.17E-01</td><td>N/A</td></tr>
<tr><td>TT</td><td>5.09E-01</td><td>1.20E-05</td><td>2.35E-05</td><td>9.47E-03</td><td>2.23E-07</td></tr>
<tr><td></td><td></td><td></td><td>TOTAL</td><td>1.27E-01</td><td>2.23E-07</td></tr>
</table>

Comments: As stated in Table 6-2.2, that all fires are assumed to damage safety related/PSA modeled cables that are assumed to transit through the area regardless of their separation from each other, severity of fires, manual or automatic suppression, etc. It is realized that not all fires in the Reactor Building will be severe. Review of fire events database (FEDB-2001) indicates that 7.5 % fires in the reactor building are severe, See Table 6-1 (c). Turbine trip initiating event (TT) is used for Unit 1 due to the severe fires. *The minor fires can be screened from further consideration.* Case 1 models U2 preferred transformer fire (frequency 8.94E-04, taken from Unit 2 IPEEE analysis, Rev. 1) in fire zone 2-3. Case 2a models minor fire in the rest of the U2 Reactor Building (frequency (1.27E-1 - 8.94E-4) x (1-0.075) = 1.17E-1), which is screened. Case 2b models severe fire in the rest of the U2 Reactor Building (other than fire zone 2-3) with an ignition frequency: (1.27E-1 - 8.94E-4) x 0.075 = 9.47E-3). Case 1 was modeled as initiator F2C1, and Case 2 as F2C2 in RISKMAN.

Conclusion: Since the fire induced CDF is less than 1E-06, fires in the Unit 2 Reactor Building can be screened from further consideration.

Table 6-2.3 Detailed Analysis					
Fire Compartments	3	**Unit 3 Reactor Building**			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
Unit 3 Shutdown Bd A3EC	A3EC	ELECT3			
Unit 3 Shutdown Bd A3ED	A3ED	ELECT3			
480V Diesel Aux. Bd 3EB	RP	ELECT3			
480V Shutdown Board 3A	RX	ELECT3			
480V Shutdown Board 3B	RY	ELECT3			
Diesel Generator 3A, 3B, 3C, 3D	GE, GF, GG, GH	ELECT3			
4KV Shutdown Board 3EB control batteries, charger (SB-3EB)	DF	ELECT3	Recovery Actions Impact		
			Shutdown Bus Recovery	SDREC	ELECT3
			250V DC DIV II Control Power Recovery	CPREC	ELECT3
Risk Evaluation					
Initiating Event	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
TT	5.09E-01	3.77E-07	7.40E-07	1.26E-01	9.36E-08

Comments: Browns Ferry Unit 3 is an operating unit and contains equipment supporting unit 1. This support is factored in the Unit 1 PSA model. A bounding case was evaluated for potential fires in the Unit 3 Reactor Building. It was assumed that the equipment located in any of the fire areas within Unit 3 Reactor Building (fire areas 12, 13, 14 and 15) could potentially fail, since the associated cables may transit through the Unit 3 Reactor Building to Unit 1. In addition, equipment located in the Diesel Generator Building (fire area 21) is also assumed damaged, since the power cables for diesel generators may transit through the Unit 3 Reactor Building on their way to Unit 1. Turbine trip (TT) for Unit 1 was assumed. This case is modeled by initiator F3 in RISKMAN.

Conclusion: Since the fire induced CDF 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.

Table 6-2.4 Detailed Analysis					
Fire Area	4	4kV Shutdown Board Room B (Unit 1 Reactor Building, 593' Elevation)			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	Mitigating Systems Impact (Top Events)	Event Tree Impact	Fire Damaged Components (Indirect Impacts)	Mitigating Systems Impact (Top Events)	Event Tree Impact
CASE 1 - MINOR FIRE					
4kV Shutdown Board B	AB	ELECT12			
CASE 2 - SEVERE FIRE					
See Section 5.1.4 Quantitative Evaluation					
Risk Evaluation					
Initiating Event CASE1 CASE 2	Initiating Event Frequency (IE)	CDF_{MS}	$CCDP_{MS}$ (CDF_{MS}/IE)	Ignition Frequency (F1)	CDF_{FIRE} ($CCDP_{MS}*F1$)
TT	5.09E-01	6.46E-07	1.27E-07	1.81E-02	2.29E-08
IMSIV	5.70E-02	9.22E-04	1.62E-02	1.34E-03	2.17E-05
			TOTAL	1.94E-02	2.17E-05

Table 6-2.4
Detailed Analysis

Comments: 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.

Case 1: Evaluates a minor fire in the area which is assumed to start in 4kV shutdown board B. The fire is then suppressed manually or allowed to burn out. Turbine trip (TT) is assumed for Unit 1. Case 1 is designated as initiator F4C1 in RISKMAN model.
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. Reactor trip with MSIV closure (IMSIV) was assumed for Unit 1. Case 2 is designated as initiator F4 in RISKMAN model.

Conclusion: Since the total core damage frequency for both of these cases is more than 1E-06, fires in this area will be further evaluated in Table 6-2.4 (a)

Fire Hazards Evaluation:
Fire area 4 was further analyzed to determine the compartment temperatures based on typical heat release rates of a single electrical cabinet in a confined space. The evaluation accounted for the volume of the room and thermal heat sink of the concrete structure to assess the upper layer hot gas temperatures. The hot gas layer temperature was estimated using the NIST fire model FAST Version 3.1.7 for a single compartment. Following parameters were used to quantify the fire model:
Compartment Size = 24 ft x 50 ft
Ceiling Height = 10 ft
Door Undercut = 4 ft x 1 inch (doors closed)
Fire Parameters: Slow fire, level to peak at 190 btu/sec within 600 seconds; No decay; Unconstrained (fuel controlled);
Fuel Height = 6 ft.
Simulation = 1800 seconds
Initial Room temperature = 80 deg. F
Outside temperature = 95 deg F

The results show that the upper layer hot gas temperature remains around 210 deg. F for the postulated fire scenario. This is below the damage temperature for non-qualified cables.

Table 6-2.4
Detailed Analysis

Time (Seconds)	Upper Layer Temp (K)	Upper Layer Temp (F)
0	300	80
100	323	122
200	338	149
300	350	170
400	355	179
500	358	185
600	362	192
700	365	197
800	366	199
900	366	200
1000	367	201
1100	368	202
1200	368	203
1300	369	204
1400	369	205
1500	369	206
1600	370	206
1700	370	207

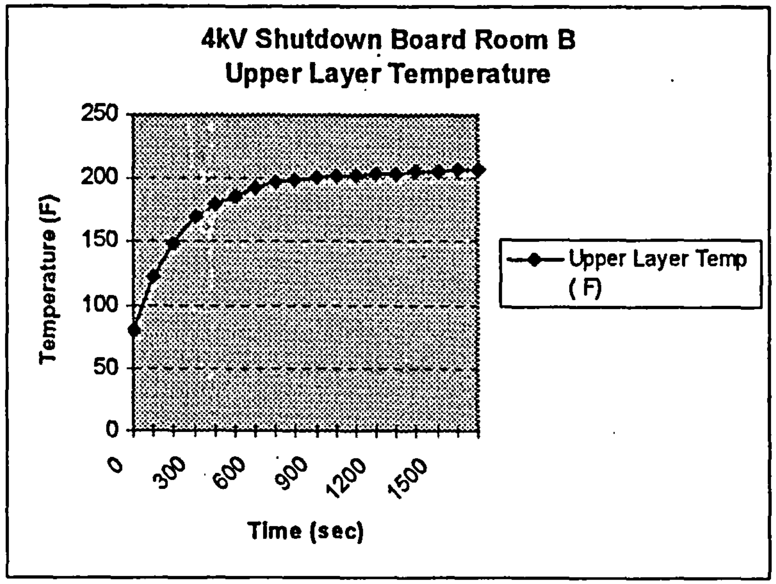


Table 6-2.4 (a)
Detailed Analysis Fire Area 4

	Ignition Frequency	Severity			Manual Suppression (fire brigade)	
		Case 1				
	IF=3.40E-3	Minor Fire (4kV SD Bd B)			(Suppressed/Unsuppressed) CCDP = 1.27E-6	
		0.931				
		Case 2			Suppressed = 1- 0.2	
		Severe Fire (Engulfing)			CCDP = 1.27E-6 (similar to Case 1)	
		0.069				
					Unsuppressed* = 0.2	
					CCDP = 1.62E-2	

*The probability of manual non-suppression (PNS) factor is taken from NRC draft Fire Protection Significance Determination Process (SDP), Appendix F. Fire propagation beyond the cabinet is assumed to take 15 minutes and fire is detected in 3 minutes by the installed smoke detectors. This is conservative, since using the NUREG 1805 spreadsheet, the detection time is less than a minute for a 200 kW HRR. For a difference of 12 minutes, the electrical fires mean non-suppression curve shows a PNS factor of 0.2.

Fire Scenario	Initiating Event	Initiating Event Frequency (IE)	CDF_{MS}	$CCDP_{MS}$ (CDF_{MS}/IE)	Ignition Frequency (F1)	Severity Factor (SF)	Probability of Non-Suppression (PNS_{manual})	CDF_{FIRE} (ΣIF*SF*PNS*CCDP)
Minor Fire (Suppressed or unsuppressed)	TT	5.09E-01	6.46E-07	1.27E-06	3.40E-03	0.93	N/A	4.02E-09
Severe Fire (Suppressed)	TT	5.09E-01	6.46E-07	1.27E-06	3.40E-03	0.07	0.80	2.38E-10
Severe Fire (Unsuppressed)	IMSIV	5.70E-02	9.22E-04	1.62E-02	3.40E-03	0.07	0.20	7.59E-07
							SUM	**7.64E-07**

Comments: Neglecting the contribution of transient combustibles due to adequate plant control procedures, the ignition frequency in the area is based on electrical cabinets which is 3.40E-3 (Appendix B). Assume that the fire starts in 4kV SDBD. A minor fire affects the 4kV board only, whereas, a severe fire affects all other components in the room.

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

Table 6-2.5 Detailed Analysis					
Fire Area	5	4kV Shutdown Board Room A and 250V Battery Room (Unit 1 RB, EL 621')			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	Mitigating Systems Impact (Top Events)	Event Tree Impact	Fire Damaged Components (Indirect Impacts)	Mitigating Systems Impact (Top Events)	Event Tree Impact
CASE 1 - MINOR FIRE					
4kV Shutdown Board A	AA	ELECT12			
CASE 2 - SEVERE FIRE					
See Section 5.1.5 Quantitative Evaluation					
Risk Evaluation					
Initiating Event CASE1 CASE 2	Initiating Event Frequency (IE)	CDF_{MS}	$CCDP_{MS}$ (CDF_{MS}/IE)	Ignition Frequency (F1)	CDF_{FIRE} $(CCDP_{MS}*F1)$
TT	5.09E-01	3.83E-06	7.53E-06	2.19E-02	1.64E-07
IMSIV	5.70E-02	4.82E-04	8.46E-03	1.72E-03	1.46E-05
			TOTAL	2.36E-02	1.47E-05

Table 6-2.5 Detailed Analysis
Comments: Review of the results from the screening quantification of this area (see Section 5.1.5) 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. <u>Case 1</u>: Evaluates a minor fire in the area which is assumed to start in 4kV shutdown board A. The fire is then suppressed manually or allowed to burn out. Turbine trip (TT) is assumed for Unit 1. This case is modeled as initiator F5C1 in RISKMAN. <u>Case 2</u>: A fire starts anywhere in fire area 5 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. The fire is then suppressed manually or allowed to burn out. Reactor trip with MSIV closure (IMSIV) is assumed for Unit 1. This case is modeled as initiator F5 in RISKMAN. **Conclusion:** Since the total core damage frequency for both of these cases is more than 1E-06, fires in this area will be further evaluated in Table 6-2.5 (a). **Fire Hazards Evaluation:** Fire area 5 was further analyzed to determine the compartment temperatures based on typical heat release rates of a single electrical cabinet in a confined space. The evaluation accounted for the volume of the room and thermal heat sink of the concrete structure to assess the upper layer hot gas temperatures. The hot gas layer temperature was estimated using the NIST fire model FAST Version 3.1.7 for a single compartment. Following parameters were used to quantify the fire model: Compartment Size = 25 ft x 65 ft Ceiling Height = 17 ft Door Undercut = 4 ft x 1 inch (doors closed) Fire Parameters: Slow fire, level to peak at 190 btu/sec within 600 seconds; No decay; Unconstrained (fuel controlled); Fuel Height = 6 ft. Simulation = 1800 seconds Initial Room temperature = 80 deg. F Outside temperature = 95 deg F The results shown on the following sheet indicate that the upper layer hot gas temperature remains around 155 deg. F for the postulated fire scenario. This is below the damage temperature for non-qualified cables.

Table 6-2.5
Detailed Analysis

Time (Seconds)	Upper Layer Temp (K)	Upper Layer Temp (F)
0	300	80
100	307	93
200	314	107
300	323	122
400	328	132
500	332	138
600	335	143
700	336	146
800	337	148
900	338	149
1000	339	150
1100	339	151
1200	340	152
1300	340	153
1400	341	154
1500	341	154
1600	341	155
1700	342	155

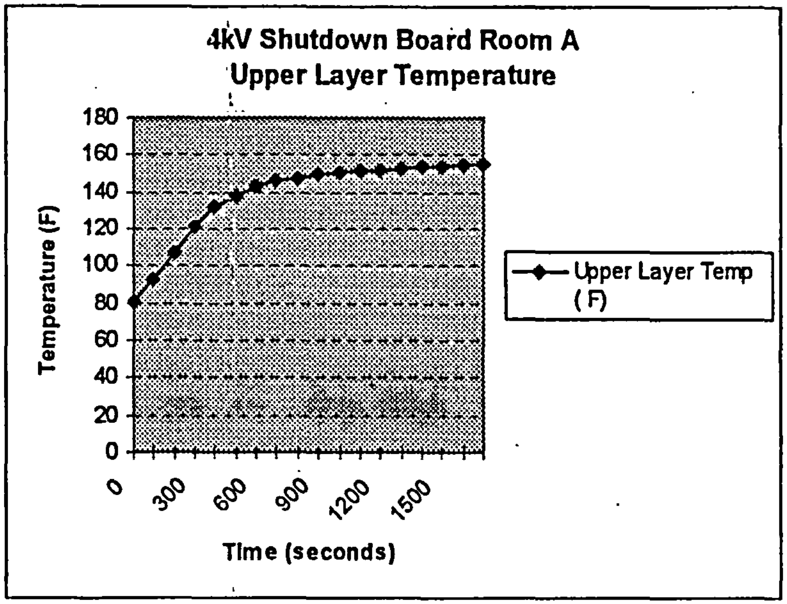


Table 6-2.5 (a)
Detailed Analysis Fire Area 5

	Ignition Frequency	Severity			Manual Suppression (fire brigade)	
		Case 1				
	IF=6.06E-3	Minor Fire (4kV SD Bd A)			(Suppressed/Unsuppressed) CCDP = 7.53E-6	
		0.927				
		Case 2			Suppressed = 1- 0.2	
		Severe Fire (Engulfing)			CCDP = 7.53E-6 (similar to Case 1)	
		0.073				
					Unsuppressed* = 0.2	
					CCDP = 8.46E-3	

*The probability of manual non-suppression (PNS) factor is taken from NRC draft Fire Protection Significance Determination Process (SDP), Appendix F. Fire propagation beyond the cabinet is assumed to take 15 minutes and fire is detected in 3 minutes by the installed smoke detectors. This is conservative, since using the NUREG 1805 spreadsheet, the detection time is less than a minute for a 200 kW HRR. For a difference of 12 minutes, the electrical fires mean non-suppression curve shows a PNS factor of 0.2.

Fire Scenario	Initiating Event	Initiating Event Frequency (IE)	CDF_{MS}	$CCDP_{MS}$ (CDF_{MS}/IE)	Ignition Frequency (F1)	Severity Factor (SF)	Probability of Non-Suppression (PNS_{manual})	CDF_{FIRE} (ΣIF*SF*PNS*CCDP)
Minor Fire (Suppressed or unsuppressed)	TT	5.09E-01	3.83E-06	7.53E-06	6.06E-03	0.93	N/A	4.23E-08
Severe Fire (Suppressed)	TT	5.09E-01	3.83E-06	7.53E-06	6.06E-03	0.07	0.80	2.66E-09
Severe Fire (Unsuppressed)	IMSIV	5.70E-02	4.82E-04	8.46E-03	6.06E-03	0.07	0.20	7.48E-07
							SUM	**7.93E-07**

Comments: Neglect the contribution of transient combustibles due to adequate plant control procedures. Since the batteries are isolated in separate rooms within the area, a fire in the battery rooms is likely to be confined to these rooms and not affect the electrical switchgear in the fire area. The ignition frequency in the area is now revised to include all components except transients and the batteries. Assume that the fire starts in 4kV SDBD A. A minor fire affects the 4kV board only, whereas, a severe fire affects all other components in the room.

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

Table 6-2.6 Detailed Analysis					
Fire Area	**7**	**480V Shutdown Board Room 1B (Unit 1 Reactor Building, 621' Elevation)**			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	Mitigating Systems Impact (Top Events)	Event Tree Impact	Fire Damaged Components (Indirect Impacts)	Mitigating Systems Impact (Top Events)	Event Tree Impact
CASE 1 - SEVERE FIRE					
See Section 5.1.7 Quantitative Evaluation					
Risk Evaluation					
Initiating Event CASE1	Initiating Event Frequency (IE)	CDF_{MS}	$CCDP_{MS}$ (CDF_{MS}/IE)	Ignition Frequency (F1)	CDF_{FIRE} ($CCDP_{MS}$*F1)
IMSIV	5.70E-02	5.96E-06	1.05E-04	3.40E-03	3.56E-07
			TOTAL	**3.40E-03**	**3.45E-07**

Comments: In Section 5.1.7, an engulfing fire scenario (RISKMAN initiator F7) was developed which fails all the equipment and cables in this area. This fire scenario represents a severe fire in the area. A minor fire scenario could have been developed by assuming only 480V Shutdown Board 1B catches fire and the fire is suppressed (but still the board fails). As discussed in Section 5.1.7, failure of 480V Shutdown Board 1B will cause failure of 480V RMOV Board 1B, and also causes failure of crosstie of Unit 2 RHR loop I to Unit 1 RHR loop II. Essentially, this "minor" fire scenario would have comparable impact and CCDP with the severe fire in Area 7. Hence, distinguishing the fire frequency between severe and "minor" fire would not reduce the CDF for this area.

However, the administrative control would make transient fire in this area not credible. We use the total cabinet fire initiating frequency (3.40E-3) as the fire initiator frequency for Fire Area 7. Please note based on Attachment B, frequency calculation, Area 7 fire frequency (total of 1.92E-2) consists only of transient fire frequency and electric cabinet fire frequency (3.40E-3). We assume this fire is a severe fire, with CCDP of 1.01E-4 (same as in Section 5.1.7). The total CDF from this are is then 3.45E-7.

Conclusion: Since the total core damage frequency in this area is less than 1E-06, fires in this area can be screened.

Table 6-2.7 Detailed Analysis					
Fire Area	**9**	**4kV Shutdown Board Room C and 250V Battery Room (Unit 2 RB, EL 621')**			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
CASE 1 - MINOR FIRE					
4kV Shutdown Board C	AC	ELECT12			
CASE 2 - SEVERE FIRE - Engulfing Case					
See 5.1.9 Quantitative Evaluation					
Risk Evaluation					
Initiating Event CASE1 CASE 2	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
TT	5.09E-01	3.08E-06	6.05E-06	2.09E-02	1.26E-07
TT	5.09E-01	3.11E-04	6.12E-04	1.65E-03	1.01E-06
			TOTAL	**2.26E-02**	**1.13E-06**

Comments:
Case 1: Evaluates a minor fire in the area which is assumed to start in 4kV shutdown board C. Smoke detectors are installed in this area and will provide early warning. The fire is then suppressed manually. Turbine trip initiator is assumed for Unit 1. This case is modeled as F9C1 in RISKMAN.
Case 2: Severe fire in this area, this is the same engulfing case in the screening analysis (RISKMAN initiator F9).

Conclusion: The total core damage frequency for these cases is slightly above 1E-06. Therefore, this area will be further evaluated in Table 6-2.7 (a).

Table 6-2.7 (a)
Detailed Analysis Fire Area 9

Ignition Frequency | Severity | Manual Suppression (fire brigade)

IF=2.26E-2 –

Case 1
Minor Fire (4kV SD Bd C)
0.927
(Suppressed/Unsuppressed)
CCDP = 6.05E-6

Case 2
Severe Fire (Engulfing)
0.073
Suppressed = 1- 0.2
CCDP = 6.05E-6 (similar to Case 1)

Unsuppressed* = 0.2
CCDP =6.12E-4 (similar to Case 2)

*The probability of manual non-suppression (PNS) factor is taken from NRC draft Fire Protection Significance Determination Process (SDP), Appendix F. Fire propagation beyond the cabinet is assumed to take 15 minutes and fire is detected in 3 minutes by the installed smoke detectors. This is conservative, since using the NUREG 1805 spreadsheet, the detection time is less than a minute for a 200 kW HRR. For a difference of 12 minutes, the electrical fires mean non-suppression curve shows a PNS factor of 0.2.

Fire Scenario	Initiating Event	Initiating Event Frequency (IE)	CDF_{MS}	$CCDP_{MS}$ (CDF_{MS}/IE)	Ignition Frequency (F1)	Severity Factor (SF)	Probability of Non-Suppression (PNS_{manual})	CDF_{FIRE} (ΣIF*SF*PNS*CCDP)
Minor Fire (Suppressed or unsuppressed)	TT	5.09E-01	3.08E-06	6.05E-06	2.26E-02	0.93	N/A	1.26E-07
Severe Fire (Suppressed)	TT	5.09E-01	3.08E-06	6.05E-06	2.26E-02	0.07	0.80	7.97E-09
Severe Fire (Unsuppressed)	TT	5.09E-01	3.11E-04	6.12E-04	2.26E-02	0.07	0.20	2.02E-07
							SUM	3.36E-07

Comments: The ignition frequency used in this calculation is the total area ignition frequency including transient and batteries, which is conservative.

Conclusion: Since the total core damage frequency for these cases is less than 1E-06, fires in this area can be screened.

Table 6-2.8.1 Detailed Analysis					
Fire Area	16-1	Control Building - 593' Elevation			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	Mitigating Systems Impact (Top Events)	Event Tree Impact	Fire Damaged Components (Indirect Impacts)	Mitigating Systems Impact (Top Events)	Event Tree Impact
CASE 1 - MINOR FIRE (Aux Inst. Rooms or Computer Rooms)					
Total loss of Feedwater	Subsumed within initiator TLFW				
CASE 2 - SEVERE FIRE (Aux Inst. Rooms or Computer Rooms)					
MSIV Closure	Subsumed within initiator IMSIV				
HPCI	HPI	HPGTET			
CASE 3 - FIRE in Other Areas (Mech. Equip. Room, Process Computer Room, Communication Room) No Plant Trip)					
None requiring plant trip					
Risk Evaluation					
Initiating Event CASE 1 CASE 2 CASE 3	Initiating Event Frequency (IE)	CDF_{MS}	$CCDP_{MS}$ (CDF_{MS}/IE)	Ignition Frequency (F1)	CDF_{FIRE} ($CCDP_{MS}*F1$)
TLFW	2.58E-02	4.45E-08	1.73E-06	8.71E-03	1.50E-08
IMSIV	5.70E-02	9.74E-07	1.71E-05	7.27E-04	1.24E-08
N/A	N/A	N/A	N/A	2.83E-02	N/A
			TOTAL	3.78E-02	2.75E-08

Comments:
Case 1: Evaluates a minor fire in the Unit 1 Aux. Inst. Room or the Unit 1/2 Computer Room which is assumed to result in loss of all feedwater. Evaluated with initiator TLFW. This case is models as initiator F16_1C1 in RISKMAN.
Case 2: Evaluates a severe fire in the Unit 1 Aux. Inst. Room or the Unit 1/2 Computer Room which is assumed to result in MSIV closure and loss of HPCI. This case is models as initiator F16_1C2 in RISKMAN.
Case 3: Fire in these areas do not affect any Unit 1/2 mitigating systems and will not result in plant trip.

There are new cables routed in this fire zone. All but one cable was screened based on satisfying one of two rules:
1. The cable connected to only Unit 2 equipment.
2. The cable was connected to the remote shutdown board of either unit. Credit for remote shutdown is not taken.

Table 6-2.8.1
Detailed Analysis

There was one cable left unscreened. This cable connected a Unit 1 panel with a Unit 2; both panels were suppression pool #1 information panels. These panels are not explicitly modeled. We note that the #2 information panels are undamaged. Based on these facts, the new cables have no affect on Unit 1.

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

Fire Hazards Assessment:

SMOKE DETECTOR ACTIVATION AND SMOKE STRATIFICATION

The upward movement of the smoke in the plume is dependent on the smoke being buoyant relative to the surroundings. Given the physical configuration (fire sources and detector location), following correlations can be used to determine the time to detector actuation and smoke stratification possibility.

References:

1. NFPA 72, Fire Alarm and Detection System.
2. NFPA 92 B, Smoke Management Systems.
3. Fire Technology, Aug 1990, Smoke management of Covered Malls and Atria.
4. Fire Technology, May 1991, Letters to editor

Ceiling Mounted Smoke Detector Response

For radius-to-ceiling height ratios less than approximately 0.6, the temperature rise of the smoke can be estimated as function of time based on theoretical generalizations of the limited amount of experimental data. For $X < 100$:

X=	$4.6 \cdot 10^{-4} Y^2 + 2.7 \cdot 10^{-15} Y^6$
where:	
X =	$tQ^{1/3} / H^{4/3}$
Y =	$DT \cdot H^{5/3} / Q^{2/3}$

and where:

t = time from ignition (sec)

Q = heat release rate (steady fire) (Btu/sec)

Table 6-2.8.1 Detailed Analysis						
H = ceiling height above fire surface (ft)						
DT = Temperature rise of gasses within ceiling jet (^{0}F)						
Stratification of Smoke						
Assuming the ambient temperature increases linearly with increasing elevation, the maximum rise of the plume						
is dependent on the convective portion of the heat release rate of the fire and temperature change from floor to ceiling.						
	$H_{max} = 74Q_c^{2/5}DT_0^{-3/5}$					
	where:					
	Q_c = Convective portion of the heat release rate (Btu/sec) (approximately 70% of total heat release rate)					
	DT_0 = difference between ambient temp. at ceiling and ambient temp. at the level of fire surface (^{0}F)					
Description:					**Data**	
Height (H):					12	
Temperature rise within ceiling jet (DT) (Temp. rise required for smoke detector activation):					18	
Heat release rate (Q):					50	
Difference between ambient temp. at ceiling and ambient temp. at the level of fire surface (DT_0):					30	
Convective heat release rate (Q_c) (Q*0.7):					35	
	Y =		83.42			
	X =		3.20			
	Activation time for smoke detector (t) =				24	**Seconds**
	Maximum Smoke Height (H_{max}) =				40	**Feet**
	(Activation time @Q=100 Btu/s is 8 sec; @190 Btu/s is 3 sec)					
	Fire Brigade manual Response =				5 to 10	**Minutes** (based on fire drills)
	Fire Damage to Adjacent Cabinet =				15	**Minutes** (based on industry tests)

Table 6-2.8.2 Detailed Analysis					
Fire Area	16-2	Control Building - 606' (Cable Spreading Room)			
	PSA MODEL IMPACTS DUE TO FIRE DAMAGE				
Fire damaged Components (Direct Impacts)	Mitigating Systems Impact (Top Events)	Event Tree Impact	Fire Damaged Components (Indirect Impacts)	Mitigating Systems Impact (Top Events)	Event Tree Impact
CASE 1 - MINOR FIRE (Single Cable Tray, Loss of Feedwater)					
Total loss of Feedwater	Subsumed within initiator TLFW				
CASE 2 - SEVERE FIRE (Two Cable Trays, MSIV Closure, HPCI/RCIC/CS failure)					
MSIV Closure	Subsumed within initiator IMSIV				
HPCI	HPI	HPGTET			
RCIC	RCI	HPGTET			
Core Spray	CS	LPGTET			
Risk Evaluation					
Initiating Event CASE 1 CASE 2	Initiating Event Frequency (IE)	CDF_{MS}	$CCDP_{MS}$ (CDF_{MS}/IE)	Ignition Frequency (F1)	CDF_{FIRE} ($CCDP_{MS}$*F1)
TLFW	2.58E-02	4.45E-08	1.73E-06	1.05E-02	1.81E-08
IMSIV	5.70E-02	1.91E-05	3.35E-04	1.54E-03	5.16E-07
			TOTAL	**1.20E-02**	**5.35E-07**

Comments:
Case 1: Evaluates a minor fire in a single cable tray which is assumed to result in loss of all feedwater. Evaluated with initiator TLFW. This case is modeled as initiator C16_2C1 in RISKMAN.
Case 2: Evaluates a severe fire (two cable trays affected) which is assumed to result in MSIV closure, loss of HPCI, Loss of RCIC and CS. This case is modeled as initiator C16_2C2 in RISKMAN.

Table 6-2.8.2
Detailed Analysis

CABLE SPREADING ROOM FIRE PROPAGATION ANALYSIS

A fire in the CSR is expected to be *slow* growing at least initially, and if not controlled may become *medium* or *fast* growing fire. It is intended to detect and suppress the fire while still in the slow growth phase. A slow growth fire is defined as a fire which takes 400 or more seconds from the time established burning takes place until the fire reaches a HRR of 1000 Btu/sec. If the fire has to be limited to a maximum of 300 Btu/s (design objective); i.e., the fire may have caused limited damage to one or two cable trays, the time to detection and suppression can be evaluated. The following calculation evaluates if the design objective is met by the installed fire suppression and detection systems:

Input Parameters (metric units)

Ceiling Height (m):		3.1		10.0ft
Ambient Temp. (C)		20		
Growth Time (s)		400	(slow fire)	
HRR (KW)		1055	(1000 Btu/sec reached in 400 sec)	
Power Law "p"		2		
HRR (KW)		315	(Design objective to limit fire size to 300 Btu/sec)	
Radial distance to Detector (r) m		3.4		11.0ft
Radial distance to Sprinkler (r) m		1.54		5.0ft
Height of Ceiling Above Fire (H) m		2.47		8.0ft

Slow Fire Intensity Coefficient

Table 6-2.8.2
Detailed Analysis

$$Q(kW) = \alpha(kW/s^2)t^p(s)........Equation1$$

α = 0.0066 kW/s^2

Time to Reach 315 kW (300 Btu/sec)

$$t(\sec) = \sqrt{\frac{Q}{\alpha}}..............Equation2$$

t (sec) = 219Sec

Lag Time Associated with Fire Detection

The total lag time ($Lag_{plume} + Lag_{ceiling\ jet}$) can be calculated as follows:
(Mowrer, F.W., "Lag times associated with suppression and detection", Fire Technology, Vol 26, No 3, 1990)

$$t = \frac{(1.4r + 0.2H)}{(0.028\alpha H)^{1/5}}...........Equation3$$

t (sec) = 24Seconds

Time to Detection (English Units)

(See FHA for Compartment 16-1 for References and associated information)

Table 6-2.8.2
Detailed Analysis

Description:	Data
Height (H):	10
Temperature rise within ceiling jet (DT) (Temp. rise required for smoke detector activation):	18
Heat release rate (Q):	300
Convective heat release rate (Q_c) (Q*0.7):	210

Y = 18.64
X = 0.16

Activation time for smoke detector (t) = 1Second

Time for Sprinkler Activation

		Data
Length (radial distance) of sprinkler from fire source centerline (L) ft.		5
Width (Distance between beams) (W) ft.		8
Distance from fire source to ceiling (H) ft		8
Detector (sprinkler) actuation temperature (Td) (^{0}F)		165
Time Constant (TC) seconds	(Ref: Table A-6E, EPRI FIVE Document for Quick Response)	30
Heat release rate (Q) Btu/sec	(design Objective)	300
Plume temperature rise (^{0}F)	(Equation 2)	476
Ceiling Jet temperature rise factor at sprinkler	(Ref: Table 6A/6B EPRI FIVE Document)	0.4
Ambient Temperature (^{0}F)		100
Temperature rise at target (^{0}F)		190
Temperature at target (^{0}F)		290

Table 6-2.8.2 Detailed Analysis
$$\frac{t_d}{\tau} = -\ln\left[1-\frac{(Td-Ta)}{(Tg-Ta)}\right] \ldots\ldots\ldots\ldots\ldots Equation1$$ $$\Delta T(^0F) = 340\frac{Q_{eff}^{2/3}(btu/sec)}{Z_{cell}^{5/3}(ft)} \ldots\ldots\ldots Equation2$$
$t_d/\tau =$ (dimensionless actuation time, Equation 1) 0.42
Estimated time for sprinkler actuation (t_d) (seconds) **13**
<u>Conclusion</u> The above evaluation shows that for slow growing fires, time to reach 300 Btu/sec fire size is 219 seconds, whereas time to detect and activate sprinklers is no more that 50 seconds. Therefore, it can be concluded that fires in the CSR can be detected and suppressed well before critical conditions are reached. Even for medium (300 sec) and fast (150 sec) developing fires the time to reach 300 Btu/sec is 164 seconds and 82 seconds. The sprinkler system is expected to control such fires.

Table 6-2.8.3 Detailed Analysis					
Fire Area	**16-3**	**Control Building - 617' (Control Room)**			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
CASE 1A - Critical Cabinet (Panel 1-9-3) Suppressed					
MSIV Closure	IVO	HPGTET			
RCIC	RCI	HPGTET			
SORV	Subsumed within initiator IOOV				
CASE 1B - Critical Cabinet (Panel 1-9-3) Unsuppressed					
Evacuate Control Room	0.064	Human error rate for successfully performing control room abandonment procedure			
CASE 2A - Any Other Panel - Suppressed					
BOP Panels	Subsumed within initiator IMSIV				
CASE 2B - Any Other Panel - Unsuppressed					
Loss of Offsite Power	Subsumed within initiator LOSP				

Table 6-2.8.3 Detailed Analysis					
Risk Evaluation					
Initiating Event CASE 1A CASE 1B CASE 2A CASE2B	Initiating Event Frequency (IE)	CDF_{MS}	$CCDP_{MS}$ $(CDF_{MS}/IE)^1$	Ignition Frequency $(F1)^2$	CDF_{FIRE} $(CCDP_{MS}*F1)$
IOOV	4.36E-02	2.16E-06	4.95E-05	2.39E-03	1.18E-07
TT	5.09E-01	N/A	6.40E-02	8.16E-06	5.22E-07
IMSIV	5.70E-02	2.42E-07	4.24E-06	1.36E-02	5.75E-08
LOSP	6.43E-03	2.17E-07	3.37E-05	4.62E-05	1.56E-09
			TOTAL	**1.60E-02**	**7.00E-07**

Note 1: CCDP for Case 1B is the failure of remote shutdown capability probability. CCDP for Case 2B uses CCDP from LOSP.
Note 2: Ignition Frequency is calculated as follows:

Case 1A = 0.016 * 0.15* (1-0.0034)
Case 1B = 0.016 * 0.15 * 0.0034
Case 2A = 0.016 * 0.85* (1-0.0034)
Case 2B = 0.016 * 0.85*0.0034

Where Frequency of fire in Unit 1 MCR electrical cabinets is 0.016; 0.15 and 0.85 are the area ratio of panels; 0.0034 is the probability of non-suppression

Comments:

Case 1A: Suppressed fire in panel 1-9-3; one stuck open relief valve; MSIV closure, RCIC failure. This case is modeled as initiator F16_3C1A.
Case 1B: Unsuppressed fire in panel 1-9-3; Evacuate control room.
Case 2A: Suppressed fire in any other panel; Loss of condensate heat sink. This case is modeled as initiator F16_3C2A.
Case 2B: Unsuppressed fire in any other panel; this case uses CCDP from LOSP initiator.

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

Table 6-2.9.1 Detailed Analysis					
Fire Area	**25-1**	**Intake Pump Station**			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
Condenser circulating water pump resulting in low condenser vacuum	Subsumed within initiator IMSIV				
Risk Evaluation					
Initiating Event CASE 1 CASE 2	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
IMSIV	5.70E-02	9.67E-08	1.70E-06	7.77E-02	1.32E-07

Comments:
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). 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. Initiator IMSIV bounds this case (used CCDP associated with IMSIV in the base model).

Conclusion: The total core damage frequency for this scenario is less than 1E-06, however, fires in this area will be further evaluated in Table 6-2.8.1 (b)

Table 6-2.9.1 (a) Detailed Analysis					
Fire Area	**25-1**	**Intake Pump Station**			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
RHRSW and EECW Pumps –	EA, EB, EC, ED SW1A,SW1B, SW1C,SW1D, SW2A,SW2B, SW2C,SW2D	MESUPT			
Risk Evaluation					
Initiating Event Current Case Previous Case (Table 6-2.10.1 a)	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
TT	5.09E-01	1.24E-03	2.44E-03	7.43E-05	**1.81E-07**
IMSIV	5.70E-02	2.42E-07	4.24E-06	7.77E-02	**3.29E-07**
			TOTAL	**7.77E-02**	**5.10E-07**

Comments:

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 1 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 6.57E-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 2.54E-2. 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:

(6.57E-04 x 10%) + 2.54E-02 x 2.22% = 6.30E-04

Table 6-2.9.1 (a) Detailed Analysis
Only severe fires are judged to have the potential to develop to the size required to threaten both trains of RHRSW. Therefore, a severity factor of 0.118 from Table 6-1 (b) is applied. The ignition frequency is calculated as 6.30E-04 * 0.118 = 7.43E-5. This case is modeled as initiator F25_1C2 (modified from TT) in RISKMAN. The rest of frequency (7.77E-2 -7.43E-5 = 7.77E-2) has been modeled by initiator F25_1C1 (modified from IMSIV) in RISKMAN. **Conclusion:** Since the total core damage frequency for these cases is less than 1E-06, fires in this area can be screened from further consideration.

Table 6-2.9.2 Detailed Analysis					
Fire Area	**25-2**	**Pipe Tunnel**			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**	**Fire Damaged Components (Indirect Impacts)**	**Mitigating Systems Impact (Top Events)**	**Event Tree Impact**
No Mitigating Systems Impact					
Risk Evaluation					
Initiating Event Current Case Previous Case (Table 6-2.11.a)	**Initiating Event Frequency (IE)**	**CDF_{MS}**	**$CCDP_{MS}$ (CDF_{MS}/IE)**	**Ignition Frequency (F1)**	**CDF_{FIRE} ($CCDP_{MS}$*F1)**
TT	5.09E-01	1.86E-07	3.65E-07	1.09E-05	**3.96E-12**

Comments:
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 (1.09E-05), 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 turbine trip (TT) for all fires in this area. The CCDP uses that of TT initiator in the base model.

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

Table 6-2.9.3
Detailed Analysis

Fire Area	25-3	Turbine Building			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	Mitigating Systems Impact (Top Events)	Event Tree Impact	Fire Damaged Components (Indirect Impacts)	Mitigating Systems Impact (Top Events)	Event Tree Impact
Unit 2 and 3 Hydrogen Recombiners	Screened	CASE 1A - Recombiners - Unit 2 & 3 IF=1.48E-1			
Unit 1 Hydrogen Recombiners	Subsumed within Initiator TT	CASE 1B - Recombiners - Unit 1 IF=7.40E-2			
Unit 2 and 3 Lube Oil	Screened	CASE 2A - Lube Oil Fire - Unit 2 & 3 IF=2.4E-2			
Unit 1 Lube Oil	Subsumed within Initiator TT	CASE 2B - Lube Oil Fire - Unit 1 IF=1.2E-2			
Turbine Operating Deck	Screened	CASE 3A - Turbine Deck - Unit 2 & 3 IF=2.68E-2		CASE 3A-1 - Turbine Deck - Unit 2 & 3 Minor Fire IF=2.68E-2*0.58=1.55E-2	
				CASE 3A-2 - Turbine Deck - Unit 2 & 3 Severe Fire IF=2.68E-2*0.42=1.13E-2	
Turbine Operating Deck	Subsumed within Initiator LOSP	CASE 3B - Turbine Deck - Unit 1 IF=1.34E-2			
Other Areas	Screened	CASE 4 - Other Areas Fire 2.61E-1			

Table 6-2.9.3
Detailed Analysis

Risk Evaluation

Initiating Event	Initiating Event Frequency (IE)	CDF_{MS}	$CCDP_{MS}$ (CDF_{MS}/IE)	Ignition Frequency (F1)	CDF_{FIRE} ($CCDP_{MS}$*F1)
N/A (1A)	N/A	N/A	N/A	1.48E-01	
TT (1B)	5.09E-01	1.40E-07	2.75E-07	7.40E-02	2.03E-08
N/A (2A)	N/A	N/A	N/A	2.40E-02	
TT (2B)	5.09E-01	1.40E-07	2.75E-07	1.20E-02	3.30E-09
N/A (3A-1)	N/A	N/A	N/A	1.55E-02	
N/A (3A-2)	N/A	N/A	N/A	1.13E-02	
LOSP (3B)	6.43E-03	2.17E-07	3.37E-05	1.34E-02	4.52E-07
N/A (4)	N/A	N/A	N/A	2.61E-01	
			TOTAL	**5.59E-01**	**4.76E-07**

Comments:
The following cases describe the various fire scenarios postulated for the Turbine Building:
CASE 1A: Fire affects Unit 2 and 3 recombiners. Plant trip not expected.
CASE 1B: Fire affects Unit 1 recombiners. Plant trip is assumed. RISKMAN initiator is F25_3C1B.
CASE 2A: Lube oil fire in Units 2 and 3. Plant trip is assumed.
CASE 2B: Lube oil fire in Unit 1. Plant trip is expected. RISKMAN initiator is F25_3C2B.
CASE 3A-1: Minor fire on the Turbine Deck of Units 2 and 3. Plant trip is not expected for Unit 1.
CASE 3A-2: Severe fire on the Turbine Deck of Units 2 and 3. Plant trip is not expected for Unit 1.
CASE 3B: Fire on the Turbine Deck of Unit 1. Plant trip expected, use CCDP associated with LOSP in the base model.
CASE 4: Fire in all other areas. Plant trip not expected.

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

Table 6-2.10 Detailed Analysis					
Fire Area	N/A	Yard Areas			
PSA MODEL IMPACTS DUE TO FIRE DAMAGE					
Fire damaged Components (Direct Impacts)	Mitigating Systems Impact (Top Events)	Event Tree Impact	Fire Damaged Components (Indirect Impacts)	Mitigating Systems Impact (Top Events)	Event Tree Impact
CASE 1 - Yard fire propagating to Turbine Building					
Yard Transformers - loss of condensate heat sink, feedwater	Subsumed within initiator IMSIV				
CASE 2 - Yard Fire					
Yard Transformers - Modeled as Loss of Off-site Power	Subsumed within initiator LOSP				
CASE 3 - Other Yard Transformers					
Other Yard Transformers	Subsumed within initiator TT				
Risk Evaluation					
Initiating Event CASE 1 CASE 2 CASE 3	Initiating Event Frequency (IE)	CDF_{MS}	$CCDP_{MS}$ (CDF_{MS}/IE)	Ignition Frequency (F1)	CDF_{FIRE} ($CCDP_{MS}$*F1)
IMSIV	5.70E-02	9.67E-08	1.70E-06	2.60E-03	4.41E-09
LOSP	6.43E-03	2.17E-07	3.37E-05	5.10E-03	1.72E-07
TT	5.09E-01	1.86E-07	3.65E-07	2.60E-02	9.48E-09
			TOTAL	3.37E-02	1.77E-07

Comments:
Case 1: Yard transformer fire propagating to the Turbine Building; reactor trips and MSIV closes. Use CCDP for IMSIV in base model.
Case 2: Yard transformer fire resulting in loss of offsite power. Use CCDP for LOSP in the base model.
Case 3: Other yard transformers; manual shutdown. Use CCDP for TT in the base model.
As a sensitivity analysis, the single unit ignition frequencies were multiplied by 3, assuming fires in Units 2 and 3 yard areas will also have a similar impact on Unit 1. The new ignition frequencies for the above 3 cases will then be 7.80E-3, 1.53E-2 and 7.80E-2 respectively. The total ignition frequency will then be 1.01E-01. The resulting CDF is 5.58E-07. This value is also less than the screening value, therefore, the yard areas can remain screened.
Conclusion: Since the total core damage frequency for these cases is less than 1E-06, fires in this area can be screened from further consideration.

7.0 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 significant fire-induced vulnerabilities associated with the continued operation of Browns Ferry Unit 1.

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 1 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, it is not the intent here to sum up the fire induced 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 calculation 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 1 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				
Fire Area / Zone	Description	Qualitative Analysis (Phase I)	Quantitative Analysis (Phase II)	
			Initial Screening	Detailed Analysis
1-1	Unit 1 Reactor Building, 519' through 565' Elevations (West side of Torus Area and Main Floor)	See Note 1	6.08E-02	See Note 2
1-2	Unit 1 Reactor Building, 519' through 565' Elevations (East side of Torus Area and Main Floor)	See Note 1	3.52E-02	See Note 2
1-3	Unit 1 Reactor Building, 593' Elevation, North Side	See Note 1	2.32E-02	See Note 2
1-4	Unit 1 Reactor Building, 593' Elevation, South Side and RHR Heat Exchanger Rooms	See Note 1	2.12E-02	See Note 2
1-5	Unit 1 Reactor Building, 621' Elevation and North Side of 639' Elevations	See Note 1	4.88E-02	See Note 2
1-6	Unit 1 Reactor Building, South Side of 639' Elevation	See Note 1	3.08E-02	See Note 2
2	Unit 2 Reactor Building	See Note 1	1.27E-01	2.23E-07
3	Unit 3 Reactor Building	See Note 1	1.26E-01	9.36E-08
4	4kV Shutdown Board Room B (Unit 1 Reactor Building, 593' Elevation)	See Note 1	3.14E-04	7.64E-07
5	4kV Shutdown Board Room A and 250V Battery Room (Unit 1 RB, EL 621')	See Note 1	1.99E-04	7.93E-07
6	480V Shutdown Board Room 1A (Unit 1 Reactor Building, 621' Elevation)	See Note 1	1.11E-07	
7	480V Shutdown Board Room 1B (Unit 1 Reactor Building, 621' Elevation)	See Note 1	2.01E-06	3.56E-07
8	4kV Shutdown Board Room D (Unit 2 Reactor Building, 593' Elevation)	See Note 1	1.02E-07	
9	4kV Shutdown Board Room C and 250V Battery Room (Unit 2 RB, EL 621')	See Note 1	1.38E-05	3.36E-07
10	480V Shutdown Board Room 2A (Unit 2 Reactor Building, 621' Elevation)	See Note 1	1.15E-08	
11	480V Shutdown Board Room 2B (Unit 2 Reactor Building, 621' Elevation)	See Note 1	4.63E-09	
12	Shutdown Board Room F (Unit 3 Reactor Building, 593' Elevation)	See Note 1	1.15E-08	
13	Shutdown Board Room E (Unit 3 Reactor Building, 621' Elevation)	See Note 1	5.71E-09	
14	480V Shutdown Board Room 3A (Unit 3 Reactor Building, 621' Elevation)	See Note 1	5.11E-09	
15	480V Shutdown Board Room 3B (Unit 3 Reactor Building, 621' Elevation)	See Note 1	5.45E-09	

Table 7-1 Summary of Results				
Fire Area / Zone	Description	Qualitative Analysis (Phase I)	Quantitative Analysis (Phase II)	
			Initial Screening	Detailed Analysis
16-1	Control Building - 593' Elevation	See Note 1	3.78E-02	2.75E-08
16-2	Control Building - 606' (Cable Spreading Room)	See Note 1	1.20E-02	5.35E-07
16-3	Control Building - 617' (Control Room)	See Note 1	6.92E-02	7.00E-07
17	Unit 1 Battery and Battery Board Room, Control Building 593' Elevation	See Note 1	2.40E-07	
18	Unit 2 Battery and Battery Board Room, Control Building 593' Elevation	See Note 1	1.54E-08	
19	Unit 3 Battery and Battery Board Room, Control Building 593' Elevation	See Note 1	2.74E-08	
20	Unit 1 and 2 Diesel Generator Building	See Note 1	4.77E-08	
21	Unit 3 Diesel Generator Building	See Note 1	1.15E-07	
22	4kV Shutdown Board Room 3EA and 3EB, 583' Elevation, Unit 3 Diesel Generator Building	See Note 1	7.35E-09	
23	4kV Shutdown Board Room 3EC and 3ED, 583' Elevation, Unit 3 Diesel Generator Building	See Note 1	1.08E-08	
24	4kV Bus Tie Board Room, 565' Elevation, Unit 3 Diesel Generator Building	See Note 1	6.49E-07	
25-1	Intake Pump Station	See Note 1	7.77E-02	5.10E-07
25-2	Pipe Tunnel	See Note 1	1.09E-05	3.96E-12
25-3	Turbine Building	See Note 1	5.59E-01	4.76E-07
N/A	Yard Area (See Note 3)	See Note 1		1.77E-07

Notes:

1. *No areas were screened in Phase I.*

2. *The Unit 1 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:*

RHR Pumps 1A and 1C	*1.29E-07 (Fire Source 1-1-5)*
240V Lighting Transformer TL1A	*5.76E-07 (Fire Source 1-5-2)*

VFD 1A (Panel) 1.70E-07 (Fire Source 1-6-2)
VFD 1B (Panel) 1.70E-07 (Fire Source 1-6-3)

For completeness, unqualified cables were also included in Table 6-2.1 (d), with an upper bound core damage frequency of 1.5E-07.

3. *No fire area or zone was assigned to the yard area, though, for completeness, potential fires in this area are evaluated in Section 6.2.10. This evaluation gave a total upper bound core damage frequency for this area of 1.77E-07.*

Insights

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.
- 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 four cases (4kV shutdown board rooms A, B, 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. 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.

Fire Area/Zone CDF Summation

Due to the progressive nature of this screening methodology, the various fire area/zones have been screened out at different level of detail. Some areas have been screened assuming engulfing fire, whereas others have been screened by considering the fire severity factors or by taking credit for automatic or manual suppression. Therefore, the CDF values for each area/zone/fire source may not represent a uniform means of comparison. Keeping this in mind, the following table sums up the CDF values for each fire zone/area screened out during the detailed analysis.

Table 7-2 Summary of Results			
Fire Area / Zone	Description	Fire Induced CDF	Sub Total
1-1-1	480V RMOV Board 1C	3.55E-08	
1-1-2	480V RB Vent Board 1B	1.42E-09	
1-1-3	250 RMOV Board 1C	1.45E-09	
1-1-4	Core Spray Pumps 1A and 1C	1.05E-08	
1-1-5	RHR Pumps 1A and 1C	1.29E-07	
1-1-6	RCIC Pump	5.27E-09	
1-1-7	HPCI Pump	2.05E-08	
1-1-8	1-LPNL-925-0340 ES Div I & II Panel	7.03E-08	2.74E-07
1-2-1	Core Spray Pumps 1B and 1D	1.75E-09	
1-2-2	RHR Pumps 1B and 1D	3.58E-08	3.76E-08
1-3-1	RCW Pump 1A	9.15E-10	9.15E-10
Fire Zone 1-4	No Credible Fire Induced CDF	0.00E+00	0.00E+00
1-5-1	240V Lighting Board 1A	4.92E-09	
1-5-2	240V Lighting Transformer TL1A	5.76E-07	
1-5-3	4kV to 480V Transformer TS1A	2.58E-09	
1-5-4	4kV to 480V Transformer TS1B	5.71E-10	
1-5-5	4160V RPT Board 1-1 (Panel 1 and Panel 2)	7.59E-08	
1-5-6	4160V RPT Board 1-2 (Panel 1 and Panel 2)	1.89E-09	
1-5-7	RCIC Control Panel 1-25-31	4.92E-09	
1-5-8	Panel 25-3 (Filter Demin)	0.00E+00	6.67E-07
1-6-1	4kV to 480V Emergency Transformer TS1E (Oil)	2.86E-08	
1-6-2	VFD 1A (Panel)	1.70E-07	
1-6-3	VFD 1B (Panel)	1.70E-07	3.69E-07
2	Unit 2 Reactor Building	2.23E-07	2.23E-07
3	Unit 3 Reactor Building	9.36E-08	9.36E-08
4	4kV Shutdown Board Room B (Unit 1 Reactor Building, 593' Elevation)	7.64E-07	7.64E-07
5	4kV Shutdown Board Room A and 250V Battery Room (Unit 1 RB, EL 621')	7.93E-07	7.93E-07
6	480V Shutdown Board Room 1A (Unit 1 Reactor Building, 621' Elevation)	1.11E-07	1.11E-07
7	480V Shutdown Board Room 1B (Unit 1 Reactor Building, 621' Elevation)	3.56E-07	3.56E-07
8	4kV Shutdown Board Room D (Unit 2 Reactor Building, 593' Elevation)	1.02E-07	1.02E-07
9	4kV Shutdown Board Room C and 250V Battery Room (Unit 2 RB, EL 621')	3.36E-07	3.36E-07

Table 7-2
Summary of Results

Fire Area / Zone	Description	Fire Induced CDF	Sub Total
10	480V Shutdown Board Room 2A (Unit 2 Reactor Building, 621' Elevation)	1.15E-08	1.15E-08
11	480V Shutdown Board Room 2B (Unit 2 Reactor Building, 621' Elevation)	4.63E-09	4.63E-09
12	Shutdown Board Room F (Unit 3 Reactor Building, 593' Elevation)	1.15E-08	1.15E-08
13	Shutdown Board Room E (Unit 3 Reactor Building, 621' Elevation)	5.71E-09	5.71E-09
14	480V Shutdown Board Room 3A (Unit 3 Reactor Building, 621' Elevation)	5.11E-09	5.11E-09
15	480V Shutdown Board Room 3B (Unit 3 Reactor Building, 621' Elevation)	5.45E-09	5.45E-09
16-1	Control Building - 593' Elevation	2.75E-08	2.75E-08
16-2	Control Building - 606' (Cable Spreading Room)	5.35E-07	5.35E-07
16-3	Control Building - 617' (Control Room)	7.00E-07	7.00E-07
17	Unit 1 Battery and Battery Board Room, Control Building 593' Elevation	2.40E-07	2.40E-07
18	Unit 2 Battery and Battery Board Room, Control Building 593' Elevation	1.54E-08	1.54E-08
19	Unit 3 Battery and Battery Board Room, Control Building 593' Elevation	2.74E-08	2.74E-08
20	Unit 1 and 2 Diesel Generator Building	4.77E-08	4.77E-08
21	Unit 3 Diesel Generator Building	1.15E-07	1.15E-07
22	4kV Shutdown Board Room 3EA and 3EB, 583' Elevation, Unit 3 Diesel Generator Building	7.35E-09	7.35E-09
23	4kV Shutdown Board Room 3EC and 3ED, 583' Elevation, Unit 3 Diesel Generator Building	1.08E-08	1.08E-08
24	4kV Bus Tie Board Room	6.49E-07	6.49E-07
25-1	Intake Pump Station	5.10E-07	5.10E-07
25-2	Pipe Tunnel	3.96E-12	3.96E-12
25-3	Turbine Building	4.76E-07	4.76E-07
N/A	Yard Area	1.77E-07	1.77E-07
	Total Unit 1 CDF		**7.71E-06**

8.0 NEW AND REMAINING ISSUES (PHASE III)

(This section is taken directly from the original Unit 2 submittal and has not been updated. It is included in this calculation for completeness of the IPEEE Fire analysis and is for information only.)

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 1 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 Seismic Degradation of Fire Suppression Systems

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.

Response *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 (6.1.3).*

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.

Response *Fire brigade training records are required to be maintained, as specified in Volume 2 of the Browns Ferry Fire Protection Report, Section IV.*

8.2.4 Total Environment Equipment Survival

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 Potential Adverse Effects on Plant Equipment by Combustion Products

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).

Response *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 modeling 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 modeling 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 6.2.*

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.

<u>Response</u> *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 1 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 6.2.8.1.

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.

Response *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 1 Reactor Building fire sources (see Section 6.2).*

The Fire Events Database (NSAC/178L) was used as a basis for fire size for fires analyzed in Section 6.1. 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 6.2.8.1.

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 judgment, 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 ERPI 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.*

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ATTACHMENT A
HEAT RELEASE RATES (HRR)

Electrical Cabinets

Sandia National Laboratories (SNL) and Technical Research Center of Finland (VIT) have conducted tests in the measurement of Heat release rates for closed panels. In the SNL tests closed/ventilated panel HRR values up to 265 BTU/s were measured (i.e., Considering SNL Scoping Test 10) and in the VTT Finland tests, maximum HRR values of up to 380 BTU/s were measured (i.e., considering VTT panel test #1) (References 30, 31 and 32). The maximum HRR of a closed but ventilated electrical panel is assumed to be 190 BTU/s, the midrange of the available test data. Refer to question number 4 and its response in Reference 43 regarding HRR for electrical cabinets. Also Reference 6, Section 4.12 recommends using a HRR value of 200 kW (190 Btu/sec) for cabinets containing non-qualified cables.

Description	Number of Vertical Section	Combustible Loading (Btu)	Heat Release Rate (Btu/sec)	Comments
480V RMOV Bd 1C	11	6.16E+6	190	No vent openings
480V RB Vent Bd 1B	12	6.72E+6	190	No vent openings
250v RMOV Bd 1C	10	6.72E+6	190	No vent openings
1-LPNL-025-0031	2	1.40E+6	190	No vent openings
4KV RPT Bd 1-1, 1-2	2	1.40E+6	190	Vent openings
Panel 25-3	1	7.00E+5	190	No vent openings
Panel 25-9	1	7.0E+4	190	No vent opening
240V Lighting Bd 1A	10	5.60E+5	190	No vent openings

Dry Type Transformers

Dry type cast-resin transformers hardly experience any escalation of fire beyond the original source. Also there is no comparable risk of explosion. Transformers will be considered similar to a closed non-qualified electrical cabinet. HRR will be computed based on the combustible loading of the transformer at 4E-4 x Fuel load (Btu/sec) or 400 Btu/sec for 1 million Btu loading or 40 Btu/sec per 100,000 Btu (Reference 5, Table E-1 and Section E.2) but no less than 95 Btu/sec.

Description	Combustible Loading (Btu)	HRR Criteria	Heat Release Rate (Btu/sec)
4kV-480V Transformer TS1A	2.8E+5	40 Btu/sec per 100,000 Btu (not less than 95 Btu/sec)	112
4kV-480V Transformer TS1B	2.8E+5	40 Btu/sec per 100,000 Btu (not less than 95 Btu/sec)	112
240V Lighting Transformer	5.6E+5	40 Btu/sec per 100,000 Btu (not less than 95 Btu/sec)	224

Electrical Motors - Ventilation Subsystems, Pumps

For combustible material involves electrical windings, heat release rates should be smaller than a small cabinet fire (< 65 Btu/sec) (Reference 5). HRR will be computed based on the combustible loading of the motor at 4E-4 x Fuel load (Btu/sec) or 400 Btu/sec for 1 million Btu loading or 40 Btu/sec per 100,000 Btu.

For RBCCW pumps 1A and 1B, and RCW pumps 1A and 1B, the pump motors have power less than 100 H.P. each. Hence heat release rate for small electric fire (i.e., 190 BTU/sec) is used for each pump.

SLC Pumps A and B (Unit 1 Reactor Building El 639-north side)		
Oil:		
Spill Area	54 ft^2 per gal	(Reference 1, Attachment 10.4, Table 3 for Pennzoil 30-HD
Pool area for ½ pint spill	3.3ft^2	(45 ft /gal * 1 gal/16 pints)
Unit Heat Release Rate	135 Btu/s ft^2	Reference 5, Table E-1 (pumps)
Peak Heat Release Rate	450 Btu/sec	(135 * 3.3)
Motor:		
Combustible Loading	2.80E+5 Btu	
HRR per unit	40 Btu/sec per 100,000 Btu	
Peak Heat Release Rate	112 Btu/sec	

ATTACHMENT B
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 taken from References 4 and 27, related system flow diagrams and plant walkdowns and is summarized in Table B-1.
- Cables heat of combustion - This information was taken from References 4 and 27 and is summarized in Table B-2.

When performing ignition frequency calculation for Unit 1, efforts were made to assure completeness of fire ignition source equipment identification. An EXCEL table was developed based on the Unit 1 cable loading file (an electronic version of Reference 27). For each piece of equipment in that file, proper disposition was provided, i.e., screened based on certain criteria (small combustible loading, or non EPRI source equipment type, etc.), or retained for inclusion in the source equipment tally. For details of screening and categorization of the Unit 1 equipment, please refer to the EXCEL completeness table (Reference 44).

In addition, the Unit 1 cable loading mini-calculation (Reference 27) was processed electronically, to identify all the cable tray loadings in various Unit 1 areas, and added to the total plant cable loading (this is reflected in Table B-2).

The generic fire frequencies and weighting factors used in these calculations are in accordance with Reference 1. Where specific frequencies were not provided for plant areas, such as computer rooms, mechanical equipment room, etc., conservative assumptions were made.

The performance of these calculations consist of two main steps:

- The fire ignition frequency that can be assigned to specific plant areas (plant locations), such as switchgear area, is allocated to similar areas within the plant.
- The identified plant wide components, e.g., battery chargers, transformers, etc., 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 shown at the end of Attachment B (after Table B-2).

Table B-1
Plant Wide Components

Location	Unit	Number of Components	Fire Area	Comments
Fire Protection Panels				
Reactor Building	1	4	1	Based on MD-N0999-2003-0046 Rev. 2 (Unit 1 Loading Mini Cal)
Reactor Building	2	3	2	Includes only operating panels
Reactor Building	3	3	3	Includes only operating panels
Diesel Generator Building	1 /2	2	20	FP panel and CO_2 switch panel
Diesel Generator Building	3	2	21	FP panel and CO_2 switch panel
Intake Pump Station	0	1	25-1	
Control Building	0	6	16	Includes 1 on El 617, 2 on El 606, 1 on El 593, 1 in process computer room and 1 in the Unit 3 computer room.
Turbine Building	0	4	25-3	
Total		25		
Battery Chargers				
Diesel Generator Building	1 /2	8	20	
Diesel Generator Building	3	9	21	
Battery & Battery Bd. Room	1	3	17	250V, 48V and 24V neutron
Battery & Battery Bd. Room	2	3	18	250V, 48V and 24V neutron
Battery & Battery Bd. Room	3	3	19	250V, 48V and 24V neutron
Communication Battery Board Room	3	3	16	
Shutdown Board Room C	2	2	9	
Shutdown Board Room A	1	2	5	Based on MD-N0999-2003-0046 Rev. 2 (Unit 1 Loading Mini Cal)
Turbine Building	1	1	25-3	El 586
Total		34		
Transformers				
Battery & Battery Bd. Room		2	17	
Diesel Generator Building		3	20	
Electrical Board Room El 621	3	1	13	Dry type transformer
Electrical Board Room El 593	3	1	12	Dry type transformer
Shutdown Bd Room A El 621	1	1	5	
Reactor Building El 541	1	1	1	Dry type transformer
Reactor Building El 519	1	1	1	Dry type transformer
Reactor Building El 519	3	1	3	
Reactor Building El 593	2	1	2	Unit 2 preferred AC Transformer
Reactor Building El 621	1	3	1	4kV/480V & 240V Lighting Bd
Reactor Building El 621	2	3	2	4kV/480V & 240V Lighting Bd
Reactor Building El 621	3	3	3	4kV/480V & 240V Lighting Bd

Table B-1
Plant Wide Components

Location	Unit	Number of Components	Fire Area	Comments
Reactor Building El 639	1	1	1	4kV/480V Transformer
Reactor Building El 639	2	2	2	4kV/480V & 240V Lighting Bd
Reactor Building El 639	3	1	3	4kV/480V Transformer
Turbine Building El 617	0	1	25-3	GE Transformer
Turbine Building El 604	0	18	25-3	
Turbine Building El 584	0	1	25-3	
Intake Pump Station	0	3	25-1	
Total		48		
Air Compressors				
Reactor Building	1	0	1	
Reactor Building	2	3	2	
Reactor Building	3	3	3	
Turbine Building El 604	0	6	25-3	
Turbine Building El 565	0	10	25-3	
Control Building El 606	0	1	16-2	Mechanical Equipment Room
Total		23		
Ventilation Subsystems				
Reactor Building	1	19	1	
Reactor Building	2	20	2	Includes 4 for SDBR C/D
Reactor Building	3	21	3	Includes 4 for SDBR E/F
Turbine Building	1	36	25-3	Includes 1 booster fan
Turbine Building	2	38	25-3	Includes AHU for CB
Turbine Building	3	37	25-3	Includes record storage Bd. Rm.
Control Building	0	30	16	El 617
		2	16	El 606
		1	4	El 593
		1	12	El 593
		3	16	El 593
		4	5	El 621
		4	9	El 621
Radwaste		17		
Diesel Generator Building	1/2	20	20	
Diesel Generator Building	3	28	21	
Intake Pump Station		8	25-1	
Total		289		

Table B-1 Plant Wide Components				
Location	**Unit**	**Number of Components**	**Fire Area**	**Comments**
RPS MG Sets				
Control Building	0	6	16	
Turbine Building	0	2	25-3	
Reactor Building 639	1	0	1	U1 Recirc MG Sets Removed
Reactor Building 639	2	0	2	U2 Recirc MG Sets Removed (DCN 50869 Rev B)
Reactor Building 639	3	0	3	U3 Recirc MG Sets Removed (DCN 51312 Rev A)
Reactor Building 621	1	0	1	U1 LPCI MG Sets Removed
Reactor Building 621	2	4	2	
Reactor Building 621	3	4	3	
MG Set Room	1	1	17	
MG Set Room	2	1	18	
MG Set Room	3	1	19	
Total		19		
Offgas Recombiners				
Unit 1	1	1	25-3	
Unit 2	2	1	25-3	
Unit 3	3	1	25-3	
Total		3		

Table B-2
Cables Heat of Combustion
(References 4 and 27)

DESCRIPTION	BTU	TOTAL (BTU)
REACTOR BUILDING, UNIT 1 (FIRE AREA 1)		
EL 639	205,503,311	
EL 621	340,372,052	
EL 593	951,726,437	
EL 565	829,992,355	
		2,327,594,155
REACTOR BUILDING, UNIT 2 (FIRE AREA 2)	1,239,318,566	1,239,318,566
REACTOR BUILDING, UNIT 3 (FIRE AREA 3) (Assume combustible heat load similar to unit 2)	1,239,318,566	1,239,318,566
4kV SHUTDOWN BOARD ROOM B (UNIT 1 REACTOR BLDG, EL 593', FIRE AREA 4)	16,050,038	16,050,038
4kV SHUTDOWN BOARD ROOM A AND 250V BATTERY ROOM (UNIT 1 REACTOR BLDG, EL 621', FIRE AREA 5)	66,689,320	66,689,320
480V SHUTDOWN BOARD ROOM 1A (UNIT 1 REACTOR BLDG, EL 621', FIRE AREA 6)	11,375,057	11,375,057
4 KV SHUTDOWN BD C AND 250V BATTERY ROOM (UNIT 2 REACTOR BLDG, EL 612', FIRE AREA 9)	5,000,000	5,000,000
CONTROL BUILDING (FIRE AREA 16)		
Fire Zone 16-3 Control Room (Control Bldg., EL 617')		
Unit 1/2 main control room	70,340,799	
Mechanical equipment room EL 617	30,000,000	
Fire Zone 16-2 Cable Spreading Room (Control Bldg. EL 606')		
Cable spreading room A	799,723,278	
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	
Fire Zone 16-1 Control Bldg. EL 593'		
Auxiliary instrument room 1	61,492,871	
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	
		1,680,366,750
BATTERY BOARD ROOM 2 (FIRE AREA 18)	506,520	506,520
DIESEL GENERATOR BUILDING UNIT 1/2 (FIRE AREA 20)		
Pipe tunnel	21,093,135	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	15,279,390
4KV SHUTDOWN BD 3EC AND 3ED (FIRE AREA 23)	14,582,610	14,582,610
4KV BUS TIE BD (FIRE AREA 24)	7,614,810	7,614,810
TURBINE BUILDING (FIRE AREA 25)		
Unit 1,2,3 turbine building (fire compt. 25-3)	6,264,112,490	
Pipe tunnel (fire compt. 25-2)	5,000,000	

Table B-2 Cables Heat of Combustion (References 4 and 27)		
DESCRIPTION	**BTU**	**TOTAL (BTU)**
Intake pump station (fire compt. 25-1) Radwaste building	271,630,557 167,463,262	6,708,206,309
TOTAL CABLES COMBUSTIBLE LOAD		**13,388,182,616**
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.		

PHASE II (STEP 1) EVALUATION
PLANT LOCATION: REACTOR BUILDING (FIRE ZONE 1-1)

Components	Generic Fire Frequency	Location Weighting Factor (WL)	Number of Components in Fire Area	Total Number in All Plant locations(1)	Ignition Source Weighting factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location:						
Electric Cabinets	5.70E-02	1	4	11	3.64E-01	2.07E-02
Pumps	3.50E-02	1	6	11	5.45E-01	1.91E-02
Plant Wide:						
Fire Protection Panel	1.30E-03	3	3	25	1.20E-01	4.68E-04
RPS MG Set	3.40E-03	3	0	19	0.00E+00	0.00E+00
Non-Qualified Cable	8.40E-03	3	415	13388	3.10E-02	7.81E-04
Non-Qualified JB	1.30E-03	3	415	13388	3.10E-02	1.21E-04
Transformers	1.40E-02	3	0	48	0.00E+00	0.00E+00
Battery chargers	5.50E-03	3	0	34	0.00E+00	0.00E+00
Air Compressors	5.90E-03	3	0	23	0.00E+00	0.00E+00
Vent. Subsystems	1.60E-02	3	4	289	1.38E-02	6.64E-04
			Sum of Ignition Sources Weighting Factor for Transient Source	Total Number of Compartments		
Transients	3.60E-02	3	5	34	1.47E-01	1.59E-02
Cable Fire-Welding	1.30E-03	3	1	34	2.94E-02	1.15E-04
Transient Fire-Welding	3.40E-02	3	1	34	2.94E-02	3.00E-03
				TOTAL		6.08E-02

Note: (1) The electric cabinet and pump count in unit 1 fire zones includes only significant fire sources (i.e., those components that have the potential to develop credible fire scenarios). Other similar non-significant fire sources have not been considered in the count to avoid unnecessary assignment of fire frequency to those components. This approach is different from unit 2/3 calculations which considered all components (significant and non-significant) in the count.

FIRE AREA /COMPARTMENTATION IGNITION FREQUENCY
PHASE II (STEP 1) EVALUATION
PLANT LOCATION: REACTOR BUILDING (FIRE ZONE 1-2)

Components	Generic Fire Frequency	Location Weighting Factor (WL)	Number of Components in Fire Area	Total Number in All Plant locations(1)	Ignition Source Weighting factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location:						
Electric Cabinets	5.70E-02	1	0	11	0.00E+00	0.00E+00
Pumps	3.50E-02	1	4	11	3.64E-01	1.27E-02
Plant Wide:						
Fire Protection Panel	1.30E-03	3	0	25	0.00E+00	0.00E+00
RPS MG Set	3.40E-03	3	0	19	0.00E+00	0.00E+00
Non-Qualified Cable	8.40E-03	3	415	13388	3.10E-02	7.81E-04
Non-Qualified JB	1.30E-03	3	415	13388	3.10E-02	1.21E-04
Transformers	1.40E-02	3	2	48	4.17E-02	1.75E-03
Battery chargers	5.50E-03	3	0	34	0.00E+00	0.00E+00
Air Compressors	5.90E-03	3	0	23	0.00E+00	0.00E+00
Vent. Subsystems	1.60E-02	3	5	289	1.73E-02	8.30E-04
			Sum of Ignition Sources Weighting Factor for Transient Source	Total Number of Compartments		
Transients	3.60E-02	3	5	34	1.47E-01	1.59E-02
Cable Fire-Welding	1.30E-03	3	1	34	2.94E-02	1.15E-04
Transient Fire-Welding	3.40E-02	3	1	34	2.94E-02	3.00E-03
				TOTAL		**3.52E-02**

Note: (1) The electric cabinet and pump count in unit 1 fire zones includes only significant fire sources (i.e., those components that have the potential to develop credible fire scenarios). Other similar non-significant fire sources have not been considered in the count to avoid unnecessary assignment of fire frequency to those components. This approach is different from unit 2/3 calculations which considered all components (significant and non-significant) in the count.

FIRE AREA /COMPARTMENTATION IGNITION FREQUENCY
PHASE II (STEP 1) EVALUATION
PLANT LOCATION: REACTOR BUILDING (FIRE ZONE 1-3)

Components	Generic Fire Frequency	Location Weighting Factor (WL)	Number of Components in Fire Area	Total Number in All Plant locations(1)	Ignition Source Weighting factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location:						
Electric Cabinets	5.70E-02	1	0	11	0.00E+00	0.00E+00
Pumps	3.50E-02	1	1	11	9.09E-02	3.18E-03
Plant Wide:						
Fire Protection Panel	1.30E-03	3	0	25	0.00E+00	0.00E+00
RPS MG Set	3.40E-03	3	0	19	0.00E+00	0.00E+00
Non-Qualified Cable	8.40E-03	3	238	13388	1.78E-02	4.48E-04
Non-Qualified JB	1.30E-03	3	238	13388	1.78E-02	6.93E-05
Transformers	1.40E-02	3	0	48	0.00E+00	0.00E+00
Battery chargers	5.50E-03	3	0	34	0.00E+00	0.00E+00
Air Compressors	5.90E-03	3	0	23	0.00E+00	0.00E+00
Vent. Subsystems	1.60E-02	3	3	289	1.04E-02	4.98E-04
			Sum of Ignition Sources Weighting Factor for Transient Source	Total Number of Compartments		
Transients	3.60E-02	3	5	34	1.47E-01	1.59E-02
Cable Fire-Welding	1.30E-03	3	1	34	2.94E-02	1.15E-04
Transient Fire-Welding	3.40E-02	3	1	34	2.94E-02	3.00E-03
				TOTAL		2.32E-02

Note: (1) The electric cabinet and pump count in unit 1 fire zones includes only significant fire sources (i.e., those components that have the potential to develop credible fire scenarios). Other similar non-significant fire sources have not been considered in the count to avoid unnecessary assignment of fire frequency to those components. This approach is different from unit 2/3 calculations which considered all components (significant and non-significant) in the count.

FIRE AREA /COMPARTMENTATION IGNITION FREQUENCY
PHASE II (STEP 1) EVALUATION
PLANT LOCATION: REACTOR BUILDING (FIRE ZONE 1-4)

Components	Generic Fire Frequency	Location Weighting Factor (WL)	Number of Components in Fire Area	Total Number in All Plant locations[1]	Ignition Source Weighting factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location:						
Electric Cabinets	5.70E-02	1	0	11	0.00E+00	0.00E+00
Pumps	3.50E-02	1	0	11	0.00E+00	0.00E+00
Plant Wide:						
Fire Protection Panel	1.30E-03	3	1	25	4.00E-02	1.56E-04
RPS MG Set	3.40E-03	3	0	19	0.00E+00	0.00E+00
Non-Qualified Cable	8.40E-03	3	714	13388	5.33E-02	1.34E-03
Non-Qualified JB	1.30E-03	3	714	13388	5.33E-02	2.08E-04
Transformers	1.40E-02	3	0	48	0.00E+00	0.00E+00
Battery chargers	5.50E-03	3	0	34	0.00E+00	0.00E+00
Air Compressors	5.90E-03	3	0	23	0.00E+00	0.00E+00
Vent. Subsystems	1.60E-02	3	3	289	1.04E-02	4.98E-04
			Sum of Ignition Sources Weighting Factor for Transient Source	Total Number of Compartments		
Transients	3.60E-02	3	5	34	1.47E-01	1.59E-02
Cable Fire-Welding	1.30E-03	3	1	34	2.94E-02	1.15E-04
Transient Fire-Welding	3.40E-02	3	1	34	2.94E-02	3.00E-03
				TOTAL		2.12E-02

Note: (1) The electric cabinet and pump count in unit 1 fire zones includes only significant fire sources (i.e., those components that have the potential to develop credible fire scenarios). Other similar non-significant fire sources have not been considered in the count to avoid unnecessary assignment of fire frequency to those components. This approach is different from unit 2/3 calculations which considered all components (significant and non-significant) in the count.

FIRE AREA /COMPARTMENTATION IGNITION FREQUENCY
PHASE II (STEP 1) EVALUATION
PLANT LOCATION: REACTOR BUILDING (FIRE ZONE 1-5)

Components	Generic Fire Frequency	Location Weighting Factor (WL)	Number of Components in Fire Area	Total Number in All Plant locations(1)	Ignition Source Weighting factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location:						
Electric Cabinets	5.70E-02	1	5	11	4.55E-01	2.59E-02
Pumps	3.50E-02	1	0	11	0.00E+00	0.00E+00
Plant Wide:						
Fire Protection Panel	1.30E-03	3	0	25	0.00E+00	0.00E+00
RPS MG Set	3.40E-03	3	0	19	0.00E+00	0.00E+00
Non-Qualified Cable	8.40E-03	3	340	13388	2.54E-02	6.40E-04
Non-Qualified JB	1.30E-03	3	340	13388	2.54E-02	9.90E-05
Transformers	1.40E-02	3	3	48	6.25E-02	2.63E-03
Battery chargers	5.50E-03	3	0	34	0.00E+00	0.00E+00
Air Compressors	5.90E-03	3	0	23	0.00E+00	0.00E+00
Vent. Subsystems	1.60E-02	3	3	289	1.04E-02	4.98E-04
			Sum of Ignition Sources Weighting Factor for Transient Source	Total Number of Compartments		
Transients	3.60E-02	3	5	34	1.47E-01	1.59E-02
Cable Fire-Welding	1.30E-03	3	1	34	2.94E-02	1.15E-04
Transient Fire-Welding	3.40E-02	3	1	34	2.94E-02	3.00E-03
				TOTAL		4.88E-02

Note: (1) The electric cabinet and pump count in unit 1 fire zones includes only significant fire sources (i.e., those components that have the potential to develop credible fire scenarios). Other similar non-significant fire sources have not been considered in the count to avoid unnecessary assignment of fire frequency to those components. This approach is different from unit 2/3 calculations which considered all components (significant and non-significant) in the count.

FIRE AREA /COMPARTMENTATION IGNITION FREQUENCY
PHASE II (STEP 1) EVALUATION
PLANT LOCATION: REACTOR BUILDING (FIRE ZONE 1-6)

Components	Generic Fire Frequency	Location Weighting Factor (WL)	Number of Components in Fire Area	Total Number in All Plant locations(1)	Ignition Source Weighting factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location:						
Electric Cabinets	5.70E-02	1	2	11	1.82E-01	1.04E-02
Pumps	3.50E-02	1	0	11	0.00E+00	0.00E+00
Plant Wide:						
Fire Protection Panel	1.30E-03	3	0	25	0.00E+00	0.00E+00
RPS MG Set	3.40E-03	3	0	19	0.00E+00	0.00E+00
Non-Qualified Cable	8.40E-03	3	206	13388	1.54E-02	3.88E-04
Non-Qualified JB	1.30E-03	3	206	13388	1.54E-02	6.00E-05
Transformers	1.40E-02	3	1	48	2.08E-02	8.75E-04
Battery chargers	5.50E-03	3	0	34	0.00E+00	0.00E+00
Air Compressors	5.90E-03	3	0	23	0.00E+00	0.00E+00
Vent. Subsystems	1.60E-02	3	1	289	3.46E-03	1.66E-04
			Sum of Ignition Sources Weighting Factor for Transient Source	Total Number of Compartments		
Transients	3.60E-02	3	5	34	1.47E-01	1.59E-02
Cable Fire-Welding	1.30E-03	3	1	34	2.94E-02	1.15E-04
Transient Fire-Welding	3.40E-02	3	1	34	2.94E-02	3.00E-03
				TOTAL		**3.08E-02**

Note: (1) The electric cabinet and pump count in unit 1 fire zones includes only significant fire sources (i.e., those components that have the potential to develop credible fire scenarios). Other similar non-significant fire sources have not been considered in the count to avoid unnecessary assignment of fire frequency to those components. This approach is different from unit 2/3 calculations which considered all components (significant and non-significant) in the count.

FIRE AREA /COMPARTMENTATION IGNITION FREQUENCY
PHASE II (STEP 1) EVALUATION
PLANT LOCATION: REACTOR BUILDING (FIRE AREA 2)

Components	Generic Fire Frequency	Location Weighting Factor (WL)	Number of Components in Fire Area(1)	Total Number in All Plant locations(1)	Ignition Source Weighting factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location:						
Electric Cabinets	5.70E-02	1				5.70E-02
Pumps	3.50E-02	1				3.50E-02
Plant Wide:						
Fire Protection Panel	1.30E-03	3	3	25	1.20E-01	4.68E-04
RPS MG Set	3.40E-03	3	4	19	2.11E-01	2.15E-03
Non-Qualified Cable	8.40E-03	3	1239	13388	9.25E-02	2.33E-03
Non-Qualified JB	1.30E-03	3	1239	13388	9.25E-02	3.61E-04
Transformers	1.40E-02	3	6	48	1.25E-01	5.25E-03
Battery chargers	5.50E-03	3	0	34	0.00E+00	0.00E+00
Air Compressors	5.90E-03	3	3	23	1.30E-01	2.31E-03
Vent. Subsystems	1.60E-02	3	20	289	6.92E-02	3.32E-03
			Sum of Ignition Sources Weighting Factor for Transient Source	Total Number of Compartments		
Transients	3.60E-02	3	5	34	1.47E-01	1.59E-02
Cable Fire-Welding	1.30E-03	3	1	34	2.94E-02	1.15E-04
Transient Fire-Welding	3.40E-02	3	1	34	2.94E-02	3.00E-03
				TOTAL		**1.27E-01**

Note: (1) Since the ignition frequency for plant location components is area based, individual component count is not necessary. Therefore, no entries have been made for "Number of Components in Fire Area" and "Total Number in all Plant Locations".

FIRE AREA /COMPARTMENTATION IGNITION FREQUENCY
PHASE II (STEP 1) EVALUATION
PLANT LOCATION: REACTOR BUILDING (FIRE AREA 3)

Components	Generic Fire Frequency	Location Weighting Factor (WL)	Number of Components in Fire Area(1)	Total Number in All Plant locations(1)	Ignition Source Weighting factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location:						
Electric Cabinets	5.70E-02	1				5.70E-02
Pumps	3.50E-02	1				3.50E-02
Plant Wide:						
Fire Protection Panel	1.30E-03	3	3	25	1.20E-01	4.68E-04
RPS MG Set	3.40E-03	3	4	19	2.11E-01	2.15E-03
Non-Qualified Cable	8.40E-03	3	1239	13388	9.25E-02	2.33E-03
Non-Qualified JB	1.30E-03	3	1239	13388	9.25E-02	3.61E-04
Transformers	1.40E-02	3	5	48	1.04E-01	4.38E-03
Battery chargers	5.50E-03	3	0	34	0.00E+00	0.00E+00
Air Compressors	5.90E-03	3	3	23	1.30E-01	2.31E-03
Vent. Subsystems	1.60E-02	3	21	289	7.27E-02	3.49E-03
			Sum of Ignition Sources Weighting Factor for Transient Source	Total Number of Compartments		
Transients	3.60E-02	3	5	34	1.47E-01	1.59E-02
Cable Fire-Welding	1.30E-03	3	1	34	2.94E-02	1.15E-04
Transient Fire-Welding	3.40E-02	3	1	34	2.94E-02	3.00E-03
				TOTAL		**1.26E-01**

Note: (1) Since the ignition frequency for plant location components is area based, individual component count is not necessary. Therefore, no entries have been made for "Number of Components in Fire Area" and "Total Number in all Plant Locations".

FIRE AREA /COMPARTMENTATION IGNITION FREQUENCY
PHASE II (STEP 1) EVALUATION
PLANT LOCATION:SWITCHGEAR ROOM (FIRE AREA 4)

Components	Generic Fire Frequency	Location Weighting Factor (WL)(1)	Number of Components In Fire Area(2)	Total Number in All Plant locations(2)	Ignition Source Weighting factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location:						
Electric Cabinets	1.70E-02	0.2				3.40E-03
Plant Wide:						
Fire Protection Panel	1.30E-03	3	0	25	0.00E+00	0.00E+00
RPS MG Set	3.40E-03	3	0	19	0.00E+00	0.00E+00
Non-Qualified Cable	8.40E-03	3	16	13388	1.20E-03	3.01E-05
Non-Qualified JB	1.30E-03	3	16	13388	1.20E-03	4.66E-06
Transformers	1.40E-02	3	0	48	0.00E+00	0.00E+00
Battery chargers	5.50E-03	3	0	34	0.00E+00	0.00E+00
Air Compressors	5.90E-03	3	0	23	0.00E+00	0.00E+00
Vent. Subsystems	1.60E-02	3	1	289	3.46E-03	1.66E-04
			Sum of Ignition Sources Weighting Factor for Transient Source	Total Number of Compartments		
Transients	3.60E-02	3	4	34	1.18E-01	1.27E-02
Cable Fire-Welding	1.30E-03	3	1	34	2.94E-02	1.15E-04
Transient Fire-Welding	3.40E-02	3	1	34	2.94E-02	3.00E-03
				TOTAL		1.94E-02

Note: (1) *The" Location Weighting Factor" for "Plant Locations" is the number of units (3) divided by the number of switchgear rooms (15)*

(2) *Since the ignition frequency for plant location components is area based, individual component count is not necessary. Therefore, no entries have been made for "Number of Components in Fire Area" and "Total Number in all Plant Locations".*

FIRE AREA /COMPARTMENTATION IGNITION FREQUENCY
PHASE II (STEP 1) EVALUATION
PLANT LOCATION:SWITCHGEAR ROOM (FIRE AREA 5)

Components	Generic Fire Frequency	Location Weighting Factor (WL)(1)	Number of Components in Fire Area(3)	Total Number in All Plant locations(3)	Ignition Source Weighting factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location:						
Electric Cabinets	1.70E-02	0.2				3.40E-03
Batteries(2)	1.70E-02	1				1.70E-03
Plant Wide:						
Fire Protection Panel	1.30E-03	3	0	25	0.00E+00	0.00E+00
RPS MG Set	3.40E-03	3	0	19	0.00E+00	0.00E+00
Non-Qualified Cable	8.40E-03	3	67	13388	5.00E-03	1.26E-04
Non-Qualified JB	1.30E-03	3	67	13388	5.00E-03	1.95E-05
Transformers	1.40E-02	3	1	48	2.08E-02	8.75E-04
Battery chargers	5.50E-03	3	2	34	5.88E-02	9.71E-04
Air Compressors	5.90E-03	3	0	23	0.00E+00	0.00E+00
Vent. Subsystems	1.60E-02	3	4	289	1.38E-02	6.64E-04
			Sum of Ignition Sources Weighting Factor for Transient Source	Total Number of Compartments		
Transients	3.60E-02	3	4	34	1.18E-01	1.27E-02
Cable Fire-Welding	1.30E-03	3	1	34	2.94E-02	1.15E-04
Transient Fire-Welding	3.40E-02	3	1	34	2.94E-02	3.00E-03
				TOTAL		**2.36E-02**

Note: (1) *The" Location Weighting Factor" for "Plant Locations" is the number of units (3) divided by the number of switchgear rooms (15)*

(2) *Due to the presence of some 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.*

(3) *Since the ignition frequency for plant location components is area based, individual component count is not necessary. Therefore, no entries have been made for "Number of Components in Fire Area" and "Total Number in all Plant Locations".*

FIRE AREA /COMPARTMENTATION IGNITION FREQUENCY
PHASE II (STEP 1) EVALUATION
PLANT LOCATION:SWITCHGEAR ROOM (FIRE AREA 6)

Components	Generic Fire Frequency	Location Weighting Factor (WL)(1)	Number of Components in Fire Area(2)	Total Number in All Plant locations(2)	Ignition Source Weighting factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location:						
Electric Cabinets	1.70E-02	0.2				3.40E-03
Plant Wide:						
Fire Protection Panel	1.30E-03	3	0	25	0.00E+00	0.00E+00
RPS MG Set	3.40E-03	3	0	19	0.00E+00	0.00E+00
Non-Qualified Cable	8.40E-03	3	11	13388	8.22E-04	2.07E-05
Non-Qualified JB	1.30E-03	3	11	13388	8.22E-04	3.20E-06
Transformers	1.40E-02	3	0	48	0.00E+00	0.00E+00
Battery chargers	5.50E-03	3	0	34	0.00E+00	0.00E+00
Air Compressors	5.90E-03	3	0	23	0.00E+00	0.00E+00
Vent. Subsystems	1.60E-02	3	0	289	0.00E+00	0.00E+00
			Sum of Ignition Sources Weighting Factor for Transient Source	Total Number of Compartments		
Transients	3.60E-02	3	4	34	1.18E-01	1.27E-02
Cable Fire-Welding	1.30E-03	3	1	34	2.94E-02	1.15E-04
Transient Fire-Welding	3.40E-02	3	1	34	2.94E-02	3.00E-03
				TOTAL		**1.92E-02**

Note: (1) *The" Location Weighting Factor" for "Plant Locations" is the number of units (3) divided by the number of switchgear rooms (15)*

(2) *Since the ignition frequency for plant location components is area based, individual component count is not necessary. Therefore, no entries have been made for "Number of Components in Fire Area" and "Total Number in all Plant Locations".*

FIRE AREA /COMPARTMENTATION IGNITION FREQUENCY
PHASE II (STEP 1) EVALUATION
PLANT LOCATION:SWITCHGEAR ROOM (FIRE AREA 7)

Components	Generic Fire Frequency	Location Weighting Factor (WL)(1)	Number of Components in Fire Area(2),(3)	Total Number in All Plant locations(3)	Ignition Source Weighting factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location:						
Electric Cabinets	1.70E-02	0.2				3.40E-03
Plant Wide:						
Fire Protection Panel	1.30E-03	3	0	25	0.00E+00	0.00E+00
RPS MG Set	3.40E-03	3	0	19	0.00E+00	0.00E+00
Non-Qualified Cable	8.40E-03	3	0	13388	0.00E+00	0.00E+00
Non-Qualified JB	1.30E-03	3	0	13388	0.00E+00	0.00E+00
Transformers	1.40E-02	3	0	48	0.00E+00	0.00E+00
Battery chargers	5.50E-03	3	0	34	0.00E+00	0.00E+00
Air Compressors	5.90E-03	3	0	23	0.00E+00	0.00E+00
Vent. Subsystems	1.60E-02	3	0	289	0.00E+00	0.00E+00
			Sum of Ignition Sources Weighting Factor for Transient Source	Total Number of Compartments		
Transients	3.60E-02	3	4	34	1.18E-01	1.27E-02
Cable Fire-Welding	1.30E-03	3	1	34	2.94E-02	1.15E-04
Transient Fire-Welding	3.40E-02	3	1	34	2.94E-02	3.00E-03
				TOTAL		**1.92E-02**

Note: (1) *The" Location Weighting Factor" for "Plant Locations" is the number of units (3) divided by the number of switchgear rooms (15)*

(2) *No Plant Wide ignition sources (except transients) exist in this area.*

(3) *Since the ignition frequency for plant location components is area based, individual component count is not necessary. Therefore, no entries have been made for "Number of Components in Fire Area" and "Total Number in all Plant Locations".*

FIRE AREA /COMPARTMENTATION IGNITION FREQUENCY
PHASE II (STEP 1) EVALUATION
PLANT LOCATION:SWITCHGEAR ROOM (FIRE AREA 8)

Components	Generic Fire Frequency	Location Weighting Factor (WL)(1)	Number of Components in Fire Area(2),(3)	Total Number in All Plant locations(3)	Ignition Source Weighting factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location:						
Electric Cabinets	1.70E-02	0.2				3.40E-03
Plant Wide:						
Fire Protection Panel	1.30E-03	3	0	25	0.00E+00	0.00E+00
RPS MG Set	3.40E-03	3	0	19	0.00E+00	0.00E+00
Non-Qualified Cable	8.40E-03	3	0	13388	0.00E+00	0.00E+00
Non-Qualified JB	1.30E-03	3	0	13388	0.00E+00	0.00E+00
Transformers	1.40E-02	3	0	48	0.00E+00	0.00E+00
Battery chargers	5.50E-03	3	0	34	0.00E+00	0.00E+00
Air Compressors	5.90E-03	3	0	23	0.00E+00	0.00E+00
Vent. Subsystems	1.60E-02	3	0	289	0.00E+00	0.00E+00
			Sum of Ignition Sources Weighting Factor for Transient Source	Total Number of Compartments		
Transients	3.60E-02	3	4	34	1.18E-01	1.27E-02
Cable Fire-Welding	1.30E-03	3	1	34	2.94E-02	1.15E-04
Transient Fire-Welding	3.40E-02	3	1	34	2.94E-02	3.00E-03
				TOTAL		**1.92E-02**

Note: (1) *The" Location Weighting Factor" for "Plant Locations" is the number of units (3) divided by the number of switchgear rooms (15)*

(2) *No Plant Wide ignition sources (except transients) exist in this area.*

(3) *Since the ignition frequency for plant location components is area based, individual component count is not necessary. Therefore, no entries have been made for "Number of Components in Fire Area" and "Total Number in all Plant Locations".*

FIRE AREA /COMPARTMENTATION IGNITION FREQUENCY
PHASE II (STEP 1) EVALUATION
PLANT LOCATION:SWITCHGEAR ROOM (FIRE AREA 9)

Components	Generic Fire Frequency	Location Weighting Factor (WL)(1)	Number of Components in Fire Area(3)	Total Number in All Plant locations(3)	Ignition Source Weighting factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location:						
Electric Cabinets	1.70E-02	0.2				3.40E-03
Batteries(2)	1.70E-02	1				1.70E-03
Plant Wide:						
Fire Protection Panel	1.30E-03	3	0	25	0.00E+00	0.00E+00
RPS MG Set	3.40E-03	3	0	19	0.00E+00	0.00E+00
Non-Qualified Cable	8.40E-03	3	5	13388	3.73E-04	9.41E-06
Non-Qualified JB	1.30E-03	3	5	13388	3.73E-04	1.46E-06
Transformers	1.40E-02	3	0	48	0.00E+00	0.00E+00
Battery chargers	5.50E-03	3	2	34	5.88E-02	9.71E-04
Air Compressors	5.90E-03	3	0	23	0.00E+00	0.00E+00
Vent. Subsystems	1.60E-02	3	4	289	1.38E-02	6.64E-04
			Sum of Ignition Sources Weighting Factor for Transient Source	Total Number of Compartments		
Transients	3.60E-02	3	4	34	1.18E-01	1.27E-02
Cable Fire-Welding	1.30E-03	3	1	34	2.94E-02	1.15E-04
Transient Fire-Welding	3.40E-02	3	1	34	2.94E-02	3.00E-03
				TOTAL		2.26E-02

Note: (1) *The" Location Weighting Factor" for "Plant Locations" is the number of units (3) divided by the number of switchgear rooms (15)*

(2) *Due to the presence of some 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.*

(3) *Since the ignition frequency for plant location components is area based, individual component count is not necessary. Therefore, no entries have been made for "Number of Components in Fire Area" and "Total Number in all Plant Locations".*

FIRE AREA /COMPARTMENTATION IGNITION FREQUENCY
PHASE II (STEP 1) EVALUATION
PLANT LOCATION:SWITCHGEAR ROOM (FIRE AREA 10)

Components	Generic Fire Frequency	Location Weighting Factor (WL)(1)	Number of Components in Fire Area(2,3)	Total Number in All Plant locations(3)	Ignition Source Weighting factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location:						
Electric Cabinets	1.70E-02	0.2				3.40E-03
Plant Wide:						
Fire Protection Panel	1.30E-03	3	0	25	0.00E+00	0.00E+00
RPS MG Set	3.40E-03	3	0	19	0.00E+00	0.00E+00
Non-Qualified Cable	8.40E-03	3	0	13388	0.00E+00	0.00E+00
Non-Qualified JB	1.30E-03	3	0	13388	0.00E+00	0.00E+00
Transformers	1.40E-02	3	0	48	0.00E+00	0.00E+00
Battery chargers	5.50E-03	3	0	34	0.00E+00	0.00E+00
Air Compressors	5.90E-03	3	0	23	0.00E+00	0.00E+00
Vent. Subsystems	1.60E-02	3	0	289	0.00E+00	0.00E+00
			Sum of Ignition Sources Weighting Factor for Transient Source	Total Number of Compartments		
Transients	3.60E-02	3	4	34	1.18E-01	1.27E-02
Cable Fire-Welding	1.30E-03	3	1	34	2.94E-02	1.15E-04
Transient Fire-Welding	3.40E-02	3	1	34	2.94E-02	3.00E-03
				TOTAL		**1.92E-02**

Note: *(1) The" Location Weighting Factor" for "Plant Locations" is the number of units (3) divided by the number of switchgear rooms (15)*

(2) No Plant Wide ignition sources (except transients) exist in this area.

(3) Since the ignition frequency for plant location components is area based, individual component count is not necessary. Therefore, no entries have been made for "Number of Components in Fire Area" and "Total Number in all Plant Locations".

FIRE AREA /COMPARTMENTATION IGNITION FREQUENCY
PHASE II (STEP 1) EVALUATION
PLANT LOCATION:SWITCHGEAR ROOM (FIRE AREA 11)

Components	Generic Fire Frequency	Location Weighting Factor (WL)(1)	Number of Components in Fire Area(2),(3)	Total Number in All Plant locations(3)	Ignition Source Weighting factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location:						
Electric Cabinets	1.70E-02	0.2				3.40E-03
Plant Wide:						
Fire Protection Panel	1.30E-03	3	0	25	0.00E+00	0.00E+00
RPS MG Set	3.40E-03	3	0	19	0.00E+00	0.00E+00
Non-Qualified Cable	8.40E-03	3	0	13388	0.00E+00	0.00E+00
Non-Qualified JB	1.30E-03	3	0	13388	0.00E+00	0.00E+00
Transformers	1.40E-02	3	0	48	0.00E+00	0.00E+00
Battery chargers	5.50E-03	3	0	34	0.00E+00	0.00E+00
Air Compressors	5.90E-03	3	0	23	0.00E+00	0.00E+00
Vent. Subsystems	1.60E-02	3	0	289	0.00E+00	0.00E+00
			Sum of Ignition Sources Weighting Factor for Transient Source	Total Number of Compartments		
Transients	3.60E-02	3	4	34	1.18E-01	1.27E-02
Cable Fire-Welding	1.30E-03	3	1	34	2.94E-02	1.15E-04
Transient Fire-Welding	3.40E-02	3	1	34	2.94E-02	3.00E-03
				TOTAL		**1.92E-02**

Note: (1) *The" Location Weighting Factor" for "Plant Locations" is the number of units (3) divided by the number of switchgear rooms (15)*

(2) *No Plant Wide ignition sources (except transients) exist in this area.*

(3) *Since the ignition frequency for plant location components is area based, individual component count is not necessary. Therefore, no entries have been made for "Number of Components in Fire Area" and "Total Number in all Plant Locations".*

FIRE AREA /COMPARTMENTATION IGNITION FREQUENCY
PHASE II (STEP 1) EVALUATION
PLANT LOCATION:SWITCHGEAR ROOM (FIRE AREA 12)

Components	Generic Fire Frequency	Location Weighting Factor (WL)(1)	Number of Components in Fire Area(2)	Total Number in All Plant locations(2)	Ignition Source Weighting factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location:						
Electric Cabinets	1.70E-02	0.2				3.40E-03
Plant Wide:						
Fire Protection Panel	1.30E-03	3	0	25	0.00E+00	0.00E+00
RPS MG Set	3.40E-03	3	0	19	0.00E+00	0.00E+00
Non-Qualified Cable	8.40E-03	3	0	13388	0.00E+00	0.00E+00
Non-Qualified JB	1.30E-03	3	0	13388	0.00E+00	0.00E+00
Transformers	1.40E-02	3	1	48	2.08E-02	8.75E-04
Battery chargers	5.50E-03	3	0	34	0.00E+00	0.00E+00
Air Compressors	5.90E-03	3	0	23	0.00E+00	0.00E+00
Vent. Subsystems	1.60E-02	3	1	289	3.46E-03	1.66E-04
			Sum of Ignition Sources Weighting Factor for Transient Source	Total Number of Compartments		
Transients	3.60E-02	3	4	34	1.18E-01	1.27E-02
Cable Fire-Welding	1.30E-03	3	1	34	2.94E-02	1.15E-04
Transient Fire-Welding	3.40E-02	3	1	34	2.94E-02	3.00E-03
				TOTAL		**2.03-02**

Note: (1) *The" Location Weighting Factor" for "Plant Locations" is the number of units (3) divided by the number of switchgear rooms (15)*

(2) *Since the ignition frequency for plant location components is area based, individual component count is not necessary. Therefore, no entries have been made for "Number of Components in Fire Area" and "Total Number in all Plant Locations".*

FIRE AREA /COMPARTMENTATION IGNITION FREQUENCY
PHASE II (STEP 1) EVALUATION
PLANT LOCATION:SWITCHGEAR ROOM (FIRE AREA 13)

Components	Generic Fire Frequency	Location Weighting Factor (WL)(1)	Number of Components in Fire Area(2)	Total Number in All Plant locations(2)	Ignition Source Weighting factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location:						
Electric Cabinets	1.70E-02	0.2				3.40E-03
Plant Wide:						
Fire Protection Panel	1.30E-03	3	0	25	0.00E+00	0.00E+00
RPS MG Set	3.40E-03	3	0	19	0.00E+00	0.00E+00
Non-Qualified Cable	8.40E-03	3	0	13388	0.00E+00	0.00E+00
Non-Qualified JB	1.30E-03	3	0	13388	0.00E+00	0.00E+00
Transformers	1.40E-02	3	1	48	2.08E-02	8.75E-04
Battery chargers	5.50E-03	3	0	34	0.00E+00	0.00E+00
Air Compressors	5.90E-03	3	0	23	0.00E+00	0.00E+00
Vent. Subsystems	1.60E-02	3	0	289	0.00E+00	0.00E+00
			Sum of Ignition Sources Weighting Factor for Transient Source	Total Number of Compartments		
Transients	3.60E-02	3	4	34	1.18E-01	1.27E-02
Cable Fire-Welding	1.30E-03	3	1	34	2.94E-02	1.15E-04
Transient Fire-Welding	3.40E-02	3	1	34	2.94E-02	3.00E-03
				TOTAL		**2.01E-02**

Note: (1) *The" Location Weighting Factor" for "Plant Locations" is the number of units (3) divided by the number of switchgear rooms (15)*

(2) *Since the ignition frequency for plant location components is area based, individual component count is not necessary. Therefore, no entries have been made for "Number of Components in Fire Area" and "Total Number in all Plant Locations".*

FIRE AREA /COMPARTMENTATION IGNITION FREQUENCY
PHASE II (STEP 1) EVALUATION
PLANT LOCATION:SWITCHGEAR ROOM (FIRE AREA 14)

Components	Generic Fire Frequency	Location Weighting Factor (WL)(1)	Number of Components in Fire Area(2),(3)	Total Number in All Plant locations(3)	Ignition Source Weighting factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location:						
Electric Cabinets	1.70E-02	0.2				3.40E-03
Plant Wide:						
Fire Protection Panel	1.30E-03	3	0	25	0.00E+00	0.00E+00
RPS MG Set	3.40E-03	3	0	19	0.00E+00	0.00E+00
Non-Qualified Cable	8.40E-03	3	0	13388	0.00E+00	0.00E+00
Non-Qualified JB	1.30E-03	3	0	13388	0.00E+00	0.00E+00
Transformers	1.40E-02	3	0	48	0.00E+00	0.00E+00
Battery chargers	5.50E-03	3	0	34	0.00E+00	0.00E+00
Air Compressors	5.90E-03	3	0	23	0.00E+00	0.00E+00
Vent. Subsystems	1.60E-02	3	0	289	0.00E+00	0.00E+00
			Sum of Ignition Sources Weighting Factor for Transient Source	Total Number of Compartments		
Transients	3.60E-02	3	4	34	1.18E-01	1.27E-02
Cable Fire-Welding	1.30E-03	3	1	34	2.94E-02	1.15E-04
Transient Fire-Welding	3.40E-02	3	1	34	2.94E-02	3.00E-03
				TOTAL		**1.92E-02**

Note: (1) *The" Location Weighting Factor" for "Plant Locations" is the number of units (3) divided by the number of switchgear rooms (15*

(2) *No Plant Wide ignition sources (except transients) exist in this area.*

(3) *Since the ignition frequency for plant location components is area based, individual component count is not necessary. Therefore, no entries have been made for "Number of Components in Fire Area" and "Total Number in all Plant Locations".*

FIRE AREA /COMPARTMENTATION IGNITION FREQUENCY
PHASE II (STEP 1) EVALUATION
PLANT LOCATION:SWITCHGEAR ROOM (FIRE AREA 15)

Components	Generic Fire Frequency	Location Weighting Factor (WL)(1)	Number of Components in Fire Area(2,3)	Total Number in All Plant locations(3)	Ignition Source Weighting factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location:						
Electric Cabinets	1.70E-02	0.2				3.40E-03
Plant Wide:						
Fire Protection Panel	1.30E-03	3	0	25	0.00E+00	0.00E+00
RPS MG Set	3.40E-03	3	0	19	0.00E+00	0.00E+00
Non-Qualified Cable	8.40E-03	3	0	13388	0.00E+00	0.00E+00
Non-Qualified JB	1.30E-03	3	0	13388	0.00E+00	0.00E+00
Transformers	1.40E-02	3	0	48	0.00E+00	0.00E+00
Battery chargers	5.50E-03	3	0	34	0.00E+00	0.00E+00
Air Compressors	5.90E-03	3	0	23	0.00E+00	0.00E+00
Vent. Subsystems	1.60E-02	3	0	289	0.00E+00	0.00E+00
			Sum of Ignition Sources Weighting Factor for Transient Source	Total Number of Compartments		
Transients	3.60E-02	3	4	34	1.18E-01	1.27E-02
Cable Fire-Welding	1.30E-03	3	1	34	2.94E-02	1.15E-04
Transient Fire-Welding	3.40E-02	3	1	34	2.94E-02	3.00E-03
				TOTAL		**1.92E-02**

Note: *(1) The" Location Weighting Factor" for "Plant Locations" is the number of units (3) divided by the number of switchgear rooms (15)*

(2) No Plant Wide ignition sources (except transients) exist in this area.

(3) Since the ignition frequency for plant location components is area based, individual component count is not necessary. Therefore, no entries have been made for "Number of Components in Fire Area" and "Total Number in all Plant Locations".

FIRE AREA /COMPARTMENTATION IGNITION FREQUENCY
PHASE II (STEP 1) EVALUATION
PLANT LOCATION:CONTROL BUILDING EL 593 (FIRE COMPARTMENT 16-1)

Components	Generic Fire Frequency	Location Weighting Factor (WL)(1)	Number of Components in Fire Area(3)	Total Number in All Plant locations(3)	Ignition Source Weighting factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location:						
Electric Cabinets	1.60E-02	1				1.60E-02
Pumps(2)						
Plant Wide:						
Fire Protection Panel	1.30E-03	3	3	25	1.20E-01	4.68E-04
RPS MG Set	3.40E-03	3	6	19	3.16E-01	3.22E-03
Non-Qualified Cable	8.40E-03	3	140	13388	1.05E-02	2.64E-04
Non-Qualified JB	1.30E-03	3	140	13388	1.05E-02	4.08E-05
Transformers	1.40E-02	3	0	48	0.00E+00	0.00E+00
Battery chargers	5.50E-03	3	3	34	8.82E-02	1.46E-03
Air Compressors	5.90E-03	3	0	23	0.00E+00	0.00E+00
Vent. Subsystems	1.60E-02	3	3	289	1.04E-02	4.98E-04
			Sum of Ignition Sources Weighting Factor for Transient Source	Total Number of Compartments		
Transients	3.60E-02	3	4	34	1.18E-01	1.27E-02
Cable Fire-Welding	1.30E-03	3	1	34	2.94E-02	1.15E-04
Transient Fire-Welding	3.40E-02	3	1	34	2.94E-02	3.00E-03
				TOTAL		**3.78E-02**

Note: (1) *This location in the control building is not categorized as a Plant Location/Building in Table 1.2 of Reference 1. The ignition frequency attributed due to electrical cabinets is assumed similar to a control building.*

(2) *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.*

(3) *Since the ignition frequency for plant location components is area based, individual component count is not necessary. Therefore, no entries have been made for "Number of Components in Fire Area" and "Total Number in all Plant Locations".*

FIRE AREA /COMPARTMENTATION IGNITION FREQUENCY
PHASE II (STEP 1) EVALUATION
PLANT LOCATION:CABLE SPREADING ROOMS A & B EL 606 (FIRE COMPARTMENT 16-2)

Components	Generic Fire Frequency	Location Weighting Factor (WL)(1)	Number of Components in Fire Area(3)	Total Number in All Plant locations(3)	Ignition Source Weighting factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location:						
Electric Cabinets	2.50E-03	3				7.50E-03
Plant Wide:						
Fire Protection Panel	1.30E-03	3	2	25	8.00E-02	3.12E-04
RPS MG Set	3.40E-03	3	0	19	0.00E+00	0.00E+00
Non-Qualified Cable	8.40E-03	3	1440	13388	1.08E-01	2.71E-03
Non-Qualified JB	1.30E-03	3	1440	13388	1.08E-01	4.19E-04
Transformers	1.40E-02	3	0	48	0.00E+00	0.00E+00
Battery chargers	5.50E-03	3	0	34	0.00E+00	0.00E+00
Air Compressors	5.90E-03	3	1	23	4.35E-02	7.70E-04
Vent. Subsystems	1.60E-02	3	2	289	6.92E-03	3.32E-04
			Sum of Ignition Sources Weighting Factor for Transient Source(2)	Total Number of Compartments		
Transients	3.60E-02	3		34	0.00E+00	0.00E+00
Cable Fire-Welding	1.30E-03	3		34	0.00E+00	0.00E+00
Transient Fire-Welding	3.40E-02	3		34	0.00E+00	0.00E+00
				TOTAL		**1.20E-02**

Note: (1) *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 barriers. For calculation purposes the two CSR's will be considered as one room. Therefore, the weighting factor will be 3 (3 units/1CSR).*

(2) *Contribution due to transient combustibles is being neglected due to restrictions imposed by plant procedures.*

(3) *Since the ignition frequency for plant location components is area based, individual component count is not necessary. Therefore, no entries have been made for "Number of Components in Fire Area" and "Total Number in all Plant Locations".*

FIRE AREA /COMPARTMENTATION IGNITION FREQUENCY
PHASE II (STEP 1) EVALUATION
PLANT LOCATION:CONTROL ROOMS (FIRE COMPARTMENT 16-3)

Components	Generic Fire Frequency	Location Weighting Factor (WL)(1)	Number of Components in Fire Area(2)	Total Number in All Plant locations(2)	Ignition Source Weighting factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location:						
Electric Cabinets	1.60E-02	3				4.80E-02
Plant Wide:						
Fire Protection Panel	1.30E-03	3	1	25	4.00E-02	1.56E-04
RPS MG Set	3.40E-03	3	0	19	0.00E+00	0.00E+00
Non-Qualified Cable	8.40E-03	3	100	13388	7.47E-03	1.88E-04
Non-Qualified JB	1.30E-03	3	100	13388	7.47E-03	2.91E-05
Transformers	1.40E-02	3	0	48	0.00E+00	0.00E+00
Battery chargers	5.50E-03	3	0	34	0.00E+00	0.00E+00
Air Compressors	5.90E-03	3	0	23	0.00E+00	0.00E+00
Vent. Subsystems	1.60E-02	3	30	289	1.04E-01	4.98E-03
			Sum of Ignition Sources Weighting Factor for Transient Source	Total Number of Compartments		
Transients	3.60E-02	3	4	34	1.18E-01	1.27E-02
Cable Fire-Welding	1.30E-03	3	1	34	2.94E-02	1.15E-04
Transient Fire-Welding	3.40E-02	3	1	34	2.94E-02	3.00E-03
				TOTAL		**6.92E-02**

Note: (1) *For Plant Locations, the calculated weighting factor should be 1 (3 units/3control rooms). However, since the three control rooms are located in one area without being separated by a fire barrier (i.e., there is a 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).*

(2) *Since the ignition frequency for plant location components is area based, individual component count is not necessary. Therefore, no entries have been made for "Number of Components in Fire Area" and "Total Number in all Plant Locations".*

FIRE AREA /COMPARTMENTATION IGNITION FREQUENCY
PHASE II (STEP 1) EVALUATION
PLANT LOCATION:BATTERY AND BATTERY BOARD ROOMS UNIT 1 (FIRE AREA 17)

Components	Generic Fire Frequency	Location Weighting Factor (WL)(1)	Number of Components In Fire Area(2)	Total Number in All Plant locations(2)	Ignition Source Weighting factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location:						
Electric Cabinets	5.70E-02	0.25				1.43E-02
Batteries	1.70E-02	1				1.70E-02
Plant Wide:						
Fire Protection Panel	1.30E-03	3	0	25	0.00E+00	0.00E+00
RPS MG Set	3.40E-03	3	1	19	5.26E-02	5.37E-04
Non-Qualified Cable	8.40E-03	3	0	13388	0.00E+00	0.00E+00
Non-Qualified JB	1.30E-03	3	0	13388	0.00E+00	0.00E+00
Transformers	1.40E-02	3	2	48	4.17E-02	1.75E-03
Battery chargers	5.50E-03	3	3	34	8.82E-02	1.46E-03
Air Compressors	5.90E-03	3	0	23	0.00E+00	0.00E+00
Vent. Subsystems	1.60E-02	3	0	289	0.00E+00	0.00E+00
			Sum of Ignition Sources Weighting Factor for Transient Source	Total Number of Compartments		
Transients	3.60E-02	3	4	34	1.18E-01	1.27E-02
Cable Fire-Welding	1.30E-03	3	1	34	2.94E-02	1.15E-04
Transient Fire-Welding	3.40E-02	3	1	34	2.94E-02	3.00E-03
				TOTAL		5.08E-02

Note: (1) *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 only. The cabinet contribution will be assumed 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 frequencies of the two rooms.*

(2) *Since the ignition frequency for plant location components is area based, individual component count is not necessary. Therefore, no entries have been made for "Number of Components in Fire Area" and "Total Number in all Plant Locations".*

FIRE AREA /COMPARTMENTATION IGNITION FREQUENCY
PHASE II (STEP 1) EVALUATION
PLANT LOCATION:BATTERY AND BATTERY BOARD ROOMS UNIT 2 (FIRE AREA 18)

Components	Generic Fire Frequency	Location Weighting Factor (WL)(1)	Number of Components in Fire Area(2)	Total Number in All Plant locations(2)	Ignition Source Weighting factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location:						
Electric Cabinets	5.70E-02	0.25				1.43E-02
Batteries	1.70E-02	1				1.70E-02
Plant Wide:						
Fire Protection Panel	1.30E-03	3	0	25	0.00E+00	0.00E+00
RPS MG Set	3.40E-03	3	1	19	5.26E-02	5.37E-04
Non-Qualified Cable	8.40E-03	3	1	13388	7.47E-05	1.88E-06
Non-Qualified JB	1.30E-03	3	1	13388	7.47E-05	2.91E-07
Transformers	1.40E-02	3	0	48	0.00E+00	0.00E+00
Battery chargers	5.50E-03	3	3	34	8.82E-02	1.46E-03
Air Compressors	5.90E-03	3	0	23	0.00E+00	0.00E+00
Vent. Subsystems	1.60E-02	3	0	289	0.00E+00	0.00E+00
			Sum of Ignition Sources Weighting Factor for Transient Source	Total Number of Compartments		
Transients	3.60E-02	3	4	34	1.18E-01	1.27E-02
Cable Fire-Welding	1.30E-03	3	1	34	2.94E-02	1.15E-04
Transient Fire-Welding	3.40E-02	3	1	34	2.94E-02	3.00E-03
				TOTAL		4.91E-02

Note: (1) *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 only. The cabinet contribution will be assumed 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 frequencies of the two rooms.*

(2) *Since the ignition frequency for plant location components is area based, individual component count is not necessary. Therefore, no entries have been made for "Number of Components in Fire Area" and "Total Number in all Plant Locations".*

FIRE AREA /COMPARTMENTATION IGNITION FREQUENCY
PHASE II (STEP 1) EVALUATION
PLANT LOCATION:BATTERY AND BATTERY BOARD ROOMS UNIT 3 (FIRE AREA 19)

Components	Generic Fire Frequency	Location Weighting Factor (WL)(1)	Number of Components in Fire Area(2)	Total Number in All Plant locations(2)	Ignition Source Weighting factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location:						
Electric Cabinets	5.70E-02	0.25				1.43E-02
Batteries	1.70E-02	1				1.70E-02
Plant Wide:						
Fire Protection Panel	1.30E-03	3	0	25	0.00E+00	0.00E+00
RPS MG Set	3.40E-03	3	1	19	5.26E-02	5.37E-04
Non-Qualified Cable	8.40E-03	3	0	13388	0.00E+00	0.00E+00
Non-Qualified JB	1.30E-03	3	0	13388	0.00E+00	0.00E+00
Transformers	1.40E-02	3	0	48	0.00E+00	0.00E+00
Battery chargers	5.50E-03	3	3	34	8.82E-02	1.46E-03
Air Compressors	5.90E-03	3	0	23	0.00E+00	0.00E+00
Vent. Subsystems	1.60E-02	3	0	289	0.00E+00	0.00E+00
			Sum of Ignition Sources Weighting Factor for Transient Source	Total Number of Compartments		
Transients	3.60E-02	3	4	34	1.18E-01	1.27E-02
Cable Fire-Welding	1.30E-03	3	1	34	2.94E-02	1.15E-04
Transient Fire-Welding	3.40E-02	3	1	34	2.94E-02	3.00E-03
				TOTAL		**4.91E-02**

Note: (1) 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 only. The cabinet contribution will be assumed 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 frequencies of the two rooms.

(2) Since the ignition frequency for plant location components is area based, individual component count is not necessary. Therefore, no entries have been made for "Number of Components in Fire Area" and "Total Number in all Plant Locations".

FIRE AREA /COMPARTMENTATION 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)(1)	Number of Components in Fire Area(2)	Total Number in All Plant locations(2)	Ignition Source Weighting factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location:						
Diesel Generators	3.90E-02	4				1.56E-01
Electrical Cabinets	7.20E-03	4				2.88E-02
Plant Wide:						
Fire Protection Panel	1.30E-03	3	2	25	8.00E-02	3.12E-04
RPS MG Set	3.40E-03	3	0	19	0.00E+00	0.00E+00
Non-Qualified Cable	8.40E-03	3	21	13388	1.57E-03	3.95E-05
Non-Qualified JB	1.30E-03	3	21	13388	1.57E-03	6.12E-06
Transformers	1.40E-02	3	3	48	6.25E-02	2.63E-03
Battery chargers	5.50E-03	3	8	34	2.35E-01	3.88E-03
Air Compressors	5.90E-03	3	0	23	0.00E+00	0.00E+00
Vent. Subsystems	1.60E-02	3	20	289	6.92E-02	3.32E-03
			Sum of Ignition Sources Weighting Factor for Transient Source	Total Number of Compartments		
Transients	3.60E-02	3	5	34	1.47E-01	1.59E-02
Cable Fire-Welding	1.30E-03	3	1	34	2.94E-02	1.15E-04
Transient Fire-Welding	3.40E-02	3	1	34	2.94E-02	3.00E-03
				TOTAL		**2.14E-01**

Note: (1) *BFN has two separate diesel generator buildings, one for units 1 and 2 and the other for unit 3. Each diesel building has 4 diesel generators and associated electrical and mechanical systems. Thus fire frequency for one BFN diesel generator building will be 4 times as much as the generic fire frequency (i.e., Plant Location weighting factor of 4).*

(2) *Since the ignition frequency for plant location components is area based, individual component count is not necessary. Therefore, no entries have been made for "Number of Components in Fire Area" and "Total Number in all Plant Locations".*

FIRE AREA /COMPARTMENTATION 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)(1)	Number of Components in Fire Area(2)	Total Number in All Plant locations(2)	Ignition Source Weighting factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location:						
Diesel Generators	3.90E-02	4				1.56E-01
Electrical Cabinets	7.20E-03	4				2.88E-02
Plant Wide:						
Fire Protection Panel	1.30E-03	3	2	25	8.00E-02	3.12E-04
RPS MG Set	3.40E-03	3	0	19	0.00E+00	0.00E+00
Non-Qualified Cable	8.40E-03	3	35	13388	2.61E-03	6.59E-05
Non-Qualified JB	1.30E-03	3	35	13388	2.61E-03	1.02E-05
Transformers	1.40E-02	3	0	48	0.00E+00	0.00E+00
Battery chargers	5.50E-03	3	9	34	2.65E-01	4.37E-03
Air Compressors	5.90E-03	3	0	23	0.00E+00	0.00E+00
Vent. Subsystems	1.60E-02	3	28	289	9.69E-02	4.65E-03
			Sum of Ignition Sources Weighting Factor for Transient Source	Total Number of Compartments		
Transients	3.60E-02	3	5	34	1.47E-01	1.59E-02
Cable Fire-Welding	1.30E-03	3	1	34	2.94E-02	1.15E-04
Transient Fire-Welding	3.40E-02	3	1	34	2.94E-02	3.00E-03
				TOTAL		**2.13E-01**

Note: (1) *BFN has two separate diesel generator buildings, one for units 1 and 2 and the other for unit 3. Each diesel building has 4 diesel generators and associated electrical and mechanical systems. Thus fire frequency for one BFN diesel generator building will be 4 times as much as the generic fire frequency (i.e., Plant Location weighting factor of 4).*

(2) *Since the ignition frequency for plant location components is area based, individual component count is not necessary. Therefore, no entries have been made for "Number of Components in Fire Area" and "Total Number in all Plant Locations".*

FIRE AREA /COMPARTMENTATION 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)(1)	Number of Components in Fire Area(2)	Total Number in All Plant locations(2)	Ignition Source Weighting factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location:						
Electric Cabinets	1.70E-02	0.2				3.40E-03
Plant Wide:						
Fire Protection Panel	1.30E-03	3	0	25	0.00E+00	0.00E+00
RPS MG Set	3.40E-03	3	0	19	0.00E+00	0.00E+00
Non-Qualified Cable	8.40E-03	3	15	13388	1.12E-03	2.82E-05
Non-Qualified JB	1.30E-03	3	15	13388	1.12E-03	4.37E-06
Transformers	1.40E-02	3	0	48	0.00E+00	0.00E+00
Battery chargers	5.50E-03	3	0	34	0.00E+00	0.00E+00
Air Compressors	5.90E-03	3	0	23	0.00E+00	0.00E+00
Vent. Subsystems	1.60E-02	3	0	289	0.00E+00	0.00E+00
			Sum of Ignition Sources Weighting Factor for Transient Source	Total Number of Compartments		
Transients	3.60E-02	3	4	34	1.18E-01	1.27E-02
Cable Fire-Welding	1.30E-03	3	1	34	2.94E-02	1.15E-04
Transient Fire-Welding	3.40E-02	3	1	34	2.94E-02	3.00E-03
				TOTAL		**1.93E-02**

Note: *(1)* *The" Location Weighting Factor" for "Plant Locations" is the number of units (3) divided by the number of switchgear rooms (15)*

(2) *Since the ignition frequency for plant location components is area based, individual component count is not necessary. Therefore, no entries have been made for "Number of Components in Fire Area" and "Total Number in all Plant Locations".*

FIRE AREA /COMPARTMENTATION 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)(1)	Number of Components in Fire Area(2)	Total Number in All Plant locations(2)	Ignition Source Weighting factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location:						
Electric Cabinets	1.70E-02	0.2				3.40E-03
Plant Wide:						
Fire Protection Panel	1.30E-03	3	0	25	0.00E+00	0.00E+00
RPS MG Set	3.40E-03	3	0	19	0.00E+00	0.00E+00
Non-Qualified Cable	8.40E-03	3	15	13388	1.12E-03	2.82E-05
Non-Qualified JB	1.30E-03	3	15	13388	1.12E-03	4.37E-06
Transformers	1.40E-02	3	0	48	0.00E+00	0.00E+00
Battery chargers	5.50E-03	3	0	34	0.00E+00	0.00E+00
Air Compressors	5.90E-03	3	0	23	0.00E+00	0.00E+00
Vent. Subsystems	1.60E-02	3	0	289	0.00E+00	0.00E+00
			Sum of Ignition Sources Weighting Factor for Transient Source	Total Number of Compartments		
Transients	3.60E-02	3	4	34	1.18E-01	1.27E-02
Cable Fire-Welding	1.30E-03	3	1	34	2.94E-02	1.15E-04
Transient Fire-Welding	3.40E-02	3	1	34	2.94E-02	3.00E-03
				TOTAL		**1.93E-02**

Note: (1) *The" Location Weighting Factor" for "Plant Locations" is the number of units (3) divided by the number of switchgear rooms (15)*

(2) *Since the ignition frequency for plant location components is area based, individual component count is not necessary. Therefore, no entries have been made for "Number of Components in Fire Area" and "Total Number in all Plant Locations".*

FIRE AREA /COMPARTMENTATION IGNITION FREQUENCY
PHASE II (STEP 1) EVALUATION
PLANT LOCATION:SWITCHGEAR ROOM, UNIT 3 4KV BUS TIE BOARD ROOM (FIRE AREA 24)

Components	Generic Fire Frequency	Location Weighting Factor (WL)(1)	Number of Components in Fire Area(2)	Total Number in All Plant locations(2)	Ignition Source Weighting factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location:						
Electric Cabinets	1.70E-02	0.2				3.40E-03
Plant Wide:						
Fire Protection Panel	1.30E-03	3	0	25	0.00E+00	0.00E+00
RPS MG Set	3.40E-03	3	0	19	0.00E+00	0.00E+00
Non-Qualified Cable	8.40E-03	3	8	13388	5.98E-04	1.51E-05
Non-Qualified JB	1.30E-03	3	8	13388	5.98E-04	2.33E-06
Transformers	1.40E-02	3	0	48	0.00E+00	0.00E+00
Battery chargers	5.50E-03	3	0	34	0.00E+00	0.00E+00
Air Compressors	5.90E-03	3	0	23	0.00E+00	0.00E+00
Vent. Subsystems	1.60E-02	3	0	289	0.00E+00	0.00E+00
			Sum of Ignition Sources Weighting Factor for Transient Source	Total Number of Compartments		
Transients	3.60E-02	3	4	34	1.18E-01	1.27E-02
Cable Fire-Welding	1.30E-03	3	1	34	2.94E-02	1.15E-04
Transient Fire-Welding	3.40E-02	3	1	34	2.94E-02	3.00E-03
				TOTAL		**1.92E-02**

Note: *(1)* *The" Location Weighting Factor" for "Plant Locations" is the number of units (3) divided by the number of switchgear rooms (15)*

(2) *Since the ignition frequency for plant location components is area based, individual component count is not necessary. Therefore, no entries have been made for "Number of Components in Fire Area" and "Total Number in all Plant Locations".*

FIRE AREA /COMPARTMENTATION IGNITION FREQUENCY
PHASE II (STEP 1) EVALUATION
PLANT LOCATION:INTAKE PUMP STATTION (FIRE COMPARTMENT 25-1)

Components	Generic Fire Frequency	Location Weighting Factor (WL)(1)	Number of Components in Fire Area(2)	Total Number in All Plant locations(2)	Ignition Source Weighting factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location:						
Electric Cabinets	5.50E-03	3				1.65E-02
Fire Pumps	2.40E-03	3				7.20E-03
Others	8.00E-03	3				2.40E-02
Plant Wide:						
Fire Protection Panel	1.30E-03	3	1	25	4.00E-02	1.56E-04
RPS MG Set	3.40E-03	3	0	19	0.00E+00	0.00E+00
Non-Qualified Cable	8.40E-03	3	271	13388	2.02E-02	5.10E-04
Non-Qualified JB	1.30E-03	3	271	13388	2.02E-02	7.89E-05
Transformers	1.40E-02	3	3	48	6.25E-02	2.63E-03
Battery chargers	5.50E-03	3	0	34	0.00E+00	0.00E+00
Air Compressors	5.90E-03	3	0	23	0.00E+00	0.00E+00
Vent. Subsystems	1.60E-02	3	8	289	2.77E-02	1.33E-03
			Sum of Ignition Sources Weighting Factor for Transient Source	Total Number of Compartments		
Transients	3.60E-02	3	7	34	2.06E-01	2.22E-02
Cable Fire-Welding	1.30E-03	3	1	34	2.94E-02	1.15E-04
Transient Fire-Welding	3.40E-02	3	1	34	2.94E-02	3.00E-03
				TOTAL		7.77E-02

Note: (1) The" Location Weighting Factor" for "Plant Locations" is the number of units (3) divided by the number of intake pump stations (1).

(2) Since the ignition frequency for plant location components is area based, individual component count is not necessary. Therefore, no entries have been made for "Number of Components in Fire Area" and "Total Number in all Plant Locations".

FIRE AREA /COMPARTMENTATION IGNITION FREQUENCY
PHASE II (STEP 1) EVALUATION
PLANT LOCATION:PIPE TUNNEL (FIRE COMPARTMENT 25-2)

Components	Generic Fire Frequency	Location Weighting Factor (WL)(1)	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:						
Fire Protection Panel	1.30E-03	3	0	25	0.00E+00	0.00E+00
RPS MG Set	3.40E-03	3	0	19	0.00E+00	0.00E+00
Non-Qualified Cable	8.40E-03	3	5	13388	3.73E-04	9.41E-06
Non-Qualified JB	1.30E-03	3	5	13388	3.73E-04	1.46E-06
Transformers	1.40E-02	3	0	48	0.00E+00	0.00E+00
Battery chargers	5.50E-03	3	0	34	0.00E+00	0.00E+00
Air Compressors	5.90E-03	3	0	23	0.00E+00	0.00E+00
Vent. Subsystems	1.60E-02	3	0	289	0.00E+00	0.00E+00
			Sum of Ignition Sources Weighting Factor for Transient Source	Total Number of Compartments		
NO TRANSIENTS						
				TOTAL		1.09E-05

Note: (1) The" Location Weighting Factor" for "Plant Wide" component is the number of units (3).

FIRE AREA /COMPARTMENTATION IGNITION FREQUENCY
PHASE II (STEP 1) EVALUATION
PLANT LOCATION:TURBINE BUILDING (FIRE COMPARTMENT 25-3)

Components	Generic Fire Frequency	Location Weighting Factor (WL)(1)	Number of Components in Fire Area(2)	Total Number in All Plant locations(2)	Ignition Source Weighting factor (WI)	Fire Compartment Fire Frequency (F1)
	a	b	c	d	c/d	a*b*c/d
Plant Location:						
Boiler	1.30E-03	3				3.90E-03
Electric Cabinets	2.50E-02	3				7.50E-02
Feedwater Pumps	1.20E-02	3				3.60E-02
Other Pumps	1.70E-02	3				5.10E-02
T/G Excitor	5.70E-03	3				1.71E-02
T/G Oil	1.20E-02	3				3.60E-02
T/G Hydrogen	7.70E-03	3				2.31E-02
Plant Wide:						
Fire Protection Panel	1.30E-03	3	4	25	1.60E-01	6.24E-04
RPS MG Set	3.40E-03	3	2	19	1.05E-01	1.07E-03
Non-Qualified Cable	8.40E-03	3	6264	13388	4.68E-01	1.18E-02
Non-Qualified JB	1.30E-03	3	6264	13388	4.68E-01	1.82E-03
Transformers	1.40E-02	3	20	48	4.17E-01	1.75E-02
Battery chargers	5.50E-03	3	1	34	2.94E-02	4.85E-04
Air Compressors	5.90E-03	3	16	23	6.96E-01	1.23E-02
Vent. Subsystems	1.60E-02	3	128	289	4.43E-01	2.13E-02
Off-Gas/H2 Recombiner	7.40E-02	3	3	3	1.00E+00	2.22E-01
			Sum of Ignition Sources Weighting Factor for Transient Source	Total Number of Compartments		
Transients	3.60E-02	3	8	34	2.35E-01	2.54E-02
Cable Fire-Welding	1.30E-03	3	1	34	2.94E-02	1.15E-04
Transient Fire-Welding	3.40E-02	3	1	34	2.94E-02	3.00E-03
				TOTAL		**5.59E-01**

Note: (1) The" Location Weighting Factor" for "Plant Wide" component is the number of units (3).

(2) Since the ignition frequency for plant location components is area based, individual component count is not necessary. Therefore, no entries have been made for "Number of Components in Fire Area" and "Total Number in all Plant Locations".

ATTACHMENT C
FIRE DAMAGE ANALYSIS

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 1 Reactor Building. The heat release rates and combustible values used in these calculations are taken from Attachment A, Heat Release Rates.

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
Four transformers listed in Table B-1 (Plant Wide Component List) have been identified as potential fire sources.

4kV/480V Transformers TS1A and TS1B	(Fire zone 1-5)
240V Lighting Transformer TL1A	(Fire zone 1-5)
4kV/480V Emergency Transformers TS1E, (oil)	(Fire zone 1-6)

In addition, there are two dry-type light transformers at el 519' and 541'; these have no PSA impact.

RPS MG Sets
The LPCI MG sets were removed. The recirc pump MG sets were abandoned.

Air Compressors
The drywell/Torus compressor is removed from Fire Zone 1-1. There are no air compressors modeled.

Fire Protection Panels
Four fire protection panels were identified in Attachment B. 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
Nineteen ventilation subsystems have been identified in Attachment B, Plant Wide Component List. The six air cooling units associated with the RHR and Core Spray Pumps are evaluated along with the respective pumps. The four following systems are evaluated as potential fire sources.

Shutdown Board Room 1B HVAC Compressor Motor (2) and Fan Motor (2) (Fire zone 1-4)

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 11	480V RMOV Board 1C
	480V RB Vent Board 1B
	250V RMOV Board 1C
	1-PNLA-25-340 ES Div I and II panel
Fire Zone 1-5	240V Lighting Board 1A
	1-LPNL-025-0031 (RCIC Aux Control Panel)
	4kV RPT Board 1-1, Panel 1 and 2
	4kV RPT Board 1-2, Panel 1 and 2
	Panel 25-3 (Filter demin)
Fire Zone 1-6	VFD 1A (panel)
	VFD 1B (panel)

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 1-1	Core Spray Pumps 1A and 1C RHR Pumps 1A and 1C HPCI Pump RCIC Pump
Fire Zone 1-2	Core Spray Pumps 1B and 1D RHR Pumps 1B and 1D
Fire Zone 1-3	RCW Pump 1A

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 °F (Based on non-qualified cables)
- Critical Heat Flux = 0.5 Btu/sec/ft^2 (Based on non-qualified cables)
- Heat Loss Fraction = 0.70
- Area Geometry (See accompanying spread sheets)
- Heat Release Rates (See accompanying spread sheets)
- Heat of combustion (From Reference 4)

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. The following spreadsheets are used to calculate the damage height, critical radial distance and potential for the targets being in the hot gas layer or ceiling jet.

The damage threshold elevation (Z_{crit}) and the critical radial distance (R_{crit}) are calculated based on FIVE guidance. The ceiling jet sub layer can form if the damage threshold elevation exceeds the ceiling height. It should be noted that the damage elevation is based on the combined affects of the ambient temperature rise due to complete pyrolysis of the fire source and the direct affect of the fire plume. When a hot upper layer is formed, the plume temperature is affected since the plume now includes added enthalpy by entraining hot layer gas as it moves through the upper layer to the ceiling. The critical temperature rise ΔT_{crit} (Damage temp. - Ambient temp.) is adjusted to account for the hot gas layer temperature rise ($T_{hgl-incr}$).

Table C.2-1 summarizes the results of the Zone of Influence (ZOI) calculations. The individual fire source calculations are performed in Table C.2-2. Where the fire source resulted in hot gas layer conditions, detailed fire hazards evaluation was performed in Section 6.2.

Table C.2-1
Zone of Influence (ZOI) Summary

Ignition Sources (Unit 1)	Damage Height (ft)	Critical Radial Distance (ft)
480V RMOV BD 1C	8.9	3.5
480V RB Vent BD 1B	11.7	4.9
250V RMOV BD 1C	11.8	4.9
1-LPNL-925-340 ES Div I & II Panel	9.0	3.5
EI 541: GE Dry-type Transformer[1]	8.3	3.3
519.0-R-1B: Dry-type Transformer[1]	7.0	2.7
SRM-1RM Drive Control Panel 25-14[1]	9.0	3.5
SRM Preamp Panel 25-27[1]	9.0	3.5
RCW Pumps 1A	8.5	3.5
RCW Pumps 1B[1]	8.5	3.5
Panel 25-6A RPS & NSSS RPS II[1]	8.6	3.5
Panel 25-6-001[1]	8.5	3.5
Panel 25-5A RPS & NSSS RPS II[1]	8.6	3.5
Panel 25-218[1]	8.5	3.5
RBCCW Pump 1A/1B[1]	8.5	3.5
240V Lighting BD 1A	9.3	3.5
240V Lighting Transformer TL-1A[2]	14.0	5.3
4kV-480V Emergency Transformer 1A & 1B	9.7	3.8
4KV RPT BD 1-1/1-2	10.5	3.5
1-LPNL-025-0031: RCIC Backup Control	11.8	3.5
Panel 25-3 (filter demin.)	10.5	3.5
Panel 25-9 (Sample panel)[1]	8.5	3.5
SLC Pump A or B (oil)[1]	16.9	5.4
Panel 1-25-213 & 1-25-222[1]	9.1	3.5
VFD 1A/1B (Panel)	See Note 3	3.5
Core Spray Pumps 1A/1C	Fire damage limited to loss of associated Loop 1 components and RCIC Pump.	
RHR Pumps 1A/1C	Fire damage limited to loss of associated RHR components and HPCI Pump	
RCIC Pump	Fire damage limited to loss of Loop 1 CS components and RCIC Pump.	

Table C.2-1
Zone of Influence (ZOI) Summary

Ignition Sources (Unit 1)	Damage Height (ft)	Critical Radial Distance (ft)
HPCI Pump	Fire damage limited to loss of RHR 1A/1C components and HPCI Pump	
Core Spray Pumps 1B/1D	Fire damage limited to loss of associated Loop 2 CS components.	
RHR Pumps 1B/1D	Fire damage limited to loss of associated RHR components.	
CRD Pumps 1A/1B	Walkdown confirms no credible fire scenario because pumps are located on grating (hence cannot form oil pool), and in isolated area.	
Shutdown Board 1B Room HVAC Fan Motor (1A and 1B)	The fan motor is totally enclosed within a large steel housing. Negligible amount of combustibles within the housing. No fire impact is postulated.	
4kV-480V Emergency Transformer TS1E (oil)	Oil Transformer TS1E fire will cause hot gas layer, see section 6.2.1 for details.	
Primary Containment Purge Filter Unit	The purge unit is used to purge containment during plant shutdown. It is not operated continuously. The heating of the charcoal beds due to any radioactive decay heat buildup will be negligible. A fire is therefore, not postulated. Even if the fire occurs, it will be contained within the stainless steel housing.	

Note 1: Zone of influence were calculated; walkdown found no fire impact.
2: Zcrit > Ceiling height, hot gas layer will form. See section 6.2.1 for details.
3: Large amount of combustible, form hot gas layer.

Table C.2-2 (1)			
Location: 1-1	**Reactor Building, Unit 1, EL 565**		
Fire Source: 1-1-1	**480V RMOV Board 1C**		
Floor Area	ft^2	15984	<-User Input
Maximum Ambient Temperature (Ta)	(^0F)	100	<-User Input
Critical Radial Flux to Target	$Btu/s/ft^2$	0.5	<-User Input
Radiant Fraction of Heat Release		0.4	<-User Input
Critical Damage Temperature (Tc)	(^0F)	425	<-User Input
Height from Fire Source to Ceiling, H	ft.	22	<-User Input
Fire heat release rate (Q)	Btu/sec	190	<-User Input
Location factor (LF)		1	<-User Input
Effective heat release rate (Q_{eff})	Btu/sec	190	Q*LF
Critical temperature rise $(\Delta T)_{crit}$	(^0F)	296	T_c-T_a-$T_{hgl\text{-}incr}$
Damage Threshold Elevation (Z_{crit})	**ft**	**8.9**	Equation 1 & 2
Radiant Heat Release rate	Btu/s	76.0	Q_{eff}·Radiant Frac.
Critical Radial Flux to Distance	**ft**	**3.5**	(Equation 5)
Total Heat (Qtot to HGL)	Btu	560000	<-User Input
Estimated Heat Loss Fraction (HLF)		0.70	<-User Input
Calculated Qnet	Btu	168000	Qtot (1-HLF)
Calculated Enclosure Volume, V	ft^3	351648	
Calculated Qnet/V	Btu/ft^3	0.48	
HGL Temperature Increase ($T_{hgl\text{-}incr}$)	F	29	Equation 3 and 4
Hot Gas Layer Temp. (T_{hgl})	**F**	**129**	Ta+HGL Temp incr.

Table C.2-2 (2)			
Location: 1-1	**Reactor Building, Unit 1, EL 565**		
Fire Source 1-1-2	**480V RB Vent Board 1B**		
Floor Area	ft^2	15984	<-User Input
Maximum Ambient Temperature (Ta)	(^{0}F)	100	<-User Input
Critical Radial Flux to Target	Btu/s/ft^2	0.5	<-User Input
Radiant Fraction of Heat Release		0.4	<-User Input
Critical Damage Temperature (Tc)	(^{0}F)	425	<-User Input
Height from Fire Source to Ceiling, H	ft	22	<-User Input
Fire heat release rate (Q)	Btu/sec	190	<-User Input
Location factor (LF)		2	<-User Input
Effective heat release rate (Q_{eff})	Btu/sec	380	Q*LF
Critical temperature rise $(\Delta T)_{crit}$	(^{0}F)	296	T_c-T_a-$T_{hgl-incr}$
Damage Threshold Elevation (Z_{crit})	**ft**	**11.7**	Equation 1 & 2
Radiant Heat Release rate	Btu/s	152.0	Q_{eff}-Radiant Frac.
Critical Radial Flux to Distance	**ft**	**4.9**	(Equation 5)
Total Heat (Qtot to HGL)	Btu	560000	<-User Input
Estimated Heat Loss Fraction (HLF)		0.70	<-User Input
Calculated Qnet	Btu	168000	Qtot (1-HLF)
Calculated Enclosure Volume, V	ft^3	351648	
Calculated Qnet/V	Btu/ft^3	0.48	
HGL Temperature Increase ($T_{hgl-incr}$)	F	29	Equation 3 and 4
Hot Gas Layer Temp. (T_{hgl})	**F**	**129**	Ta+HGL Temp incr.

Table C.2-2 (3)			
Location: 1-1	**Reactor Building, Unit 1, EL 565**		
Fire Source 1-1-3	**250V RMOV Board 1C**		
Floor Area	ft^2	15984	<-User Input
Maximum Ambient Temperature (Ta)	(^0F)	100	<-User Input
Critical Radial Flux to Target	$Btu/s/ft^2$	0.5	<-User Input
Radiant Fraction of Heat Release		0.4	<-User Input
Critical Damage Temperature (Tc)	(^0F)	425	<-User Input
Height from Fire Source to Ceiling, H	ft	22	<-User Input
Fire heat release rate (Q)	Btu/sec	190	<-User Input
Location factor (LF)		2	<-User Input
Effective heat release rate (Q_{eff})	Btu/sec	380	Q*LF
Critical temperature rise $(\Delta T)_{crit}$	(^0F)	290	T_c-T_a-$T_{hgl\text{-}Incr}$
Damage Threshold Elevation (Z_{crit})	**ft**	**11.8**	Equation 1 & 2
Radiant Heat Release rate	Btu/s	152.0	Q_{eff}-Radiant Frac.
Critical Radial Flux to Distance	**ft**	**4.9**	(Equation 5)
Total Heat (Qtot to HGL)	Btu	672000	<-User Input
Estimated Heat Loss Fraction (HLF)		0.70	<-User Input
Calculated Qnet	Btu	201600	Qtot (1-HLF)
Calculated Enclosure Volume, V	ft^3	351648	
Calculated Qnet/V	Btu/ft^3	0.57	
HGL Temperature Increase ($T_{hgl\text{-}incr}$)	F	35	Equation 3 and 4
Hot Gas Layer Temp. (T_{hgl})	**F**	**135**	Ta+HGL Temp incr.

Table C.2-2 (4)			
Location: 1-1 (or 1-2)	**Reactor Building, Unit 1, EL 565**		
Fire Source 1-1-8 or 1-2-3	**Panel 1-25-340 ES Div I and II**		
Floor Area	ft^2	15984	<-User Input
Maximum Ambient Temperature (Ta)	(^{0}F)	100	<-User Input
Critical Radial Flux to Target	Btu/s/ft^2	0.5	<-User Input
Radiant Fraction of Heat Release		0.4	<-User Input
Critical Damage Temperature (Tc)	(^{0}F)	425	<-User Input
Height from Fire Source to Ceiling, H	ft	22	<-User Input
Fire heat release rate (Q)	Btu/sec	190	<-User Input
Location factor (LF)		1	<-User Input
Effective heat release rate (Q_{eff})	Btu/sec	190	Q*LF
Critical temperature rise $(\Delta T)_{crit}$	(^{0}F)	289	T_c-T_a-$T_{hgl-incr}$
Damage Threshold Elevation (Z_{crit})	**ft**	9.0	Equation 1 & 2
Radiant Heat Release rate	Btu/s	76.0	Q_{eff}*Radiant Frac.
Critical Radial Flux to Distance	**ft**	3.5	(Equation 5)
Total Heat (Qtot to HGL)	Btu	700000	<-User Input
Estimated Heat Loss Fraction (HLF)		0.70	<-User Input
Calculated Qnet	Btu	210000	Qtot (1-HLF)
Calculated Enclosure Volume, V	ft^3	351648	
Calculated Qnet/V	Btu/ft^3	0.60	
HGL Temperature Increase ($T_{hgl-incr}$)	F	36	Equation 3 and 4
Hot Gas Layer Temp. (T_{hgl})	**F**	**136**	**Ta+HGL Temp incr.**

Table C.2-2 (5)			
Location: 1-2	**Reactor Building, Unit 1, EL 541**		
Fire Source	**El 541: GE Dry Type XFMR**		
Floor Area	ft^2	15984	<-User Input
Maximum Ambient Temperature (Ta)	(^{0}F)	100	<-User Input
Critical Radial Flux to Target	Btu/s/ft^2	0.5	<-User Input
Radiant Fraction of Heat Release		0.4	<-User Input
Critical Damage Temperature (Tc)	(^{0}F)	425	<-User Input
Height from Fire Source to Ceiling, H	ft	22	<-User Input
Fire heat release rate (Q)	Btu/sec	168	<-User Input
Location factor (LF)		1	<-User Input
Effective heat release rate (Q_{eff})	Btu/sec	168	Q*LF
Critical temperature rise $(\Delta T)_{crit}$	(^{0}F)	304	T_c-T_a-$T_{hgl\text{-}Incr}$
Damage Threshold Elevation (Z_{crit})	**ft**	**8.3**	Equation 1 & 2
Radiant Heat Release rate	Btu/s	67.2	Q_{eff}·Radiant Frac.
Critical Radial Flux to Distance	**ft**	**3.3**	(Equation 5)
Total Heat (Qtot to HGL)	Btu	420000	<-User Input
Estimated Heat Loss Fraction (HLF)		0.70	<-User Input
Calculated Qnet	Btu	126000	Qtot (1-HLF)
Calculated Enclosure Volume, V	ft^3	351648	
Calculated Qnet/V	Btu/ft^3	0.36	
HGL Temperature Increase ($T_{hgl\text{-}incr}$)	F	21	Equation 3 and 4
Hot Gas Layer Temp. (T_{hgl})	**F**	**121**	Ta+HGL Temp incr.

Table C.2-2 (6)			
Location: 1-2	**Reactor Building, Unit 1, EL 519**		
Fire Source	**519.0-R-1B: Dry Type XFMR**		
Floor Area	ft^2	15984	<-User Input
Maximum Ambient Temperature (Ta)	(^0F)	100	<-User Input
Critical Radial Flux to Target	Btu/s/ft^2	0.5	<-User Input
Radiant Fraction of Heat Release		0.4	<-User Input
Critical Damage Temperature (Tc)	(^0F)	425	<-User Input
Height from Fire Source to Ceiling, H	ft	22	<-User Input
Fire heat release rate (Q)	Btu/sec	112	<-User Input
Location factor (LF)		1	<-User Input
Effective heat release rate (Q_{eff})	Btu/sec	112	Q*LF
Critical temperature rise ($(\Delta T)_{crit}$)	(^0F)	311	T_c-T_a-$T_{hgl-incr}$
Damage Threshold Elevation (Z_{crit})	**ft**	**7.0**	Equation 1 & 2
Radiant Heat Release rate	Btu/s	44.8	Q_{eff}-Radiant Frac.
Critical Radial Flux to Distance	**ft**	2.7	(Equation 5)
Total Heat (Qtot to HGL)	Btu	280000	<-User Input
Estimated Heat Loss Fraction (HLF)		0.70	<-User Input
Calculated Qnet	Btu	84000	Qtot (1-HLF)
Calculated Enclosure Volume, V	ft^3	351648	
Calculated Qnet/V	Btu/ft^3	0.24	
HGL Temperature Increase ($T_{hgl-incr}$)	F	14	Equation 3 and 4
Hot Gas Layer Temp. (T_{hgl})	**F**	**114**	Ta+HGL Temp incr.

Table C.2-2 (7)			
Location: 1-2	**Reactor Building, Unit 1, EL 565**		
Fire Source	**SRM-1RM Drive Control Panel 25-14**		
Floor Area	ft^2	15984	<-User Input
Maximum Ambient Temperature (Ta)	(^{0}F)	100	<-User Input
Critical Radial Flux to Target	Btu/s/ft^2	0.5	<-User Input
Radiant Fraction of Heat Release		0.4	<-User Input
Critical Damage Temperature (Tc)	(^{0}F)	425	<-User Input
Height from Fire Source to Ceiling, H	ft	22	<-User Input
Fire heat release rate (Q)	Btu/sec	190	<-User Input
Location factor (LF)		1	<-User Input
Effective heat release rate (Q_{eff})	Btu/sec	190	Q*LF
Critical temperature rise $(\Delta T)_{crit}$	(^{0}F)	289	T_c-T_a-$T_{hgl-incr}$
Damage Threshold Elevation (Z_{crit})	**ft**	**9.0**	Equation 1 & 2
Radiant Heat Release rate	Btu/s	76.0	Q_{eff}-Radiant Frac.
Critical Radial Flux to Distance	**ft**	**3.5**	(Equation 5)
Total Heat (Qtot to HGL)	Btu	700000	<-User Input
Estimated Heat Loss Fraction (HLF)		0.70	<-User Input
Calculated Qnet	Btu	210000	Qtot (1-HLF)
Calculated Enclosure Volume, V	ft^3	351648	
Calculated Qnet/V	Btu/ft^3	0.60	
HGL Temperature Increase ($T_{hgl-incr}$)	F	36	Equation 3 and 4
Hot Gas Layer Temp. (T_{hgl})	**F**	**136**	Ta+HGL Temp incr.

Table C.2-2 (8)			
Location: 1-2	**Reactor Building, Unit 1, EL 565**		
Fire Source	**SRM PREAMP Panel (1-LPNL-925-0027)**		
Floor Area	ft^2	15984	<-User Input
Maximum Ambient Temperature (Ta)	(^0F)	100	<-User Input
Critical Radial Flux to Target	$Btu/s/ft^2$	0.5	<-User Input
Radiant Fraction of Heat Release		0.4	<-User Input
Critical Damage Temperature (Tc)	(^0F)	425	<-User Input
Height from Fire Source to Ceiling, H	ft	22	<-User Input
Fire heat release rate (Q)	Btu/sec	190	<-User Input
Location factor (LF)		1	<-User Input
Effective heat release rate (Q_{eff})	Btu/sec	190	Q*LF
Critical temperature rise $(\Delta T)_{crit}$	(^0F)	289	T_c-T_a-$T_{hgl\text{-}incr}$
Damage Threshold Elevation (Z_{crit})	**ft**	**9.0**	Equation 1 & 2
Radiant Heat Release rate	Btu/s	76.0	Q_{eff}-Radiant Frac.
Critical Radial Flux to Distance	**ft**	**3.5**	(Equation 5)
Total Heat (Qtot to HGL)	Btu	700000	<-User Input
Estimated Heat Loss Fraction (HLF)		0.70	<-User Input
Calculated Qnet	Btu	210000	Qtot (1-HLF)
Calculated Enclosure Volume, V	ft^3	351648	
Calculated Qnet/V	Btu/ft^3	0.60	
HGL Temperature Increase ($T_{hgl\text{-}incr}$)	F	36	Equation 3 and 4
Hot Gas Layer Temp. (T_{hgl})	**F**	**136**	Ta+HGL Temp incr.

Table C.2-2 (9)			
Location: 1-3	**Reactor Building, Unit 1, EL 593**		
Fire Source: 1-3-1 for RCW Pump 1A	**RCW Pump 1A/1B**		
Floor Area	ft^2	12138	<-User Input
Maximum Ambient Temperature (Ta)	(^{0}F)	100	<-User Input
Critical Radial Flux to Target	Btu/s/ft^2	0.5	<-User Input
Radiant Fraction of Heat Release		0.4	<-User Input
Critical Damage Temperature (Tc)	(^{0}F)	425	<-User Input
Height from Fire Source to Ceiling, H	ft	22	<-User Input
Fire heat release rate (Q)	Btu/sec	190	<-User Input
Location factor (LF)		1	<-User Input
Effective heat release rate (Q_{eff})	Btu/sec	190	Q*LF
Critical temperature rise $(\Delta T)_{crit}$	(^{0}F)	320	T_c-T_a-$T_{hgl\text{-}incr}$
Damage Threshold Elevation (Z_{crit})	**ft**	**8.5**	Equation 1 & 2
Radiant Heat Release rate	Btu/s	76.0	Q_{eff}·Radiant Frac.
Critical Radial Flux to Distance	**ft**	**3.5**	(Equation 5)
Total Heat (Qtot to HGL)	Btu	75000	<-User Input
Estimated Heat Loss Fraction (HLF)		0.70	<-User Input
Calculated Qnet	Btu	22500	Qtot (1-HLF)
Calculated Enclosure Volume, V	ft^3	267036	
Calculated Qnet/V	Btu/ft^3	0.08	
HGL Temperature Increase ($T_{hgl\text{-}incr}$)	F	5	Equation 3 and 4
Hot Gas Layer Temp. (T_{hgl})	**F**	**105**	Ta+HGL Temp incr.

*Note: RCW pump motor is < 100 HP; use small electric fire heat release rate, 190 BTU/s.
Assume RCW pump total heat is comparable with RBCCW pump value (75,000 BTU)*

Table C.2-2 (10)			
Location: 1-3	**Reactor Building, Unit 1, EL 593**		
Fire Source	**Panel 25-6A RPS & NSSS RPS II**		
Floor Area	ft^2	12138	<-User Input
Maximum Ambient Temperature (Ta)	(^{0}F)	100	<-User Input
Critical Radial Flux to Target	Btu/s/ft^2	0.5	<-User Input
Radiant Fraction of Heat Release		0.4	<-User Input
Critical Damage Temperature (Tc)	(^{0}F)	425	<-User Input
Height from Fire Source to Ceiling, H	ft	22	<-User Input
Fire heat release rate (Q)	Btu/sec	190	<-User Input
Location factor (LF)		1	<-User Input
Effective heat release rate (Q_{eff})	Btu/sec	190	Q*LF
Critical temperature rise $(\Delta T)_{crit}$	(^{0}F)	311	T_c-T_a-$T_{hgl\text{-}incr}$
Damage Threshold Elevation (Z_{crit})	**ft**	**8.6**	Equation 1 & 2
Radiant Heat Release rate	Btu/s	76.0	Q_{eff}-Radiant Frac.
Critical Radial Flux to Distance	**ft**	**3.5**	(Equation 5)
Total Heat (Qtot to HGL)	Btu	210000	<-User Input
Estimated Heat Loss Fraction (HLF)		0.70	<-User Input
Calculated Qnet	Btu	63000	Qtot (1-HLF)
Calculated Enclosure Volume; V	ft^3	267036	
Calculated Qnet/V	Btu/ft^3	0.24	
HGL Temperature Increase ($T_{hgl\text{-}incr}$)	F	14	Equation 3 and 4
Hot Gas Layer Temp. (T_{hgl})	**F**	**114**	Ta+HGL Temp incr.

Table C.2-2 (11)			
Location: 1-3	**Reactor Building, Unit 1, EL 593**		
Fire Source	**Panel 25-6-001**		
Floor Area	ft^2	12138	<-User Input
Maximum Ambient Temperature (Ta)	(^{0}F)	100	<-User Input
Critical Radial Flux to Target	Btu/s/ft^2	0.5	<-User Input
Radiant Fraction of Heat Release		0.4	<-User Input
Critical Damage Temperature (Tc)	(^{0}F)	425	<-User Input
Height from Fire Source to Ceiling, H	ft	22	<-User Input
Fire heat release rate (Q)	Btu/sec	190	<-User Input
Location factor (LF)		1	<-User Input
Effective heat release rate (Q_{eff})	Btu/sec	190	Q*LF
Critical temperature rise $(\Delta T)_{crit}$	(^{0}F)	316	T_c-T_a-$T_{hgl-incr}$
Damage Threshold Elevation (Z_{crit})	**ft**	**8.5**	Equation 1 & 2
Radiant Heat Release rate	Btu/s	76.0	Q_{eff}*Radiant Frac.
Critical Radial Flux to Distance	**ft**	**3.5**	(Equation 5)
Total Heat (Qtot to HGL)	Btu	140000	<-User Input
Estimated Heat Loss Fraction (HLF)		0.70	<-User Input
Calculated Qnet	Btu	42000	Qtot (1-HLF)
Calculated Enclosure Volume, V	ft^3	267036	
Calculated Qnet/V	Btu/ft^3	0.16	
HGL Temperature Increase ($T_{hgl-incr}$)	F	9	Equation 3 and 4
Hot Gas Layer Temp. (T_{hgl})	**F**	**109**	Ta+HGL Temp incr.

Table C.2-2 (12)			
Location: 1-4	**Reactor Building, Unit 1, EL 593**		
Fire Source	**Panel 25-5A RPS & NSSS RPS II**		
Floor Area	ft^2	12138	<-User Input
Maximum Ambient Temperature (Ta)	(^{0}F)	100	<-User Input
Critical Radial Flux to Target	$Btu/s/ft^2$	0.5	<-User Input
Radiant Fraction of Heat Release		0.4	<-User Input
Critical Damage Temperature (Tc)	(^{0}F)	425	<-User Input
Height from Fire Source to Ceiling, H	ft	22	<-User Input
Fire heat release rate (Q)	Btu/sec	190	<-User Input
Location factor (LF)		1	<-User Input
Effective heat release rate (Q_{eff})	Btu/sec	190	Q*LF
Critical temperature rise $(\Delta T)_{crit}$	(^{0}F)	311	T_c-T_a-$T_{hgl\text{-}incr}$
Damage Threshold Elevation (Z_{crit})	**ft**	**8.6**	Equation 1 & 2
Radiant Heat Release rate	Btu/s	76.0	Q_{eff}-Radiant Frac.
Critical Radial Flux to Distance	**ft**	**3.5**	(Equation 5)
Total Heat (Qtot to HGL)	Btu	210000	<-User Input
Estimated Heat Loss Fraction (HLF)		0.70	<-User Input
Calculated Qnet	Btu	63000	Qtot (1-HLF)
Calculated Enclosure Volume, V	ft^3	267036	
Calculated Qnet/V	Btu/ft^3	0.24	
HGL Temperature Increase ($T_{hgl\text{-}incr}$)	F	14	Equation 3 and 4
Hot Gas Layer Temp. (T_{hgl})	**F**	**114**	Ta+HGL Temp incr.

Table C.2-2 (13)			
Location: 1-4	**Reactor Building, Unit 1, EL 593**		
Fire Source	**Panel 25-218**		
Floor Area	ft^2	12138	<-User Input
Maximum Ambient Temperature (Ta)	(^{0}F)	100	<-User Input
Critical Radial Flux to Target	Btu/s/ft^2	0.5	<-User Input
Radiant Fraction of Heat Release		0.4	<-User Input
Critical Damage Temperature (Tc)	(^{0}F)	425	<-User Input
Height from Fire Source to Ceiling, H	ft	22	<-User Input
Fire heat release rate (Q)	Btu/sec	190	<-User Input
Location factor (LF)		1	<-User Input
Effective heat release rate (Q_{eff})	Btu/sec	190	Q*LF
Critical temperature rise $(\Delta T)_{crit}$	(^{0}F)	316	T_c-T_a-$T_{hgl\text{-}incr}$
Damage Threshold Elevation (Z_{crit})	**ft**	**8.5**	Equation 1 & 2
Radiant Heat Release rate	Btu/s	76.0	Q_{eff}-Radiant Frac.
Critical Radial Flux to Distance	**ft**	**3.5**	(Equation 5)
Total Heat (Qtot to HGL)	Btu	140000	<-User Input
Estimated Heat Loss Fraction (HLF)		0.70	<-User Input
Calculated Qnet	Btu	42000	Qtot (1-HLF)
Calculated Enclosure Volume, V	ft^3	267036	
Calculated Qnet/V	Btu/ft^3	0.16	
HGL Temperature Increase ($T_{hgl\text{-}incr}$)	F	9	Equation 3 and 4
Hot Gas Layer Temp. (T_{hgl})	**F**	**109**	Ta+HGL Temp incr.

Table C.2-2 (14)			
Location: 1-4	**Reactor Building, Unit 1, EL 593**		
Fire Source	**RBCCW Pump 1A/1B**		
Floor Area	ft^2	12138	<-User Input
Maximum Ambient Temperature (Ta)	(^0F)	100	<-User Input
Critical Radial Flux to Target	$Btu/s/ft^2$	0.5	<-User Input
Radiant Fraction of Heat Release		0.4	<-User Input
Critical Damage Temperature (Tc)	(^0F)	425	<-User Input
Height from Fire Source to Ceiling, H	ft	22	<-User Input
Fire heat release rate (Q)	Btu/sec	190	<-User Input
Location factor (LF)		1	<-User Input
Effective heat release rate (Q_{eff})	Btu/sec	190	Q*LF
Critical temperature rise $(\Delta T)_{crit}$	(^0F)	320	T_c-T_a-$T_{hgl\text{-}incr}$
Damage Threshold Elevation (Z_{crit})	**ft**	**8.5**	Equation 1 & 2
Radiant Heat Release rate	Btu/s	76.0	Q_{eff}-Radiant Frac.
Critical Radial Flux to Distance	**ft**	**3.5**	(Equation 5)
Total Heat (Qtot to HGL)	Btu	75000	<-User Input
Estimated Heat Loss Fraction (HLF)		0.70	<-User Input
Calculated Qnet	Btu	22500	Qtot (1-HLF)
Calculated Enclosure Volume, V	ft^3	267036	
Calculated Qnet/V	Btu/ft^3	0.08	
HGL Temperature Increase ($T_{hgl\text{-}incr}$)	F	5	Equation 3 and 4
Hot Gas Layer Temp. (T_{hgl})	**F**	**105**	Ta+HGL Temp incr.

Note: RBCCW pump motor is < 100 HP; use small electric fire heat release rate, 190 BTU/s. Assume RBCCW pump total heat is comparable with Unit 2 value (75,000 BTU)

Table C.2-2 (15)			
Location: 1-5	**Reactor Building, Unit 1, EL 621**		
Fire Source 1-5-1	**240V Lighting Board 1A**		
Floor Area	ft^2	10062	<-User Input
Maximum Ambient Temperature (Ta)	(^{0}F)	100	<-User Input
Critical Radial Flux to Target	Btu/s/ft^2	0.5	<-User Input
Radiant Fraction of Heat Release		0.4	<-User Input
Critical Damage Temperature (Tc)	(^{0}F)	425	<-User Input
Height from Fire Source to Ceiling, H	ft	13	<-User Input
Fire heat release rate (Q)	Btu/sec	190	<-User Input
Location factor (LF)		1	<-User Input
Effective heat release rate (Q_{eff})	Btu/sec	190	Q*LF
Critical temperature rise $(\Delta T)_{crit}$	(^{0}F)	273	T_c-T_a-$T_{hgl\text{-}Incr}$
Damage Threshold Elevation (Z_{crit})	**ft**	**9.3**	Equation 1 & 2
Radiant Heat Release rate	Btu/s	76.0	Q_{eff}-Radiant Frac.
Critical Radial Flux to Distance	**ft**	**3.5**	(Equation 5)
Total Heat (Qtot to HGL)	Btu	373333	<-User Input
Estimated Heat Loss Fraction (HLF)		0.70	<-User Input
Calculated Qnet	Btu	112000	Qtot (1-HLF)
Calculated Enclosure Volume, V	ft^3	130806	
Calculated Qnet/V	Btu/ft^3	0.86	
HGL Temperature Increase ($T_{hgl\text{-}Incr}$)	F	52	Equation 3 and 4
Hot Gas Layer Temp. (T_{hgl})	**F**	**152**	Ta+HGL Temp incr.
Note: Qtot is based on one section fire from 3 vertical sections (counted from left to right)			

Table C.2-2 (16)			
Location: 1-5	**Reactor Building, Unit 1, EL 621**		
Fire Source 1-5-2	**240V Lighting Transformer TL-1A**		
Floor Area	ft^2	10062	<-User Input
Maximum Ambient Temperature (Ta)	(^{0}F)	100	<-User Input
Critical Radial Flux to Target	Btu/s/ft^2	0.5	<-User Input
Radiant Fraction of Heat Release		0.4	<-User Input
Critical Damage Temperature (Tc)	(^{0}F)	425	<-User Input
Height from Fire Source to Ceiling, H	ft	13	<-User Input
Fire heat release rate (Q)	Btu/sec	224	<-User Input
Location factor (LF)		2	<-User Input
Effective heat release rate (Q_{eff})	Btu/sec	448	Q*LF
Critical temperature rise $(\Delta T)_{crit}$	(^{0}F)	244	T_c-T_a-$T_{hgl\text{-}incr}$
Damage Threshold Elevation (Z_{crit})	**ft**	**14.0**	Equation 1 & 2
Radiant Heat Release rate	Btu/s	179.2	Q_{eff}-Radiant Frac.
Critical Radial Flux to Distance	**ft**	**5.3**	(Equation 5)
Total Heat (Qtot to HGL)	Btu	560000	<-User Input
Estimated Heat Loss Fraction (HLF)		0.70	<-User Input
Calculated Qnet	Btu	168000	Qtot (1-HLF)
Calculated Enclosure Volume, V	ft^3	130806	
Calculated Qnet/V	Btu/ft^3	1.28	
HGL Temperature Increase ($T_{hgl\text{-}Incr}$)	F	81	Equation 3 and 4
Hot Gas Layer Temp. (T_{hgl})	**F**	**181**	Ta+HGL Temp incr.
Note: Plume height is higher than ceiling, hot gas layer will be formed. See Section 6.2.1.			

Table C.2-2 (17)			
Location: 1-5	**Reactor Building, Unit 1, EL 621**		
Fire Source 1-5-3 or 1-5-4	**4KV-480V Transformer (TS1A or TS1B)**		
Floor Area	ft^2	10062	<-User Input
Maximum Ambient Temperature (Ta)	(^0F)	100	<-User Input
Critical Radial Flux to Target	$Btu/s/ft^2$	0.5	<-User Input
Radiant Fraction of Heat Release		0.4	<-User Input
Critical Damage Temperature (Tc)	(^0F)	425	<-User Input
Height from Fire Source to Ceiling, H	ft	13	<-User Input
Fire heat release rate (Q)	Btu/sec	112	<-User Input
Location factor (LF)		2	<-User Input
Effective heat release rate (Q_{eff})	Btu/sec	224	Q*LF
Critical temperature rise $(\Delta T)_{crit}$	(^0F)	286	T_c-T_a-$T_{hgl\text{-}incr}$
Damage Threshold Elevation (Z_{crit})	**ft**	**9.7**	Equation 1 & 2
Radiant Heat Release rate	Btu/s	89.6	Q_{eff}-Radiant Frac.
Critical Radial Flux to Distance	**ft**	**3.8**	(Equation 5)
Total Heat (Qtot to HGL)	Btu	280000	<-User Input
Estimated Heat Loss Fraction (HLF)		0.70	<-User Input
Calculated Qnet	Btu	84000	Qtot (1-HLF)
Calculated Enclosure Volume, V	ft^3	130806	
Calculated Qnet/V	Btu/ft^3	0.64	
HGL Temperature Increase ($T_{hgl\text{-}incr}$)	F	39	Equation 3 and 4
Hot Gas Layer Temp. (T_{hgl})	**F**	**139**	Ta+HGL Temp incr.

Table C.2-2 (18)			
Location: 1-5	**Reactor Building, Unit 1, EL 621**		
Fire Source 1-5-5 or 1-5-6	**4KV RPT Board 1-1 or Board 1-2**		
Floor Area	ft^2	10062	<-User Input
Maximum Ambient Temperature (Ta)	(^{0}F)	100	<-User Input
Critical Radial Flux to Target	Btu/s/ft^2	0.5	<-User Input
Radiant Fraction of Heat Release		0.4	<-User Input
Critical Damage Temperature (Tc)	(^{0}F)	425	<-User Input
Height from Fire Source to Ceiling, H	ft	13	<-User Input
Fire heat release rate (Q)	Btu/sec	190	<-User Input
Location factor (LF)		1	<-User Input
Effective heat release rate (Q_{eff})	Btu/sec	190	Q*LF
Critical temperature rise $(\Delta T)_{crit}$	(^{0}F)	223	T_c-T_a-$T_{hgl\text{-}Incr}$
Damage Threshold Elevation (Z_{crit})	**ft**	**10.5**	Equation 1 & 2
Radiant Heat Release rate	Btu/s	76.0	Q_{eff}-Radiant Frac.
Critical Radial Flux to Distance	**ft**	**3.5**	(Equation 5)
Total Heat (Qtot to HGL)	Btu	700000	<-User Input
Estimated Heat Loss Fraction (HLF)		0.70	<-User Input
Calculated Qnet	Btu	210000	Qtot (1-HLF)
Calculated Enclosure Volume, V	ft^3	130806	
Calculated Qnet/V	Btu/ft^3	1.61	
HGL Temperature Increase ($T_{hgl\text{-}incr}$)	F	102	Equation 3 and 4
Hot Gas Layer Temp. (T_{hgl})	**F**	**202**	Ta+HGL Temp incr.
Note: Qtot is based on one section fire from 2 vertical sections (panel 1 and panel 2)			

Table C.2-2 (19)			
Location: 1-5	**Reactor Building, Unit 1, EL 621**		
Fire Source 1-5-7	**Panel 1-25-31 RCIC Backup Control**		
Floor Area	ft^2	10062	<-User Input
Maximum Ambient Temperature (Ta)	(^{0}F)	100	<-User Input
Critical Radial Flux to Target	Btu/s/ft^2	0.5	<-User Input
Radiant Fraction of Heat Release		0.4	<-User Input
Critical Damage Temperature (Tc)	(^{0}F)	425	<-User Input
Height from Fire Source to Ceiling, H	ft	13	<-User Input
Fire heat release rate (Q)	Btu/sec	190	<-User Input
Location factor (LF)		1	<-User Input
Effective heat release rate (Q_{eff})	Btu/sec	190	Q*LF
Critical temperature rise $(\Delta T)_{crit}$	(^{0}F)	184	T_c-T_a-$T_{hgl\text{-}incr}$
Damage Threshold Elevation (Z_{crit})	**ft**	**11.8**	Equation 1 & 2
Radiant Heat Release rate	Btu/s	76.0	Q_{eff}-Radiant Frac.
Critical Radial Flux to Distance	**ft**	**3.5**	(Equation 5)
Total Heat (Qtot to HGL)	Btu	933333	<-User Input
Estimated Heat Loss Fraction (HLF)		0.70	<-User Input
Calculated Qnet	Btu	280000	Qtot (1-HLF)
Calculated Enclosure Volume, V	ft^3	130806	
Calculated Qnet/V	Btu/ft^3	2.14	
HGL Temperature Increase ($T_{hgl\text{-}incr}$)	F	141	Equation 3 and 4
Hot Gas Layer Temp. (T_{hgl})	**F**	**241**	Ta+HGL Temp incr.
Note: There are 2 vertical sections, left side is twice as large as right side. Assume left section catches fire.			

Table C.2-2 (20)			
Location: 1-5	**Reactor Building, Unit 1, EL 621**		
Fire Source 1-5-8 -	**Panel 25-3 Filter Demin**		
Floor Area	ft^2	10062	<-User Input
Maximum Ambient Temperature (Ta)	(^{0}F)	100	<-User Input
Critical Radial Flux to Target	Btu/s/ft^2	0.5	<-User Input
Radiant Fraction of Heat Release		0.4	<-User Input
Critical Damage Temperature (Tc)	(^{0}F)	425	<-User Input
Height from Fire Source to Ceiling, H	ft	13	<-User Input
Fire heat release rate (Q)	Btu/sec	190	<-User Input
Location factor (LF)		1	<-User Input
Effective heat release rate (Q_{eff})	Btu/sec	190	Q*LF
Critical temperature rise $(\Delta T)_{crit}$	(^{0}F)	223	T_c-T_a-$T_{hgl-incr}$
Damage Threshold Elevation (Z_{crit})	**ft**	**10.5**	Equation 1 & 2
Radiant Heat Release rate	Btu/s	76.0	Q_{eff}-Radiant Frac.
Critical Radial Flux to Distance	**ft**	**3.5**	(Equation 5)
Total Heat (Qtot to HGL)	Btu	700000	<-User Input
Estimated Heat Loss Fraction (HLF)		0.70	<-User Input
Calculated Qnet	Btu	210000	Qtot (1-HLF)
Calculated Enclosure Volume, V	ft^3	130806	
Calculated Qnet/V	Btu/ft^3	1.61	
HGL Temperature Increase ($T_{hgl-incr}$)	F	102	Equation 3 and 4
Hot Gas Layer Temp. (T_{hgl})	**F**	**202**	Ta+HGL Temp Incr.
Note: Qtot is based on the fact that panel has a single vertical section			

Table C.2-2 (21)			
Location: 1-5	**Reactor Building, Unit 1, EL 621**		
Fire Source	**Panel 25-9 Sample Panel**		
Floor Area	ft^2	10062	<-User Input
Maximum Ambient Temperature (Ta)	(^0F)	100	<-User Input
Critical Radial Flux to Target	$Btu/s/ft^2$	0.5	<-User Input
Radiant Fraction of Heat Release		0.4	<-User Input
Critical Damage Temperature (Tc)	(^0F)	425	<-User Input
Height from Fire Source to Ceiling, H	ft	13	<-User Input
Fire heat release rate (Q)	Btu/sec	190	<-User Input
Location factor (LF)		1	<-User Input
Effective heat release rate (Q_{eff})	Btu/sec	190	Q*LF
Critical temperature rise $(\Delta T)_{crit}$	(^0F)	316	T_c-T_a-$T_{hgl\text{-}incr}$
Damage Threshold Elevation (Z_{crit})	**ft**	**8.5**	Equation 1 & 2
Radiant Heat Release rate	Btu/s	76.0	Q_{eff}·Radiant Frac.
Critical Radial Flux to Distance	**ft**	3.5	(Equation 5)
Total Heat (Qtot to HGL)	Btu	70000	<-User Input
Estimated Heat Loss Fraction (HLF)		0.70	<-User Input
Calculated Qnet	Btu	21000	Qtot (1-HLF)
Calculated Enclosure Volume, V	ft^3	130806	
Calculated Qnet/V	Btu/ft^3	0.16	
HGL Temperature Increase ($T_{hgl\text{-}incr}$)	F	9	Equation 3 and 4
Hot Gas Layer Temp. (T_{hgl})	**F**	**109**	Ta+HGL Temp incr.
Note: Qtot is based on the fact that panel has a single vertical section			

Table C.2-2 (22)			
Location: 1-5	**Reactor Building, Unit 1, EL 639**		
Fire Source	**SLC Pump A or Pump B (oil)**		
Floor Area	ft^2	2451	<-User Input
Maximum Ambient Temperature (Ta)	(^0F)	100	<-User Input
Critical Radial Flux to Target	$Btu/s/ft^2$	0.5	<-User Input
Radiant Fraction of Heat Release		0.4	<-User Input
Critical Damage Temperature (Tc)	(^0F)	425	<-User Input
Height from Fire Source to Ceiling, H	ft	18	<-User Input
Fire heat release rate (Q)	Btu/sec	450	<-User Input
Location factor (LF)		1	<-User Input
Effective heat release rate (Q_{eff})	Btu/sec	450	Q*LF
Critical temperature rise $(\Delta T)_{crit}$	(^0F)	179	T_c-T_a-$T_{hgl\text{-}incr}$
Damage Threshold Elevation (Z_{crit})	**ft**	**16.9**	Equation 1 & 2
Radiant Heat Release rate	Btu/s	180.0	Q_{eff}-Radiant Frac.
Critical Radial Flux to Distance	**ft**	**5.4**	(Equation 5)
Total Heat (Qtot to HGL)	Btu	326250	<-User Input
Estimated Heat Loss Fraction (HLF)		0.70	<-User Input
Calculated Qnet	Btu	97875	Qtot (1-HLF)
Calculated Enclosure Volume, V	ft^3	44118	
Calculated Qnet/V	Btu/ft^3	2.22	
HGL Temperature Increase ($T_{hgl\text{-}incr}$)	F	146	Equation 3 and 4
Hot Gas Layer Temp. (T_{hgl})	**F**	**246**	Ta+HGL Temp incr.

Table C.2-2 (23)			
Location: 1-5	**Reactor Building, Unit 1, EL 639**		
Fire Source	**Panel 1-25-213 or 1-25-222**		
Floor Area	ft^2	2451	<-User Input
Maximum Ambient Temperature (Ta)	(^{0}F)	100	<-User Input
Critical Radial Flux to Target	Btu/s/ft^2	0.5	<-User Input
Radiant Fraction of Heat Release		0.4	<-User Input
Critical Damage Temperature (Tc)	(^{0}F)	425	<-User Input
Height from Fire Source to Ceiling, H	ft	18	<-User Input
Fire heat release rate (Q)	Btu/sec	190	<-User Input
Location factor (LF)		1	<-User Input
Effective heat release rate (Q_{eff})	Btu/sec	190	Q*LF
Critical temperature rise $(\Delta T)_{crit}$	(^{0}F)	284	T_c-T_a-$T_{hgl\text{-}incr}$
Damage Threshold Elevation (Z_{crit})	**ft**	**9.1**	Equation 1 & 2
Radiant Heat Release rate	Btu/s	76.0	Q_{eff}-Radiant Frac.
Critical Radial Flux to Distance	**ft**	**3.5**	(Equation 5)
Total Heat (Qtot to HGL)	Btu	100000	<-User Input
Estimated Heat Loss Fraction (HLF)		0.70	<-User Input
Calculated Qnet	Btu	30000	Qtot (1-HLF)
Calculated Enclosure Volume, V	ft^3	44118	
Calculated Qnet/V	Btu/ft^3	0.68	
HGL Temperature Increase ($T_{hgl\text{-}incr}$)	F	41	Equation 3 and 4
Hot Gas Layer Temp. (T_{hgl})	**F**	**141**	Ta+HGL Temp incr.

Table C.2-2 (24)			
Location: 1-6	**Reactor Building, Unit 1, EL 639**		
Fire Source	**VFD 1A/1B (Panel)**		
Floor Area	ft^2	8109	<-User Input
Maximum Ambient Temperature (Ta)	(^{0}F)	100	<-User Input
Critical Radial Flux to Target	Btu/s/ft^2	0.5	<-User Input
Radiant Fraction of Heat Release		0.4	<-User Input
Critical Damage Temperature (Tc)	(^{0}F)	425	<-User Input
Height from Fire Source to Ceiling, H	ft	13	<-User Input
Fire heat release rate (Q)	Btu/sec	190	<-User Input
Location factor (LF)		1	<-User Input
Effective heat release rate (Q_{eff})	Btu/sec	190	Q*LF
Critical temperature rise $(\Delta T)_{crit}$	(^{0}F)	See Note 1	T_c-T_a-$T_{hgl\text{-}incr}$
Damage Threshold Elevation (Z_{crit})	**ft**	**See Note 1**	Equation 1 & 2
Radiant Heat Release rate	Btu/s	76.0	Q_{eff}*Radiant Frac.
Critical Radial Flux to Distance	**ft**	3.5	(Equation 5)
Total Heat (Qtot to HGL)	Btu	54502295	<-User Input
Estimated Heat Loss Fraction (HLF)		0.70	<-User Input
Calculated Qnet	Btu	16350689	Qtot (1-HLF)
Calculated Enclosure Volume, V	ft^3	105417	
Calculated Qnet/V	Btu/ft^3	155.10	
HGL Temperature Increase ($T_{hgl\text{-}incr}$)	F	See Note 1	Equation 3 and 4
Hot Gas Layer Temp. (T_{hgl})	**F**	**See Note 1**	Ta+HGL Temp incr.
Note 1: The Unit 1 Restart DCN Combustible Loading Change Summary (includes Sargent & Lundy Mods) specifies 109,004,590 BTUs of total heat of combustion for VFD's 1A and 1B. So each VFD (treated as panel) has 54,502,295 BTUs. This will cause hot gas layer to form in fire zone 1-6.			

Referenced Evaluations

$$\Delta T(^0F) = 340\frac{Q_{eff}^{2/3}(btu/sec)}{H_{target}^{5/3}(ft)} \ldots\ldots\ldots(1)$$

$$Zcrit = 33\frac{Q_{eff}^{2/5}}{\Delta T_{crit}^{3/5}} \ldots\ldots\ldots(2)$$

$$\Delta T = T_0\left[e^{(Q_{net}/V\rho_0 C_p T_0)} - 1\right] \ldots\ldots\ldots(3)$$

$$Q_0 = \rho_0 T_0 C_p V = 0.071 * 0.24 * 560 * V = 9.54V \ldots\ldots\ldots(4)$$

$$R_{crit} = \sqrt{\frac{Q_{rad}}{4\pi q_{crit}}} \ldots\ldots\ldots(5)$$

ENCLOSURE 3

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN) UNIT 1

RESPONSE TO NRC GENERIC LETTER (GL) 88-20, SUPPLEMENT 4
INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE) FOR
SEVERE ACCIDENT VULNERABILITIES

SUMMARY OF COMMITMENTS

1. TVA will complete corrective actions to address the seismic-induced fire vulnerability associated with the emergency lighting battery racks located in the BFN Unit 1 cable spreading room prior to restart of BFN Unit 1.