Station Support Department

10CFR50.54(f)
GL 88-20 Sup. 4

PECO ENERGY

PECO Energy Company
Nuclear Group Headquarters
965 Chesterbrook Boulevard
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June 26, 1995

Docket Nos. 50-352
50-353

License Nos. NPF-39
NPF-85

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

SUBJECT: Limerick Generating Station, Units 1 and 2
Response to NRC Generic Letter 88-20 Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities"

Reference:

1) Letter from G. J. Beck (PECO Energy) to NRC dated September 18, 1992

2) Letter from G. A. Hunger, Jr. (PECO Energy) to NRC dated February 4, 1994

3) Letter from G. A. Hunger, Jr. (PECO Energy) to NRC dated December 19, 1994

Dear Sir:

NRC Generic Letter (GL) 88-20 Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," dated June 28, 1991, requested PECO Energy Company (PECO Energy) to perform an Individual Plant Examination of External Events (IPEEE) of the Limerick Generating Station (LGS) and submit the results to the NRC pursuant to 10CFR50.54(f).

Reference letter 1 committed PECO Energy to submitting a final report to the NRC by June 30, 1995. Accordingly, the LGS IPEEE Final Report is attached. The IPEEE was conducted in accordance with the objectives of GL 88-20 Supplement 4 and the guidance provided in NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities."

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Reference letters 2 and 3 were submitted in response to Generic Letter 92-08, "Thermo-Lag 330-1 Fire Barriers," and included details of an integrated analysis program prepared in response to the Thermo-Lag issue. This integrated analysis program includes using the fire safe shutdown re-analysis completed as part of this IPEEE final report to minimize our reliance on Thermo-Lag to achieve safe shutdown. The integrated analysis program in response to GL 92-08, and the Thermo-Lag issue, will also use insights gained from this IPEEE final report to prioritize resolution of Thermo-Lag assemblies that remain after the fire safe shutdown re-analysis.

As a result of conducting the LGS IPEEE, PECO Energy identified seismic event and fire event findings. Equipment, structures, and systems at LGS are seismically very rugged; however, actions are being taken to address minor housekeeping and maintenance issues related to the seismic analysis such as unrestrained tools, lockers, hoist controllers and lifting devices for low voltage switchgear. Fire brigade drill activities and fire brigade awareness will be increased for 3 areas in the common control structure. Actions credited in the fire analysis such as improved transient combustible controls, creation of transient combustible free zones and formal designation of certain fire rated doors as "fire" doors will be implemented at LGS.

If you have any questions concerning this submittal, or require additional information, please contact us.

Very truly yours,

G. A. Hunger, Jr.

G. A. Hunger, Jr.,
Director
Licensing Section

Attachment

Enclosure: Affirmation

cc: T. T. Martin, Administrator, Region I, USNRC (w/ attachment and enclosure)
N. S. Perry, USNRC Senior Resident Inspector, LGS "
R. R. Janati, Commonwealth of Pennsylvania "

COMMONWEALTH OF PENNSYLVANIA :

: SS

COUNTY OF CHESTER :

W. H. Smith, III, being first duly sworn, deposes and says:

That he is Vice President of PECO Energy Company, the Applicant herein; that he has read the response to Generic Letter No. 88-20 Supplement 4 involving Limerick Generating Station, and knows the contents thereof; and that the statements and matters set forth therein are true and correct to the best of his knowledge, information and belief.

William H Smith

Vice President

Subscribed and sworn to

before me this 26th day

of June 1995.

Erica A. Santori

Notary Public

Notarial Seal
Erica A. Santori, Notary Public
Tredyffrin Twp., Chester County
My Commission Expires July 10, 1995

Limerick Generating Station
Units 1 and 2

Individual Plant Examination for External Events

PECO Energy Co.
June 1995

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1.0 EXECUTIVE SUMMARY

1.1 Background and Objectives

This report documents the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities performed for PECO Energy's Limerick Generating Station, (LGS) Units 1 and 2. The objective of the IPEEE is to perform an individual plant examination of external events to identify vulnerabilities, if any, to severe accidents and report the results together with any licensee-determined improvements and corrective actions to the Commission in accordance with the requirements of the Nuclear Regulatory Commission's Generic Letter (GL) No. 88-20, Supplement 4 (Ref. 1.1-1)

The general purpose of the IPEEE is similiar to that of the internal event IPE which is for each licensee:

* to develop an appreciation of severe accident behavior

* to understand the most likely severe accident sequences that could occur at its plant under full power operating conditions

* to gain a qualitative understanding of the overall likelihood of core damage and radioactive material release

* if necessary, to reduce the overall likelihood of core damage and radioactive material releases by modifying hardware and procedures that would help prevent or mitigate severe accidents

Another objective specified in the generic letter is to coordinate the examination with other external event programs including USI A-45, the Eastern U.S. Seismicity Issue and the "Fire Risk Scoping Study".

In its Supplement 4 to GL 88-20, the NRC recommended that only five events be included in the IPEEE. However, the NRC cautioned that the licensee should confirm that no plant-unique external events known to the licensee with the potential to initiate severe accidents are excluded from the IPEEE. The five external events requested to be assessed were:

(1) Seismic Events

(2) Internal Fires

(3) High Winds and Tornadoes

(4) External Floods

(5) Transportation and Nearby Facility Accidents

In addition, the NRC identified certain examination methods as being acceptable for performance of the IPEEE. For seismic events, the methods were either a seismic probabilistic risk assessment with enhancements or one of two seismic margin assessments with enhancements. Internal fire events could be treated by performing a Level 1 fire PRA as described in NUREG/CR-2300 (ref. 1.1-2), a simplified fire PRA as described in NUREG/CR-4840 (ref. 1.1-3), or possibly a Fire Induced Vulnerability Evaluation (FIVE) (ref. 1.1-4) which was under review by the NRC at the time the Generic Letter was issued. (The FIVE approach was subsequently approved by the NRC.) Finally, a screening type approach was identified for high winds, floods and transportation and nearby facility accidents.

PECO Energy chose the EPRI seismic margin assessment method for addressing seismic events, the FIVE approach for evaluating internal fires, and the screening technique for high winds, floods and transportation and nearby facility accidents. No plant unique external events were identified for examination.

LGS was previously evaluated for external events in the Severe Accident Risk Assessment report (ref. 1.1-5) which was reviewed in NUREG/CR-3493 (ref. 1.1.6).

1.2 Plant Familization

The Limerick Generating Station, owned and operated by the PECO Energy Company, is located on the east bank of the Schuylkill River in the Limerick Township of Montgomery County, Pennsylvania. The Station is approximately 4 miles downriver from Pottstown, 35 miles upriver from Philadelphia, and 49 miles above the confluence of the Schuylkill with the Delaware River. The site contains 595 acres (423 acres in Montgomery County and 172 in Chester County).

The Limerick Generating Station consists of two boiling water reactor (BWR) generating units. Each unit was originally designed to operate at a rated core thermal power of 3293 MWt with a corresponding gross electrical output of 1092 MWe. In February, 1995, Unit 2's thermal power was increased 5% to 3458 MWt with a new corresponding gross

electrical output of 1163 MWe. Unit 1 is scheduled to be rerated to 105% thermal power following its next refueling outage in January-February 1996.

The Units are BWR 4's having the following safety systems:

- turbine driven high pressure coolant injection (HPCI) system
- turbine driven reactor core isolation cooling (RCIC) system
- automatic depressurization system (ADS)
- motor driven core spray system
- low pressure coolant injection (LPCI) mode of the Residual Heat Removal system
- Four emergency diesel generators per unit

The Units have Mark II containment systems designed to limit the release of radioactive materials to the environs in the unlikely event of a breach of the reactor system. The design consists of two barriers, the primary and the secondary containment. The primary containment is a steel-lined reinforced concrete structure of the over-and-under configuration. The secondary containment is the concrete reactor enclosure, which surrounds the reactor systems, the primary containment, and the fuel storage areas.

Figure 1.2-1 shows the general site arrangement and Figure 1.2-2 shows the general reactor enclosure arrangement.

1.3 Overall Methodology

In its Supplement 4 to GL 88-20, the NRC identified certain examination methods as being acceptable for performance of the IPEEE. From the various acceptable methods, PECO Energy has chosen the EPRI seismic margin method for addressing seismic events, the EPRI developed Fire Induced Vulnerability Evaluation (FIVE) methodology for evaluating internal fires, and the screening technique for high winds, floods and transportation and nearby facility accidents.

Seismic Analysis - The seismic margin methodology developed by EPRI is documented in EPRI NP-6041-SL (ref. 1.3-1). The recommendations and guidelines presented in that report were used extensively during the

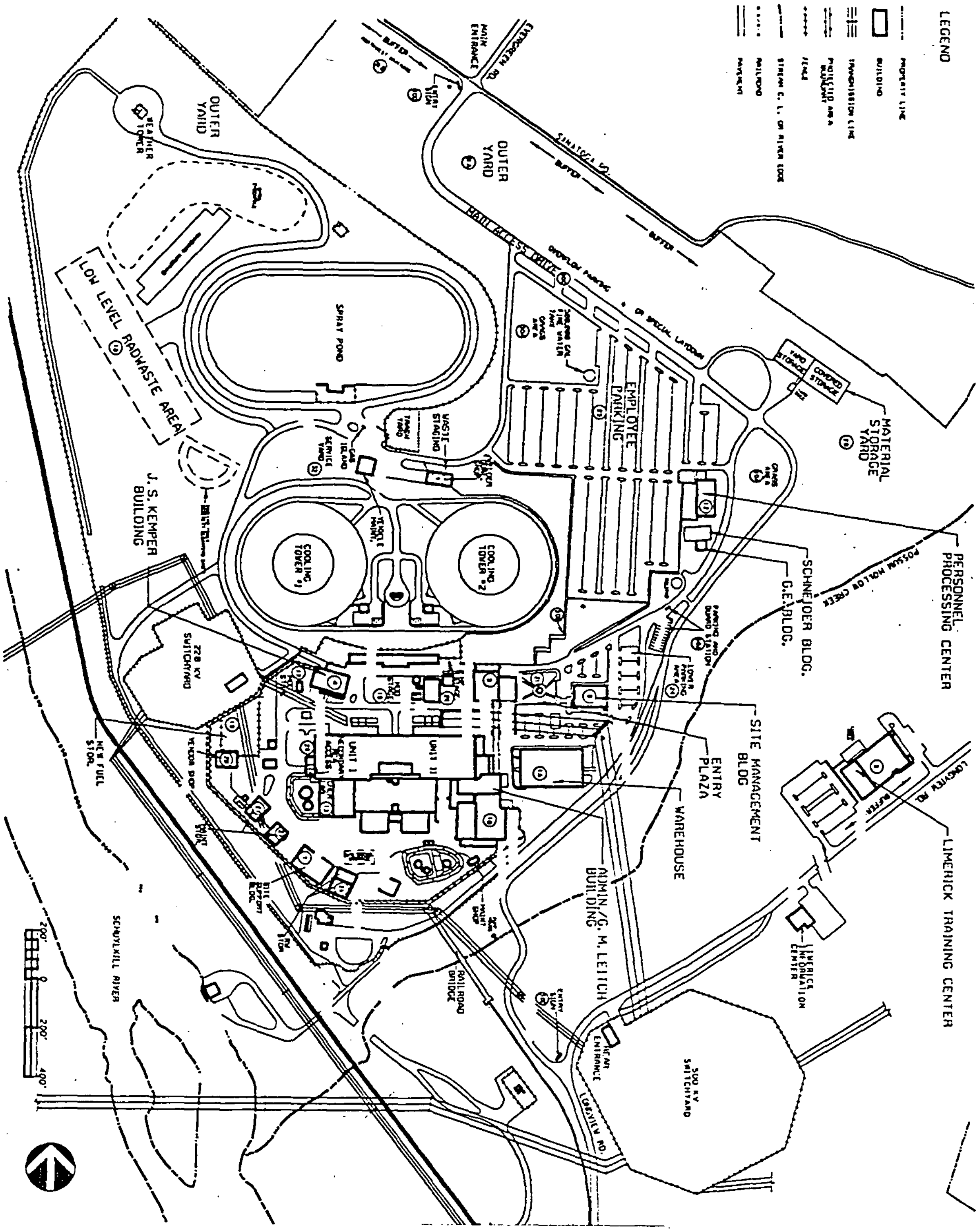

FIGURE 1.2-1
GENERAL SITE ARRANGEMENT

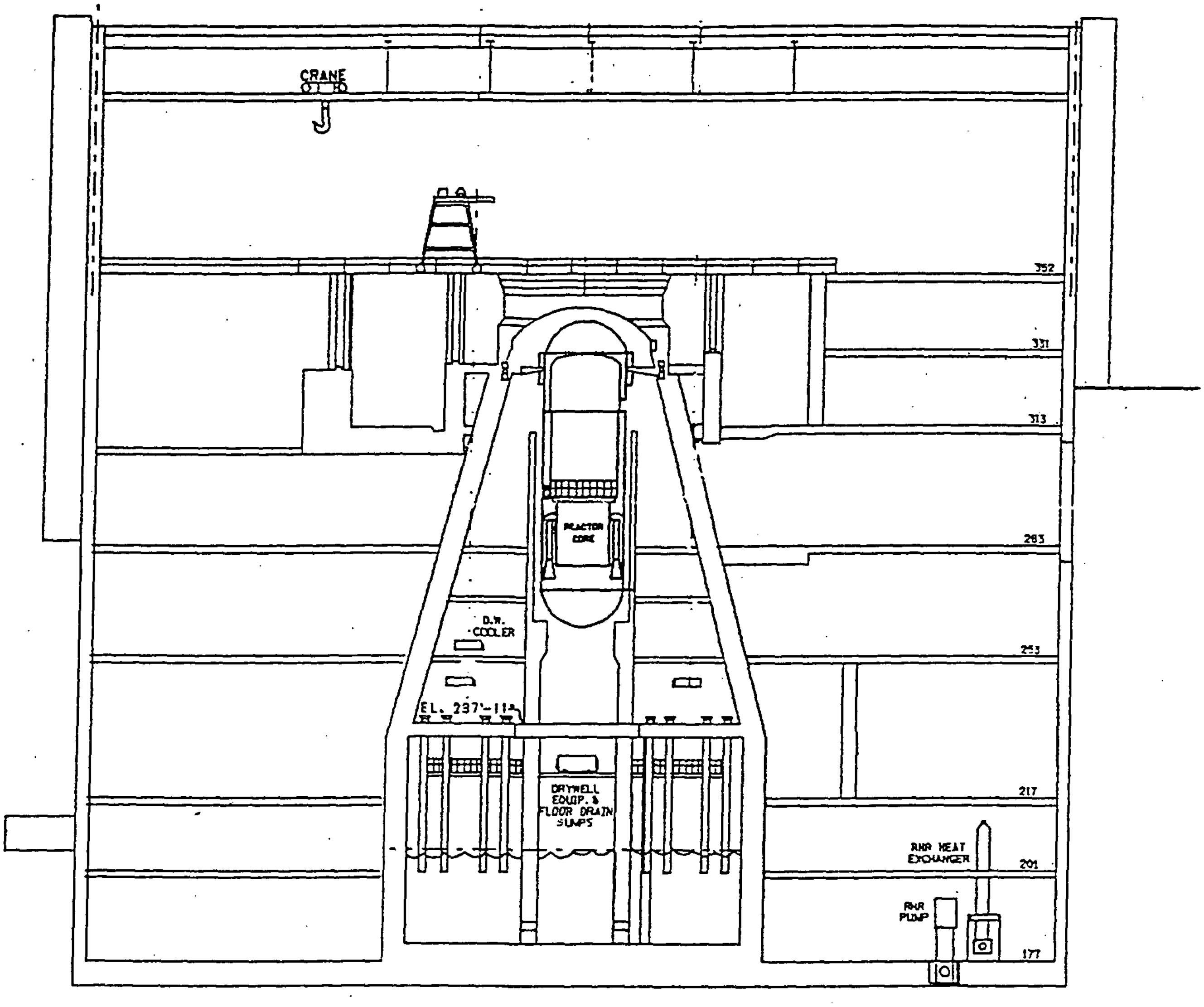

FIGURE 1.2-2
GENERAL REACTOR ENCLOSURE ARRANGEMENT

LGS seismic margin assessment. Since the seismic hazard is low for LGS, the examination was performed using a reduced-scope seismic margins approach emphasizing plant walkdowns. PECO informed the NRC of its decision to conduct a reduced-scope seismic margins analysis in a letter to the NRC dated July 28, 1994.

Fire Risk Analysis - Because it provides a comprehensive approach for screening plant areas for fire risk, FIVE was used to identify fire areas of potential risk significance. FIVE was also used to calculate area fire ignition frequencies and to provide hazard analysis for non-screened and modified critical fire areas. FIVE worksheets and equations together with a modified Level 1 PSA plant model were used to quantify the fire induced safe shutdown system unavailabilities. Plant walkdowns were conducted to obtain inplant data and to address the Fire Risk Scoping Study issues.

High winds, floods and transportation and nearby facility accidents - The evaluation of these external events follows the methodology recommended by the NRC is GL 88-20, Supplement 4 as well as Section 5 of NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities" (Ref. 1.3-2). The recommended methodology consists of a progressive screening approach shown in Figure 5.1 of NUREG-1407 and repeated herein as Figure 1.3-1.

PECO Energy personnel were involved in all phases of the evaluation. A peer review has been conducted.

1.4 Summary of Major Findings

The IPEEE analysis was performed in accordance with GL 88-20, Supplement 4 and NUREG-1407 with the results described below.

1.4.1 Seismic Event Findings

The Seismic Review Team (SRT) has concluded that the equipment, structures and distributed systems at LGS are seismically very rugged. This was attributed to the conservative nature of the original design. During the walkdowns the SRT noted some housekeeping and maintenance issues such as unrestrained tools, lockers, hoist controllers and lifting devices for the low voltage switchgear. Most of these housekeeping and maintenance issues have since been corrected by the station. Actions are being taken to address the houskeeping and maintenance issues related to the seismic analysis.

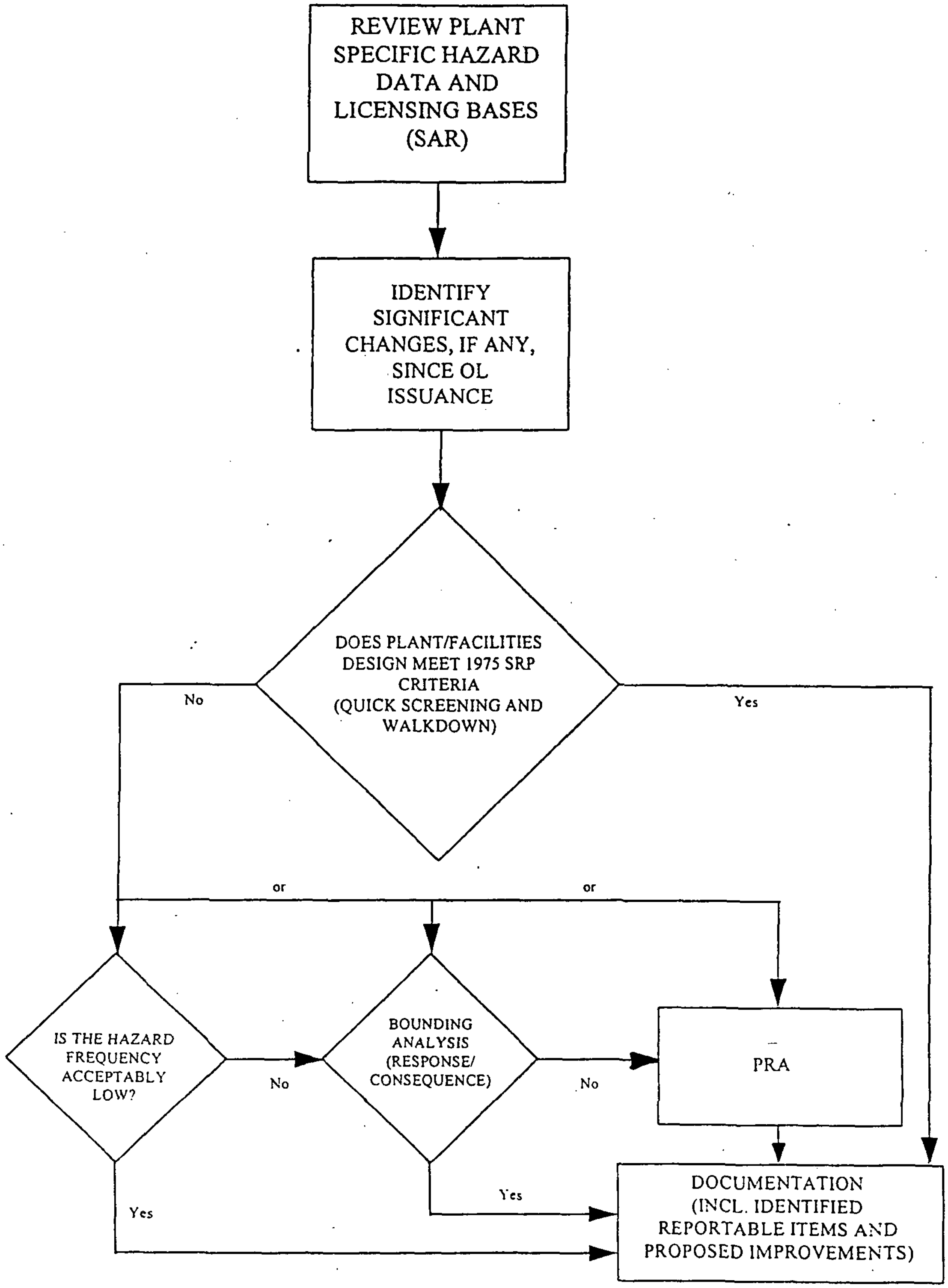

FIGURE 1.3 1
SCREENING APPROACH FOR WINDS, FLOODS AND OTHER EXTERNAL EVENTS

1.4.2 Fire Event Findings

The FIVE Methodology was used to evaluate 127 fire compartments. Of the 127 compartments all but 21 passed the initial screening and were removed from further consideration. Four of these 21 compartments screened after analysis of the qualification of combustible free zones. The remaining 17 compartments required detailed fire interaction analysis. Four compartments screened by crediting revisions to plant combustible control procedures, 1 screened with planned plant modification or structural analysis, and 7 screened with detailed fire interaction analysis following the FIVE methodology guidelines. Two additional areas screened by taking credit for the detailed fire analysis of these areas performed for the SARA report. The following 3 compartments located in the common control structure could not be screened.

Fire Area	Description
2	13 kV switchgear room
20	Unit 1 Static inverter room
26	Remote Shutdown Room

PSA techniques were used to determine the unavailability of selected systems for these compartments. Based on the result, the type of fire risk reduction measures appropriate for the compartment was identified. To manage the risk the station will increase the fire brigade drill activities and brigade awareness in these areas.

Actions credited in the fire analysis; improved transient combustible controls, creation of transient combustible free zones and formal designation of certain fire rated doors as "fire" doors will be implemented at the station. Credit was also taken for certain plant modifications related to the resolution of the Thermo-lag issue.

1.4.3 Other Event Findings

The other external events which were evaluated were high winds and tornadoes, external floods, and transportation and nearby facility accidents.

The design of the LGS plant facilities meets the NRC's 1975 Standard Review Plan criteria for each of the other external events evaluated. Additionally, 1) the site review has shown that no significant changes have occurred since the operating license was issued and 2) a site walkdown and drive around found no outdoor facilities that could be affected by high winds, onsite storage of hazardous materials, and offsite developments. Therefore, in accordance with guidance of NUREG-1407, the contribution from the each significant external hazard to core damage is less that 1.0E-6 and the IPEEE screening criteria for other external events is met.

2.0 EXAMINATION DESCRIPTION

2.1 Introduction

PECO Energy originally developed an external events Probabilistic Risk Assessment for the Limerick Generating Station titled "Severe Accident Risk Analysis (SARA) (ref 1.1-5) and submitted it to the NRC in 1983. The SARA analyzed the risk from seismic events, fires, flooding, tornadoes, turbine missiles, and transportation and related accidents in the vicinity of the plant. The SARA was reviewed by Brookhaven National Laboratory and the NRC (ref. 1.1-6, NUREG/CR-3493). The overall conclusion of this review was that the SARA appeared to use state-of-the-art methodologies for evaluation of the core melt frequency due to seismic and fire initiating events.

The seismic, fires and internal flooding analysis from the SARA were updated in 1989 in support of the Limerick Unit 2 licensing process (ref. 2.1-1) and the revised accident frequencies for these external initiators were submitted to the NRC in the PECO Energy response to a request for additional information regarding consideration of severe accident mitigation design alternatives (ref. 2.1-2).

In Section 3.1.2 of NUREG-1407 the NRC allows the use of an existing seismic/fire PRA for the IPEEE provided that the PRA reflects the current as-built and as-operated condition of the plant and that some of the deficiencies of past PRAs are adequately addressed. Rather than attempt to verify that SARA reflects current design and additionally, backfit the newer requirements into the SARA analysis, PECO opted to perform an entirely new evaluation (IPEEE) to meet the requirements of the Generic Letter and accompanying NUREG-1407 guidance.

2.2 Conformance with Generic Letter and Supporting Material

The PECO Energy Company's response to GL 88-20 dated December 26, 1991 described the methodology which the Company planned to use for its IPEEE submittal. The proposed methodology was subsequently found to be acceptable by the NRC in its letter to the Company dated June 18, 1992. Subsequently, the Company notified the NRC that it was changing the scope of seismic review from focused scope to reduced scope (Ref. 2.2-1). This change was based on the new seismic hazard estimates developed by LLNL and published in NUREG-1488 (ref. 2.2-2).

In accordance with these commitments, this report provides the information requested in Generic Letter 88-20, Supplement 4, and in NUREG-1407. The seismic analysis was completed using the Reduced-Scope EPRI Seismic Margins Method described in Section 3.2.3 of the NUREG. The fire analysis was based on the FIVE Methodology which was reviewed by the NRC and determined to be conditionally acceptable for IPEEE purposes in a letter from the NRC to NUMARC dated August 21, 1991 (ref. 2.2-3). The high winds, floods, and transportation and nearby facility accident analysis followed the progressive screening approach described in Section 5.2 of NUREG-1407.

The are some deviations from the standard Table of Contents provided in Table C.1 of NUREG-1407. Since only the Seismic Margins Method was used for the seismic analysis, it is described in a Section numbered 3.1, not 3.1b. Furthermore, two new sections have been included. Section 9 provides a comparison between the IPEEE and the previous 1983 LGS external events analysis provided in the LGS Severe Accident Risk Assessment. Section 10 was added to provide a list of references, abbreviations, and acronyms. Within Section 5, High Winds, Floods and Others a new subsection 5.0, Screening, has been added to describe the process used to ensure that all significant external events relevant to the LGS site were evaluated and addressed. Finally, to enhance readability, Sections 6.3 and 6.4 were combined so that each review comment is followed by its resolution.

Achievement of the goals of the IPEEE is discussed below:

Goal #1 Develop an appreciation of severe accident behavior

During the greater than three year development of the IPEEE, the PECO Energy staff worked with the contractor in all phases of the seismic analysis, including generation of the success path

component list, walkdown preparation, walkdowns and evaluation of the seismic capacity of the plant structures and components. Through this involvement PECO Energy personnel gained a further appreciation for the importance of proper equipment anchorage, spatial separation, and attention to housekeeping.

For the IPEEE fire risk evaluation, PECO Energy selected the FIVE methodology. In-house performance of the FIVE analysis integrated the work of PSA, safe shutdown and fire protection personnel which enhanced PECO's understanding of the risk due to fires at LGS.

Since the high winds, external flooding, transportation and nearby facility accidents analysis could be accomplished through a screening approach, no additional severe accident insights were achieved.

Members of the LGS operating and engineering staff provided review of the IPEEE report in support of the IPEEE development team. Senior Nuclear Generation Group and Corporate management is cognizant of the IPEEE and established the development and completion of the IPEEE as a 1995 Department goal.

Goal #2 Understand the most likely severe accident sequences that could occur at Limerick under full power operation

In implementing the FIVE methodology, three fire areas were identified which did not meet the screening criteria. These areas were further analyzed using probabilistic methods to enhance out understanding of these fire initiated sequences.

Goal #3 Gain a qualitative understanding of the overall likelihood of core damage and radioactive material release

The outputs of the approaches used to meet the requirements of the Generic Letter were an enumeration of potential plant vulnerabilities to external events. Since no vulnerabilities were determined to exist at LGS, the overall likelihood of core damage and radioactive material release due to external events is extremely low.

Goal #4 If necessary, reduce the overall likelihood of core damage and radioactive material releases by modifying hardware and procedures that would help prevent or mitigate severe accidents

Hardware and procedural changes resulting from the IPEEE process were identified and reviewed within PECO Energy to determine which changes would be beneficial and cost effective in preventing or mitigating externally caused severe accidents. These changes are identified in Report Section 7, Plant Improvements and Unique Safety Features.

2.3 General Methodology

2.3.1 Seismic Methodology Outline

The seismic events portion of the LGS IPEEE was examined through the use of the EPRI Seismic Margin Method (SMM) as enhanced by NUREG-1407 and GL 88-20, Supplement 4, and is documented herein consistent with the reporting requirements of a reduced-scope SMM review. Specifically, the seismic IPEEE includes the following elements:

1) Selection of Alternate Success Paths

2) Walkdown

3) Screening Criteria (Use of Screening Tables)

4) Seismic Input

5) Evaluation of Outliers

These elements were addressed in the seismic IPEEE to meet the applicable program requirements through the use of Seismic Margin Assessment Procedures as established in EPRI NP-6041-SL, Revision 1, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin."

2.3.2 Fire Methodology Outline

The Fire Induced Vulnerability Evaluation (FIVE) methodology was selected as the method to satisfy the requirements of GL 88-20, Supplement 4. The FIVE methodology was used to identify fire areas of potential risk significance, calculate fire area ignition frequencies, and provide hazards analysis for resulting critical areas. The quantification of safe shutdown system unavailability was obtained by propagating fire induced system failures through a modified PSA plant model.

The FIVE methodology included the following elements:

1) A qualitative analysis screening plant areas whose loss due to fire will have no impact on the ability to achieve and maintain safe shutdown;

2) A quantitative screening based on fire ignition frequencies and the availability of safe shutdown equipment outside the fire area;

3) A fire damage evaluation that involves a detailed assessment of the effects of a fire for those areas that did not initially screen;

4) Finally, a fire scenario evaluation and quantification is used to assess the probability of not being capable of performing a safe shutdown for a particular area.

2.3.3 Other Event Methodology Outline

The high winds, floods and other events portion of the LGS IPEEE was examined through the use of the progressive screening-type approach defined in NUREG-1407, Section 5, and in GL 88-20, Supplement 4. Results are documented herein consistent with the reporting guidelines of Appendix of NUREG-1407.

The required steps included in the screening are the following:

1) Review Plant Specific Hazard Data and Licensing Bases (FSAR)

2) Identify Significant Changes, if any, since OL Issuance

3) Does Plant/Facilities Design Meet 1975 SRP Criteria? (Quick Screening & Walkdown)

For those events not screened out at this stage, the following optional steps were considered for further examination:

4) Is the Hazard Frequency Acceptably Low?

5) Bounding Analysis (Response/Consequence)

6) PSA

The final step is:

7) Documentation (includes Identified Reportable Items and Proposed Improvements)

2.4 Information Assembly

2.4.1 Plant Layout and Containment Information

The Limerick Generating Station plant layout and reactor enclosure (containment building) information used in conducting the IPEEE are contained in the UFSAR. Plant layout information is contained in UFSAR Section 1.2 and reactor enclosure information is contained in Sections 3.2 through 3.7.

2.4.2 Documentation Used

In selecting the systems and seismic success paths to accomplish the seismic safe shutdown functions identified in Section 3.1.2 of this report, PECO Energy reviewed the Limerick Special Events Procedures along with the shutdown methods identified in the Limerick Fire Protection Evaluation Report (FPER). Technical input (e.g. P&ID's, FPER, and procedures) for developing the seismic success paths had a freeze date of September 1, 1992. Plant design basis documentation was reviewed by one or more members of the Seismic Review Team (SRT). This included the UFSAR, civil/structural drawings, specifications, mechanical system and component documents, P&IDs, piping isometrics, electrical load lists, seismic analyses, floor response spectra, seismic anchorage analyses and design, electrical one line diagrams, equipment qualification reports, and seismic qualification review team (SQRT) forms. Documentation reviewed and used by the SRT, systems engineers, fire protection engineers and other external event reviewers during the IPEEE are referenced in the applicable sections of this report where appropriate.

Technical input for the fire analysis had a freeze date of July 1994. This included the UFSAR, plant drawing schematics and calculation.

Documentation used for high winds, floods and other events had a freeze date of September 1, 1992, and included the UFSAR, SER and its supplements and various site procedures and specifications.

2.4.3 Coordination Among External Events

The SRT and fire protection engineers collaborated on evaluating the Sandia/NRC Fire Risk Scoping Study Seismic/Fire Interaction Issue. This evaluation included walkdowns to review the potential breakage of flammable liquid sources, the potential for seismic actuation of fire suppression systems, and for the survivability of any fire suppression systems which might have been in close proximity to safe shutdown path components.

3 SEISMIC ANALYSIS

In 25 years of commercial operation of nuclear power plants, the requirements for seismic design have evolved from the application of commercial building codes to major structures to very detailed analysis and testing of all safety related structures, equipment, instrumentation, controls and their associated interconnecting electrical cables, etc.

Current earthquake design practices for nuclear plants, similar to those used at Limerick Generating Stations (LGS) Units 1 and 2, result in substantial conservatism in excess of what is necessary to assure that plants can be safely shut down in the event of a postulated safe shutdown earthquake (SSE). This design philosophy is recognized to provide significant reserve margin in the seismic capability of structures, systems, and components. The reserve capacity can be attributed to the inherent conservatism included in the design basis analysis techniques and design approaches.

Limerick Generating Station (LGS) was designated a focused scope plant with a Review Level Earthquake (RLE) of 0.3g per NUREG-1407 (ref. 1.3-2) and Generic Letter 88-20 Supplement 4 (ref. 1.1-1).

In the initial response transmitted to the NRC, PECO Energy indicated that the EPRI Seismic Margin Methodology approach would be utilized to address the seismic portion of the IPEEE.

In the course of the seismic reviews and based on new seismic hazard data published by the Lawrence Livermore National Laboratories (LLNL) in NUREG-1488 (ref. 3.0-1) in October 1993, it became very evident that the seismic risk for the LGS may have been overstated.

The new hazard data indicates that the seismic risk for LGS is less than the seismic risks for most plants originally included in the "reduced scope" bin by the NRC in the generic letter supplement and NUREG-1407. Accordingly, PECO Energy informed the NRC (ref. 2.2-1) that a "reduced scope" evaluation would be performed for LGS. The amended approach uses the methodologies suggested in Generic Letter 88-20, Supplement 4 and NUREG-1407.

Although a "reduced scope" IPEEE assessment is performed at the plant design basis seismic input level (Safe Shutdown Earthquake, SSE), the seismic capability screening walkdowns consider the guidance of EPRI NP-6041-SL. Accordingly, components were preliminary screened to a minimum Review Level Earthquake (RLE) of 0.3g pga. Items that did

not pass the preliminary 0.3 pga screening were screened further to the actual LGS SSE.

Therefore for a reduced scope evaluation, the following guidelines apply:

(1) The RLE is the applicable SSE ground response spectra.
(2) The post walkdown evaluation will not develop HCLPF capacities for components and structures.
(3) Soil failure evaluation is not required.
(4) Relay Evaluation - No action is required.
(5) Containment Performance evaluation only requires retention of the walkdown of the containment systems to prevent early failures.

3.0 Methodology Selection

The approach used by PECO Energy for LGS in selecting the appropriate review method is consistent with the generic insights used in the EPRI seismic margin methodology development as discussed in EPRI NP-6041-SL (ref. 1.3-1) and EPRI NP-7498 (ref. 3.0-2) based on judgment concerning the seismic capability of its plant and the well founded conclusions derived from earthquake engineering experience.

The seismic evaluations at LGS were performed in accordance with the EPRI Seismic Margins Analysis Methodology. This methodology examines the capability of the plant in terms of seismic ruggedness of components in a minimal set of equipment required to safely shut down the plant. A detailed description of the methodology is provided in EPRI NP-6041-SL, Revision 1 (ref. 1.3-1). Details of the examinations conducted for LGS are summarized in this report.

3.1 EPRI Seismic Margins Method

The EPRI Seismic Margin Assessment (SMA) consists of the following eight steps:

(1) Selection of the seismic margin earthquake (SME)
(2) Selection of the Assessment Team
(3) Preparatory work prior to Walkdowns
(4) Systems and Element Selection

(5) Seismic Capability Walkdown
(6) Subsequent Walkdowns (as-needed)
(7) Seismic Margin Assessment (SMA) Work
(8) Documentation

Step 1 involves the specification of the earthquake for which the seismic margin assessment is to be conducted. For LGS, the RLE is the applicable SSE ground response spectra.

The seismic margin assessment team consisted of a combination of PECO Energy and VECTRA Technologies Inc. systems and seismic capability engineers. The make-up and qualification of key individuals of the team is discussed in detail in sections 3.1.2 and 3.1.4.

Preparatory work steps included detailed reviews of the original plant seismic design bases, acquaintance with the design documentation, performance of preliminary walkdowns, and data gathering. The systems selection is summarized in section 3.1.2. The Walkdowns and associated evaluations are discussed in sections 3.1.3 and 3.1.4.

3.1.1 Review of Plant Information

This section will provide a brief summary of plant information and original seismic design criteria that were reviewed in preparation for performing the screening and walkdowns of structures, components, and systems. In addition, documents which were utilized in the performance of the seismic capability work that are of a general nature, i.e., specification for design of anchor bolts, specifications for the design of structural steel, etc., are listed as references 3.1-12 through 3.1-22. Details on the screening and walkdown are provided in section 3.1.4. LGS is a modern vintage plant which employed very rigorous seismic design and construction methods.

3.1.1.1 General Plant Description

Each of the LGS units employs a GE BWR-4 designed to operate at rated core thermal powers of 3293 MWt (Unit 1) and 3458 MWt (Unit 2) (100% steam flow) with a corresponding gross electrical output of 1092 MWe for Unit 1 and 1163 MWe for Unit 2. Approximately 37 MWe are used for auxiliary power, resulting in a net electrical output of 1055 MWe (Unit 1) and 1126 MWe (Unit 2).

The LGS Units have Mark II containments. The units are similar in design and layout.

Limerick Generating Stations is located on the east bank of the Schuylkill River in the Limerick Township of Montgomery County, Pennsylvania, approximately 4 river miles downriver from Pottstown, 35 river miles upriver from Philadelphia, and 49 river miles above the confluence of the Schuylkill with the Delaware River. The site contains 595 acres (423 acres in Montgomery County and 172 in Chester County).

The site is located in the gently rolling countryside, traversed by numerous valleys containing small creeks or streams that empty into the Schuylkill River. Two parallel streams, Possum Hollow Run and Brooke Evans Creek cut through the site in wooded valleys, running southwest into the Schuylkill River.

The area surrounding the site can be generally classified as rural and open. A large portion of the land is used for agricultural purposes with the remainder of the area being either vacant or woodland with scattered commercial and residential development taking place.

The main access to the plant is from U.S. Highway 422 which runs east and west about one mile north of the site. Access to the site and all activities thereon are under the control of PECO Energy.

The site is situated in the Triassic Lowland section of the Piedmont Physiographic Province. This section is characterized by a gently rolling land surface on an eroded low plateau.

The rocks in the region surrounding the site include Precambrian and Lower Paleozoic crystalline rocks and folded sedimentary strata, and essentially unfolded Triassic sedimentary rocks and igneous intrusions. The Triassic rocks belong to the Newark Group, which is divided into the basal Stockton Formation and the Brunswick, Lockatong, and Hammer Creek Lithofacies.

Bedrock at the site underlies a thin cover of residual soil. The Brunswick red siltstone, sandstone and shale is the predominant bedrock formation. Gray shale and argillite of the Lockatong Lithofacies, light gray sandstones and conglomerates of the Hammer Creek Lithofacies, and intruded diabase and associated hornfels are also found in the area. The strata exhibit gentle homoclinal dips to the north and northwest. The thickness of the Newark Group overlying the Paleozoic and Precambrian basement rocks at the site is on the order of 8000 feet.

The dominant structural feature of the region is the Regional Appalachian Orogenic Belt. This belt is marked by the northeast-southwest orientation of the axes and lineation of most of the structural features and stratigraphic

contacts.

3.1.1.2 Plant Seismic Design Basis

3.1.1.2.1 Ground Response Spectra

The site design response spectra used are illustrated in Figures 3.1.1-1 and 3.1.1-2. These spectra are for the horizontal components of the Operating Basis Earthquake (OBE) and the Safe Shutdown Earthquake (SSE) respectively. The response spectra for the SSE are normalized to a maximum ground acceleration of 15% of gravity. The values for the vertical component of the design response spectra are 2/3 of the horizontal design response spectra described above. The response spectra are based on data developed from records of previous earthquake activity and represent an envelope of motion expected at a sound rock site from a nearby earthquake.

Regulatory Guide 1.60 (ref. 3.1-1) (December, 1973) "Design Response Spectra for Seismic Design of Nuclear Power Plants" was not used for development of the spectra since both LGS units were docketed for construction permit review in March 1970, and the spectra were finalized in 1973. Further, a letter dated December 21, 1973 from J.M. Hendrie (NRC) to R.M. Collins (Bechtel) states that Regulatory Guide 1.60 is applicable only to the plants docketed for construction permit review after April 1, 1973.

The site-dependent analysis has developed the seismic response spectra from site-related information. This approach, used in lieu of the response spectra specification RG 1.60, was found to be acceptable to the NRC as stated in the SER (ref. 5.1-7). In addition, the NRC noted that the 0.15g Newmark SSE spectrum, which is less than but close to the 84^{th} percentile, is an adequate representation of the ground motion for the site.

Therefore the use of site-dependent analysis and damping values which were typically lower than RG 1.61 ensures that the seismic design parameters of Category I structures, systems and components are defined adequately to form a conservative basis for the design of these structures, systems, and components to withstand seismic loadings.

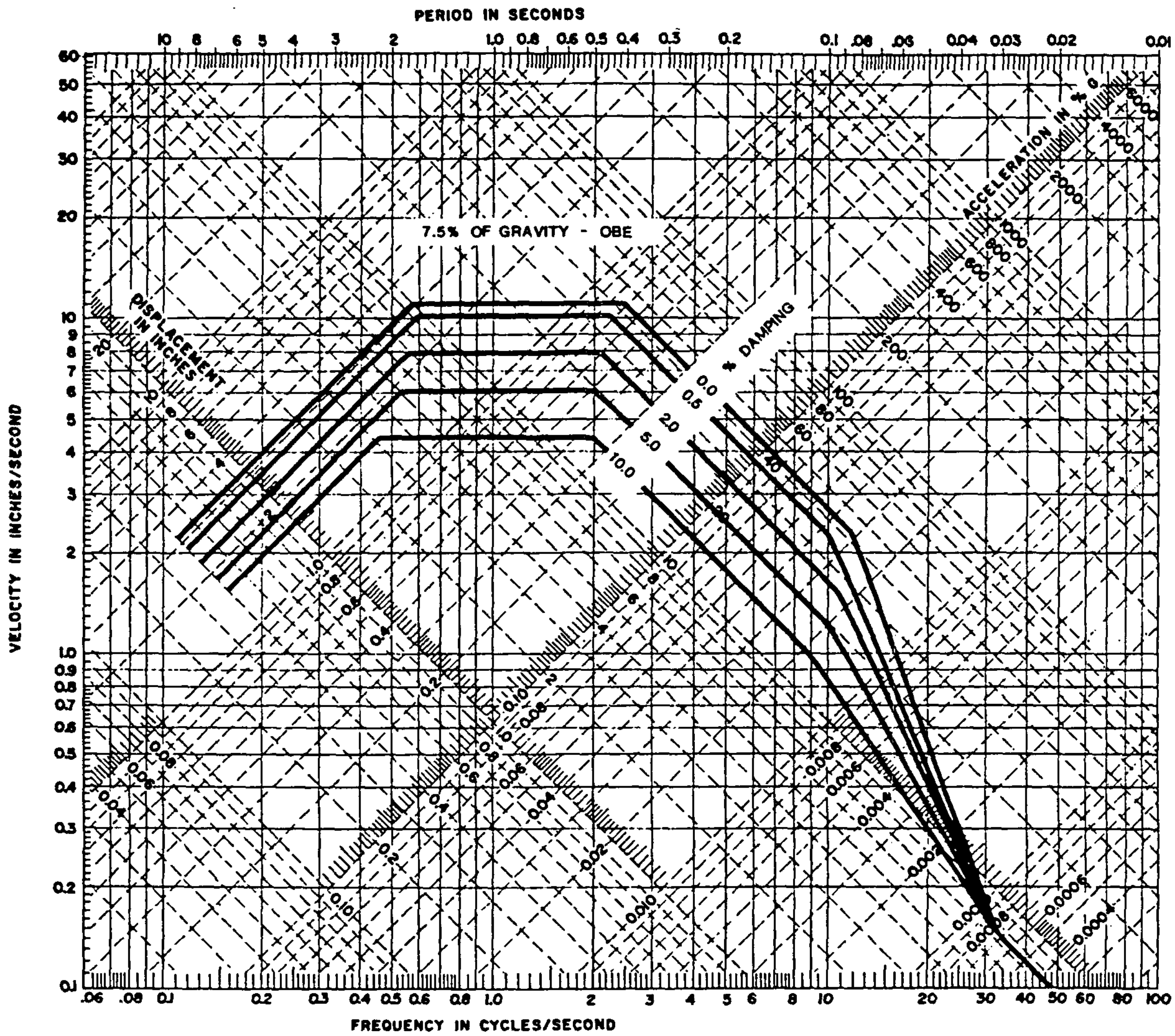

FIGURE 3.1.1-1

DESIGN RESPONSE SPECTRA FOR OPERATING BASIS EARTHQUAKE (HORIZONTAL COMPONENT)

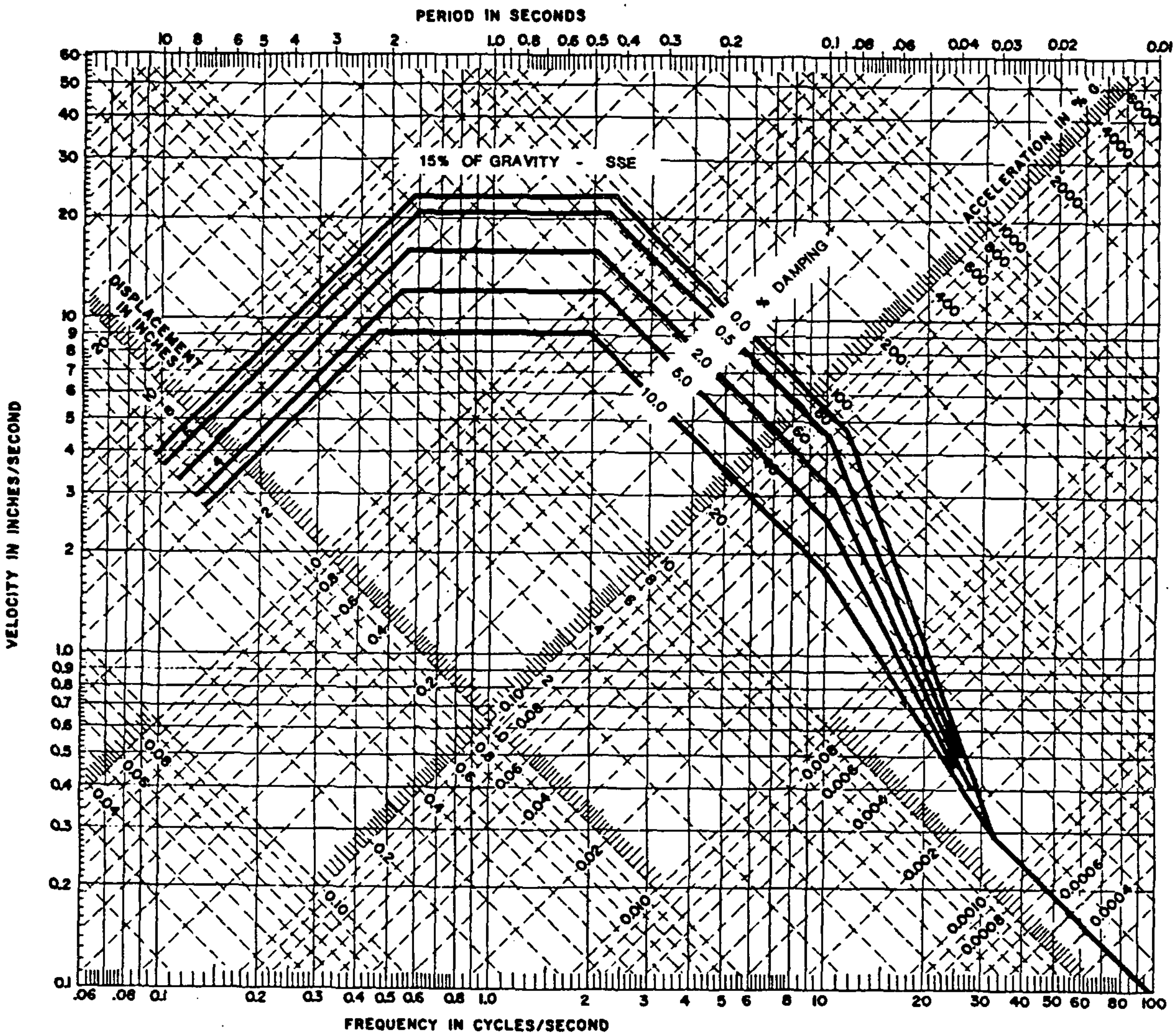

FIGURE 3.1.1-2

DESIGN RESPONSE SPECTRA FOR SAFE SHUTDOWN EARTHQUAKE (HORIZONTAL COMPONENT)

3.1.1.2.2 Generation of In-Structure Response Spectra

A time history analysis of the Seismic Category I structures was performed to generate the response spectra at the various mass points of the model. This section presents a description of the time histories, structural damping values, dynamic models, and soil structure interaction considerations used in the time history analysis. In addition, a description of the methodology used to construct the node point/floor response spectra is presented.

3.1.1.2.2.1 Synthetic Time Histories

A synthetic time history of motion was generated by modifying the actual records of the 1952 Taft earthquake according to the techniques proposed in ref. 3.1.2. This synthetic time history was then further modified to develop the time history which corresponds to the design response spectra. The duration of the time history is 15 seconds. A comparison of the time history response spectra and the design response spectra for 1%, 2%, 3%, 5% and 7% damping values was performed. The spectra are computed at the frequency values as given in Table 5-1 of BC-TOP-4A, "Seismic Analysis of Structures and Equipment for Nuclear Power Plants" (ref. 3.1-3).

3.1.1.2.2.2 Critical Damping Values (NSSS)

The damping values indicated in Table 3.1.1-1 are used in the response analysis of various structures and systems, and in preparation of floor response spectra used as forcing inputs for piping and equipment analysis or testing. It can be seen that the values given in Table 3.1.1-1 are somewhat less than those given in Regulatory Guide 1.61 (October 1973). The calculated responses are, therefore, conservative.

3.1.1.2.2.3 Critical Damping Values (Non-NSSS)

Critical damping values expressed as a percentage of critical damping and used for seismic Category I structures, equipment and piping for both OBE and SSE are given in Table 3.1.1-2.

All the values shown in Table 3.1.1-2 are equivalent to or more conservative han those in Regulatory Guide 1.61 with the exception of the SSE value for welded steel structures. The damping value of 5% (PSAR Table C.2.1) is based on information given in ref. 3.1.1. The 5% value has been used, with appropriate design margins, because the stress levels for SSE conditions are allowed to approach the yield point.

TABLE 3.1.1-1

CRITICAL DAMPING VALUES FOR NSSS MATERIALS (2) (3)
FROM LGS UFSAR SECTION 3.1

Structure or Component	Operating Basis[1] Earthquake	Safe Shutdown Earthquake
Reinforced Concrete Structures	2.0	5.0
Welded Structural Assemblies	1.0	2.0
Bolted or Riveted Structural Assemblies	2.0	3.0
Vital Piping Systems	0.5	1.0
Drywell (Coupled)	2.0	5.0
PRV Support Skirt, Shroud Head, Separator,	2.0	2.0
CRD Housings	3.5	3.5
Fuel	7.0	7.0
Steel Frame Structures	2.0	3.0

(1) Values expressed as percent of critical damping.
(2) Other values may be used if they are indicated to be reliable by experiment or study.
(3) Alternative critical damping values for piping systems may be used as described in the UFSAR.

TABLE 3.1.1-2

CRITICAL DAMPING VALUES FOR NON-NSSS MATERIALS (2)
FROM LGS UFSAR SECTION 3.2

Structure or Component	Operating Basis[1] Earthquake	Safe Shutdown Earthquake
Equipment	0.5	1
Piping Systems	0.5	1
Welded Steel Structures	2	5
Bolted Steel Structures	3	7
Reinforced Concrete Structures	2	5
Primary Containment	3	7

(1) Values expressed as percent of critical damping.

(2) Alternative critical damping values for piping systems may be used as described in the UFSAR.

3.1.1.2.3 Equipment and Piping Systems Damping

The existing LGS design basis generally uses conservative damping values in the analysis and design of equipment and piping systems. Typical design basis values for percent of critical damping with regard to equipment and large diameter piping is 1 percent for SSE.

3.1.1.2.4 Dynamic Models of Seismic Category I Structures

In the analysis of seismic Category I structures, two distinct objectives must be satisfied:

- Development of in-structure seismic response characteristics, where necessary, for use in the analysis and design of seismic Category I systems, equipment and components.

- Determination of seismic force distribution within the various structures resulting from the design criteria free-field seismic input, for use in the design of seismic Category I structures.

Two analytical procedures were employed to determine the seismic responses to the Category I structures. In general, a modal response spectrum analysis was used to compute the in-structure seismic responses, including nodal accelerations, nodal displacements and member forces. Alternatively, a time history analysis procedure was used to generate the in-structure seismic responses discussed above. In addition, time history analysis was used to generate all floor response spectra. The mathematical idealization of the structural characteristics of the various seismic Category I structures was accomplished by a lumped parameter beam-stick model. The general analytical methods and modeling techniques used in these analyses are in accordance with BC-TOP-4A. The seismic design criteria is defined in terms of the OBE and SSE design response spectra, the synthetic time history and the soil structure interaction parameters used for development of floor response spectra for equipment assessment.

3.1.1.2.5 Soil Structure Interaction

Since the seismic Category I structures are founded on competent bedrock, a soil spring approach to characterize soil structure interaction is not used in the dynamic analysis. A simplified lumped mass method using a fixed base model is used. However, for a more refined analysis of the containment and reactor enclosure, the underlying foundation medium is considered to interact with the structure. The equivalent soil spring constant and damping coefficient are computed in accordance with the

formulae of Table 3-2 of BC-TOP-4A and the analysis carried out by the methods discussed in Appendix D of BC-TOP-4A (ref 3.1-3). The resulting structure-foundation interaction coefficients are listed in Table 3.1.1-3. This analysis approach is conservative for a rock founded structure since the generated spectra of the base mat slab will typically be higher than the free-field input spectra.

3.1.1.2.6 Development of Floor Response Spectra

3.1.1.2.6.1 <u>Floor Response Spectra (NSSS)</u>

The simultaneous use of three components of earthquake motion was not a design basis requirement of the construction permit for the LGS plants, however, the NSSS systems and components are evaluated to the requirements of Regulatory Guide 1.92 (ref. 3.1-4).

Response Spectrum Method

Response spectra are developed considering three components of earthquake motion. The individual responses in each orthogonal direction are combined by SRSS of the colinear contribution due to three directions of earthquake motion. These are used to predict the total response at each frequency.

Time History Method

When the time history method of analysis is used, one of the following options is used to obtain peak value of any particular response of interest:

- When maximum colinear contributions due to the three directions of earthquake motion are calculated separately, the total response is obtained as the SRSS combination of the colinear values.

- When colinear time history responses from each of the three components of the earthquake motion are calculated individually by the step-by-step method and then combined algebraically at each time step, the maximum response is obtained as the peak value from the combined time solution.

- When a response at each time step is calculated directly based on the simultaneous application of the three earthquake components, the maximum response is determined by scanning the combined time history solution.

TABLE 3.1.1-3

STRUCTURE-FOUNDATION INTERACTION COEFFICIENTS

STRUCTURE	MOTION	EQUIVALENT SPRING CONSTANT	EQUIVALENT DAMPING COEFFICIENT
Primary Containment	Translational	4.15×10^{7} k/ft	2.01×10^{5} k-sec/ft
	Rocking	8.12×10^{10} k-ft/rad	7.82×10^{7} k-ft-sec/rad
	Vertical	4.87×10^{7} k/ft	3.48×10^{5} k-sec/ft
Reactor Enclosure and Control Structure	Translational: E-W	8.17×10^{7} k/ft	8.98×10^{5} k-sec/ft
	N-S	8.55×10^{7} k/ft	8.79×10^{5} k-sec/ft
	Rocking: E-W	2.04×10^{12} k-ft/rad	9.23×10^{9} k-ft-sec/rad
	N-S	6.22×10^{11} k-ft/rad	2.63×10^{9} k-ft-sec/rad
	Vertical	9.90×10^{7} k/ft	1.74×10^{6} k-sec/ft

The components of earthquake motion must be statistically independent for Options 2 and 3 above. Also, the time history method precludes the need to consider closely spaced modes.

3.1.1.2.6.2 **Floor Response Spectra (Non-NSSS)**

The time history method of analysis was used to develop the floor response spectra. A discussion of the technique of finding the nodal time history and then producing the spectrum may be found in Sections 4.2 and 5.2 of BC-TOP-4A.

The LGS nuclear power plant structure, systems and components important to safety were designed to withstand the effects of a safe shutdown earthquake (SSE) without the loss of capability to perform their safety function and are designated seismic Category I. The structures, systems and components are identified in Table 3.2-1 of the UFSAR and are described below.

3.1.1.3 Safety Related Category I Structures

3.1.1.3.1 Shear Walls, Footings and Shield Walls

These types of civil structures within the powerblock at LGS are designed for a minimum SSE of 0.15g. Category I structures outside the powerblock are designed for an SSE of 0.15g.

3.1.1.3.2 Category I Concrete and Steel Frame Structures

The seismic Category I structures that are considered in the IPEEE assessment of the Limerick Generating Station include the Primary Containment and Internal Structures, the Secondary Containment (Reactor Enclosure and refueling area), the Control structure, the Diesel Generator enclosure, the Spray Pond Pump structure, the Spray Pond and several miscellaneous structures detailed later in this section. These structures house or support Category I equipment and, therefore, maintaining their structural integrity is considered essential for the ability to safely shut down the plant in the event of a design basis earthquake or review level earthquake.

3.1.1.3.2.1 Primary Containment

The primary containment is divided by a horizontal diaphragm slab into two major regions: the drywell and the suppression chamber. The drywell encloses the reactor vessel, reactor recirculation system and associated piping and valves. The suppression chamber stores a large volume of water.

The primary containment is in the form of a truncated cone over a cylindrical section, with the drywell being the upper conical section and the suppression chamber being the lower cylindrical section. These two sections comprise a structurally integral, reinforced concrete pressure vessel, lined with welded steel plates and provided with a steel domed head for closure at the top of the drywell. The diaphragm slab is a reinforced concrete slab structurally connected to the containment wall.

The primary containment is structurally separated from the surrounding reactor enclosure.

Section 3.8.1.3 of the UFSAR describes the loading combinations used for the design and analysis of the containment. The containment is also analyzed and designed for hydrodynamic loads resulting from MSRV discharge and LOCA phenomena. These loads are combined with the OBE and SSE.

The design of the containment internal structures (i.e., diaphragm slab, reactor pedestal, reactor shield wall, suppression chamber columns, drywell platforms, seismic truss and reactor vessel stabilizer) considers the effects of all appropriate loading conditions with the major/significant load contributions coming from the design basis accident pressure, accident temperature gradients, or missile/pipe rupture loadings. The dynamic effects of seismic loads are appropriately addressed and included in the containment internal structures design, however, the contributions from such loadings are typically minimal and overshadowed by the more significant contributions from the severe accident loadings mentioned above.

Section 3.1.5 of this report describes the containment performance review performed as part of the IPEEE EPRI SMA.

3.1.1.3.2.2 Secondary Containment

The reactor enclosure surrounds the primary containment in each unit and, with the refueling area, provides secondary containment. The secondary containment houses the auxiliary systems of the NSSS, the spent fuel pool, the refueling facility and equipment essential to the safe shutdown of the reactor. The secondary containment is structurally integral with the control structure described below.

The secondary containment, up to and including the roof slab, is of reinforced concrete construction. Exterior bearing walls are reinforced concrete and are additionally designed as shear walls to resist lateral loads. The floors and roof are constructed of reinforced concrete, supported by steel beams and column framing systems. The concrete slabs are designed as diaphragms to transmit lateral loads to the shear walls. The structural steel beams and girders are supported by either structural steel columns or reinforced concrete bearing walls. The steel columns are supported by base plates attached to the foundation. The reinforced concrete walls and floors meet structural as well as radiation shielding requirements. At certain locations, concrete block masonry walls are used to provide better access for erecting and installing equipment. The block walls also meet the structural and the radiation shielding requirements.

The refueling facility is located above the reactor enclosures. It consists of the spent fuel pool, the steam dryer and separator storage pool, the reactor well, the cask loading pit, the skimmer surge tank vaults, a 48' long refueling platform crane and a 129' long reactor enclosure crane. The facility is supported by end bearing walls and by two post-tensioned concrete girders with grouted tendons. The girders run east-west and span over the primary containments without intermediate supports. Each girder spans approximately 162' and is 6' wide. The depth is 46' at the supports and is reduced to 26' at midspan, where the girders straddle the containments. The ends of the girders are supported by concrete pilasters. A gap between the bottom of the girders and the top of the containments ensures that loads from the refueling facility are not transferred to the containment. The walls and slabs of the spent fuel pool, the cask loading pit, the reactor cavity, and the steam dryer and separator storage pool are lined on the inside with a stainless steel liner plate. The refueling facility meets the radiation shielding requirements.

The reactor enclosure crane consists of a main and an auxiliary hoist, with capacities of 125 tons and 15 tons respectively. The crane is used during maintenance and refueling operations. It spans approximately 129' and is 28' above the refueling floor. The crane is mounted on two 175 lb. rails,

supported by a pair of runway girders. The runway girders are supported by a series of built-up columns spaced at 27' centers, which in turn are supported by bearing walls.

The reactor enclosure is separated from the primary containment by a gap filled with compressible material. A gap is also provided at the interface of the secondary containment with the diesel generator, radwaste and turbine enclosures.

3.1.1.3.2.3 <u>Control Structure</u>

The control structure is a reinforced concrete enclosure structurally integrated with the secondary containment. The bearing walls are of reinforced concrete and are additionally designed as shear walls to resist lateral loads. The floors and roofs are constructed of reinforced concrete supported by steel beams and are designed as diaphragms to transmit lateral loads to the shear walls. The beam spans in the north-south direction and are supported at the ends by the bearing walls. The reinforced concrete walls and floors meet structural as well as radiation shielding requirements. At certain locations, concrete block masonry walls are used to provide better access for erection and installation of equipment. The block walls also meet the structural and radiation shielding requirements. The control structure is separated from the turbine enclosure by a seismic gap.

3.1.1.3.2.4 <u>Diesel Generator Enclosure</u>

The diesel generator enclosures house the standby diesel generators, which are essential for safe shutdown of the plant.
Concrete walls, each 2' thick, separate each diesel enclosure into four cells, one for each of the four diesel generators provided per unit. Each diesel generator unit is enclosed in its own concrete missile-protected cell. The diesel generator enclosure is a reinforced concrete structure on wall foundations. The bearing walls are of reinforced concrete and are additionally designed as shear walls to resist lateral loads. The floors and roof are constructed of reinforced concrete supported by steel beams and are designed as diaphragms to transmit lateral loads to the shear walls. The north side of the enclosure bears on the pipe tunnel beneath. At certain locations, concrete masonry walls are used to provide better access for erection and installation of equipment. The diesel generators are supported by the floors.

3.1.1.3.2.5 Spray Pond Pump Structure

The spray pond pump structure contains the ESW and RHRSW pumps, auxiliary equipment and related piping and valves.

The spray pond pump structure is a two story reinforced concrete structure. The bearing walls are of reinforced concrete and are additionally designed as shear walls to resist lateral loads. The operating floor and roof are constructed of reinforced concrete supported by steel beams and are designed as diaphragms to transmit lateral loads to the shear walls. A mezzanine floor composed of grating over steel beams is provided to support the heating and ventilating equipment. An intermediate floor in the wing areas is provided to support valves and piping.

3.1.1.3.2.6 Spray Pond

The spray pond serves as the ultimate heat sink for the plant and is designed so that normal operating water is retained in excavation only, that is, not by constructed embankments.

An emergency spillway is provided at the north side of the pond. The only anticipated use of this spillway is either during a malfunction of the blowdown line or during postulated conditions of heavy rainfall. The emergency spillway is designed to ensure that the maximum water level does not adversely affect the spray pond system, and to direct run-off water away from safety related facilities in a controlled manner. The roadway surrounding the remainder of the spray pond provides a minimum freeboard of 4'.

The spray network piping, which is located above the water, is supported by reinforced concrete columns. The columns are founded on bedrock or on concrete fill on top of bedrock.

3.1.1.3.2.7 Miscellaneous Structures

Subgrade pits, manholes and tunnels which contain safety related components are constructed of reinforced concrete.

Safety related piping, tanks and electrical ducts which are not located inside structures, are buried underground with adequate cover for missile protection. Additionally, soil erosion due to failure of non-seismic piping has also been considered. The integrity of safety related seismic Category I buried pipe will not be impaired through soil erosion by a failure of one buried non-seismic Category I pipe. This conclusion was based on the

following conditions:

- All but 170' of common trench has been constructed in rock, where erosion of the supporting medium would be insignificant.

- The 170 feet of trench not constructed in rock is Type 1 fill.

The non-pressure pipes (gravity lines), such as the blowdown line and waste and storm lines, do not pose a significant erosion problem. The non-seismic Category I pressure lines consist of a 36" Schuylkill River makeup water line and a 12" fire line.

Failure of the non-seismic Category I pressure pipe may create progressive erosion in the Type 1 fill. It is anticipated that water under pressure would penetrate to the surface, creating a progressively enlarging crater. However, because the water will flow in the direction of least resistance, once the water penetrates to the surface, the crater will be enlarged at a relatively slow pace. The maximum acceptable unsupported length of safety related pipe in the trench is conservatively estimated to be in excess of 30', based on the maximum allowable spans given in the ASME code. A considerably long time would be required to exceed this span capacity.

- Instrumentation would give indication in the control room if a break occurred in the non-seismic Category I pressure pipe. Loss of flow from the makeup water line to the cooling tower would result in an alarm in the control room when low level is reached in the cooling tower basin. It is conservatively estimated that low level would be reached within 30 minutes.

  Low pressure in the 12" fire line, following a break, starts a fire pump that gives an alarm in the control room without a fire signal.

  Following an SSE, if either alarm described above is activated, personnel will investigate for evidence of a faulty condition in the pipelines described above and will initiate any necessary corrective action.

- The procedures for operator action to a seismic event include the requirement that, within two hours after an SSE, personnel will investigate for evidence of a faulty condition in the pipelines described above and will initiate any necessary corrective action.

3.1.1.4 Non-Safety Related Structures

3.1.1.4.1 Non-Safety Related Seismic Category I Structures

The non-safety related seismic Category I structures at LGS consists of the radwaste enclosure and the offgas portion of the radwaste enclosure.

The radwaste enclosure is designed in accordance with seismic Category I criteria even though its integrity is not required to protect the reactor coolant pressure boundary or to ensure the capability to safely shut down the reactor. In addition, its failure would not result in potential offsite exposures comparable to the guideline exposures of 10CFR100. The radwaste enclosure houses systems for receiving, processing and temporarily storing the radioactive waste products generated during the operation of the plant.

The radwaste enclosure, which includes the offgas enclosure, is a reinforced concrete structure. The bearing walls are of reinforced concrete and are additionally designed as shear walls to resist lateral loads. The exterior walls are waterproofed and are designed for hydrostatic effects as necessary. The floors and roof in the main portion of the radwaste enclosure are constructed of reinforced concrete supported by beam and column framing systems and are designed as diaphragms to resist lateral loads. The columns are supported by base plates on the foundation. The floors and roof of the offgas portion of the radwaste enclosure are of reinforced concrete supported by steel beams and bearing walls. The reinforced concrete walls and floors meet structural as well as radiation shielding requirements. At certain locations, concrete block masonry walls are used to provide better access for erection and installation of equipment. The block walls also meet the structural and radiation shielding requirements.

The radwaste enclosure is separated from the turbine enclosure and reactor enclosure by seismic gaps.

3.1.1.4.2 Non-Safety Related Non-Seismic Category I Structures

The turbine enclosure is the only non-safety related non-seismic Category I structure in close proximity to seismic Category I structures. The remaining non-Category I structures were designed for seismic loads according to the Uniform Building Code (UBC). The non-Category I structures were analytically checked to ensure that they will not collapse on, or otherwise impair the integrity of, adjacent seismic Category I structures when subjected to the design loads.

The turbine enclosure is divided into two units separated by an expansion

joint. It houses two inline turbine generator units and auxiliary equipment including condensers, condensate pumps, moisture separators, air ejectors, feedwater heaters, reactor feed pumps, MG sets for reactor recirculation pumps, interconnecting piping and valves, switchgear and heating and ventilating equipment.

Two 110 ton overhead cranes are provided above the operating floor for servicing both turbine generator units. Two reinforced concrete tunnels, one for each unit, are provided for the offgas pipelines at the foundation level, running from the area around the control structure to the radwaste enclosure.

The turbine enclosure rests on a reinforced concrete mat foundation. The superstructure is framed with structural steel and reinforced concrete. Rigid steel frames support the two turbine enclosure cranes and resist all transverse (north-south) lateral loads. Steel bracings resist longitudinal (east-west) lateral loads above the operating floor. Below this level, reinforced concrete shear walls transfer all lateral loads to the foundations.

Seismic separation gaps are provided at the interface of the turbine enclosure with the reactor, control and radwaste enclosures.

The floors of the turbine enclosure are of reinforced concrete supported by structural steel beams and are designed as diaphragms for lateral load transfer to the shear walls. The roof is built up roofing on metal decking.

Exterior walls are covered by non-structural precast reinforced concrete panels.

Interior walls, required for radiation shielding or fire protection, are constructed of reinforced concrete block. These walls are not used as elements of the load resistant system.

The seismic Category II turbine enclosure may undergo some plastic deformation under seismic loading resulting from an SSE, but the plastic deformation is limited to a ductility factor of 2. The portions of the turbine enclosure which support the main steam lines are designed so that the main steam lines and their supports maintain their integrity under the seismic loading resulting from an SSE.

The turbine generator units are supported on free-standing reinforced concrete pedestals. The mat foundations for the pedestals are founded on rock at the same levels as the basemat for the turbine enclosure. Separation joints are provided between the pedestals and the turbine

enclosure floors and walls, to prevent transfer of vibration to the enclosure. The operating floor of the turbine enclosure is supported on vibration damping pads at the top edge of the pedestal.

3.1.1.5 Mechanical/Electrical NSSS and BOP Components

Seismic qualification of equipment is performed by analysis, dynamic testing, or a combination of analysis and dynamic testing. Seismic qualification of equipment by analysis is utilized when the equipment can be adequately represented by a model and the analysis can determine its structural and functional adequacy. The analysis is provided by either an equivalent static analysis or a dynamic analysis. The equivalent static analysis method can be used when the natural frequency of the equipment is not determined. Seismic qualification of equipment by dynamic testing is provided when qualification by analysis is insufficient to determine either the structural and/or functional adequacy. Typical testing methods include single frequency single axis tests, single frequency dwell tests, and multifrequency tests.

The seismic qualification and coumentation procedures used for Category I mechanical and electrical equipment and equipment supports meets as a minimum the requirements of IEEE 344-1971 (ref. 3.1-24) which is the plant commitment, and as such does not demonstrate compliance with Regulatory Guide 1.100. However, the dynamic qualification requirements committed for LGS ensure an acceptable basis for qualifying the equipment.

The NRC performed an audit in 1984 of selected equipment items to develop the basis for the staff judgement on the completeness and adequacy of the implementation of the entire seismic and dynamic qualification program. The results of the NRC audit by the Seismic Qualification Review Team (SQURT) is given in reference 5.1-7. [Limerick SSER3]. The purpose of the audit was to determine the extent to which the qualification of equipment meets the current (1984) licensing criteria as described in RG 1.100, RG 1.92, SRP Section 3.10 and IEEE 344-1975 standards. Based on the audit the NRC staff concluded that the applicant's equipment the applicant's equipment seismic and dynamic qualification program was satisfactorily defined and implemented according to the intent of the current staff licensing criteria.

Sections 3.9.2 and 3.10.2 of the UFSAR defines the methods and procedures for dynamic testing and analysis of mechanical and electrical equipment.

NSSS supplied Class 1E equipment having primarily an active electrical

safety function was tested in compliance with IEEE 344-1971. In regard to compliance with RG 1.100, the analysis and testing for the seismic qualification of non-NSSS Class 1E instruments and electrical equipment required, to function during and after an SSE are in compliance with IEEE 344-1971 for components purchased before issuance of IEEE 344-1975 and are in compliance with IEEE 344-1975 for components purchased after its issuance. The seismic qualification of equipment is documented in qualification reports.

Mechanical equipment were typically qualified by analyses using ASME stress limits or stresses within the elastic range for nonpressure retaining components. Electrical equipment was typically qualified by testing or a combination of test and analysis.

3.1.1.6 Distribution Systems

Seismic Category I distribution systems at LGS include piping, electrical raceways, electrical conduit, and HVAC ductwork. Varying methods of seismic qualification are provided for these systems in addition to their supports.

Seismic qualification of Category I piping systems is provided by dynamic analysis. The response spectrum method is the primary analytical technique used in the analysis.

Supports for the Category I piping systems are analyzed and designed to withstand the resulting pipe loadings from the piping analysis. Normal/upset, emergency, and faulted conditions are appropriately considered in accordance with applicable ASME code sections.

Electrical raceways systems, that is, cable tray, conduit and wireway gutter systems, are seismically qualified by either the capacity evaluation method or the static analysis method. The capacity evaluation methodology considers a comparison of seismic loading, based upon the raceway frequency, support frequency, and design spectra, to the raceway allowable capacity. Maximum raceway loadings calculated by the equivalent static load method are evaluated against maximum raceway capacity determined by testing or analysis for the static analysis method.

Raceway supports were typically analyzed and designed in groupings of generic support configurations for loadings based on the maximum allowed raceway span. The majority of the support configurations are braced trapezes, direct attachment, or simple beams. Damping of 10% of the critical is used for the design of cable tray support systems, 7% damping

is used for conduit and wireway gutter trapeze type support systems. The recommended damping values for cable trays and conduit systems were developed based on a test program as described in the UFSAR.

Seismic qualification of electrical conduits is accomplished by evaluating the conduit structural capacity based on an allowable span approach. To ensure compliance with the conduit seismic qualification, conduit supports are provided along the conduit at locations that do not exceed the calculated allowable spans. Conduit supports are analyzed by the response spectrum method or by the equivalent static load method.

HVAC ductwork is seismically qualified by dynamic as well as static analysis. The equivalent static load method is typically utilized with the SSE damping value considered as 7 percent. The effect of seismic stress on the ductwork is usually very low compared to other design parameters such as vacuum or internal pressure. HVAC ductwork supports are typically analyzed by the response spectrum method.

3.1.1.7 Seismic Spatial Systems Interaction

3.1.1.7.1 Proximity Effects

The Turbine Enclosure, auxiliary boiler enclosure, and the administration building are the only major non-Category I structures which are adjacent to seismic Category I structures. These non-Category I structures are designed for seismic loading in accordance with the UBC. In addition, the Turbine Enclosure was dynamically analyzed to ensure the capacity to withstand a SSE without collapsing on or impairing the integrity of the adjacent reactor and control structures. Similarly, the other non-Category I structures were analytically evaluated to ensure that they will not collapse on or otherwise impair the integrity of adjacent seismic Category I structures when subjected to the design loads.

Structural separations have been provided to ensure that interaction between Category I and non-Category I structures does not occur. The minimum separation gap between the buildings is twice the relative displacement for the design seismic loads.

Seismic Category I design requirements were extended to the first seismic restraint beyond the defined boundaries.

3.1.1.7.2 II/I Criteria

Components and their supporting structures that are not seismic Category

I, but are located in the vicinity of seismic Category I items, are identified as seismic Category IIA. These components are either designed to seismic Category I criteria or are reviewed to identify the equipment whose failure could result in a loss of function capability for seismic Category I structures, equipment or systems that are required for an SSE. Components identified by this review are considered safety impact items and are either analytically checked to confirm their integrity against collapse when subjected to seismic loading from the SSE or are separated from seismic Category I equipment by a barrier.

The Safety Impact Program (ref. 3.1-16) for LGS describes the approach taken to ensure that the interface between Category I and non-Category I structures and plant equipment has been adequately addressed.

Therefore, LGS can be considered as being constructed with a significant awareness of and concern for II/I issues. Programs developed and implemented since commercial operation of LGS provide adequate assurance of continued compliance with the requirements of the safety impact program and the subsequent resolution of II/I concerns.

The fire piping at Limerick Generating Station uses a dry preaction system. The main headers up to the deluge valves were found to be well supported. The dry piping from the deluge valve to the various sprinkler heads were typically rod hung and used threaded couplings. A review of various control panels associated the fire protection system shows that these are small wall mounted units that are typically rigid. The relays that are used in the Pyrotronics controls are the AROMAT Socket type relays. These have recently been tested by EPRI (ref. 3.1-23) in EPRI NP-7147-SL Volume 2, Seismic Ruggedness of Relays mounted on representative printed circuit boards that are part of the control circuit. The reported GERS level of 9g indicating highly rugged relays that are not expected to cause sny inadvertent actuation of the deluge valve. BAsed on this review and the review of the field conditions of the fire piping, the SRT concluded that this piping does not represent a credible interaction or flooding source.

3.1.1.8 <u>Sources of Conservatism</u>

Previous design practices generally yield a substantial reserve margin between the required seismic demand and seismic capability of nuclear power plant structures and equipment. The required design SSE peak ground acceleration is only 0.15g for powerblock structures, systems, and components at LGS and resulting in-structure peak spectra demands are generally less than 3.0g. However, it is not unusual to find equipment that was dynamically qualified to acceleration levels greater than 10g. In

addition to the usual sub-structure amplifications, the following list provides some sources of conservatism in the seismic design criteria and associated methodology based on an overview of LGS seismic design basis documents. Most are typical of the methods in practice, codes in effect, and USNRC regulations in effect at the time LGS was designed (mid-late 1970s). But, some are unique for LGS.

(1) The design basis ground spectra is conservative for this site based on the recent seismic hazard estimates. The NUREG/CR-0098 (ref. 3.1-8) shape used is enriched between 8 and 10 Hz. Also the spring elements used to model the rock interface have introduced additional conservatism into the input motion.

(2) A single broad frequency content synthetic earthquake acceleration time history was derived for use as an input to generate floor response spectra. Spectra were generated from this single time history analysis which is typically 25 to 35 % more conservative than multiple time history analyses.

(3) Seismic design of Category I structures was performed by using linear elastic techniques. However, experience tells us that past near failures involve some degree of yielding, which results in nonlinear inelastic energy absorption. The original seismic design documents did not account for these inelastic energy absorption mechanisms and consequently substantial factors of safety were built in at various design states.

(4) For seismic equipment qualification by testing, the test response spectra usually envelop the required response spectra over the frequency range of interest with a reserve margin.

(5) For dynamic qualification of similar pieces of equipment, dynamic demand was usually calculated by conservatively enveloping demand at different floor locations. This usually results in unrealistic dynamic demand with more than one peak and broad frequency content.

3.1.2 System Analysis

3.1.2.1 Selection of Success Path Systems

This section presents the systems analysis and component selection process used to develop the Success Path Component List (SPCL) used in the SMA. This process begins with the identification of important plant safety functions and ends with the selection of specific components in

specific systems required to fulfill these functions and safely shutdown the plant.

Technical input utilized for this project has a design document "freeze" date of September 1, 1992. However, any plant modifications made after this date utilized the same design practices as described in Section 3.1.1-2. Documents required for clarification or completion of the evaluation issued after September 1, 1992 were used and so noted.

3.1.2.2 Systems Evaluation Team

The primary SMA team sub-group responsible for systems evaluation is identified as the Systems Evaluation Team. The Limerick Generating Station systems evaluation team was organized with personnel having significant experience, understanding, and familiarity with systems necessary for the operation of nuclear generating stations. All members of the systems evaluation team received the appropriate EPRI and SQUG sponsored training courses regarding the selection and identification of systems and components required to accomplish the safe shutdown objectives.

3.1.2.3 Analytical Assumptions

The development of the SPCL assumes that the RLE occurs, offsite power is lost for 72 hours and a small break LOCA (SBLOCA) is likely to occur in containment. It is required that the chosen success paths be able to bring the plant to and hold it in a stable shutdown for 72 hours.

The bases and further discussion of these assumptions are discussed in more detail in EPRI Report NP-6041-SL. Loss of off-site power is a realistic consequence of the SME. Ceramic insulators in switchyards are historically weak in seismic events. The assumption of a small break LOCA is made to avoid extensive walkdowns of small lines inside containment. The "LOCA" in the assumption is the combined leakage from instrument line breaks, not failures of primary system piping. Loss of shutdown decay heat removal affects the ability to protect the containment from over-pressure failure. Discussion of seismically induced fires is covered under the Fire Risk Scoping Study issues in Section 4.

3.1.2.4 Safe Shutdown (SSD) Functional Success Path Determination

Given the above deterministic constraints, acceptable "paths" for achieving and maintaining hot or cold shutdown for 72 hours must be identified. The EPRI SMA methodology specifies that the intent of the IPEEE seismic

evaluation is satisfied if a preferred and alternate SSD path are identified. However, NUREG-1407 states that a more complete set of potential paths should be identified initially, and that these potential paths should be narrowed down to two. In contrast to a PRA which is geared toward estimating plant damage states, the SMA is based on showing successful defense of fission product barriers. Thus, the SMA paths desired are those which end with fuel cladding, reactor vessel, and primary containment intact.

Safety functions are defined as groups of systems and actions that prevent undesirable effects, such as damage to the core, containment failure, and/or release of radioactivity to the environment. Since the focus of the SMA is to identify a set of actions that results in a stable plant condition following the RLE, only the safety functions that preclude core damage are addressed. As defined in EPRI Report NP-6041-SL (ref. 1.3-1), these core protection safety functions are as follows:

- Reactivity Control
- Reactor Coolant Pressure Control
- Reactor Coolant Inventory Control
- Decay Heat Removal

Failure of any one of these functions could result in subsequent core damage.

A successful path involves first, successful reactivity control. Reactivity control is an important safety function because the amount of heat that must be removed from the core is determined by how well this function is performed. Reactivity control is established by control rod insertion subsequent to a reactor trip signal.

With the reactor subcritical, only core decay heat must be removed. Reactor inventory (level) and pressure must be controlled to facilitate heat removal. Inventory requirements are increased with the SBLOCA.

3.1.2.5 SSD Systems

To derive component level "Success Path Component Lists" (SPCLs) for the functional success paths, the frontline systems and their support systems must be identified. Because ECCS equipment at LGS is seismically qualified, and because such equipment is familiar to operators for use in an emergency, the frontline systems identified are those in place already to handle emergencies. Operators are familiar with the use of the ECCS through regular training, and the LGS Emergency Operating Procedures direct operators in its use. Although additional equipment is in place which

is capable of responding to the event, the emphasis on success paths leads us to defer its consideration. The ECCS have the additional benefit of requiring fewer support systems and human actions which simplifies both the analysis and the walkdowns.

3.1.2.5.1 SSD Frontline Systems

The frontline systems selected for satisfying the four safety functions (reactivity, vessel inventory, vessel pressure, and decay heat removal control) are the following: Note that systems descriptions are provided in section 3.2.1 of the LGS IPE.

- Control Rod Drive System (CRD)
- Automatic Depressurization System (ADS)
- High Pressure Coolant Injection System (HPCI)
- Reactor Core Isolation Cooling System (RCIC)
- Main Steam System (MS)
- Residual Heat Removal System - Low Pressure Coolant Injection mode (RHR-LPCI)
- Residual Heat Removal System - Shutdown Cooling mode (RHR-SDC)
- Residual Heat Removal System - Suppression Pool Cooling mode (RHR-SPC)
- Residual Heat Removal System - Alternate Shutdown Cooling mode (RHR-ASC)

3.1.2.5.2 SSD Support Systems

The support systems required are those which provide power (AC, DC, pneumatic), component cooling (ESW, RHRSW), room cooling, or Instrumentation and/or control to both frontline other support systems. A benefit of using the ECCS for frontline systems is that fewer support systems are required. For the frontline systems selected, the following support systems are required (Note that descriptions of these support systems are provided in Section 3.2.1 of the Limerick IPE):

- Emergency Service Water System (ESW)
- Residual Heat Removal Service Water System (RHRSW)
- Emergency Diesel Generator System (DG)
- HPCI, RCIC and RHR Pump Room Cooling
- Diesel Generator Building HVAC System
- Class 1E AC Distribution System
- Class 1E DC Distribution System

3.1.2.5.3 Preferred and Alternate Success Paths

The various combinations of the above systems that meet the four functions form the full set of possible success paths. Two paths, the preferred and the alternate, were chosen from these possibilities and are based on the LGS Appendix R safe shutdown methodology. These paths are listed in Table 3.1.2-1 and the systems are briefly described below.

The preferred method has reactivity control via the control rods, inventory control via HPCI / RCIC systems, overpressure protection via all of the SRVs, depressurization control via the ADS SRVs and heat removal by the "A" loop of suppression pool cooling and shutdown cooling. The alternate method has reactivity control again via the control rods, inventory control by the "C" LPCI / "D" LPCI systems, overpressure protection via all the SRVs, depressurization control via the ADS SRVs, and heat removal by the "B" loop of suppression pool cooling and alternate shutdown cooling. The chosen paths maximize equipment diversity.

TABLE 3.1.2-1

PREFERRED AND ALTERNATE SHUTDOWN PATHS

Function	Preferred Path System(s)	Alternate Path System(s)
Reactivity Control	Control Rod Drive HCUs and Scram Discharge Volume	Control Rod Drive HCUs and Scram Discharge Volume
Pressure Control	All SRV's	All SRVs
Inventory Control	HPCI / RCIC [1]	"D" LPCI / "C" LPCI
Heat Removal[2]	A Suppression Pool Cooling A Shutdown Cooling	B Suppression Pool Cooling B Alternate Shutdown Cooling
Support Systems	Diesels D*1 & D*3 Diesel Enclosure Ventilation A & C RHRSW A&C ESW Supp. Pool Level, Temp. Inst. Reactor Level/Press Instrumentation Class 1E DC	Diesels D*22 & D*24 Diesel Enclosure Ventilation B & D RHRSW B & D ESW ADS N_2 Bottles Supp. Pool Temp/Level Inst. Reactor Level/Press Instrumentation Class 1E DC

1 Given a SB LOCA, the ADS system would be required to depressurize the reactor below the RHR pump shutoff head so that LPCI "C" or "D" can be initiated.

2 As stated in EPRI NP-7498, evaluation of systems and equipment whose functionality is required to prevent long term containment failure is not necessary because previous PRAs indicate that risk to the public due to severe accident sequences involving failure of long term containment integrity is low. Therefore, review of the spray systems can be excluded from the scope of the IPEEE. However, the primary containment spray system up to the inboard isolation valve (containment side) was included on the SPCL and a seismic review was performed by the SRT.

3.1.2.5.3.1 Reactivity Control

Adequate shutdown margin will be established and maintained by the use of the control rods. The initial control of reactivity using the control rods and the Hydraulic Control Units (HCUs) is considered single failure proof. The actual mechanism of the control rod insertion (i.e., manual or automatic scram) was not considered in this analysis. The components which comprise the Control Rod Drive System (outside of the control rods and HCUs) and Reactor Protection System are not included on the SPCL.

No direct means of verifying reactivity within the reactor has been included on the SPCL due to their vulnerability to failure during a seismic event. The ATWS (Anticipated Transient Without Scram) analyses performed to support the LGS IPE indicate that the suppression pool heatup resulting from conditions involving the failure to fully shutdown the reactor can be significant. Suppression pool heatup is a parameter that will require specific operator response as directed in the Emergency Operating Procedures (EOPs). Increases in suppression pool temperature beyond those normally expected after a transient (decay heat) will indicate that the reactor was not fully shutdown. Therefore, the suppression pool temperature response as measured by temperature indication credited in the SPCL will be used as an indication that the reactor successfully scrammed.

3.1.2.5.3.2 Reactor Coolant Pressure Control

The primary method used in depressurizing the reactor vessel is the Automatic Depressurization System. Normally, the ADS valves automatically open in their SRV mode to maintain reactor pressure at their assigned setpoint. However, these valves can be manually actuated from the control room if necessary to maintain pressure within limits or depressurize the reactor.

Overpressure protection for the reactor vessel is provided by the operation of the main steam relief valves (SRVs). This function occurs automatically when reactor pressure reaches the setpoint for each SRV. The SRV discharge vacuum relief valves prevent drawing suppression pool water into the SRV discharge lines following termination of blowdown. Reactor coolant pressure control is monitored using instrumentation loops XR-42-1(2)R623A and XR-42-1(2)R623B.

3.1.2.5.3.3 Reactor Coolant Inventory Control

Reactor Water Level Control

The inventory of water within the Reactor is supplied from the Suppression Pool and will be maintained by the Reactor Core Isolation Cooling (RCIC) System, High Pressure Coolant Injection (HPCI) System and Low Pressure Coolant Injection (LPCI) mode of the Residual Heat Removal (RHR) Systems. The HPCI and RCIC Systems are normally aligned to take suction from the Condensate Storage Tank (CST). When the water level in the CST falls below a predetermined level or the suppression pool level is high, the HPCI pump suction is automatically transferred to the suppression pool. Level transmitters for the CST and suppression pool have been included on the SPCL to eliminate the need for manual operator action in the event the CST fails as a result of the seismic event. Both HPCI and RCIC pump suctions can be manually aligned to the suppression pool using components on the SPCL.

The HPCI System is the preferred method for ensuring that the reactor core is adequately cooled in the event of a small break in the reactor coolant pressure boundary. The HPCI System will maintain reactor water level (including a small break LOCA) when reactor pressure is greater than the RHR pump shutoff head (LPCI mode). Injection water is piped to the reactor via the Core Spray sparger pipe and through the feedwater sparger.

The RCIC System provides an additional method for maintaining reactor water level. Injection water is piped to the reactor via a reactor feedwater line. The inventory of water within the reactor will be maintained by the RHR System (LPCI mode) when reactor pressure is reduced below the RHR pump shutoff head using the ADS valves. When LPCI operates in conjunction with reactor depressurization, the effective core cooling capability of the LPCI System is extended to all break sizes because depressurization rapidly reduces reactor pressure to the LPCI operating range. Makeup water is supplied to the reactor vessel from the suppression pool by operating RHR in the LPCI mode (4 loops).

Reactor water level is monitored using instrumentation loops XR-42-1(2)R623A and XR-42-1(2)R623B.

Discharges from the Reactor Coolant System

A review was performed on Limerick to determine the High/Low Pressure Interfaces for this analysis. All lines connecting to the reactor coolant pressure boundary that are not required for either reactor pressure or

inventory control will be isolated either automatically, or manually (operator action) utilizing plant emergency procedures, to prevent losses from the reactor coolant system. This includes the following potential leak paths:

- Main Steam System
- Reactor Head Vent
- Reactor Shutdown Cooling
- Reactor Water Clean-up

3.1.2.5.3.4 Decay Heat Removal

Since the main condenser may not be available following an SME, the primary method for removing reactor decay heat will be through the RHR System.

The RHR System is utilized for decay heat removal in various operating modes to provide Suppression Pool Cooling, Containment Spray, Shutdown Cooling and Alternate Shutdown Cooling.

Heat is removed from the Suppression Pool following blowdown from the SRVs and/or operation of HPCI or RCIC pumps by operating the RHR System in the Suppression Pool Cooling mode. In this mode, water from the suppression pool is circulated through an RHR heat exchanger, and returned to the suppression pool.

Pressure and temperature build up in the Containment can be reduced by aligning the RHR system for Containment spray mode. In this mode, water from the suppression pool is cooled by an RHR heat exchanger and circulated to spray headers in Containment. However, as stated in EPRI NP-7498 (ref. 3.0-2), evaluation of systems and equipment whose functionality is required to prevent long term containment failure is not necessary because previous PRAs indicate that risk to the public due to severe accident sequences involving failure of long term containment integrity is low. Therefore, review of the spray systems can be excluded from the scope of the IPEEE. However, the primary containment spray system up to the inboard isolation valve (containment side) was included on the SPCL and a seismic review was performed by the SRT.

Once the reactor has been depressurized below 75 psig, the RHR system can be aligned for the Shutdown Cooling mode which circulates water from the reactor through the RHR heat exchanger and back to the reactor.

An Alternate Shutdown Cooling mode has been defined from the Fire Protection Evaluation Report. This method uses an RHR pump to circulate

water from the suppression pool through an RHR heat exchanger and discharge into the reactor vessel through either the LPCI injection line or the shutdown cooling return line to the Reactor Recirculation loop. Water from the reactor vessel is returned to the suppression pool by opening a minimum of two ADS valves. The reactor water level is allowed to increase and fill the steam line with water spilling back to the suppression pool through the open ADS SRV(s).

For the RHR System modes described above, heat is removed from the RHR heat exchanger by the RHRSW system which transfers heat at the Spray Pond.

3.1.2.5.4 Nonseismic Failures and Human Actions

3.1.2.5.4.1 Non-seismic Failures

To address the issue of non-seismic caused component or system unavailability, the following method has been used. In developing this methodology, the focus was on the following statement provided in Section 3.2.5.8 of NUREG-1407:

"The redundancies along a given success path should be specifically analyzed and documented when they exist".

The success paths selected for Limerick were based upon the Success Path Logic Diagram (SPLD) that was developed at the beginning of the SPCL phase of the project. The SPLD provided for two separate success paths to ensure that each safe shutdown function could be performed. Using this approach, if one system were to fail, or not be available, the alternate success path could be utilized. In order to ensure additional reliability of a given success path, all redundant components within a particular multi-train system that makes up a portion of the success path were included on the Success Path Component List (SPCL). This approach is consistent with the NRC direction in NUREG-1407.

It was identified, from the LGS IPE Report, that the unavailability of the RCIC system was greater than that which is recommended in EPRI Report NP-6041-SL for a single train system. If RCIC is unavailable and HPCI is not available, reactor depressurization would be required (via the ADS system) to reduce reactor pressure below the RHR pump shutoff head for LPCI initiation.

3.1.2.5.4.2 Human Actions

As recommended in EPRI Report NP-6041-SL, this issue can best be handled with a Systems Evaluation Team (SET) that is familiar with the operations of the plant and the procedures utilized to shutdown the plant. In addition to the systems team, the plant's Operations Department was also called upon to review and provide comment on both the system and component selection (SPLD and SPCL) for compatibility with normal and emergency procedures.

Systems selected for performance of the safe shutdown functions were based upon the SET's knowledge of the plant's normal and special events and emergency operating procedures. Using this approach, the project considered the availability of instrumentation, plant procedures and operator training on these procedures. The procedures are all available in the Main Control Room; the operators are trained on these procedures, and the procedures contain all operator actions required in the success path.

In addition to the above, PECO Energy reviewed human interactions in the LGS Level 1 IPE. Considerable effort was expended in this portion of the IPE to ensure that the assumptions used were accurate and the models were complete. The operator actions were grouped into 3 classes of interaction for the study; pre-cursor, recovery and post-initiating actions. For the purposes of this IPEEE effort, the pre-cursor actions would be the same.

The post-initiating event operator actions are consistent with the activities which will need to be performed following a seismic event. The results of this review did not show operator actions to adversely affect the plants ability to remove decay heat, or increase the total core damage frequency. Operator actions required for the Alternate Shutdown Cooling System are similar to those analyzed for Shutdown Cooling, and Suppression Pool Cooling in the LGS IPE and therefore should not adversely affect the plants ability to remove decay heat, or increase the total core damage frequency.

Based upon the system selection for IPEEE, use of existing operating procedures and considering the results of LGS IPE, the probability of human actions are considered to be low enough so as not to affect the LGS SMA.

3.1.2.5.5 Success Path Component List Development

With the frontline and support systems identified in a preferred and alternate success path, the components within these systems must be determined so

that seismic evaluations may proceed. These components are the mechanical (pumps, valves, tanks, heat exchangers, compressors, etc.) and the electrical (buses, transformers, switchgear, instruments, relays, power supplies, batteries, etc.) equipment necessary for system function. Using the P&IDs, electrical one line and schematic diagrams, emergency operating procedures, and the results of previous work for the IPE and Appendix R evaluations, a component list (so called "Success Path Component List") is generated. It is the detailed seismic margin assessment of these components which allows PECO Energy to determine success relative to surviving the RLE.

The basic procedure followed to determine the SPCL is outlined in the steps below:

(1) Select a system from the preferred or alternate success path;
(2) Identify the components in the selected system by review of the IPE, emergency operating procedures, and Appendix R systems analyses;
(3) Review P&IDs and electrical one line and schematic diagrams to ensure all components are accounted for; and
(4) Look up components in the PECO Energy equipment data base for location information.

Seismically qualified systems and components are used typically in the SPCL. The use of qualified components minimizes the amount of judgment which must be used to determine seismic capacity of the components.

The "rule of the box" was applied in development of the SPCL, particularly for instruments and associated electrical components. The "rule of the box" states that components mounted on or in larger pieces of equipment do not have to be considered separately. This means that if the instrument racks and cabinets housing electrical components are seismically strong, anchored correctly, and free from seismic interaction concerns, the electrical equipment on/within them is assumed to survive the RLE ("rule of the box"). Thus, if an instrument/electrical component resides on a piece of equipment or within a cabinet or on a rack, only the equipment, cabinet, or rack is listed.

3.1.2.5.6 Success Path Component List

The Success Path Component List provides the component lists for all frontline systems and support systems, including all instruments (I&C). All components listed are required to support one of the two paths described in Section 3.1.2.

A number of equipment items/types are particularly important because they are common to several systems and their seismic common cause failure would affect these multiple systems. These include the batteries/distribution panels, safety relief valves and emergency diesel generators.

3.1.2.6 Relay Evaluation

The relay evaluation is not required for a reduced scope submittal.

3.1.2.7 Summary of SMA Systems Analysis

The safe shutdown paths selected and the corresponding SPCL equipment satisfy the constraints imposed by the SMA methodology and are capable of bringing the Limerick Generating Station to safe shutdown after the RLE. The SPCL equipment selection is derived from a review of the four safety functions whose satisfactory fulfillment is required for hot shutdown: reactivity control, reactor coolant system pressure control, reactor coolant system inventory control, and decay heat removal. Seismic margins assessment of this equipment provides an estimate of the seismic capacity of the selected safe shutdown paths.

To cope with a LOOP all equipment is either AC power independent or powered from the emergency diesel generators. It is pointed out, however, that the safe shutdown paths selected are not dependent on LOOP. The IPE and additional IPEEE examinations confirm that the equipment selected is sufficient for coping with a LOOP and a small break LOCA for 72 hours. With power from the emergency diesel generators and only a small break, adequate vessel make-up is assured by the RCIC, HPCI or LPCI pumps. The break is not so large as to render reactor vessel make-up sources inoperable and does not disable a loop of RHR. With the reactor shutdown (i.e., rod insertion), decay heat can be removed via the RHR system.

Sufficient redundancy is provided by inclusion of the RCIC and HPCI pumps for maintaining high pressure vessel inventory in the event of a small LOCA. Other systems are multi-train, providing protection at least to the single active failure depth. Only seismically qualified ECCS equipment is required to cope with the RLE. Emergency procedures exist for the use of ECCS equipment and are part of operator training. Thus, protection against non-seismic failures and adequacy of human actions is provided for. The possibility of seismically induced fires is evaluated in Section 4.8.

3.1.3 Analysis of Structure Response

As stated in Section 3.1, LGS has chosen to submit a response for the IPEEE consistent with the requirements contained in NUREG-1407 and GL 88-20 Supplement 4 for a reduced scope plant. Therefore the RLE is the applicable SSE ground response spectra.

3.1.3.1 Structural Damping

The Limerick Generating Station design basis considered the following structural damping values:

Reinforced Concrete and Welded Steel	5%
Bolted Steel	7%

Recommended damping values for the seismic design of nuclear power plants are presented in Regulatory Guide 1.61 (ref. 3.1-2). The recommended value for reinforced concrete structures is 7% for the SSE. Along with these recommended values goes a stipulation that the maximum combined stresses due to static, seismic and other dynamic loads should be close to half of yield stress and yield stress for the SSE.

The structural damping values used were taken to be those values used in the design basis evaluations to be consistent with the original SSE levels.

3.1.4 Evaluation of Seismic Capacities of Components and Plant

3.1.4.1 Seismic Margin Assessment and Screening

The SMA approach for LGS assesses the appropriate equipment, systems, and structures to show that these components meet the design basis requirements.

Equipment identified in the SPCL is sorted by equipment type or class into categories that are consistent with those developed by EPRI. The safe shutdown equipment categories, as adopted from EPRI NP-6041 and expanded for LGS, are identified in Table 3.1.4-1.

Various methods are available for determining the seismic functional capability levels of equipment. These methods include those recommended by EPRI and SQUG in addition to seismic capability data found in the specific equipment qualification reports. A preferred sequence of consideration and utilization for these available methodologies during the SMA is:

- Screening criteria in Table 2-4 of EPRI NP-6041
- Original equipment qualification reports

Several approaches are available for the assessment of the equipment anchorage adequacy. These approaches include, but are not limited to, the following:

- Existing anchorage analysis/test qualification levels for the design basis event that are acceptable and meet the LGS commitments;

- Anchorage analysis qualification calculations that may be reworked with appropriate refinements/ enhancements to show acceptance for the SSE event; and

- Generic bounding calculations for typical holddown details (i.e., expansion anchors, fillet welds, grouted anchors, etc.).

The identification of potential seismic interaction issues is a key element of the seismic margin walkdowns. Resolution of seismic interaction items/concerns is either provided by engineering review/judgment during the equipment walkdown phase or by engineering analysis during the seismic margin evaluation phase. In addition, generic bounding analysis calculations, such as the evaluation of masonry walls, were prepared for resolving seismic interaction concerns.

The approach taken for the seismic margin evaluation of structures and distributed systems is also in accordance with the recommendations and guidelines presented in EPRI NP-6041.

The selected approach for the seismic margin evaluation emphasized thorough walkdowns and reviews will result in complete, accurate, and plant-specific conclusions relative to seismic margins.

3.1.4.1.1 Screening Criteria

In the course of the seismic margin evaluation, an effort to eliminate (screen-out) elements from detailed review is provided by considering conservative seismic capacity screening criteria. The screening criteria are based upon experience and judgment concerning the seismic ruggedness of the component and its ability to withstand an SSE. Tables 2-3 and 2-4 of EPRI NP-6041 provide recommended screening criteria for civil structures as well as equipment and subsystems. These recommendations for screening are based upon SPRA studies, actual earthquake experience data, seismic qualification data, generic equipment ruggedness spectra

(GERS), and the combined judgment and experience of the expert panel on the quantification of seismic margins in NUREG/CR-4334 (ref. 3.1-10). In general, the Table 2-3 and 2-4 screening guidelines are the basic screening tools used in the seismic margin review.

The screening of equipment elements was the responsibility of the SRT. The screening process involves several key considerations prior to reaching a final decision. A plant walkdown by the SRT was required to confirm the seismic ruggedness of each equipment item in conjunction with the recommended screening guidelines and criteria. In addition, the screening caveats or footnote restrictions from the screening criteria tables were evaluated to satisfy their concern/requirements or provide a resolution if they could not be satisfied. Seismic interaction was addressed for all equipment items regardless of the screening criteria and guidance presented.

Once the decision was by the SRT to screen an equipment element, no further evaluations or review were required, indicating that the equipment is considered seismically rugged and capable of withstanding the SSE level event without adverse effect to the safety of the plant. Elements that are not capable of being screened are reviewed and evaluated in further detail to assure that the plant design basis ground motion level can be adequately accommodated.

3.1.4.1.2 Walkdown Team

The Limerick Generating Station seismic margin assessment team was organized with personnel having significant experience, understanding, and familiarity with the seismic analysis/design and systems requirements of nuclear generating stations. The primary SMA team sub-group responsible for the seismic capability evaluations is identified as the Seismic Review Team (SRT). All members of the SRT received the appropriate EPRI and SQUG sponsored training courses regarding seismic evaluation, walkdown screening, and seismic capability. SRT members and key seismic capability individuals are introduced below with a brief summary of their experience and qualifications.

Charbel M. Abou-Jaoude (SRT Member)

Mr. Abou-Jaoude is a Project/Service Area Manager in the Engineering Analysis Department at VECTRA Technologies, Inc. He has a B.E., Mechanical Engineering from the American University of Beirut and an M.S., Civil Engineering from the University of Michigan. He has more than ten years experience in the dynamic analysis of equipment, systems, and

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Mr. Antonopoulos is a Technical Lead Engineer in the Engineering Analysis Department at VECTRA Technologies, Inc. He has a B.S., Civil/Structural Engineering from the University of Massachusetts Dartmouth and an M.S., Structural Engineering from Northeastern University. He has more than twenty years experience in the design and analysis of nuclear power plant structures and components. He is well versed in the Generic Implementation Procedure developed by the Seismic Qualification Utility Group and the methodologies developed by the industry for response to Generic Letter 88-20, Supplement 4. Mr. Antonopoulos is a registered professional engineer in the states of Massachusetts and Rhode Island.

Richard E. Daniels (SRT Member)

Mr. Daniels is a structural engineer in the Programs & Procedures Section, Nuclear Engineering Division, PECO Energy Company. He has a B.S. and an M.S., Civil Engineering from Drexel University. He has more than twenty-five years of experience in civil, structural, and seismic engineering, most of which is in the nuclear power industry. He is well versed in the Generic Implementation Procedure developed by the Seismic Qualification Utility Group and the methodologies developed by the industry for response to Generic Letter 88-20, Supplement 4. He is currently PECO Energy Company's representative to SQUG. Mr. Daniels is a registered professional engineer in the state of Pennsylvania.

Daniel J. Fiorello (SRT Member)

Mr. Fiorello is a Senior Structural Engineer in the Technical Support Section, Nuclear Engineering Division of PECO Energy Company. He has a B.E., Civil Engineering from Villanova University and a M.S., Civil Engineering from the University of Pennsylvania. He has more than eighteen years experience in the design and analysis of nuclear power plant structures and components. He is well versed in the Generic Implementation Procedure developed by the Seismic Qualification Utility Group and the methodologies developed by the industry for response to Generic Letter 88-20, Supplement 4. He also served as PECO Energy

Company's representative to SQUG for two years. Mr. Fiorello is a registered professional engineer in the state of Pennsylvania.

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Mr. Flaherty is a Technical Services Manager in the Engineering Analysis Department at VECTRA Technologies, Inc. He has a B.S., Civil Engineering from Northeastern University and an M.S., Civil Engineering from Tufts University. He has more than twenty-five years experience in the seismic design and analysis of structures and equipment. He has extensive knowledge in the area of seismic design and equipment qualification. Mr. Flaherty is a licensed professional engineer in the state of Massachusetts.

Peter Guglielmino (SRT Member)

Mr. Guglielmino is a Business Area Manager in the Engineering Analysis Department at VECTRA Technologies, Inc. He has a B.S. and an M.S., Civil Engineering from Northeastern University. He has more than twenty years of experience in civil, structural, and seismic engineering. He is well versed in the Generic Implementation Procedure developed by the Seismic Qualification Utility Group and the methodologies developed by the industry for response to Generic Letter 88-20, Supplement 4. Mr. Guglielmino served as an SRT member during the Limerick Generating Station seismic margins assessment and has participated in similar roles at other power plants.

Robert P. Kennedy (Project Consultant)

Dr. Kennedy is a nationally recognized consultant to the nuclear industry in the areas of structural mechanics and seismic analysis issues. He has over twenty-five years of experience in static and dynamic analysis plus design of special purpose civil and mechanical-type structures for the nuclear, petroleum, and defense industries. Dr. Kennedy served as Chairman, Senior Seismic Review and Advisory Panel (SSRAP), jointly advising both nuclear power utilities and the US NRC on issues relating to seismic ruggedness of existing nuclear power plants. He is a member of NRC Expert Panel on Seismic Margin for nuclear power plants and co-author of Electric Power Research Institute (EPRI) Seismic Margin Research Program. He provided technical direction on seismic fragility portion of seismic probabilistic risk assessments for 23 nuclear power plants. Dr. Kennedy participated in the preliminary walkdown during the Limerick Generating Station Seismic Margin Assessment.

Miguel Manrique (SRT Member)

Mr. Manrique is a Technical Services Manager in the Engineering Analysis Department at VECTRA Technologies, Inc. He has a B.S., Civil Engineering from University of Rhode Island and a M.E., Structural Engineering and Mechanics from University of California, Berkeley. He has more than seventeen years experience in structural analysis and earthquake engineering of structures and equipment. He completed the Seismic Qualification Utility Group sponsored Walkdown Screening and Seismic Evaluation Training Course as well as the IPEEE Training Course and is certified as a seismic capability engineer. He has extensive knowledge in the area of seismic analysis and has performed or directed several soil-structure interaction analyses along with post-earthquake investigation of power facilities. Mr. Manrique is a licensed professional engineer in the state of California.

3.1.4.1.3 Walkdown Procedure

A major portion of the Seismic Margins Assessment was the walkdown of equipment on the SPCL. The effort performed at LGS was in accordance with EPRI NP-6041. Certain preparations were required prior to performing the walkdown to obtain maximum benefit from the walkdown.

- The physical location of the equipment was established based on a review of equipment arrangement drawings.
- Pertinent existing design basis documentation (including SQRT summaries, qualification reports, anchorage calculations, and equipment foundation drawings) were reviewed.
- A review of the various seismic IPEEE reference books was made to obtain a thorough understanding of the screening basis provided for the various equipment classes, structures and distributed systems.
- A review of the plant seismic design basis was performed.
- A comparison of some component seismic demand to seismic capacity was performed.
- Arrangements were made with plant personnel to open electrical equipment to inspect the internals.
- Copies of Screening and Evaluation Sheets (SEWS) from Appendix F of EPRI NP-6041 and walkdown checklists were obtained for the equipment class being walked down.

3.1.4.1.4 Equipment Walkdowns

3.1.4.1.4.1 Purpose

The purpose of the walkdown was to assist in addressing the three parts of equipment assessment (i.e., functional capability, anchorage adequacy, and seismic interaction).

3.1.4.1.4.2 Functional Capability

Any adverse seismic features associated with the equipment which could affect the functional capability of the equipment were identified during the walkdown. To address the functional capability of the equipment certain equipment caveats based on earthquake experience data and Appendices A and F of EPRI NP-6041 were reviewed. As a minimum, the caveats noted in Appendix F of EPRI NP-6041 and Part B of the SEWS sheets and under Section "i" of the walkdown checklists were reviewed during the walkdown.

3.1.4.1.4.3 Anchorage Adequacy

During the walkdown the equipment anchorage (type, number, size, etc.) was reviewed for conformance with the design documents and qualification reports. In some instances, the approach used for evaluating the anchorage required that additional anchorage information be obtained.

A review of the original anchorage design basis calculations and acceptance criteria was performed. The reviews performed by the SRT of the anchorage drawings, calculations and associated design margins enabled the SRT members to exercise judgment in the screening of most equipment anchorage configurations.

For elements not initially screened out during the seismic walkdown detailed evaluations were performed. Only a limited number of components required detailed evaluations.

3.1.4.1.4.4 Seismic Interaction

Seismic interaction addressed the effects of items external to the equipment being evaluated for any adverse impact on the safety-related function of the evaluated equipment. Interactions are typically grouped into proximity, seismic Category II/I, and spray or flooding. Proximity refers to the potentially adverse effect from the seismic motion of one component or element into another. Seismic Category II/I refers to the potential for failure of a seismic Category II component and its subsequent effect on a required Category I component. Spray and flooding may result from failure of either

Category II or I components. Specific seismic interaction items looked for during the walkdowns include the following:

- Good housekeeping practices, e.g., portable equipment, ladders, cleaning items, etc. properly stored/ secured.
- Adequate anchorage of nearby Non-Q equipment to prevent impacting an item listed on SPCL.
- Potential impact effects of swinging items (piping, light fixtures, doors, etc.) on other safety-related items listed on SPCL.
- Adequate space between item listed on SPCL and other items to prevent impact from differential displacement or out-of-phase response (e.g., adequate space between panels/racks and the wall). In lieu of this adjacent items should be secured together (e.g., adjacent panels should be bolted together).
- Flexible connections between points of differential movement:
  - ♦ Flex conduit attached to panels and instrument racks
  - ♦ Flex tubing/airline connections (i.e., there should be proper offsets in tubing runs to allow for differential displacement)
  - ♦ Flexible bellows
- Particular emphasis was placed on "soft" targets on essential components. Soft targets include but are not limited to such items as instrumentation, glass or ceramic components, Victaulic™ or threaded piping systems, and switches or relays.

3.1.4.1.4.5 Sampling

A detailed review of at least one component for each equipment type in an equipment class was performed (e.g., for the battery rack class of equipment, one 125 VDC battery rack assembly was reviewed in detail as a minimum). However, all accessible components were "walked by." The "walk by" considered the three parts of equipment assessment (functional capability, anchorage, and seismic interaction) but emphasized a confirmation that the construction pattern was typical and looked for unique seismic interaction concerns for each equipment item. A very limited number of equipment items were not walked by because they are located in contaminated areas, moderate to high radiation areas, or areas which are difficult to gain physical access. A sampling of the distribution systems (e.g., piping, cable trays, conduit, and HVAC ducting) was performed by the SRT in the areas containing essential equipment. Given the plant vintage and the design criteria, this was judged to be sufficient.

The walkdown approach discussed above is in agreement with the walkdown methodology and philosophy presented in Chapter 2 of EPRI NP-6041.

3.1.4.1.4.6 Documentation

In general, the walkdowns were documented on the Screening and Evaluation Sheets (SEWS) found in Appendix F of NP-6041. The detailed review for at least one item in an equipment class was documented on the SEWS and the "walk-by" of remaining items in the equipment class were documented by reference to the walkdown item on the screening evaluation sheets. In all cases two SRT engineers signed off the walkdown screening documentation.

3.1.4.2 Equipment Category Evaluations

As noted previously, the equipment on the SPCL was evaluated on an equipment category basis (Table 3.1.4-1). A summary of each equipment category evaluation is presented in this section (Table 3.1.4-2). This summary information is presented in a systematic/form like manner for each equipment category. This provides a concise synopsis on major equipment classes, their configurations, any unique features, method of anchorage, and a summary of the screening results; it also highlights issues that the SRT focused on during their reviews.

Only housekeeping and maintenance issues were identified as a result of this effort. These items are being tracked to assure that adequate resolution is provided. The issues are summarized in Table 3.1.4-3.

TABLE 3.1.4-1

LIST OF SAFE SHUTDOWN EQUIPMENT CATEGORIES

Category Number	Description
1	Motor Control Centers / Low & Medium Voltage Switchgear
2	Transformers
3	Horizontal Pumps
4	Vertical Pumps
5	Fluid (Air / Hyd)-Operated Valves
6	Motor-Operated Valves
7	Solenoid Operated Valves
8	Fans / Air Handlers
9	Chillers
10	Air Compressors
11	Motor-Generators
12	Distribution Panels
13	Batteries and Racks
14	Battery Chargers and Inverters
15	Engine-Generators
16	Instrument on Racks
17	Temperature Sensors / Local Instruments (Not on Racks)
18	Control Panels and Cabinets
19	Vertical Tank or Heat Exchanger
20	Horizontal Tank or Heat Exchanger
21	Vertical Suspended Tank
22	Buried Tanks
23	Conduit and Cable Tray Raceways
24	Piping
25	NSSS Components & Primary Loop
26	Traveling Screens & Sluice Gates
27	Control Rod Drive Assemblies
28	Building Seismic Gaps
29	Control Room Ceiling
30	Automatic Transfer Switches
31	Wall Mounted Contactor, Transmitter, Power Supply, etc.
32	Strainers and Filters
33	Building Penetrations of Underground Utilities

TABLE 3.1.4-2

EQUIPMENT CATEGORY EVALUATION

(1) Motor Control Centers/Low & Medium Voltage Switchgear

Number of Items in Success Path Component List: 58

Category is composed of medium and low voltage switchgear and motor control centers (MCC) in the reactor enclosure, control enclosure and spray pond pumphouse.

Walkdown Description/Screening Results

A. 4.16 kV Medium Voltage Switchgear (SWGR).

ABB (BBC/ITE SWGR) Model 5HK350, Horizontally racking SWGR.

Switchgears are mounted at elevation 239' of the Control Structure (~ 60' from the base of the building, within 40' of grade). The SWGR is welded to embedded C3 x 6.0 channels.

20A118 was selected for detailed walkdown, all other units were walked-by. 20A118 was opened and inspected in detail to verify consistency with design documents. The inspection also verified that adjacent cubicles were bolted together, and that electrical devices (such as relays, switches, terminal blocks, etc.) were in place and properly secured. Flexibility of base members and the adequacy of the load path were reviewed and judged to be acceptable. There were no excessive cut-outs; some cubicles had externally attached switches which are light weight and positively secured.

The switchgear assembly was screened based on the walkdown screening guidelines and the review of the original seismic qualification documentation.

The anchorage was judged to be adequate for the RLE, based on the review of the civil detail and the general configuration of the SWGR; based on the low aspect ratio, no net tension will develop and the welds provide sufficient margin to accommodate the resulting shear loads.

The walkdown identified a general housekeeping concern for the Unit 2 SWGRs; this may have been due to the Unit 2 outage which was

in progress at the time of the walkdown. The items of concern were unrestrained spare breakers and unrestrained ladders which may cause adverse interactions.

Overhead fluorescent lights are suspended with chains. The S-hook on the chain links were not always completely closed. The SRT judged that the light fixtures could not become unsecured in a seismic event since the S-hooks are partially closed and a vertical positive slippage of the light from the hooks is not credible for a chain type support. The walk-by which was conducted on Unit 1 during plant operation noted that all breakers were positively restrained and no similar housekeeping concerns were noted for Unit 1.

B. 480V Load Centers or Low Voltage Switchgear (LV SWGR).

ABB (BBC/ITE) K-Line load centers with K-6005 breakers. Two sections 42"W x 58"D x 90"H containing the individual breaker cubicles adjacent to the 4.16 kV to 480V step down transformer (100"W).

The load centers are mounted at elevations 253', 283' and 313' of the reactor enclosure. The breaker compartments are reinforced at the 4 outside corners with 3 x 3 x 3/16 angles which are then welded to two C4 x 7.25 channels that are embedded in the slabs with 7/8" nelson studs.

Load center 20B202 and the associated transformer (20X202) was selected for detailed inspection in view of outage component out of service schedule. The walkdown verified the anchorage details for the transformer and the two angles inside the bus box section. The individual breakers were examined to confirm that a seismic lateral restraint is provided at the back. The various sections were verified to be bolted together. There were no excessive cut-outs or externally attached items.

The load centers were screened based on the walkdown reviews and the review of the seismic qualification.

The load centers and transformers were selected for a detailed evaluation of the anchorage based on the configuration and the

building elevation for 20B202 and 20X202 (313'). The evaluation demonstrated that the anchorage was adequate for the RLE.

The walkdowns had identified some general housekeeping concerns in the vicinity of the Unit 2 load centers 201, 202 and 204. All items were judged to be related to ongoing outage activities. Also, the overhead lifting device, for both units, was noted as being unrestrained and may result in impact on equipment.

C. 440V AC Power MCCs are Cutler Hammer/Unitrol and 250V DC MCCs are Westinghouse Units

All MCCs are top braced to adjacent walls or to substantial structural members (braced tube supports - for spray pond MCCs and B223/4). Each MCC section is a standard 20"D x 20"W x 90"H, with typical line-ups consisting of 2 sections or more. The base channels are welded to embeds on reinforced concrete pads.

20B213 was one of the units selected for full walkdown. Also 20B219 which was being maintained as part of outage activities was inspected in detail for internal construction and device design and attachment. Breaker compartments have removable buckets which had positive restraints, adjacent sections were bolted together, and all devices were found secured with vendor provided hardware. The load path given the top bracing was very rugged, there were no large cut-outs or other items of concern found in the design and support configuration of MCCs.

The walkdowns identified one area in the reactor building of each unit at elevation 217' (Rooms 304 & 370) that contained housekeeping interaction concerns. The areas are used for the reactor building dress out and have scaffolding staging with high aspect ratios and minimal restraints. Similar Health Physics stations were found at rooms 506E and 508E at elevation 283' of the reactor building.

Based on the results of the walkdown, existing design basis documentation and screening criteria in Table 2-4, Appendix A and Appendix F of EPRI NP-6041, Rev. 1, the components were screened out, except as noted, and judged to meet the SSE by the SRT.

TABLE 3.1.4-2 (cont.)

Elements Not Screened Out

Systems Interaction : The noted Unit 2 housekeeping concerns.
The lifting devices for the LV SWGR also need to be restrained when not in use.

Assessment of Elements Not Screened Out

Housekeeping issues need to be addressed procedurally and implemented during normal plant operation and where required during outages. The lifting devices also need to be restrained.

TABLE 3.1.4-2 (cont.)

(2) Transformers

Number of Items in Success Path Component List: 28

This category includes 750-1000kVA and 25-75 kVA transformers.

Walkdown Description/Screening Results

The larger transformers are the step down units associated with the load centers and are ABB (BBC/ITE) Dry type. The coils are connected together at the top and the base channels are bolted to the metal clad base framing which is then welded to an embedded C4 x 7.25.

The other transformers are the typical Square D 30 kVA power distribution units. The base channel for these units has stiffeners to avoid reliance on weak way bending; the channels are either anchored or welded to the floor slabs or pads.

Transformers 20X202 and 10X109 were selected for detailed inspection. The clearance between the bus bars and the metal clad were checked, the hold down bolts and load path were inspected. The large transformers did not have a bracing for the top of the coils. Therefore these units were selected by the SRT for further evaluations to confirm the load paths and anchorage adequacy. The original qualification package was reviewed. The evaluation of the critical components showed them capable to accommodate the SEE loads.

There were no seismic spatial interactions identified.

Based on the results of the walkdown, existing equipment documentation and screening criteria in Table 2-4, Appendix A and Appendix F of EPRI NP-6041, Rev. 1, the transformers were screened out and judged to meet the RLE by the SRT.

Elements Not Screened Out

- None were identified

Assessments of Elements Not Screened Out

N/A

TABLE 3.1.4-2 (cont.)

(3) Horizontal Pumps

Number of Items in Success Path Component List: 8

This category includes horizontal pumps and turbines.

Walkdown Description/Screening Results

The RCIC and HPCI pumps and associated turbines are located at elevation 177' of the Reactor Enclosure for Units 1 and 2.

The Unit 2 RCIC and HPCI pumps and turbine were selected for a detailed walkdown. Unit 1 components were walked-by.

The RCIC pump (20P203) and turbine (20S212) are mounted on separate skids which are anchored on two separate concrete pedestals. Shear pins and keyway are used for alignment.

Anchorage was judged acceptable by SRT based on the walkdown and review of the existing civil structural anchor bolt calculations (File No. 22.8.C) for equipment foundations. In addition, concrete pedestals are adequately reinforced (i.e., tied down) to concrete floor and/or foundations per review of LGS design documents. Attached piping is well supported.

The HPCI booster pump (20P204) and turbine (20S211) are mounted on separate skids which are anchored on two separate concrete pedestals. The booster pump is skid mounted. The driver and pump are mounted on separate bases. However, the driver and pump are rigidly attached and adequate. Shear pins are installed. Suction and discharge piping is well supported.

Anchorage was judged acceptable by SRT based on the walkdown and review of the existing civil/structural anchor bolt calculations for equipment foundations. In addition, concrete pedestals are adequately reinforced (i.e., tied down) to concrete floor and/or foundations per review of LGS civil/structural drawings.

The auxiliary skid mounted components for both the HPCI and RCIC pumps and turbines were evaluated during the walkdown and judged to be acceptable.

TABLE 3.1.4-2 (cont.)

A walkby of the Limerick Unit 1 HPCI and RCIC pumps and turbines was performed by the SRT and judged to be similar to Unit 2 components and therefore acceptable. No seismic interaction problems were identified with these pumps.

Based on the screening criteria contained in Table 2-4, Appendix A, and Appendix F of EPRI NP-6041, Rev. 1 the above listed pumps were screened-out and judged to meet the RLE by the SRT.

Elements Not Screened Out

- None were identified

Assessment of Elements Not Screened Out

N/A

(4) **Vertical Pumps**

Number of components in Success Path Component List: 24

This category includes vertical pumps.

Walkdown Description/Screening Results

A. Residual Heat Removal (RHR) Pump

The RHR pumps are mounted at elevation 177' in the Reactor Enclosure for Limerick Units 1 and 2.

2AP202 was selected for a detailed walkdown, all other RHR pumps were walked-by. The walkdown utilized existing design documentation (pump vendor drawings, structural drawings for anchorage and seismic qualification reports).

During the walkdown it was noted that the anchor bolt pattern was not symmetric (i.e., no anchor bolts on face of flange at sump). Bechtel calculation 22.8C was reviewed by the SRT and it was determined that the design basis calculation for the RHR pumps account for the non-symmetric bolt pattern. The pumps are of good seismic design and adequately supported. The attached piping system is well supported and has adequate flexibility, thereby minimizing nozzle loading. All electrical conduits are well supported and have adequate flexibility and the pumps do not contain soft targets.

No seismic interaction problems were identified with these pumps.

Based on the screening criteria contained in Table 2-4, Appendix A, and Appendix F of EPRI NP-6041, Rev. 1, the RHR pumps were screened-out from further review by the SRT.

B. RHR Service Water Pumps and Emergency Service Water Pumps

The RHR Service water pumps and Emergency Service Water Pumps are mounted at elevation 268' of the Spray Pond Building. The pumps are "Common" and utilized for both Units.
Pump OAP506 was selected for a detailed walkdown. All other RHR

service water pumps and emergency service water pumps were walked-by. The walkdown utilized existing design documentation (pump vendor drawings, structural drawings for anchorage and seismic qualification reports).

The pumps are vertical pumps with intermediate supports. Some local corrosion of anchor bolt plate was noted but judged to be acceptable by SRT.

Based on the screening criteria contained in Table 2-4, Appendix A, and Appendix F of EPRI NP-6041, Rev. 1, the RHR service water pumps and emergency service water pumps were screened-out and judged to meet the RLE by the SRT.

There were no seismic interaction concerns identified during the walkdown.

C. Diesel Generator Diesel Oil Transfer Pump

The Diesel Generator diesel oil transfer pumps are mounted in the yard at elevation 206' in a valve pit which is part of the Diesel Oil Storage Tank Underground Structure.

Since the oil transfer pumps are located in a "Confined Workspace", special authorization is required for access. The SRT believes that a document review is sufficient to determine the structural adequacy of the transfer pumps and vault based on walkdowns performed to date which show a strong correlation between "as built" and "as-design" components and the overall structural adequacy at Limerick.

The Diesel Oil Storage Tank Structures Yardwork plans and sections are shown on Bechtel drawings C-1063, Rev. 10 and C-1064, Rev. 15. The valve pit is a reinforced concrete box structure. The analysis and evaluation of valve pit is given in Bechtel design basis calculation 47.10A, revision 2. The analysis is conservative.

The oil transfer pump is attached to the oil storage tank by means of a flanged connection to the oil tank. The flange is a 24" inside diameter by 28" outside diameter flange with four -5/8 inch diameter bolts.

TABLE 3.1.4-2 (cont.)

The diesel oil transfer pumps are Crane Company Size 1S. The pump is a vertical discharge type. A dynamic analysis of the pump was performed for the design basis evaluation. Verification of the analysis was performed by Bechtel and is given in Calculation EQG D-8-1. This evaluation demonstrates that all stresses are well below the code allowable values. A qualification Summary Report M79-20-3BC-1 performed by Bechtel demonstrates that the pump meets the requirements of the SQRT Program.

The SRT judged the anchorage to be sufficient.

Based on the screening criteria contained in Table 2-4, Appendix A, and Appendix F of EPRI NP-6041, Rev. 1, the oil transfer pumps and screened-out and judged to meet the RLE by the SRT.

Elements Not Screened Out

- None were identified

Assessment of Elements Not Screened Out

N/A

TABLE 3.1.4-2 (cont.)

(5) **Fluid (Air/Hyd) Operated Valves**

Number of components in Success Path Component List: 114

This category includes air and hydraulically operated valves.

Walkdown Description/Screening Results

110 of the 114 SPCL Fluid Operated Valves were walked down. Safety related mechanical equipment at LGS were procured with appropriate seismic design requirements. Dynamic qualification was performed by testing, dynamic analysis or a combination of test and analysis. Valves were generally procured to withstand a minimum of 3 g's in each direction (most valves had qualifications for 4.5 g's or more). Given the existence of qualification data on all valves, the walkdown screening did not require explicit verification of the valve operator height restriction (based on pipe size) specified in EPRI NP-6041.

The walkdowns focused on seismic spatial interactions, the flexibility of attached tubing/electrical conduit, and the potential for any differential displacements between the piping and any actuator supports.
Valves HV-51-2F041A and HV-51-2F041C were identified as having seismic spatial interaction concerns due to small clearances and required further review.

Except as noted above, there were no other seismic spatial interaction concerns identified where on-the-spot resolution was not possible. Several instances of close proximity effects were noted; in all cases the SRT was able to judge the available clearances as either sufficient or that any resulting impact would be insignificant and of no consequences to the operability of the valve.

In all cases attached lines had adequate flexibility. For valves which require back-up air supply (such as the Main Steam Safety Valves), attached tubing was tracked back to the accumulator tanks to verify structural adequacy and screen for any seismic interaction concerns.

The four inaccessible valves were screened based on similarity to other valves and the favorable screening results of the remainder of the valve population.

TABLE 3.1.4-2 (cont.)

Based on the documentation reviews performed, the results of the walkdowns, and the screening criteria in Table 2-4, Appendix A, and Appendix F of EPRI NP-6041, Rev. 1, all fluid operated valves were screened-out from further review and judged acceptable by the SRT.

Elements Not Screened Out

- System Interaction: Valves HV-51-2F041A and HV-51-2F041C

Assessment of Elements Not Screened Out

Subsequent to the seismic walkdown, the SRT performed an evaluation to determine the significance of the potential system interaction. The evaluation shows that there is a adequate clearance between the valves (HV-51-2F041A and HV-51-2F041C) and the structural members to prevent seismic interaction during a SSE and therefore the potential for system interaction is no longer a concern.

TABLE 3.1.4-2 (cont.)

(6) **Motor Operated Valves**

Number of items in Success Path Component List: 90

This category includes motor operated valves.

Walkdown Description/Screening Results

Eighty seven (87) of the 90 SPCL MOV's were walked down. During the walkdown, two valves were found to be disassembled for maintenance work. Safety related mechanical equipment at LGS were procured with appropriate seismic design requirements. Dynamic qualification was performed by testing, dynamic analysis or a combination of test and analysis. Valves were generally procured to withstand a minimum of 3 g's in each direction (most valves had qualifications for 4.5 g's or more). Given the existence of qualification data on all valves the walkdown screening did not require explicit verification of the valve operator weight and height restriction (based on pipe size) specified in EPRI NP-6041.

The walkdowns focused on seismic spatial interactions, the flexibility of attached electrical conduit, and the potential for any differential displacements between the piping and any actuator supports.

There were no seismic spatial interaction concerns identified where on-the-spot resolution was not possible. Several instances of close proximity effects were noted; in all cases the SRT was able to judge the available clearances as either sufficient or that any resulting impact would be insignificant and of no consequences to the operability of the valve.

In all cases attached electrical conduits had adequate flexibility. There were a number of heavy valve actuators which were internally supported. In all instances the piping was also supported to the same structure in close proximity to the valve body and therefore judged to be acceptable by the SRT.

The 3 inaccessible and 2 disassembled valves were screened based on a drawing review, similarity to other valves and the favorable screening results of the remainder of the valve population.

Based on the documentation reviews performed and the results of the walkdowns, all motor operated valves were screened-out from further

review and judged to be adequate for the plant RLE (0.15g pga).

Elements Not Screened Out

- None were identified

Assessment of Elements Not Screened Out

N/A

(7) **Solenoid Operated Valves**

Numver of Components in Success Path Component List: 4

This category includes solenoid operated valves.

Walkdown Description/Screening Results

Detailed walkdowns were performed by the SRT for the solenoid operated valves (SOVs). Other SOVs were walked by, but are not listed since they were tracked as "Rule of the Box" with the associated fluid operated valves they control.

The SOVs are of compact design and were screened out based on the adequate flexibility of attached lines. No seismic interaction concerns were identified during the walkdown.

Based on the results of the walkdown, and screening criteria in Table 2-4, Appendix A, and Appendix F of EPRI NP-6041, Rev. 1, the SOVs were screened out from further review by the SRT.

Elements Not Screened Out

- None were identified.

Assessment of Elements Not Screened Out

N/A

TABLE 3.1.4-2 (cont.)

(8) **Fans/Air Handlers**

Number of components in Success Path Component List: 58

This category includes fans and unit coolers.

Walkdown Description/Screening Results

A. Diesel Generator Ventilation Air Exhaust Fans

The diesel generator ventilation air exhaust fans are mounted to structural support steel and platform steel at elevation 225'-8" of the Diesel Generator Building for Units 1 and 2. There are two diesel generator ventilation air exhaust fans per diesel unit.

2AV512 was selected for detail walkdown, all other components were walked-by. The plant walkdown utilized existing design basis documentation (e.g., component drawing, structural drawings for anchorage and qualification reports).

It was observed during the walkdown that the fans are well supported from structural steel to limit lateral displacement under seismic loading. Internal devices appeared seismically adequate. None of the fans were mounted on vibration isolators.

Ancillary equipment was verified to be adequately supported and adequate flexibility in interconnecting components, such as temperature elements and ducting to accommodate relative movement.

The anchorage was visually inspected and found to match details shown on design drawings and considered in existing qualification documentation. There were no seismic interaction concerns identified.

The screening criteria in Table 2-4, Appendix A, and Appendix F of EPRI NP-6041, Rev. 1 in conjunction with a walkdown serves as the basis for the acceptance of the units. On this basis, the units were screened-out and judged to meet the RLE by the SRT.

TABLE 3.1.4-2 (cont.)

B. RHR Pump Room Unit Coolers

The RHR pump room unit coolers are mounted in the Reactor Enclosure for LGS Unit 2 at elevation 191' for 2AV210, 2BV210, 2CV210 and 2DV210. 2EV210, 2FV210, 2GV210 and 2HV210 are mounted at elevation 182'-10". The locations and elevations are similar for Unit 1. The coolers are bolted to stiffener plates which are welded to the structural steel platforms.

The fans and coolers are manufactured by American Air Filter Company.

2AV210 was selected for detailed walkdown, all other units were walked-by. The seismic walkdown utilized existing documentation (e.g., component drawings, structural drawings for anchorage, and qualification reports).

The anchorage was visually inspected and found to match details shown on design drawings and judged to be acceptable.

During the seismic walkdown it was noted that the fan coolers coil tubing is copper. The SRT was concerned that failure of the copper tubing could become a "source" for flooding of the rooms.

The SRT performed a review of the design basis documentation to determine the structural adequacy of the coolers and tubing. Based on the review of the documentation which included Bechtel calculation EQG-D-62-3, dated 6/14/89, Seismic Qualification Report NESE 188, Rev. 4, nozzle loads from attached piping are low due to piping system being well supported. Therefore the SRT judged that the coil copper tubing is adequately designed and is not a source for flooding of the room.

During the walkdown, the SRT noted a potential interference between the copper coil on Unit 1HV210 and structural steel. The clearance between coil header and steel bracket bolt is 1/8" The SRT reviewed the existing Qualification Data Package (Package D-62). The natural frequency of the fan coil assembly is 11.5 Hz side to side and 12.4 Hz front to back. The displacement for 1g at 11 Hz is 0.08". For the equipment location, the spectral demand for the RLE is less than 1g. Therefore, the existing 1/8" gap is sufficient.

TABLE 3.1.4-2 (cont.)

There were no other seismic interaction concerns identified.

The temperature element for this unit is locally mounted, has adequate flexibility and is covered by the rule of the box. All attached piping and electrical conduit are well supported and have adequate flexibility. There were no seismic interaction concerns.

Based on the results of the walkdowns, coupled with a review of existing qualification documentation and Table 2-4, Appendix A, and Appendix F of EPRI NP-6041, Rev. 1, the RHR fans and coolers were screened-out from further review and judged to meet the RLE by the SRT.

C. RCIC Pump and Turbine Room Unit Cooler

The RCIC pump and turbine room unit coolers are located in the Reactor Enclosure at elevation 191' for 2AV208 and elevation 184' for 2BV208. Similar location and elevations apply for LGS Unit 1.

The fans and coolers are similar to the RHR pump room unit coolers and have the same type of anchorage. A walk-by was performed. The anchorage was visually inspected and judged to be acceptable. All attached piping and electrical conduit are well supported and have adequate flexibility. There were no seismic interaction concerns.

Based on the results of the walk-by and similarity of these units to the RHR unit coolers, the RCIC room unit coolers were screened-out and judged to meet the RLE by the SRT.

D. HPCI Pump and Turbine Room Unit Coolers.

The HPCI pump and turbine room unit coolers are located in the Reactor Enclosure at elevation 177'. Both unit coolers are floor mounted. Similar location and elevation apply for LGS Unit 1 HPCI room coolers. The fans and coolers are similar to the RHR pump room unit coolers.

A walk-by was performed by the SRT. The anchorage for 2AV209 is starting to exhibit signs of rust but is not a near term problem. The SRT judged the anchorage to be acceptable. All attached piping

TABLE 3.1.4-2 (cont.)

and electrical conduit was well supported and had adequate flexibility.

There were no seismic interaction concerns. Based on the similarity to the RHR room unit coolers, the SRT screen-out the HPCI room unit coolers and they were judged to meet the RLE by the SRT.

E. Core Spray Pump Room Unit Coolers

The core spray room unit coolers are located in the Reactor Enclosure for LGS Unit 2 at elevation 190' for 2AV211 through 2DV211 and at elevation 177' for 2EV211 through 2HV211. The locations and elevations are similar for LGS Unit 1.

The fans and coolers are similar to the RHR pump room unit coolers and have the same type of anchorage. A walkby was performed by the SRT for all core spray pump room unit coolers. The anchorage was visually inspected and judged to be acceptable. All attached piping and electrical conduit was well supported and had adequate flexibility. There were no seismic interaction concerns.

Based on the results of the walkby and the similarity of these units to the RHR room unit coolers, the core spray room unit coolers were screened out and judged to meet the RLE by the SRT.

F. Spray Pond Room Unit Coolers

The spray pond room unit coolers are located in the Spray Pond Structure. The fans and coolers are similar to the RHR pump room unit coolers and have the same type of anchorage. A walkdown was performed by the SRT for both room unit coolers. The anchorage was visually inspected and judged to be acceptable. All attached piping and electrical conduit was well supported and had adequate flexibility. There were no seismic interaction concerns.

Based on the results of the walkdown and the similarity of these room unit coolers to the RHR room unit coolers, the spray pond room unit coolers were screened out and judged to meet the RLE by the SRT.

TABLE 3.1.4-2 (cont.)

Elements Not Screened Out

- None were identified

Assessment of Elements Not Screened Out

N/A

TABLE 3.1.4-2 (cont.)

(9) **Chillers**

Number of components in Success Path Component List: 2

This category includes chillers.

Walkdown Description/Screening Results

The chillers are located in the control structure and are mounted to the floor. 0AK112 was selected for a detailed walkdown, 0BK112 was walked by.

The control room cabinet for the chiller is mounted on isolation pads. The SRT judged this not to be a concern since the chiller is required to maintain pressure boundary only. Cabinet is screened out based on system functionality requirements. The compressor and chiller are supported by tube-steel with anchor bolts and judged to be acceptable.

The SRT noted general housekeeping issues in the area. Compressed gas bottles are loose and near chillers. The walkdown was performed during the Unit 2 refueling outage and the SRT judged these to be outage related and therefore not a concern.

Attached piping was well supported to minimize nozzle loadings and there were no differential movements noted. Anchorage details were verified to match those considered in existing qualification documentation. There were no seismic interaction concerns identified.

The screening criteria contained in Table 2-4, Appendix A, and Appendix F of EPRI NP-6041, Rev. 1, in conjunction with the walkdown, serves as the basis for the acceptance of the units. On this basis, the units were screened-out from further review and judged to meet the SSE by the SRT.

Elements Not Screened Out

- None were identified

Assessment of Elements Not Screened Out

N/A

(10) **Air Compressors**

Number of components in Success Path Component List: 16

This category includes air compressors.

Walkdown Description/Screening Results

Colt Industries : QR-25 Series Compressor, Model D#90 with 20 Hp. Motor

Starting Air Compressors are mounted at elevation 217' of the Diesel Generator Building for Units 1 and 2. There are two starting air compressors for each diesel.

2A1K513 was selected for detailed walkdown, all other units were walked-by. Six 5/8" diameter cast in place anchor bolts, three on each side of skid are used to attach the compressor skid to a twelve inch high pedestal that is integral with the base mat. Air lines and electrical conduit are well supported and have adequate flexibility. There are no vibration isolators on the skid.

Concrete cracks on the top surface were noted by the SRT team on several of the pedestals, however, no spalling of the concrete was evident. Due to their location and size, the SRT judged these cracks to be acceptable and believe they will not impair the strength of the cast-in-place anchor bolts.

Based on the results of the walkdown, coupled with a review of existing qualification documentation and Table 2-4 of EPRI NP-6041, Rev. 1, all air compressors were screened-out from further review and judged to meet the RLE by the SRT.

Elements Not Screened Out

- None were identified

Assessment of Elements Not Screened Out

N/A

TABLE 3.1.4-2 (cont.)

(11) **Motor Generators**

Number of items in Success Path Component List: 0

Walkdown Description/Screening Results

There are no motor-generators on the SPCL for LGS Units 1 and 2.

Elements Not Screened Out

- None were identified

Assessment of Elements Not Screened Out

N/A

TABLE 3.1.4-2 (cont.)

(12) **Distribution Panels**

Number of items in Success Path Component List: 54

This category includes DC floor mounted distribution panels, wall mounted distribution panels and 120V floor mounted distribution panels.

Walkdown Description/Screening Results

The floor mounted DC distribution Panels are manufactured by B.K Electric (ASCo). Typical configurations included a 2 section line-up; each section is 40"W or 30"W x 24"D x 90" H. Panels are top braced and have 8 ½" anchors at the base pad.

2BD105 was selected for detailed walkdown, all other units were walked-by. 2BD105 was opened and inspected in detail to verify consistency with design documents. The inspection also verified the overall adequacy of the load path, the mounting of the breakers, and bolting of adjacent sections. The base anchorage configuration results in 2" eccentricity (e.g. induces bolt bending). A detailed evaluation of the anchorage was performed to confirm that the eccentrically loaded bolts were adequate for the SSE.

These panels were screened based on the EPRI screening guidelines and review of original qualification documentation.

Wall mounted distribution panels were also manufactured by B.K Electric (ASCo). 2 predominant configurations are used 29"W x 10"D x 93"H and 29"W x 10"D x 60"H. Panels are either directly bolted to the wall or to 2 unistrut members for the smaller units.

These wall mounted units are very widely represented in the experience data base and were screened-out on the walkdowns.

The AC (208Y/120V) Floor mounted Distribution Panels are manufactured by Cutler Hammer. Typical units consisted of 2 sections, are extensively welded at the base to a transfer channel and top braced. These units are similar in construction and configuration to MCC's.

20Y104 was selected for detailed review and inspection; the inspection confirmed consistency with design documents and provided verification of good overall design for load path and mounting of internal components and

devices.

During the walkdown by the SRT, it was noted that light fixtures supports near 20Y206 and 20Y207 have open S-hooks. Due to the potential interactions if fixtures became loose, these components were not screened out.

The AC distribution panels were screened based on the EPRI walkdown screening guidelines and review of original seismic qualification documentation.

The anchorage was also judged to be adequate for the SSE based on the braced configuration and the extensive weld detail at the base.

The screening criteria in Table 2-4, Appendix A, and Appendix F of EPRI NP-6041, Rev. 1, in conjunction with a walkdown, serves as the basis for the acceptance of the components. On this basis, the components were screened-out and judged to meet the SSE by the SRT.

Elements Not Screened Out

- Systems interaction: Open S-hooks were in the vicinity of 20Y207 and 20Y206 components.

Assessment of Elements Not Screened Out

Open S-hooks were crimped closed.

(13) **Batteries & Racks**

Number of items in Success Path Component List: 12

This category includes batteries and battery racks.

Walkdown Description/Screening Results

The batteries and racks are located in the Control Structure at elevation 217' and 239'. The racks are mounted to the floor.

2DD101 was selected for a detailed walkdown, all other batteries and racks were walked-by. The battery cells were found to be restrained in both horizontal directions by siderails. The racks were of the step tier design. Rack framework completely encased each row of battery cells (i.e., had side rails) and each cell was adequately separated by a spacer (except as described below). Siderails were of rugged seismic design. Racks were braced in both lateral directions between adjacent "bays" of the framing, providing an adequate load path and rigidity to resist lateral loading and to minimize movements. The racks were bolted to the concrete floor. A generic anchor bolt calculation was performed assuming the maximum weight. The anchorage was visually inspected and found to match details shown on design drawings and considered in existing seismic qualification documentation. No concrete condition concerns were identified.

For Racks 2B2D101, 2A2D101 and 1A2D101 it was noted during the walkdown that the spacer pads were loose and a small gap (1/8" - 1/4") existed between the pad and the batteries for the following:

a. Cell 56 and 57 for 2B2D101
b. Cell 55 and 56 for 2A2D101
c. Cell 41 and 42 for 1A2D101

The gaps were judged as being of no consequence for the SSE but the SRT recommended that this be identified as a maintenance item.

Near Rack 2CD101, it was noted that the overhead light fixture had an open "S" hook on one end. If the fixture falls during seismic event, it could impact batteries. Therefore, 2CD101 was not screened-out.

TABLE 3.1.4-2 (cont.)

Other general housekeeping issues were noted. Emergency eye wash stations do not have two way vertical restraint. However, SRT judged that the tank would not cause any interaction concerns. Trash cans are free standing but located at a significant distance from battery racks do not pose an interaction concern.
No other significant seismic interaction problems were identified during the walkdowns.

However, it was noted that many of the battery rooms contain block walls. A generic calculation was performed on block walls throughout the plant and were shown to be acceptable, and therefore, the presence of blockwalls near the battery racks was not considered as an item to preclude screening.

2CD101 was not screened out based on the screening criteria in Table 2-4, Appendix A and Appendix F of EPRI NP-6041, Revision 1. All other batteries and battery racks were screened-out from further review by the SRT.

Elements Not Screened Out

- 2B2D101, 2A2D101, 1A2D101 and 2CD101 were not screened-out as discussed above.

Assessment of Elements Not Screened Out

As a good seismic design practice, it was recommended that a maintenance request be issued for racks 2B2D101, 2A2D101 and 1A2D101 to replace spacer pads with thicker spacer pads to reduce the gap between the batteries identified above. For rack 2CD101, the "S" hooks on the light fixture were crimped closed.

(14) Battery Chargers & Inverters

Number of items in Success Path Component List: 16

This category includes battery chargers and inverters.

Walkdown Description/Screening Results

The battery chargers are located in the Control Structure at elevation 217' and 239'. The inverters are located at elevation 254' of the Control Structure.

Battery Charger 2DD103 was selected for detailed walkdown, all other battery chargers were walked-by. Several of the units were opened to verify mounting of internal devices and transformers. All of the small electrical devices were mounted with a full compliment of hardware.

The walkdown verified that none of the unit transformers were missing hold down bolts.

The battery chargers and inverters are solid state, floor mounted units. Transformers are located near the base of the unit. All cabinets were found to have adequate lateral load paths to the base of the cabinet. There were not significant cut-outs in the cabinets. Cabinet doors were tightly secured by latches or fasteners. The anchorage associated with the battery chargers were determined to be adequate.

Inverter 2AD160 was selected for detail walkdown, all other inverters were walked-by. The anchorage for 2AD160 and 2BD160 was not screened-out due to anchorage issues. Shims were used to elevate the cabinet at all four anchor bolt locations. This can result in bending of anchor bolts under the action of lateral seismic loads.

The SRT noted a potential seismic interaction concern with 2BD160. An unanchored transformer was stored adjacent to inverter and could impact the inverter.

All other inverters were walked-by, and anchorage was judged acceptable by SRT and no other significant seismic interaction concerns were noted.

It was noted during the SRT walkdown that many of the rooms where the

battery chargers and inverters are located contain block walls. A generic block wall calculation was performed by the SRT. The calculation demonstrated that the block walls can be screened-out.

Except as noted above, all other battery chargers and inverters were screened-out from further review based on Table 2-4, Appendix A, and Appendix F of EPRI NP-6041, Rev. 1 and judged to meet the RLE by the SRT.

Elements Not Screened Out

- Inverters 2AD160 and 2BD160 were not screened out due to anchorage issues.
- 2BD160 was not screened out due to spare transformer unanchored.

Assessment of Elements Not Screened Out

For inverters 2AD160 and 2BD160, the SRT performed an engineering evaluation subsequent to the walkdowns. The evaluations demonstrate that the anchorage for the inverters are acceptable for a SSE event. In addition the adjacent spare transformer is judged to represent an insignificant interaction concern for this inverter given its relatively small size and weight. Therefore the inverters were screened out from further review by the SRT.

(15) Engine Generators

Number of items in Success Path Component List: 8

This category includes diesel generators.

Walkdown Description/Screening Results

Diesel Generators are Fairbanks Morse Model 387D8; 3964 HP and are mounted at elevation 214' of the Diesel Generator Building Unit 1 and 2.

2AG501 was selected for detailed walkdown, all others were walked by. The walkdown verified that the engine and generator are attached to a common stiff skid and have an adequate structural assembly for resisting lateral loads. No concerns were identified with respect to potential relative motion of interconnecting fuel, lube oil, and water cooling lines. Appurtenances were verified to be attached with stiff supporting members. The units are not supported on vibration isolators.

Ancillary equipment was verified to be adequately supported and to have adequate flexibility in interconnecting components, such as piping and conduit, to accommodate relative movements.

The anchorage was visually inspected and found to match details shown on design drawings and considered in existing qualification documentation. No concrete condition concerns were identified.

There were no seismic interaction concerns identified. Fire protection piping overhead was reviewed to determine whether it is a normally charged (wet) system. It was found to be normally dry, thereby eliminating any flooding or cascading concerns.

The walkdown identified a general housekeeping concern for the Unit 2 engine-generators. The overhead crane for Unit 2 was found to be parked near the south side of the building. Crane pendant (controller) hangs near panel 2ETB-AG501 which contains relays. Swinging of the pendant could hit the panel and trip relays. A similar situation was noted for all Unit 2 overhead cranes in the Diesel Generator Bays.

For Unit 1, at time of walkdowns, the overhead crane was parked at the north end of the building, therefore it was not an issue. However, there is no procedure for locating (parking) the crane after its use and securing the pendant. Procedures should be established for the location of the crane when it is not in use.

The screening criteria of Table 2-4, in conjunction with a walkdown of the component serves as the basis for the acceptance of the units. On this basis, the units were screened out from further evaluation by the SRT.

Elements Not Screened Out

- None were identified

Assessment of Elements Not Screened Out

N/A

(16) **Instruments on Racks**

Number of items on Success Path Component List: 36

This category consists of instruments on racks.

Walkdown Description/Screening Results

All Instruments racks on the SPCL are located in Reactor Enclosure Building at various elevations, are of similar design and are built by GE. The instruments on racks were walked down by the SRT to assess their seismic capability with an emphasis on component attachment to the rack and rack anchorage.

The instrument racks at Limerick Generating Station are primarily built up of structural angles welded together with adequate cross bracing. The instrument racks are secured to the floor using concrete expansion anchor bolts. The walkdown verified that the device mountings were seismically rugged and properly secured to the rack. Anchorages were verified to match the generic details specified on design drawings and considered in existing qualification documentation. No concrete condition concerns were identified.

There were no seismic spatial interaction concerns identified during the walkdowns. For many of the instruments, the related tubing was traced back to the root valve or the nearest penetration to verify no interaction concerns. Attached lines were well supported and had adequate flexibility to accommodate relative movements.

Overhead fire protection piping was identified in several areas and contained threaded joints. However, it was verified that the system is normally dry and there is no concern for flooding or cascading sources on the instrument on racks.

The walkdowns had identified some general housekeeping concerns in the vicinity of the Unit 2 instrument racks 20C019, 20C016, 20C035 and 20C038. Most concerns were judged to be related to ongoing outage activities and were not tracked as outliers. Also, the overhead lifting device (monorail and controller) were unrestrained and could cause some minor impact. Monorail control and monorail hoist should be parked away from racks. This was noted for 20C016 and 20C019.

TABLE 3.1.4-2 (cont.)

Based on the results of the walkdown, and screening criteria in Table 2-4, Appendix A, and Appendix F of EPRI NP-6041, Rev. 1, the Instruments on Racks were screened out from further evaluation by the SRT.

Elements Not Screened Out

For 2OC016 and 2OC019, the unrestrained monorail and controller are tracked as housekeeping concerns.

Assessment of Elements Not Screened Out (N/A)

Monorail control and hoist will be parked away from racks and secured when not in use.

TABLE 3.1.4-2 (cont.)

(17) **Temperature Sensors/Local Instruments (Not on Racks)**

Number of components on Success Path Component List: 149

This category includes temperature sensors and other local instrumentation.

Walkdown Description/Screening Results

A. Temperature Sensors

Temperature elements TE-41-101A through 101H and TE-41-103A through 103H are located in the suppression pool for Unit 1. Similar temperature elements 201A, etc. are located in the suppression pool for Unit 2.

A detailed walkdown was performed for the Unit 1 suppression pool. A walkdown was not performed for the Unit 2 suppression pool.

The walkdown utilized existing documentation (e.g. component drawings, structural drawings for anchorage, and qualification reports) since the majority of the temperature elements are under water. The suppression pool temperature elements are adequately supported and encapsulated in protective tube throughout entire length. No seismic interaction concerns were observed. The temperature elements were qualified to IEEE 344-1975.

All other temperature sensors were walked by and were judged to be adequately supported and no seismic interaction concerns were identified.

Based on the results of the walkdown, and screening criteria in Table 2-4, Appendix A, and Appendix F of EPRI NP-6041-SL, Rev. 1, the temperature sensors were screened out from further evaluation by the SRT.

B. Local Instruments

For Limerick Unit 2, local instruments such as level switches, pressure transmitters, and level indicators which were not rack mounted were walked down by the SRT. A walk-by of local instruments not rack mounted was performed for Limerick Unit 1

since the SRT judged that instrumentation and their supports were similar in design and location.

Local instrumentation is typically wall mounted to channels or floor mounted to tube steel which has adequate anchorage The walkdowns utilized existing documentation (e.g. structural drawings for anchorage and equipment qualification reports). All instrumentation is of good seismic design and the devices did not contain soft targets. All attached lines were well supported and had adequate flexibility to accommodate relative movement. There were no seismic spatial interaction concerns identified during the walkdown.

Overhead fire protection piping was identified in several areas. However, it was verified that the system is normally dry and there is no concern for flooding or spilling onto instruments.

The walkdowns identified general housekeeping issues in the vicinity of the Unit 2 instruments for PT-55-2N008, PT-55-2N051 and PSL-12-202B. Most concerns were judged to be related to ongoing refueling outage activities.

Based on the results of the walkdown, equipment qualification documentation and screening criteria in Table 2-4, Appendix A, and Appendix F of EPRI NP-6041-SL, Rev. 1, the local instruments were screened out from further review by the SRT.

Elements Not Screened Out

- None were identified

Assessment of Elements Not Screened Out

N/A

TABLE 3.1.4-2 (cont.)

(18) **Control Panels & Cabinets**

Number of Components in Success Path Component List: 161

This category includes control panels and cabinets.

Walkdown Description/Screening Results

A walkdown of 160 of the 161 control and instrumentation panels on the SPCL was conducted to assess seismic adequacy. The main purpose of the walkdown was to verify that internal devices were properly mounted and secured, were not excessively flexible, appeared seismically rugged, and had an adequate load path to the cabinet structural frame and anchorage. In addition, the SRT Team verified that cabinets were not excessively flexible, had no large cutouts in the lower half, were of similar configuration to NEMA standards, drawers and equipment on slides were restrained from falling out, and doors were secured by latches or fasteners. To that end, a representative sample of the control and instrumentation panels on the SPCL were opened for visual inspection of internal device mounting, structural adequacy and anchorage.

The cabinets found in the LGS Units 1 and 2 can be grouped into three categories, Hoffman type panels located throughout the plant, vertical boards and console panels located in the control room and auxiliary equipment room, and HVAC enclosure vertical panels, and similar type control board panels, located primarily in the diesel generator areas.

A. Hoffman Type Panels

The Hoffman Type control and instrumentation panels are flush mounted or attached to unistrut channels which in turn are mounted to concrete and masonry block walls with expansion anchors or through bolts. In some instances they are mounted to structural support steel. The panel sizes vary in dimension with the maximum size being in the 36"W x 42"H x 22"D range. 0AC564, 1AD106, 10TB-053, 2DD104, 2BD106 and 2BD104 panels are representative at LGS Unit 1 and 2 plants.

These panels are used as junction/termination boxes and enclosures for devices such as relays, fuses, switches, indicator lights or instruments and cable connectors.

The SRT, prior to the walkdowns, reviewed applicable dynamic qualification of Hoffman type panels and anchorage design of civil calculation 101.49 to become familiar with the structural adequacy of the panels.

The anchorage of the panels and the attachment of the various devices were visually inspected and found structurally acceptable since they match details shown on design drawings and existing qualification documentation. There were no seismic interaction concerns identified.

Based on the results of the walkdowns, coupled with a review of existing qualification documentation and Table 2-4, Appendix A, and Appendix F of EPRI NP-6041-SL, Rev. 1, the Hoffman type panels were screened-out from further review by the SRT.

B. Vertical Board and Console Panels

The majority of the vertical board panels are located in the control room and auxiliary equipment room at elevations 269' and 289' respectively. The panels are manufactured by General Electric and Rockwell. The width of the panels varies from 30" to 36", the height is 90" and the length varies from 24" to 158". Typically several panels are located next to each other to form a large panel section. When this situation occurs the individual panels are bolted to each other along the vertical direction at each corner.

The console panels are located in the control room and are manufactured by General Electric. The width and height of the console panel is 72" and 61" respectively, and the length varies from 29" to a maximum of 168". Panels are bolted to each other when they are located next to each other in a similar fashion to the vertical boards.

The vertical board and console panels contain control and instrumentation devices such as switches, pushbuttons, panel lights, recorders, relays, controllers, solid state circuit boards, power supplies, wiring and terminal blocks and are manufactured with heavy gauge sheet metal and structural framing members which provide a good load path to the foundation.

20C626, 0BC667, 0DC667, 2AC696, 20C201, 20C609 and 20C788 were selected for detailed walkdown, all other panels were walked-by. These panels were opened and inspected in detail to verify consistency with design documents. The inspection also verified that adjacent panels were bolted together, and that electrical devices were in place and properly secured on front plates which have intermediate stiffeners. Flexibility of base members and the adequacy of the load path were reviewed and judged to be acceptable. There were no excessive cutouts.

The anchorage was visually inspected and found to match details shown on design drawings and judged to be acceptable for the RLE.

The walkdown identified a general housekeeping concern with respect to systems interaction effects for the control room panels. The items of concern were free standing equipment and personnel lockers with storage boxes on top that were located in the rear, side or a few feet in front of the panels. Panels 10C626 and 2AC696 were used to track all the housekeeping issues of the control room. In addition, remote shutdown vertical board panels 10C201 and 20C201 which are located in Room 540 are identified as having interaction concerns with tech spec binders and a safe shutdown equipment box which is heavy and not secured and could impact a panel during a seismic event.

Based on the results of the walkdowns, coupled with a review of existing qualification documentation and Table 2-4, Appendix A, and Appendix F of EPRI NP-6041, Rev. 1, the vertical boards and console panels were screened-out from further review by the SRT except where noted.

C. Diesel Generator HVAC and Instrument Control Panels

The Diesel Generator Enclosure HVAC control panels are manufactured by MCC Powers, are walk through type and contain relays. Their size is 72"H x 48"W x 42"D and are floor mounted.

The Diesel Generator instrument control panels are adjacent to the generator transformer and excitation panels and are manufactured by Allied and Besler Electric respectively. Their size is 90"H x 72"W x 48"D, are floor mounted and contain transformers, rectifiers, control

and power chassis, and other electrical devices.

The SRT, prior to walkdowns, reviewed applicable dynamic qualification of MCC Powers panels (QAP 569, Report No. A-421-81-02) to become familiar with the structural adequacy of the panels.

2AC563, 2AC514 and 2AG502 were selected for detailed walkdown, all other panels were walked-by. These panels were opened and inspected in detail to verify consistency with design documents. The inspection also verified that adjacent panels were bolted together, and that electrical devices (such as relays, switches, terminal blocks, etc.) were in place and properly secured. Flexibility of base members and the adequacy of the load path were reviewed and judged to be acceptable. There were no excessive cutouts..

The anchorage was judged to be adequate for the RLE, based on the review of the details shown on design drawings. The panels were either bolted to structural steel or welded to embedded plates on the floor.

The walkdown identified panel 2AC563 as requiring maintenance action due to the light mounting inside the panel being broken and hanging by 1 screw. In addition, a housekeeping concern was identified for panel 2AG502 with the overhead crane controller not being tied down when the crane is not used.

Based on the results of the walkdowns, coupled with a review of existing qualification documentation and Table 2-4, Appendix A, and Appendix F of EPRI NP-6041-SL, Rev. 1, the diesel generator HVAC and instrument control panels were screened-out from further review by the SRT except as noted.

Elements Not Screened Out

- Systems Interaction: The noted housekeeping concerns were tracked as outliers on the following components: 10C626, 2AC696, 20C201 and 10C201.

TABLE 3.1.4-2 (cont.)

Component 2AC563 requires a maintenance request.

Component 2AG502 could be impacted by overhead crane controller not being tied down.

Assessment of Elements Not Screened Out

- Housekeeping issues need to be addressed procedurally during normal plant operation and where required during outages.
- For component 2AC563, broken light mounting was repaired.
- Diesel generator overhead crane needs to be parked at north end of building when not in use. Controller should be secured when not in use.

(19) **Vertical Tanks or Heat Exchangers**

Number of Components in Success Path Component List: 42

This category includes vertical tanks, heat exchangers and drain pots.

Walkdown Description/Screening Results

A. Diesel Generator Starting Air Reservoirs

The diesel generator starting air reservoirs are mounted at elevation 217' of the Diesel Generator Building for Units 1 and 2. There are two starting air reservoirs per diesel unit. The air reservoir is a 37 cubic foot vertical tank supported at the base with four 3/4" diameter cast-in-place anchor bolts.

2A1T558 was selected for detail walkdown, all other components were walked-by. The plant walkdown utilized existing design documentation (vessel drawings, structural drawings for anchorage, and qualification reports).

During the walkdown for Limerick Unit 2 it was noted that some bolts lacked full thread engagement and nuts have been plug welded to threaded bolts. This occurred on 2B1T558, 2D1T558 and 2D2T558. The SRT reviewed documentation that was performed by Bechtel during the Construction Phase of Limerick. Field Change Request (FCR) C-7032F was reviewed. The FCR defines the basis for acceptance of lack of thread engagement and provides the justification for the dispositioning of the lack of thread engagement. The SRT agrees that the lack of tread engagement has been adequately dispositioned. Therefore, the anchorages were verified to match the design details considered in existing qualification documentation and the anchorages appeared to be stiff with no excessive prying.

There were no seismic interaction concerns for Limerick Unit 1. The attached safety valve lines are flexible and are judged to be acceptable. Based on the screening criteria in Table 2-4, Appendix A, and Appendix F of EPRI NP-6041, Rev. 1, all of the Unit 1 air reservoirs were screened-out from further review by the SRT. However, for Limerick Unit 2, the SRT noted that the safety relief

TABLE 3.1.4-2 (cont.)

valve lines near the top of the tank are supported at the wall near the valve and there is little flexibility. If the tank has excessive displacement, the rigid support could cause the line to fail. This condition was noted for all air reservoir tanks in Unit 2, so the air reservoirs were not screened-out.

B. Diesel Generator Day Tank

The diesel generator day tanks are mounted at elevation 220' of the Diesel Generator Building for Units 1 and 2. The tank is Vertical and supported at the base with four 1 inch diameter cast-in-place anchor bolts. The tank has a lateral brace located approximately at two-thirds of height attached to concrete wall.

2AT528 was selected for detail walkdown, all other day tanks were walked-by. Tanks 2AT528, 2BT528, 2CT528 and 2DT528 anchorage typically were missing one to one and a half thread on nut engagement. This was judged to be acceptable by SRT since tank has top lateral support.

The plant walkdown included a review of documentation. The walkdown was primarily a verification that the as-installed configuration matched design documents. The overall appearance of the day tanks, including connections of the vessel to the supporting members, was reviewed for weak links. All attached piping and electrical conduit were judged to be flexible. All rule of the box components were adequately attached to the tank.

Based on the screening criteria contained in Table 2-4, Appendix A and Appendix F of EPRI NP-6041, Rev. 1, the day tanks were screened-out from further review by the SRT.

There were no seismic interaction concerns identified during the walkdown.

During the walkdown of Limerick Unit 1 and 2, it was noted that the rooms housing the diesel Generator day tanks also contain lube oil storage tanks. Although the lube oil tanks are adequately supported, the SRT is concerned that these tanks contain a level gage on the side of the tank. The level gage is glass, however the vessel has isolation valves which will limit spillage if the glass fails. This issue

is addressed in the Fire portion of the IPEEE.

C. Residual Heat Removal Heat Exchangers

The Residual Heat Removal Heat Exchangers are mounted at elevation 177' of the Reactor Enclosure for Limerick Unit 1 and 2. 2AE205 was selected for detail walkdown, all other heat exchangers were walked-by. The plant walkdown utilized design documentation (e.g., heat exchanger detail drawings, structural drawings for anchorage, and qualification reports) and was primarily a verification that the as-installed configuration matched design documents. The overall appearance of the heat exchangers, including connections of the vessel to the supporting members was reviewed for obvious weak links.

Anchorages were verified to match the design details in existing qualification documentation. No concerns were identified with respect to anchorage. Anchorage appeared to be stiff with no excessive prying.

There were no seismic interaction concerns identified during the walkdown.

It was noted the 1AE205 and 1BE205 for Limerick Unit 1 were replaced during the 1994 outage.

Based on the screening criteria in Table 2-4, Appendix A, and Appendix F of EPRI NP-6041, Rev. 1, all of the heat exchangers were screened-out from further review by the SRT.

D. PCIG/ADS Nitrogen Bottles

The PCIG/ADS Nitrogen Bottles are located at elevation 217' of the Reactor Enclosure for Limerick Units 1 and 2. 2BS252 which includes 2BS252-1, 2BS252-2, 2BS252-3 and PCV-59-252B-1, 2 and 3, was selected for a detailed walkdown, all others were walked-by.

The nitrogen bottles are located in structural housing. The typical detail for the structure is given on drawing C-869, Sheet 2, Type IV. The nitrogen bottles are well supported top and bottom. Some bolts on top bracket were missing but the bracket was judged to be

acceptable by the SRT.

Anchorages were verified to match the design details in existing qualification documentation, tubing was flexible and well supported. No concerns were identified. There were no seismic interactions concerns identified during the walkdown.

The walkdown of Limerick Unit 2 occurred during a plant refueling outage. Portable tool boxes and miscellaneous equipment and tools were noted in the area to support outage activities. This was judged to be acceptable by SRT since the equipment was required to support the outage.

A walkdown of Limerick Unit 1 by the SRT during non-outage conditions verified that housekeeping, in general, is acceptable in similar areas.

Based on the screening criteria in Table 2-4, Appendix A, and Appendix F of EPRI NP-6041, Rev. 1 all of the PCIG/ADS Nitrogen Bottles were screened-out from further review by the SRT.

E. Drain Pots - Diesel Generator Area

Drain pots were identified on the SPCL for LGS Units 1 and 2 at elevation 217' of the Diesel Generator Building. The pots are vertical 8" diameter pipes approximately 3' high attached to the floor with 4 - ½" diameter bolts. These are passive components and were judged by the SRT to be acceptable.

Elements Not Screened Out

The Diesel Generator Starting Air Reservoirs 2A1T558, 2A2T558, 2B1T558, 2B2T558, 2C1T558, 2C2T558, 2D1T558 and 2D2T558 are all considered outliers due to the attached safety valve line not being flexible.

Assessment of Elements Not Screened Out

Lube oil tank spillage is addressed in the Fire section of the IPEEE report.

TABLE 3.1.4-2 (cont.)

A engineering evaluation from the starting air reservoirs was performed by the SRT subsequent to the seismic walkdowns to determine relative displacement between the top of the tank and the safety valve support. The evaluation shown that the maximum relative displacement is less than 1/16" during an SSE. There is adequate flexibility between the tank and safety valve to accommodate this movement and the SRT has judged that the starting air reservoirs are screened-out from further review.

(20) **Horizontal Tanks or Heat Exchangers**

Number of Components in Success Path Component List: 40

This category includes horizontal tanks and heat exchangers.

Walkdown Description/Screening Results

A. Diesel Generator Exhaust Silencer

The diesel Generator Exhaust Silencers are mounted at elevation 217' of the Diesel Generator Building for Units 1 and 2.

AS575 was selected for detail walkdown, all other components were walked-by. The seismic walkdown utilized existing design documentation (e.g. component drawings, structural drawings for anchorage, and qualification reports).

The anchorage was visually inspected and found to match details shown on design drawings and judged to be acceptable. All intake and exhaust ducting is well supported and had adequate flexibility. There were no seismic interaction concerns identified.

Based on the results of the walkdown, coupled with a review of existing qualification documentation and Table 2-4 of EPRI NP-6041, Rev. 1, all diesel generator exhaust silencers were screened-out from further review by the SRT.

B. Diesel Generator Jacket Water Expansion Tanks

The Diesel Generator Jacket Water Expansion Tanks are mounted at elevation 234' of the Diesel Generator Building for Units 1 and 2.

2AT564 was selected for detail walkdown, all others were walked-by. The seismic walkdown utilized existing design documentation (e.g. component drawings, structural drawings for anchorage and qualification reports).

The diesel Jacket water expansion tanks are mounted to the platform at elevation 234'. Eight, 3/4" diameter anchor bolts are used to attach the tank to the platform steel.

Attached piping is well supported and had adequate flexibility. All electrical cable and conduit are well supported and have adequate flexibility. Level sight glass is eleven inches long and is supported by tube steel frame back to platform. The tank supports are structurally good. Sight glass is a soft target and small relative displacements of attached piping could cause failure. However, lower valve on sight glass piping has an automatic ball check valve and will prevent spillage if sight glass fails. All other Limerick Unit 2 diesel water expansion tanks and supports are similar.

The Limerick Unit 1, the diesel water tanks are mounted to the platform at elevation 234' with eight, 3/4" diameter bolts. The tanks and supports are structurally good. The sight glass for the Limerick Unit 1 diesel water tanks are independently supported at the wall at the same elevation as the tank. The SRT believes that this design will lead to excessive relative displacements during a seismic event and potentially cause failure of the sight glass. However, lower valve on sight glass piping has an automatic ball check valve and will prevent spillage if sight glass fails. Bechtel calculation C-G-007-2, Rev. 1, for the Supplemental Qualification for the Jacket water expansion tank qualifies the water tank. There were no other seismic interaction concerns identified during the walkdown.

C. HPCI/RCIC Turbine Barometric Condenser.

The HPCI and RCIC Turbine Barometric condensers are mounted at elevation 177' in the Reactor Enclosure.

20E209, RCIC turbine barometric condenser and 20E210 HPCI turbine barometric condenser were selected for detailed walkdown, all other units were walked-by.

The anchorage was visually inspected and judged to be acceptable by the SRT. There were no seismic interaction concerns identified.

Based on the results of the walkdown, and the screening criteria of Table 2-4, Appendix A, and Appendix F of EPRI NP-6041, Rev. 1, the units were screened-out from further review by the SRT.

D. Drain Pots

Drain Pots are mounted at elevation 177' of the Reactor Enclosure for each unit. These are passive components, in-line with the piping system. All drain pots were walked-down by the SRT for Limerick Unit 2 and a walk-by was performed for Limerick Unit 1.

Based on the results of walkdown, and Table 2-4, Appendix A, and Appendix F of EPRI NP-6041, Rev. 1, all drain pots were screened-out from further review by the SRT.

E. Nuclear Boiler Vessel Condensing Chamber

The vessel is an in-line piping component. A walk-by was performed for Limerick Unit 1 component XY-42-1D002 and judged acceptable by SRT. The Limerick Unit 2 component could not be located and a detailed document review was performed by the SRT. The SRT believes that the good correlation between drawings and as built conditions found throughout the plant and also because of the well designed piping systems observed, it is concluded that there would not be any interaction effects which would fail the pressure boundary maintained by the condenser chamber.

Therefore, all condensing chambers were screened-out from further review by the SRT.

F. Main Steam Relief Valve (MSRV) Accumulator Tanks.

The Main Steam Relief Valve Accumulator Tanks are located in the Primary Containment Reactor enclosure at elevation 286'.

2ET003 was selected for detailed walkdown, all other accumulator tanks were walked-by.

The accumulator tanks are 35" long and are mounted to clamp beams (saddles) by U-bolts. The clamp beams are fillet welded to the containment platform beams. Anchorage was judged acceptable by the SRT based on the walkdown and review of the design basis documents. Air lines are well supported and have adequate flexibility.

TABLE 3.1.4-2 (cont.)

Based on the screening criteria in Table 2-4, Appendix A, and Appendix F of EPRI NP-6041, Revision 1, the accumulators were screened out from further review by the SRT.

Elements Not Screened Out

- None were identified

Assessment of Elements Not Screened Out

N/A

(21) **Horizontal Suspended Tank**

Walkdown Description/Screening Results

There are no horizontal suspended tanks on the SPCL for LGS Unit 1 and Unit 2.

Elements Not Screened Out

- None were identified

Assessment of Elements Not Screened Out

N/A

(22) **Buried Tanks**

Number of Components in Success Path Component List: 8

This category includes buried tanks.

Walkdown Description/Screening Results

All tanks on the SPCL, except for those underground, were walked down by the SRT.

The Diesel Generator Diesel Oil Storage Tanks are buried tanks. Access into the valve pit requires special authorization since this is a "Confined Workspace", therefore the valve pit was not walked down. A detailed review of the design basis documentation was performed by the SRT. A general area review walkdown was performed to verify that there is no potential for interactions that could negatively affect the tanks or connections. There are no overhead systems or equipment that might impact tanks. The SRT believes that the walkdowns performed to date show a strong correlation between "as-built" and "as-design" for LGS Units 1 and 2. Dwgs. C-1063, Rev. 10 and C-1064, Rev. 15 show the general layout for the tanks and valve pits. The tanks rest on two saddles. Anchor straps are used to tie down the tanks. The saddles rest on a reinforced concrete base mat. The analysis for the slab, saddle, tie down straps, and valve pit and hatch covers are contained in Bechtel calc. 47.10A, Rev. 2. The hatch covers are adequately attached to the reinforced concrete box

TABLE 3.1.4-2 (cont.)

structures. Tank analysis is given in Bechtel calc. EQG D-209-1. Qualification Report D-209 demonstrates acceptability of tank.

Based on the results of the general area walkdown, detailed review of existing documentation, and screening criteria in Table 2-4, Appendix A, and Appendix F of EPRI NP-6041, Rev. 1, the buried tanks were screened-out from further review by the SRT.

Elements Not Screened Out

- None were identified

Assessments of Elements Not Screened Out

N/A

(23) **Conduit and Cable Tray Raceways**

Raceway reviews are documented in Section 3.1.4.4.

(24) **Piping**

Piping reviews are documented in Section 3.1.4.4.

(25) **NSSS Components & Primary Loop**

Reviews for NSSS items are documented in Section 3.1.4.5.

(26) **Traveling Screens & Sluice Gates**

Walkdown Description/Screening Results

There are no traveling screens and sluice gates on the SPCL for LGS Unit 1 and Unit 2.

Elements Not Screened Out

- None were identified

Assessments of Elements Not Screened Out

N/A

(27) **Control Rod Drive Assemblies**

Reviews for the CRDs and HCUs are documented in Section 3.1.4.5.

(28) **Building Seismic Gaps**

Building seismic gaps were reviewed with structures and are documented in Section 3.1.4.3.

(29) **Control Room Ceiling**

The control room ceiling review is discussed in Section 3.1.4.5.

(30) **Automatic Transfer Switches**

Number of Components in Success Path Component List: 4

This category includes automatic transfer switches.

Walkdown Description/Screening Results

All the automatic transfer switches (ATS) on the SPCL were walked down by the SRT. They are located at El. 254' in rooms 452 and 453 of the Control Structure.

The ATS panels are attached to vertical unistrut members which are welded to base plates that are anchored to the concrete floor with expansion anchors. The unistrut members are laterally supported by the respective inverters for the ATS. ATS 20NAD160 and 20NBD160 are not screened out since they interact with the inverters and inverters 2AD160 and 2BD160 are not screened out due to anchorage concerns. Refer to the Battery Chargers and Inverters equipment class summary for additional information on the inverters.

Except as noted above, all other ATS were screened-out from further review based on the walkdown, Table 2-4, Appendix A, and Appendix F of EPRI NP-6041-SL, Rev. 1 and judged to be acceptable by the SRT.

Elements Not Screened Out

Automatic Transfer Switches 20NAD160 and 20NBD160 are laterally supported to inverters 2AD160 and 2BD160 respectively, which are not screened out due to anchorage concerns. Therefore, the switches are also tracked as not screened out.

Assessment of Elements Not Screened Out

An Engineering evaluation was performed by the SRT subsequent to the seismic walkdowns which qualifies the inverters. Therefore, the automatic transfer switches are screened out from further review and are acceptable

for a SSE event.

(31) **Wall Mounted Contactor, Transmitter, Power Supply, etc.**

Walkdown Description/Screening Results

Elements Not Screened Out

Assessment of Elements Not Screened Out

N/A

(32) **Strainers and Filters**

Number of Components in Success Path Component List: 40

This category includes strainers and filters

Walkdown Description/Screening Results

A. Strainers

The RHR, HPCI and RCIC suppression pool suction strainers are under water in the suppression pool and therefore were not inspected. A walkdown of the Unit 1 suppression pool was performed to review and evaluate potential systems interaction effects on the strainers from sources above the water surface. No interaction concerns were identified as the result of the walkdown.

The SRT prior to the walkdown reviewed in detail applicable dynamic qualification packages, and design drawings, to become familiar with the structural adequacy of the strainers. The SRT believes that the walkdowns performed show a strong correlation between "as-built" and "as-design" for LGS Units 1 and 2. The strainers are located on 24", 16" and 6" diameter pipes at tee connections that are supported from the suppression pool; the strainers are bolted to the tee via flange connections.

Based on the results of the suppression pool walkdown, a detailed review of existing documentation, and screening criteria in Table 2-4, Appendix A, and Appendix F of EPRI NP-6041-SL, Rev. 1, the

strainers were screened-out from further review by the SRT.

B. Filters

All Filters on the SPCL were walked down by the SRT.

The Diesel Engine Inlet Air Filters are contained within tanks which in turn are anchored with 4 - 1" diameter bolts to concrete. Tank details are shown on Drawing 8031-M-71-195-6 and anchor bolt details on Drawings C-664 and C-615 Sheet 1. The Fuel Filters are inline components on 2" diameter pipes which are very well supported under the grating in the diesel generator building.

The anchorage was visually inspected and found to match details shown on design drawings. There were no seismic interaction concerns identified.

The screening criteria in Table 2-4, Appendix A, and Appendix F of EPRI NP-6041-SL, Rev. 1 in conjunction with the walkdown serves as the basis for the acceptance of the filters. On this basis, the filters were screened-out from further review by the SRT.

Elements Not Screened Out

- None were identified

Assessment of Elements Not Screened Out

N/A

(33) **Building Penetrations of Underground Utilities**

Penetrations are discussed in Section 3.1.4.3.

TABLE 3.1.4-3

HOUSEKEEPING AND MAINTENANCE CONCERNS

Equipment Type	Room/Area	Description of Concern
Medium Voltage SWGR	428, 429, 430,431	Housekeeping Concern - Unrestrained breakers - Open S-Hooks - Free Standing Ladder
Low Voltage SWGR	638, 475, 402, 506, 602	Unrestrained lifting device
Motor Control Centers	370, 508, 304, 506	General Housekeeping
Main Control Boards 10C626 & 2AC696	Control Room	General Housekeeping
Distribution Panels 20Y207 & 20Y206	619E, 625	- Open S-Hooks
Batteries and Racks 2CD101	508	Open S-Hooks
Engine Generator 2AG502 (Typical)	Diesel Gen. Bldg. 311A-311D 315A-315D	- Overhead crane controller not tied down - (screened but being tracked)
Instruments on Racks 20C016 and 20C019	370E, 370W	N2 Bottle, I&C Trolley, Monorail controller
Control Panels & Cabinets 10C201, 20C201	542	- General housekeeping and Open S-Hooks -Outlier due to Tech Spec/SSE Box free.
Chillers 0AK112, 0BK112	258, 263	N2 Bottles (screened but tracked as outage issue)
Batteries - 2B2D101, 2A2D101, 1A2D101	426, 427, 436	Gaps Between Spacers
Vert Tanks / Heat Exchangers 2BS252 (Typ)	370E	Missing & loose bolts on N2 bottle stand. (screened but being tracked)

3.1.4.3 Structures

This section documents the assessment and walkdown of the LGS structures, performed in March of 1993.

The seismic Category I structures that are considered in the IPEEE assessment of the Limerick Generating Station include the Primary Containment and Internal Structures, the Secondary Containment (Reactor enclosures and refueling area), the Control structure, the Diesel Generator enclosure, the Spray Pond Pump structure, the Spray Pond and several miscellaneous structures detailed later in this section. These structures house or support Category I equipment and, therefore, maintaining their structural integrity is considered essential for the ability to safely shut down the plant in the event of a design basis earthquake.

3.1.4.3.1 Seismic Loads

As discussed earlier in 3.1.1.2, the design basis ground response spectrum (5% damped) for the LGS safety-related structures is shown in Figures 3.1.1-1 and 3.1.1-2 compared with the 84% NUREG/CR 0098 response spectrum. It can be seen that the design basis response spectrum is essentially equal to the NUREG/CR 0098 and enhanced in the 8 Hz to the 10 Hz range. The Safe Shutdown Earthquake (SSE) design ground response spectra is anchored to maximum horizontal and vertical ground accelerations equal to 0.15g and 0.10g, respectively.

Seismic loads for Seismic Category II structures are based on the UBC 1970 Edition Seismic Zone 1 requirements. Although the Turbine Enclosure is a Seismic Category II structure, certain portions of this structure were designed to withstand the SSE without exceeding yield strength; e.g., essentially as a Seismic Category I structure.

Furthermore, structural separations were designed to assure that interaction between Seismic Category I and Non-Seismic Category I structures does not occur. Non-Seismic structures, such as the Turbine Enclosure, were designed in such a way that movements or deflections of a Non-Seismic Category I building due to design, environmental or accident conditions do not endanger the function of any Seismic Category I building.

3.1.4.3.2 Load Combinations

Seismic Category I structures were designed for load combinations of gravity, thermal, seismic and accident loads and were generally proportioned to maintain elastic behavior. Tables 3.8-8 through 3.8-10 of the

UFSAR describe the load combinations and acceptance criteria used for the Primary Containment, the Reactor Enclosure and Control Structure and the other Category I structures, respectively.

3.1.4.3.3 Governing Codes

The LGS civil structures were designed to the following Codes:

American Institute of Steel Construction (AISC), "Specification for the Design, fabrication and Erection of Structural Steel for Buildings", dated February 12, 1969, Supplement No. 1 (AISC, dated November 1, 1970), Supplement 2 (AISC, dated December 8, 1971), Supplement 3 (AISC, dated June 12, 1974), depending on the date fabrication or construction was done.

American Concrete Institute (ACI), "Building Code Requirements for Reinforced Concrete" (ACI-318-63). All structures and portions of structures designed after June 1972 conform to ACI 318-71. All Seismic Class I structures and components have been designed to ACI 318-71.

In accordance with the screening guidelines in EPRI NP-6041 Table 2-3, these structures can be screened-out with a HCLPF capacity of 0.3g pga without any seismic margin evaluation if the design considers an SSE of 0.10g or greater. The screening basis for LGS structures is summarized in Table 3.1.4-4. On this basis, the Category I structures identified above at LGS are considered seismically rugged and were screened-out from further review by the SRT. Thus, the drawing review and walkdown are geared to confirm good seismic detailing and to identify areas that could pose a seismic concern (if any) and that may not be considered in the original design.

A brief general description of the seismic Category I structures included in the SMA follows.

TABLE 3.1.4-4

SCREENING BASIS FOR CIVIL STRUCTURES

Type of Structure	Basis for Screening
Concrete Containment	Generally has HCLPF>>0.3g. Therefore, it can be screened out at the 0.3g RLE. Drawing review and confirmatory walkdown.
Containment Internal Structure	Designed for SSE of 0.15g>0.1g; thus, no detailed evaluation is required for 0.3g RLE. Drawing review and confirmatory walkdown.
Shear walls, footings and shield walls	Designed for SSE of 0.15g>0.1g; thus, no detailed evaluation is required for 0.3g RLE. Drawing review and confirmatory walkdown.
Diaphragms	Designed for SSE of 0.15g>0.1g; thus, no detailed evaluation is required for 0.3g RLE. Drawing review and confirmatory walkdown.
Category I concrete frame structures	Designed for SSE of 0.15g>0.1g; thus, no detailed evaluation is required for 0.3g RLE. Drawing review and confirmatory walkdown.
Category I steel frame structures	Designed for SSE of 0.15g>0.1g; thus, no detailed evaluation is required for 0.3g RLE. Drawing review and confirmatory walkdown.
Impact between structures	Considered not to be a real problem at the 0.3g level earthquake. No evaluation required for 0.3g RLE. Drawing review and confirm gap sizes during walkdown.
Category II structures with safety-related equipment or with potential to fail Category I structures (Turbine Enclosure).	Evaluation not required provided the structure is capable of meeting the 1985 UBC Zone 4 requirements. Not applicable for LGS.

3.1.4.3.4 Primary Containment

The primary containment is in the form of a truncated cone over a cylindrical section, with the drywell being the upper conical section and the suppression chamber being the lower cylindrical section. These two sections comprise a structurally integral, reinforced concrete pressure vessel, lined with welded steel plates and provided with a steel domed head for closure at the top of the drywell. The diaphragm slab is a reinforced concrete slab structurally connected to the containment wall. The primary containment is divided by a horizontal diaphragm slab into two major regions: the drywell and the suppression chamber. The drywell encloses the reactor vessel, reactor recirculation system and associated piping and valves. The suppression chamber stores a large volume of water.
The primary containment is structurally separated from the surrounding reactor enclosure.

In accordance with Table 2-3 of EPRI NP-6041, concrete containment structures generally have a capacity in excess of 0.3g pga due to the heavy wall construction and heavy reinforcement designed to resist extreme accident loads such as LOCA. Therefore, the LGS Primary Containment is considered seismically rugged and is screened-out from any further review by the SRT.

In general, the design of the containment internal structures considers the effects of all appropriate loading conditions with the major/significant load contributions coming from the design basis accident pressure, accident temperature gradients, or missile/pipe rupture loadings. The dynamic effects of seismic loads are appropriately addressed and included in the containment internal structures design, however, the contributions from such loadings are typically minimal and overshadowed by the more significant contributions from the severe accident loadings mentioned above.

In accordance with Table 2-3 of EPRI NP-6041, containment internal structures can be screened-out from further review provided they are designed for an SSE of 0.1g or greater. The containment internal structures are essentially designed for an SSE of 0.15g. Therefore, the LGS containment internal structures are considered seismically rugged and screened out from further review by the SRT. Furthermore, since the structure was designed by dynamic analysis using the provisions of ACI 318-71 (ref. 3.1-5), it could be screened to an even higher level of 0.5g pga in accordance with EPRI NP-6041 Table 2-3 screening guidelines.

Section 3.1.5 describes the Containment Performance review performed as part of the IPEEE EPRI SMA.

3.1.4.3.5 Miscellaneous Structures

Subgrade pits, manholes and tunnels which contain safety related components are constructed of reinforced concrete.

Safety related piping, tanks and electrical ducts which are not located inside structures, are buried underground with adequate cover for missile protection. Additionally, soil erosion due to failure of non-seismic piping has also been considered. The integrity of safety related seismic Category I buried pipe will not be impaired through soil erosion by a failure of one buried non-seismic Category I pipe.
In addition to the screening reviews based on Table 2-3 of EPRI NP-6041, drawing reviews and confirmatory walkdowns were performed and are summarized as follows:

3.1.4.3.6 Drawing Review

The drawing review comprised of a qualitative review of the structure's seismic load paths and the distribution of the seismic load through each path. In some cases, results from previous seismic evaluations were used to confirm load paths and identify elements with the lowest seismic capacity. The review paid particular attention to those lowest seismic capacity elements. Reinforcing detailing was reviewed; in particular the connection details between the shear walls and floor slabs and between shear walls and their foundation base. These areas were reviewed for general adequacy and to ensure ductility.

As described earlier, the structural system for the LGS structures consists of reinforced concrete bearing walls which are also designed as shear walls to collect, resist and transmit the lateral loads down to the foundation. The floor slabs and roof are constructed of reinforced concrete supported by steel beams and, in some cases, a column framing system. The reinforced concrete floor slabs are designed as diaphragms capable of transmitting the lateral loads to the shear walls.

Floor slab structural drawings were also reviewed. In general, floors were constructed of reinforced concrete supported by steel beams. They were designed as diaphragms to transmit lateral loads to the shear walls. Large floor cutouts that could reduce the floor capacity appear to have been fully considered in the original seismic design as there is adequate steel around these cutouts to provide ductility to the load path if the SME loads were to exceed the ultimate or yield capacity.
The connection details of the braced frames were reviewed for structural adequacy and in particular, to ensure ductility to withstand overload.

In general, from the drawing review, it was concluded that the LGS structures are of robust seismic design and they can be screened-out for the specified 0.15g RLE without further evaluation. Items looked for during the civil structures walkdown include: confirmation of specific details contained in design drawing, potential seismic interactions and verification of the adequacy of the existing seismic gaps.

3.1.4.3.7 Walkdown

The walkdown of the civil structures was focused on identification of special situations that may have been overlooked during the design and that could pose a concern from a seismic standpoint. These included verification of gap dimensions between buildings, ensuring that potential differential displacements between separate but adjacent structural systems do not cause a seismic interaction, ensure that areas of potential gap closure and impact do not contain impact sensitive equipment, and check that the effects of potential relative displacement between buildings are properly considered in the design of the important plant systems.

The Primary Containment structure and the Reactor Enclosure are independent structures separated by a seismic gap filled with compressible material. Horizontal frequencies are 5.8 Hz for the Primary Containment and 3.1 Hz (north-south) and 3.8 Hz (east-west) for the Reactor Enclosure. Verification was made that adequate seismic gaps exist between the Primary Containment structure and the Reactor Enclosure and between the Control Room structure and the Turbine Enclosure. The minimum horizontal separation is 2 inches at the top of the pipe tunnel at elevation 289'. Based on a review of the existing documentation impact is not likely to occur at this location for the 0.15g SSE.

The seismic gaps between buildings were inspected during the walkdown. Gap sizes were confirmed and sensitive equipment located in the proximity of the gap locations was identified. Success path equipment located in the proximity of gaps are located in Room 580E (MCC 20B214) at elev. 283' in the Reactor Enclosure where the gap is about 2 inches. Similarly, in Room 580W there is a gap of 2.5 inches going on horizontally along the slab elevation and 3.5 inches to 4 inches going vertically along the drywell wall. These gaps are more than adequate to accommodate any differential displacements, and impact between the two structures is unlikely at the 0.15g SSE.

Other sensitive equipment at this location (Room 580W) is MCC 20B213 but it is located sufficiently away from the gap location (on the opposite side of the Primary Containment wall) and it is judged not to be of concern from

the impact standpoint.

The Control Structure was walked-down to ensure that possible impact from the Turbine Enclosure Roof (Elevation 324'-4") would not affect equipment located on the Control Structure side. At elevation 304', a 4-inch gap separates the Control Structure and the Turbine Enclosure Auxiliary Bay. The length of this gap was walked down and confirmed that no success path equipment is located in this area.

The RHRSW piping system was selected for walkdown to assess and verify that the effects from potential differential displacements between buildings were properly considered in the design of the LGS commodities. Penetration of the RHRSW piping as it goes out of the Pipe Tunnel at elevation 198' was inspected. The RHRSW piping penetration at this location has a 2-inch gap around the pipe which isolates the piping movements from the building response. The piping is supported on a separate concrete anchor located adjacent to the pipe tunnel. This separate anchor block is founded on essentially the same media as the pipe tunnel structure; e.g., bedrock subgrade with Class A concrete backfill material bearing on bedrock. Thus, possible failure modes associated with large relative displacement motion and lack of support flexibility and potential differential settlement between the soil-supported component and structure are not likely. In summary, good seismic design practices were observed during the walkdown and the piping was confirmed to have enough flexibility to accommodate any potential differential movement.

Based on the evaluations and reviews documented above, the LGS civil structures and building penetrations of underground utilities are judged to be acceptable by the SRT.

3.1.4.4 <u>Distributed Systems</u>

Walkdown of distributed systems such as piping, electrical raceways, conduits and HVAC systems were performed in conjunction with the walkdowns associated with other classes of components. This approach allowed the SRT to assess the structural adequacy of the distributed systems and to visually inspect the potential for seismic interaction of the distributed systems on the various classes of components being examined.

3.1.4.4.1 Piping Systems

The expert Panel on the Quantification of Seismic Margin (ref. 3.1-10), referred to hereafter as the "Panel", believes that, in general, piping systems in nuclear power plants have HCLPF capacities greater than 0.5

g pga. However, earthquake experience data has indicated that certain construction details have led to damage in industrial facilities. Examples of potential failure modes include:

- Valve failure by impact resulting from large displacements of flexible pipes;
- Pipe failure caused by large displacement of inadequately anchored equipment;
- Failure of small, stiff pipes attached to large, flexible pipes;
- Failure of piping between buildings as a result of large relative displacement caused by rocking or sliding of the buildings; and
- Failure of brittle connections (e.g., threaded pipe), eroded or corroded piping, and brittle cast iron piping.

All of these failure modes are related to displacement effects as opposed to inertia-induced stresses.

Walkdowns of piping systems in the areas where SPCL components are located were conducted in conjunction with equipment walkdowns to assess the potential for piping failures. Primary emphasis was placed on identifying the potential for displacement-induced effects on the piping. Prior to performing the plant walkdowns, the SRT became familiar with the plant design basis for pipe stress by reviewing relevant portions of the Updated Final Safety Analysis Report, pipe stress acceptance criteria, and plant design calculations.

The capability assessment criteria used for the evaluation of NSSS piping systems are given in Table 3A-19 of the UFSAR. The table is in agreement with a conservative general interpretation of the NRC Technical position given for Class 1, 2 and 3 components and components' supports.

BOP piping systems in the containment and reactor enclosure are analyzed in accordance with ASME Section III, Division 1 (1971 Edition with Addenda through Winter 1972 for Class 2 and 3 piping, and 1977 Edition through Summer 1979 Addenda for Class 1 piping). The design damping values are ½% and 1% for OBE and SSE respectively and 2% for combination of seismic and hydrodynamic loads.

The design basis for damping values are quite conservative when compared to 5 percent damped spectral accelerations recommended in EPRI NP-6041.

Experience from past earthquakes in industrial facilities indicates that piping is rugged and can generally resist earthquakes of 0.5g pga. Experience

has also shown that even if piping supports fail, failure of piping does not necessarily follow.

During the walkdown of the various equipment classes, the SRT reviewed the configuration of the piping for any potential failure modes discussed above. The condition of the piping systems on the success path were reviewed as well as general BOP piping during the walkdown. It was noted that the piping systems were well designed and built, the piping systems are very well supported and no differential anchor movement concerns were identified. Branch line piping was examined and found to have adequate flexibility.

Location of valve operators and other piping components, in relationship to structures, was observed and checked with the walkdowns of the valves. For the locations where valves were in close proximity to structures, the SRT evaluated the condition to determine the acceptability of the component.

As documented in the previous section (3.1.4.3), the RHRSW piping system was reviewed to evaluate the effects of potential differential displacements between buildings. No concerns were identified. Also underground piping was reviewed as discussed in reference 3.1-25 and was found to be acceptable.

Based on the results of the walkdown and a review of piping design criteria and calculations, the SRT judged the piping to be seismically rugged and screened-out from further review.

3.1.4.4.2 Electrical Raceways

According to Table 2-4 of EPRI NP-6041, cable trays can be screened-out from further review and assigned a minimum HCLPF capacity of 0.30g pga. However, EPRI NP-6041 also recommends that example cable trays and supports be inspected during the plant walkdowns to verify that they are adequately supported. In the past, cable trays have been constructed differently from design details. Therefore, a sample walkdown is necessary in assessing their as-built seismic capacity.

The "Panel" believes that cable trays have HCLPF capacities of at least 0.3g pga. Earthquake experience data indicates that there have been few failures of cable trays and supports for ground motions up to 0.5g pga. Of some concern are rigid "boot" connection details where vertical struts are bolted into a boot with a single bolt. In strong motion earthquakes, unbraced struts may undergo significant displacement. A potential failure

mode for a rigid "boot" connection is for the vertical strut to "walk" out of the boot due to the cyclical loading.

As part of the SRT walkdowns, a visual inspection of representative cable trays and supports was conducted at LGS to assess as-built seismic capacity. Particular emphasis was placed on reviewing the supports for long unbraced struts and the use of rigid "boots". Typically one or more cable tray was reviewed in each of the plant rooms containing equipment on the SPCL, as appropriate. The SRT noted that trapeze type supports were abundantly used. Supports were well braced both in-plane and out-of-plane.

The inspection also addressed such items as cable fill, cable ties, tray hold down, support anchorages, and tray spans. Prior to performing the plant walkdowns, the SRT became familiar with the plant design basis for cable trays and supports by reviewing relevant portions of the UFSAR and plant design drawings and calculations.

Cable tray supports at LGS are primarily constructed of unistrut members. Much of the cable tray raceway installation at LGS was based on conservative tray weight and span limitations using generic (or standard) support details. Standard support details were typically evaluated using maximum values for tray weights and spans and conservative static or equivalent static analysis method with spectral accelerations based on 10 percent damping.

The design basis damping value is quite conservative when compared to 15 percent spectral accelerations recommended in EPRI NP-6041.
Based on the results of the walkdown and review of design details and calculations, the SRT judged the cable tray raceways at LGS to be seismically rugged and screened-out from further review.

3.1.4.4.3 Electrical Conduit

According to Table 2-4 of EPRI NP-6041, conduits can be screened-out from further review and assigned a minimum HCLPF capacity of 0.30g pga. However, EPRI NP-6041 also recommends that sample conduit raceways be inspected during the plant walkdown to verify that they are adequately supported.

The "Panel" did not make recommendations on the HCLPF capacity for electrical conduit. EPRI NP-6041 indicates that earthquake experience data has shown that electrical conduit has not been vulnerable for seismic events up to 0.5g pga.

A visual inspection of representative conduits and supports was conducted at LGS to assess as-built seismic capacity. Particular emphasis was placed on reviewing conduit span lengths and support anchorages. Prior to performing the plant walkdown, the SRT became familiar with the plant design basis for conduit and supports by reviewing relevant portions of the UFSAR and plant design drawings and calculations. Typically one or more conduit was reviewed in each of the rooms containing equipment on the SPCL, as appropriate. The SRT noted that conduit was well supported. Supports were typically compact and did not support an overabundance of conduits. Supports with long members were typically braced. No specific vulnerabilities were noted during the walkdowns.

Conduit supports at LGS are typically constructed of unistrut members. Much of the conduit installation at LGS is based on conservative conduit size/weight and span limitations using generic (or standard) support details. In addition to the conservative design basis spectra for LGS, sources of conservatism in the design include:

- Generic support designs are based on worse case loading (conduit size/weight and spans). The likelihood of individual supports being consistently loaded in this worst case fashion is remote.
- Acceleration values used for generic support designs are typically based on the maximum worst case for permissible locations of the support.
- Generic supports are provided with an array of potential configurations (.e.g., variable member sizes, lengths). Designs typically consider the worst case combinations of these parameters.

On this basis, the SRT judged the conduit systems seismically rugged and screened-out from further review.

3.1.4.4.4 HVAC Systems

EPRI NP-6041 recommends that a representative sample of HVAC ducting be walked down to confirm seismic adequacy.

The "Panel" has indicated that all components in HVAC systems have HCLPF capacities of at least 0.30g pga. Reviews of damage reports for major earthquakes have not indicated ducting to be a problem itself. HVAC related problems generally are associated with loss of anchorage of fans and blowers and possibly fan blade misalignment. For those reasons, it was concluded that the dominant failure modes for HVAC systems are anchor bolts and support failures.

A visual inspection of representative HVAC duct and supports was conducted at LGS to assess as-built seismic capacity. Emphasis was placed on systems originating in the higher elevations of Category I structures where seismic amplification is greatest. In addition, HVAC related equipment on the SPCL was walked down to verify anchorage and assess the duct/equipment interface. Prior to performing the plant walkdown, the SRT became familiar with the plant design basis for HVAC duct and supports by reviewing relevant portions of the UFSAR, design specifications, and design drawings and calculations. In general, ducting was traced from major equipment through several rooms. The SRT noted that ductwork was well supported. Supports were positioned at regular intervals to yield high natural frequencies in the flexural mode. Duct supports were judged seismically rugged and were well braced for lateral loads. No relative movement concerns were noted at duct/equipment interfaces or building cross-overs. Some minor seismic interaction concerns were noted but were judged to be acceptable since only local dents in the ductwork could result.

HVAC supports at LGS are primarily constructed of structural members (angles, wide flanges, tube steel) welded and/or bolted together; some configurations also incorporated the use of unistrut members. Much of the installation of HVAC ducting was based on conservative duct size and span limitations using generic (standard) support details. Conservatism similar to those noted for electrical raceways (cable trays) and conduit were also noted in HVAC designs.

Based on the results of the walkdown and review of design documents, the SRT judged that HVAC duct systems are seismically rugged and screened-out from further review.

3.1.4.5 <u>Other Components</u>

3.1.4.5.1 NSSS Primary Coolant System and Supports

The nuclear steam supply system (NSSS) for the Limerick Generating Station is a boiling water reactor (BWR) which was designed and supplied by the General Electric Corporation.

In accordance with EPRI NP-6041 Table 2-4, the NSSS primary coolant system (piping and vessels) can be screened out, provided the piping is not suspected of intergranular stress corrosion cracking (IGSCC). Also, visual inspection of NSSS is not required since sufficient detail is available in the design basis documentation to assess capacity. In addition, the supports for NSSS components

can be screened out if they are designed for combined dynamic SSE and pipe break analysis loadings.

Programs to mitigate any concerns of IGSCC in the NSSS piping welds for LGS are in place. The NSSS supports were designed for combined dynamic loading from SSE and postulated pipe break. In addition the SRT confirmed the presence of lateral supports for the recirculation pumps during the Drywell Walkdowns.

Based on the above, the NSSS primary coolant system and the NSSS supports are screened-out from further review by the SRT.

3.1.4.5.2 Reactor Internals

This review is not required for a reduced scope plant.

3.1.4.5.3 CRD Mechanisms and HCUs.

The guidance provided in EPRI NP-6041 is that the control rod drive (CRD) housings and mechanisms do not require an evaluation for 0.3g if the CRD housing has lateral seismic supports.

At LGS the hydraulic control units (HCUs) are located outside the Drywell at Elevation 253' of the Reactor Building. The base of the HCU is bolted to an embedment in the slab. The HCU components and piping are laterally braced by a substantial frame structure; this frame is built up around tube steel posts anchored to the floor slabs and interconnected with various structural members including unistrut channels. The SRT also performed a walkdown of the HCU's and identified no concerns with the units.

Inside the Drywell the SRT was able to gain visual access under the vessel and confirmed the presence of the hangers providing lateral restraints to the CRD hydraulic system piping.

In addition, the SRT reviewed the SARA report (ref. 2.1-2) and the supporting calculations. The SARA report included fragility estimates for the CRD guide tubes, housing, housing supports, and piping; also included was the fragility value for the HCU. The HCU had the lowest fragility estimate with a corresponding $HCLPF_{84}$ = 0.39g.

Based on the above reviews, the CRD and HCU were screened-out from further review by the SRT.

3.1.4.5.4 Control Room Ceiling

The LGS control room ceiling was designed, furnished and installed in accordance with Specification A.24 (ref. 3.1-22). The design criteria addressed the pertinent seismic design requirements as documented in Section 5.0 of the specification.

The installed system was supplied by Sanders & Thomas, Inc., and consisted of a two level suspended system. A braced unistrut frame welded to the structural I beams is used as the primary support, with a second tier for accommodating the light diffusers suspended with short rods from the main unistrut grid.

During plant walkdowns, the SRT also made arrangements with plant maintenance to gain access to the ceiling. The inspection provided the SRT with a confirmation of good lateral bracing being used, and the adequacy of the connections and restraints for the lower diffusers.

Based on these reviews and guidance of EPRI NP-6041 the control room ceiling was screened-out from further review by the SRT.

3.1.4.5.5 Masonry Block Walls

In accordance with the recommendations of EPRI NP-6041, a seismic review of masonry walls is only required for a RLE that exceeds the plant SSE. Therefore this review is not required for LGS.

3.1.4.5.6 Soils Evaluation

A soils evaluation is not required for a reduced scope plant.

3.1.4.6 <u>Summary and Conclusions</u>

Based on the results of the seismic margin reviews discussed earlier, it is very evident that the Limerick Generating Station equipment, structures, and distributed systems are seismically very rugged, owing to conservative original designs. All designs were found to have substantial reserve margins. Given the original plant design criteria which specifically addressed compliance with Regulatory Guide 1.29 (ref. 3.1-11) no instances were found where interactions were caused by permanent plant commodities or structures. All items which were not screened out during the seismic walkdown are listed in Table 3.1.4-3. No single unscreened item was identified that was related to original design or a permanently

installed commodity. The SRT therefore believed that with the resolution of the items listed in Table 3.1.4-3 the plant has sufficient margin to withstand the RLE.

3.1.5 Analysis of Containment Performance

This section describes the method of approach and a summary of conclusions for assessing the Containment Performance for the Limerick Generating Station Units 1 and 2 in accordance with Generic Letter 88-20, Supplement 4 for a reduced scope plant.

Per NUREG-1407, for a reduced scope plant, only the retention of the walkdown of containment systems necessary to prevent early failure is required. The primary purpose of the walkdowns is to identify anchorage and spatial interactions that may exist.

3.1.5.1 Methodology

EPRI Report NP-7498 discusses the technical approach to be used for containment evaluation. Specifically, Appendix D, "Containment Performance Requirements", describes the proposed industry approach to reviewing containment performance for the seismic portion of the IPEEE process for full and focused scope plants. Page D-5 recommends "that the SMA demonstrate with a high degree of confidence that those containment related functions that are necessary to prevent early containment failure survive the SME (early means roughly the 12 hours following the Seismic Margin Earthquake (SME))."

For a focused scope SMA this would include:

(1) A successful containment isolation
(2) The demonstration that the SME does not fail the containment structural integrity (including containment penetrations such as piping, instrumentation, electrical, drywell personnel air-lock and equipment hatches, drywell head and wetwell access hatches.)
(3) The demonstration that SME (or seismically included relay chatter) does not result in containment bypass.

Although a "reduced scope" IPEEE assessment is performed at the plant design basis seismic input level (Safe Shutdown Earthquake, SSE) for LGS, the seismic capability screening walkdowns consider the guidance of EPRI NP-6041-SL. Accordingly, components are screened to a minimum RLE of 0.3g pga.

3.1.5.2 Description of the Primary Containment

The Limerick Mark II primary containment is divided by a horizontal

diaphragm slab into two major regions: the drywell and the suppression chamber. The drywell encloses the reactor vessel, reactor recirculation system, and associated piping and valves. The suppression chamber stores a large volume of water.

The primary containment, shown on Figure 3.8-1 of the UFSAR is in the form of a truncated cone over a cylindrical section, with the drywell being the upper conical section and the suppression chamber being the lower cylindrical section. These two sections comprise a structurally integral, reinforced concrete pressure vessel, lined with welded steel plates and provided with a steel domed head for closure at the top of the drywell. Connection of the drywell head to the top of the drywell wall is shown on Figure 3.8-9 of the UFSAR. The diaphragm slab is a reinforced concrete slab structurally connected to the containment wall, as shown on Figure 3.8-10 of the UFSAR. The primary containment is structurally separated and therefore seismically isolated from the surrounding reactor enclosure. Seismic impact on the structure itself is expected to be minimal.

3.1.5.2.1 Penetrations

Services and communications between the inside and the outside of the containment are performed through penetrations. Basic penetration types include pipe penetrations, electrical penetrations and access hatches (equipment hatches, personnel lock, suppression chamber access hatches, and CRD removal hatch). Each penetration consists of a pipe sleeve with an annular ring welded to it. The ring is embedded in the concrete wall, and provides an anchorage for the penetration to resist normal operating and accident loads. The pipe sleeve is also welded to the containment liner plate to provide a leak-tight penetration.

Meridional and hoop reinforcement is bent around typical penetrations. Additional local reinforcement in the hoop and diagonal directions is added at all large penetrations. Local thickening of the containment wall at penetrations is generally not required.

The containment hatches and drywell head seals do not rely on gas pressure, electricity, or other active means of function. Seals are formed by bolting and mechanical deformation of "O" rings. As with hatches, the containment penetrations/penetration seals are passive, i.e. they do not rely on pneumatic pressure or electricity to function.

3.1.5.2.1.1 Pipe Penetrations

There are two basic types of pipe penetrations. For piping systems

containing high temperature fluids, a sleeved penetration is furnished, providing an air gap between the containment concrete wall and the hot pipe. This air gap is large enough to maintain the concrete temperature below 200°F in the penetration area. A flued head outside the containment connects the process pipe to the pipe sleeve. For piping systems containing low temperature fluid, a separate sleeve for the penetration is not furnished. For this type of penetration, the process pipe is welded directly to the two ends of the embedded pipe penetration. Typical pipe penetration details are shown on Figure 2.8-21 of the UFSAR.

3.1.5.2.1.2 <u>Electrical Penetrations</u>

Figure 3.8-22 of the UFSAR shows a typical electrical penetration assembly used to extend electrical conductors through the containment. The penetrations are hermetically sealed and provide for leak testing at design pressure.

3.1.5.2.1.3 <u>Equipment Hatches and Personnel Lock</u>

Two equipment hatches, with inside diameters of 12 feet, are furnished in the drywell wall. One of these equipment hatches includes a personnel lock. Figures 3.8-23 and 3.8-32 of the UFSAR show details of an equipment hatch. Additional meridional, hoop, helical, and shear reinforcement is used to accommodate local stress concentrations at the opening. The primary containment wall is thickened at the equipment hatches to accommodate the additional rebars. As described above, the seals for the penetrations are passive and do not use inflatable seals.

3.1.5.2.1.4 <u>Suppression Chamber Access Hatches</u>

Two access hatches, with internal diameters of 4'-4", are furnished in the suppression chamber wall. Additional local reinforcement in the meridional and diagonal directions is added.

3.1.5.2.1.5 <u>CRD Removal Hatch</u>

One three (3) foot diameter CRD removal hatch is furnished in the drywell wall to permit transfer of the CRD assemblies into and out of the drywell. The hatch is furnished with a double-gasketed flange and a bolted flat cover. Figure 3.8-34 of the UFSAR shows details of the CRD removal hatch.

3.1.5.2.2 Diaphragm Slab and Downcomers

The diaphragm slab serves as a barrier between the drywell and the suppression chamber. It is a reinforced concrete circular slab, with an outside diameter of 88 feet and a thickness of 3'-6".

The diaphragm slab is supported by the reactor pedestal, the containment wall, and 12 steel columns. The diaphragm slab is penetrated by 87, 24-inch diameter downcomers. The downcomers are 24" outside diameter pipe by 3/8" thickness. Additional reinforcement is furnished at downcomer penetrations. A 1/4 inch thick, carbon steel liner plate is provided on top of the diaphragm slab and is anchored to it. The liner plate prevents bypass flow around the downcomers during a LOCA.

The diaphragm slab is attached to the containment wall by a structural weldment at the junction of the two components. Radial force and bending moment, carried by the diaphragm slab main reinforcement, are transferred to the containment wall by cadwelding the diaphragm slab rebar to the top and bottom flanges of the structural weldment. The top and bottom flanges of the structural weldment are embedded in the containment concrete wall, and are anchored using structural steel anchors. Flexural shear in the diaphragm slab is transferred to the containment wall through the web of the structural weldment, which is welded to opposite sides of the thickened containment liner plate.

In order to maintain the integrity of the wetwell and the drywell with respect to each other, the downcomers and the drain lines in the reactor pedestal must remain intact. The vacuum breakers on the downcomers are also included on the SPCL as the only active components that provide isolation between the wetwell and the drywell.

Four downcomers are capped and contain the vacuum breakers. Figure 3A-461 of the UFSAR shows the location of the capped downcomers. Figure 4A-462 of the UFSAR shows the detail of the capped downcomers and vacuum breaker locations.

3.1.5.2.3 Reactor Pedestal Drain Lines

The Limerick primary containment does not have downcomers located in the reactor pedestal. Although the pedestal area does not contain downcomers, it does contain four drain lines connected to a drain sump (tank). It is postulated that during a core melt, the drain lines will melt when the corium drops to the floor, creating a direct pathway to the suppression

pool.

The reactor pedestal is shown on Figure 3.8-38 of the UFSAR.

3.1.5.2.4 Suppression Pool

The suppression pool acts as the heat sink for all of the following: a Loss of Coolant Accident (LOCA) within the drywell; a safety relief valve lift; or, the HPCI and RCIC turbine exhaust. Energy is transferred to the suppression pool by the discharge piping from the reactor pressure relief valves, the drywell downcomers, the HPCI and RCIC systems turbine exhaust pipes, and the RHR heat exchangers relief valves. The exhaust steam is discharged below the water surface and is condensed. The SRV discharge piping is used as the energy transfer path for any condition which requires safety relief valve operation. The drywell downcomers are the energy transfer path for energy released to the drywell during a LOCA.

3.1.5.2.5 Drywell Head Assembly

The drywell head lower flange is anchored to the top of the drywell wall by rigid attachments to 108 meridional reinforcing bars in the inner curtain of the containment wall. The drywell head provides a removable closure at the top of the primary containment for reactor access during refueling operations. The drywell head consists of a 2:1 hemi-ellipsoidal head and a cylindrical lower flange. Figure 3.8-31 of the UFSAR shows the detail of the head assembly. Seals are formed by bolting and mechanical deformation of "O" rings.

3.1.5.3 Applicable Codes, Standards and Specifications

This sub-section provides a brief summary of plant information and original dynamic design criteria that were reviewed in preparation for performing the screenings and walkdowns.

3.1.5.3.1 Loads and Loading Combinations

Table 3.8-2 of the UFSAR lists the loading combinations used for the design and analysis of the containment. Loading combinations listed in ASME Section III, Division 2 were considered for the containment design. Table 3.8-21 of the UFSAR identifies and explains differences between the loads listed in Table 3.8-2 and the ASME Code.

Section 3.8 of the UFSAR describes the procedures used for the design and analysis of the containment. For a description of the design and

analysis procedures that consider the effects of hydrodynamic loads resulting from MSRV discharge and LOCA phenomena, refer to Section 3.8 of the UFSAR and Appendix 3A of the UFSAR.

3.1.5.3.2 Structural Acceptance Criteria

The primary containment wall, the diaphragm slab, and the reactor pedestal are designed for the factored load combinations listed in Table 3.8-2 of the UFSAR in accordance with the ultimate strength method of ACI 318 (1971) and therefore also meet the ductility detailing requirements.

The codes, standards and specifications used in the design and construction of the primary containment are listed in Table 3.8-1 of the UFSAR. The primary containment is also analyzed and designed for hydrodynamic loads resulting from MSRV discharge and LOCA phenomena. Appendix 3A, " Design Assessment Report", of the UFSAR describes the basis for the hydrodynamic evaluation.

3.1.5.4 <u>System Considerations</u>

Appendix 2 of Generic Letter 88-20, Supplement 4 requires that an evaluation be made of containment performance for external events. The evaluation must identify vulnerabilities that involve early failure of containment functions. These functions include:

(1) Containment integrity, including specific systems depending on containment design (e.g., suppression pool) and prevention of bypass functions.

(2) Containment isolation.

The system considerations which lead to the selection of structures and equipment necessary for primary containment performance are summarized below.

As required by GL 88-20, Supplement 4, the analyses performed for the internal events Individual Plant Examination (Level II IPE) were used to determine the scope of systems to be included as required for shutdown during a seismic event.

3.1.5.4.1 Primary Containment Integrity

To determine the effectiveness of the requirements in assuring adequate containment performance, a systematic review of the primary containment

challenges associated with a spectrum of severe accident types was assembled in the Internal Events IPE. The systems and functions that were identified in the IPE as required to mitigate the effects of the event upon primary containment and, therefore prevent primary containment failure, are the following:

- Reactivity Control - CRD System
- Reactor Pressure Control - ADS, and SRV
- RHR in the Suppression Pool Cooling Mode
- RHR in the Drywell Spray Mode
- The Downcomers
- Vacuum Breakers on the Downcomers
- Drain Lines in the Pedestal
- SRV Discharge Lines

The components required for reactivity control, reactor pressure control, and for the RHR system in the Suppression Pool Cooling mode have been selected for inclusion in the Success Path Component List (SPCL). These systems provide the necessary containment heat removal and pressure suppression functions and reactor reactivity and pressure control functions to prevent containment failure.

Isolation of primary containment and containment bypass are addressed below.

3.1.5.4.2 Primary Containment Isolation

Valves involved in the primary containment isolation system are expected to be seismically rugged (NUREG/CR-4734). Seismic failures of actuation and control systems are more likely to cause isolation system failure and were included in the examination. NUREG-1407 states that for valves relying on a backup air system, the air system should also be included in the seismic examination. In the Limerick configuration, the valves provide a passive function for isolation (i.e., they must remain closed) and loss of the air supply to the valves will not affect the position of the valve. The valves will fail closed on loss of air. Therefore, the backup air supply to these valves was screened from further evaluation.

The internal events level II IPE was again used to scope the valves that are important to containment isolation. The level II IPE begins with a state of plant damage. Therefore, it is assumed that containment isolation valves receive isolation signals, and operators receive instructions to verify isolation. Thus, containment isolation will not fail if:

- At least one isolation valve in each piping penetration closes and remains closed.

- Each of the containment access penetrations remains sealed.

Tables 6.2-17 and 6.2-25 from the UFSAR list the Limerick primary containment penetrations. The first table is a list of all containment penetrations. The second table lists only piping penetrations, and includes details about isolation valve types, positions, actuation methods, and isolation signals. From a licensing perspective, any isolation failure constitutes a system failure. However, of all the possible system failures, only a few significantly alter the course of a core damage sequence. These are:

- Isolation failures of a large penetration open to the drywell. After a LOCA or RPV rupture, these failures provide an immediate release to the reactor enclosure. The flow path is sufficiently large to prevent containment over-pressure failures.
- Isolation failures of large penetrations open to the wetwell. This failure is equivalent to wetwell venting. It prevents containment over-pressure and creates a release pathway through the suppression pool. However, if coupled with drywell vacuum breaker failures, this flow path becomes equivalent to a drywell isolation failure.

By focusing on these particular failures, the set of all containment penetrations can be reduced to a small number of lines. Those penetrations having no significant role in characterizing source terms include:

<u>Penetrations for RCS Connections</u>. These include feedwater and main steam lines, CRD lines, HPCI/RCIC steam lines, RHR shutdown cooling lines and low pressure ECCS injection lines. The Level I portion of the Limerick IPE addresses failure of the main steam lines, feedwater lines, HPCI and RCIC lines (Section 3.1.3.24) and considers the likelihood very low. The likelihood of the reactor coolant system failing to isolate appears small compared to other primary containment isolation failures. This is born out by the LGS IPE Level II Analysis (Section 4.5, IS - containment isolation mode description). Therefore, RCS connection penetrations will not be analyzed further.

<u>Penetrations for Closed Loop Piping Systems</u>. These penetrations create a release path only if:

a. Piping failures occur both inside and outside containment.

b. Containment isolation valves fail to close.

Assuming some degree of independence between failures, the joint probability of such a release path appears negligible.

Penetrations for Small (less than or equal to 1" diameter) Piping. Test, vent, drain, and similar types of branch lines have manually closed (and locked closed after use) isolation valves and a screwed cap outboard of the containment boundary. This combination of hardware and administrative control makes isolation failure very unlikely. It also appears such failures could not prevent other containment failures, given the rate and magnitude of energy released to containment during a core melt event. The Level II IPE (Section 4.4.2) analysis dismisses small (less than 2" diameter) lines penetrating containment because the release magnitude is too low to be of significance.

Table 3.1.5-1 of this document lists the piping penetrations which remain after applying these screening criteria to Table 6.2-25 of the UFSAR. The valves on this list were selected for inclusion on the SPCL. All of the potential drywell release paths through the penetrations have two valves in series. All of the valves except for two are normally closed and all of the valves fail in the closed position. Given this valve alignment and configuration, the valves provide a passive function (i.e., they must remain closed). Therefore, there is no analysis required for either the electrical motive power to the valves or the air supply to the valves since loss of either of these functions will not affect the position of the valve. Also, since a prior relay review performed on Limerick identified no "bad actor" relays, the control and actuation circuitry associated with these valves cannot cause a chatter induced opening of the valves. In fact, since the valves are part of the containment isolation system, the only signal they would receive would be a close signal, keeping the valves in the desired state.

TABLE 3.1.5-1

**PRIMARY CONTAINMEMT
ISOLATION VALVES IMPORTANT TO SAFE SHUTDOWN**

Penetration X-25	Penetration X-26
HV-57-1(2)35	HV-57-1(2)15
HV-57-1(2)21	HV-57-1(2)11
HV-57-1(2)23	HV-57-1(2)14
HV-57-1(2)31*	HV-57-1(2)61
HV-57-1(2)63	HV-57-1(2)17
HV-57-1(2)09*	FV-C-DO-1(2)01
FV-C-DO-1(2)01B	

Penetration X-201-A	Penetration X-202
HV-57-1(2)09*	HV-57-1(2)12
HV-57-1(2)47	HV-57-1(2)62
HV-57-1(2)24	HV-57-1(2)05
HV-57-1(2)31*	HV-57-1(2)04
HV-57-1(2)64	HV-57-1(2)18
HV-57-1(2)69	HV-57-1(2)66

Penetration X-231-A	Penetration X-231-B
HV-61-1(2)10	HV-61-1(2)30
HV-61-1(2)11	HV-61-1(2)31

*Piping is routed through more than one penetration

Primary Containment Vacuum Relief Valves
PSV-1(2) 137A-1 and 2
PSV-1(2) 137B-1 and 2
PSV-1(2) 137C-1 and 2
PSV-1(2) 137D-1 and 2
As shown on Dwg 8031-M-57, Sht. 2, Rev. 35

3.1.5.4.3 Primary Containment Bypass

In order to preclude bypass of the primary containment function, the integrity of the drywell and wetwell must be maintained with respect to each other and to outside containment. In order to maintain the integrity of the wetwell and drywell with respect to each other, the downcomers, SRV discharge lines and the drain lines in the reactor pedestal must remain functional. Since these structures are not active individual pieces of equipment, they will not be included on the SPCL. However, they were included in the SRT walkdown to verify seismic adequacy. The drywell/wetwell vacuum breakers on the downcomers have been selected for inclusion on the SPCL as the only active components that provide isolation between the drywell and wetwell. The drywell/wetwell vacuum breakers are listed in Table 3.1.5.-1

3.1.5.5 Seismic Capability Walkdown

As recommended in NUREG-1407 for a reduced scope SMA, a walkdown was performed of the identified systems to identify anchorage and/or spatial interaction problems.

Table 2-3 of EPRI Report NP-6041 lists the elements which should be screened from a seismic margin review when performed in conjunction with a "walkdown" of plant specific elements to evaluate for unusual conditions (e.g., special interaction, unique penetration configuration), because of their generically good performance at this review level.

As part of the seismic capability walkdowns, the seismic review team (SRT) performed "walkdowns" of the containment including the containment structure, access hatches, suppression pool, primary containment vacuum relief valves, drain lines in the reactor pedestal, SRV discharge lines and isolation valves important to safe shutdown.

Walkdowns were performed by the SRT of these elements for LGS Unit 1 during February 1994 and March 1995. Walkdowns were performed for Unit 2 during February and March 1993 and March 1995, except for the suppression pool and reactor pedestal drain lines. A drawing review of these elements for Unit 2 was performed and was compared to the Unit 1 walkdowns and found to be similar to LGS Unit 1 and therefore acceptable. As part of the walkdowns the piping penetrations and valves listed in Table 3.1.5-1 were inspected to ensure that they are seismically rugged and that there are no spatial interaction issues. As stated in Section 3.1.5.4, the valves provide a passive function.

Based on the SRT walkdowns and design basis document review, it was noted that the plant was well constructed, there is no potential for seismic interactions, nor any unusual designs or unique configurations.

3.1.5.6 Summary and Conclusions

A review of construction drawings, design basis calculations and applicable Sections of the UFSAR was performed by the SRT. This review included penetrations, the containment structure, downcomers, SRV discharge lines, access hatches and suppression pool and vacuum breakers as well as those valves listed in Table 3.1.5-1. As part of the SRT walkdowns, a seismic walkdown of the above components and structures was performed. The walkdown included an evaluation for unusual conditions; i.e., spatial interactions and penetration configuration. No unusual or unique features were noted.

Based on a review of the technical documentation in conjunction with walkdowns by the SRT, no sequence that involve containment failure modes distinctly different from those found in the IPE internal events evaluation were identified.

Therefore, the SRT judged the containment and associated elements to be seismically rugged and vulnerabilities were not found in the systems/functions which would lead to early containment failure and high consequences.

3.2 USI-A45, GI-131, and Other Seismic Safety Issues

Six programs related to seismic events requiring PECO Energy Co. action have been identified in Supplement 4 to Generic Letter 88-20 and NUREG-1407. A brief description and the proposed resolution of these programs are documented below.

(1) USI A-17, System Interactions in Nuclear Power Plants. This unresolved safety issue addresses the NRC's concern in regard to possible system interactions that could adversely affect redundancy and independence of safety systems. The seismic spatial system interaction of USI A-17 at LGS was addressed as part of original design and construction based on reference 3.1-17. In addition system interaction issues were captured in the IPEEE program through the screening and seismic margin assessment walkdowns of the items in the IPEEE safe shutdown equipment list.

(2) USI A-40, Seismic Design Criteria. This unresolved safety issue, moreover, addresses the concern in regard to seismic adequacy of safety-related above-ground large flat bottom storage tanks for SSE loading. There are no safety-related above-ground large flat bottom storage tanks at LGS, therefore, USI A-40 is not relevant to LGS. Moreover, the IPEEE seismic success paths for LGS do not include credit for any above-ground tanks.

(3) USI A-45, Shutdown Decay Heat Removal Requirement. The objective of USI A-45 is the determination of the adequacy of decay heat removal systems to ensure that LGS does not pose unacceptable risk as a result of decay heat removal system failure. The LGS IPE for internal initiating events, section 3.4.3 addresses USI A-45. The IPE presents a strategy for removal of decay heat which is capable of success despite potential increasing plant damage. The IPE states "Loss of decay heat removal capability is not considered a vulnerability and no additional insights have been found that require incorporation in plant procedures or equipment."

For seismic events, the SMA approach is based on selection of two success paths, a preferred and alternate, to achieve safe shutdown. As defined in NP-6041, the safety functions necessary to preclude core damage and thereby achieve safe shutdown consist of the reactivity control, reactor coolant pressure control, reactor coolant inventory control and decay heat removal. Therefore, as part of the SMA the adequacy of the decay heat removal function following a RLE is addressed.

As shown on Table 3.1.2-1, the preferred and alternate frontline systems selected for satisfying the decay heat removal safety function were the residual heat removal system (shutdown cooling mode) and the residual heat removal system (alternate shutdown cooling mode) respectively. Table 3.1.2-1 also lists the necessary support systems for each of the two shutdown paths. A detailed discussion of the operation of the RHR system to remove decay heat is provided in Section 3.1.2.5.3.4.

The components within the preferred and alternate paths chosen for the decay heat removal function were identified during the success path component list development and then assessed for seismic functional capability using the approach described in Section 3.1.4.1.1 which emphasized thorough walkdowns and reviews.

Since all components comprising the two success paths were found by the SRT to be acceptable for the RLE earthquake, loss of decay heat removal capability is not considered a vulnerability and no additional insights from the SMA review have been found that require incorporation in plant procedures or equipment.

(4) USI A-46, Verification of Seismic Adequacy of Equipment in Operating Plants. Generic Letter 87-02 was issued on February 19, 1987, to holders of operating licenses not reviewed to current licensing criteria on Seismic Qualification of Equipment as a result of the technical resolution of USI A-46. LGS was reviewed to the current licensing criteria on seismic qualification of equipoment prior to receiving its operating license. Hence, A-46 is not applicable to LGS.

(5) GI-131, Potential Seismic Interactions Involving the Movable In-Core Flux Mapping System Used in Westinghouse Plants. Each unit at Limerick consists of a General Electric Boiling Water Reactor. Hence, GI-131 is not applicable to LGS.

(6) The Eastern U.S. Seismicity Issue, formerly called the Charleston Earthquake issue. The IPEEE seismic margin earthquake was determined based on the probabilistic seismic hazard studies by NRC/LLNL and EPRI in resolution of this issue. Hence, the IPEEE submittal provides a resolution to the Eastern U.S. issue without any additional analyses or documentation from PECO Energy Co.

4 INTERNAL FIRES ANALYSIS

4.0 METHODOLOGY SELECTION

The Fire Induced Vulnerability Evaluation (FIVE) methodology (Ref. 1.1-4) was selected as the method to satisfy the NRC request described in Generic Letter 88-20, Supplement 4. The NRC has reviewed the EPRI developed FIVE methodology and has determined that it provides a comprehensive approach for screening plant areas for fire risk and is an acceptable method for meeting GL 88-20 requirements (Ref. 1.1-1). The FIVE methodology was used to identify fire areas of potential risk significance, calculate area fire ignition frequencies, and provide hazards analysis for resulting critical areas. The quantification of fire induced safe shutdown system unavailability was obtained by propagating fire induced system failures through a modified PSA plant model.

4.0.1 Description

The fire analysis presented in this section was performed by a team with expertise in fire modeling, system operation, fire protection/safe shutdown engineering, and PSA. The analysis is based on the current Limerick Appendix R analysis with proposed modifications designed to reduce the dependance on Thermo-lag encapsulation material and an updated IPE (LGS PSA) evaluation conducted for Limerick Generating Station. It also incorporates data from the EPRI Fire Events Database and fire hazards analysis methods from the FIVE methodology.

The process takes advantage of the existing PSA logic models and computational aids, and the FIVE methodology worksheets and equations. Plant walkdowns were conducted to provide feedback on in-plant data as it may impact the analysis and to address the Fire Risk Scoping Study issues.

Figure 4.0-1 illustrates the steps followed using the basic elements of the FIVE methodology.

The process provides the ability to focus early in the analysis on the significant fire areas and issues. PSA models are then applied on a limited number of areas found to have potential fire risk significance. Finally, in-depth fire analyses are then performed on the critical fire areas.

FIGURE 4.0-1
Fire Evaluation Methodology
(Page 1 of 3)

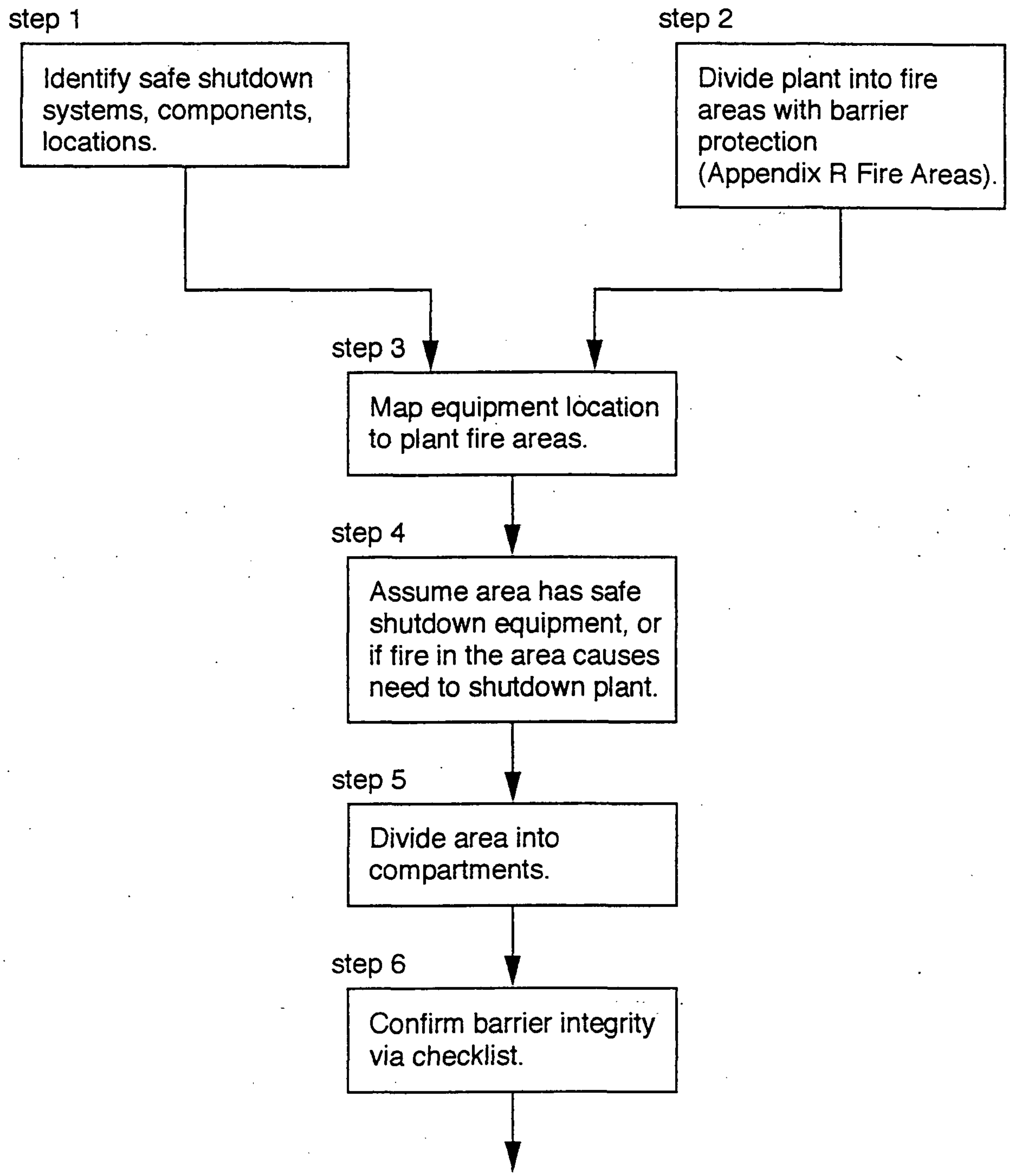


FIGURE 4.0-1 (continued)
Fire Evaluation Methodology
(page 2 of 3)

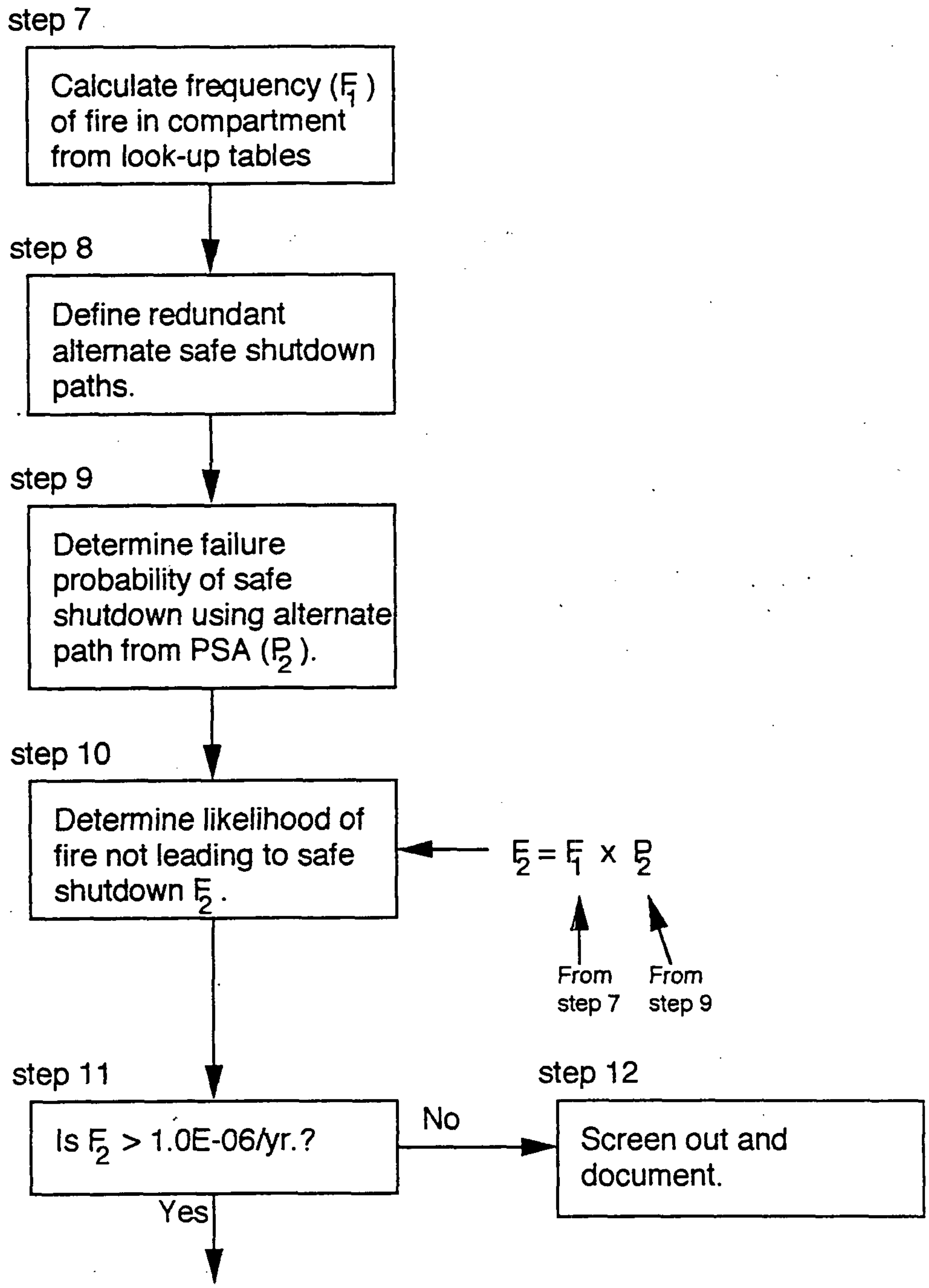
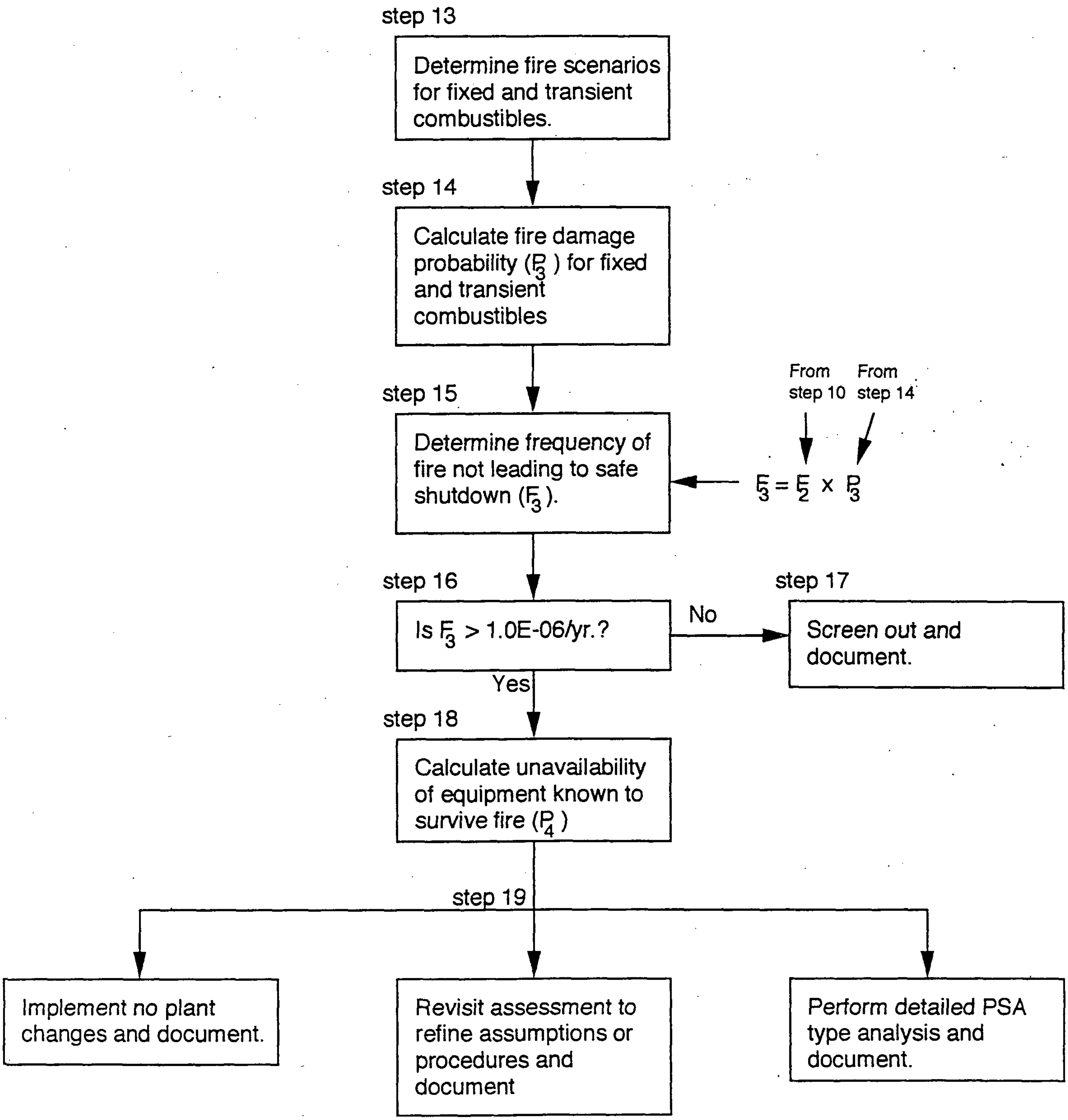


FIGURE 4.0-1 (continued)
Fire Evaluation Methodology
(page 3 of 3)



Although the FIVE methodology was followed, a few enhancements were made to facilitate the interaction between the groups performing the analyses. The methodology chosen consists of four phases:

Phase I, Qualitative/Quantitative Analysis

This phase provides a method for quickly screening plant areas whose loss due to fire will have no impact on the ability to achieve and maintain safe shutdown. Steps in this phase consist of the following:

(1) Identify fire areas to be analyzed. The Appendix R fire areas and zones were used as an initial basis for this study.

(2) Identify safe shutdown equipment in each fire area identified above. A spatial database of equipment and cable routing for the defined areas was developed. The Appendix R analysis was the principal source for equipment information, supplemented by plant arrangement drawings, cable raceway schedules, and walkdowns..

(3) Identify effects of fire in each area. For each area, identify existence of safe shutdown equipment that could be affected by a fire in the area and identify if fire results in a demand for plant shutdown. It was conservatively assumed that every fire area resulted in a plant shutdown (either automatic or manual due to Technical Specification requirements).

(4) Evaluate Fire Compartment Boundaries. The barriers that define each fire area were reviewed against the criteria provided in FIVE. This process leads to the identification of fire compartments. The remainder of the fire IPEEE analysis was based on these fire compartments rather than the Appendix R fire areas.

Phase 2, Quantitative Screening

This phase provides for screening compartments based on the fire ignition frequencies, and the availability of safe shutdown equipment outside of the fire area/compartment. Steps in this phase consist of the following.

(1) Quantification of Fire Ignition Frequencies. The fire ignition frequencies (F_1) are calculated for all fire compartments identified in Phase 1. This calculation is based on the latest industry fire frequency data from the EPRI Fire Events Database as provided in the FIVE documentation.

(2) Identify non-fire induced safe shutdown availability. The purpose of this step is to evaluate the likelihood of safe shutdown paths being unavailable (P_2) at the same time a fire occurs that disables safe shutdown equipment in the fire compartment. Appropriate reviews were performed for non-Appendix R systems to determine whether they would be disabled by the postulated fire events. Appendix R safe shutdown (SSD) paths and selected proposed paths were analyzed using SSD rules for equipment and cable selection. Not all potential systems credited in the PSA were utilized when determining the potential for screening an area.

(3) Screen compartments. Plant response to a fire in each compartment is evaluated based on the IPE PSA models and the unavailability of non-fire damaged shutdown equipment. If the product (F_2) of the compartment fire ignition frequency (F_1) and the shutdown system unavailability (P_2) is less than 1.0E-6/yr, the compartment is screened out from further analysis. This screening conservatively assumes all safe shutdown systems or associated equipment in the fire compartment are disabled by the fire.

<u>Phase 3, Fire Damage Evaluation Screening</u>

For those compartments that could not be eliminated as part of the phase 2 process, a detailed assessment is completed on the effects of a fire. Compartments may be screened during this assessment if the estimated maximum environmental condition does not exceed the damage threshold criteria for the target equipment. This methodology includes the following aspects.

(1) Fire Hazard Parameters. Fire hazard documentation is developed to provide the information necessary to support fire damage evaluation. For each compartment, this documentation contains information on fire detection, fire suppression, fire barriers, and types and amounts of combustibles.

(2) Identify Fire Scenarios. Within each fire compartment fire scenarios are identified for analysis based on the locations of credible fire sources. This step identifies the credible fire scenarios and the geometry of the area around each fire source in which targets will be damaged.

(3) Fire Suppression. Evaluation of automatic and manual suppression is performed for plant-specific areas, including fire brigade and fire detection system response times.

(4) Fire Growth and Propagation. For each compartment, target equipment sets of interest are identified. Fire growth and propagation analysis is performed based on the FIVE worksheets and heat transfer equations. For each target set, it is determined if sufficient combustible materials are present in the compartment to cause damage, and, if so, the time it will take to cause damage and the time it will take to actuate detection and suppression.

(5) Quantify Fire Damage Probability. For each identified target set and fire scenario, the probability of damage due to the fire is determined based on the FIVE equations. The compartment is screened out from further analysis if the cumulative target set damage probability (F_3) is less than 1.0E-6/yr.

<u>Phase 4, Fire Scenario Evaluation and Quantification</u>

LGS PSA models are utilized to quantify system unavailabilities for each fire scenario in the unscreened areas. This consists of four steps.

(1) Identification of Fire Induced Sequences. For each fire compartment, one or more fire induced event sequences were defined and evaluated.

(2) Fire Event Trees. Event trees for fire induced sequences were developed and used to quantify system unavailabilities.

(3) Recovery Actions. Human actions credited in the safe shutdown system analysis were compared to those in the PSA model.

(4) Probabilistic Assessments. The system unavailabilities for each fire induced sequence was quantified. Non-fire induced equipment failure probabilities from the system fire damage evaluations were calculated using a PSA model.

4.0.2 Assumptions

The following assumptions are used in this analysis:

(1) It is assumed that a reactor trip would be generated (either automatically or manually) upon significant fire initiation in all areas.

(2) A fire induced spuriously opened or stuck open relief valve will not inhibit the ability to achieve safe shutdown. The thermo-hydraulic analysis for LGS Safe Shutdown [EAS-26-0489] addresses this

issue.

(3) A period of 72 hours is assumed as the period for assessment for this analysis. This time is conservative with respect to the internal events analysis.

(4) Fire-induced disabling of the control room HVAC is not assumed to result in loss of control room habitability. The control room is constantly manned, and a heating or cooling failure would be corrected in a timely manner according to the applicable procedure.

(5) Motor control centers (MCCs) and other metal-enclosed components are not considered to be vulnerable to a low-intensity external exposure fire. However, unprotected cables entering and exiting the metal-enclosed component are considered to be vulnerable. Internal fires are conservatively assumed to render the entire MCC or cabinet and associated cables inoperable.

(6) The fire rated boundaries employed in the Appendix R analyses are used in this analysis. However, they are, examined to ensure consistency with the FIVE boundary criteria. Existing barrier deviations were evaluated for corrections.

(7) It is assumed that all automatic fire suppression systems credited in the analysis are properly designed and installed to effectively mitigate fires and that mitigation occurs at the time of system actuation.

(8) It is assumed that the fire brigade failure probabilities account for any fire-induced access difficulties. Fire brigade response times are compared with target damage times prior to assigning credit. Fire brigade response time is assumed to be equal to the manual fire suppression time.

(9) For all shutdown scenarios, off-site power may be utilized if it can be proven that a LOOP can not be induced as a result of the postulated fire.

(10) For any analyzed fire only one worst-case spurious actuation or signal is postulated (with the exception of Hi-Low pressure interfaces). Operator actions and repairs may be available to correct the actuation or signal or redundant equipment may be utilized in order to provide the required safe shutdown function. The analysis of spurious operations is identical to that performed for Appendix R

analyses.

(11) Hi-Low pressure interfaces are evaluated with regard to multiple hot shorts. All Hi-Low interface valves have been addressed by the Appendix R Studies and have been shown to perform their intended function for postulated fires.

(12) Potential modifications and operator actions have been added to the safe shutdown analysis in order to reduce the plant's reliance on the fire barrier material, Thermo-lag, but these changes are still under review by PECO Energy technical personnel. They are assumed to be acceptable and some are credited in this analysis. The modifications are described in Section 4.0.3 and the operator actions are discussed in Section 4.6.3.

(13) Fire barrier materials are credited in one compartment within Fire Area 02 for required safe shutdown systems to succeed.

(14) The unique fail-safe design, redundancy, and diversity of the systems associated with reactivity control, the Reactor Protection System (RPS) and Control Rod Drive (CRD) system, ensure that reactor shutdown can always be achieved.

(15) Compressed air is assumed to be lost, therefore, a source of compressed air will be provided for valves which must be repositioned in order to vent the suppression pool.

(16) Containment is inerted during operating, therefore, no fires are postulated in these areas.

A number of additional assumptions can be found in the specific discussion sections.

4.0.3 Status of Appendix R Modifications

For fire protection, LGS is committed to BTP CMEB 9.5.1, (ref. 4.0-1) "Guidelines for Fire Protection for Nuclear Power Plants." Appendix R (10CFR50, Appendix R) is used throughout the IPEEE guidance documentation. Therefore, to support correlation with the guidance documentation Appendix R will be used in place of BTP CMEB 9.5.1 throughout this document.

All safe shutdown modifications required for compliance with BTP CMEB 9.5.1 as documented in the UFSAR, Appendix 9A "Fire Protection

Evaluation Report" (FPER) have been completed.

The following additional modifications are proposed as a result of the LGS Thermo-lag reduction effort and have been credited in the IPEEE Internal Fires Analysis.

(1) To ensure A ESW Pump availability in additional compartments isolate the control circuit from analyzed faults on an annunciator cable. Normal and emergency pump operation will not be affected by this modification.

(2) Maintain "1A" diesel operability in additional compartments by preventing an analyzed control cable fault from inadvertently shutting down the diesel. This modification will not affect the ability of the diesel generator to perform its normal or emergency functions.

4.1 Fire Hazards Analysis

4.1.1 Critical Fire Compartments

Within the analysis, critical fire compartments are defined as those compartments which do not have a combined Fire Initiation (F_1) and Safe Shutdown Unavailability (P_2) of less than 1E-6, and therefore could not be screened from the analysis without performing a detailed fire/safe shutdown analysis.

At LGS, fire compartments were initially defined as the Appendix R fire areas. As required, these areas were redefined into fire compartments using the following criteria as defined by the FIVE Methodology:

(1) Boundaries consist of a 2-hour or 3-hour rated fire barrier on the basis of barrier effectiveness.

(2) Boundaries consist of a 1-hour rated fire barrier with a combustible loading in the exposing compartment < 80,000 Btu per sq. ft. on the basis of barrier effectiveness and combustible loading.

(3) Boundaries where the exposing compartment has a very low combustible loading < 20,000 Btu per sq. ft. and automatic fire detection on the basis that manual suppression will prevent fire spread to the adjacent compartment.

(4) Boundaries where both the exposing and exposed compartment have a very low combustible loading < 20,000 Btu per sq. ft. on the basis that a significant fire cannot develop in the area.

(5) Boundaries where automatic fire suppression is installed over combustibles in the exposing compartment on the basis that this will prevent fire spread to the adjacent compartment.

The fire compartment analysis produced the compartment breakdown (127 compartments) listed in Table 4.1-1. The shaded areas represent those compartments that did not meet the screening criteria and therefore required detailed analyses. Compartment numbers are initially based on the fire area and typically are subsets or combinations of fire areas.

TABLE 4.1-1

FIRE COMPARTMENT DESCRIPTIONS

FIRE COMPARTMENT	COMPARTMENT DESCRIPTION
1A	Corridor Areas
1B	Recombiner Rooms
1C	Recombiner Rooms
1D	Backwash Tank and Pump Rooms
1E	Recombiner Access Area
1F	H_2O_2 Analyzer Rooms
1G	H_2O_2 Analyzer Rooms
1H	H_2O_2 Analyzer Rooms
1I	H_2O_2 Analyzer Rooms
1J	H_2O_2 Analyzer Rooms
2	13 KV Switchgear Room
3	Battery Room
4	Battery Room
5	Battery Room
6	Battery Room
7	4 KV Switchgear Corridor
8	Battery Room
9	Battery Room
10	Battery Room
11	Battery Room
12	4 KV Switchgear Room
13	4 KV Switchgear Room
14	4 KV Switchgear Room

FIRE COMPARTMENT	COMPARTMENT DESCRIPTION
15	4 KV Switchgear Room
16	4 KV Switchgear Room
17	4 KV Switchgear Room
18	4 KV Switchgear Room
19	4 KV Switchgear Room
20	Unit 1 Static Inverter Room
21	Unit 2 Static Inverter Room
22	Unit 1 Cable Spreading Room
23	Unit 2 Cable Spreading Room
24	Control Room Area
25	Auxiliary Equipment Room (PGCC)
26	Remote Shutdown Panel Room
27	Control Structure Fan Room
28	SGTS Area
29	Unit 1 Suppression Chamber (Inerted)
30	Unit 1 Drywell (Inerted)
31	Unit 1 RHR Heat Exchanger and Pump Room
32	Unit 1 RHR Heat Exchanger and Pump Room
33	Unit 1 RCIC Pump Room
34	Unit 1 HPCI Pump Room
35	Unit 1 Core Spray Pump Room
36	Unit 1 Core Spray Pump Room
37	Unit 1 Core Spray Pump Room
38	Unit 1 Core Spray Pump Room
39	Unit 1 Sump Room and Passageway
40	Corridor
41	Unit 1 RECW Equipment Area
42	Unit 1 Safeguard System Access Area

FIRE COMPARTMENT	COMPARTMENT DESCRIPTION
43	Unit 1 Safeguard System Isolation Valve Area
44	Unit 1 Safeguard System Access Area
45	Unit 1 CRD Hydraulic Equipment Area
46	Unit 1 Main Steam Tunnel
47	Unit 1 Isolation Valve Compartment Areas
48A	Unit 1 Laydown and Corridor area
48B	Unit 1 RWCU Pump Area
49	Unit 1 Reactor Enclosure Fan Area
52	Unit 2 Suppression Chamber (Inerted)
53	Unit 2 Drywell (Inerted)
54	Unit 2 RHR Heat Exchanger and Pump Room
55	Unit 2 RHR Heat Exchanger and Pump Room
56	Unit 2 RCIC Pump Room
57	Unit 2 HPCI Pump Room
58	Unit 2 Core Spray Pump Room
59	Unit 2 Core Spray Pump Room
60	Unit 2 Core Spray Pump Room
61	Unit 2 Core Spray Pump Room
62	Unit 2 Sump Room and Passageway
63	Corridor
64	Unit 2 RECW Equipment Area
65	Unit 2 Safeguard System Access Area
66	Unit 2 System Isolation Valve Area
67	Unit 2 Safeguard System Access Area
68	Unit 2 CRD Hydraulic Equipment Area
69	Unit 2 Main Steam Tunnel
70	Unit 2 Isolation Valve Compartment Area
71A	Unit 2 Laydown and Corridor Area

FIRE COMPARTMENT	COMPARTMENT DESCRIPTION
71B	Unit 2 RWCU Pump Area
72	Unit 2 Reactor Enclosure Fan Area
75	Service Water Pipe Tunnel
76	Refueling Hoistway
77	South Exhaust Stack
78	Refueling Area
79A	Diesel Generator Cell
79B	Diesel Generator Tank Area
80A	Diesel Generator Cell
80B	Diesel Generator Tank Area
81A	Diesel Generator Cell
81B	Diesel Generator Tank Area
82A	Diesel Generator Cell
82B	Diesel Generator Tank Area
83A	Diesel Generator Cell
83B	Diesel Generator Tank Area
84A	Diesel Generator Cell
84B	Diesel Generator Tank Area
85A	Diesel Generator Cell
85B	Diesel Generator Tank Area
86A	Diesel Generator Cell
86B	Diesel Generator Tank Area
87A/B/C	Condensate Pump Rooms, Generator Equipment Areas, Operating Floor
90	Turbine Building Areas
91	Turbine Building Areas
96	Battery Room
103	Turbine Building Areas
104	Turbine Building Areas

FIRE COMPARTMENT	COMPARTMENT DESCRIPTION
109	Battery Room
115	Radwaste Pipe Tunnel
116	Radwaste Building
122	Unit 1 Spray Pond Pump Structure
123	Unit 2 Spray Pond Pump Structure
124	Unit 1 Diesel Generator Access Corridor
125	Unit 2 Diesel Generator Access Corridor
126	North Stack Instrument Area
127	South Stack Instrument Area
YARD A	Manholes associated with A/C Compartments
YARD B	Manholes associated with B/D Compartments
YARD C	Diesel Fuel Storage and Transfer Area
YARD D	Main Transformer Yard Area
YARD E	Water Treatment Area
YARD F	Chlorine/Water Treatment Area
YARD G	Holding Pond
YARD H	Schuylkill River Pump House
YARD I	Technical Support Center
YARD J	Circulating water Pump House
YARD K	Aux Boiler/Admin Building
YARD L	Sewage Treatment Building
YARD M	H_2 Storage Area

TABLE 4.1-2

FIRE COMPARTMENTS BY PLANT-SPECIFIC LOCATIONS

PLANT LOCATION	NUMBER OF FIRE COMPARTMENTS
Reactor Building	62*
Turbine Building	7
Radwaste	1
Switchgear	13
Battery	10
Cable Spreading Room	2
Control Room	1
Diesel Generator	16
Intake	2
Yard	13
TOTAL	127

* Does not include containment areas which are inerted.

4.1.2 Fire Initiation Database

For estimating fire ignition frequencies the FIVE methodology provides a generic database of about 800 fire events that occurred in U.S. nuclear generating units during the period 1965-1988. The FIVE process includes Ignition Source Data Sheets (ISDS) to facilitate estimating plant specific fire ignition frequencies from the generic data.

The data from the generic database provides frequencies for different fire ignition sources (equipment) by specific plant locations. Some of the components, such as air compressors, ventilation subsystems, or hydrogen tanks, are treated as plant-wide components (not connected to any specific location). Transient fires and fires caused by welding are also treated as plant-wide fires.

In order to determine the Fire Frequency (F_1) for each fire compartment at Limerick, a detailed analysis following the guidelines of the FIVE Methodology was performed. This analysis involved the identification of each ignition source in the fire compartment. This information was collated using the plant fire hazards analysis, equipment location drawings, and the Plant Information Management System (PIMS) database. Major equipment locations were confirmed using plant walkdowns. Transient ignition sources were identified by calculating a generic number (see section 4.4.1.2) which was used for all fire compartments at Limerick.

The process of calculating F_1 for each fire compartment at Limerick consisted of the following steps.

(1) <u>Select a Location and Applicable Weighting Factor</u>

(2) <u>Identify Sources</u>

(3) <u>Select Fire Frequency</u>

(4) <u>Identify Source Weighing Factor</u>

(5) <u>Determine Transient Source Factor</u> - Using the information provided in the methodology, a "generic" transient factor was determined for all plant compartments. Per plant Administrative Controls, no cigarette smoking or use of candles are permitted in plant structures; therefore, they are not included in the transient calculation. The remaining

transients, Extension Cord, Heater Overheating, and Hot Pipe were assumed to be allowed in all areas. Using these assumptions and following the methodology, a "generic" transient ignition source factor of 7.87E-2 was calculated.

(6) Calculate Ignition Source Factor (F_1)

The total quantity and location of the various types of Fire Ignition/Fuel Source components were extracted from the Component Record List (CRL) in PIMS. Plant Equipment arrangement and Layout drawings were used as necessary to identify equipment and locations.

The yard was divided into 13 areas as described Table 4.1-1. A report was generated from the CRL to list all applicable equipment in these areas. The equipment totals were then tabulated for each yard area.

The plant wide equipment totals were then tabulated for each fire compartment. A summary of the equipment totals by type is listed in Table 4.1-3.

TABLE 4.1-3

PLANT WIDE EQUIPMENT TOTALS BY TYPE

Equipment Type	Total
Fire Protection Panels	106
Reactor Protection System (RPS) M/G Sets	4
Transformers	257
Battery Chargers	23
Offgas/H_2 Recombiners	6
H_2 Tanks	12
Air Compressors	70
Ventilation Subsystems	521
Electrical Cabinets	1557 Turbine Enclosure 1718 Reactor Enclosure
Pumps	153 Turbine Enclosure 137 Reactor Enclosure

4.2 Review of Plant Information and Walkdown

Information required for inputs into the analysis were taken from plant documentation such as procedures, databases, and drawings. Specific documentation used within the analysis is identified under the individual headings as appropriate.

Walkdowns, as performed, were confirmatory in nature. Walkdowns were performed to verify equipment location, area dimensions, and other information as needed to support the fire damage analysis and the screening of fire compartments. The discussion of Fire Scoping Study issues in Section 4.8 provides walkdown information appropriate to those issues. Walkdowns were performed in accordance with PECO Energy specific procedures.

4.2.1 Cable Routing Information Verification

The accuracy of the cable routing and equipment location information is important since it provides much of the basis for the fire analysis. PECO Energy's cable management system is a living database and is used in the performance of day-to-day cable/raceway design and installation activities as well as Appendix R analysis.

Changes to the database are controlled by engineering procedures which require that all changes be originated, reviewed, and approved. Historically, the safe shutdown raceway room location data has been reviewed against the applicable plant layout drawings; when required, walkdowns have been performed to assure raceway location data correctness. The software electronically assigns the raceway room locations to the cables routed in the raceway.

To perform fire area safe shutdown analyses the same software, using the cable management database, electronically identifies safe shutdown cable failures based on room locations.

The Condensate system in the injection mode was the only system that cable requirements were not identified per Appendix R rules. The cable requirements were identified by using the electrical elementary scheme identifiers for all associated equipment schemes. The scheme identifiers are imbedded in the associated cable numbers. Using the automated cable management system the condensate system associated cables/raceways were identified. Raceway layout drawings were reviewed to ensure the

condensate system cables/raceway were not located in the areas where the system was taken credit for surviving.

4.2.1.1 Control of Appendix R Information

The PECO Energy cable management system is a controlled database with the capability to perform tasks required to analyze Appendix R compliance. To support Appendix R analysis the database contains the safe shutdown method, systems, component, cable logics, and fire area and room data. The software has the capability of performing analyses on a fire area, room, or combination of rooms basis.

Using Boolean logic the software provides a place to document the relationships between safe shutdown methods, systems, equipment, and cables. In the analysis mode the software utilizes the logics along with component and cable locations to identify the equipment, cables, systems, and methods that do not survive for a fire in a given fire area. Also provided is the identification of the course for a safe shutdown entity fail: location, supporting cable or logical relationships. The software provides the ability to perform an analysis for shutdown from the control room or the alternative control stations.

4.2.1.2 Update of Appendix R Databases

The normal update process of the safe shutdown database is continuous via the proceduralized modification and engineering change processes, therefore, the as-designed and as-built configuration of the plant is maintained. However, the safe shutdown compliance configuration as delineated in the UFSAR, Appendix 9A (FPER) is representative of the as-built configuration of the plant.

The information utilized in performing the safe shutdown system analysis for the IPEEE Internal Fires Analysis was current to July, 1994.

4.2.2 Control Room/Remote Shutdown Circuit Dependencies

Compliance with Appendix R alternative shutdown requirements is achieved with isolation and control transfer capabilities on the Remote Shutdown Panel (RSP) and a few local control stations. Alternative shutdown control power supplies meet the requirements of IN 85-09, (ref. 4.2-1) Isolation Transfer Switches and Post Fire Shutdown Capability. The Appendix R

study of remote transfer capability identified and documented the resolution of issues associated with alternative shutdown compliance. The alternative safe shutdown methodology (Method R) is based on equipment associated with electrical Division 1 power, whereas, the method used for a remote shutdown room fire is safe shutdown Method D which is based on equipment associated with electrical Division 2.

4.2.3 Walkdown Team

All walkdowns were performed using a walkdown procedure developed for the project. Procedure attachments identify the walkdown purpose, requirements for walkdown personnel, and detailed instructions on the information to be obtained. Composition of the team for each walkdown was specific to the information being obtained. Teams consisted of a minimum of two persons such that all data gathered was originated and reviewed. Personnel conducting the walkdowns had either generated the information being confirmed during the walkdown or were trained as appropriate to gather the information requested by the walkdown initiator. Information obtained during the walkdowns was either recorded on attachments to the procedure or on other appropriate documentation as deemed necessary by the person requesting the information.

4.2.4 Walkdown Findings

Walkdowns, as performed, were confirmatory in nature. The objective of performing walkdowns was to assure by physical inspection that data gleaned from existing documentation is representative of the physical condition of the plant. In cases where no specific documentation existed (e.g. actual transient combustible loading in an area), all information was gathered in the field. Walkdown data was either recorded on attachments to the procedure or documented in a manner suitable for the specific task. The following walkdowns were performed:

- Fire Ignition Source locations and quantities

  Walkdowns conducted to confirm equipment locations taken from PIMS and plant drawings. This information was used for the calculation of F_1.

- Fire Source locations and quantities

  Walkdowns conducted to confirm fire source locations and quantify relative amounts of combustibles. This information was used for fire

modeling.

- Fire Compartment Boundary Verification

  Walkdowns conducted to confirm assumptions and information used to compartmentalize fire areas.

- CFZ Boundary Confirmation

  Walkdowns conducted to verify that the CFZ boundaries are properly identified in the plant.

- SOD Walkdowns

  Walkdowns conducted to identify all intervening combustibles and verify targets within every calculated SOD. This information was used for fire modeling.

- Transient and Fixed Combustible Review

  Walkdowns conducted to verify location and quantify all fixed combustibles and expected transient combustibles within each critical fire compartment. This information was used as the basis for fire modeling in the compartment.

- Sensitive Electrical Equipment Verification

  Walkdowns were conducted to confirm that sensitive electrical equipment was well outside of the boundary of a SOD. This verification was performed to assure the equipment would not fail or cause a trip of equipment credited during the target analysis (section 4.4.0).

The discussion of Fire Scoping Study issues in Section 4.8 provides walkdown information appropriate to those issues. Walkdowns were performed in accordance with PECO Energy specific procedures.

4.3 Fire Growth and Propagation

For those fire compartments that did not meet the screening criteria of the analysis during the F_2 calculation phase, the FIVE Methodology provided qualitative fire analysis and modeling to evaluate fire damage probabilities to specific targets within a fire compartment. The methodology involved analysis of fire damage probabilities from both fixed and transient fire sources and evaluated damage from direct plume involvement, hot gas layer entrainment, and radiant energy effects of the fire source.

4.3.1 Modeling

Fire modeling was performed using the worksheets in the FIVE Methodology as guidelines. Using these worksheets, three possible fire scenarios were modeled for each fire source identified. Those scenarios were:

(1) In-the-Plume

(2) Out-of-Plume (Hot Gas Layer)

(3) Radiant Energy

The initial steps in the modeling process involved identifying fire size and growth potential, fire duration, and the "Spheres of Damage" (SODs) around each fuel source. Details of these items are discussed in the following sections.

4.3.2 Fire Size and Duration

The recommendations in the FIVE Methodology were used to calculate the type, size, and duration of each fire source in a fire compartment. The steps involved were:

(1) Identifying the potential fire sources in each critical fire compartment

(2) Determining the potential fire source as an actual source and type and amount of fuel

(3) Calculating the fire size and duration due to fire source

4.3.2.1 Identification of Potential Fuel Sources

Potential fuel sources were identified through the use of plant documentation. Table 9A-1 of the Limerick Fire Protection Evaluation Report (FPER) in addition to the specific compartment fire hazard evaluations within the FPER were used to identify fixed and expected transient combustibles. Plant equipment location drawings were also used to identify locations of plant equipment, which were a potential fire source (i.e. pumps, switchgear, MCCs, etc.). These potential fire sources were listed on the applicable FIVE tables and highlighted on drawings of each critical fire compartment.

4.3.2.2 Determination of Actual Fire Sources

Walkdowns of each critical fire compartment were performed to confirm the location and applicability of each fire source. These walkdowns were also used to confirm the existence and location of temporary/permanent storage locations throughout the plant as identified in accordance with Procedure A-30 "Plant Housekeeping". Equipment determined to present an actual fire source as classified by type, and an amount of fuel was then calculated or assumed.

4.3.2.3 Calculation of Fire Size and Duration

Once all actual fuel sources were identified, the type and amount of combustible were determined. Fixed fuel sources were classified as either electrical cabinets (cable insulation) or lube oil. For transient combustibles, a "typical" plant transient was calculated as representative of other plant transient material. The amounts of combustibles for each of these fuel sources and fire size were determined as follows. Fire size is identified as the heat release rate (HRR) of the fire source measured in British Thermal Units per second (BTU/sec). The amount of fuel or potential total energy release of the source is measured in BTU.

Electrical Cabinets

The following methodology was developed to determine the HRR and BTU content of electrical cabinets resulting from cable loading within the cabinets.

Part 1: Calculate HRR

- Assume that each cabinet is 8 ft. high; therefore, there is 8 ft of combustible per cable.

- Derate the amount of cable by a derating factor of 0.4 using the method from the National Fire Protection Handbook, Chapter 6, (ref. 4.3-1) on derated fire loads:

- Based on a review of plant data the average cable diameter is 0.6 in.

- Section 5.3 of Attachment 10.4 of the FIVE methodology states that "the pyrolyzing area (of the intervening combustible) can be set equal to the area of the fire source that is exposing the cable tray". It is assumed that the pyrolyzing area is equal to the area of the "profile" of the cable within the cabinet; therefore, A_E is determined by multiplying the cable diameter by the derated length. This value was multiplied by the ratio to determine the specific profile for a cabinet.

$$\frac{\text{actual cabinet height (ft)}}{8 \text{ ft.}}$$

<u>Part 2: Determine the Heat Release Rate/ft^2 of the Cable</u>

- From the National Bureau of Standards Information Report entitled "Heat Release Rate Characteristics of Some Combustible Fuel Sources in Nuclear Power Plants", Table 1 on page 17, (ref. 4.3-2) assume a cable HRR of 258 kW/m^2 (22.7 BTU/sec·ft^2).

Combining the information from parts 1 and 2 results in:

HRR = (0.16 ft^2/cable) x (22.7 BTU/sec·ft^2) x (No. of cables in cabinet)

or

HRR = (3.63 BTU/sec·cable) x (No. of cables in cabinet) = <u>X</u> BTU/sec

<u>Part 3: Calculate Cabinet BTU Content</u>

Cabinet BTU content calculations were performed using information from the cable management system. Running this program for each electrical cabinet produced the following data:

- Cable Designation
- To and From Destinations
- Basic Cable Code

Using this information the combustible loading per ft. of each cable in a cabinet was obtained from Attachment I of Limerick calculation LE-48, assuming a heat of combustion for cable insulation of 10,000 BTU/lb.

The BTU content for each cable type in the cabinet was calculated using the following formula:

BTU Content = (Number of cables)(BTU/Ft)(Cabinet Height)

The total BTU content of the cabinet was obtained by adding the BTU content contribution of each cable type together. Therefore, the equation for total cabinet BTU content used is:

Total Cabinet BTU Content = Σ (Number of Cables)(BTU/Ft Cable)(Cabinet Height in feet)

= X BTU

Lube Oil

Lube oil spills were calculated using the amount of oil as shown on the plant lube oil schedule. Spill sizes and heat release rates shown below were calculated following the guidance of the FIVE Methodology (Tables 2E and 3).

TABLE 4.3-1

LUBE OIL AND HEAT RELEASE RATES

Equipment	Heat Release Rate (BTU/sec)	BTU Content (BTU)
Condensate Pumps	214,650	8,168,253
RECW Pumps	1,013	7,854
SLC Pumps	36,450	282,748
Instrument Gas Compressors	7,088	55,042

Transient Combustibles

In order to determine the hazard posed by transient combustibles, a "largest" expected transient combustible was determined, then this combustible was evaluated for its affect on targets. The determination of the "largest" transient fire was as follows.

Description of Hazards

Transient combustibles expected at LGS fall into four basic categories:

(1) Flammable/Combustible Liquids

(2) Plant Trash

(3) Anti-Contamination Clothing

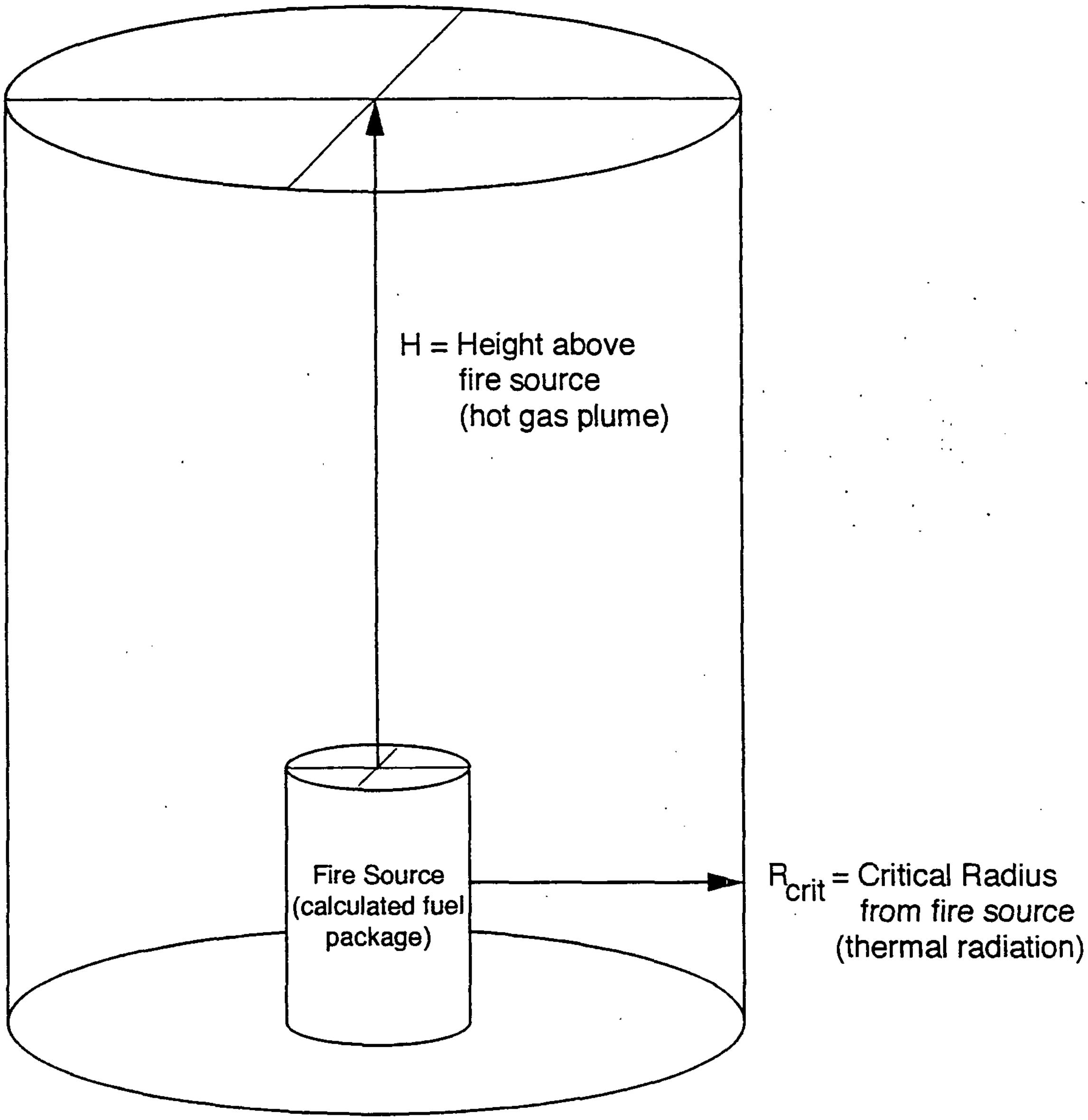
(4) Wood/Paper Materials

Per the assumptions in the FIVE Methodology flammable/ combustible liquids that are required per plant administrative procedure to be handled in approved flammable liquid cans are assumed to not be exposed. Therefore, they will not be considered as a possible transient combustible. The analysis determined that plant trash and anti-contamination clothing consist of similar materials and represent the most severe transient fire exposure. HRR and BTU content calculated are 226 BTU/sec and 1,162,086 BTU respectively.

4.3.2.3.1 Spheres of Damage (SODs)

In order to quantify the area around a particular fire source in which targets could be damaged, spheres of damage (SOD) were developed (See Figure 4.3-1). These SODs are based on the vertical height above a fuel source at which the expected temperature rise due to the plume is below the FIVE Methodology criteria of 700°F. The horizontal distance is based on the distance at which the radiant flux to a target is below the FIVE Methodology criteria of 1 BTU/s ft^2. The use of these SODs enabled the creation of maps of each fire source to be developed which could identify all target raceway and intervening combustibles within each fire source SOD. This allowed for quick determination of equipment lost and intervening combustibles involved for each specific fire scenario. SOD distances were calculated following the equations within the FIVE methodology.

FIGURE 4.3-1
SPHERE OF DAMAGE (SOD) DUE TO THERMAL RADIATION AND HOT GAS PLUME



4.3.2.3.2 Intervening Combustibles

Intervening combustibles are those combustibles which could be ignited by either a fixed or transient fire source, thus adding to the heat release rate and BTU content of the fire. These combustibles have been defined as electrical cable in both trays and gutter. Plant administrative controls provide guidance for the placement of transients in relation to these combustibles, therefore, it is assumed that there will not be involvement of intervening combustibles with a transient fire. Intervening combustibles have been evaluated for each fixed fire source within a critical fire compartment. The methodology used in analyzing the heat release rate and BTU contribution from them is as follows.

(1) HRR for cable trays and gutters is based on a bench scale HRR of 22.7 BTU/sec·ft^2.

(2) Full scale HRR is estimated to be 45% of the bench scale rate based on equation 2 in section 5.3 of the FIVE methodology; therefore, the HRR_{FS}= 10.2 BTU/sec·ft^2.

(3) Section 5.3 of the FIVE methodology user's guide states that "the pyrolyzing area (of the intervening combustible) can be set equal to the area of the fire source that is exposing the cable tray". It is assumed that the pyrolyzing area is equal to the area of the tray or gutter within the SOD of the fire source.

(4) BTU content of intervening combustibles is calculated assuming a full (40%) tray or gutter with C01 cable. See Table 4.3-2 for BTU/ft of specific tray and gutter sizes.

The heat content per unit length is determined by multiplying the number of cables by 677.0 BTU/ft·cable. 677.0 BTU/ft is obtained from the insulation weight, 0.0677 lb/ft from PECO Energy calculation LE-0048, multiplied by 10,000 BTU/lb.

(5) HRR totals and elevation of fire source due to the addition of intervening combustibles is as described in example 7 in section 10.4 of the FIVE methodology:

- Intervening combustible HRR added to source HRR
- Source elevation taken as elevation of intervening combustible

TABLE 4.3.-2

INTERVENING COMBUSTIBLE BTU/FT TRAY OR GUTTER

BASED ON CABLE TRAY OR GUTTER AT 40% FILL		
CABLE TRAY WIDTH (in)	NUMBER OF CABLES	BTU/ft
6	45	30,465
12	90	60,930
18	135	91,395
24	180	121,860
30	225	152,325
36	270	182,790
GUTTER SIZE (in x in)	NUMBER OF CABLES	BTU/ft
2 x 3	15	10,155
2.5 x 2.5	16	10,832
4 x 4	40	27,080
6 x 6	90	60,930
8 x 8	160	108,320
12 x 12	360	243,720
2 x 8	40	27,080
6 x 24	360	243,720

4.3.3 Cross Zone Fire Spread

The FIVE Methodology provides for analysis of cross zone (fire compartment) fire spread for zones separated by barriers. This analysis is discussed in Section 4.1.1 of this report. Those compartments in the Unit 1 and Unit 2 reactor buildings which directly communicate to each other due to the equipment hatchway are conservatively assumed to contain and prevent smoke and hot gas spread into this opening. This is conservative because it maximizes fire and hot gas temperatures within the compartment of fire origin.

The FIVE Methodology Section 6.2.1 discusses spatial separation in conformance with 10CFR50 Appendix R requirement III.G.2.b (20 ft. combustible free zones), stating that this arrangement should be adequate to prevent fire propagation across the zone. To verify this assumption for conditions at Limerick, analyses were performed on the nine combustible free zones (CFZs) in both the Unit 1 and Unit 2 reactor buildings. These analyses assumed the transient fire placed directly at one edge of the CFZ and calculated the radiant and hot gas layer thermal affects on the far edge of the zone. The analyses showed no propagation due to transients was anticipated. Calculations were also performed using the most severe fixed fire source at each CFZ location. These calculations also showed that with the spatial separation and available automatic detection and suppression, no fire propagation across the zone is anticipated. As a result of these conclusions, the reactor building fire compartments containing CFZs were separated into east and west zones for the purposes of calculating the unavailability of redundant safe shutdown equipment (P_2).

4.3.4 Spread of Smoke and Hot Gases

The FIVE Methodology conservatively assumes that any fire in a critical compartment will result in damage to all targets within the compartment. For compartments that do not screen ($F_2 > 1E\text{-}6$) using this assumption, the methodology provides for the detailed fire modeling process as described in previous sections. The modeling accounts for the thermal affects of smoke and hot gases generated by the fire in both the plume and hot gas layer scenarios.

4.4 Evaluation of Component Fragilities and Failure Modes

The FIVE methodology was used to evaluate fire growth and propagation. This screening methodology provides a means to make conservative estimates about conditions that could develop at a target as a result of a specified fire. These conditions were then compared with target damage threshold criteria (temperature or heat flux) and if the criteria were not exceeded, the specified fire was screened from further analysis. Otherwise, more analysis was required. If the specified fire led to a fire scenario which is an important contributor to the plant risk more detailed fire growth and propagation analysis was needed.

4.4.0 Selection of Targets for Safe Shutdown

Targets were selected in those fire compartments where prior attempts to reduce P_2 below the screening threshold was not possible. After a hot gas layer volume was developed by extending the calculated hot gas layer from floor to ceiling the raceway and equipment within that volume was determined. This conservative list of affected raceway and enabled the identification of raceway and equipment which were not affected by the extended hot gas layer and would support safe shutdown system success. Multiple electrical divisions of raceway were identified outside of the hot gas layer volume. Redundancy was ensured in successful safe shutdown methods and systems by targeting multiple divisions. Conservative assumptions in the hot gas layer volume calculation ensured that greater amounts of equipment could be shown to not reach the damage threshold, if required. The location of sensitive electrical equipment was also verified to assure failure would not occur because of the lower damage threshold typically associated with this type of equipment.

4.4.1 Fire Damage Modeling

The fire damage modeling began with the generation of the SOD for each identified fuel source in each critical fire compartment. The methodology used to develop these SODs was discussed in section 4.3.2.3.1 of this report. These SODs were developed to bound two of the three fire scenarios possible for each fuel source. Scenarios considered are:

(1) Targets located in the plume, directly above the fire source.

(2) Targets located in the hot gas layer.

(3) Targets located lateral to the fuel source, subjected to thermal radiation.

Using the FIVE Methodology damage threshold criteria of 700°F and 1 BTU/sec.ft.^{2} the SODs bounded scenarios 1 and 3 for target analysis for both fixed and transient fuel sources. (See Figure 4.3-1)

A bounding calculation was also performed on the hot gas layer scenario for each fire source. A target was assumed to exist at the plume/hot gas layer interface which is the worst case target location as the target receives the thermal affects of both the plume and hot gas layer subject. This calculation was completed to see if target damage could result due to the combined affect of the fire heat release and total BTU content. If the time to fire damage exceeded that of the fire duration, no damage to ceiling targets was postulated. If damage could occur an additional calculation comparing time to damage versus fire brigade response time was completed. This calculation gave a bounding distance at which target damage would be precluded by manual fire extinguishment. These calculations also assumed a target damage threshold of 700°F and 1 BTU/sec.ft^2.

The analyses required collection of data for the following parameters.

(1) Location of targets relative to fire sources.

(2) Damage threshold criteria for targets.

(3) Fire intensity and total energy content.

(4) Fire compartment volume and construction.

(5) Fire brigade response times and drill history.

(6) Fire suppression system information.

(7) Fire detection system information.

4.4.1.1 Fixed Combustible Modeling

Four fire compartments were identified as not containing ignition sources or fixed combustibles except for electrical cables. These compartments are:

- Fire Compartment 1E - Recombiner Access Area
- Fire Compartment 7 - 4 KV Switchgear Room Corridor
- Fire Compartment 22 - Unit 1 Cable Spreading Room
- Fire Compartment 23 - Unit 2 Cable Spreading Room

Because no ignition sources or fixed combustibles except electrical cables were present, the probability of fixed combustible exposure damage (P_f) was set equal to zero (P_f=0). These four compartments were then analyzed for transient fuel sources. Disposition of these compartments is discussed in Section 4.4.1.2 on transient combustible modeling.

The remaining 7 fire compartments listed below were analyzed for the probability of fixed combustible fire source damage.

- Fire Compartment 44 - Unit 1 Safeguard System Access Area
- Fire Compartment 45 - Unit 1 CRD Hydraulic Equipment Area
- Fire Compartment 47 - Unit 1 Isolation Valve Compt. Area
- Fire Compartment 64 - Unit 2 RECW Equipment Area
- Fire Compartment 67 - Unit 2 Safeguard System Access Area
- Fire Compartment 68 - Unit 2 CRD Hydraulic Equipment Area
- Fire Compartment 70 - Unit 2 Isolation Valve Compt. Area

Targets identified in these areas were required for safe shutdown success based on two selected divisions of safe shutdown equipment. Because of the physical relationship between the specified targets and the fixed combustibles no damage is anticipated due to fixed combustibles, in these fire compartments. Therefore, the probability of fixed combustible exposure damage (P_f) in these areas was set equal to zero (P_f = 0).

As a result of the SOD calculations 5 fire compartments could not be screened due to the size of the resultant SOD, and the inability of fire modeling to show reasonable probability of crediting redundant systems back into the analysis. The following compartments were removed from further analysis:

- Fire Compartment 2 - 13.2 KV Switchgear Room
- Fire Compartment 20 - Unit 1 Static Inverter Room
- Fire Compartment 24 - Control Room
- Fire Compartment 25 - Aux Equipment Room (PGCC)
- Fire Compartment 26 - Remote Shutdown Panel Room

A discussion on the disposition of these five fire compartments is detailed in Section 4.6.5 of this report.

4.4.1.2 Transient Combustible Modeling

The four critical fire compartments which were identified as not containing fixed combustibles were evaluated for transient combustible exposure damage. Because of the physical arrangement of the redundant safe shutdown equipment in the room it was not possible to preclude damage by location of combustibles or fire suppression (either automatic or manual). Further evaluation of the compartments led to the development of administrative controls on transient combustibles as outlined in the FIVE Methodology Section 6.3.3 to preclude transient combustible exposure damage. Because of these controls and the assumptions allowed by the FIVE Methodology, the probability of transient combustible exposure damage (P_{tc}) was set equal to zero ($P_{tc} = 0$). Since P_f for these four compartments also equalled zero, these compartments were screened from the analysis on the basis of no exposed combustibles.

The remaining seven fire compartments were analyzed for the probability of transient combustible exposure damage. The P_{tc} calculation involved the following steps:

(1) Calculation of probability of manual suppression, P_{ms}.

(2) Determination of probability of automatic suppression, P_{as}.

(3) Calculation of probability of transient combustible being located where it can cause damage, u.

(4) Determination of probability of transient combustible being exposed, p.

(5) Determination of probability of having critical amounts of transient combustibles between periodic inspections, w.

(6) Calculation of P_{tc}, where:

$$P_{tc} = (1-P_{ms}) \cdot P_{as} \cdot u \cdot p \cdot w$$

These steps are discussed in detail in the following sections.

Calculation of Probability of Manual Suppression (P_{ms})

The calculation of probability of manual suppression consists of the comparison of two numbers, the first being the time required for the fire brigade to respond to and control a fire, and the second being the time to target damage as a result of the fire. If the fire brigade can respond to and control the expected transient fire before the fire can damage the target under consideration, credit can be taken for fire brigade response and a probability for this response added into the P_{tc} equation. If the brigade response time is greater than that required for target damage the probability of manual suppression is assigned a value of 1.0.

A review of fire brigade drills conducted at LGS from 1987 up to and including the second quarter of 1994 was performed, and fire brigade response maps were plotted for all critical areas in the Unit 1 and Unit 2 reactor buildings. From these maps the fire brigade response time could be predicted for these areas along with a historical record of brigade drills for each compartment.

The fire brigade can only respond to a fire situation in the plant once they have been alerted as to the presence and location of a fire, therefore, the time to fire detection (t_d) must be calculated and added to the fire brigade response time to obtain the total time required for fire brigade response. Using detection system drawings the detector spacing for each of the critical fire areas was obtained and as a worst case the transient fire was placed at the detector spacing midpoint. This allows for conservatism in the calculation by producing the longest detector response time.

Comparing the time to detection plus time to response with the time to target damage demonstrated that in each case the fire brigade can respond to and control the fire before the target is damaged.

Therefore, fire brigade response can be credited in the analysis of transient exposure damage. Because of the guidelines within the FIVE methodology the probability of manual suppression (P_{ms}) cannot be less than 0.1, therefore, for this analysis P_{ms} for transient exposure has been set equal to 0.1 (P_{ms} = 0.1).

No credit has been taken in the analysis for automatic suppression because none of the critical fire compartments are protected by area wide automatic suppression systems. The following factors were calculated in accordance with the FIVE methodology:

(1) Probability of transient combustible being located were it can cause damage (u)

(2) Probability of transient combustibles being exposed (p)

(3) Probability of having critical amounts of transient combustibles between periodic inspections (w)

(4) Transient combustible fire exposure (P_{tc})

4.4.1.3 Calculation of Probability of Fire Damage (P_3 and F_3)

The final step in the fire modeling phase of the analysis is the calculation of P_3. The probability of fire damage to the target is the probability of fixed exposure damage plus the probability of transient exposure damage.

$$P_3 = P_f + P_{tc}$$

As discussed in section 4.4.1.1 of this report, the probability of damage due to a fixed fire exposure in the critical fire areas is zero, therefore, the P_3 value will be equal to P_{tc}. These values were then multiplied by the F_2 values calculated for each compartment to determine the value of the overall frequency of a fire occurring in a compartment and damaging safe shutdown components (F_3). The F_3 values calculated for the critical fire compartments are shown in Table 4.4-1.

TABLE 4.4-1
PROBABILITY OF FIRE DAMAGE

FIRE COMPARTMENT	F_2	P_3	F_3
44	6.9E-3	1.59E-3	1.1E-5
45	5.1E-3	9.15E-4	4.7E-6
47	6.6E-2	8.11E-4	5.4E-5
64	1.3E-4	1.14E-4	1.5E-8
67	7.1E-3	3.74E-3	2.7E-5
68	4.7E-3	6.42E-4	3.0E-6
70	6.6E-2	8.89E-4	5.9E-5

Additional probabilistic analyses were performed for these seven areas to determine the unavailability of systems within the fire compartments (P_4) that were analyzed to survive a given fire. A description of the process of targeting specific equipment in a compartment and the calculation of P_4 can be found in Section 4.6.0.3. Table 4.6-1 provides the P_4 results for those compartments.

4.4.2 Combination of Fire Induced and Non Fire-Induced Failures

Boolean logic is utilized in the cable and raceway management database to establish relationships between the safe shutdown methods, systems, components, and cables. This allows the relationships to be broken down into relatively small steps to simplify the development and utilization of the results of the analysis. The logical relationships are specified through the use of "AND" and "OR" gates; each relationship represents a logic step. If a logic step fails, then the safe shutdown feature requiring that logic step will also fail. The propagation of plant equipment failures, whether initiated by cable failures or other logic step failures will continue until all affected safe shutdown systems and methods are either failed or are determined to be unaffected by the fire.

4.4.2.1 Failure Modes of Cable and Equipment

For determining safe shutdown system success any cables within the fire compartment of concern were considered damaged and therefore failed. The failure modes of the cables were analyzed because of the potential effect outside the fire compartment of origin. The following failure modes were analyzed for cables which were determined to be damaged:

Short	-	Individual conductors within a cable short to each other.
Ground	-	Individual conductors within a cable are grounded to the supporting raceway or other ground structure.
Open	-	Individual conductors within a cable lose electrical continuity.
Hot Short	-	Individual conductors within a cable are shorted to individual conductors of a different cable. This type of short includes the case of one de-energized circuit becoming energized by shorting to an external source of electrical power through independent conductor-to-conductor shorts.

Plant equipment fails for any combination of four (4) reasons:

Location	-	The equipment is located in the fire of concern.
Failed Support Equipment	-	Equipment required to support the equipment of concern fails.
Failed Cable	-	Cables required to support the equipment of concern fail.

4.5 Fire Detection and Suppression

The fire modeling portion of this analysis credits both fire detection and fire suppression as factors in mitigating fire damage to safe shutdown targets. Suppression is credited in those cases where fire brigade response provides for manual suppression of the fire. Automatic fire suppression is not credited in the analysis of either fixed or transient fire exposure damage due to the lack of area-wide automatic suppression systems within the critical fire compartments.

4.5.1 Types of Fire Mitigation Actions Credited

Within the analysis, credit has been taken for the following actions in mitigating transient fire damage within a compartment.

(1) Fire Detection

(2) Manual Fire Suppression - Fire Brigade Response

Fire Detection

All of the critical fire compartments are provided with area wide detection systems designed to provide early warning to the Control Room of a fire situation within the plant allowing for response and fire control by the plant fire brigade. Detection systems have been designed and installed per the requirements of NFPA 72A and D with noted exceptions to provide prompt notification to the Control Room in the event of a fire. The detection systems at LGS were analyzed per the requirements of NFPA 72E, 1987 and found acceptable in a report issued in 1988. The area detection systems at LGS utilize smoke detectors arranged throughout the area with distance limitations as per manufacturers data and NFPA 72E requirements.

Fire Suppression

Automatic fire suppression is not credited within the analysis. Manual fire suppression is credited within the analysis as providing fire extinguishment of transient fire exposures to prevent target damage. Fire control and extinguishment is assumed to occur concurrent with the arrival of the plant fire brigade to the fire scene.

4.5.2 Fire Fighting Procedures

Upon being alerted to a fire situation at LGS, Control Room personnel enter into special event procedure SE-8, Fire. This procedure is designed to provide guidance to plant personnel in the event of a fire, and is to be used with plant operating and trip procedures to place the unit in a safe condition. In the event of a fire in the plant, specific safe shutdown methodologies are detailed in attachments to the main procedure.

Guidance in the F-series procedures reference the pre-fire-plan on a per fire area basis. These pre-fire-plans provide area specific information on:

- Equipment in the area
- Fire hazards and appropriate extinguishing agents
- Access routes-both preferred and secondary
- Available fire suppression equipment-both automatic and manual
- Ventilation paths
- Hazards
- Proposed fire fighting strategies
- Systems requiring management
- Safe shutdown methods
- Area map

SE-8 and the F-series pre-fire plans when used in combination provide the main control and fire brigade personnel with a comprehensive plan and methodology for mitigating the consequences of a fire in the plant.

4.5.3 Fire Brigade Training and Equipment

The plant fire brigade receives training as outlined in 10CFR50 Appendix R Section I including classroom instruction, fire fighting practice and fire drills and meetings.

Initial Classroom Instruction/Refresher

Initial classroom instruction is provided for all fire brigade members. Subjects covered follow the guidance of Appendix R Section I.1 and include:

- Fire hazards
- Fire fighting equipment location and use
- Fire fighting procedures and methods
- Fire fighting direction and coordination - brigade leaders only
- Structured fire fighting

Each fire brigade member receives a refresher classroom instruction annually which includes the above instruction in addition to updates on plant procedures and modifications which affect the operation of the fire brigade.

Fire Fighting Practice

Each fire brigade member receives live fire fighting instruction at the PECO Energy Company fire school. This hands-on training is designed to train the brigade on the proper methods of combating the various types of fires that could occur in the plant. These sessions are administered under actual fire conditions to acclimate the brigade to actual fire fighting environments.

Fire Drills

Fire Drills are conducted in accordance with 10CFR50 Appendix R, section 1.3 and follow plant surveillance test ST-7-022-551-0, Fire Drill. Fire brigade drills are conducted at the plant by a qualified individual who is knowledgeable, experienced, and suitably trained in fighting the types of fires that could occur in the plant and in using the types of equipment available in the nuclear power plant.

Drills are designed to include:

- Fire alarm and brigade notification effectiveness.
- Brigade response time.
- Brigade knowledge of duties and responsibilities.
- Main control room response.
- Adherence to procedures.
- Simulation of firefighting equipment use.
- Assessment of fire brigade leader effectiveness.

Each shift fire brigade receives a minimum of one drill per calendar year. One of these drills is a back shift unannounced fire drill. Each fire brigade member must attend two drills per calendar year.

The drills are planned and conducted for anticipated actual plant fire scenarios and are designed to test both the response of the main control room and the fire brigade. A drill critique is conducted with the brigade following the drill to analyze the brigade's conduct and make recommendations as necessary.

Fire Brigade Meetings

Fire brigade meetings are conducted for the fire brigade upon completion of the fire drill critique. Each fire brigade member attends a quarterly meeting or reads, signs and dates the quarterly meeting minutes. The meetings are used to update the brigade with plant modifications, changes to the fire protection program, fire fighting equipment and other subjects as necessary.

Fire Brigade Equipment

The plant fire brigade is provided with the equipment necessary to combat and extinguish the anticipated types of fires at the site. Personal Protective Equipment (PPE) includes helmets, coats, boots, gloves, and SCBA. Fire fighting equipment is provided at the main fire brigade locker room in the Unit 2 Turbine Aux Building 269 elevation, in addition to 3 brigade sublockers located at the following locations in the plant.

(1) Unit 1 Turbine Aux Building 217' elevation
(2) Unit 2 Turbine Aux Building 217' elevation
(3) Circulating Water Pump House

These additional lockers allow for quicker response for fire brigade members who may be in various plant locations due to shift operational responsibilities. Brigade PPE and fire fighting equipment are inventoried and are verified by periodic inspections to assure their availability to the brigade. The above described equipment, particularly with the sublockers, are adequate for response to any plant fire by the fire brigade.

4.5.4 Access Routes and Existing Barriers

The plant fire brigade is knowledgeable of the physical arrangement of the site. Individual pre-fire strategies provide detailed primary and secondary access routes to each plant area, in addition a map of the area is attached to the plan. Access routes that involve locked doors are specifically identified in the strategies with appropriate precautions and methods for access identified.

4.6 Analysis of Plant Systems, Sequences, and Response

An assessment of the availability of plant equipment, cabling and components necessary to achieve and maintain safe shutdown of the reactor was performed. The quantification of unavailability was performed using NUPRA2.2 and the 1993 LGS PSA model. The 1993 LGS PSA model updated the IPE model with plant equipment and procedure changes after the freeze date of the IPE. The potential impact of a fire was considered for all plant areas and focused primarily on the Appendix R safe shutdown equipment remaining free from fire damage. A few additional systems were also credited to supplement the Appendix R equipment.

4.6.0 Model Differences from the LGS PSA

Changes to the base LGS PSA model were made to match the assumptions and boundary conditions from the Appendix R study. The changes from the standard PSA models and assumptions can be summarized as follows:

(1) The duration of the analysis is assumed to be 72 hours versus the PSA endpoint of 20 hours;

(2) Alternate shutdown cooling is credited as a mode of RHR that provides both injection and heat removal;

(3) Cold shutdown was modeled as the endstate in the analysis (except where venting was credited for screening an area) compared to the hot shutdown endstate used in the PSA analysis;

(4) Human actions not credited in the Appendix R safe shutdown analysis were removed from the fault tree models;

(5) Pre-cursor events such as miscalibration of sensors were removed from the PSA model;

(6) Offsite power is never recovered in the 72 hour period for both fire-induced and random causes;

(7) Diversion paths are not modeled;

(8) Crossover between injection flowpaths is not credited;

(9) All systems were assumed available before the initiation of the fire

Items 5, 7, and 9 were used to create a model as closely matched as possible to the Appendix R analysis basis. Diversion paths in particular were not assumed possible due to Appendix R spurious actuation analyses.

Event trees were developed to specifically match the system\train definitions from the safe shutdown analysis (i.e. top events represented systems or trains rather than functions as modeled in the IPE). In addition, offsite power was specifically modeled as an event tree node.

4.6.0.1 Unavailability of Equipment

P_2 is defined as the aggregate unavailability of redundant/alternate safe shutdown systems not impacted by a fire in a particular compartment. In the initial phase of screening, P_2 typically represents the random unavailability of systems/components "outside" a fire area because it is assumed that all equipment and cabling in an area is destroyed by fire. Credit for systems not affected by the fire is given when all components and equipment including power, control and instrumentation cables and any local operator actions are separated from the fire location in conformance with accepted Appendix R assumptions (including operator actions) and fire protection separation criteria. In addition, fire barrier materials used to provide cable separation were not credited in the P_2 analysis.

4.6.0.2 Calculation of Unavailability

The calculation of the overall unavailability (P_2) presented in the FIVE Methodology is simply the product of the unavailability from each of the totally independent shutdown paths or systems. The shutdown paths as defined in the safe shutdown analysis are linked through common support systems such as AC and DC power, Emergency Service Water (ESW), and RHRSW. The use of PSA models and software facilitated the calculation of P_2 by identifying the minimal common failures that contribute to the overall unavailability.

Information regarding the failures of equipment and components was obtained from the safe shutdown analysis and performed for each fire compartment. Fault trees, basic event probabilities, and equations representing systems/trains were modified based on the information obtained from safe shutdown analysis. An event tree was developed to specifically model each unique set of systems categorized as successful and failed. The event tree was used to determine the possible failure combinations that could occur for those systems known to survive the

effects of a particular fire. A description of the elements of the event trees can be found in Section 4.6.2. All possible failure combinations from random (including the random probability of a LOOP occurring within the 72 hour time period) and common causes were summed to determine the overall (aggregate) unavailability (P_2) for each fire compartment. This unavailability combined with the fire frequency (F_1) for each fire compartment was used to determine if the fire area met the 1.0E-6/yr. screening criteria.

Similar fire compartments (in terms of system/equipment failures) were compared, and, where applicable, the same P_2 was applied. Based upon insights gained from the results of many of the calculations it was obvious when an area would not screen given the number and extent of the failures combined with the fire initiator frequency. Given this situation, a P_2 was not calculated and the fire compartment was passed to the next level of analysis.

4.6.0.3 <u>Calculation of Unavailability (P_4) Resulting from Target Analysis</u>

P_4 is defined as the aggregate unavailability of redundant/alternate safe shutdown systems not impacted by a fire both outside the compartment (P_2) and those that survive the effects of a fire within ("inside") the fire compartment.

An iterative process involving the identification of the cables and components (targets) with respect to the known sources of fixed and transient combustibles is used to facilitate the credit that can be given to the survivability of equipment within an area. When it is determined that the desired system/train will survive in a given fire compartment the PSA models were used to calculate the unavailability of the surviving systems from non fire-induced causes and again compared to the 1.0E-6/yr. screening criteria. The calculated P_4 for those compartments that required specific target fire analyses is shown on Table 4.6-1.

4.6.1 Dominant Sequences

A resultant Core Damage Frequency (CDF) for each fire compartment was not considered as an endstate for this FIVE screening analysis. Rather, an unavailability of selected systems was calculated using PSA modeling techniques and plant specific models.

The safe shutdown analysis, used as a starting point of the analysis, identified those systems and actions required to achieve and maintain sub-critical reactivity conditions in the reactor, reactor coolant inventory, reactor pressure control, and maintain safe and stable shutdown conditions following a fire initiated event. Numerous redundant/alternate methods were typically available to maintain each of the above stated functions.

Although CDF was not calculated, global insights can be gleaned from the impact that some failures had in some of the fire areas. Reactor coolant inventory control, pressure control, and reactor/containment heat removal were typically achieved because of the diverse systems or trains of systems that were available to fulfill the function. Fires that affected the individual loops of ESW appeared to have the largest effect when determining the screening potential of individual fire areas. ESW provides cooling water to the diesel generators, RHR pumps, and room coolers required during a loss of offsite power. The fire-induced loss of ESW had a higher impact than that of other systems because it is a support system for a number of frontline systems. Systems used to shutdown the plant when offsite power is available would not typically require cooling from ESW because the service water system would potentially be available. The ESW system is described in the LGS IPE, Section 3.2.1.5.

Similar to the results in the IPE, the unavailability of high pressure injection systems, HPCI and RCIC, from fire-induced or random failures combined with the failure to depressurize the reactor provided a lower limit on the P_2 calculations and affected the potential to screen some areas initially.

TABLE 4.6-1

UNAVAILABILITY RESULTING FROM TARGET ANALYSIS

FIRE COMPARTMENT	F_3	P_4 EAST	P_4 WEST
44	1.1E-5	1.6E-4	1.3E-4
45	4.7E-6	1.6E-4	1.3E-4
47	5.4E-5	1.6E-4	1.3E-4
64	1.5E-8	1.3E-4	
67	2.7E-5	1.6E-4	1.6E-4
68	3.0E-6	1.5E-4	1.3E-4
70	5.9E-5	1.6E-4	1.3E-4

4.6.2 Fire Event Trees

The probabilistic analysis began with the identification of the fire initiating event for each particular area (assumed to be 1.0 for consistency with the FIVE methodology and preventing double counting of the initiator frequency) leading to the requirement for plant shutdown. A generic event tree was developed to represent the potential shutdown systems available and was used as a template for individual fire areas. The event trees were then modified to specifically model each unique set of systems categorized as successful and failed for each particular fire compartment.

4.6.3 Recovery Actions

Human actions credited in the safe shutdown analysis were specifically identified for each fire area (if applicable) when used to assure success of a particular system. These actions were compared to those modeled in the PSA to assess if they were accounted for in the probabilistic models. In general, the actions credited were either outside the fire area of interest or were recovery actions well after the fire was extinguished.

With the following exceptions, operator actions credited in the IPEEE Fire Hazard Analysis are included in the LGS FPER:

(1) Venting of the Suppression Pool

In order to provide an alternate means of suppression pool cooling the six (6) inch vent path was credited. Venting will be accomplished as described in TRIP procedure T-200 with the exception that one valve is assumed to be operated remotely to initiate the venting process while all others are locally repositioned.

(2) RCIC Trip at the RCIC Turbine

To assure that the RCIC system can be secured, operators may be required to trip the RCIC turbine locally. This action is proceduralized in OT-110 and taken when reactor level exceeds Level 8.

(3) Power Feed to Pump 0CP548 (C ESW Pump)

The power feeds to pump 0CP548, normally aligned to Unit 2, provide the capability of powering this ESW pump from Unit 1 or Unit 2. Manual actions associated with the power feed transfer is proceduralized.

(4) Manual Positioning of Valve HV-51-2F014A

In order to provide RHR Service Water to the RHR heat exchanger, valve HV-51-2F014A will be manually opened to support suppression pool cooling.

4.6.4 Core Damage Frequency

The non-fire induced unavailability calculations used are not intended to calculate a Core Damage Frequency (CDF) but represent a screening method that assesses the redundancy and diversity of equipment and actions necessary for safe shutdown given a fire in a particular compartment. A CDF calculation implies all possible systems and actions are credited for shutdown. The unavailability calculations are conservative and can potentially serve as surrogates for CDF. Credit was not taken for all potential systems and operator actions.

4.6.5 Results of Probabilistic Analysis

Five compartments could not be screened due to the calculated size of the SOD, and the inability of fire modeling to show reasonable probability of crediting redundant systems back into the analysis The five compartments are:

Fire Compartment 2 - 13.2 KV Switchgear Room

Fire Compartment 20 - Unit 1 Static Inverter Room

Fire Compartment 24 - Control Room

Fire Compartment 25 - Aux Equipment Room (PGCC)

Fire Compartment 26 - Remote Shutdown Panel Room

The probabilities associated with these compartments using the FIVE process were greater than the other compartments analyzed; therefore, additional probabilistic analyses were performed to assess the significance of these compartments and the measures needed to address the issues in each compartment. A previous probabilistic analysis performed for Limerick (Ref. 1.1-5) and reviewed by the NRC (Ref. 1.1-6) was reviewed to assess if the analysis and results could be applied to these compartments. These compartments were also identified in that study as well.

4.6.5.1 Control Room

The control room (fire compartment 24) contains instrumentation and controls for almost all plant equipment. A fire in the control room could cause inadvertent tripping or actuation of equipment and may disable instrumentation needed to control the safe shutdown of both Limerick units.

Cabinet fires are postulated to occur within control room cabinets but are typically confined to where the fire originated. A small number of cabinets within the control room could impact the shutdown of the plant given a fire. The cabinet containing ECCS equipment controls is considered to have the greatest impact on the mitigative equipment needed for shutdown. The control room, is continuously manned which will facilitate the early detection and suppression of any fires that might occur.

The probability of a fire occurring in the control room affecting a particular cabinet combined with the probability of failure to detect and suppress a fire is small. In addition, a dedicated remote shutdown panel can provide shutdown capability outside the control room should evacuation be necessary. The combined probability of events needed to prevent the safe shutdown of the plant is considered to result in the probability that allows screening the control room from further evaluations. Previous analyses performed for Limerick support this conclusion.

4.6.5.2 Auxiliary Equipment Room

The auxiliary equipment room (fire compartment 25) contains relays and other signal-conditioning components and cabling required for the control of almost all plant equipment. A fire in this compartment could have a similar impact on plant equipment as the control room.

Fire detection and suppression systems in the auxiliary equipment room exist to provide indication and some mitigation should a fire occur. The probability of fire initiation combined with the failure of detection or suppression is considered to be small. The remote shutdown panel can

also be used should ECCS equipment be affected. In addition, the control room could potentially use BOP equipment to shutdown the plant. Similar to the conclusion reached in the control room, the combined probability is considered low enough to screen the auxiliary equipment room from further analysis. Again, previous analyses support this conclusion.

4.6.5.3 Remaining Compartments

The postulated failures of equipment in compartments 2, 20 and 26 are sufficient, given the fire initiation frequency, to prevent screening these areas. Given a fire and the calculated unavilability of the surviving equipment the screening values (F_2) obtained are of a magnitude that would not suggest the need for any modifications. Comparison to the NEI Guidelines on Severe Accident Issue Closure (ref. 4.6-1) would suggest administrative changes be made to emphasize prevention or control of fires in these areas. A summary of the screening process and disposition of those areas not initially screened is provided in section 4.10.

4.7 Analysis of Containment Performance

The FIVE Methodology outlines a process for quantifying the likelihood of a fire in a given compartment resulting in the inability to achieve or maintain safe and stable shutdown conditions. Implicit in the screening is the impact of the fire on the ability to maintain containment functions for those compartments that remained above the 1E-6/yr. criteria after completing the Phase 2 analysis.

The evaluation of containment performance must be evaluated against the following two criteria:

(1) Identification of a minimum set of equipment and manual actions necessary to achieve the containment function considering those lost due to the fire.

(2) An assessment of the potential for a fire in the compartment of concern to damage equipment or prohibit manual operator actions used to accomplish the containment functions of isolation and heat removal.

4.7.1 Containment Isolation Function

An assessment of the potential for fire-induced containment isolation failures or containment bypass events were performed using the following criteria:

(1) The failures of equipment associated with a particular fire compartment were reviewed to determine if any spurious opening of valves needed to maintain isolation would occur and if any manual actions would be feasible to maintain containment integrity.

(2) Fire-induced containment bypass potential is assessed by evaluating the equipment and cable failures in each of the fire compartments that did not screen.

4.7.2 Containment Heat Removal

Decay heat removal, in the context of the fire analysis, is accomplished using the RHR system with RHRSW to remove decay heat or through the use of containment venting. The RHR system is credited in removing decay heat using a number of different operating modes. Suppression pool

cooling (SPC), shutdown cooling (SDC), and alternate shutdown cooling (ASDC) are different modes considered for removing decay heat to the ultimate heat sink.

Each of the fire compartments that did not screen were evaluated to assess the degree of redundancy of heat removal that remained. The non fire-induced random failures were then evaluated to determine any insights with regard to the heat removal function.

4.7.3 Containment Performance for Non-screened Compartments

Fire Compartment 2 - 13.2 KV Switchgear Room

Fire Compartment 20 - Unit 1 Static Inverter Room

Fire Compartment 26 - Remote Shutdown Panel Room

The fire-induced failures associated with the above three areas were reviewed to determine the impact on containment performance in light of the criteria. The equipment failures involved with these specific compartments reveal that all high-low pressure piping and the valves associated with each pipe segment were not compromised by the fire, either directly or through spurious operation of valves. The high-low pressure interface analysis was required for Appendix R and was, therefore, explicitly considered when assessing the fire impact.

Containment heat removal systems were impacted differently in each of the compartments. Fire compartment 2 had the greatest impact on heat removal systems because of the electrical power feeds to a majority of the systems.

4.8 Treatment of Fire Risk Scoping Study Issues

NRC Generic Letter 88-20 Supplement 4 lists the following Fire Risk Scoping Study issues to be addressed in the IPEEE submittal on fire analysis.

(1) Seismic/Fire Interactions.

(2) Fire Barrier Qualifications

(3) Manual Fire Fighting Effectiveness

(4) Total Environment Equipment Survival

(5) Control Systems Interaction

4.8.1 Basis and Assumptions

The specific concerns regarding each of these issues are addressed in the FIVE Methodology section 7.0. The methodology outlined in this section in addition to the attributes to be addressed as outlined in Section 10.5 of the methodology were used as a basis for the evaluations of the scoping study issues. Details of the specific evaluations are addressed in the following sections.

4.8.2 Findings

A detailed discussion of the analysis and findings and conclusions for each Fire Risk Scoping Study issue is discussed in the following paragraphs.

4.8.2.1 Seismic/Fire Interactions

The seismic/fire interactions issue consists of three elements:

(1) seismically induced fires;

(2) seismic actuation of fire suppression systems;

(3) seismic degradation of fire suppression systems;

Seismically Induced Fires

As outlined in section 10.5 of the FIVE methodology, walkdowns were performed to address potential seismic/fire interaction. The results of these walkdowns show that one condition could potentially result in a seismically induced fire. The condition involves the sight glasses on the diesel generator lube oil make up tanks in the diesel day tank rooms.

These sight glasses do not have isolation valves; therefore, should they fail during a seismic event, the oil in the lube oil make up tanks would drain out onto the day tank room floor. This condition is not considered to be a significant issue for the following reasons:

(1) The lube oil make up tanks are not needed to maintain operability of the diesels.

(2) Should the sight glass fail and release oil onto the floor of the day tank room, the room design will contain the oil within the room.

(3) Also, ignition of the oil is not postulated due to the lack of ignition sources (the postulated fire for this fire area is a fuel oil or lube oil leak from the diesel engine onto the floor of the diesel engine compartment, with subsequent ignition).

(4) Should a fire occur in the day tank room, it would be contained by the 3 hour rated fire barriers that make up the day tank room walls and it would be controlled by the pre-action sprinkler system which protects the diesel generator compartment (including the day tank room).

No other situations were discovered by the seismic/fire interaction walkdown team where flammable gas or liquid storage vessels could create a significant fire hazard due to a seismic event.

Seismic Actuation of Fire Suppression Systems

The seismic/fire interaction walkdown also investigated the impact of inadvertent actuation of fire suppression systems on plant equipment. As discussed in Section 9A.3.1.2 of the LGS UFSAR, the suppression systems at LGS have been designed and located so that inadvertent operation of or a crack in the systems will not cause damage to redundant trains of safety related equipment that is needed for safe shutdown of the plant.

Seismic Degradation of Fire Suppression Systems

Fire suppression systems at LGS are designed and installed in accordance with the applicable NFPA code. This code provides for an adequate level of support based on the geological characteristics of the region. In addition, systems located in safety related areas are designed and installed with the II/I design criteria for seismic conditions as required by PECO Energy specification M-400, (ref. 3.1-16, 3.1-17) Safety Impact Review and Commodity Clearance/Structural Walkdown Program. Based on these conditions, hazards due to seismic degradation are not anticipated. Mercury switches are a special concern since they can cause equipment to spuriously operate during a seismic event. The fire protection systems were reviewed to determine if mercury switches are used. The following mercury switches were found on the fire system:

Electric motor driven fire pump (00-P512) discharge pressure switch. The pressure switch is designed to start the electric motor driven fire pump should the fire system header pressure drop to 100 psig. A seismic event could cause the 00-P512 to start spuriously which would result in the pump running at minimum flow. However, since there is a relief valve on the pump discharge piping, no damage to the pump would result. A seismic event could also prevent 00-P512 from starting. However, should this occur the diesel driven fire pump (00-P511) would still be capable of starting at 95 psig. The fire system is designed such that only one pump is required to maintain system operability. The seismic event that is postulated for LGS is expected to last less than a minute; therefore, the impact of 00-P512 possibly being unavailable is considered to be insignificant, since it is expected that the pump would resume normal operation after the seismic event ends and 00-P511 would be available throughout the entire event.

This pressure switch is only capable of starting the fire pump on low fire system pressure, spurious actuation of this pressure switch would not result in shutting down the fire pump should the pump be operating at the time of the seismic event. Spurious actuation of this pressure switch would not result in actuation of any water suppression systems.

CO_2 system refrigeration system control pressure switch. A seismic event could cause the compressor to not operate on demand, or to operate spuriously. The seismic event that is postulated for LGS is expected to last less than a minute; therefore, the impact of the CO_2 system compressor operating or not operating is considered to be insignificant, since it is expected that the system would resume normal operation after the seismic event ends. Spurious actuation of this pressure switch would not result in actuation of any CO_2 suppression systems. Based on these conditions and

the seismic walkdowns performed for fire suppression system, hazards due to seismic degradation are not anticipated.

4.8.2.2 Fire Barrier Qualifications

Rated fire barriers and components (doors, dampers, and penetration seals) are used at LGS to provide separation between redundant safe shutdown equipment and to separate safety related equipment and areas from significant fire hazards. The scoping study issues of fire barrier qualifications is concerned with adequate design, inspection, testing, and maintenance of the barriers and associated components.

Fire rated barriers are inspected and maintained through periodic testing and inspection programs controlled by plant procedures. Fire doors are also inspected and maintained in accordance with approved plant procedures. Fire dampers are inspected routinely and are maintained per plant procedures. The installation concerns addressed in the NRC Information Notice 89-52 for fire dampers have been reviewed for applicability to LGS installations and have been adequately addressed. Penetration seals are installed per manufacturers installation and design criteria and are inspected periodically. Maintenance and control of penetration seals is administered through plant procedures. The penetration seal issues discussed in Information Notice 88-04 have been reviewed for applicability to Limerick installations and have been adequately addressed.

Consistent with the recommendations of NRC Generic Letter 92-08, rated fire barriers made from the Thermal Science, Inc. (TSI) fire barrier material Thermo-Lag 330-1 have been declared inoperable. Appropriate fire protection compensatory measures have been implemented. PECO Energy's planned response to the Generic Letter identified that on a per barrier basis, this condition will remain until either the revised safe shutdown analysis (Thermo-Lag reduction) is implemented, thereby removing the requirement to have the barrier or the barrier is analyzed to meet full regulatory requirements.

4.8.2.3 Manual Fire Fighting Effectiveness

Section 10.5 of the FIVE methodology discusses a number of attributes of an acceptable fire brigade and preparedness program. A review of the plant fire brigade program at LGS was conducted comparing the attributes to the plant program. The fire brigade training, manning, and equipment meets or exceeds those discussed in the methodology, therefore, credit for

manual fire suppression is considered within the analysis. Details on fire brigade procedures, training and equipment is provided in section 4.5 of this report.

4.8.2.4 Total Environmental Equipment Survival

The Scoping study addressed three major concerns regarding equipment survival.

(1) The potential for adverse effects on plant equipment caused by combustion products.

(2) The spurious or inadvertent actuation of fire suppression systems.

(3) Operator effectiveness in performing manual safe shutdown actions.

Spurious or inadvertent actuation of fire suppression systems has been addressed in the seismic/fire interaction discussion above.

As designed the FIVE methodology does not currently allow for an evaluation of non-thermal environmental effects of smoke on equipment. However, the detrimental short term effects of smoke and hot gases on equipment are not believed to be significant for the time lines used within the analysis.

Operator effectiveness in performing manual safe shutdown actions is not considered to be affected by areas which contain smoke and hot gases. Plant shutdown in the event of a fire is controlled through plant procedures. Manual actions to support hot shutdown are not performed in the area under the influence of the fire environment. Operators receive training of fire shutdown procedures and are equipped with the knowledge and equipment necessary to respond to the fire situation and place the plant in a safe condition. Human behavior is accounted for within the analysis in the calculation of redundant safe shutdown equipment being available (P_2).

4.8.2.5 Control Systems Interaction

The intent of this issue is to verify that the ability to achieve safe shutdown from either the control room or remote shutdown room cannot be compromised by a single fire. The LGS remote shutdown panel meets the circuit isolation requirements for alternative shutdown. Additionally, for the alternative shutdown scenario the diesel generators required to support alternative shutdown are locally isolated and controlled. Both the remote

(alternative) shutdown and applicable diesel generator control power supplies meet the requirements of IN 85-09, Isolation Transfer switches and Post Fire Shutdown Capability.

Additionally, in conjunction with the fire risk analysis a fire safe shutdown re-evaluation was performed to revise the safe shutdown analysis to support a reduction of the reliance on the fire barrier material Thermo-Lag. This study verified alternative shutdown isolation and control transfer circuitry independence for the condition found after circuit isolation and control transfer.

Circuitry independence prior to control transfer is assured for those devices (e.g. the RCIC system control) that can be damaged by overcurrent. However, for valves with alternative shutdown isolation and control transfer capabilities, transfer capability scenarios that need to consider smart (defined) hot shorts or unique sequences of multiple fire-induced cable failures are not postulated to occur prior to control transfer. The probability associated with a specific set of circumstances leading to the loss of safe shutdown capability is assessed to be extremely low.

A qualitative analysis using representative probabilities for each of the events is used to support the conclusion that the assumptions of specific or a series of fire-induced cable failures affecting safe shutdown from outside the control room, in combination, are highly unlikely and are therefore, not safety significant. The probability of fire initiation in the room, the identification of a fire by plant personnel, the limited time frame in which the fire damage must occur (the time it takes to transfer control from the control room), the postulated cable damage resulting in specific circuit opens, grounds, and hot shorts that would be required to be maintained, and limitations on the specific number of conductors in a cable all contribute to the conclusion that the probability of this event occurring and propagating to a stage severely affecting the ability to safely shutdown the plant is extremely small.

4.8.3 Findings and Potential Improvements

As shown above, Limerick adequately addresses the issues related to the Fire Risk scoping Issues. No potential improvements related to the Fire Risk scoping study were identified.

4.9 USI A-45 and other Safety Issues

4.9.1 Basis and Assumptions

The purpose of USI A-45 regarding Shutdown Decay Heat Removal Requirements is to evaluate the adequacy of current designs to ensure that LWRs do not pose unacceptable risk as a result of decay heat removal failures. Those systems and components required to maintain primary and secondary coolant inventory control and to transfer heat from the reactor coolant system to an ultimate heat sink following shutdown are considered under USI A-45. The USI A-45 program also requires consideration of supporting systems, such as, component cooling water systems, emergency service water systems, and emergency on-site AC and DC power systems when evaluating the impact on decay heat removal.

Decay heat removal, in the context of the fire analysis, is accomplished using the RHR system with RHRSW to remove decay heat or through the use of containment venting. The RHR system is credited in removing decay heat using a number of different operating modes. Suppression pool cooling (SPC), shutdown cooling (SDC), and alternate shutdown cooling (ASDC) (See Section 3.1.2.5.3.4 for description) are different modes considered for removing decay heat to the ultimate heat sink. Decay heat removal was specifically addressed in the fire analysis as one requirement for an area to screen.

4.9.2 Findings and Conclusions

Based on the above, loss of decay heat removal capability is not considered a vulnerability and no additional insights from the fire assessment have been found that require incorporation in plant procedures or equipment.

4.9.3 Potential Improvements

No potential improvements were identified from the fire analyses that were unique from previous assessments of heat removal capability.

4.10 Summary of Fire Analysis

The process used in assessing the impact of postulated fires at Limerick has provided insights and an understanding of the areas of the plant that have the highest potential for affecting safe shutdown capability.

An illustration representing the screening process and the plant specific compartment analyses performed during that process is shown in Figure 4.10-1. The figure outlines the successive screening applied to specific compartments and the outcome resulting at each screening step.

21 of 127 compartments did not initially meet the 1.0E-6/yr. screening criteria. Of the initial 21, 4 compartments in the Reactor Enclosure subsequently screened because analysis demonstrated separation could be maintained between subcompartments within each of the 4 large reactor enclosure compartments by providing additional combustible control guidelines in areas within the Reactor Enclosure. A P_2 recalculated for each of the subcomponents allowed screening.

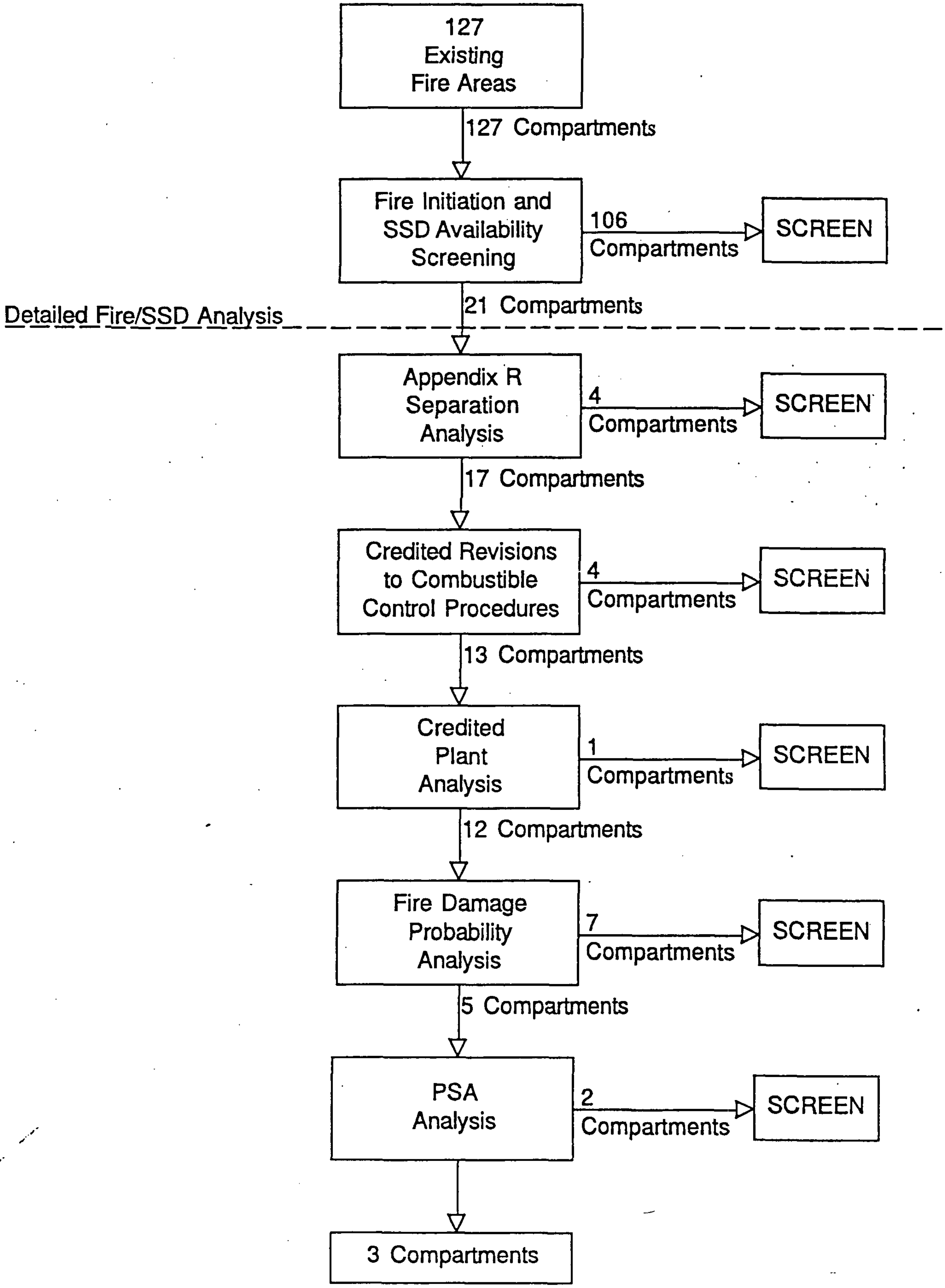
Four of the remaining 17 compartments were identified as not containing ignition sources or fixed combustibles except for electrical cable. Electrical cable is not considered a fire ignition source, and therefore will not ignite without an external fire source available. These compartments were evaluated for transient combustible exposure damage. Because of the physical arrangement of the equipment in the room it was not possible to preclude damage by location of combustibles or fire suppression. Further evaluation of the compartments led to the development of administrative controls in the form of designating these compartments as transient combustible free zones.

To support compartmentalization of the Turbine Enclosure a structural steel survivability calculation was performed to assess the potential of a fire to impact the structural capability (i.e. compartment barrier) of the room. The calculation identified that a fire in the area was ventilation controlled. Therefore, by administratively controlling the opening of the doors, which are fire rated, the Turbine Enclosure was able to be compartmentalized and subsequently screened.

Seven compartments required detailed analysis of the probability of damage to equipment within each compartment. Spatial relationships between the fixed combustibles and equipment needed for shutdown (targets) were investigated. Transient exposure damage was also assessed. This analysis combined within the calculated availability was used to screen the seven compartments.

FIGURE 4.10-1

FIRE COMPARTMENT SCREENING METHODOLOGY FLOWCHART



Two of the remaining fire compartments were screened based on probabilities supported by previous analyses.

The remaining 3 compartments, 13.2kV switchgear room (FC2), Unit 1 Static Inverter Room (FC 20), and the Remote Shutdown Panel Room (FC 26) did not screen but were of a magnitude that would warrant an increased awareness of the potential impact each could have given a fire in each compartment. Limerick will increase the fire brigade drill activities and awareness in these areas to emphasize prevention and control.

5 HIGH WINDS, FLOODS, AND OTHERS

This section addresses the Limerick Units 1 and 2 Individual Plant Examination of External Events (IPEEE) for high winds and tornadoes, external floods, transportation and nearby facility accidents and other plant unique external events in response to Generic Letter 88-20 Supplement 4. The methodology used to assess these external events' impact to plant safety was done in accordance with the guidance described in Section 5 of NUREG-1407 (ref. 1.3-2).

5.0 External Event Screening

5.0.1 Screening Methodology

High winds and tornadoes, external flooding, transportation and nearby facility accidents are identified as the significant other external events (besides seismic activity and internal fires) for inclusion in IPEEE. Evaluation of these external events incorporates the progressive screening approach shown in Figure 5.0-1 and represents a series of analyses in increasing level of detail, effort and resolution.

The methodology used to ensure that all significant external events relevant to the LGS site are evaluated and addressed, was implemented through the following review process:

1. As a first step, a complete listing of LGS external events was compiled based on the LGS UFSAR (ref. 5.0-1) and LGS Severe Accident Risk Assessment (SARA) (ref. 1.1-5). This list of events provided the basis for LGS external events evaluation. However, to confirm that no other plant unique external event with potential severe accident vulnerability was excluded from the IPEEE, other external events were compiled based on the recommendation of NUREG/CR-2300 (ref. 1.1-2). Table 10-1 of NUREG/CR-2300 contains a comprehensive listing of natural and man-made external events that are recommended for consideration on Probabilistic Risk Assessment (PRA) studies.

2. Next, each of the external events identified on this list was evaluated on the basis of an interim screening approach. The intent was to eliminate from further study those events with negligible contribution to the overall plant risk. The screening criteria were formulated to screen other significant external events from further study in accordance with the recommendations of NUREG/CR-2300 Section 10.3.1.

The screening criteria are summarized as follows:

Criterion 1: Low Frequency
The event has a significantly lower mean frequency of occurrence than other events with similar uncertainties and will not result in worse consequences than these events. For example, meteorite impact as an external event can be eliminated on the basis of low frequency of occurrence.

Criterion 2: Design Basis
The event is of equal or lower damage potential than the events for which the plant has been designed. For example, since LGS has been designed for a Design Basis Tornado of 300 miles per hour (MPH) winds and tornado missiles, consideration of hail as a missile source is not necessary.

Criterion 3: Relevance
The event cannot occur close enough to the plant to affect it. Tsunamis are rare on the East Coast and LGS is far inland, and thus, tsunami as an external event for consideration is eliminated.

Criterion 4: Inclusion
The event is included in the definition of another event. For example, release of toxic gases is included in the effects of pipeline accidents, industrial or military facility accidents and transportation accidents.

Criterion 5: Speed
The event is slow in developing (such as drought) and there is sufficient time to eliminate the source of the threat or to provide an adequate response.

Application of these screening criteria resulted in the selection of a limited number of significant events for IPEEE analysis.

5.0.2 Identification of Events and Event Screening

The complete list of external events that was compiled for the LGS external events evaluation is shown in Table 5.0-1 and has been organized to show the corresponding UFSAR and SRP sections.

The primary focus was on the review of the plant specific hazard data and plant design information for conformance to the 1975 Standard Review Plan and other regulatory requirements. Significant site changes since issuance of the operating license, if any, were also reviewed and compared for acceptance against the 1975 SRP (ref. 5.0-2) criteria.

Based on the applied screening, the following external events must be addressed as part of the LGS IPEEE program:

High Winds
This accident category will also include: Tornadoes, Hurricanes (insofar as they may induce high winds), Missiles Generated by Natural Phenomena (insofar as they may be induced by high winds and tornadoes).

Floods
This accident category will also include: Dam Failure, High River Stage, Hurricanes (insofar as they may induce intense rain), Intense Precipitation, Storm Surge, Waves, Probable Maximum Floods on Streams and Rivers.

Transportation and Nearby Facility Accidents
This accident category will also include: Aircraft Impact, Pipeline Accidents, Release of Chemicals from Storage On-site, Toxic Gas (Exposure to Hazardous Chemical Release), Missiles Generated by Events Near the Site, Explosions, and Flammable Vapor Clouds.

5.0.3 Plant Walkdowns

NUREG-1407 requires a confirmatory walkdown of the plant to provide assurance that potential vulnerabilities not identified in the plant design basis do not exist.

Walkdowns were conducted at the LGS site the week of January 9 - 11, 1995. The purpose of the plant walkdowns was to assess the vulnerability of plant structures and equipment to high winds and tornadoes, external flooding, nearby industrial, transportation and military facility accidents and to confirm the location of nearby facilities and transportation routes.

To identify potential vulnerabilities to high winds and tornadoes the walkdowns focused on the exterior of the plant power block structures, safety related components outside the power block structures and other facilities, equipment and material situated around the plant site.

The elements of the LGS wind resistant design which were assessed are:

- General ruggedness of the Category I structures to resist tornado winds
- Materials (type and quantity) around the plant site which could become tornado missiles
- Barriers and other protection to prevent the impact of tornado generated missiles on Category I structures

- Potential effects of wind/tornado damaged non-Category I structures and components on Category I structures

To identify potential vulnerabilities of plant structures and equipment to external flooding the walkdowns were focused inside the protected area and inside plant structures. The walkdown also included a general survey of the plant site and surrounding area via car outside the protected area.

The general objectives of the plant walkdown were to:

- Verify flood protection of structures and equipment as stated in the LGS UFSAR
- Perform a general assessment of the site area and surrounding areas topography to identify significant areas of runoff and restrictions or diversions to that runoff (i.e. potential for local site ponding)
- Assess overall site drainage capabilities

To identify potential vulnerabilities to transportation and nearby facility accidents and to confirm the location of facilities and transportation routes, the walkdown included a general survey of the surrounding area via car outside the protected area. The survey covered a five (5) mile radius around the LGS site and its main objective was to identify and confirm the significant nearby industrial, transportation and military facilities described in section 2.2 of the UFSAR within the five mile radius of the plant site.

In addition, any new significant facilities that were built or changes made to the plant after the LGS Operating License was issued were considered in the IPEEE evaluation. The following changes were identified:

- Chlorine Tanks have been removed from site;
- Warehouse near Spray Pond has been removed;
- Route 422 has been completed;
- Construction of residential developments within a 5 mile radius of LGS;
- The Hooker Chemical Plant changed to Occidental Chemical Corporation;
- Two small airports have been added while two others are no longer in operation and three new airways have been identified;
- The following new site structures have been erected: the Warehouse and Procurement Building, the Site Support Building, the Site Management Building, the J. S. Kemper Building, and additions to the Administration, Water Treatment, Radwaste and Technical Support Center Buildings.

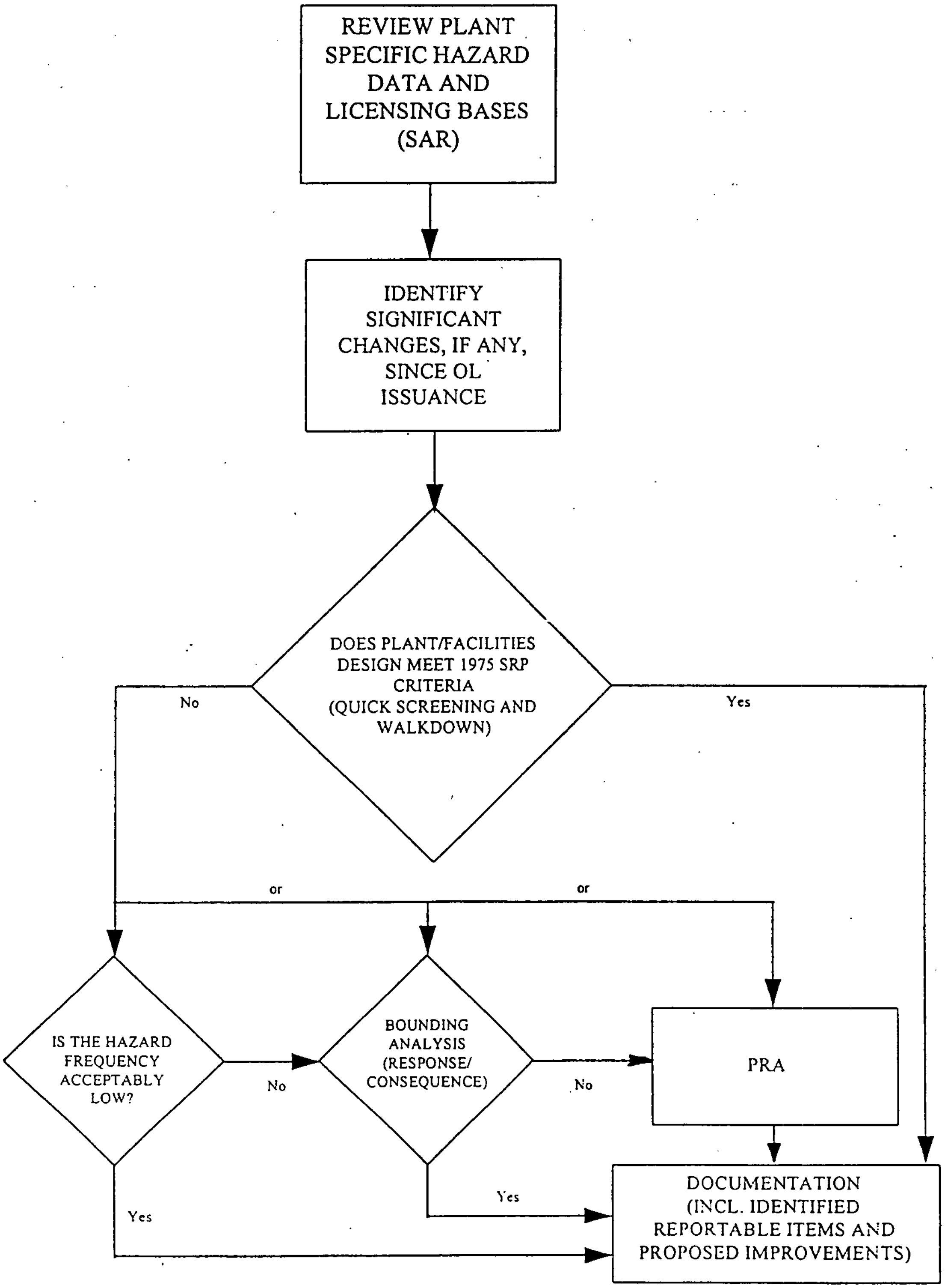


FIGURE 5.0-1
NUREG-1407 PROGRESSIVE SCREENING APPROACH

TABLE 5.0-1
OTHER EXTERNAL EVENT SCREENING

Event	Applicable Screening Criterion	Remarks	LGS UFSAR Section	1975 SRP Section
AIRCRAFT IMPACT	None	Safety-related structures at LGS are reviewed for adequacy against missiles externally generated by aircraft accidents. These will be evaluated as part of the transportation and nearby facility accidents, NUREG-1407 Section 2.5.	3.5.1.6	3.5.1.6
AVALANCHE	3	*Topography is such that no avalanche is possible.*	--	--
COASTAL EROSION	3	Not relevant to LGS site. LGS is not located in coastal plain.	--	--
DROUGHT	2 & 5	Drought is not a concern at LGS. Cooling water is provided by the Ultimate Heat Sink (UHS) (Spray Pond). The UHS is designed such that make-up water is not required for at least 30 days.	--	--
EXTERNAL FLOODING	None	To be included in IPEEE evaluation.	2.4.2 & 3.4	2.4.2 & 3.4.1
DAM FAILURE	4	(Seismic Failure) - There are three (3) major dams whose seismic failure could generate significant waves in the LGS reach of the Schuylkill River. The impact of flooding due to dam failure will be assessed as part of external flooding evaluation per NUREG-1407, Section 2.4	2.4.4	2.4.4
HIGH WINDS AND TORNADOES	None	To be included in IPEEE evaluation	2.3.1.2.1.2 & 3.3.2	2.3.1 & 3.3.2
FOG	2	Fog can increase the frequency of occurrence for other events such as aircraft, railway and highway accidents. These accidents are evaluated as part of IPEEE.	2.3.2.1.6	2.3.2

TABLE 5.0-1
OTHER EXTERNAL EVENT SCREENING

Event	Applicable Screening Criterion	Remarks	LGS UFSAR Section	1975 SRP Section
FOREST FIRE	2 & 3	LGS site is cleared of forestry and external fires are unlikely to spread onsite.	2.2.3.1.4	2.2.3
FROST	2	A survey (ref. 2.3.1-13 LGS UFSAR) indicates ice or freezing rain may occur up to four times a year. Glaze accumulation greater than 0.25 inches is expected only once per year. Loads induced on structures due to frost are much lower than snow loads. Snow loads at LGS are conservatively assumed at 103 psf. Frost may cause complete or partial loss of offsite power which is addressed in the internal events IPE.	2.3.1.2.1.5	2.3.1
HAIL	2	Hail is less damaging than the tornado missile. Therefore, hail as an external event is eliminated.	2.3.1.2.1.4	2.3.1
HIGH RIVER STAGE	4 & 5	Included under external flooding.	--	--

TABLE 5.0-1
OTHER EXTERNAL EVENT SCREENING

Event	Applicable Screening Criterion	Remarks	LGS UFSAR Section	1975 SRP Section
HIGH SUMMER TEMPERATURE/LOW WINTER TEMPERATURE	2	Temperatures in the region of LGS site rarely exceed 100°F or drop below 0°F. The impact that high or low temperatures may have is in the operation of the UHS. The spray pond at LGS is designed for two cases: maximum water consumption (evaporation, drift and seepage) and minimum heat transfer. Long-term data from the National Weather Station in Philadelphia over a period of 40 years, was used for the analysis. The worst 30-day period was used from the data for each case to determine a conservative water loss used for evaporation and drift and the minimum heat transfer. For the winter operation, the spray pond is designed to perform its safety function with an initial ice layer on the pond surface. See external events for ice effects for a detailed explanation. In addition, thermal stresses and embrittlement are insignificant and are covered by design codes and standards for LGS plant design. Based on the above, it can be concluded that temperature effects at LGS as an external event for consideration are eliminated.	2.3.1.1.3	2.3.1
HURRICANES	4	Intense rain precipitation from hurricanes is covered under external flooding while wind forces are covered under high winds and tornadoes.	2.3.1.2.1.1	2.3.1

TABLE 5.0-1
OTHER EXTERNAL EVENT SCREENING

Event	Applicable Screening Criterion	Remarks	LGS UFSAR Section	1975 SRP Section
ICE COVER	2	In the event of an accident, UHS is designed to provide sufficient cooling water to the Emergency Service Water (ESW) and Residual Heat Removal Service Water (RHRSW) systems to dissipate the heat for that accident safely. During winter operations, the spray pond is designed to perform its safety functions with an initial ice layer on the pond surface. During icing conditions, return flow to the pond is initially directed to the winter bypasses, which inject the warm return water directly to the pond volume. The bypasses are directed towards the ends of the pond to allow the return water to circulate and mix with the pond volume, and avoid hydraulic short-circuiting. The increasingly warmer pond water causes any ice layer present on the pond surface to melt. Once a hole is formed in the ice layer, a return path for spray water is available and the spray networks may be used as water temperature dictates.	2.3.1.2.1.5	2.3.1

TABLE 5.0-1
OTHER EXTERNAL EVENT SCREENING

Event	Applicable Screening Criterion	Remarks	LGS UFSAR Section	1975 SRP Section
NEARBY INDUSTRIAL, TRANSPORTATION OR MILITARY FACILITY ACCIDENTS	None	Industrial and transportation facilities within five miles of the LGS site are described in detail in the UFSAR. These facilities may cause accidents which may pose a threat of severe reactor core damage, hence, a possibly large release of radioactivity to the environment. These facilities are: 5 Oil and gas pipeline 6 Manufacturing and processing plants, including a chemical company 7 Railroad lines 8 Airports (public and private) 9 Transportation routes There are no military installations within five miles of the site and no commercial boating traffic on the Schuylkill River in the vicinity of the site except some small pleasure boating during the warmer weather. Therefore, military installations and commercial boating traffic as an external event for consideration are eliminated.	2.2	2.2.1
LANDSLIDE	2 or 3	The topography of the LGS site is such that this event cannot affect the plant. Therefore, it will not be included in IPEEE. In addition, ref. 5.0-3 investigated the stability of slopes (rock and soil) for the LGS relative to geotechnical seismic issues and concluded that: "There appears to be no geotechnical issues that would affect the success path at this plant considering an SME having a peak site acceleration of 0.3g."	2.5.5	2.5.5

TABLE 5.0-1
OTHER EXTERNAL EVENT SCREENING

Event	Applicable Screening Criterion	Remarks	LGS UFSAR Section	1975 SRP Section
LIGHTNING (SEVERE WEATHER PHENOMENON)	2	Thunderstorm and lightning are a seasonal phenomenon in the region of the LGS site. It is estimated that 27 to 32 thunderstorm days per year occur, with 90% of these occurring between the months of April and September. The primary impact of lightning (Section 2.6 of NUREG-1407) is the loss of offsite power, which is included as part of the IPE process. The primary equipment shown to be vulnerable to lightning damage is meteorological equipment which does not prevent safe shutdown of the plant and actions have been taken to improve lightning protection of the meteorological instrument towers at LGS. Therefore, it will not be included in IPEEE.	2.3.1.2.1.3	2.3.1
LOW WATER CONSIDERATIONS (RIVER WATER LEVEL)	2	Extreme low flow in the Schuylkill River does not affect the ability of any LGS safety-related facility to perform adequately, including the UHS. The UFSAR Section 9.2.6 discusses the Design Basis of the UHS. The spray pond is capable of providing cooling water for at least 30 days of operation, taking into account evaporation, drift and seepage as well as other water-loss events such as SSE, tornado, drought, etc.	2.4.11	2.4.11
METEORITE	1	This event has a very low probability of occurrence, less than 10^{-9}; therefore, is eliminated on the basis of low frequency (NUREG/CR-5042, Supplement 2).	--	--
PIPELINE ACCIDENTS (GAS, ETC.)	None	Included under Transportation and Nearby Facility Accidents	2.2.2.3	2.2.2

TABLE 5.0-1
OTHER EXTERNAL EVENT SCREENING

Event	Applicable Screening Criterion	Remarks	LGS UFSAR Section	1975 SRP Section
VOLCANIC ACTIVITY	3	Not relevant to LGS site. There are no active volcanoes nearby.	--	--
WAVES	3 & 4	The LGS site is located on the Schuylkill River which is not subject to severe wave action. In addition, wave action due to dam failure is included under external flooding.	--	--
MISSILES GENERATED BY NATURAL PHENOMENA	4	Only tornado generated missiles will be considered as a concern at LGS from natural phenomena. These will be evaluated as part of tornado loadings.	3.5.1.4	3.5.1.4
MISSILES GENERATED BY EVENTS NEAR THE SITE	4	The postulated missiles resulting from events near the site such as missiles generated by possible train explosion will be evaluated as part of transportation and nearby facility accidents.	3.5.1.5	3.5.1.5
SNOW	2	Roofs of all structures are designed for a snow load of 103 psf (conservative). Snow storms may cause loss of offsite power which is addressed in the internal event IPE; therefore, it will not be included in IPEEE.	2.3.1.2.2	2.3.1
STORM SURGE	4	Included under external flooding	--	--
TSUNAMI	3	This event cannot occur close enough to the site. Tsunamis are rare on the East Coast and LGS is far inland. Tsunami as an external event for consideration is eliminated.	2.4.6	2.4.6

TABLE 5.0-1
OTHER EXTERNAL EVENT SCREENING

Event	Applicable Screening Criterion	Remarks	LGS UFSAR Section	1975 SRP Section
TOXIC GAS (EXPOSURE TO HAZARDOUS CHEMICAL RELEASE)	4	Control Room personnel could potentially be exposed to hazardous chemical vapors from chemical spills from nearby rail lines, highways, nearby industrial facility or from onsite chemical storage. These will be evaluated per NUREG-1407 Section 2.5.	2.2.3.1.3	2.2.3
INTENSE PRECIPITATION	4	Rain precipitation is included in the assessment of external flooding. Intense local rain precipitation may cause flooding of nearby streams and of the Schuylkill River. The impact of flooding on LGS safe operation will be assessed as part of external flooding evaluation.	2.3.1.1.5 & 2.4	2.3.1 & 2.4
RELEASE OF CHEMICALS FROM STORAGE ONSITE	4	Included under Toxic Gas	--	--
RIVER DIVERSION	1	Not applicable to LGS. River is well regulated and essential water supply is not dependent upon stream channels.	2.4.9	2.4.9
SANDSTORM	1	This is not relevant at LGS site.	--	--
SEICHE	3	Seiche flooding is not applicable to the LGS site. There are no large bodies of water nearby, such as a lake. Thus, Seiche flooding as an external event for consideration is eliminated.	2.4.5	2.4.5
PROBABLE MAXIMUM FLOODS ON STREAMS AND RIVERS	4	The impact of flooding on LGS safe operation will be assessed as part of external flooding evaluation. Safety-related structures at LGS are secure from flooding. Hence, flooding protection requirements are not required. See external flooding evaluations for details.	2.4.3	2.4.3

TABLE 5.0-1
OTHER EXTERNAL EVENT SCREENING

Event	Applicable Screening Criterion	Remarks	LGS UFSAR Section	1975 SRP Section
COOLING WATER CANALS AND RESERVOIR	4	There are no canals in the cooling water system. LGS has a spray pond which serves as the UHS for the RHRSW and the ESW systems after a possible accident. The hydrologic design of the spray pond and its possible impact on flooding will be assessed as part of the external flooding evaluation per NUREG-1407 Section 2.4	2.4.8	2.4.8
CHANNEL DIVERSION	1 & 4	Not applicable to LGS. Essential water supply at LGS is not dependent upon stream channels.	2.4.9	2.4.9
EXPLOSIONS	4	Explosion can potentially occur due to accidents on the nearby railway line, highway or pipelines. Therefore, these will be evaluated per NUREG-1407 Section 2.5	2.2.3.1.1	2.2.3
FLAMMABLE VAPOR CLOUDS	4	These can result from rupture of a natural gas pipeline adjacent to the site. Therefore, these will be evaluated per NUREG-1407 Section 2.5.	2.2.3.1.2	2.2.3

5.1 High Winds

High or extreme winds present a potential threat to a nuclear power plant. The components of the general atmospheric circulation which give rise to extreme winds are extratropical cyclones, tropical cyclones, and tornadoes. The intensity and occurrence frequency of winds which are generated by these components are a function of the climatic conditions of the geographic area in which the plant is situated.

Winds have a number of effects on structures within their path. They can apply effective external pressures to structures, they can create external/internal pressure differentials in closed structures, and they can generate missiles which are carried with potentially damaging kinetic energies. The winds associated with tornadoes are typically the most intense and highest in magnitude. Tornadoes can eject large damaging missiles with high kinetic energies. Tornadoes can provide the controlling wind related loads which must be considered in the wind resistant design of nuclear power plant structures and the protection of components required to safely operate the plant.

5.1.1 Plant Design Basis

This section provides a comparison of the wind resistant design of LGS Units 1 and 2 to the requirements of the SRP. Design and construction of the LGS Units was initiated prior to the NRC's issuance of the SRP. As a result, it was anticipated that there may be differences in the LGS design and the SRP requirements.

The comparison focuses on three principal elements of the SRP criteria for the extreme wind design of nuclear power plants:

- Definition of climatic conditions, average and extreme, which may affect the plant site. This includes determination of the 100-year return period "fastest mile of wind" and the design basis tornado characteristics (SRP Sections 2.3.1 and 2.3.2)
- Evaluation of high wind loading (SRP Section 3.3.1)
- Evaluation of tornadic wind loading including potential tornado generated missiles (SRP Sections 3.3.2 and 3.5.1.4)

The following sections provide the details of the comparison of the LGS design with the SRP criteria.

5.1.1.1 Design Wind Velocity and Loadings

The basis for the design wind velocity provided in the SRP is ANSI A58.1 Minimum Design Loads for Buildings and Other Structures (ref. 5.1-1). This standard provides the extreme fastest mile wind speed for a 100-year return period. The variation in wind velocity with height above ground and factors to account for the fluctuating nature of wind velocities (gust factors) are also provided.

All exposed structures at LGS are designed to withstand a basic wind velocity of 90 mph at 30 feet above ground (UFSAR Section 3.3.1). The recurrence interval of this wind velocity was estimated to be at least 100 years using the guidance provided in ASCE Paper 3269, "Wind Forces on Structures" (ref. 5.1-2). A gust factor of 1.1 was used in conjunction with this basic wind velocity and the variation of wind velocity with height was determined and is given in Table 3.3-1 of the UFSAR. This is in agreement with the SRP requirements.

In addition, the 100 year recurrence interval fastest mile of wind to be expected at LGS is 82 mph (UFSAR Section 2.3.1.2.5). This value is obtained from the work of Thom, "New Distribution of Extreme Winds in the United States", Journal of Structural Division, ASCE, 1968, and is valid 30 feet above the ground (ref. 5.1-3).

Section 3.3.1 of the SRP provides guidance for the criteria for the design of structures which must withstand the effects of the design wind. The design wind has previously been defined as the fastest mile of wind with a 100-year recurrence interval. To adequately account for the appropriate wind loading, the wind velocity variation with height, and fluctuating nature of wind velocities (gust factors) must be considered. These factors combine to provide a design wind definition as a function of the height above ground. The SRP requirements and the LGS design basis (UFSAR Section 3.3.1.2) for the design wind definition were compared and were found to be in agreement with each other.

Based on the above comparison, it can be concluded that the LGS basic wind speed and loadings meet the SRP requirements and can be screened per NUREG-1407.

5.1.1.2 Tornado Velocity and Loadings

In the 1975 SRP Sections 2.3.1 and 3.3.2, the NRC provided specific criteria for establishing the parameters for the tornado design of nuclear power plants. These criteria included the specific requirements for tornado

design provided in Regulatory Guides 1.117 and 1.76 (ref. 5.1-4 & 5.1-5). Additionally, the SRP provided guidance on the spectrum of potential missiles to be considered in the tornado design of the plant in Section 3.5.1.4.

In Regulatory Guide 1.76, the NRC adopted the regionalization scheme proposed by Markee, in Technical Basis for Interim Regional Tornado Criteria (ref. 5.1-6). Using this scheme, the LGS site falls into Tornado Intensity Region I. For each of the tornado intensity regions, Markee has provided values to develop a definition for a design basis tornado in terms of six tornado parameters. These parameters are:

- Maximum wind speed
- Rotational speed
- Translational speed (maximum and minimum)
- Radius of maximum rotational speed
- Pressure drop
- Rate of pressure drop

The tornado resistant design of the LGS site was completed prior to the introduction of the aforementioned regionalization and the issuance of Reg. Guide 1.76. The parameters of the design basis tornado were based on the state of tornado knowledge at the time. A comparison of design basis tornado characteristics provided in Reg. Guide 1.76 and the design basis of LGS is provided in Table 5.1-1.

From Table 5.1-1, it is noted that there are differences in the definition of the design basis tornado used for LGS and that specified in the SRP. Specifically, the three parameters which require further discussion are the maximum wind speed used for design, the radius of maximum rotational speed, and the rate of pressure drop.

The radius of maximum rotational speed is not significant to the LGS design. Category 1 structures for LGS were designed considering a uniform pressure resulting from the 300 mph wind velocity. Conservatively, the variation in resultant wind pressure with distance from the funnel center, as the funnel moves along the structure, was not considered.

The LGS design considers Category I structures to be unvented and thus structures were designed to withstand the maximum pressure drop of 3.0 psi. Blow-out panels are provided, where required, to lower the design differential pressure. The rate of depressurization would have no significant effect on the external structural elements of tornado resistant structures, since these structures were designed for the maximum pressure differential

due to a tornado.

The maximum tornado wind speed used for the design of Category 1 structures at LGS is lower than that provided by the SRP design basis tornado (DBT). This may suggest that the LGS design is not as robust as that suggested by the SRP. The following discussion provides the basis for screening out the LGS design basis tornado and tornado loading from further consideration.

The NRC states in Regulatory Guide 1.76, "If a DBT proposed for a given site is characterized by less conservative values for the parameters than the regional values in Table I, a comprehensive analysis should be provided to justify the selection of the less conservative design basis tornado".

Section 3.3.2.1 of the UFSAR performed a comprehensive analysis of the LGS site to justify the design basis tornado criteria that was used for the LGS. Tornado data was obtained from the National Severe Storms Forecast Center (NSSFC) for the years 1950 through 1981 for an area of 125 nautical miles in radius centered in Pottstown, Pennsylvania. The tornado data obtained is shown in Table 3.3-3 of the UFSAR. The UFSAR determined that the LGS site specific design basis tornado maximum wind speed is 280 mph, which is less than the UFSAR design basis tornado speed of 300 mph. Therefore, the site specific tornado is less than the design basis tornado.

Additional tornado data between 1981 through 1990 was obtained from the NSSFC for review for the LGS IPEEE and is shown in Table 5.1-2. The additional data of 49 tornadoes are in good agreement with the UFSAR data and support the conclusions reached in the UFSAR analysis.

In addition, the NRC, in NUREG-0991, "Safety Evaluation Report Related to the Operation of LGS Units 1 and 2" (ref. 5.1-7), has reviewed the design basis tornado requirements and concluded in the Limerick SER Section 2.3.1 as follows:

"The characteristics of the design-basis tornado considered by the applicant for the Limerick plant are different from the recommendations of RG 1.76, "Design Basis Tornado for Nuclear Power Plants", for this region of the country. The applicant's design-basis tornado has a 300-mph rotational velocity, with a translational velocity of 60 mph, a total pressure drop of 3 psi, and a rate of pressure drop of 1 psi per second. The recommended values in RG 1.76 are a 290-mph rotational velocity, a 70-mph translational velocity, a total pressure drop of 3 psi, and a rate of pressure drop of 2 psi per second. In a letter from E.J. Bradley to A. Schwencer, dated July 15,

1983, the applicant provided a comprehensive analysis of tornado parameters based on local tornado data. The results of this analysis show that the applicant's design-basis tornado parameters meet the RG 1.76 and WASH 1300 criterion of probability of occurrence of 10^{-7} per year or less. This analysis has also been confirmed by an independent NRC staff analysis. Therefore, the NRC staff concludes that adequate tornado parameters have been considered in plant design."

It can be concluded from the above discussion that LGS meets the DBT wind speed and loading requirements set forth in SRP acceptance criteria and can be screened per NUREG-1407.

The SRP, Sections 3.3.2 and 3.5.1.4, require that nuclear plants protect safety-related equipment against damage from missiles which might be generated by the design basis tornado. The criteria and procedures utilized for the design of Category I structures, shields, and barriers to withstand the effects of these missiles are provided in SRP Section 3.5.3.

Section 3.5.1.4 of the LGS UFSAR addresses missiles generated by the design basis tornado. UFSAR Table 3.5-4 lists the missiles that were used in the design and assessment of Category I structures. Exterior wall and roof thicknesses have been evaluated for the UFSAR missiles and are capable of withstanding all the missiles. The minimum thickness for walls is 24" and for roofs is 18" which exceeds the minimum acceptable missile barrier thickness requirements specified in Table 1 of the SRP (NUREG-0800, July 1981) section 3.5.3.

A comparison of the missiles considered in the design of LGS Category I structures, shields, and barriers and the minimum spectrum of missiles required by the SRP is provided in Table 5.1-3. It is evident from Table 5.1-3 that the missiles considered in the LGS design bound the energies and damage potential associated with the spectrum of SRP missiles.

LGS is also in conformance with Reg. Guide 1.117 regarding systems to be protected from tornado missiles. The exception is the ultimate heat sink. A detailed explanation and write-up in Section 3.5.1.4 of the UFSAR is provided to demonstrate the conservatism built into the system to lower the probability of a tornado missile accident which might cause damage.

The intent of the acceptance criteria of SRP Section 3.5.1.4 for tornado missiles is met. That is, the effects of design basis events have been adequately considered. The effect of these accidents on the safety-related structures have been performed and measures to mitigate the consequences have been taken.

Based on the above comparison, it can be concluded that LGS meets the tornado missile and loading requirements set forth in the SRP acceptance criteria and can be screened per NUREG-1407.

SRP Section 3.3.2 provides the requirements for developing and combining the three basic components of tornado loading; 1) effective pressures due to wind velocity, 2) differential pressures between the interior and exterior of the structure, and 3) impact forces resulting from tornado missiles. Additionally, the SRP provides requirements for the combination of these components.

Section 3.3.2.2 of the UFSAR describes the criteria of determining tornado forces on structures. Tornado wind velocity was calculated into effective pressure applied to structures based on ref. 5.1-2. Velocity and velocity pressure were assumed constant with height and a gust factor of 1.0 was used. The maximum differential pressure drop of 3.0 psi was used as loading on fully enclosed structures. Tornado missile dynamic loads were transformed into effective loads to determine the structural response of elements subjected to missile impingement. All of the above design basis tornado loadings were considered to act simultaneously.

The LGS design utilized criteria consistent with the SRP for the development of effective pressures due to tornado wind and appropriately combining the tornado wind load with tornado differential pressure load and tornado missile load.

There were no missiles identified during the walkdown which would potentially be more damaging than those which are addressed by the LGS UFSAR Design Basis or the SRP. In addition, the perimeter of the power block structures was walked down to determine if there were any credible unprotected openings in the Category I structures large enough to allow the entry of tornado missiles. None were found.

Based on the above discussion, it can be concluded that the LGS design basis tornado and associated tornado loadings meet the SRP requirements and can be considered screened per NUREG-1407.

Table 5.1-1

COMPARISON OF REG. GUIDE 1.76 AND LGS DESIGN BASIS TORNADO

Tornado	Rotational Speed (mph)	Translational Speed (mph)		Maximum Wind Speed (mph) (for design)	Radius of Maximum Rotational Speed (ft)	Pressure Drop (psi)	Rate of Pressure Drop (psi/sec)
		Maximum	Minimum				
LGS Design Basis	300	60	N/A[1]	300[2]	N/A[1]	3.0	1.0
Reg. Guide 1.76 (Region I)	290	70	5	360	150	3.0	2.0

1 These parameters have not been specified.

2 For Limerick, the design basis tornado maximum wind speed is based on a site specific evaluation.

Table 5.1-2

WIND SPEED AND CUMULATIVE WIND SPEED DISTRIBUTION FOR TORNADOES WITHIN 1° SQUARE AROUND LIMERICK SITE[1]

Wind Speed Classification	No. of Tornadoes	Percent of Total
F5 (261 - 318 mph)	0	0.0
F4 (207 - 260 mph)	0	0.0
F3 (158 - 206 mph)	1	2.0
F2 (113 - 157 mph)	16	32.7
F1 (73 - 112 mph)	23	46.9
F0 (40 - 72 mph)	9	18.4
F-1 (<40 mph)	0	0.0
TOTAL	49	100

1 Tornado data between 1981 - 1990; Data obtained from National Climatic Data Center, Asheville, North Carolina

Table 5.1-3

COMPARISON OF SRP AND LGS DESIGN BASIS TORNADO MISSILES

LGS Missile	SRP Missile	LGS/SRP Weight (lb)	LGS Impact Area (ft^2)	LGS Horizontal[4] Velocity (mph)	SRP Horizontal Velocity (mph)	SRP Height Above Ground (ft)
1. Wood Plank (4" x 12" x 12')	A	200	0.333	300	288	ANY
2. Steel Pipe (3" dia x 10', Sch. 40)[3]	B	78	0.067	144	144	ANY
3. Automobile[2] [3]	G	4000	20	72	72	30
4. Steel Rod (1" dia x 3')	C	8	0.007	216	216	ANY
5. Utility Pole (13½" dia x 35', not more than 30' above all grade elevations within ½ mile of the plant)	F	1490	1.266	144	144	30
6. Steel Pipe (6" dia x 15', Sch. 40) [1]	D	285	0.239	144	144	ANY
7. Steel Pipe (12" dia x 15', Sch. 40) [1]	E	743	0.886	144	144	ANY

NOTES:

(1) The design basis for LGS included only missiles 1, 2, 3, 4 and 5. All safety-related structures and openings have been assessed for the effects of missiles 6 and 7.

(2) LGS was originally designed for a postulated automobile missile not more than 25 ft above grade for all safety-related structures. All safety-related structures have been reassessed for the effect of the automobile at elevations up to 30 ft above grade levels within ½ mile of the plant.

(3) LGS was originally designed for postulated missile velocities equal to 100 mph for the 3 in diameter steel pipe and 50 mph for the automobile. All safety related structures have been reassessed for the revised velocities shown on the table.

(4) These missiles are considered to be capable of striking in all directions with vertical velocities equal to 80% of the horizontal velocities.

5.2 Floods

Extreme floods (or high water level) present a potential threat to a nuclear power plant. High water levels at a plant site can be caused by a single source or a combination of sources: stream flooding, surges, seiches, tsunamis, dam failures, landslides, and ice melt. The water levels associated with stream flooding or stream flooding coincident with upstream dam failure are, however, generally much higher than those associated with other sources for inland plant sites. Therefore, stream flooding or stream flooding coincident with upstream dam failure typically provides the controlling water level which must be used in the design of nuclear power plant structures and the protection of components required to safely operate the plant.

Floods have a number of effects on structures within their path. They can apply effective external pressures to structures (hydrostatic loads), they can create buoyant forces (uplift) on closed structures, and they can apply dynamic forces generated by wave activity. Flooding, rather than normal groundwater elevation, often provides the controlling flood related loads which must be considered in the design of nuclear power plant structures and the protection of components required to safely operate the plant.

The following sections provide a discussion of the licensed extreme flood design basis for LGS Units 1 and 2 with the requirements of the 1975 Standard Review Plan (SRP).

5.2.1 SRP Requirements

SRP requirements for external floods are addressed in Sections 2.4.2 through 2.4.7. This section summarizes and identifies the individual types of flood-producing phenomena applicable to the LGS site, or combination thereof, considered in establishing the flood design bases for safety-related plant features.

Section 2.4.2 provides the acceptance criteria for providing information on flood history, flood design consideration and effect of intense precipitation. Flood history and the potential for flooding are to be reviewed for the following sources and events applicable to the LGS site:

1. Stream flooding
2. Seismically-induced dam failure
3. Ice loadings from water bodies

SRP acceptance criteria for these events are covered in the following sections:

- Probable Maximum Flood (PMF) on Streams and Rivers (Section 2.4.3)
- Potential Dam Failures (Seismically Induced) (Section 2.4.4)
- Ice Flooding (Section 2.4.7)

Flood design considerations are provided by estimating controlling flood levels. Documentation and justification of the estimated controlling levels are to be provided for review.

Effect of intense precipitation is to be provided. For example, the Probable Maximum Precipitation (PMP) and the capacity of site drainage facilities (including drainage from the roofs of buildings and site ponding). Conclusions relating to the potential for any adverse effects of blockage of site drainage facilities by debris, ice or snow should be based upon conservative assumptions of storm and vegetation conditions likely to exist during storm periods.

Appropriate sections of Reg. Guide 1.59 (ref. 5.2-2) were used to provide regulatory guidance. For example, for estimating the design basis for flooding, the worst single phenomena and combination of less severe phenomena were considered. Reg. Guide 1.135 (ref. 5.2-1) describes methods for determining normal water levels. Reg. Guide 1.29 (ref. 5.2-2) identifies the safety-related structures, systems and components and Reg. Guide 1.102 (ref. 5.2-3) describes acceptable flood protection to prevent the safety-related facilities from being adversely affected.

Publication of the U.S. Geological Survey (USGS), National Oceanic and Atmospheric Administration (NOAA), Soil Conservation Service (SCS), U.S. Army Corps of Engineers, applicable state and river basis authorities and other similar agencies shall be used to obtain data related to hydrology and extreme events in the region.

5.2.2 Plant Design Basis

The hazard to LGS from all sources of water that are located outside the plant, including flooding due to severe local precipitation were considered. The LGS UFSAR attempts to systematically respond to the SRP requirements by addressing plant specific flooding vulnerabilities that may result in severe accidents. Section 2.4 of the UFSAR provides the design basis flooding-produced phenomena based on the plant and site physical, topographical and geological conditions.

Section 2.4.1.1 provides hydrologic descriptions of the site and facilities. The locations of the site and major plant structures with respect to the surrounding topography are shown in Figure 2.4-1 of the UFSAR.

Hydrology characteristics are discussed in 2.4.1.2. Plant site location with respect to the Schuylkill River is given. Specifics about the river are also provided. For example, the watershed of the Schuylkill River lies entirely in southeastern Pennsylvania. The basin is about 80 miles long by 25 miles wide and encompasses an area of 1909 square miles above its confluence with the Delaware River in Philadelphia. The principal towns and cities along the river are also provided. The principal uses of the river are municipal and industrial water supply. The river is also used for recreational fishing and boating.

Section 2.4.1.2.2 discusses existing and proposed water control structures. Figure 2.4-3 of the UFSAR shows the location of 23 small dams upstream of LGS. Heights, volume and drainage areas are given in Table 2.4-2. Based on the UFSAR, none of these dams are close enough and large enough to threaten LGS in the event of failure. There are, however, three significant water control structures upstream of LGS. These are: Ontelaunee, Blue Marsh and Maiden Creek dams. Their locations are also given in Figure 2.4-3. Their general design characteristics are summarized in Table 2.4-3 and are further discussed in detail in Section 2.4.4.

Flood history is discussed in detail in Section 2.4.3.5.2. This section reads as follows:

> "Flood-producing storms in this area are normally associated with tropical disturbances. Although flooding from snow melt occurs annually, snow melt run-off usually has not been associated with major historic floods. Peak stages and discharges published by the USGS and the U.S. Army Corps of Engineers for the major historic floods are given in Table 2.4-6 for several stations on the Schuylkill River. At Pottstown, the 1902 flood, with a peak discharge of 53,900 cfs, was the highest known until June, 1972. However, the Reading and Philadelphia data indicate that the 1902 flood was very likely exceeded in 1850 and 1869 and may have been exceeded in 1757 and 1839.
>
> In June, 1972, Hurricane Agnes produced the flood of record on many Pennsylvania streams. The flow at Pottstown has been evaluated as 95,900 cfs by the USGS. Figure 2.4-10 shows the flood frequency curve for the Schuylkill River at

Pottstown. This curve is based on composite regional flood discharge relationships given in "Magnitude and Frequency of Floods in the United States", Water Supply Paper No. 1672, Part 1-B USGS (1968). It is not expected that the 1972 flood alters these regional relationships."

The UFSAR also discusses potential flood-producing phenomena as recommended by SRP Section 2.4.2. For example:

Stream Flooding

UFSAR Section 2.4.3 discusses the Probable Maximum Flood (PMF) for the Schuylkill River at LGS. The original PMFs were included in the PSAR. The PMF developed by the U.S. Army's Corps of Engineers for Pottstown was estimated at 356,000 cfs with a stage at the site of 158 feet. However, due to the effect of Hurricane Agnes (1972) and the authorization and construction of Blue Marsh and Maiden Creek dams, a new flood analysis was required. Reg. Guide 1.59 Revs. 1 and 2 were issued giving the option of using either detailed flood routing studies (Appendix A) or enveloping maps for determining peak PMF flows (Appendix B). The latter method was used, although more conservative, giving a value of 500,000 cfs versus the original estimate of 356,000 cfs. PMF computed for the Schuylkill River at LGS shows that the maximum resulting stage of the Schuylkill River at el. 181'. The PMF for Possum Hollow Run is discussed in detail in Section 2.4.2.3.5. The conclusion states that the PMF for Possum Hollow Run coincident with the Standard Projected Flood (SPF) in the Schuylkill River would not flood any safety-related structures.

Potential Dam Failures, Seismically Induced

UFSAR Section 2.4.4 discusses the potential failure of three major dams upstream of LGS whose seismic failure could generate significant waves in the LGS reach of the Schuylkill River. Table 2.4-2 lists minor dams that are either too small or too remote to cause significant flooding at LGS in the event of seismic failure. Table 2.4-3 lists the three major dams of concern. The dams are: Ontelaunee, Blue Marsh and Maiden Creek dams. Due to their design parameters, these dams cannot be considered seismic qualified. Their failure is considered simultaneously with the LGS SPF.

Sections 2.4.4.1 through 2.4.4.3.2 provide detailed discussions and analysis sequences on the potential dams' failure. Dam failure permutation and unsteady flow analysis was performed with a final conclusion that the most severe seismic dam break permutation of the three dams: Blue Marsh, Ontelaunee and Maiden Creek, would not endanger safety-related

structures. A simplified analysis, presented in the UFSAR, is justifiable because the plant area is high above the Schuylkill River.

Flood Design Considerations

This section addresses flood design considerations, including GL 89-22. Flood design consideration is provided in UFSAR Section 2.4.2.2. Some of the key points addressed in this section are: the Design Basis Flood Levels (DBFL) with respect to the Schuylkill River is conservatively estimated at el. 207'. This stage is derived from an SPF, combined with the wave crest from three simultaneous dam breaks and the 1 percent wave run-up generated by a 40 mph wind. Without the wave, the maximum level is estimated at el. 201'. The lowest grade level entrance to any safety-related structure is at el. 217', which is 10 feet above the DBFL. Therefore, Schuylkill River floods cannot affect any of the safety-related facilities.

The Schuylkill River PMF is conservatively estimated at 500,000 cfs, based on Appendix B of Reg. Guide 1.59, Rev. 2. When combined with a simultaneous dam break flood wave due to a PMF-induced failure of Ontelaunee dam, the highest stage obtained at LGS was el. 181'. This is well below the stage obtained from the multiple dam break as given above. Also, an analysis of the percolation is provided to ensure that flood water would not reach the nearest safety-related structure inland through the embankments. It was concluded that dam break flood wave would not affect hydrostatic pressure on the foundation of safety-related structures.

UFSAR Section 2.4.2.3 addresses the effects of local intense precipitation. The PMP was obtained from NOAA updated subsequent to Hurricane Agnes. The updated values are given in Table 2.4-7. Site drainage is extensively discussed in the following sections: 2.4.2.3, 2.4.2.3.1, 2.4.2.3.1.1, 2.4.2.3.1.2, 2.4.2.3.1.3, 2.4.2.3.2, 2.4.2.3.3, 2.4.2.3.3.1 and 2.4.2.3.3.2.

UFSAR Section 2.4.2.3.4 provides the criteria for roof loads on safety-related structures that are due to PMP onsite.

New precipitation data for the Allentown and Philadelphia, Pennsylvania areas for years 1963 through 1992 obtained from NOAA local climatological data are shown on Tables 5.2.2-1 through 5.2.2-4 and are compared to the LGS UFSAR values. The mean monthly and annual precipitation data for rainfall and snowfall for Allentown and Philadelphia areas are in close agreement with the mean values shown in the UFSAR. The annual and greatest rainfall monthly, as well as annual and greatest snowfall monthly,

values shown in the UFSAR are greater than, or equal to, the new values obtained from the NOAA publications for the subject areas.

The greatest 24-hour rainfall value used in the UFSAR is 5.89 inches while the new value for the Allentown area is 7.85 inches. Similarly, the UFSAR snowfall value is 21.0 inches vs. the 25.2 inches from the Allentown area. The minor changes in the short duration precipitation near the Limerick site will not appreciably change the Schuylkill River PMF that was conservatively estimated at 500,000 cfs in Section 2.4.2.2 of the UFSAR, based on Appendix B of Reg. Guide 1.59, Rev. 2. In addition, the lowest grade level entrance to any safety-related structure is at elevation 217', which is 10 feet above the DBFL for the Limerick site and thus, a safe margin from flooding exists. Thus, it is concluded that the intent of the GL 89-22 is met by LGS.

In addition, the short duration precipitation increase discussed above will not affect the Limerick site and roof drainage. Section 2.4.2.3.4 of the UFSAR has calculated a 24-hour PMP of 34.4 inches, assuming all roof drains and scuppers are blocked. The highest parapet on any safety-related structure is less than the maximum PMP height; therefore, the design basis for roof load is equivalent to this maximum water depth and the design basis for the LGS safety-related structures is not affected by the increase in short duration precipitation. Thus, it is concluded that the intent of the GL 89-22 is met by LGS.

The plant walkdown which considered the effect of the new site structures listed in section 5.0.3 confirmed that safety-related structures have adequate flood protection and are in agreement with the UFSAR Design Basis Criteria. For example, failure of Units 1 and 2 CST, Refueling Water Storage Tank, and Auxiliary Boiler Fuel Oil Storage Tank will not cause flooding of safety-related structures, systems, or components because the contents of these tanks would be contained within seismic Category IIA earth dikes. Continued compliance with the UFSAR design basis is maintained through Specification 8031-G-32, "General Requirements for Site Flooding Protection for Limerick Generating Station, Units 1 and 2."

Based on the above comparison, it can be concluded that LGS meets the external flooding and associated loading requirements set forth in SRP acceptance criteria and can be screened per NUREG-1407.

Table 5.2.2-1

DISTRIBUTION OF PRECIPITATION (RAIN) FOR ALLENTOWN, PA

COMPARISON OF UFSAR DATA (1944 - 1976) WITH NOAA DATA (1963 - 1992)[1]

	Total Precipitation (inches of water)	
	Mean	Maximum
January	3.15 (3.19)*	8.42 (6.16)
February	2.84 (2.94)	5.44 (5.44)
March	3.50 (3.66)	6.68 (7.21)
April	3.67 (3.84)	7.87 (10.09)
May	4.23 (3.86)	10.62 (7.88)
June	3.70 (3.69)	8.58 (8.58)
July	4.38 (4.30)	10.42 (10.42)
August	4.26 (4.28)	9.42 (12.10)
September	3.96 (4.03)	8.87 (7.69)
October	2.82 (2.74)	5.70 (6.84)
November	3.76 (3.66)	9.69 (9.69)
December	3.52 (3.71)	7.89 (7.89)
ANNUAL	43.80 (43.90)	55.85 (55.85)

Greatest Rainfall	Monthly	12.10 (12.10)	Aug. 1955
	24 hours	7.85	Sept. 1985
		(4.79)	Aug. 1955

* ( ) Values in parenthesis are UFSAR Design Basis Data from Table 2.3.1-5.

[1] "Local Climatological Data, Annual Summary With Comparative Data"

Table 5.2.2-2

DISTRIBUTION OF PRECIPITATION (SNOW, SLEET) FOR ALLENTOWN, PA

COMPARISON OF UFSAR DATA (1944 - 1976) WITH NOAA DATA (1963 - 1992)[2]

	Snow & Sleet (inches)	
	Mean	Maximum
January	8.5 (7.7)	24.1 (24.1)
February	8.9 (8.6)	29.5 (22.4)
March	5.5 (6.1)	17.4 (30.5)
April	0.7 (0.4)'	13.4 (3.1)
May	T[1] (T)[1]	T[1] (T)[1]
June	0.0 (0.0)	0.0 (0.0)
July	0.0 (0.0)	0.0 (0.0)
August	0.0 (0.0)	0.0 (0.0)
September	0.0 (0.0)	0.0 (0.0)
October	0.1 (T)[1]	1.4 (1.4)
November	1.3 (1.4)	7.8 (7.8)
December	6.2 (7.4)	28.4 (28.4)
ANNUAL	31.1 (31.6)	67.2 (67.2)

Greatest Snowfall	Monthly	30.5	Mar. 1958
		(43.2)	Jan. 1925
	24 hours	25.2	Feb. 1983
		(17.5)	Mar. 1958

[1] Trace amount

( ) Values in parenthesis are UFSAR Design Basis Data from Table 2.3.1-5.

[2] "Local Climatological Data, Annual Summary With Comparative Data"

Table 5.2.2-3

DISTRIBUTION OF PRECIPITATION (RAIN) FOR PHILADELPHIA, PA

COMPARISON OF UFSAR DATA (1872 -1976) WITH NOAA DATA (1963 - 1992)[1]

	Total Precipitation (inches of water)	
	Mean	Maximum
January	3.22 (3.17)*	8.86 (6.06)
February	3.02 (3.10)	6.44 (5.43)
March	3.51 (3.51)	7.01 (6.27)
April	3.33 (3.28)	8.12 (6.68)
May	3.46 (3.35)	7.03 (7.41)
June	3.57 (3.65)	7.88 (7.88)
July	4.17 (4.10)	9.44 (8.33)
August	4.40 (4.48)	9.61 (9.70)
September	3.37 (3.40)	8.70 (8.78)
October	2.77 (2.80)	5.12 (5.21)
November	3.12 (3.07)	9.06 (9.06)
December	3.20 (3.19)	7.37 (7.23)
ANNUAL	41.13 (41.10)	(54.41) (--)

Greatest Rainfall	Monthly	9.70	Aug. 1955
		(12.10)	Aug. 1911
	24 hours	5.68	Aug. 1971
		(5.89)	Aug. 1898

* ( ) Values in parenthesis are UFSAR Design Basis Data from Table 2.3.1-4.

(1) "Local Climatological Data, Annual Summary With Comparative Data"

Table 5.2.2-4

DISTRIBUTION OF PRECIPITATION (SNOW, SLEET) FOR PHILADELPHIA, PA

COMPARISON OF UFSAR DATA (1943 -1976) WITH NOAA DATA (1963 - 1992)[2]

	Snow & Sleet (inches)	
	Mean	Maximum
January	6.5 (5.4)*	23.4 (19.7)
February	6.3 (6.1)	27.6 (18.4)
March	3.6 (3.8)	13.4 (13.4)
April	0.3 (0.2)	4.3 (4.3)
May	T [1] (T) [1]	T [1] (T) [1]
June	0.0 (0.0)	0.0 (0.0)
July	0.0 (0.0)	0.0 (0.0)
August	0.0 (0.0)	0.0 (0.0)
September	0.0 (0.0)	0.0 (0.0)
October	0.0 (T) [1]	0.0 (T) [1]
November	0.7 (0.7)	8.8 (8.8)
December	3.5 (4.2)	18.8 (18.8)
ANNUAL	20.8 (20.4)	54.9 (--)

Greatest Snowfall	Monthly	27.6	Feb. 1979
		(31.5)	Feb. 1899
	24 hours	21.3	Feb. 1983
		(21.0)	Dec. 1909

* ( ) Values in parentheses are UFSAR design basis data from Table 2.3.1-4.

[1] Trace amount

[2] "Local Climatological Data, Annual Summary With Comparative Data"

5.3 Transportation and Nearby Facility Accidents

Transportation and nearby facilities present a potential threat to the safe operation of a nuclear power plant. Consequences of transportation accidents or accidents at nearby industrial or military facilities can involve direct collision, pressure loading, missile impact, fire, vapor cloud detonation, and/or drifting of toxic fumes into the control room leading to potential degradation of plant facilities and equipment or incapacitation of plant operators.

The severity of events and the potential vulnerability of the LGS Unit 1 and Unit 2 power plant due to transportation and nearby facility accidents is evaluated in this section. Transportation routes and industrial or military facilities within a five mile radius of the plant site were considered. The U.S. Nuclear Regulatory Commission 1975 Standard Review Plan (SRP) and Regulatory Guide 1.70 (ref. 5.3-1) provide guidance for evaluating such events and their impact on the plant.

5.3.1 SRP Requirements

Standard Review Plan requirements for identification of potential hazards in site vicinity are addressed in SRP Sections 2.2.1, 2.2.2, 2.2.3, 3.5.1.4, 3.5.1.5, 3.5.1.6, 3.5.2, and 6.4.

These potential hazards are reviewed against locations and separation distances from the site of industrial, military, transportation facilities and routes in the vicinity of the site. These facilities and routes include air, ground, water traffic, pipelines, manufacturing, processing and storage facilities.

Section 2.2.1 and 2.2.2 provide the acceptance criteria for the identification of potential external hazards. The criteria is as follows:

1. Data in the SAR adequately describes the locations and distances of industrial, military and transportation facilities in the vicinity of the plant, and is in agreement with data obtained from other sources, when available.
2. Descriptions of the nature and extent of activities conducted at nearby facilities, including the products and materials likely to be processed, stored, used, or transported, are adequate to permit evaluations of possible hazards.
3. Where potentially hazardous materials may be processed, stored, used or transported in the vicinity of the plant, sufficient statistical data on such materials are provided to establish a basis for

evaluating the potential hazard to the plant.

The remaining SRP sections identified above provide the specific acceptance criteria for the events required to be described by Sections 2.2.1 and 2.2.2. These are:

- Evaluation of Potential Accidents (Section 2.2.3)
- Missiles Generated by Natural Phenomena (Section 3.5.1.4)
- Site Proximity Missiles (Section 3.5.1.5)
- Aircraft Hazards (Section 3.5.1.6)
- Structures, Systems and Components to be Protected from Externally Generated Missiles (Section 3.5.2)
- Habitability Systems (Section 6.4)
- Barrier Design Procedures (Section 3.5.3)

5.3.2 Plant Design Basis

The LGS UFSAR Sections 2.2.1 and 2.2.2 describe in detail the type of facilities and major transportation routes within 5 miles of the site. These are:

- Oil and natural gas pipelines
- Transportation routes
- Manufacturing and processing plants, including a chemical company (formerly Hooker, now Occidental)
- Airports (public and private)
- Railroad lines (Conrail)

There are no military installations within 5 miles of the site and no commercial boating traffic on the Schuylkill River in the vicinity of the site except some small pleasure boating occurring in warmer weather.

UFSAR Figure 2.2-1 shows all transportation routes. Figures 2.2-1 and 2.2-4 and Table 2.2-2 show oil and natural gas pipelines located within 5 miles of the LGS site. There is one quarry; Pottstown Trap Rock Quarry, Inc., located 0.8 miles from the site. The location of the quarry is shown on Figure 2.2-2. Industries located within 5 miles of the site are listed in Table 2.1-17 while the locations and descriptions of airports are provided in Section 2.2.2.5.

The description and extent of activities conducted at these nearby facilities is discussed in detail in various sections of the UFSAR. For example, Section 2.2.1 describes the location, and types, of transportation routes within 5 miles. Section 2.2.2.1 describes industries within 5 miles, Table

2.1-17 lists all industries with ten or more employees, products, and location for each establishment.

A description of the hazardous materials stored near LGS is provided in Section 2.2.2.2. Those stored by Hooker (Occidental) Chemical Company are listed in Table 2.2-1. Occidental Chemical has reduced the number of product lines formerly manufactured by Hooker Chemical with an accompanying reduction in the number of stored hazardous chemicals. Explosives and hazardous materials that may be transported on the highways and railroads are discussed in Sections 2.2.3.1.1 and 2.2.3.1.3, respectively.

Natural gas pipelines operated by Columbia Gas Transmission Company and an oil and gasoline pipeline operated by Atlantic Richfield Company (ARCo) are discussed in Section 2.2.2.3. Other pipelines, pipe sizes, ages, operating pressure, operated by various companies, are listed in Table 2.2-2.

Airports are discussed in Section 2.2.2.5. All landing fields within 10 miles of the site are listed in Table 2.2-3. The aircraft crash probability analysis from operation of these airports and airways is provided in Section 3.5.1.6 using the procedures of SRP Section 3.5.1.6.

Explosions, exposures to hazardous chemical release, fires, collision with the intake structure and liquid spills are evaluated for potential accidents such that design basis events are established. The detailed evaluation is provided in Section 2.2.3.1. For example, explosions that can potentially occur due to accidents on the nearby railway line, highways, or pipelines has been performed in conformance with Reg. Guide 1.91 (ref. 5.3-2) methodology. The potential rupture of one of the several nearby natural gas pipelines and subsequent explosion of a gas or vapor cloud has also been postulated. A gas-air mixture approximately 4 times the requirements of Reg. Guide 1.91 (Rev. 1) is conservatively used to develop the explosive pressure for structural assessment.

Exposure of control room personnel to hazardous chemical vapors from a chemical spill is discussed in detail in Section 2.2.3.1.3. Such spills could occur on the rail line, one of several highways close by, nearby industrial facilities, or from onsite chemical storage. Acceptable toxic incapacitation levels were based on compliance with the Reg. Guide 1.78 (ref. 5.3-3) requirement of 2 minutes for operator protective action. Potential chemical hazards were identified by compiling a list of toxic chemicals that could pose a vapor hazard based on Reg. Guide 1.78, NUREG-0570 and other sources. Surveys were then conducted to determine which of these are

actually stored or shipped within 5 miles of the LGS site, with what frequency and in what quantities. An analysis was done to determine which of these chemicals, if spilled, could exceed toxic incapacitation levels in the control room. The analysis was conducted in accordance with the assumption and methodologies recommended by Reg. Guide 1.78 and NUREG-0570. As a result of the analyses, six potentially hazardous chemicals requiring monitoring were identified. These chemicals are listed in Table 2.2-6 and are:

- Ammonia
- Chlorine
- Ethylene Oxide
- Formaldehyde
- Vinyl Chloride
- Phosgene

To mitigate the consequences of such exposure to hazardous chemicals, control room operators are trained and periodically tested on their ability to put on breathing apparatus within 2 minutes after initiation of the toxic chemical alarm. The control room will then be manually isolated by the operators as described in Section 6.4.3.2.3. If chlorine is detected with the control room HVAC in the normal operating mode, automatic isolation of the control room will occur as described in Section 6.4.3.2.1. Section 6.4.1 of the UFSAR discusses the design basis of the LGS Habitability Systems.

The LGS toxic chemical analysis complies with the intent of Reg. Guide 1.78. The analysis goes beyond the methodologies outlined in this guide in the following areas:

- In addition to the chemicals listed on Table C-1 of Reg. Guide 1.78, other chemicals were investigated to determine if potential hazards existed. A total of 153 chemicals were evaluated.
- The models of NUREG-0570 were used to determine the concentrations of hazardous chemicals in the control room.
- The more stringent TLV levels were initially used instead of the Reg. Guide 1.78 Table C-1 toxicity limits to determine which chemicals were potentially hazardous. Table C-2 of Reg. Guide 1.78 was not used to determine which chemicals were hazardous.

The UFSAR also discusses specific acceptance criteria for transportation and nearby facility accidents as recommended by SRP Sections 2.2.1 and 2.2.2. For example:

Missiles Generated by Events Near the Site

This section addresses the SRP Section 3.5.1.5, "Site Proximity Missiles" requirements. The safety-related structures, system and components were reviewed for adequacy against missiles externally generated by railroad explosions. The safety-related facilities are either designed to resist the externally generated missiles or are protected by missile-resistant barriers. The barriers designed to resist externally generated missiles, and the corresponding systems and components, are listed in Table 3.5.7 of the UFSAR. Therefore, the acceptance criteria of the SRP are satisfied and LGS is in conformance with the requirements for missiles generated by events near the site.

Aircraft Hazards

Based on the airport and aircraft information provided in Section 2.2.2, an analysis has been performed using the methodology of SRP Section 3.5.1.6 to demonstrate that the probability of an aircraft accident causing damage to safety-related equipment is lower than the acceptance criteria of SRP Section 3.5.1.6.

UFSAR Sections 3.5.1.6.1 through 3.5.1.6.4 contain the details and assumptions of the analysis. Table 3.5-1 lists the types of aircraft operating out of nearby airports. Based on the estimated maximum kinetic energy at impact, the Learjet is chosen as the design aircraft for the design of safety-related structures. The conclusion of the analysis is as follows:

> "The total probability per year of an aircraft impact resulting in offsite radiological consequences in excess of 10CFR100 guidelines is 9.56×10^{-8} per year. It is concluded that the acceptance criteria of SRP Section 3.5.1.6 are met."

The SARA Report also concludes that aircraft impact at LGS is a negligible contributor to core melt frequency. It agrees with the licensing analysis described in UFSAR which concludes that the frequency of an aircraft impact at LGS that could create a potential nuclear safety hazard is less than 10^{-7} per year.

Structures, Systems and Components to be Protected from Externally Generated Missiles

Seismic Category I and safety-related structures, equipment, and systems are protected from missiles through basic station component arrangement so that the missiles do not cause failure of these structures.

In addition, where it is impossible to provide protection through plant layout, suitable physical barriers are provided to isolate the missile or to shield the critical system or component. Redundant seismic Category I components are also suitably protected such that a single missile cannot simultaneously damage a critical system component and its backup system.

UFSAR Table 3.2-1 provides a tabulation of safety-related structures, systems and components along with their applicable Seismic category and quality group classification. Structures and barriers designed to provide protection from external missiles are listed in Table 3.5-7. Based on a review of these tables, it is concluded that the intent of Reg. Guides 1.13 and 1.27, with respect to providing adequate external missile protection for structures, systems and components important to safety, is met.

Based on the above comparison, it can be concluded LGS meets the transportation and nearby facility accidents requirements set forth in SRP acceptance criteria and can be screened per NUREG-1407.

5.3.2.1 SRP Review Results

A thorough review of Sections 2.2.1, 2.2.2, and 2.2.3 of the UFSAR shows that the acceptance criteria of the corresponding SRP sections are satisfied and that LGS site is in conformance with requirements for the identification of potential hazards in site vicinity and evaluation of potential accidents.

In addition to the UFSAR, the SARA report (ref. 1.1-5) contains a detailed bounding analysis of chemicals, explosives and aircraft crash accidents that could occur near the site. The SARA report conclusion on toxic vapor analysis is: "The bounding estimate of the total predicted frequency of core melt resulting from an accidental release of toxic vapors in the neighborhood of the LGS is 6.3×10^{-8} per year . . ." This frequency is far lower than IPEEE reporting requirements.

5.3.2.2 Plant Walkdown Results

Prior to performing the plant walkdown a review of all existing information pertaining to the LGS plant was conducted in order to familiarize the walkdown personnel with the site. Local government agencies such as Chamber of Commerce, Limerick Fire Department, Pennsylvania Department of Transportation, U.S. Army Corps of Engineers, and the U.S. Military were contacted to identify any new facilities and transportation routes near the site.

The result of the plant site survey via car and investigation can be summarized below:

- Location and transportation routes located within 5 miles of the site are in agreement with Section 2.2.1 of UFSAR and are shown on Figure 2.2-1.
- Oil and natural gas pipelines located within 5 miles of the site are in agreement with UFSAR Table 2.2-4 and Figures 2.2-1 and 2.2-4.
- Location of the two major airports near the site; Pottstown-Limerick Airport and Pottstown Municipal Airport are in agreement with Section 2.2.2.5 of UFSAR.

- Industries within 5 miles of LGS, with ten or more employees, are listed in Table 2.1-17 of UFSAR. UFSAR Section 2.2.2.1 describes the industries nearest to the site. The walkdown confirmed the general location of these industries with the following exceptions:

    a) The Hooker Chemical Company Plant is renamed Occidental Chemical Corporation.
    b) Eastern Warehouses, Inc. Industrial Park appears to be closed.

In general, the plant site walkdown confirmed that the LGS UFSAR Design Basis Evaluation of nearby industrial, transportation and military facilities was accurate and complete. There were no unique concerns or considerations for the LGS plant site identified via the plant site walkdown.

It can be concluded from this review that all design basis analyses meet or exceed the requirements set forth in SRP acceptance criteria for transportation and nearby facility accidents. Therefore, the transportation and nearby facility accidents can be screened per NUREG-1407.

5.4 Others

Section 2.0 of NUREG-1407 identifies specific events evaluated for inclusion in the IPEEE program. Based on the evaluations conducted, the following events were identified in NUREG-1407 for consideration by all licensees in the IPEEE; seismic events, internal fires, high winds and tornadoes, external floods, and transportation and nearby facility accidents. However, NUREG-1407 also requires that each individual licensee confirm that no plant unique external events known to the licensee with potential severe accident vulnerability are being excluded from the IPEEE. As part of the response to the IPEEE, a comprehensive screening of external events as discussed in sections 5.0.1, 5.0.2 and 5.0.4 of this report was performed to assure that no unique events were excluded from the evaluation.

Based on the applied screening and plant walkdown, no unique events were identified for inclusion in the LGS IPEEE. Thus, the scope of the IPEEE for "High Winds, Floods, and Others" as implemented for LGS Unit 1 and Unit 2 included consideration of high winds and tornadoes, floods, transportation and nearby facility accidents, and confirmation that no other plant unique external events with potential severe accident vulnerability were excluded per NUREG-1407 requirements. Therefore, it can be concluded that the LGS IPEEE meets the requirements set forth in NUREG-1407 for high winds, floods, and other events.

6. LICENSEE PARTICIPATION AND INTERNAL REVIEW TEAM

As in the IPE, the maximum benefits from the performance of the IPEEE are obtained if the licensee staff is involved in all aspects of the examination. Such involvement typically provides a more accurate picture of the as-built, as-designed facility and helps the integration of knowledge gained into plant equipment and procedures by allowing early ownership of the IPEEE process and results. This section describes the PECO Energy involvement in the IPEEE and its review, including major comments and resolutions.

6.1 IPEEE Program Organization

The Limerick IPEEE is a follow-up to the original Severe Accident Risk Assessment (SARA) completed in 1983. As such there is a long historical involvement by PECO Energy personnel in the examination of external events at the Limerick Generating Station.

PECO Energy's Nuclear Engineering Division (NED) personnel were primarily involved in the IPEEE development. Two utility structural engineers participated in the seismic analysis and one utility structural engineer participated in the other external events analysis. In addition, the NED structural engineers served on the Seismic Review Team (SRT). Two fire safe shutdown and three probabilistic safety assessment engineers took part in the fire risk analysis and in the selection of the safe shutdown success paths.

The PECO Energy engineers participating in the IPEEE include those involved in the IPE, Appendix R analyses/compliance and SQUG (at PECO Energy's other nuclear plant). Additional expertise was available to the IPEEE team when required. For example, expertise of on-site engineers was used throughout the IPEEE including access to and evaluation of the success path component list (SPCL) components during the seismic margin assessment and evaluation of fire barriers and detection and suppression systems. Management personnel at LGS were also instrumental in review and approval of potential plant improvements for reducing the risk from external events.

Outside consultants were used to supplement the PECO Energy IPEEE team. The consultants consisted primarily of civil/seismic engineers with a wealth of background doing IPEEE examinations and developing plant specific A-46 (SQUG) programs. Consultants were used in the SMA because of the increased level of expert judgement required and to

ensure that a least two seismically knowledgeable engineers were involved in all seismic walkdowns.

Overall there was an approximate 30/70 split between utility personnel and contractor (VECTRA Technologies, Inc.) support in performance of the IPEEE. The seismic and other events evaluation was primarily performed by VECTRA, while PECO Energy was primarily involved in the fire risk analysis. There was equal involvement between the two parties in producing the success path component list (SPCL).

Dr. Robert Kennedy provided consulting services for the seismic portion of the IPEEE and participated in the initial plant walkdowns. NUS Corporation, the contractor for the SARA Report, provided a comparison of the original external events analysis results with the present (IPEEE) results. This comparison is presented in Section 9 of this report.

6.2 Composition of Independent Review Team

Independent review of the IPEEE was achieved in three ways.

(1) PECO Energy contracted with a consulting firm, Programmatic Solutions, Inc. to perform an independent review of the procedures and processes used by the IPEEE team in developing and performing the SMA and other events portions of the IPEEE. The independent review team consisted of Mr. Harry Johnson and Mr. Greg Rahner. Mr. Johnson's review encompassed all seismic and other event evaluation portions of the examination including the project plan, general walkdowns at LGS to develop a sense for the conclusions and recommendations presented by the IPEEE team, and review of the documentation and draft report. Mr. Rahner's review covered all system aspects of the SMA examination including the SPCL development documentation, and a review of the documentation and draft report.

(2) The fire risk analysis was independently reviewed by Mr. Paul Guymer a Senior Executive Consultant with the NUS Corporation. Mr. Guymer performed the original fire risk analysis for LGS in the 1983 SARA Report.

(3) The IPEEE report was also reviewed by Site Engineering structural and fire protection personnel, Licensing, and Nuclear Group management. In addition, potential changes to plant equipment and procedures (including training) to reduce the risk

from external initiating events received extensive review as part of the PECO Energy modification and procedure processes.

6.3 Areas of Review, Major Comments and Comment Response

The Independent Reviewers provided comments on the Seismic Analysis (Section 3), the Internal Fire Analysis (Section 4), and on High Winds, Floods and Others (Section 5). This section summarizes the major technical comments of Messrs. Johnson and Guymer. Mr. Greg Rahmer who reviewed the system aspects of the SMA had no significant comments on the report. To enhance readability, each reviewer comment is followed by its resolution. Editorial comments and those of a minor technical nature received from the independent reviewers were evaluated and, where appropriate, revisions to the IPEEE text have been made but not included below.

6.3.1 Comments of Mr. Harry Johnson on the Seismic Analysis

6.3.1.1 Since the LGS spectrum is not Reg. Guide 1.61, any comparison of Reg Guide 1.61 damping is not relevant and should be deleted.

Response: The analysis does not intend that there is a true comparison with Reg Guide 1.61. Rather, Reg. Guide should only be considered a point of reference. The text in Section 3.1.1.2. was revised to reflect this.

6.3.1.2 It is incorrect to state that the seismic qualification and documentation procedures used for Category I .... meet the provisions of IEEE 344-1975 and Reg Guide 1.100 (see UFSAR Sections 3.10.2.1 and 3.10.2.2). How did this misunderstanding affect preparations for the walkdown?

Response: Section 3.10.1 of Limerick SSER No. 3 provides the NRC Staff's evaluation of PECO Energy's program for qualification of safety related electrical and mechanical equipment for seismic and dynamic loads. The evaluation included a plant site audit to determine the extent to which the qualification of equipment, as installed at Limerick 1, met the licensing criteria described in Reg. Guides 1.100 and 1.92, Standard Review Plan (SRP) Section 3.10 , and IEEE 344-1975 standards and as noted in the Outstanding Issues Section of the SSER closed the seismic/dynamic and enviromental qualification of equipment issue for Limerick 1. Subsequently, in Limerick SSER No. 9 the NRC Staff verified that the seismic and dynamic qualification of Limerick 2 equipment was performed by an extension of the Unit 1 program.

Additional information was included in Section 3.1.1.5 to address Mr. Johnson's comment. Through its review of the referenced SSER's, the walkdown team correctly understood the licensing basis of the plant.

6.3.1.3 The report incorrectly states that the components and their supporting structures ... are identified as seismic Category IIA. In fact, only the support for the component is evaluated. In addition, per Spec 8031-M-4400-2 for many of the commodities the evaluation is performed by rule. Therefore, did the walkdown team correctly understand the actual design basis for Category IIA?

Response: Section 3.1.1.7.2 is correct as worded and in addition is consistent with UFSAR Section 3.2.

6.3.1.4 The report lists several approaches for the assessment of anchorage adequacy. For a reduced plant it is not clear how the first two approaches differ. Also, what specifically were the seismic IPEEE activities performed for anchorage?

Response: The second approach which involves use of the anchorage analysis qualification calculations to show acceptablility at the SSE can be used if the initial walkdown failed to screen the equipment. This approach was used to show that the anchorage of inverters 2AD160 and 2BD160 is acceptable at the SSE.

6.3.1.5 In Section 3.1.4.2(A) what is the basis for the conclusion that the "S" hook on the chain links not being closed is unlikely to cause the light fixture to become unsecured in a seismic event?

Response: The SRT judged that the light fixtures could not become unsecured in a seismic event since the s-hooks are partially closed and a vertical positive slippage of the light from the hooks is not credible for a chain type support. This explanation was added to Section 3.1.4.

6.3.1.6 What is the basis for the SRT judgement that the cracks found on several of the air compressor pedestals are acceptable?

Response: The SRT's judgement was based on the location and size of the cracks. This clarification was added to the table which documents the results of the walkdown.

6.3.1.7 Fire protection piping, if normally dry, was eliminated by the SRT as a flooding or cascading concern. However, was the dry system reviewed to assure that the non-seismic fire detection system, relays, and deluge

valves will remain functional in an earthquake and not release water into the dry system?

Response: The SRT concluded that the fire protection system piping does not represent a credible interaction or flooding source. The detailed rationale for this conclusion was added to Section 3.1.1.7.2.

6.3.2 Comments of Mr. Paul Guymer on the Internal Fires Analysis

6.3.2.1 The analysis assumes a fire induced reactor trip. Were other events considered?

Response: Assumptions regarding spurious ADS actuation were added to section 4.0.2.

6.3.2.2 Explain why only one worst case spurious actuation signal was considered. Appendix R requires consideration of multiple spurious actuation for hi-lo interfaces.

Response: Explanation was added that multiple hot shorts were considered for hi-low pressure interfaces. A reference to the hi-low pressure interface study was added to address why a LOCA resulting from a hi-low pressure interface failure does not have to be postulated.

6.3.2.3 LGS fire areas in the Reactor Enclosure (area 44, 45, and 47, as well as the equivalent unit 2 areas) are connected by an open equipment hatch. Explain how these, and any other non-rated boundaries between fire areas, are treated within the context of the FIVE fire area definition.

Response: The treatment of the open equipment hatch in the units 1 and 2 reactor enclosures is discussed in Section 4.3.3 of the report.

6.3.2.4 For compartment boundaries which were screened on the basis of low combustible loading, walkdowns should have confirmed no combustible concentrations near those boundaries and no combustible continuity. This should be noted in the section which discusses walkdown findings.

Response: Walkdowns were completed to confirm locations and quantities of combustibles in regard to fire compartment boundaries. This information has been added to section 4.2.4 of the report which discusses walkdown findings.

6.3.2.5 Were the guidelines provided in the FIVE training course, the Fire PRA Implementation Guide and NSAC 178L, used in counting ignition sources?

Response: The FIVE methodology was followed in the counting and identification of ignition sources, as stated in the report. Guidelines provided in the FIVE training course were used as a resource in the process.

6.3.2.6 The fire modeling approach does not appear to discuss the possibility of damage to cables located outside the plume, but in close proximity to the ceiling, and therefore in the ceiling jet. This would only apply in situations where the temperature of the plume at the ceiling was substantially greater than the failure temperature of the cable.

Response: As stated in Section 4.4.1 of the report, boundary calculations were performed in each critical fire compartment to analyze the combined thermal effects of the plume and hot jet layer on targets.

6.3.2.7 How were the characteristics of fixed ignition source fires, other than cabinets and lube oil spills, determined? Battery chargers, transformers, inverters and electric motors are examples.

Response: All identified fire sources in critical compartments were analyzed. If specific examples stated were not identified within the report, it is because they were not identified as being located within the compartments being analyzed.

6.3.2.8 Since the method used to calculate cabinet fire sizes is somewhat unique (and does not follow the suggested FIVE HRR) it would add credibility if some comparisons were made with the approach suggested in the EPRI Fire PRA Implementation Guide and the Sandia Cabinet Fire test results. Further explanation of the derating factor would be helpful.

Response: The method for calculating HRR and BTU content of cabinet fires used in the analysis is well documented within the report. The HRR for cabinets referenced in the FIVE methodology was not used as it is overly conservative and is not representative of typical HRR that would be expected at the plant. The method used to derate the cabinets is well documented in the reference.

6.3.2.9 Plant administrative controls are used as a basis for completely discounting the possibility of transient combustible sources causing ignition of intervening combustibles. This may be non-conservative and

not consistent with the approach intended by the authors of the FIVE methodology.

Response: Administrative controls were analyzed and credited as allowed and stated within the FIVE methodology.

6.3.2.10 How was the potential for damage due to the ceiling jet addressed?

Response: The potential impact of the ceiling jet on targets is discussed in detail in section 4.4.1.

6.3.2.11 At relatively low temperatures (150 deg F) sensitive electrical equipment may fail or trip. In relatively small compartments hot gas layer temperatures may reach or exceed these levels. Consideration given to such failure mechanisms should be discussed in the analysis.

Response: No "sensitive" targets that were required to survive were identified within the analyzed fire compartments.

6.3.2.12 The relevance of the Control/Remote Shutdown Circuit Dependencies section is lost for those reviewers who are not familiar with the "Remote Transfer Capability Study". Further explanation is suggested.

Response: Details were added to the subject section to clarify circuit independence and transfer issues.

6.3.2.13 Has the evaluation of control system interactions considered the issue raised by the NRC in IN 92-18?

Response: Design and compliance with SSD alternative shutdown requirements were added to the discussion in Section 4.8.2.5.

6.3.2.14 Did the cross zone fire spread analysis account for the potential for intervening combustible involvement?

Response: Section 4.3.3 discusses possible fire spread across combustible-free zones and open hatchways. Any intervening combustibles introduced into these areas are administratively controlled. These controls were taken into account as allowed within the FIVE metholology.

6.3.3 Comments from Mr. Harry Johnson on High Winds, Floods and Others

7.0 PLANT IMPROVEMENTS AND UNIQUE SAFETY FEATURES

7.1 Unique Safety Features

In order to deal with a wide range of external events, the Limerick design incorporates the following unique safety features:

- Four electrical divisions per unit each with its own diesel generator (for AC) and battery (for DC).
- High degree of compartmentalization in the reactor enclosure reducing the potential for fire damage.
- Conformance to BTP CMEB 9.5.1, Guidelines for Fire Protection for Nuclear Power Plants.
- Location at an elevation above the Schuylkill River sufficient to preclude river flooding.
- Detailed seismic analysis and testing of all safety related structures, equipment, instrumentation, controls and their associated interconnecting electrical cables.

The combination of these features enhances the LGS capability to withstand external events.

7.2 Plant Improvements

The criteria used by PECO Energy to define vulnerability for each of the major areas of external event is as follows:

- A seismic vulnerability is component, system or structure that does not comply with its seismic design bases and provides a major accident risk.
- A fire vulnerability is any fire compartment that is well above the compartment screening criteria of 1E-6.
- A high winds, floods and external event vulnerability is a external event that cannot be shown to meet the requirements of the 1975 Standard Review Plan.

Based on these criteria no vulnerabilities to seismic, fires, high winds or floods or "others" were found to exist.

Actions being taken as a result of the seismic walkdown are categorized as housekeeping and maintenance. The various housekeeping concerns such as unrestrained lifting devices, open S-hooks, and free standing equipment and maintenance concerns (Table 3.1.4-3) such as rusting support bolts and loose battery rack spacer pads will be corrected by December 1995.

Actions credited in the fire analysis but not yet implemented are the following with committed completion dates.

- The station will designate fire compartments 1,7, 22 and 23 as transient combustible free zones by June 1996.

- All wood scaffolding has been replaced with metal scaffolding and procedures will be revised to prevent further use of wood scaffolding by December 1995.

- The combustible control procedure will be revised to provide more conservative combustible control guidelines in safety related areas within the reactor enclosures. The completion date for this procedure revision is December 1995.

- Additional doors will be administratively controlled by the Hazard Barrier Procedure as "fire" doors to limit the amount of air available for combustion. The existing doors are fire rated. The completion date for the procedure revision is December 1995.

Also, as described in Section 4, the analysis of the fire risk took credit for a small number of plant changes being made to address resolution of the Thermo-lag issue. These plant changes are scheduled for completion in association with the safe shutdown Thermo-lag reduction reanalysis.

To manage the risk in the unscreened fire areas, because fixed combustibles present a fire risk impact in fire compartments 2, 20 and 26, the station will increase the fire brigade drill activities and brigade awareness in these areas. The completion date for this is December 1995. These fire compartments 2, 20, and 26 were reviewed against the NEI91-04 (ref. 4.6-1) closure guidance and procedural change is the appropriate level or action.

8.0 SUMMARY AND CONCLUSIONS

An individual plant examination of external events for severe accident vulnerabilities was performed for the Limerick Generating Station. The objective of performing the IPEEE was to identify vulnerabilities, if any, to severe accidents and to report the results together with any licensee determined improvements and corrective actions to the NRC in accordance with the requirements of GL 88-20 Supplement 4.

The IPEEE was divided into three major portions: a seismic analysis, an internal fire analysis, and an examination of high winds, floods and other events.

The IPEEE process has identified the following:

- housekeeping and maintenance concerns related to seismic events that will be resolved and,

- various plant improvements required to allow certain fire areas to be screened out will be implemented. Credit was taken for these improvements in the internal fires analysis.

Improvements to remedy the identified houskeeping and maintenace concerns and those improvements credited in the internal fires analysis will be completed by December 1995, except for designation of transient combustible free zones which will be completed in June 1996. The overall conclusion of the three evaluations was that the Limerick Generating Station has no vulnerabilities to external events.

Consideration of USI A-45, "Shutdown Decay Heat Removal Requirements" was included in the both the IPEEE seismic and internal fires analyses. No significant or unique vulnerabilities were identified in the decay heat removal function. Thus, USI A-45 has been resolved for the Limerick Generating Station. Also, per Section 6.3.3.2 of NUREG-1407, completion of the IPEEE resolves the Eastern U.S. Seismicity Issue for Limerick Generating Station.

9.0 COMPARISON BETWEEN LGS IPEEE AND LGS SARA (SUPPLEMENT 2)

9.1 Introduction

The Limerick Severe Accident Risk Assessment (SARA) (ref. 1.1-5) analysis was performed to estimate the probability of core damage and the offsite consequences from externally initiated events. SARA provided a quantitative estimate of risk whereas the screening methodologies selected for the IPEEE analysis provide similar conclusions without the quantitative rigor. This section will provide a generalized comparison between the previous analyses and the conclusions reached from the IPEEE analyses.

9.2 Seismic Analysis Comparison

The LGS SARA estimates the contribution to the frequency of core damage and to the various accident classes (public risk) from earthquake-induced accident sequences. The process produced a seismic PRA that integrated the seismology of the occurrence frequencies and magnitudes of earthquakes that can affect the region of the plant site with fragility evaluations of the components represented in the system fault trees. Additionally, the system fault trees were expanded to include other components and structures that had potential to significantly influence the likelihood of core damage from seismic events.
For those components that the SARA concluded were significant earthquake-induced failures the associated median ground acceleration capacities are listed in Table 3-1 (of the SARA). Values range from 0.20g to 1.56g. Excluding ceramic insulators in the 500/230-kV switchyard and the condensate storage tank, which are not included in the IPEEE SPCL, the lowest median ground acceleration capacity is 0.67g.

The seismic IPEEE is an EPRI Seismic Margins Assessment (SMA). The review level earthquake was set at the SSE level, 0.15g, and all components on the SPCL were successfully screened for original design criteria and construction implementation. The only issues that emerged from the walkdowns were housekeeping and maintenance which are not structural in nature. Consequently, the conservative deterministic failure margin (CDFM) HCLPF for the plant has been determined to be at least 0.15g.

The above noted median ground acceleration capacities for (SARA) significant earthquake induced failures were converted to CDFM HCLPFs. For all of the components that are also on the IPEEE SPCL, the resulting values are at least 0.30g, which is consistent with the conclusions of the SMA screening.

9.3 Fire Analysis Comparison

9.3.1 Screening Analysis

The IPEEE study identified some fire compartments that were not recognized in the SARA analysis but could potentially have an effect on the ability of the plant to shutdown. These compartments are:

IE	Recombiner Access Area
23	Unit 2 Cable Spreading Room
43	Unit 1 Safeguard System Isolation Valve Area
48A	Unit 1 Laydown and Corridor Area
64	Unit 2 RECW Equipment Area
66	Unit 2 System Isolation Valve Area
67	Unit 2 Safeguard System Access Area
70	Unit 2 Isolation Valve Compartment Area
71A	Unit 2 Laydown and Corridor Area
87	Condensate Pump Rooms, Generator Equipment Areas, Operating Floor

The IPEEE study did not screen out any of the fire compartments that were identified as being potentially significant in the SARA study.

9.3.2 Detailed Fire Analysis

Differences in the results of the detailed analysis are anticipated due to changes in modelling assumptions and data input. The principal differences are subdivided into those which would tend to increase the predicted fire risk relative to the SARA study, and those which would have the opposite effect.

Increase in predicted fire risk

(1) The SARA study took credit for 3 hour fire rated wrap material but accounted for the probability of failure due to improper installation. The IPEEE study took no credit for fire wrap material (except fire compartment 2).

(2) The fire frequencies derived in the IPEEE study are based on a more recent and comprehensive fire event database than that developed specifically for the SARA study. Consequently, the fire frequencies used in the IPEEE are higher than those used in the SARA study.

Decrease in predicted fire risk

(1) The IPEEE analysis was able to utilize a cable management database which permitted specific cable trays and conduits to be identified as potential targets. In the LGS SARA study, it was conservatively assumed that, in the event of damage to any cable raceway, the entire electrical division associated with that raceway would be disabled. This is the principle reason why fixed fire sources, located in fire compartments such as 44, 45 and 47, were insignificant in the IPEEE study but were relatively significant in the SARA study.

(2) The IPEEE study was able to take credit for new combustible control procedures which prohibit the storage of transient combustibles in specific compartments (e.g. the cable spreading room). In other compartments (e.g. safeguards access area), their storage is restricted to areas well away from potential targets. The IPEEE contribution to risk from transient combustible fires predicted in the IPEEE, is therefore less than that evaluated in the SARA study.

(3) Self ignited cable fires were a major source considered as part of the SARA study. However, consistent with the FIVE methodology, such fires are not considered credible in the IPEEE study on the basis that all cable at the LGS station is IEEE.

9.4 Other Event Comparison

Both reports addressed in detail the following events: external flooding, tornadoes, toxic vapors, missiles generated by events near the site and aircraft hazards. Both the IPEEE and the SARA concluded that none of these events are a significant risk contributor.

9.5 Summary

The LGS SARA study and the LGS IPEEE both reveal LGS's long standing ability to withstand external events.

10.0 REFERENCES, ABBREVIATIONS AND ACRONYMS

10.1 REFERENCES

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10.2 ACRONYMS AND ABBREVIATIONS

ACI	American Concrete Institute
ADS	Automatic Depressurization System
ANS	American Nuclear Society
ANSI	American National Standard Institute
AOV	Air Operated Valve
ARCo	Atlantic Richfield Company
ASCE	American Society of Civil Engineers
ASDC	Alternate Shutdown Cooling
ASME	American Society of Mechanical Engineers
ATS	Automatic Transfer Switch
ATWS	Anticipated Transient Without Scram
BOP	Balance of Plant
BTU	British Thermal Unit
BWR(OG)	Boiling Water Reactor (Owners Group)
CCW	Component Cooling Water
CDF	Core Damage Frequency
CDFM	Conservative Deterministic Failure Margin
CFZ	Combustible Free Zone
CRD	Control Rod Drive
CRL	Component Record List
CS	Core Spray
CST	Condensate Storage Tank
DBE	Design Basis Earthquake
DBFL	Design Basis Flood Level
DBT	Design Basis Tornado
DG	Diesel Generator
DHR	Decay Heat Removal
DW	Drywell

ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EPG	Emergency Procedure Guideline
EOP	Emergency Operating Procedure
EPRI	Electrical Power Research Institute
EPS	Electric Power System
ESF	Engineered Safety Feature
ESW	Emergency Service Water
ET	Event Tree
FCR	Field Change Request
FIVE	Fire Induced Vulnerability Evaluation
FPER	Fire Protection Evaluation Report
FSAR	Final Safety Analysis Report
FT	Fault Tree
FW	Feedwater
GE	General Electric
GERS	Generic Equipment Ruggedness Spectra
GL	Generic Letter
HCLPF	High Confidence of Low Probability or Failure
HCU	Hydraulic Control Unit
HELB	High Energy Line Break
HPCI	High Pressure Coolant Injection
HRA	Human Reliability Analysis
HRR	Heat Release Rate
HVAC	Heating Ventilating and Air Conditioning
I&C	Instrumentation & Controls
IEEE	Institute of Electrical & Electronics Engineers, Inc.
ILRT	Integrated Leak Rate Test
INDMS	Integrated Nuclear Data Management System

INEL	Idaho National Engineering Laboratory
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination of External Events
ISDS	Ignition Source Data Sheets
LCO	Limiting Condition of Operation
LGS	Limerick Generating Station
LLNL	Lawrence Livermore National Laboratory
LOCA	Loss of Coolent Accident
LOOP	Loss of Offsite Power
LPCI	Low Pressure Coolant Injection
LPRM	Low Pressure Radiation Monitor
LV SWGR	Low Voltage Switch Gear
LWR	Light Water Reactor
MCC	Motor Control Center
MCR	Main Control Room
MOV	Motor Operator Valve
MS	Main Steam
MSIV	Main Steam Isolation Valve
MWe	MegaWatt electric
MWt	MegaWatt thermal
NFPA	National Fire Protection Association
NEMA	National Electrical Manufacturer's Association
NOAA	National Oceanic and Atmospheric Administration
NPRDS	Nuclear Plant Reliability Data System
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NSSFC	National Severe Storms Forecast Center
NSSS	Nuclear Steam Supply System
OBE	Operating Basis Earthquake

P&ID	Piping & Instrument Diagram
PCIG	Primary Containment Instrument Gas
PECO	PECO Energy Company
PIMS	Plant Information Management System
PMF	Probable Maximum Flood
PMP	Probable Maximum Precipitation
PPE	Personal Protective Equipment
PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment
PSAR	Preliminary Safety Analysis Report
PSI	Pounds per Square Inch
PWR	Pressurized Water Reactor
RCIC	Reactor Core Isolation Cooling
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RECW	Reactor Enclosure Cooling Water
RG	Regulatory Guide
RHR(SW)	Residual Heat Removal (Service Water)
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RSP	Remote Shutdown Panel
RWCU	Reactor Water Cleanup
RWST	Refuel Water Storage Tank
SARA	Severe Accident Risk Assessment
SBLOCA	Small Break Loss of Coolant Accident
SBO	Station Blackout
SCS	Soil Conservation Survey
SDC	Shutdown Cooling
SET	Systems Evaluation Team

SEWS	Screening & Evaluation Worksheet
SFPE	Society of Fire Protection Engineers
SGTS	Standby Gas Treatment System
SLCS	Standby Liquid Control System
SMA	Seismic Margins Analysis
SME	Seismic Margins Earthquake
SMM	Seismic Margins Method
SNL	Sandia National Laboratory
SPC	Suppression Pool Cooling
SPCL	Success Path Component List
SOD	Sphere of Damage
SPF	Standard Projected Flood
SPI	Suppression Pool Indication
SPLD	Success Path Logic Diagram
SQRT	Seismic Qualification Review Team
SQUG	Seismic Qualification Users Group
SRP	Standard Review Plan
SRSS	Square Root of Sum of Squares
SRT	Seismic Review Team
SRV	Safety Relief Valve
SSD	Safe Shutdown
SSE	Safe Shutdown Earthquake
SSRAP	Senior Seismic Review and Advisory Panel
SWGR	Switchgear
TAF	Top of Active Fuel
T/G	Turbine Generator
TRIP	Transient Response Implementation Procedure
TRS	Test Response Spectra
UBC	Uniform Building Code

UFSAR	Updated Final Safety Analysis Report
UHS	Ultimate Heat Sink
USGS	United States Geological Survey
USI	Unresolved Safety Issue
WW	Wetwell