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V. S. BOYER SR. VICE PRESIDENT NUCLEAR POWER

October 10, 1980 Docket Nos. 50-352 50-353

Mr. Robert L. Tedesco, Asst. Director U. S. Nuclear Regulatory Commission Division of Licensing Office of Nuclear Reactor Regulation Wash. Jton, DC 20555

Subject: Preliminary Risk Assessment--Limerick Generating Station

Dear Mr. Tedesco:

As requested at our meeting of October 2, 1980, attached is an outline of the final report on the Limerick Generating Station preliminary risk assessment. As is evident by this outline, the final report will be quite comprehensive and, based on time allowance for internal review, will be available for submittal to the Commission by April, 1981.

We will deliver to you by the end of November, 1980, supporting information being used in the analysis. This information will include:

> All fault trees All event trees General Electric and Bechtel plant drawings used in the analysis Assumptions Ground rules Success criteria Limerick transmission system description Loss of offsite power data Limerick specifications and differences compared to the WASH 1400 BWR Concrete analysis Core radiological content Containment specifications and data Limerick population data Peach Bottom diesel data Peach Bottom HPCI/RCIC data

This information will provide the background detail being used in the analysis in addition to the five years of meteorology data sent to you on October 7, 1980.

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Sincerely, J. Boye

LIMERICK GENERATING STATION PRELIMINARY PROBABILISTIC RISK ASSESSMENT OUTLINE OF FINAL REPORT

SUMMARY

- 1.0 INTRODUCTION
 - 1.1 Background
 - 1.2 Methodology and Report Structure
 - 1.3 Relationship of this Report to WASH-1400
 - Overview of Criticism of WASH-1400
 Methods Used to Address the WASH-1400 Criticism
 - 1.4 Limitations of the Methodology
 - 1.5 Groundrules and Assumptions

2.0 LIMERICK PLANT DESCRIPTION

- 2.1 Overview Description
- 2.2 Safety Features (Accident Mitigators)
- 2.3 Comparison of Limerick and the WASH-1400 BWR
- 2.4 Meteorology, Population and Site-Specific Considerations
- 3.0 ACCIDENT ANALYSES: STATE-OF-THE-ART LIMERICK CALCULATION
 - 3.1 Sources and Mobility of Radioactivity
 - 3.2 Initiators that Could Result in Local Release
 - 3.3 Release Sequence Classes
 - 3.4 Accident Sequence Event Trees Including Dominant Sequences
 - 3.5 Probabilities Associated with Event Tree Nodes
 - 3.6 Radioactive Release Fractions Associated with Dominant Accident Sequences (INCOR)
 - 3.7 Consequences Associated with Accident (CRAC Early Fatalities)
 - 3.8 Dominant Event Sequences
 - 3.9 Uncertainties
- 4.0 COMPARISON WITH WASH 1400
 - 4.1 WASH 1400 BWR at the Limerick Site

4.2 Limerick Plant at Limerick Site

| 4.2.1 | Design Differences |
|-------|-------------------------|
| 4.2.2 | Site Differences |
| 4.2.3 | Data Differences |
| 4.2.4 | Methodology Differences |

5.0 SUMMARY AND OBSERVATIONS

6.0 REFERENCES

APPENDIX A: PROBABILITY DATA BASE

- A.1 Accident Initiators
- A.2 Equipment and Component Failure Rate and Repair Rate Data and Treatment
- A.3 Human Error and Failure Rate Data
- A.4 Maintenance Procedures, Technical Specification and Operating Procedures
- A.5 Diesel Dependency Model and Data
- A.6 Loss of Offsite Power

APPENDIX B: SYSTEM DESCRIPTIONS AND FAULT TREE LOGIC MODELS

B.1 High Pressure Coolant Systems

| B.1.1 | HPCI |
|-------|-----------|
| B.1.2 | RCIC |
| B.1.3 | CRD |
| B.1.4 | Feedwater |

B.2 Low Pressure Coolant Systems and Pressure Reduction System

| B.2.1 | ADS |
|-------|------------|
| B.2.2 | LPCI |
| B.2.3 | LPCS |
| B.2.4 | Condensate |

B.3 Heat Removal

| B.3.1 | RHR Service Water |
|-------|--------------------|
| B.3.2 | Condenser |
| B.3.3 | Ultimate Heat Sink |

- B.4 Containment
- B.5 Electric Power, Instrumentatic, and Control
- B.6 Service Water -- HVAC Pump doom Cooling
- B.7 Reactor Protection System
- B.8 Standby Liquid Control System

- 2 -

B.9 Generic Component Fault Tree Models

- 3 -

B.10 Functional Level Fault Tree

B.11 Sequence Fault Trees

APPENDIX C: INCOR

- C.1 Introduction
- C.2 Code Description

| C.2.1 | CONTEMPT-LT |
|-------|-------------|
| C.2.2 | BOIL |
| C3 | PVMELT |
| C.2.4 | INTER |

C.3 IACOR Models (Sequences)

| C.3.1 | Core MeltRPV FailureConcrete Meltthrough |
|-------|--|
| | before Containment Failure |
| C.3.2 | Containment Failure Precedes Core MeltRPV |
| | FailureConcrete Meltthrough |
| C.3.3 | Anticipated Transient Without Scram-Core Melt Before |
| | Containment Failure |
| C.3.4 | Anticipated Transient Without Scram-Containment |
| | Failure Before Core Melt |

APPENDIX D: IN-PLANT RADIOACTIVITY TRANSPORT

- D.1 Review of Literature
- D.2 CORRAL Calculation
- D.3 Summary of Release Amounts by Class and Mitigation System Failure

APPENDIX E: EX-PLANT CONSEQUENCE ANALYSIS

- E.1 Demography, Meteorology, and Other Site-Specific Considerations
- E.2 CRAC Calculations
- E.3 Evacuation Model
- APPENDIX F: DISTRIBUTIONS OF UNCERTAINTIES
- APPENDIX G: DISCUSSION OF MEAN VALUES
- APPENDIX H: REVIEW OF SELECTED CONTAINMENT PHENOMENA
- APPENDIX I: LIMITATIONS OF PROBABILISTIC RISK ASSESSMENT