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SECRETARY'S OFFICE
Public Utility Commission

PENNSYLVANIA PUBLIC UTILITY COMMISSION
v.
PHILADELPHIA ELECTRIC COMPANY,
DOCKET NO. R-850152

LIMERICK 1 AND COMMON PLANT EXPLANATION OF REASONS FOR COST AND SCHEDULE GROWTH, INCLUDING DATA ON THE EFFECT OF NEW AND REVISED REGULATORY REQUIREMENTS

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September 27, 1985

I. INTRODUCTION

This Exhibit has been prepared to explain the reasons for cost and schedule growth of Limerick 1 and Common Plant. The cost analyses supporting Section II were developed jointly by TB&A and PECO. TB&A, however, provided overall direction for this analysis. The analysis presented in Section III was prepared by PECO. Both analyses and the specific contributions made by the individual organizations identified are described in the testimony of PECO witnesses Love, Kemper, Clarey, Helwig, and Sproat.

As demonstrated in Section II of this Exhibit, a primary cause of cost and schedule growth of Limerick 1 and Common Plant was the issuance of new and revised NRC regulatory requirements in the form of new regulations, regulatory guides, standard review plans, generic letters, NUREG documents and I&E bulletins and information notices, etc. The extent and timing of these new NRC requirements and the manner of their application to Limerick is described in the testimony of Dr. Roger J. Mattson, a former senior NRC official.

Section III provides additional data with respect to the impact of these new requirements upon the Project. Section III also presents an analysis of the dates upon which a number of specific, new NRC requirements were imposed on the project and the time periods necessary for completion of the required engineering and construction. Additionally, data is presented with respect to specific plant additions necesary for compliance with these requirements and the relationship of commodity and manhour growth to changes in regulatory requirements. The data presented in Section III is further discussed in the testimony of witnesses Boyer, Clarey, Helwig, and Sproat.

In addition to regulatory impacts, further causes of cost and schedule growth are shown to be funding constraints upon the project, labor unavailability, and less than anticipated labor effectiveness. Data relative to these causes are also set forth in both Section II and Section III of the Exhibit and are discussed in the testimony of witnesses Boyer, Kemper, Paquette, Clarey, and Love.

II. EXPLANATION OF REASONS FOR COST GROWTH

A. Introduction

This explanation is presented by dividing cost growth into the following categories:

- Cost Impact of Regulatory and Other Externally-Imposed Conditions
- Cost Impact of Design Changes to Facilitate Plant Operability and Reliability
- Cost Increases Due to Estimate Refinements and Other Causes
- Cost Impact of Unanticipated Escalation

A summary of the costs determined to be associated with each category is presented in Schedule 1.

B. Cost Impact of Regulatory and Other Externally-Imposed Changes

1. TMI-2 ANALYSES AND DESIGN CHANGES $57.5 Million

The Three Mile Island, Unit 2 (TMI-2) incident had a significant impact on light water reactor (LWR) construction and operations cost. Following TMI-2, various panels within the NRC, such as the Lessons Learned Task Force

Schedule 1

LIMERICK UNIT 1 & COMMON
COST GROWTH RECONCILIATION

		($ in Millions)
Original PECO Capital Authorization Unit 1 and Common, Direct Costs		$344.1
Regulatory and other Externally - Imposed Conditions		
- TMI-2 Analyses and Design Changes	57.5	
- Plant Staffing, Startup and Training	181.6	
- Seismicity	119.6	
- Impact of Mark II	136.1	
- Fire Protection & Electrical Separation	65.8	
- Equipment Qualification	38.6	
- Anticipated Transients Without Scram	20.9	
- ALARA & OSHA	39.6	
- ASME Code Requirements	20.9	
- Security Requirements	43.5	
- Intergranular Stress Corrosion Cracking	10.2	
- Licensing Costs	62.9	
- Miscellaneous Other NRC Regulations	116.3	
- Non-NRC Requirements	63.8	
- Cost Impact of Schedule Delays Due to Licensing Delays and Other Factors	385.1	
		1362.4
Design Changes To Facilitate Operability and Reliability		111.9
Estimate Refinements and Other Causes		208.6
Unanticipated Escalation		322.1
DIRECT COST RECONCILIATION TOTAL		$2349.1

and Emergency Preparedness Task Force, reviewed the incident and issued findings and recommendations in the form of NUREG reports. These requirements were for the most part detailed in NUREGs -0578, -0654, -0660, -0696, -0700, -0737, revisions to Regulatory Guide 1.97, amendment of 10CFR50 Appendix E, and Supplement 1 to NUREG -0737 which were issued during the period July 1979 to May 1983.

Examples of Limerick design changes as a result of these requirements are as follows:

- Post-Accident Sampling System (PASS) added in accordance with NUREG-0737, Item II.B.3;
- Accident monitoring equipment upgraded and new equipment added per NUREG-0737, Item II.F.1 and Regulatory Guide 1.97, Rev. 2;
- Automatic closing of drywell purge isolation valves on a high radiation signal added in response to NUREG-0737, Item II.E.4.2(7);
- A control room design review (CRDR) was performed and resultant changes implemented as per NUREG-0737, Item I.D.1, as further clarified by Supplement 1 to NUREG-0737;
- Technical Support Center (TSC) added in accordance with NUREG-0737, Item III.A.1.2;
- Computer and software for offsite dose assessment added in compliance with NUREG-0737, Item III.A.2.2, clarified by NUREG-0737, Supplement 1;
- Added emergency microwave telecommunications system between PECO main office, plant, and Emergency Operations Facility (EOF) in accordance with NUREG-0654, Item E.1;
- Relocation of vent radiation monitoring equipment to provide post-accident access per NUREG-0737, Items II.F.1 and II.B.1;
- System leakage study performed and modifications implemented as per NUREG-0737, Item III.D.1.1;
- Emergency Response Facility Data System (ERFDS) added in

compliance with NUREG-0737, Item I.D.2 to provide safety parameter display system (SPDS), clarified by Supplement 1 to NUREG-0737;

- Modifications to the high pressure coolant injection system and reactor core isolation cooling system (HPCI/RCIC) in accordance with NUREG-0737, Items II.K.3.15 and II.K.3.22;

- Emergency public notification system added per 10CFR50, Appendix E;

- Modifications to the Automatic Depressurization System (ADS) logic per NUREG-0737, Item II.K.3.18;

- Added Seismic Class I air supply system to the ADS valves per NUREG-0737, Item II.K.3.28;

- Purchase and installation of a Safety Relief Valve Position Indication System per NUREG-0737, Item II.D.3;

- Isolation signal reset interlocks added to safety feature systems per NUREG-0737, Item II.E.4.2;

- Operations Support Center added per NUREG-0737, Item III.A.1.2;

- Development and implementation of emergency procedures per NUREG-0737, Item I.C.1, as further clarified by Supplement 1 to NUREG-0737.

The effects of the TMI-2 incident upon project cost and schedule exceed actual hardware and engineering analyses accounted for in this subsection. Substantial cost additions set forth in Subsection 2 of this testimony, reflecting increases in plant staffing, startup and training, are also attributable to new requirements or revised interpretations of existing requirements adopted because of TMI-2. Further, one of the most significant indirect impacts of TMI-2 was a heightened level of regulatory awareness and scrutiny, which significantly impacted project cost.

2. PLANT STAFFING, STARTUP, AND TRAINING $181.6 Million

Project costs increased substantially relative to the original December 1970 estimate due to increased requirements in the areas of staffing, startup, and training. For instance, startup test procedure preparation and review process requirements are far more extensive than those in effect at the time of the original cost estimate. Hence, manpower requirements were considerably greater than originally estimated. Additionally, requirements for operations personnel, operating procedures, and operator training have greatly expanded since the TMI-2 incident.

The following are examples of changes in staffing, startup, and training due to increased requirements:

- Expanded startup and power ascension testing programs required in accordance with Regulatory Guide 1.68, Revision 2, Revision of Standard Review Plan Section 14 and Regulatory Guide 1.108;
- Independent Safety Engineering Group required per NUREG-0737, Item I.B.1.2;
- Fire Protection assistant added to the plant operations staff in response to SRP Section 9.5.1 and BTP CMEB 9.5-1;
- Shift Technical Advisors added per NUREG-0737, Item I.A.1.1;
- Reactor operator training and qualifications upgraded per NUREG-0737, Item I.A.2.1;
- Shift staffing, overtime, and turnover practices revised per NUREG-0737, Items I.A.1.3. and I.C.2;
- Operations Experience Assessment Committee established in response to NUREG-0737, Item I.C.5;
- Increased training for the mitigation of core damage events provided per NUREG-0737, Item II.B.4;

- Staffing increases needed to handle expanded NRC requirements for documentation and reporting.

- Increased size of administration complex based on additional staffing requirements.

- Increased number of procedures required due to NRC expansion of the scope of plant Technical Specifications.

3. <u>SEISMICITY</u> **$119.6 Million**

Major requirements for seismic design classification impacting Limerick are detailed in Regulatory Guide 1.29. This guide, adopted in June 1972 and revised as late as 1978, describes an acceptable method of identifying and classifying those features which should be designed to allow the plant to be safely shutdown in the event of an earthquake. Compliance with Regulatory Guide 1.29 was extremely costly in that numerous supports for piping, electrical equipment, and HVAC were impacted in terms of requirements for additional engineering analysis, and labor manhours required for the installation and modification of supports and hangers.

The AEC/NRC dictated several significant changes to the Limerick seismic design basis after the Preliminary Safety Analysis Report (PSAR) was docketed in February of 1970. Many of these changes were due to NRC and ACRS concerns about the ability of nuclear power plants to withstand seismic events, concurrent with keeping safety-related systems and components functional. These changes, all of which defined more conservative seismic loads to be used for the design of Limerick, included:

- Vertical direction seismic analysis for all safety-related structures;

- Twenty-five (25) percent increase in maximum ground acceleration for the design basis earthquake;

- Three earthquake component combinations for design of piping systems per requirements of Regulatory Guide 1.92.

These criteria changes resulted in structural design changes, while the increase in seismic response had a major impact in terms of costs on plant systems, components and equipment which necessitated increases in strength, number and size of components and supports.

Examples of design analyses and field changes made at Limerick in response to the seismic design requirement changes include:

- Upgrading of HVAC ductwork (relative to Peach Bottom) to Seismic Class I in the reactor enclosure and control structure per PSAR commitment and Regulatory Guide 1.29;

- Increases in structural steel in the reactor enclosure to satisfy seismic criteria;

- Increased metal conduit and cable tray installation cost to allow for Seismic Class I supports due to increased seismic response spectra and ground acceleration value;

- Increased installation manhours for all conduit, instrument tray, piping and tubing supports due to increased congestion and complexity;

- Control room ceilings upgraded from Seismic Class II to Special Class IIA in compliance with Regulatory Guide 1.29;

- Hanger and snubber quantity increases resulting from decreased hanger spacing due to revised seismic loads;

- Changes in unit cost and fabrication of large pipe and hanger material due to reassessment of dynamic loads.

These new or revised seismic design requirements were imposed during the period June 1972 to September 1978.

4. IMPACT OF MARK II $136.1 Million

A new category of load phenomena associated with BWR Mark II containments was identified in 1975. This was comprised of the hydrodynamic loading conditions during a safety relief valve (SRV) discharge or loss-of-coolant accident (LOCA). These hydrodynamic SRV and LOCA loads have come to be known simply as Mark II loads.

Various requirements have been issued by the NRC concerning acceptable means of addressing these loads by owners of BWR Mark IIs. These include NUREGs -0487, -0763, -0783, -0802, and -0808. As a result of these requirements, a number of analyses and design changes were made at Limerick. Examples of these include:

- Engineering to evaluate the suppression pool swell phenomenon, perform containment analysis, provide technical support to the BWR Mark II Owner's Group, and conduct plant unique analysis and support design. These analyses included:
  - fatigue evaluation of downcomer and SRV lines;
  - analysis of wetwell and drywell vacuum breakers;
  - analysis of shroud support legs;
  - asymmetric SRV acceleration time histories and response spectra analysis.
- Modifications to seismic structures adjacent to the containment to account for hydrodynamic loadings;
- Field modifications and installation costs associated with the re-evaluation of design loading requirements for pipe supports and piping systems, including the addition of hangers and other materials such as snubbers and struts.

These Mark II-related NRC requirements were issued during the period 1978 to 1982.

5. FIRE PROTECTION AND ELECTRICAL SEPARATION

$65.8 Million

Following the Browns Ferry Nuclear Power Plant fire in 1975, the NRC made a comprehensive evaluation of the design criteria for fire protection at nuclear power plants. The major regulatory requirement that resulted from this review was 10CFR50.48. Appendix R to 10CFR50, although not directly applicable to Limerick by regulation, was the basis for Standard Review Plan Section 9.5.1 and Branch Technical Position (BTP) CMEB 9.5-1, against which the Limerick design was reviewed. The requirements contained in this document are intended to ensure that, in the event of a fire, personnel and plant equipment would be adequate to safely shutdown the reactor, and to maintain the plant in a safe shutdown condition.

Electrical separation requirements in the form of Regulatory Guide 1.75 also impacted the Limerick design by requiring the redesign and rework of raceway and control systems to achieve physical independence of safety-related circuits and electrical equipment.

The following are examples of the regulatory impacts stemming from BTP CMEB 9.5-1 (fire protection), and Regulatory Guide 1.75 (electrical separation):

BTP CMEB 9.5-1

- Fire protection changes and modifications pertaining to the separation of safe shutdown circuits, namely: fireproofing of cable tray, added cable tray supports, additional sprinkler systems, modifications to existing fire protection systems, increased structural steel fireproofing, penetration sealing and fire protection of wall gaps, and relocation of equipment from hazardous areas;

- Addition of automatically initiated, permanently installed Halon 1301 fire suppression system on the PGCC.

Regulatory Guide 1.75

- Raceway separation tests were conducted, and other electrical separation design changes incorporated;
- PGCC, NSSS, and balance of plant panel modifications due to separation requirements;
- Q-listed equipment inspection and NRC reviews to verify conformance with Class 1E equipment internal separation requirements.

These increased fire protection and electrical separation requirements were adopted during the period 1975 to 1981 with major modifications spanning from 1982 to 1984.

6. ENVIRONMENTAL AND DYNAMIC EQUIPMENT QUALIFICATION — $38.6 Million

Limerick safety-related equipment, located inside as well as outside primary containment, is required to be capable of maintaining its functional operability for its installed life under all service conditions postulated to occur during normal operation and as a result of design basis events. Equipment qualification is the documented demonstration of this capability.

Major regulatory requirements affecting environmental and dynamic equipment qualification costs at Limerick include NUREG-0588, Regulatory Guide 1.100 and 10CFR50.49. The latter regulation specified the requirements to be met to demonstrate the environmental qualification of electrical equipment important to safety located in a potentially harsh environment, and expanded the scope of qualification and documentation required for safety-related equipment.

Costs due to environmental and dynamic equipment qualification on Limerick include:

- Qualification and/or modification, replacement, or relocation of equipment to meet environmental conditions in compliance with NUREG-0588 and 10CFR50.49;

- Plant design changes associated with replacement or relocation of equipment due to qualification;

- Evaluations to determine reactor building environmental conditions under various conditions;

- Qualification and/or modification, replacement or relocation of equipment to meet dynamic conditions in compliance with Regulatory Guide 1.100 including analyses of equipment for Mark II dynamic loads and seismic qualification (SQRT) reviews.

These requirements were imposed, during the period 1980 to 1983.

7. ANTICIPATED TRANSIENTS WITHOUT SCRAM — $20.9 Million

An Anticipated Transient Without Scram (ATWS) occurs when the control rods fail to insert as a result of an undefined common mode failure when a reactor scram signal is initiated.

In December 1978, Volume 3 of NUREG-0460, "Anticipated Transients Without Scram for Light-Water Reactors" was issued describing the proposed plant modifications the NRC staff considered necessary to reduce the risk from ATWS to an acceptable level. In response to the NRC's concern with the ATWS issue, the following features were included in the Limerick design:

- Lower main steam line isolation valve isolation setpoint;

- High pressure coolant injection system flow split;

- Redundant reactivity control system for:

  (1) Alternate rod insertion
  (2) Recirculation pump trip
  (3) Automatic initiation of standby liquid control with improved injection flow path;

- Additional uninterruptible power supply.

8. AS LOW AS REASONABLY ACHIEVABLE (ALARA) AND OSHA $39.6 Million

The NRC requires that all nuclear power plants be designed so that radiation protection measures are taken to ensure that internal and external dosages to plant personnel and contractors as a result of plant conditions, will be within the limits established by 10CFR20. This regulation addresses and details the requirements for:

- Determination of accumulated doses;
- Individual exposures to concentrations of radioactive materials in the air in restricted areas;
- Radioactivity in effluents to unrestricted areas;
- Surveys of radiation hazards;
- Personnel monitoring;
- Caution signs, labels, signals, and controls;
- Waste disposal;
- Records, reports, and notification;
- Enforcement.

Regulatory Guide 8.8 provides more detailed guidance relevant to attaining goals and objectives for planning, designing, constructing, operating, and decommissioning a light-water reactor nuclear power plant. The goals of both Regulatory Guide 8.8 and 10CFR20 are to keep the annual integrated (collective) dose to station personnel as low as is reasonably achievable (ALARA).

Modifications to the Limerick Station design as a result of these requirements include:

- Reviews, identification, purchase, and installation of radiation shielding associated with potential radiological exposure areas;

- Modifications to the breathing air system to prevent radioactive contamination;
- Special coatings in radiation areas to facilitate decontamination and cleanup;
- Relocation of instruments to low radiation areas.

Compliance with OSHA requirements for worker safety also impacted station costs. Examples of required compliance measures include:

- Eye protection for craftsmen;
- Increased lighting fixture quantities;
- Added room signs, identification signs, directional and elevation signs;
- Additional personnel access platforms.

9. ASME CODE REQUIREMENTS $20.9 Million

ASME Code requirements increased substantially during the design, procurement, and construction of the Limerick project. At the time of initial project design and cost estimates, the ASME Code Sections pertaining to Nuclear Power Plant Components were in the developmental stage and thus, their full impact could not be accurately anticipated. Examples of areas where the evolution of applicable ASME code requirements increased project costs are as follows:

- Preparation of nuclear piping design specifications;
- Additional weld preparation for non-destructive examination;
- Additional non-destructive examination;
- Additional stress analyses of piping and hangers;
- Preservice inspection.

In addition to these costs, ASME code changes caused substantial increases in the cost of equipment, field fabrication and documentation which could not be specifically accounted for in the project cost records. These costs are absorbed in other entries included in the subsections on Estimate Refinements and Other Cost Increases.

10. SECURITY REQUIREMENTS $43.5 Million

The original scope of Limerick security provisions consisted of standard industrial property fence and general site access control. With the promulgation of 10CFR73.55 in February 1977, however, significant changes in physical protection were required at all nuclear facilities. The current Limerick security system consists of an integrated computer-based system interfacing with extensive access control and alarming hardware, perimeter monitoring and alarming, TV surveillance, and personnel processing facilities.

The required extensive security system increased the costs of engineering, construction, startup and testing, and carries a constant maintenance requirement. Further, associated security personnel required to support the system prior to commercial operation have increased substantially over original project estimates.

11. INTER-GRANULAR STRESS CORROSION CRACKING $10.2 Million

In response to BWR operating experience with inter-granular stress corrosion cracking (IGSCC), susceptible materials in the Limerick primary coolant and ECCS piping systems were replaced with IGSCC resistant materials during the construction phase as opposed to a more costly replacement of the materials when the plant is operational. Subsequent regulatory requirements for dealing with this problem contained in NUREG-0313 and NUREG-1061, were satisfied by this action.

12. LICENSING COSTS — $62.9 Million

Changes and additional evaluations which were required during the Limerick licensing process contributed towards additional project cost increases. Examples of these necessitated changes are:

- Use of a new format and context requirement for preparation of the FSAR;
- Additional geological investigations;
- Completion of a control systems interaction study in response to questions regarding USI A-49;
- Preparation and defense of a Probabilistic Risk Analysis (PRA);
- Completion of an Independent Design Verification Program (IDVP);
- Hardening of the reactor enclosure to withstand aircraft impact;
- Missile protection of the refueling floor;
- Increased cost of snubber exercising per I&E Bulletin 81-01.

13. MISCELLANEOUS OTHER NRC REQUIREMENTS — $116.3 Million

A large number of plant design changes were required due to changes in NRC requirements which do not have a central theme or purpose, such as those itemized in separate sections above. Examples of such changes are indicated below:

- Increased monitoring of effluents per 10CFR50, Appendix I;

- Modifications to facilitate containment leak testing per 10CFR50, Appendix J;
- Addition of an ultimate heat sink complying with Regulatory Guide 1.27;
- Expanded interpretation of quality assurance program requirements per 10CFR50, Appendix B;
- Concrete anchor bolt test program per I&E Bulletin 79-02;
- Additional diesel generators provided per Regulatory Guide 1.81;
- Addition of an MSIV leakage collection system per Regulatory Guide 1.96;
- Inclusion of an augmented offgas system per 10CFR50.34a;
- Improvement of leakage detection capability per Regulatory Guide 1.45;
- Use of high density spent fuel storage racks (because reprocessing is unavailable);
- Masonry wall modifications per I&E Bulletin 80-11;
- Addition of a loose parts monitoring system in compliance with Regulatory Guide 1.133;
- Analysis of high and moderate energy line breaks and subsequent plant changes per BTP ASB 3-1, and BTP MEB 3-1;
- Design changes for control room habitability per Regulatory Guide 1.78;
- Addition of radwaste solidification equipment per 10CFR61.

14. <u>NON-NRC REQUIREMENTS</u> $63.8 Million

Requirements imposed by entities other than the NRC caused increases in project costs. Revised or new requirements from the EPA, PaDER, and ANI in areas such as effluent limitations, erosion control, flood plain management, and worker safety all contributed to project cost increases. In addition, various unanticipated increases in taxes and other items were imposed upon the project.

15. COST IMPACT OF SCHEDULE EXTENSIONS DUE TO LICENSING DELAYS AND OTHER FACTORS $385.1 Million

Delays in the construction and operating schedules resulted in time related direct cost increases due to the prolonged period of equipment maintenance and project staffing. Schedule extensions were experienced as a result of:

- Delays in receipt of a Construction Permit from the NRC;
- Cash flow and financing constraints;
- Limited availability of craft manpower;
- Delays in receipt of the low power operating license from the NRC;
- The cumulative effects of regulatory change on project engineering and construction;
- Delays in receipt of the full power operating license from the NRC.

C. Cost Impact Of Design Changes To Facilitate Operability And Reliability $111.9 Million

The three basic sources of non-regulatory-imposed design and work scope changes on the Limerick Project were those initiated by the architect/engineer/constructor, vendors, and PECO. Many changes in this cost causation category were identified as being the result of operating experiences derived from PECO's Peach Bottom BWR Units in addition to general industry operating experience. The objective in making these changes was to enhance plant safety, reliability, and operability. Examples of such design and work scope changes are:

- Change from hydraulic to mechanical snubbers;

- Design modifications to small pipe vents and drains;

- Revised turbine drain design to prevent water induction into the turbine based on a GE recommendation as a result of industry operating experience;

- Addition of protective coating for the entire suppression pool;

- Rerouting of plant heating pipe, addition of steam traps and condensate return units;

- Modifications to reactor feed pump seals, main condenser internals, loop seals, vents and drains, HVAC systems and turbine drains;

- Addition of an equipment vibration monitoring system;

- Utilization of three additional safety relief valves in lieu of safety valves.

D. Cost Increases Due To Estimate Refinement And Other Causes — $208.6 Million

Costs included in this section represent estimate adjustments and re-evaluations, pricing changes, and other subcontract changes not wholly attributable to specific causation factors. Certain of these cost changes resulted, albeit indirectly, from changing regulatory requirements and other uncontrollable factors described elsewhere.

Examples of cost changes associated with this category are:

- Addition of stainless steel liner for reactor well due to estimate omission;

- Re-evaluation of the concrete batch plant support operations, including additional requirements for concrete trucks;

- Reduced weather protection costs based on fewer winter placements;

- Credit for pre-commercial power generation;
- Estimate adjustments based on actual contract awards.

E. Cost Impact Of Unanticipated Escalation — **$322.1 Million**

Costs included within this category are changes in the determination of escalation due to unanticipated inflation made throughout the project, unanticipated escalation associated with schedule extensions, and unanticipated changes in labor rates.

III. EFFECT OF NEW OR CHANGED NRC REQUIREMENTS UPON THE PROJECT

A. Summary of Analysis

An analysis was performed by PECO to evaluate the time durations necessary to identify, design and construct specific plant additions and modifications required by new or revised NRC requirements. The specific NRC requirements addressed in this analysis are those concerning anticipated transients without scram (ATWS); control room panel modifications; Mark II modifications; equipment qualification; TMI-2 additions and modifications such as a) emergency response facilities, b) safety parameter display system, c) radiological meteorological monitoring system, d) post-accident sampling system and e) SRV position indication; seismicity, fire protection; high and moderate energy line breaks; security; "as-built" pipe and pipe hanger reviews; masonry walls; and ASME Section III requirements.

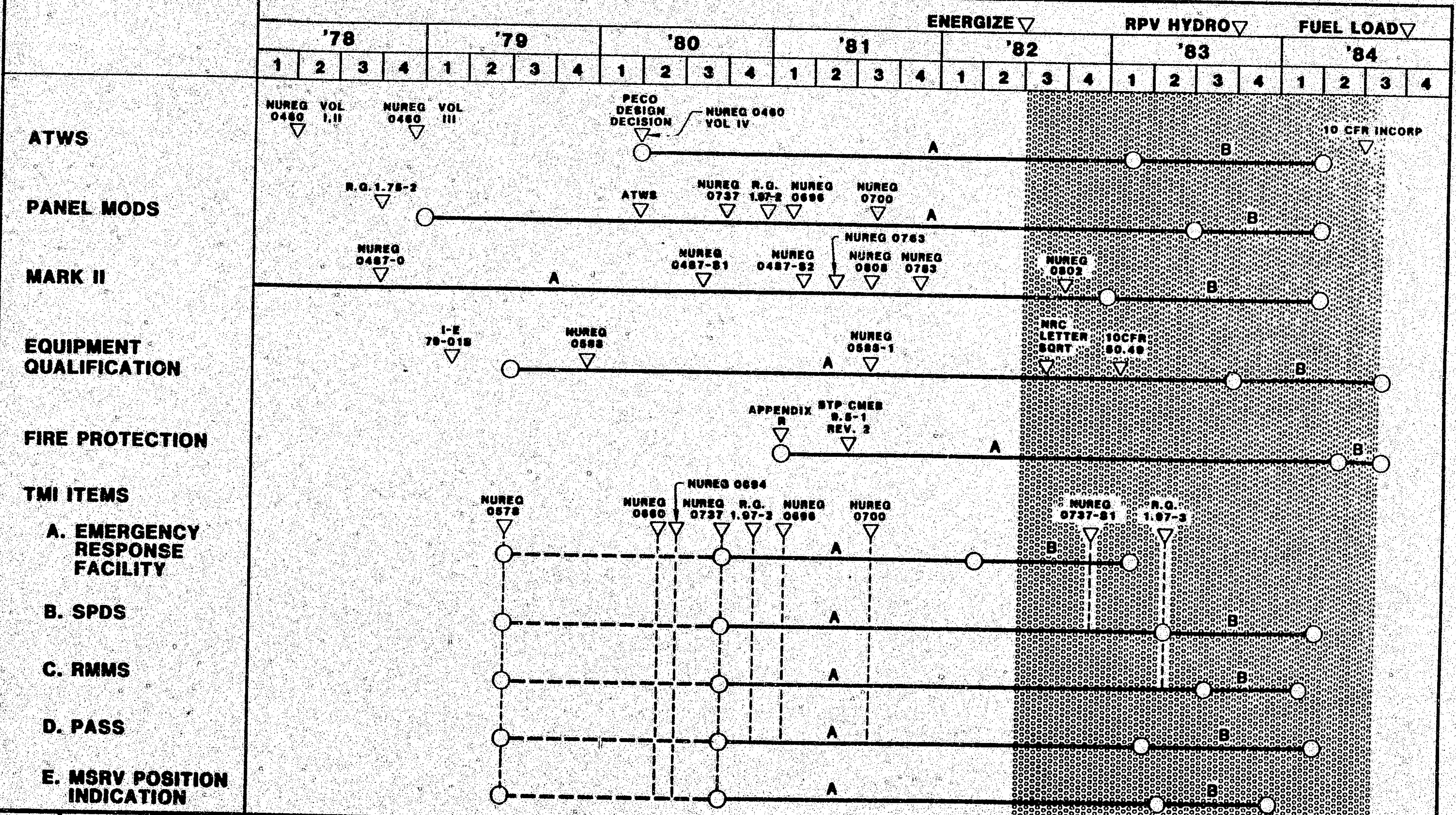
The analysis identifies when major regulatory requirements affecting the project were imposed by the NRC, and when engineering and construction activities required as a result of these regulatory changes were completed in relation to the overall project schedule. The results of this analysis are shown in Schedule 2.

The information necessary for the completion of this analysis was obtained from a variety of sources. Information regarding the timing of specific regulatory requirements was obtained through reviews of project documentation and discussions with the cognizant PECO engineers. Durations for engineering activities were obtained through review of applicable engineering schedules supplemented by discussions with cognizant project personnel. Finally, construction activity durations were developed from a review of construction schedules and dialogues with cognizant site personnel. Time periods required for engineering are illustrated on Schedule 2 by line "A", while the durations necessary for procurement and construction are designated as line "B".

The analysis demonstrates the impact of the volatile regulatory environment on the completion of project construction and startup activities. As such, it demonstrates that a large number of regulatory changes occurred between 1979 and 1984, necessitating that engineering and construction activities extend well into the startup and preoperational test period. Because of the timing of these mandated changes, engineering manhours were frequently expended for the redesign of systems and equipment as well as for the design of new systems not originally anticipated. Hence, to minimize the impact on overall project completion, a "fast track" approach was required, i.e. construction activities were initiated as soon as a sufficient portion of the design was complete. This resulted in the removal and modification of previously installed items, the installation of additional materials and equipment in already congested plant areas due to the advanced stage of construction, and delays in overall construction completion and

SCHEDULE 2

SCHEDULE EFFECT OF REGULATORY CHANGES



LEGEND: A - DESIGN PERIOD; B - MAT'L DEL & CONST COMPLETION PERIOD; - - - PRELIM. ENGR.; STARTUP TESTING; ▽ - ISSUE DATE

DESCRIPTION	REV	PLAN	SCHED	APPR	DATE
EXHIBIT 2 - SCHED 2	0				8/26/81

PROJECT: LIMERICK 1

TITLE: SCHEDULE EFFECT OF REGULATORY CHANGES

JOB NO.

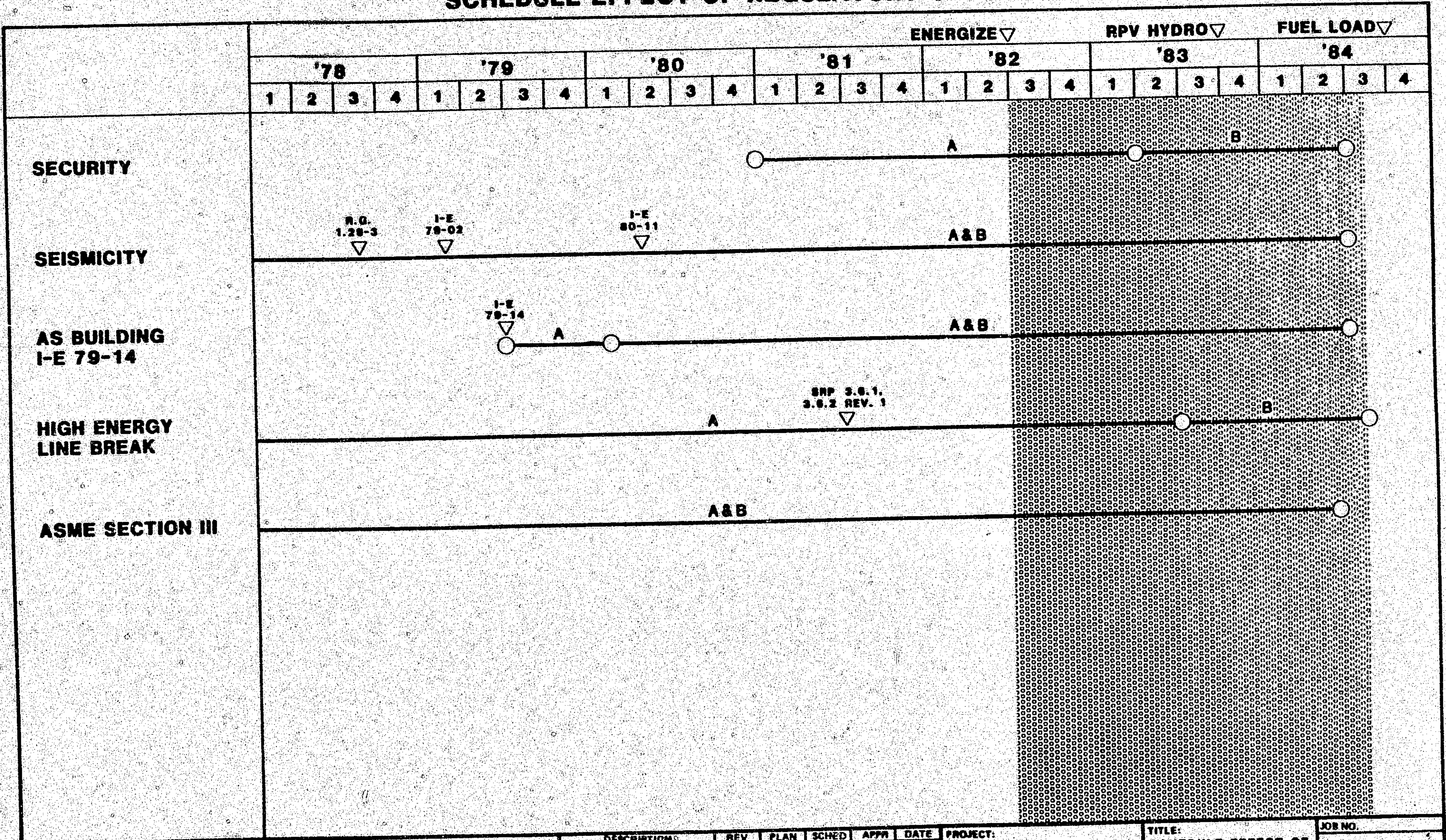
SCHED NO. 2

SHEET 1 OF 2

8 7 6 5 4 3 2 1

SCHEDULE 2

SCHEDULE EFFECT OF REGULATORY CHANGES



LEGEND: A - DESIGN PERIOD
B - MAT'L DEL & CONST COMPLETION PERIOD
- PRELIM. ENGR.
- STARTUP TESTING
▽ - ISSUE DATE

DESCRIPTION	REV	PLAN	SCHED	APPR	DATE
EXHIBIT 2 - SCHED 2	0				6/26/80

PROJECT: LIMERICK 1

TITLE: SCHEDULE EFFECT OF REGULATORY CHANGES

JOB NO.
SCHED NO. 2
SHEET 2 OF 2

8 7 6 5 4 3 2 1

turnover of systems to the startup group. The cumulative level of effort required for the resolution of these items prohibited acceleration of construction completion and startup testing activities much beyond that which was achieved. Further, the late imposition of many of these regulatory mandated activities caused less than optimal construction sequencing which adversely affected unit rates.

B. Details of Analysis

This subsection provides additional data with respect to the effect upon Limerick 1 and Common Plant of the new or revised regulatory requirements identified on Schedule 2. These specific requirements and systems are discussed in detail to provide explanations of their impact on the project.

1. Analysis of Effects of Specific TMI Regulatory Requirements

Following the incident at TMI-2, the activities of various NRC task forces resulted in rapidly changing requirements in a number of different plant areas. For example, on September 13, 1979, the NRC Staff issued recommendations concerning shift technical advisors, shift supervisor responsibilities, operating procedures, reactor coolant system vents, plant shielding, post-accident surveys, relief and safety valve testing and position indication, auxiliary feedwater system control and instrumentation, power for pressurizer heaters, hydrogen recombiner penetrations, accident monitoring instrumentation, instrumentation for the detection of inadequate core cooling, emergency support facilities, and in-plant radiation monitoring.

The various task force recommendations were codified in May 1980 as NUREG-0660, "TMI-2 Action Plan". The stated objective of the Action Plan was to provide a comprehensive and integrated plan for the actions judged necessary by the NRC Staff to resolve safety concerns raised by the TMI-2 incident. Some of these requirements were imposed in June 1980, in NUREG-0694, "TMI-Related Requirements for New Operating Licenses." Following NRC review of these matters, NUREG-0737, "Clarification of TMI Action Plan Requirements" was issued on October 31, 1980. NUREG-0737 incorporated some of the Staff recommendations contained in NUREG-0660 and 0694. It effectively superseded NUREG-0694 and was the vehicle for implementing the licensing requirements contained in NUREG-0660. In addition, the NRC Staff was reorganized to add a new division of human factors safety, and an emergency preparedness program office was established in the Office of Inspection and Enforcement with related licensing responsibilities transferred to that group.

NUREG-0737 is applicable to both operating plants and plants under construction and includes information about schedules, applicability, methods of implementation review, submittal dates and clarification of technical positions. While a large number of other NRC documents provide guidelines, clarifications, and implementation details for post-TMI requirements, NUREG-0737 provides the basic action items. Nevertheless, compliance with these basic action items required that the applicant or licensee review the guidelines of the many additional NRC documents.

The NRC's objective in requiring these changes was to reduce the frequency of events which might present challenges to plant safety systems and to ensure proper operator reactions to such challenges when they do occur. Since the TMI incident was complicated by shortcomings in operator performance, the NRC sought to minimize the potential for human error by improving emergency procedures, training, staffing and the ability of plant operators to understand and diagnose plant conditions.

The first supplement to NUREG-0737 was issued in a generic letter dated December 17, 1982, entitled "Requirements for Emergency Response Capability". NUREG-0737, Supplement 1 proposed requirements for additional instrumentation, communication equipment, design reviews and emergency operating procedure improvements. Supplement 1 also proposed new requirements for the Safety Parameter Display System and Emergency Response Facilities.

Additional requirements were imposed through revisions of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident" (Revision 2, December 1980). Regulatory Guide 1.97 required that qualified instrumentation be provided for monitoring numerous selected plant conditions (i.e., pressure, temperature, radiation) and required the installation of accident monitoring instrumentation that was supplied with Class 1E power, environmentally and seismic qualified, and completely separate from those instruments used under normal operating conditions.

The following is a brief description of the major design changes resulting from TMI-related requirements and their impact on the Limerick Project.

(a) <u>Emergency Response Facilities</u>. The incident at TMI led to studies which identified the need for improvements in the response of operations management to accidents at nuclear power plants including, in particular, the upgrade and/or establishment of dedicated emergency response facilities. NUREG-0737, NUREG-0696 and 10CFR50 Appendix E, as amended, specify the requirements for these emergency facilities.

Numerous design upgrades and plant modifications were implemented as a result of these requirements including installation of an operational support center near the control room from which plant personnel can be assigned to support emergency operations, and construction of a dedicated, fully-equipped Technical Support Center (TSC) to further improve accident management. Due to space limitations in the plant, establishing a TSC required construction of a new facility. NRC requirements also specified that other permanent plant equipment be located in the TSC building, such as the related Emergency Response Facility Data System (ERFDS) and Radiation and Meteorological Monitoring System (RMMS), as well as an inverter to power these systems.

(b) <u>Emergency Response Facility Data System (ERFDS)</u>. ERFDS is a computer-based system which provides the Safety Parameter Display System (SPDS) in the Control Room required by NUREG-0737, Supplement 1. Its function is to collect, transmit and display all of the Regulatory Guide 1.97 parameters in the Control Room, TSC and EOF per NUREG-0696. Additionally, it collects and records transient data from process systems and piping system startup tests as required by Regulatory Guide 1.68.

The implementation of ERFDS required the installation of two additional data acquisition computers in the TSC, two CRT's in the Control Room and one CRT in both the TSC and the EOF, plus electrical equipment for gathering data from over 700 locations in the plant. The system required approximately 3000 feet of raceway to run the approximately 90,000 feet of electrical and fiber optic cables to transmit data from the data collection points to the computers and CRT's. Fifty control room panel modifications involving a large number of wiring changes were required to connect to other systems.

(c) Radiation and Meteorological Monitoring System (RMMS). The RMMS is a computer-based dose assessment system which provides the capability for near real-time estimates of atmospheric dispersion and offsite dose consequences resulting from airborne radioactive releases from the station under emergency conditions. The requirements for this system were imposed by NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," NUREG-0737, and Regulatory Guide 1.97.

The RMMS consists of a computer in the TSC which gathers data from the station radiation monitoring system and meteorological monitoring equipment installed on Weather Towers 1 and 2. Using this data, the computer plots travel patterns of airborne radioactivity releases on the CRTs in the Control Room, TSC and EOF and calculates resulting offsite doses. Implementation of this system required approximately 38,000 feet of cable to be installed. This system was turned over for startup testing in March 1984 and would not have supported an earlier startup schedule because of the time required for development and testing of the system.

(d) Post-Accident Sampling System (PASS). Due to concerns pertaining to the ability to obtain and analyze representative liquid and gaseous samples following an accident, requirements for a post-accident sampling system were set forth in NUREG-0737, Item II.B.3 and a number of subsequent NRC clarifications.

In response to these requirements and concerns PECO organized a number of utilities operating BWR's to develop a generic, cost-effective approach to post-accident sampling. The generic design concept developed by the owners group called for a sampling and analysis system that would provide all necessary post-accident information while minimizing equipment complexity, total installed cost, space and maintenance requirements. A "system level" design specification was prepared and presented to the NRC Staff by General Electric and PECO to obtain early Staff concurrence with the design approach. Following NRC Staff acceptance, the PASS design was completed and a prototype was fabricated and tested within a period of five months. Each utility participating in the owners' group subsequently contracted with GE for equipment fabrication.

The PASS system at Limerick consists of numerous sampling taps into existing process pipelines in the Reactor Enclosure and routing of sample lines to a collection station in the Control Enclosure. Installation of this system required over 1400 feet of tubing and 700 feet of cable. The PASS system was completed and turned over to the startup organization in March 1984.

(e) <u>Safety Relief Valve Position Indication</u>. The position indication for the Main Safety Relief Valves (MSRV) was another design change required after the incident at TMI and was imposed in NUREG-0737. This design change involved the installation of acoustical monitors downstream of the MSRV's to detect flow and provide reliable indication of valve position. The acoustical monitors are part of an instrument loop which transmits valve position to indicators in the Control Room. This modification required installation of over 6000 feet of cable and approximately 500 feet of conduit.

2. Fire Protection

Throughout the early 1970s, fire protection was largely left by the NRC to the insurers whose primary interest was property loss prevention. During this time period, the only regulatory requirements concerning fire protection in a nuclear facility were contained in GDC-3 in Appendix A to 10CFR50, which stated:

> "Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistent materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Fire-fighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components."

On March 22, 1975, a fire occurred at the Tennessee Valley Authority's Browns Ferry Station and, as a result, fire protection for nuclear power plants became a major regulatory issue. In August 1976, more than a year after the Browns Ferry fire, the NRC issued Appendix A of Branch Technical Position (BTP) APCSB 9.5-1, providing detailed guidance for fire protection for nuclear plants under construction prior to July 1, 1976. In March 1978, Revision 1 to BTP APCSB 9.5-1 was issued providing additional guidance. In February 1981, almost five years after the BTP was originally issued, the NRC issued 10CFR50.48 and 10CFRPart 50, Appendix R. Appendix R by its terms is applicable to nuclear power plants licensed to operate prior to January 1, 1979. In July 1981, the BTP was again revised and reissued as BTP CMEB 9.5-1. This was a major revision to the BTP and imposed many of the requirements of Appendix R on plants then under construction. As a result, more rigorous requirements had to be implemented at Limerick, especially in the areas of fire brigade training and administrative controls, safe shutdown capability (i.e. fire protection of safe shutdown components) and cable penetration seal qualification. Engineering for these changes was not complete until mid-1984 while construction of additions and modifications was not complete until September 1984.

The following major changes were made at Limerick in the early 1980s in response to the new or revised NRC fire protection requirements.

(a) Raceway Encapsulation. To meet the safe shutdown requirements of Appendix R, over 1200 feet of raceway and over 1500 feet of conduit had to be encapsulated with a fire barrier material. Also, because of the added weight of the fire barrier, all the supports of the encapsulated raceway had to be re-analyzed, which required a significant engineering effort. In some cases, the supports had to be strengthened, thus requiring additional cost.

Another impact of these installations was the effect on asbestos worker availability. Raceway encapsulation was installed from November 1983 to September 1984, during the same period when other major activities, such as pipe insulation and penetration sealing, also required significant numbers of asbestos workers. The demand for asbestos workers from these activities applied tremendous pressure on the local union to supply manpower. The union was required to recruit workers from other union locals, as well as supply temporary work cards to non-asbestos workers. This necessitated increased supervisory and manual labor costs and resulted in higher unit installation rates.

(b) Sprinkler Additions. Additional sprinkler systems required at Limerick included four water curtains and seven sprinkler systems. Each of these systems was necessary to comply with requirements to separate safe shutdown area fire zones. The additional sprinkler systems were comprised of piping, seismically qualified hangers, spray heads, control valves, detection and alarm systems, conduit and wiring. The difficulties associated with the addition of these sprinkler systems included: 1) all other commodities were previously installed causing these systems to be fabricated at the site to avoid interferences; 2) installation of the new systems required the erection of large scaffolding structures, core drilling, and penetration sealing in conflict with the facility turnover schedule, special coating activities, and clean-up effort; and 3) functional testing of these systems had to be coordinated with the fire protection startup testing schedule which was already compressed due to other design changes and plant priorities.

(c) <u>Structural Steel Coating</u>. Another requirement was the determination that all structural steel forming or supporting safe shutdown fire barriers must have protection equivalent to that required of the barrier (this is generally interpreted as 3-hours). PECO initiated an extensive program to identify those steel members that actually required fire protection. This included a modeling and calculational program that was developed to reduce the amount of protection modifications which had to be performed. This latter program, which was the first of its kind to receive NRC approval, ultimately resulted in the installation of additional sprinklers and fireproofing of a reduced number of steel members.

The steel members identified as requiring fire protection by the modeling and calculational analysis had to be further evaluated to determine the best method for providing fire protection. Coating of structural steel with a fireproofing material was the most widely used method, and required in many instances a labor-intensive method of application, namely hand-troweling. Troweling the fire-proofing material onto the steel was required due to the congested environment that existed this late in the construction schedule. The coating effort also involved the erection of large scaffolding structures to gain access to the overhead steel requiring coating. The use of scaffold was in conflict with the facility turnover schedule and associated clean-up effort, and thus had a delaying effect on these programs.

Additionally, three sprinkler systems were added to protect structural steel in those areas where coating was determined to be impractical. These additions required the installation of sprinkler piping, new detection systems, spray heads, control valves, remote control panels, alarm systems and related conduit and wire for each fire zone. The installation of conduit and pipe also required core boring walls and sealing the penetrations once the commodities were installed. The difficulties associated with the addition of these sprinkler systems included installation problems due to area congestion and tight clearances, and correct spray head placement. The installations and functional testing of these systems occurred during the startup phase of the project, and required additional coordination and integration with the overall startup program.

(d) Penetration Sealing Program. Much of this program was the result of fire protection requirements. All penetrations through barriers separating the various fire zones had to be sealed. Internal conduit in fire barriers required a 3-hour rated fire seal rather than the smoke and hot gas seal originally planned. As a result, approximately 8,000 seals had to be upgraded. Additionally, certain safe shutdown areas had to be reviewed for the potential of flooding and their seals upgraded for water integrity. This program required that pipe movement evaluations be performed which caused the costly and time consuming addition of link seals and boots to some piping and sleeves. Difficulty was encountered in the installation of seal materials due to the close proximity of pipe supports to the penetrations. The design of the penetration seals was further complicated by ALARA, HELB, and MELB design requirements.

(e) <u>Halon Fire Suppression System</u>. The auxiliary equipment room contains pre-fabricated raised floor sections with mounted relay panels (i.e. floor modules), termination cabinets at the north and south ends of the floor modules, and a site-fabricated raised floor to the north, south and west of the termination cabinets with additional floor modules installed on the west side area. Since GE had conducted tests which showed that a fire in the floor modules would self-extinguish due to oxygen starvation, the original project design called for only Class B heat and smoke detectors in the floor modules. However, in 1978, the NRC required that a Halon fire suppression system also be installed in their evaluation of NEDO-10466, Rev. 2, "Power Generation Control Complex Design Criteria and Safety Evaluation." As a result of fire protection requirements for safe shutdown capability, the remote shutdown panels had to be separated from the remainder of the auxiliary equipment room by 3-hour rated fire walls. This necessitated a redesign of both the fire detection and Halon systems to create a separate zone for the new room.

Further, during the late 1970s, GE and the NRC continued development of a general licensing document for the auxiliary equipment room floor module and termination cabinet fire detection systems. The final document, NEDO-10466A, required a Class A detection system in the floor modules and termination cabinets. The NRC required Limerick to upgrade its system to the final document. GE issued the design for the addition of smoke detectors to each termination cabinet and to upgrade the detection system in November 1983. This and the above described additional work required modification of the systems which had previously been installed and required partial removal of the floor module walking surface, thus disrupting various preoperational tests in progress at the relay panels and restricting access of startup testing personnel. The combined effect of these changes was to delay Halon system preoperational testing until late summer 1984.

3. Control Room Panel Modifications

The control room and auxiliary equipment room contain numerous control panels which provide information on the status of plant systems and permit operators to manually control those systems. The control room is the nerve center of the plant, both during operations and startup testing, with the auxiliary equipment room containing the control logic for all plant systems. Beginning in late 1978, the NRC Staff issued a number of publications which required modifications to either the information displays or the control instruments on these panels. The number of requirements increased substantially following the TMI-2 incident. Major regulatory changes requiring panel modifications include: 1) Regulatory Guide 1.75, "Physical Independence of Electric Systems", Rev. 2, issued in September 1978; 2) ATWS requirements described later; 3) the human factors review of control rooms required by NUREG-0737; 4) Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and

Following an Accident", Rev. 2, issued in December 1980; 5) NUREG-0696, "Functional Criteria for Emergency Response Facilities", issued July 1980; and 6) Supplement 1 to NUREG-0737 issued December 1982. The required modifications included the addition of new switches, indicators, lights, relays and associated wiring, and the physical separation of wiring and components of different electrical divisions.

Due to the timing of these new requirements evaluation and design of the necessary changes continued through mid-1983, with installation at the site continuing through April 1984 as shown on Schedule 2. Installation of these modifications significantly disrupted the construction sequence and delayed startup testing. Indeed, in a number of instances, modifications had to be made to previously tested panels, thus requiring retesting and greater administrative control of the modifications.

4. Mark II Modifications

With the identification of previously undefined Mark II hydrodynamic loads in 1975, an extensive industry-wide effort was initiated to resolve concerns raised over this issue. PECO took a lead role in this effort through the use of consultants and participation in industry groups. Due to the complexity of the problem, the engineering design process required approximately seven years before final load criteria became available. These final criteria were the most precise and least conservative design criteria which would resolve the Mark II concerns.

During the engineering design process, a number of publications were issued by the NRC which reflected the evolutionary process of the Mark II problem resolution. In October 1978, bounding load design criteria for LOCA related loads was issued in NUREG-0487, "Mark II Containment Lead Plant Program - Load Evaluation and Acceptance Criteria, Generic Technical Activity A-8". These criteria were revised and finalized through a supplement to NUREG-0487 in August 1980. Long-term program criteria related to LOCA loads were published in August 1981 in NUREG-0808, "Mark II Containment Program Load Evaluation and Acceptance Criteria (A-8)", in November 1981 in NUREG-0783, "Suppression Pool Temperature Limits for BWR Containments (A-39)", and in October 1982 in NUREG-0802, "Safety Relief Valve Quencher Loads: Evaluation for BWR Mark II and III Containments (A-39)". Mark II modifications added approximately 750 new hangers and over 1400 modifications to existing hangers. The construction effort was increased due to requirements for additional scaffolding, remobilization of crafts, additional manual manhours required for the new and revised hangers, and increased non-manual manhours required for additional layout, work packages, and inspections. In the suppression pool, structural modifications were required for the downcomer bracing system to accommodate the Mark II loads.

5. Equipment Qualification

Equipment qualification refers to the process of demonstrating, either through testing or by analysis, that equipment will operate satisfactorily in the plant environment to which it may be exposed after a design basis earthquake or postulated accident such as a loss of coolant accident. The primary licensing requirements for equipment qualification in nuclear facilities are contained in General Design Criterion (GDC) 4 of Appendix A to 10 CFR Part 50, adopted in

1971. For many years, utilizing equipment of the highest industrial quality analyzed for anticipated service conditions was considered to satisfy GDC 4. Since the mid-1970s, however, licensing requirements for equipment qualification have changed and expanded considerably to require more vigorous analysis and/or testing of all safety-related equipment. PECO was actively involved with the generation of the industry qualification standards in the late 1960s and 1970s, and recognized the need to impose qualification requirements on Limerick purchase orders early in the project.

In 1977, NRC research program test results cast doubt on the validity of the methods being relied upon by the nuclear industry to establish the environmental qualification of electrical equipment. In response to this discovery, the NRC Staff began to develop extensive, detailed requirements in the area of equipment qualification. This resulted in the issuance of NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment", in late 1979. On May 23, 1980, the NRC issued a Memorandum and Order which expressed dissatisfaction with the industry's efforts in the environmental qualification area, and ordered its Staff to use NUREG-0588 in licensing reviews of plants. The Commission also at this time initiated a rulemaking on environmental qualification of safety-grade electrical equipment. This rulemaking eventually resulted in the issuance of 10CFR50.49 in January 1983 and the revision of Regulatory Guide 1.89, "Environmental Qualification of Certain Electrical Equipment Important to Safety for Nuclear Power Plants" (Rev. 1) in June 1984. In a parallel effort, the NRC also issued Regulatory Guide 1.100, "Seismic Qualification of Electric Equipment for Nuclear Power Plants," Rev. 1, in August 1977, which provided revised criteria for the determination of acceptable seismic qualifications.

The overall trend of these changing NRC requirements has been to require equipment qualification by testing rather than by analysis wherever possible, to require more conservative definitions of possible post-accident environmental conditions, to exactly prescribe testing sequences and methodologies to be used, and to insist on extremely detailed and thorough record-keeping.

PECO instituted a program to review all existing environmental qualification documentation and identified all deficiencies against the new criteria. When deficiencies were found, programs for their correction and resolution were formulated and carried out. PECO also participated in an industry effort, headed by the Electric Power Research Institute (EPRI), to develop a data base of equipment that had been qualified by various utilities and testing companies. Using this information, PECO could determine whether a specific piece of equipment had previously been qualified by others, thereby minimizing the cost and time needed to qualify the particular piece of equipment. As a result of the evolving regulatory requirements, the number of items to be qualified, and the evolving state-of-the-art, engineering for equipment qualification requirements was not completed until September 1983 as shown on Schedule 2.

The major field impact of environmental qualification requirements was equipment replacements and modifications. Environmentally qualified replacements included over 75 instruments associated with various valves, heaters and panels for standby gas treatment filters, diaphragm replacements on numerous valves, and various HPCI and RCIC electrical components. The dynamic qualification modifications associated with Regulatory Guide 1.100 included the addition of structural bracing on HVAC systems serving areas containing safety-related equipment, filter units, and equipment in the standby gas treatment system, control room emergency fresh air intake, reactor building recirculation filters, drywell coolers, reactor building fan cabinets and in-duct radiation monitors. Modifications were also required to stiffen and brace individual pieces of equipment such as the HPCI and RCIC turbines, suppression pool suction strainers, spent fuel pool gates, air accumulator tanks, instrumentation support racks, electrical motor control centers, etc. Also, in the case of the primary containment vacuum relief valves and MSRV discharge line vacuum relief valves, internal parts had to be replaced with new parts.

In addition to the cost impact of these modifications and replacements, the changes were required during the system turnover period when administrative controls made work more difficult. For example, because testing was underway, detailed reviews for safety blocks and equipment tagging were necessary to ensure the safety of the installation crews. Stringent cleanliness precautions were also instituted, and tighter controls over modifications and replacements were employed to ensure that appropriate retesting was performed and previous testing was not invalidated. These safety and quality restrictions led to added non-manual manhours in addition to the added manual manhours (i.e. increased unit rates) because labor effectiveness was lower due to plant congestion.

6. <u>Anticipated Transients Without Scram (ATWS)</u>

An ATWS occurs when the control rods, i.e., rods which absorb the neutrons which produce the nuclear chain reaction and are used to terminate that reaction, fail to insert into the reactor. In 1978, after a lengthy evaluation, the NRC Staff issued NUREG-0460, Volumes I, II and III. This publication outlined alternative hardware systems and operating procedures which could be employed to prevent or mitigate the consequences of an ATWS. At this time, however, the NRC Staff did not select a particular alternative as its preferred approach. Rather, the debate over the proper equipment systems and/or procedures to be employed to resolve the ATWS concern continued through the first quarter of 1980 when the NRC Staff issued NUREG-0460, Volume IV. In that publication, the Staff adopted a series of both hardware changes and operating procedures as its preferred approach in resolving ATWS. The NRC utimately confirmed the requirements applicable to Limerick in a final, generic rulemaking on ATWS in June 1984.

Throughout the period of its evaluation by the NRC, PECO remained informed of the nature of the ATWS concern and of the alternative solutions being proposed to mitigate that concern. In March 1980, PECO initiated the design effort to resolve ATWS concerns at Limerick. This program involved monthly multi-discipline design and schedule control meetings with both GE and Bechtel. Detailed design was performed by GE and Bechtel, with the participation of PECO engineers. All related activities remained under direct PECO control. By late 1981, the conceptual design of the needed systems was complete and by early 1983, as shown in Schedule 2, all detailed design drawings were finished. The system fully complied with the requirements imposed by the NRC in NUREG-0460, Volume IV.

The system developed for Limerick to respond to the NRC's ATWS concern is called the Redundant Reactivity Control System (RRCS). The RRCS includes instrumentation which monitors reactor vessel pressure and water level, neutron flux and other parameters. This data is fed to microprocessors located in the auxiliary equipment room. If the instrumentation shows that a situation exists where the reactor should be shutdown, but the neutron flux is still high, the RRCS logic will cause the control rods to be inserted. At the same time, a timer is started which will automatically initiate the Standby Liquid Control System which injects boron into the reactor if the control rods have not been inserted. This system also initiates a reduction of feedwater flow, isolates Reactor Water Cleanup and trips the Recirculation Pumps.

Construction of this system required the installation of over 35,000 feet of cable and approximately 5000 feet of conduit as well as numerous control panel modifications to connect the RRCS to other systems. Two additional sets of 4 KV switchgear were also installed in the Reactor Enclosure and a third pump and associated piping were added to increase the capacity and reliability of the Standby Liquid Control System. The necessity to make the control panel modifications, in particular, delayed turnover for startup testing of the reactor water cleanup and the standby liquid control systems. This out-of-sequence installation activity disrupted and increased the cost of startup testing while also adversely affecting the unit rates for cable and conduit installation.

Installation of the RRCS, as shown in Schedule 2, was completed in April 1984, and, given the timing of the imposed NRC requirements and equipment delivery times, could not have been completed to support a significantly earlier fuel load date than that actually achieved.

7. Seismicity

General Design Criteria 2 in Appendix A to 10CFR50 and 10CFR100, Appendix A (adopted in 1973) set forth the principal seismic and geologic standards which guide the evaluation of the suitability of the seismic design bases for a nuclear power plant. Seismic design criteria affect most quantities of material used in the construction of the plant, such as structural steel, rebar, concrete and piping. Further, the increased size and complexity of individual components and the difficulty encountered in commodity installation requires the expenditure of additional manhours. Seismic design also affects the amount of engineering manhours necessary to complete the final design due to added complexity and the need for additional analyses.

At the request of the AEC and the Advisory Committee on Reactor Safeguards (ACRS) in 1971, the Company committed to more conservative seismic design criteria than had been committed to in the initial PSAR for Limerick, docketed on February 27, 1970. In December 1974, the NRC issued Regulatory Guide 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis", which required assurance that non-safety-related items would not affect the safe operation of safety-related items following a seismic event. This Regulatory Guide was revised to include new requirements in February 1976. Finally, additional requirements were imposed by I&E Bulletin 79-02, "Pipe Support Base Plate Designs Using Concrete Expansion Bolts" in March 1979, and by I&E Bulletin 80-11, "Masonry Wall Design" in May 1980 (i.e. described below).

The increase in seismic design criteria and the hardening of the outer walls to mitigate the effects of potential missile impact combined to significantly contribute to increased Limerick construction cost and complexity. The actual quantity of reinforcing steel in Seismic Category I structures was approximately 350 pounds per cubic yard versus 250 pounds per cubic yard budgeted in the original project cost estimate (a 38% increase). This reinforcing resulted in increased complexity of reinforcing steel erection and cadweld splicing, form tie installation, embedment placement, preplacement cleanup, and concrete placement, all of which adversely affected unit rates.

In addition, Regulatory Guide 1.29, adopted in June 1972, was revised three times as new requirements were issued with the last revision occurring in September 1978. It required that all non-safety-related equipment and commodities whose failure could affect safety-related equipment be installed to seismic standards. Seismic installations require not only additional supports but also larger, stiffer and more complicated supports than normal non-seismic installations. In general, seismic supports are items fabricated from structural steel shapes (i.e. structural tubing, wide flange, angles, plate and bar stock) which require extensive cutting, fitting and welding in the field. Non-seismic pipe hangers are usually assembled by threading and bolting standard hanger components purchased from a vendor. Seismic supports for large pipe required 2.6 times more labor hours than for non-seismic supports, while seismic small pipe installations required 2.7 times more labor hours than non-seismic installations, adversely affecting both total labor manhours and unit rates. Unit rates for other commodities (i.e. light fixtures, conduit, etc.) were similarly impacted.

Regulatory Guide 1.29 also imposed additional Quality Assurance requirements which increased Field Engineering and Quality Control inspections. As a result of these requirements, increased costs were experienced in identifying the increased scope, maintaining the scope as the design was evolving, preparation of an inspection program, inspections, maintenance and collation of the inspection date.

8. <u>High Energy Line Break (HELB)/Moderate Energy Line Break (MELB).</u>

HELB's and MELB's are postulated accidents used in the design and safety reviews of nuclear power plants. The piping in a nuclear power plant is classified as high or moderate energy piping, based on the temperature and pressure of the fluid contained in the piping. A failure of high energy piping would result in the release of high temperature or pressure water or steam, and depending on the way the piping is assumed to fail, might also possibly result in violent motion of the pipe (i.e., pipe whip) and the generation of missiles. In reality the probability of a break occurring in nuclear power plant piping is quite small because of the conservative design practices, the high quality of fabrication and materials used, and the frequent inspection of such piping carried out during the plant's operating life.

General Design Criterion 4 in Appendix A to 10 CFR 50, adopted in 1971, provided the initial design requirements for the protection against the effects of HELB's and MELB's. In November 1975, the NRC issued Standard Review Plan (SRP) Sections 3.6.1, "Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," and 3.6.2, "Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping" to provide a basis for HELB/MELB design.

Following a postulated pipe break, a high energy line is predicted to whip about violently like a fire hose due to the force of the escaping fluid. Pipe whip restraints are large structural steel components that limit the movement of the pipe and protect safe shutdown components. The design effort for such pipe whip restraints is significant, involving calculations of whip force, structural loads, and evaluation of as-built configurations. Construction of the restraints requires significant effort since the restraints must be large to withstand the predicted loads. Additionally, since the piping cannot be held rigidly due to normal expansion and contraction during operation, the gaps between the piping and the restraints must be small to reduce the whip forces to manageable levels. A high degree of accuracy and thus additional time (i.e. increased unit rates) are required for installation.

Jet impingement loads are also predicted to occur as a result of the high velocity fluid escaping from the HELB. Protection against jet impingement is provided by specific shields designed to withstand the jet force or by locating vulnerable components away from the postulated water or steam jets.

These HELB/MELB requirements resulted in the addition of 400 tons of pipe whip restraint throughout the primary containment building, reactor building and turbine building. This required additional embeds in concrete walls, heavier building steel and modifications to previously installed steel to accommodate the loads on the restraints. This program also required steel plate shims and energy absorbing components to be installed between the pipes and the restraints to provide the proper spacing to absorb the pipe whip loads for the postulated break. The shimming process required extensive measurements and templating of the affected areas, machining of the steel shims and installation of the shims to rigorous tolerances. These latter steps were new and extensive and added considerable manual and non-manual manhours to the project.

HELB/MELB considerations also necessitated the inclusion of steam release paths to the atmosphere to prevent compartment overpressurization in the event of a break. This required additional concrete walls, penetration seals and blowout panels.

9. As-Built Piping Reconciliation - I&E Bulletin 79-14.

In 1979, the NRC Staff identified a number of operating plants with installations of piping and supports which did not agree with design drawings. The NRC Office of Inspection and Enforcement issued I&E Bulletin 79-14 "Seismic Analysis For As-Built Safety-Related Piping Systems" on July 18, 1979. This Bulletin required that licensees inspect their facilities to ensure that the actual piping system configurations conformed to design drawings. Plans for an "as-built" reconciliation program were in existence at Limerick at the time, but had to be substantially modified and strengthened to comply with I&E Bulletin 79-14.

The initial project plans included inspections by field engineers, inspection and acceptance by quality control personnel, and a final walkdown by the stress engineers who designed the piping systems. The final program, which reflected the requirements of I&E Bulletin 79-14, was far more extensive, including the inspection of individual hanger components by field engineers and quality control engineers, and the assembly of installation drawings which corresponded to an engineering calculation. Designers then performed additional stress analyses for the as-installed configuration and issued Stress Reconciliation Notices (SRN) to the field including revised installation drawings incorporting the field changes on which the stress reanalysis was based. Final field verification was required before the SRN was completed. The as-built process was required prior to hydrostatic testing and resulted in delays in system turnovers to startup.

These efforts also resulted in significant increases in required engineering and non-manual manhours.

10. Security Requirements.

The security system originally designed for Limerick consisted of fencing around the site and standard access controls. However, in February 1977, the NRC issued 10 CFR 73.55, "Requirement for Physical Protection of Licensed Activities in Nuclear Power Reactors Against Radiological Sabotage", which substantially increased the security requirements at nuclear power plants. Engineering for these security changes was completed in early 1983 and the final system was completely installed by mid-1984.

The security system at Limerick consists of approximately 50 microwave zones, 50 closed circuit TV (CCTV) cameras, 250 door monitors with the majority being controlled, and numerous barriers monitored with breakwires and shaker sensors. The monitoring and control information is fed through seven multiplexers to a redundant computer system located in the Secondary Alarm Station (SAS). The system is monitored in the Central Alarm Station (CAS) and in the SAS. The guard force in the CAS and SAS also monitor the TV cameras. A total of approximately 56,000 feet of embedded conduit, 20,000 feet of exposed conduit, 1300 feet of wireway, 150 feet of cable tray, and 350,000 feet of cable were required to connect the monitoring and control elements to the computers.

Two guard stations were also required in the Technical Support Center and Administration Building for personnel access. The guard stations include explosive and metal detectors, X-ray machines, and bullet-proof glass-enclosed badging facilities. A double fence around the plant boundary as well as numerous interim barriers were required to prevent unauthorized access to the plant.

11. <u>NRC I&E Bulletin No. 80-11 (Masonry Walls).</u>

Concrete masonry walls are constructed using precast concrete blocks which are mortared together to form an integral structural unit. They use hollow or solid blocks and may be designed and constructed as reinforced walls, wherein steel reinforcing bars are used. Masonry walls are used extensively in nuclear power plants where offices, laboratories, washrooms, kitchen facilities, etc., are provided and also in safety-related structures. Concrete masonry walls are used in Category I structures only where the strength of reinforced concrete is not required (i.e., interior walls for radiation shielding, fire protection or ventilation boundary requirements for partitions, etc.). Safety-related conduit, junction boxes, instrumentation sensing lines, etc. may be attached to these walls and safety-related equipment, piping, cable trays, ventilation ducts, etc. may be installed near them.

In May 1980, the NRC Office of Inspection and Enforcement issued I&E Bulletin 80-11 which identified NRC concerns with the structural integrity of concrete masonry walls with Seismic Category I piping attached to them. I&E Bulletin 80-11 required utilities to identify all masonry walls in their facilities where failure might affect safety-related systems, re-evaluate the design and construction practices employed in building such walls, justify the acceptance criteria used in the re-evaluation, and perform a confirmatory masonry wall test program, if necessary, to justify the re-evaluation acceptance criteria. In July 1981, the NRC Staff published a Branch Technical Position, "SEB Interim Criteria for Safety-Related Masonry Wall Evaluation", Revision 1, which required that all previously constructed unreinforced masonry walls be re-evaluated to determine the need for reinforcement modifications and that all new walls be built using reinforced concrete or reinforced concrete masonry.

Pursuant to I&E Bulletin 80-11, engineering identified approximately 130 concrete-masonry walls at Limerick requiring reevaluation. As-built drawings were used to confirm safety and non-safety related attachments on the walls. The concrete walls were placed into groups based on similar characteristics and one wall representative of each group was analyzed. Results of these analyses indicated that only one wall required modification.

In addition to these reanalyses, it was necessary to develop a program which would ensure that future attachment of safety and non-safety related items would meet the requirements set forth in the Bulletin. This program was continued through project completion and became a restraint on installation of field-run commodities in late 1980 and 1981, adding substantially more time and non-manual manhours than had been anticipated for field installed commodities routed on masonry walls.

12. ASME Section III Requirements

The NRC's regulations in 10CFR50.55a, issued in June 1971 require the use of ASME Section III criteria for the design and analysis of piping and pipe supports in nuclear plants. Under Section III, a "Design by Analysis" approach is imposed under which every detail of the piping system design must be supported by documented calculations. This approach requires much more rigorous testing and documentation to prove compliance with the Code than was used for the design of piping and pipe supports in earlier vintage nuclear power plants.

Field installation, examination, and testing requirements for ASME Section III are also more restrictive. Much reduced installation tolerances are utilized and additional piping must be examined with techniques such as radiography, liquid penetrant and magnetic particle testing. These techniques require that the surface condition of the finished weld be smoothed to eliminate the potential masking of subsurface defects. Additionally, Section III piping is subject to baseline and periodic inservice inspection per Section XI. In order to perform the required ultrasonic examinations, weld surfaces had to be ground to an almost mirror-like finish, a process which is highly manhour intensive.

C. <u>Conclusions</u>

Based on the preceding information in Section III, it is concluded that:

- There were a large number of new regulatory requirements late in the project which resulted in significant increases in commodities, manual and non-manual manhours, and unit installation rates.

- The late imposition of these regulatory changes necessitated the adoption of a "fast track" approach for the implementation of the required changes into the already congested plant. This approach necessitated that installation occur as the design was being finalized.

- These late changes had a significant adverse impact on system completion and preoperational testing activities.

- The substantial level of intervention in the Limerick licensing process and the resultant intensity of the NRC review necessitated the resolution of essentially all of these regulatory items prior to issuance of the low power license.

- Two particular NRC-mandated design changes (ATWS and RMMS) could not have been completed to support an earlier fuel load date because of the time required for design, procurement and installation of the necessary equipment.

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Q. IR-OCA-4-22. With respect to Mr. Clarey's Schedule 2, please confirm that the listed plants are as follows:

A. LaSalle
B. Susquehanna
C. Grand Gulf
D. Limerick
E. Fermi
F. WPPSS 2
G. Perry
H. Clinton
I. Hope Creek
J. River Bend
K. Shoreham
L. Nine Mile Point 2

If any plant is incorrect, please designate which plant is designated incorrectly, and state the correct plant.

A. IR-OCA-4-22. The above list is correct as presented.

Responsible Witness: J.J. Clarey, Superintendent, Limerick Section