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PENNSYLVANIA PUBLIC UTILITY COMMISSION

v.

PHILADELPHIA ELECTRIC COMPANY

DOCKET No. R-850152

Testimony and Exhibits of

Stephen H. Hanauer

Limerick I and Common

Delay and Cost Increase

Caused by

Errors in Mark II Containment Design

December 1985

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TABLE OF CONTENTS

TABLE OF CONTENTS (continued)

EXHIBITS

SHH-1	Qualifications and Experience of Stephen H. Hanauer
SHH-2	Typical Mark II Pressure Suppression Containment
SHH-3	Milestone Dates for Individual Mark II Plants
SHH-4	Chronology of Mark II Load Definition Program in Response to Identification of Design Errors
SHH-5	PECO Statement on Delay Caused by Mark II-IR-OCA-4-47
SHH-6	PECO Statement on Regulatory Requirements -IR-OCA-4-18.
SHH-7	PECO Statement on Mark II Cost Breakdown - First 2 pages of OCA-4-45.

1. INTRODUCTION AND SUMMARY

1.1 Introduction

Q. Please state your name and business address.

A. My name is Stephen H. Hanauer. My business address is Technical Analysis Corporation, 6723 Whittier Avenue, McLean, Virginia.

Q. On whose behalf are you testifying?

A. I am testifying at the request of the Pennsylvania Office of Consumer Advocate.

Q. What is the purpose of your testimony?

A. My testimony deals with a major error that occurred in the design of the Limerick plant. I explain the nature of the error, estimate its effect on the Limerick plant, and evaluate the responsibility for causing the resulting cost increase.

Q. What are your qualifications to testify as an expert?

A. I testified previously before the Pennsylvania Public Utility Commission In Docket No. P-830453, in Philadelphia Electric Company's Case ECR-8. I have testified before the electrical ratemaking commissions of the States of California, Delaware, Indiana, Maryland, Missouri, and New Jersey. I also performed consulting services for the Public Utilities Commission of Ohio.

During the 12 years I was at the U.S. Nuclear Regulatory Commission, I held several senior technical and managerial positions in that Federal agency and its predecessor, the U.S. Atomic Energy Commission, including Director of Safety Technology. I have also been Professor of Nuclear Engineering at the University of Tennessee and a group leader in the Instrumentation and Controls Division of Oak Ridge National Laboratory. A personal resume is given in Exhibit SHH-1 attached to my testimony.

Q. What experience have you had in evaluating regulatory effects in nuclear power plant construction projects?

A. In my work for State regulatory agencies, I have reviewed regulatory effects in the cost and schedule of the following plants:

Callaway (Missouri)
Diablo Canyon (California)
Marble Hill (Indiana)
River Bend (Texas)
San Onofre (California)
Vogtle (Georgia)
Wolf Creek (Missouri)
Zimmer (Ohio)

Some of these reviews are still underway.

In my earlier work at the AEC and the NRC, the utilities licensed by these agencies often discussed with us the effect of our regulation on the costs and schedules of their projects.

As an engineer at Oak Ridge National Laboratory, I was associated with the design and construction of several nuclear facilities. These were research and test reactors, smaller and simpler than a nuclear power plant. Because of their experimental nature, however, some of these plants presented severe challenges and commanded the attention of some of the best technologists of their time.

Q. What is the basis of your knowledge and understanding of Mark II containments?

A. The AEC Advisory Committee on Reactor Safeguards reviewed all aspects of proposed nuclear power plants, including containment. The Mark II containment was first proposed to and reviewed by the Committee during my tenure as a member and Chairman of the committee. As Technical Advisor to the AEC Director of Regulation (later, the NRC Executive Director for Operations), I followed the evolution of such major technical problems as the GE containment design inadequacy. Later, as manager of NRC Unresolved Safety Issues and then, Director of Safety Technology, I supervised the project managers responsible for resolving the Mark I and Mark II containment problems.

1.2 <u>Summary</u>

Q. Dr. Hanauer, please summarize your testimony.

A. After explaining the design and function of the Limerick containment, a GE Mark II, I explain how the errors made in its design were identified, the substance of the errors, and the responsibilities of GE and PECO as I see them.

(Sections 2 and 3 of my testimony.) The significance of the errors to public safety is given in Section 4. Section 5 describes the NRC requirements for Mark II containments and the nuclear power industry programs that were carried out to bring the Mark II plants into conformance with the requirements. The effect of these programs, and their implementation at Limerick, on the plant cost and schedule are described in Section 6. These are compared with two other Mark II plants in Section 7. My findings and conclusions are given in Section 8.

Q. Please state your findings and conclusions.

A. My findings and conclusions are the following:

1. All nuclear power plants are required to provide a containment system. For Limerick, the Company chose a Mark II Pressure Suppression Containment as recommended by General Electric.

2. General Electric, in developing the pressure suppression containment concepts and basic designs, imprudently failed to adequately measure, predict, or specify the loads and forces which this type of containment would develop in the quenching process, both from loss-of-coolant accidents and from safety/relief valve operation during normal operation and anticipated transients. The technology was available in the 1960's to make such measurements and specifications.

3. The errors were not detected by General Electric, the plant owners or the regulatory authorities until actual plant damage occurred during transients, and additional tests disclosed that the problem would exist for both safety/relief valve discharge and accident circumstances.

4. If the errors had not been identified and corrected, and an accident had occurred, the containment may well have failed to perform its safety function.

5. The NRC required the errors to be corrected.

6. The inadequate GE load specification was not an act of God or a force majeure; it was a technical error. The error in load specifications was eventually corrected. Several Mark II plants have been successfully constructed and licensed by the NRC on the basis of the corrected load specifications. The technical problem was--is--solvable.

7. The utilities owning Mark II plants undertook a long and expensive program of theoretical and experimental work to develop acceptable load definitions. These were finally accepted by the NRC in 1981 and 1982, although some load specifications were provisionally approved as early as 1978. Each Mark II owner had to apply the revised load definitions to re-analyze its plant and make the necessary modifications. These expenditures were necessary under the circumstances,

but the circumstances should never have been allowed to arise.

8. For Limerick, the Mark II program is estimated by the Company to have cost $136.1 million in direct costs. This was not an externally imposed regulatory cost, as stated by PECO, but was made necessary by failure of the Limerick design to conform to the NRC requirements already in existence.

9. The Mark II cost, that would not have had to be spent if the plant had been designed correctly to start with, was unreasonably and imprudently spent and should be disallowed.

10. The Mark II problem is stated by the Company not to have significantly delayed completion of Limerick Unit 1 and Common.

11. The Mark II problem could have been resolved at Limerick by early- to mid-1982. It should not have constrained any later completion date, provided that the Company had earlier taken the appropriate actions to achieve such early resolution.

12. The non-Mark II regulatory requirements cited by PECO could also have been resolved at Limerick by early- to mid-1982.

2. CONTAINMENT FUNCTION

Q. What is the purpose of a containment?

A. Like all nuclear power reactors, the Limerick reactor* is enclosed within a containment system. The purpose of the containment is to isolate the radioactivity within the core and the primary reactor fluids from the outside world in the event of any kind of incident or accident.

Q. How does the Limerick containment work?

A. All containments provide an enclosure around the reactor and the primary system to "contain" radioactive fluids. The Limerick plant uses a containment concept developed at General Electric called "pressure suppression," which includes, inside the containment enclosure, a large pool of water to absorb the heat and steam released from any leaks in the primary reactor system. The Limerick plant uses the Mark II configuration of the GE pressure suppression containment. The containment system includes the large pool of water, called the suppression pool, and a separate large enclosed space, called the drywell, in which the reactor and the primary system are located. The drywell is connected to the suppression pool by large pipes called downcomers, which have open ends located far below the surface of the water in the suppression pool.

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* For brevity, I will describe a single Limerick unit; the two units are essentially identical in containment design.

Steam or hot water coming out of a leak or break in the primary system is directed by the downcomers into the suppression pool, where the relatively cool water absorbs the heat and turns the steam back into additional water. This process is called "quenching".

In a boiling water reactor, such as Limerick, there are valves on the main steam pipes which open automatically if the pressure gets too high, and let out some of the steam to lower the pressure and protect the primary system against overpressure. These valves open not only during accidents, but during certain plant maneuvers which are expected typically a few times a year in a power plant. The steam coming out of these safety/relief valves is another source of high temperature fluid. Pipes from these valves are used to carry this steam below the surface of the suppression pool so the water can absorb the energy in this steam just as it would in an accident. The suppression pool thus serves to quench hot water and steam from two sources: (1) from the safety/ relief valves during plant maneuvers and certain accidents, via the safety/relief valve discharge pipes; and (2) from leaks or breaks in the primary system (and other classes of accidents) via the downcomers.

Q. What does the Limerick containment look like?

A. Over the past three decades, GE has developed three different configurations for its pressure suppression containment system, called Mark I, Mark II, and Mark III. Limerick uses the Mark II system, which is shown schematically in Exhibit SHH-2. The suppression pool is in a cylindrical tank called the suppression chamber at the bottom of the containment building. The water of the suppression pool fills about half of the suppression chamber. The drywell is cone-shaped and sits on top of the suppression chamber so that the floor of the drywell is the ceiling of the suppression chamber. You can see the downcomers and the safety/relief valve discharge pipes leading into the suppression pool in Exhibit SHH-2.

3. ERRORS MADE IN MARK II CONTAINMENT DESIGN

Q. What is the purpose of this section of your testimony?

A. I explain the nature of the errors in the Mark II design and how they were discovered.

Q. When was the Mark II containment concept developed?

A. The basic design of the Mark II containment system was done in the 1960's, and was based principally on tests run in the late 1950's and early 1960's to supply information for the Mark I pressure suppression design.

Q. How were the errors discovered?

A. Three events in the early and mid 1970's showed that the design of the Mark II containment system - and the Mark I system as well--was in error. The designs did not adequately account for the forces developed during the quenching process.

3.1 Wuergassen Reactor Containment Failure

Q. What happened in this event?

A. In April 1972, the Wuergassen plant in Germany was substantially damaged in an unanticipated transient event. This plant has a boiling water reactor and a pressure suppression containment somewhat different in configuration from the one in Limerick, but operating according to the same principles. The Wuergassen accident involved a prolonged opening of one of the safety/relief valves. The resulting steam discharge was successfully quenched in the suppression pool. However, because the safety/relief valve failed to reclose, the quenching process continued much longer than is usual in such events. This heated up the suppression pool water as time went by, and it was observed that the quenching process involved progressively louder noises from the suppression pool. These noises were evidence of peaks in local pressure inside the pool. As the suppression pool water got hotter, the pressure pulses got larger, until finally the pool wall was seriously damaged by the large forces from the pressure pulses.

3.2 Browns Ferry Tests

Q. Why were these tests undertaken?

A. As a result of the Wuergassen incident, and some less damaging experiences in U.S. plants, a series of tests began in 1973 at Browns Ferry. This is a U.S. boiling water reactor with a General Electric Mark I pressure suppression containment. The purpose of the tests was to measure the quenching forces from safety/relief valve openings. For these tests, a safety/relief valve was deliberately opened and measurements were made at various plant conditions.

Q. What were the results of these tests?

A. The Browns Ferry results were consistent with the Wuergassen experience. Higher suppression pool water temperatures produced larger local pressure pulses and larger forces on the suppression chamber wall. As the pool temperatures increased in later tests, the forces became so large that the tests were stopped for fear of damaging the facility if the pool temperature was raised still more, as had originally been planned.

Q. What conclusions were drawn from these data?

A. It was clear from the Wuergassen accident and Browns Ferry tests that the quenching of steam from the safety/relief valve openings could, under some conditions, produce very large forces which were potentially destructive to the-suppression chamber, and far outside the original design basis of the pressure suppression containment system.

3.3 General Electric Mark III Tests

Q. Why were these tests performed?

A. In the period 1972-74, General Electric performed a series of experiments on pressure suppression as part of their development of the Mark III system. General Electric was designing this new configuration to have both safety and economy advantages over the Mark I and II designs. Additional tests were needed to determine the forces and loads for design. These tests were directed toward postulated loss-of-coolant accidents, involving leaks or breaks in the primary system in the drywell. Such an event involves quenching of steam from the downcomers, rather than the safety/relief valve discharge pipes as in the Wuergassen accident and the Browns Ferry tests discussed above.

Q. What results were obtained?

A. The GE Mark III tests also showed that under some circumstances, very large local pressure pulses could occur. These could produce very large loads on the suppression chamber walls, and also on other parts of the pressure suppression system. Additionally, the resulting shaking of the containment structure could impose additional loads on equipment located in the drywell or supported by any part of the containment structure. These results were confirmed by some testing programs in other countries.

Thus, by 1974 or 1975, it was clear that the design of the GE pressure suppression system was seriously deficient.

3.4 The Substance of the Error

Q. What was the nature of the error?

A. The original design considered the mechanical loads associated with accidents in the drywell, including pressure and temperature increases, earthquake loads, the load from the jet of water coming out of the downcomer, and the usual dead loads, hydrostatic loads and test loads required for the design of any structure. However, the design did not include any allowance for the large dynamic loads from the quenching process which had now been shown to occur for both safety/relief valve events and accidents in the drywell. Therefore, there was no assurance that the containment structures would hold together and perform their safety function on the occurrence of a safety/relief valve discharge or a loss-of-coolant accident.

The basic problem with the Mark II containment was the inadequate load specification provided in its original design. Had the containment and the equipment in it been built as originally specified, there is a substantial probability that safety/relief valve actuations would have caused damage. The safety/relief valve quenchers, the suppression chamber, and the equipment in the drywell were inadequately designed to withstand the shaking caused by the quenching process. This could likely have occurred during normal operation and anticipated transients over the lifetime of the plant, which routinely involve multiple actuations of the safety/relief valves. The plants where damage has actually been experienced are earlier pressure suppression containment designs, where the drywell is supported separately from, and relatively independently of, the suppression chamber. How much more severe this might have been in the Mark II plants such as Limerick, where much of the equipment in the reactor building is essentially built using the suppression chamber as a foundation, is not known.

Q. What new technical facts were discovered? What wasn't known when the original designs were developed?

A. No new laws of nature were discovered that showed the need for new design loads. The forces generated during quenching have been known for a long time. During some of the GE pressure suppression tests in the early 1960's, noises were heard indicating large pressure pulses, which would lead to large potential forces. The technology for measuring pressures and forces was available then, and could and should have been employed in the tests.

Q. Was this a generic problem applicable to the entire nuclear power industry?

A. No. Only plants using GE pressure suppression containments were affected by the design errors. The Mark II plants are listed in Exhibit SHH-3. Six of these plants are in operation: LaSalle 1 and 2, Limerick 1, Susquehanna 1 and 2, and Washington Nuclear 2.

3.5 Responsibility for Errors

Q. Who made the errors?

A. General Electric, in developing the pressure suppression containment concepts and basic designs, did not adequately measure, predict, or specify the loads and forces which this type of containment would develop in the quenching process. This includes both loss-of-coolant accidents and safety/relief valve operation during normal operation and anticipated transients. The inadequacy was present in all three configurations of the General Electric pressure suppression containment design. It was not identified by General Electric, the plant owners or the regulatory authorities until actual plant damage occurred during transients, and additional tests disclosed that the problem would exist for both safety/relief valve discharge and accident circumstances.

Q. What are the provisions of the contract between PECO and GE, as they relate to the Limerick Mark II containment?

A. The contract between PECO and GE for "the nuclear systems for Limerick Generating Station" (August 15, 1969) provides that GE furnish PECO with design and operating requirements for the balance of plant, including the containment structures, systems and components furnished by PECO and its contractors.

Appendix A to the contract contains a table giving "Division of Responsibility" between GE and PECO. (Contract Appendix A, pages A.3-1 through A.3-9) The containment is not listed explicitly. It is presumably included under "Other plant systems and equipment, Item XX." (page A.3-9) For this item, the contract gives the following responsibilities:

Scope Design - This denotes responsibility for the design analysis and calculations to establish the essential parameters and requirements for the basic equipment and plant design. This includes the preparation of outline and arrangement drawings, specifications, piping and instrumentation diagrams, process flow diagrams, functional diagrams, and heat balances.

The principal responsibility lies with the Purchaser, but certain requirements which affect the safety or performance of the nuclear system will be furnished by General Electric. Design specifications and procedures prepared by the Purchaser will be made available to General Electric for review and comment.

<u>Detail Design</u> - This denotes responsibility for finalizing the equipment and plant design by performing the necessary analysis and calculations and by preparing the necessary drawings, specifications and instructions, such as detail, final arrangement and assembly drawings; purchasing, installation and testing specifications, and operating instructions. The Purchaser has sole responsibility.

The contract states, "Seller's sole liability with respect to the Plant Requirements shall be limited to correcting, at Seller's expense, substantial error or omissions therein." (Contract, page I-1).

Q. How much information is included in the contract documents?

A. The containment system requirements are given in three pages in Appendix B to the contract (pages 1-1 through 1-3) which reference four drawings in Appendix A. Appendix B includes the statement:

> Design methods that include design features and practices applied to previous General Electric reactor containments are available for information. These features and practices are compatible with the equipment and auxiliary systems that make up the General Electric scope of supply.

(Contract, Appendix B, page 1-2)

This presumably is the way GE transmitted the containment load specification to PECO.

Q. Who made the errors?

A. The errors were omissions of essential loads in the GE information furnished to PECO, that was used by PECO and Bechtel as a basis for the design of the containment. The errors were made by General Electric. These errors then resulted in incorrect design of the containment structures, systems and components, and also other equipment subject to shaking because it is attached to, or supported by the containment.

Q. Who was held responsible by NRC for the errors?

A. The NRC regulates the owners of nuclear power plants. For each approved plant, a single utility is issued the various required NRC licenses. The utility has the obligation of showing that operation of the plant will not be inimical to the health and safety of the public. (Code of Federal Regulations, Title 10, Chapter 1, Part 50, Section 50.40c) When the Mark II design errors were discovered and the NRC (earlier, AEC) was notified, the agency sent letters to the utility licensees and applicants for all the plants with GE pressure suppression containments. Each such utility was required to provide information demonstrating the adequacy of its containment design in the light of the new information.

Q. What was the NRC's role in the development and approval of the Mark II concept and load definition?

NRC's sole role in the Mark II was its safety regulatory responsibility. This means that a utility proposing to build and operate a nuclear power plant that has a Mark II containment must convince the NRC of its safety acceptability. As part of its licensing review, NRC evaluates the conformance of the Mark II containment to the NRC regulatory requirements.

NRC played no role in the development by GE of the Mark II concept and load definitions. The first proposed Mark II was reviewed extensively by the NRC in connection with the application for a Construction Permit for Shoreham. Years later, the NRC Operating License reviews for all the Mark II plants included reviews of containment design in the light of the problems identified in the interim.

Q. Were these errors detected in the various owner and regulatory reviews?

A. No. Neither the Quality Assurance program of the owner and his contractors nor the regulatory review by the AEC and its Advisory Committee on Reactor Safeguards identified the errors.

A. Were the costs of the Mark II redesign and modifications prudently incurred, in your opinion?

A. No. It is true that the costs had to be paid once the Mark II errors were identified; in this sense, performing (and paying for) the Mark II program was necessary. But it should not have been necessary. Sufficient theoretical and experimental work should have been done to enable developing an acceptable Mark II design in the first place. GE's failure to do so was unreasonable and imprudent.

Q. What, in your opinion, is PECO's responsibility?

A. As the owner of Limerick, PECO is responsible to the NRC for conformance with the regulations of that agency. PECO is responsible to its ratepayers and Limerick's neighbors for the safety, reliability and economy of Limerick's operation. Allocation between PECO and its contractor GE of the responsibility owed the ratepayers for the cost of errors in Mark II design is irrelevant, in my opinion. It should be settled between PECO and GE, with or without legal action. I understand that another Mark II plant, Zimmer, is the subject of a lawsuit between the utility owners and GE. I recommend that PECO not be allowed to collect the cost of these unreasonable Mark II errors from the ratepayers; my recommendation is independent of how much of the disallowance PECO can collect from GE.

4. SAFETY SIGNIFICANCE

Q. What is the safety significance of the Mark II containment design errors?

A. In the unlikely event of a loss-of-coolant accident at Limerick, the pressure suppression containment may well have failed to perform its necessary safety function. We now know that the plant as originally designed could not have withstood the forces which are now calculated to occur in the event of such an accident. This type of accident is a much less likely occurrence than the safety/relief valve discharge which would also cause excessive loads, because of the low probability of the severe postulated initiating leak or accident. However, it should be noted that the water in the suppression pool serves as the primary source for the emergency core cooling system. It is just this emergency core cooling system which must function to prevent core melting if an accident were to occur. Therefore, failure of the suppression pool that let the water run out would probably lead to failure of the emergency core cooling system, which is needed and provided to cool the core in the event of just the loss-of-coolant accident that caused the pool failure.

The NRC has required all plants with pressure suppression containments to make extensive engineering analyses and hardware changes. For the operating plants, this has meant long outages; for the plants under construction, this has meant redesign, rework, and delays in completion.

5. PROGRAMS TO CORRECT MARK II DESIGN

5.1 NRC Requirements

Q. When did the NRC first act on the design errors in GE pressure suppression containments?

A. In April 1975, the NRC sent letters to all the utilities having pressure suppression containment systems, requesting that they provide information demonstrating the adequacy of their containment designs. These letters reflect NRC concerns about the need to evaluate the containment response to the recently identified dynamic loads.

Q. What are the NRC requirements for Mark II containments?

A. The NRC has many requirements for containments, including Mark II. The general requirements are that the structures, systems and components necessary to effect the containment function must withstand, with sufficient strength margin and acceptably low leakage, the conditions of normal operations, plant transients and all accidents within the plant's safety design basis. (Code of Federal Regulations, Title 10, Chapter 1, Part 50, Appendix A, Criteria 50-57; NRC Standard Review Plan, Section 6.2)

When the Mark II containment was first proposed in the Shoreham application in the late 1960's, the utility and General Electric presented calculations and experimental results to show compliance with all AEC requirements. After some modifications and additional commitments were made, the design (in its then conceptual state) and the criteria proposed were accepted by the AEC and the Advisory Committee on Reactor Safeguards.

Q. How did the NRC requirements change after the errors were discovered?

A. The basic requirements didn't change at all. When it was discovered that the Mark II containments didn't meet the existing requirements, the NRC issued letters to all Mark II utilities. (NRC letters to Mark II licensees, April 17, 1975 (Safety/Relief valve discharges); April 18, 1975 (Accidents)). These letters requested the utilities to provide information demonstrating the adequacy of their containment designs, in the light of the recently identified errors.

The following sections of my testimony describe briefly the programs used by the utilities in responding to this problem.

5.2 Industry Program

Q. What was the situation of the Mark II plants in April 1975?

A. No Mark II plants were in operation in 1975. Eleven units at eight U.S. sites had been approved and were under construction. Since 1976, seven of these units have been completed (including Limerick Unit 1) one is under-construction, two have been cancelled, and one has been substantially delayed (Limerick Unit 2). The Mark II plants are listed in Exhibit SHH-3.

Q. How did the utilities respond to the NRC letters?

A. The owners of the Mark II plants under construction realized that a long and expensive effort would be required. They formed the Mark II Owners Group to coordinate, direct, and finance the necessary analytical and experimental programs for developing corrected specified loads for redesign of these containment systems. This did prove to be a long and expensive process. The NRC-approved load specification and acceptance criteria for drywell accident forces were eventually published in NUREG-0808, issued in August 1981, over six years after the April 1975 letters to the utilities. For safety/relief valve forces, NUREG-0802 was issued in October 1982. A recent bibliography lists 163 technical references as part of the Mark II program.

Q. Why did the resolution take so long?

A. The Mark II program included actions by GE, plant owners, consultants to the plant owners, and the NRC in several planned steps.

1) GE developed an analytical model of pool swelling.

2) GE and others ran tests of the effect of pool swelling, condensation oscillation, and chugging. These tests required building large thermohydraulic test facilities, and conducting many experimental runs.

3) GE and others prepared reports based on the tests, proposing formulas for defining the loads that would occur on structures, systems and components as a result of the various phenomena. There was some iteration between testing and theoretical development.

4) Kraftwerk Union, a consultant to Pennsylvania Power and Light Company, performed additional, independent tests to determine load definitions resulting from safety/relief valve discharge. Pennsylvania Power and Light Company published the load definitions in the revised Design Assessment Report of Susquehanna.

5) The NRC staff reviewed the technical results. After further testing required by the NRC had been completed and the results incorporated into the load specifications, the NRC approved, with some modifications, the proposed load definition formulas.

6) Plant designers applied the approved formulas to the design or redesign of plant components.

5.2.1 Mark II Lead Plant Program

Q. Why was a "Lead Plant Program" developed?

A. The Mark II program was logical, but the owners recognized from the start that it would take too long. The Mark II plants were licensed for construction during the 1972-74 period and scheduled for completion (at the time they were licensed for construction) from 1975 (LaSalle) to 1980 (Susquehanna 2, Limerick 2).

The owners of the Zimmer, Shoreham, and LaSalle plants foresaw that the technical resolution of this problem was likely to take too long compared to the schedules for these plants. The suppression chamber structure forms the bottom of the reactor building. Any substantial change in this prestressed concrete structure would have to come early in the construction process, or would be difficult, time-consuming, expensive, and perhaps even impossible to accomplish later in the project. Changes in downcomers or safety/relief valve discharge piping would, in principle, be easier to accomplish later during construction, although one could also foresee potential problems in finding adequate foundations to install bracing that might later be found to be needed.

To accommodate the licensing needs of the early Mark II plants, the program was divided into a Lead Plant Program and a Long Term Program. The three lead plants were LaSalle, Shoreham, and Zimmer.

Q. Was Limerick a part of the lead plant program?

A. No. PECO decided not to implement the lead plant loads, but to wait for the results of the long term program. The later results were expected to justify lower loads. Other delays in the Limerick schedule allowed PECO to delay designing much of the containment system until the later results were available. (PECO Statement 8, pages 149-153.) However, some modifications were still required at Limerick during 1981-83 as a result of recent new data and resulting changes in the load definitions. (See Exhibit SHH-4; reference PECO response to Data Requests OCA-4-5 and 4-46.)

Q. What was the scope of the lead plant program?

A. Preliminary load definition formulas developed for the three lead plants were intended to be acceptable more quickly, since they were believed to be more conservative from the safety standpoint than load definition formulas that would be developed later for the other plants. This meant that the available test data would be more limited for the lead plant load definitions, and that a bounding or conservative interpretation of these data would be required. Furthermore, this meant that the lead plant results would be subject to later modification if experimental or theoretical results later showed this to be necessary. Thus, the lead plant program involved a certain risk that, in spite of the caution and conservatism used, some further modifications might be required later. This would involve the expense and delay of modifications late in the construction process.

In addition, the lead plant program involved another kind of risk: the conservative load specification could turn out to be more conservative than necessary.

Q. Why were these risks incurred?

A. The Mark II lead plant owners concluded that it was necessary for them to take these risks, once the design errors had been identified, because they foresaw that their plants would be unacceptably delayed if they had to wait for the completion of the long term program.

Q. Did the lead plant program succeed in its objectives?

A. The lead plant program did not fully succeed. The NRC did not issue its "approval" report, NUREG-0487, until October 1978. This was already late in the design and construction of the lead plants. (See Exhibits SHH-3 and 4) Moreover, the NRC "approval" was hedged in three ways:

1. The NRC increased the conservatism of some of the load definition formulas proposed by the industry; and

2. More future testing was stated by the NRC to be required; and

3. Definitions for some known loads were left out of the Lead Plant program reports, yet had to be resolved for design and construction of these plants to be completed.

In this connection, the NRC staff stated:

> "Several pool dynamic related loads were not included in the Mark II owners' generic program. These will be reviewed on a plant unique basis as a part of the staff's review of the individual design assessment reports."
>
> (NRC, NUREG-0487, page IV-10)

Thus, the loads were <u>not</u> fully defined and accepted in NUREG-0487.

Q. Was that the end of the Lead Plant Program?

A. No. Even the NRC's meagre "approval" of NUREG-0487 was partly rescinded in September 1980 when Supplement 1 to NUREG-0487 stated that recent test results required reassessment of certain load definitions.

Supplement 1 had two parts:

(a) The Mark II owners had proposed some alternative pool dynamic load specifications different from, and less conservative than, those approved in the original NUREG-0487. In Supplement 1, the staff reviewed these proposals, accepted some, accepted some others with modifications and constraints, and found still others not acceptable.

(b) The NRC staff also pointed out that one of the risks had, in fact, turned into a reality. They stated:

> "The results of recent large scale tests have led the Mark II owners to reassess the original lead plant condensation oscillation, and chugging pool boundary load specifications."
>
> (NRC, NUREG-0487, Supplement 1, page IV-1)

This meant that some of the design basis loads were called into question by the recent test results.

Q. Did that complete the Lead Plant Program?

A. No. The results of later tests were discussed by the NRC staff in Supplement 2 to NUREG-0487, issued in February 1981. The staff concluded that the recent large scale tests pointed out deficiencies in the lead plant load specifications because these specifications did not bound the data at all the frequencies and amplitudes that were observed in the new tests. The Owners Group was obliged yet again to develop revised "interim" load specifications which the NRC staff judged were reasonable and acceptable for use in evaluating the lead plants. However, Supplement 2 discussed the application of these load specifications only to the LaSalle plant, with application to other plants left for later evaluation in plant specific documents.

5.3 <u>Completion of the Mark II Program</u>

Q. What was done by the utilities to solve the Mark II problem for non-lead plants?

A. Concurrent with the Lead Plant Program, the Mark II Long Term Program was being pursued by the Mark II Owners Group and their contractors. In fact, the distinction between the programs became blurred after a while. The NRC reports resolving the two programs (Lead Plant and Long Term) cite many of the same technical documents, and were issued less than one year apart.

Q. What did the NRC conclude?

A. The Long Term Program was concluded with the issuance by the NRC of NUREG-0808 for accident loads in August 1981, and NUREG-0802 for safety/relief valve loads in October 1982. In their evaluation, the NRC concluded that the condensation oscillation "interim load specification" used for the lead plants is satisfactory but that the lead plant "interim" chugging load specification was not. Therefore, each of the lead plant utilities was required to reassess its containment still another time, using the NUREG-0808 loads.

Q. Is that where the Mark II stands now?

A. Yes. The two NRC reports concluded that the load definitions, as refined and revised, are adequate in the light of both the theoretical and experimental work of the Owners Group and its contractors and the confirmatory work performed by the NRC staff and its contractors. Work by others was also considered by the NRC, including German tests and calculations performed for U.S. and German utilities, and a Japanese technical program.

Q. Then what had to be done?

A. Following the NRC approval of the load definitions, each Mark II utility had to apply the definitions to its particular configuration, design details and dimensions. It was necessary to calculate the specific loads that would be imposed by the quenching process on the containment and all other equipment influenced by these loads, and the shaking forces that go with them.

Q. What effect did this have on the plants?

A. If there ever are any new Mark II plants, these quenching loads will be factored into their design basis. The actual Mark II plants (those that haven't been cancelled) were all in the late stages of construction in 1981-2 when the load definitions were approved. It was therefore required that the results of the quenching load calculations for each Mark II plant, corrected in accordance with the approved loads, be applied to the plant structures, systems and components that would be affected by them. Any item that could not adequately withstand the newly defined or redefined loads, in combination with other required loads, had to be redesigned and, if necessary, reworked in the field.

6. EFFECT ON LIMERICK

6.1 Introduction

It is clear from the foregoing that during the period of design and construction of the Limerick Station, the design loads of the containment structure were being repeatedly changed. Exhibit SHH-4 attached to my testimony shows a chronology of significant events in the Mark II program and also milestones for Limerick, LaSalle and Susquehanna.

Mark II containment dynamic loads were not accepted by the NRC (even in a preliminary way) until 1978. This means that the basic structural design and construction of Limerick had to proceed in the presence of the uncertainties that have been described in the previous sections.

Uncertainty in the dynamic load specification had an influence on the design and construction of large segments of the plant beyond the design of the containment structure itself. Since the drywell in the Mark II arrangement is mounted on and supported by the suppression chamber, the dynamic forces within the suppression pool and on the suppression chamber are transmitted to the drywell and to the equipment mounted within the drywell or attached to its structure. These forces are very large. The shaking they cause in the containment structure produces shaking forces on the reactor vessel, the reactor core, and the primary system piping, all of which are supported from the drywell.

These forces are similar to earthquake forces, in that they produce back-and-forth and up-and-down motions that tend to shake the different parts of the reactor system. However, they are in some respects larger than the earthquake forces, and occur at different frequencies. Therefore, some parts of the Limerick plant, like all Mark II plants, must be designed as though it were in a highly seismic area, even though the actual earthquake design requirements of the Limerick Station are not particularly severe.

The seismic design of the safety-related parts of the Limerick plant is based on a maximum earthquake acceleration of 0.15g, where g is the acceleration of the earth's gravitational field, 32 feet/second^2. The maximum design earthquake that is used at each plant to define that maximum seismic acceleration is called the Safe Shutdown Earthquake. (Code of Federal Regulations, Title 10, Chapter 1, Part 100, Appendix A.)

The first two Mark II plants to be completed and licensed by the NRC for operation have seismic design requirements comparable to those for Limerick. For the LaSalle plant, the maximum acceleration is 0.2g; for Susquehanna, 0.1g for structures with foundations on bedrock (including the reactor building) and 0.15g for structures with foundations on soil.

The shaking forces from quenching in Mark II plants are generally comparable to or larger than earthquake forces for items strongly affected by quenching. Equipment supports in the drywell at Limerick, and other Mark II plants I have seen, are massive indeed, as though the plant were in an area of large earthquake potential.

The necessity for designing the plant to withstand these large shaking forces, and the uncertainty in how large these forces would be during the time the plant was being designed and constructed, were the cause of required redesign and rework of a variety of plant components.

6.2 Effect of Mark II on Limerick Schedule

Q. How did the Mark II problem impact the Limerick schedule.

A. The Company has stated that the Mark II problem delayed the Limerick project by two months or less. The statement, furnished in response to Data Request OCA-4-47, is attached as Exhibit SHH-5.

Some of the actual times of the required Limerick load definition, analysis and plant modifications were furnished by the Company in response to our data requests. Others have been obtained from NRC documents. They are shown on Exhibit SHH-4 attached to this testimony.

Q. Please summarize the salient features of the Mark II work schedule.

A. The following is a brief narrative summary of the data given in the chronology of Exhibit SHH-4.

- In 1975, PECO suspended containment work in light of the recently identified errors in the GE load definitions. Work was resumed two months later with a strengthened design.

- Starting with Forecast 1 in August 1975, Bechtel added warnings of cost increases and potential schedule impacts caused by Mark II.

- Mark II redesign was begun in January 1979. In Forecast 4 (December 1979) and Forecast 5 (late 1980), Mark II analysis was stated by Bechtel to be behind schedule.

- The safety/relief valve load definition was approved by the NRC in September 1980.

- In late 1980, Bechtel recommended beginning accident load analysis even at the risk of reanalysis and rework. The NRC only approved accident load definitions in August 1981. PECO issued Revision 4 of the Limerick Dynamic Forcing Function Report in November 1981 and the Limerick Containment Design Assessment Report in March 1982.

- Bracing for downcomers and safety/relief valve discharge pipes in the suppression pool was designed by November 1980 and installed during late 1982 and early 1983.

- The revised snubber supports for large piping were designed starting in the latter half of 1981 and installed during the two-year period ending in mid-1983.

- Steel support beams inside containment were modified in late 1981.

- Modifications to supports for small piping, heating and ventilating equipment and electrical equipment were begun in early 1982, and completed with system turnover.

(PECO response to Data Request OCA-4-5.)

Q. How much earlier could Limerick Unit 1 and Common have been completed in the light of the Mark II problem?

A. An as-could-have-been schedule is presented in the Direct Testimony of J.J. O'Brien for the Consumer Advocate in this case. The Mark II effect on that schedule is based on my analysis of the LaSalle and Susquehanna schedules in Section 7 of this testimony. I have concluded that, as far as Mark II is concerned, Limerick Unit 1 and Common could have been completed, ready for licensing and fuel loading, as early as the LaSalle and Susquehanna plants, which were licensed for operation in April and July 1982, respectively.

6.3 Effect of Mark II on Limerick Cost

Q. What effect did the Mark II problem have on the cost of Limerick Unit 1 and Common?

A. In PECO Exhibit No. 2, the Company gives $136.1 million as the direct cost, including engineering, analysis, Owners Group support, and plant modifications. (Pages 8, 33-4) These included approximately 750 new hangers and over 1400 modifications to existing hangers, replacement of the safety/relief valve discharge quenchers, modified bracing for the downcomers and discharge pipes, and modified supports for the snubbers used to dampen shaking forces on equipment and piping. (Page 34; Forecasts)

Q. What cost breakdown has been furnished?

A. In response to a data request, PECO supplemented the information in PECO Exhibit 2 with a detailed breakdown into 15 subheads, supplemented by about 50 pages from Forecasts, Trends, and other cost reconciliation documents. (PECO response to Data Request OCA-4-45.) Exhibit SHH-7 gives the overall cost breakdown portion of this PECO document.

With each of the 15 categories is associated an attachment, the front page of which lists the components of cost increase for the category. The following pages in the attachment are copies of the referenced documents for each component.

Q. Was the cost of the Mark II program a regulatory cost, externally imposed?

A. Not in my opinion. PECO classifies the Mark II costs under "Regulatory and other Externally Imposed Changes". (PECO

Exhibit 2, pages 2a, 8.) The Company states that "new NRC requirements were issued during the period 1978 to 1982." (PECO Exhibit 2, page 8.) But the expensive new Mark II programs and plant modifications were made necessary by the failure of the Limerick design to conform to the NRC requirements that existed before the original design. The "new requirements" cited by PECO are NRC's response to these errors. NRC approved, with some modifications, the acceptance criteria proposed by the Mark II Owners Group to correct the errors.

7. COMPARISON OF MARK II SCHEDULES FOR LIMERICK AND OTHER PLANTS

7.1 Purpose of the Comparison

Q. Why do you compare schedules?

A. The purpose of this comparison is to contribute to answering the question, when could Limerick Unit 1 and Common have been completed? The bulk of this analysis is given in the Direct Testimony of Mr. J. J. O'Brien. However, one of the constraints on such an analysis is the completion of the Mark II program in a faster Limerick construction schedule. To gain insight into this question, I studied the LaSalle and Susquehanna plants. Additionally, possible, non-Mark II regulatory requirements constraints are reviewed, also, in Section 7.4.

Q. Why did you choose those plants to study?

A. I chose those plants because they were the first Mark II plants to be licensed for operation by the NRC. Such licensing included review and approval by the NRC of the containment load definitions, the necessary re-analysis to include quenching loads, and the required modifications of the plants to withstand the quenching loads on all safety-related structures, systems and components.

Therefore, NRC licensing of LaSalle in April 1982 represents the earliest actual approval of a Mark II plant's containment design, with inclusion of the dynamic quenching loads that had been under review since 1975.

7.2 <u>LaSalle Plant Mark II Program</u>

Q. When was the LaSalle containment design approved by the NRC?

A. The NRC staff issued its Safety Evaluation Report in March 1981. The NRC review of dynamic loads concludes with the following statement:

> In conclusion, we conducted an assessment of the LaSalle facility against our generic acceptance criteria. We also reviewed those few areas where alternative criteria have been proposed. In addition, we completed our review of pool dynamic loads that were relegated to plant unique reviews. In each of these areas, we concluded that the pool dynamics load utilized by the applicant are conservative and therefore acceptable.
>
> (NRC, LaSalle SER, page 6-32)

It is clear from the text of the SER review (and from its date) that the review was based on the Mark II Lead Plant load definitions as accepted in NUREG-0487 and its two supplements. Other criteria used in NRC's review included some LaSalle plant - specific proposals and one Susquehanna method, all accepted by NRC.

Later, in the February 1982 Supplement 1 to the Safety Evaluation Report (SSER-1), the NRC required the LaSalle containment to be re-analyzed in a confirmatory review against the Mark II Long Term Program load specifications, recently issued in NUREG-0808. (SSER-1, page 6-1) The NRC stated in SSER-1 that the operating license for LaSalle Unit 1 would include a condition to require this re-analysis within one year of the utility's receipt of NUREG-0808 in September 1981. In fact, the initial LaSalle Unit 1 operating license was issued in April 1982, before the year was up. The initial NRC approval for plant operation was, therefore, based on the Lead Plant Program load definition.

Q. Does this mean that Limerick could have received a license in April 1982?

A. In my opinion, yes, if certain conditions had prevailed. These conditions are:

1. The plant would have to be completed and all non-Mark II licensing requirements fulfilled.

2. The Mark II containment would have to show acceptable performance when analyzed using the LaSalle methods and acceptance criteria, or other acceptable methods and

criteria based on the Lead Plant Program and available data.

Q. Did Limerick meet these conditions in April 1982?

A. No. The plant was not completed, and the Limerick Mark II containment was not designed or analyzed using the Lead Plant criteria. Thus, the load definitions to which the two plants were actually designed and built are not the same.

Q. How would the Mark II requirements have constrained the Limerick plant if it had been completed in April 1982?

A. If the Limerick plant had been completed in April 1982, and met all non-Mark II requirements, then the Mark II problem could have been resolved by that date on Limerick in the same way as it was on LaSalle. This would have required the Company to have decided, much earlier than 1982, to apply the Lead Plant Program criteria and to make, in the Limerick plant, the modifications necessary to meet those criteria.

7.3 Susquehanna Plant Mark II Program

Q. Why did you analyze the Susquehanna plant Mark II program as well as the LaSalle program?

A. The completion and licensing of Susquehanna Unit 1 in July 1982, only a few months after LaSalle, confirms that Mark II problems were not an insuperable obstacle in mid-1982.

Q. When was the Susquehanna Mark II design approved, and on what basis?

A. The Susquehanna SER was issued in April 1981. Even though Susquehanna was not a designated lead plant, its containment was reviewed using the Lead Plant Program criteria, except for five Susquehanna load definitions for which the NRC review was not available in the SER. (Section 6.2.1.8, pages 6-15 through 6-29) The NRC conclusion is given below:

> In conclusion, we conducted an assessment of the Susquehanna facility against our generic acceptance criteria. We also reviewed those few areas where alternative criteria have been proposed. In addition, we completed our review of pool dynamic loads that were relegated to plant unique reviews. In each of these areas, we concluded that the pool dynamics load utilized by the applicants are conservative and therefore acceptable, except for:
>
> 1 - Downcomer lateral load specification (Generic Review);
> 2 - Steam Condensation load specification (Plant Unique);
> 3 - Pool temperature transients involving safety relief valve discharge (Plant Unique);
> 4 - Quencher Air Clearing Load (Plant Unique); and
> 5 - Steam Condensation submerged drag loads (Plant Unique).
>
> We will report our findings regarding these items in a supplement to this Safety Evaluation Report.
>
> (Susquehanna SER, page 6-29)

In its June 1981 SSER-1, the NRC accepted Items 1, 2, and 4 above. Item 5 was resolved in September 1981. (SSER-2) These items were resolved and approved by the NRC before Susquehanna Unit 1 began operation in July 1982. Item 3 was finally resolved in SSER-6 in March 1984, on the basis of compliance with the criteria of NUREG-0783, issued in October 1981.

Q. What was the effect of Mark II on the Susquehanna schedule?

A. Pipe hanger design, installation and inspection was the critical path to completion of construction of Susquehanna Unit 1, and Mark II loads were the reason this activity was on the critical path. (Management Analysis Corporation, An Historical Assessment of the Susquehanna Nuclear Project, An Update 1980-82, November 4, 1982, pages 7, 9-11, and 22.)

Q. How do Susquehanna and Limerick differ?

A. Their designs are similar in many ways. The Susquehanna seismic design basis has a peak acceleration of 0.1 g for the reactor building and containment, which is less than the comparable value of 0.15g for Limerick. For both plants, the reactor building is founded on bedrock.

However, the Mark II design bases for these plants differ. Susquehanna was not a member of the lead plant group, although the Susquehanna design used some lead plant load definitions. Although Susquehanna was a member of the Mark II Owners Group, that utility also funded a program separate from the Mark II Owners Group. It was carried out at Kraftwerk Union (KWU) in Germany, and resulted in different and more severe condensation oscillation and chugging loads that those specified for Limerick. (SSER-3, Appendix I) The bracing in and adjacent to the Susquehanna containment is noticeably heavier than that in Limerick.

Also, KWU designed and tested for Susquehanna a safety/relief valve discharge quencher whose design and load definitions were different than the quencher designed by GE for the Owners Group. In the end, many Mark II plants, including Limerick, used the KWU/Susquehanna quencher.

Q. What do you conclude from the Susquehanna Mark II experience?

A. Even with its plant-unique Mark II criteria, the Susquehanna containment design was approved and the plant licensed in July 1982, about three months after LaSalle. This reinforces my previous conclusion that the Mark II problem could have been resolved at Limerick by sometime in early- to mid-1982, and would not have constrained any later Limerick completion date, provided that the appropriate decisions and actions had been taken earlier.

7.4 <u>Possible Constraints Associated With Non-Mark II Regulatory Requirements</u>

Q. What is the purpose of this section of your testimony?

A. To supplement my review of possible Mark II schedule constraints, I considered whether other NRC requirements would have constrained the Limerick schedule to a date later than early- to mid-1982.

Q. Which NRC requirements did you consider?

A. I used the NRC requirements listed in PECO Exhibit 2, Schedule 1. (page 2a) The last "Miscellaneous" NRC item in that list is broken down in nine pages of details in PECO's response to Data Request OCA-4-29. (Pages 28-36)

Q. Are any of these requirements special or peculiar to Limerick?

A. PECO states that requirements are the same as for other GE BWR Mark II plants. (PECO response to Data Request OCA-4-18, attached as Exhibit SHH-6.) PECO also notes that the schedule and/or scope of a number of requirements was different for Limerick.

Actually, I believe at least one NRC requirement was imposed on Limerick and not on other Mark II plants; conducting a probabilistic risk assessment. However, it was completed in 1981, with later supplements.

Q. Would any of these requirements have constrained Limerick Unit 1 and Common completion in early- or mid-1982?

A. I don't believe they would. Both LaSalle and Susquehanna complied with NRC requirements when they were licensed to operate. The Limerick items with different scopes and/or schedules could have been accommodated at Limerick similarly to their accommodation at LaSalle and Susquehanna. The Limerick probabilistic risk assessment was completed well before 1982.

8. FINDINGS AND CONCLUSIONS

Q. Dr. Hanauer, please state your findings and conclusions.

A. My findings and conclusions are the following:

1. All nuclear power plants are required to provide a containment system. For Limerick, the Company chose a Mark II Pressure Suppression Containment as recommended by General Electric.

2. General Electric, in developing the pressure suppression containment concepts and basic designs, imprudently failed to adequately measure, predict, or specify the loads and forces which this type of containment would develop in the quenching process, both from loss-of-coolant accidents and from safety/relief valve operation during normal operation and anticipated transients. The technology was available in the 1960's to make such measurements and specifications.

3. The errors were not detected by General Electric, the plant owners or the regulatory authorities until actual plant damage occurred during transients, and additional tests disclosed that the problem would exist for both safety/relief valve discharge and accident circumstances.

4. If the errors had not been identified and corrected, and an accident had occurred, the containment may well have failed to perform its safety function.

5. The NRC required the errors to be corrected.

6. The inadequate GE load specification was not an act of God or a force majeure; it was a technical error. The error in load specifications was eventually corrected. Several Mark II plants have been successfully constructed and licensed by the NRC on the basis of the corrected load specifications. The technical problem was--is--solvable.

7. The utilities owning Mark II plants undertook a long and expensive program of theoretical and experimental work to develop acceptable load definitions. These were finally accepted by the NRC in 1981 and 1982, although some load specifications were provisionally approved as early as 1978. Each Mark II owner had to apply the revised load definitions to re-analyze its plant and make the necessary modifications. These expenditures were necessary under the circumstances, but the circumstances should never have been allowed to arise.

8. For Limerick, the Mark II program is estimated by the Company to have cost $136.1 million in direct costs. This was not an externally imposed regulatory cost, as stated by PECO, but was made necessary by failure of the Limerick design to conform to the NRC requirements already in existence.

9. The Mark II cost, that would not have had to be spent if the plant had been designed correctly to start with, was unreasonably and imprudently spent and should be disallowed.

10. The Mark II problem is stated by the Company not to have significantly delayed completion of Limerick Unit 1 and Common.

11. The Mark II problem could have been resolved at Limerick by early- to mid-1982. It should not have constrained any later completion date, provided that the Company had earlier taken the appropriate actions to achieve such early resolution.

12. The non-Mark II regulatory requirements cited by PECO could also have been resolved at Limerick by early- to mid-1982.

Q. Does that complete your testimony?

A. Yes, it does, for the present.

STEPHEN H. HANAUER
Vice President

TECHNICAL ANALYSIS CORPORATION

Technical Analysis Corporation was formed in October, 1981, to provide technical expertise to state agencies, law firms and other clients. It has worked on energy, environmental and communications problems. As Vice President of TAC, Dr. Hanauer participates in a wide variety of technical activities and directs all work on nuclear generating plants. He frequently testifies before regulatory bodies on nuclear and fossil-fueled generation station performance. He is prepared to conduct studies on a wide variety of problems: design, construction and operation of nuclear and other power plants; instrumentation, control and safety issues; and engineering technology.

Since joining TAC in 1982, Dr. Hanauer has presented written and oral testimony in over a dozen electrical rate cases before State Public Utility Commissions. Most of these were reviews of utility management of plant operation and recoverability of replacement energy costs for outages. Some other cases have involved reviews of utilities' plant construction management, involving the prudence of cost overruns and schedule delays.

EDUCATION

Ph.D. (Physics), 1960, University of Tennessee
M.S. (Electrical Engineering), 1949, Purdue University
B.S. (Electrical Engineering), 1948, Purdue University

PREVIOUS EXPERIENCE

1981 to 1982

Director, Division of Safety Technology, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission.

Supervised organization charged with identifying, evaluating, and resolving technical safety issues for nuclear power plants, including;

- o application of probabilistic risk assessment to safety issues;
- o Standard Review Plan for safety evaluation of license applications;
- o Standard Technical Specifications to govern plant operation.

- generation of research information needs and application of research results to nuclear plant licensing;
- review of codes, standards, regulation criteria, and guides for application to nuclear plant licensing;
- assessment of safety priority of over a hundred generic issues based on risk, cost, and benefit;
- overall risk assessment of operation of Indian Point nuclear power station;
- multidisciplinary technical study of pressurized thermal shock and proposed action to resolve the issue.

1980 to 1981

Director, Division of Human Factors Safety, U.S. Nuclear Regulatory Commission.

Following the Three Mile Island accident, started an agency program to review the operational and human factors aspects of nuclear power plant safety, including:

- qualifications of operation personnel;
- operating procedures;
- man-machine interface;
- organization and management of the plant and the utility's nuclear operations.

1979 to 1980

Director, Unresolved Safety Issues Program, U.S. Nuclear Regulatory Commission.

Supervised 20 task groups charged with resolving generically the most important and urgent nuclear reactor safety issues. Work included multidisciplinary technical studies and development of licensing requirements.

1978 to 1979

Assistant Director for Plant Systems, Division of Systems Safety, U.S. Nuclear Regulatory Commission.

Directed three branches in developing nuclear power plant licensing requirements and implementing safety reviews of construction permits and operating license applications: the Instrumentation and Control System Branch; the Power Systems Branch; and the Auxiliary System Branch.

1970 to 1978

Technical Advisor to the Executive Director for Operations, U.S. Nuclear Regulatory Commission. Principal staff advisor to agency executive officer, primarily on technical safety issues.

- Directed task force which developed licensing requirements for emergency core cooling systems; prepared testimony, which was defended in 54 days of cross examination, confirmed in Commission rulemaking, and upheld on Federal court review.

- Organized, staffed, and supervised Applied Statistics Branch;

- Directed small staff in studies for senior management of agency;

- Led an agency staff review group for the Reactor Safety Study, the first probabilistic risk assessment of nuclear power;

- Headed the Special Review Group on the fire at Browns Ferry Nuclear Plant.

1976 to present

Lecturer and Adjunct Professor of Mechanical Engineering, The Catholic University of America, Washington, D.C.

Teaches graduate courses in nuclear engineering and supervises thesis research.

1965 to 1970

Professor of Nuclear Engineering, The University of Tennessee.

Instructed graduate and undergraduate students; supervised master's theses and doctoral dissertations; performed research in theoretical and experimental nuclear reactor technology.

1950 to 1965

Physicist and Development Engineer, Physics Division and Instrumentation and Controls Division, Oak Ridge National Laboratory.

Developed nuclear reactor instrument systems and components; participated in design, installation and operation of instrumentation and control systems in a dozen Oak Ridge reactors; performed research in reactor instrumentation, reactor noise analysis, and reactor safety considerations.

1965 to 1970

Member, Advisory Committee on Reactor Safeguards, U.S. Atomic Energy Commission (permanent review committee established by law to provide independent review of license applications and safety issues). Vice Chairman, 1968; Chairman, 1969.

1962 to 1978

Chairman, Subcommittee 45A, Reactor Instrumentation, International Electrotechnical Commission.

1962 to 1978

Member, Committee N42, Nuclear Instrumentation, American National Standards Institute.

1965 to 1972

Registered Professional Engineer, State of Tennessee, No. 5697.

AWARDS AND COMMENDATIONS

Tenured Professor, The University of Tennessee.
AEC Special Achievement Award (1974).
AEC Distinguished Service Award (1975).
Bonus, Senior Executive Service, U.S. Civil Service (1980).

TESTIMONY

Testified numerous times before the Atomic Energy Commission, the Nuclear Regulatory Commission, the Advisory Committee on Reactor Safeguards, Congressional Committees, and State Public Utility Commissions.

SPEECHES

Has given annual lectures on technical aspects of nuclear plant safety at the Massachusetts Institute of Technology, Georgia Institute of Technology, and Northwestern University, as well as many popular lectures to diverse audiences.

PUBLICATIONS

1. Stephen H. Hanauer, "The Human Factor in Nuclear Power Plant Safety: Progress Since Three Mile Island," Progress in Nuclear Energy <u>10</u>, 299-347 (1982).
2. S. H. Hanauer, "U.S. Nuclear Safety: Review and Experience," Proc. Intern. Conf. Nuclear Power and its Fuel Cycle, Salzburg, 2-13 May 1977, V. 5, pp 3-16, International Atomic Energy Agency, Vienna (1977).
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7. C. O. Thomas, Jr., S. H. Hanauer, and N. P. Baumann, "Experimental Investigation of the Effects of Space and Reflection on the Count Rate Fluctuation Spectra of a Large Nuclear Reactor," Trans. Am. Nucl. Soc. <u>14</u>, 858 (1971).
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10. J. R. Trinko, Jr., S. H. Hanauer, D. P. Roux, and J. T. De Lorenzo, "Use of a Pulse Detector for Reactor Noise Measurements," Nucl. Appl. Tech. <u>8</u>, 522 (1970).

11. J. R. Penland and S. H. Hanauer, "Shutdown Reactivity Measurements at the ORR Using Noise Analysis," Trans. Am. Nucl. Soc. 11 (1968).

12. A. R. Buhl, S. H. Hanauer, and N. P. Baumann, "An Experimental Investigation of Spatial Effects on the Neutron Fluctuation Spectra of a Large Reactor," Nucl. Sci. Engr. 34, 98 (1968).

13. S. H. Hanauer and C. S. Walker, "Principles of Design of Reactor Protection Instrument Systems," Nuclear Safety 9, 28 (1968).

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17. W. E. Ford III, and S. H. Hanauer, "Measurement of Gamma Effects on Observed Subcritical Power Spectral Density," Trans. Am. Nucl. Soc. 9, 508 (1968).

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25. D. P. Roux and S. H. Hanauer, "Use of Reactor Gamma Radiation as a Reactor Control and Safety Parameter," Trans. Am. Nucl. Soc. 6, 74 (1963).

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28. L. D. Roberts, J. W. T. Dabbs, F. J. Walter, S. H. Hanauer, and G. W. Parker, "Nuclear Orientation Experiments with U-233, U-234, U-235, and Np-237 Isotopes," Radioisotopes in the Physical Sciences and Industry," IAEA, Vienna, 1962.

29. S. H. Hanauer, "Instrument for Containment," Nuclear Safety 3, No. 1, 41 (1961).

30. S. H. Hanauer, "Safety System Interconnections Between Experiment and Reactor," Nuclear Safety 2, No. 1, 29 (1960).

31. S. H. Hanauer, J. W. T. Dabbs, L. D. Roberts, and G. W. Parker, "Angular Distribution of Alpha Particles Emitted by Oriented Np-237 Nuclei," Physical Review 124, 1512 (1961).

32. Elbert P. Epler, S. H. Hanauer, and Lester C. Oakes, "Nuclear Reactor Control System," U.S. Patent 2,911,344 (1959).

33. J. B. Ruble and S. H. Hanauer, "A High-Temperature Fission Chamber," Preprint 56, Session XV, Nuclear Engineering and Science Conference, American Institute of Chemical Engineers (1958).

34. E. P. Epler and S. H. Hanauer, "Control and Automatic Startup of the Geneva Conference Reactor," IRE Convention Record V. 4, Part 9, 90 (1956).

TESTIMONY BEFORE STATE REGULATORY AGENCIES

California Public Utility Commission

Docket A.82-02-40 and A.82-03-63
Southern California Edison Company and
San Diego Gas and Electric Company (San Onofre) 1983-5

Delaware Public Service Commission

Docket No. 82-45 Delmarva Power and Light Company 1983

Indiana Public Service Commission

Cause No. 36818-S1 Public Service Company of Indiana (Marble Hill) 1983

Maryland Public Service Commission

Case No.			
	7238-X	Baltimore Gas & Electric Company	1983
	7238-Y/Z	Baltimore Gas & Electric Company	1983
	7239-O	Delmarva Power and Light Company	1982
	7239-P	Delmarva Power and Light Company	1983
	7239-Q	Delmarva Power and Light Company	1984
	7239-R	Delmarva Power and Light Company	1985
	7240-H	Potomac Electric Power Company	1983
	7240-I	Potomac Electric Power Company	1983
	7241-H	Potomac Edison Company	1982
	7241-I	Potomac Edison Company	1983

Missouri Public Service Commission

ER-84-168	Callaway In-Service	1984
ER-84-168 & EO-85-17	Callaway Construction	1984
ER-85-128 & EO-84-147	Wolf Creek In-Service	1985
ER-85-128 & EO-85-185	Wolf Creek Regulatory Change	1985

New Jersey Board of Public Utilities

Docket No. 831-25 Public Service Electric and Gas Company (Salem) 1983-4

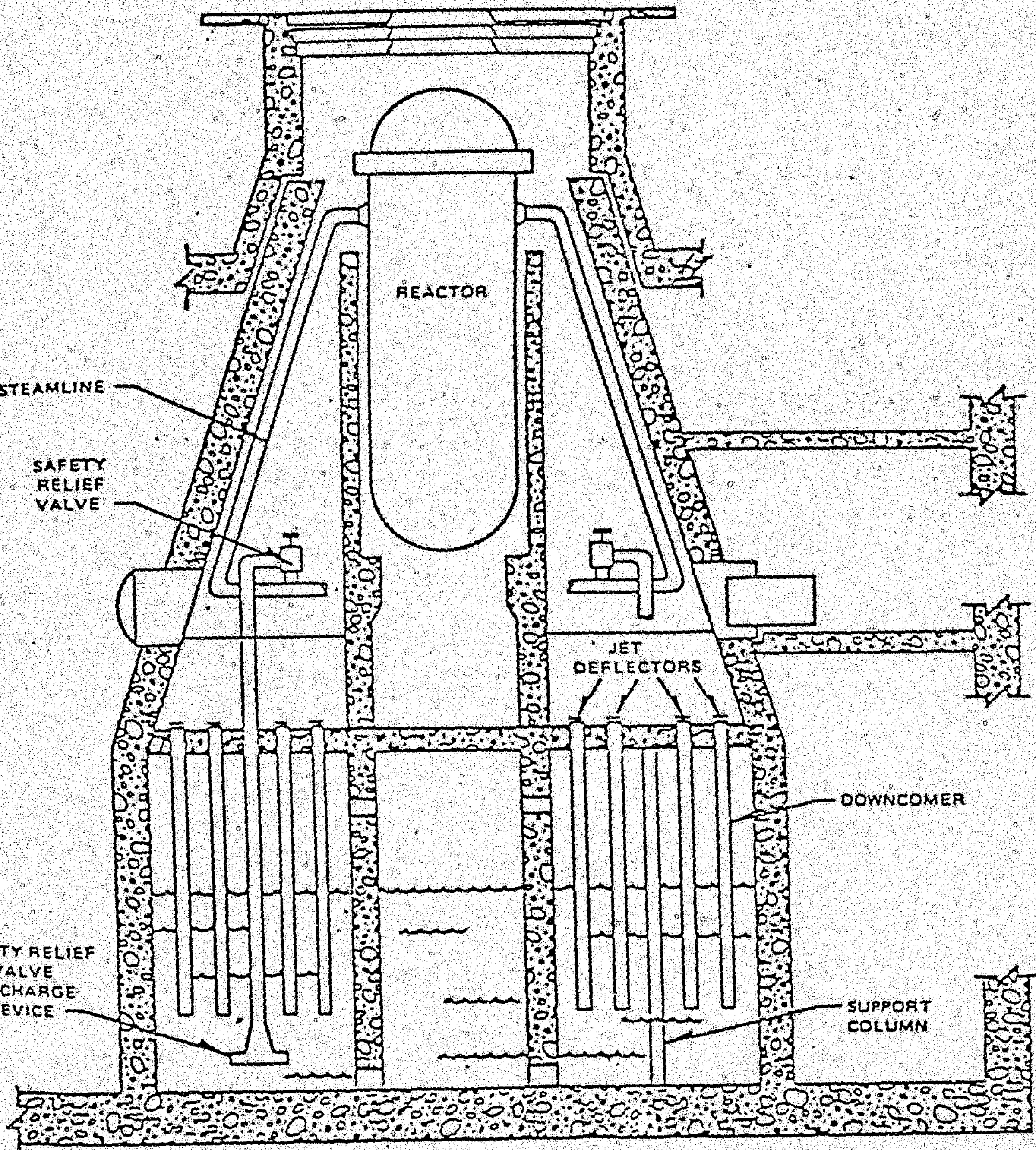
Public Utilities Commission of Ohio

Case 83-1321-EL-C01 Cincinnati Gas and Electric Co. (Zimmer) 1984

Pennsylvania Public Utility Commission

Docket No. P-830453	Philadelphia Electric Company (Salem)	1984
Docket No. R-850152		
Philadelphia Electric Company (Limerick base rate)		1984 (ongoing)

TYPICAL MARK II PRESSURE SUPPRESSION CONTAINMENT



Milestone Dates for Individual Mark II Plants

Plant	Date of CP	Schedule For Fuel Load at CP Date	Actual Operating License Date	Structural Concrete		Large Pipe		Large Pipe Hangers	
				Start	Finish	Start	Finish	Start	Finish
Zimmer	10/72	77	Cancelled	6/73	?	?	—	3/75	—
WNP 2	3/72	77	12/83	5/73	3/78	7/75	9/82E	7/75	12/82E
Shoreham	4/73	77	12/84	3/73	12/77	5/75	10/79	7/75	9/82E
LaSalle 1	9/73	?	4/82	6/74	9/76	11/76	7/78	4/79	?
LaSalle 2	9/73	?	12/83	7/74	9/76	11/76	3/82	?	11/82E
Susquehanna 1	11/73	79	7/82	9/74	8/78	8/74	4/82	8/77	?
Susquehanna 2	11/73	80	3/84	12/76	?	?	6/82	8/77	7/83E
Bailly 1	5/74	?	Cancelled	—	—	—	—	—	—
Limerick 1	6/74	79	10/84	7/74	11/80	1/76	2/83E	1/76	4/83E
Limerick 2	6/74	80	Deferred	9/74	11/80	1/76	—	1/76	—
Nine Mile 2	6/74	78	?	5/77	?	6/76	?	9/77	12/84E

Note: E = Estimate Given in NRC Yellow Book as of June 1982.
? = Information not available or incorrect.

CHRONOLOGY OF MARK II LOAD DEFINITION PROGRAM IN RESPONSE TO IDENTIFICATION OF DESIGN ERRORS

1958-62	General Electric	Pressure suppression containment testing
1969	Shoreham	Approval of Mark II conceptual design by AEC staff and ACRS
April 1972	Wuergassen (German Plant)	Blowdown Event - Containment damaged by quenching forces
1972-75	General Electric	Tests on Mark III containment showing that Mark II load definition was inadequate
1973	Browns Ferry	Tests result in very large forces
October 1974	General Electric	Issues Safety Information letter
April 1975	NRC	Generic NRC letter identifies Mark II problem
April 1975	Limerick	Hold on containment work
April 1975	Susquehanna	Suspends containment construction
June 1975	Limerick	Releases hold on containment work; makes design stronger
October 1975	Mark II Owners Group	Issues Dynamic Forcing Function Report, Revision O
December 1975-June 1976 (March 1976)	Utilities Limerick	Issue Plant Unique Design Assessment Reports (later revised)
May 1976	Mark II Owners Group	Issues Containment Supporting Program Report, Revision O

October 1976	Mark II Owners Group	Issues Revision 2 of Dynamic Forcing Function Report
May 1977	Mark II Owners Group	Divides program into lead plant and long term programs
August 1977	Limerick	Begins Mark II design based on existing models
August 1977	Mark II Owners Group	Issues Lead Plant Topical Report on certain loads
September 1977	General Electric	Identifies new larger loads from multiple SRV actuations
November 1977	Susquehanna	Quencher design
April 1978	Susquehanna	Issues Design Analysis report, subject to future test results
June 1978	Mark II Owners Group	Issues Revision 3 of Dynamic Forcing Function Report
October 1978	NRC	Issues NUREG-0487 "Approved" lead plant load definitions, but left out some loads and required further tests
October 1978	NRC	Tells owners to use square-root-of-sum-of-squares method to combine dynamic loads
January 1979	Limerick	Begins Mark II redesign
March 1979	Limerick	Establishes asymmetric pool boundary loads
October 1979	Susquehanna	Establishes quencher load spec.
November 1979	Limerick	Decides not to use lead plant loads but wait for better definition

January 1980	LaSalle	Issues Design Analysis Report Revision 7, which turned out not to need later revision
March 1980	Limerick	Begins modifications in primary containment
July 1980	Limerick	Establishes need for modifications to Cond-Osc and chugging loads
August 1980	Limerick	Establishes interim condensation - Osc & chugging loads
August 1980	Susquehanna	KWU test results show loads higher than design
September 1980	NRC	Issues Suppl. 1 NUREG-0487 states approval of load definition requires reassessment of recent test results SRV (but <u>not</u> LOCA) loads later confirmed OK
November 1980	Limerick	Issues to field design for bracing in suppression pool
December 1980	Limerick	Conducts in-plant test
January 1981	Susquehanna	3000 hangers to go
February 1981	NRC	Issues Suppl. 2 NUREG-0487 new chugging load definition, still subject to future test results
March 1981	Susquehanna	Discovers (again) structural loads exceed design
April 1981	Susquehanna	Decides to use more severe KWU loads rather than justify switch to OG loads

June 1981-December 1982	Limerick	Issues to field design of large pipe snubber supports
July 1981-August 1983	Limerick	Fabricate and install large pipe snubber supports
August 1981	NRC	Issues NUREG-0808, revised load definitions for LOCA
August 1981	Susquehanna	Hangers lagging
September 1981	Limerick	Issues to field design strengthens supports
November 1981	Limerick	Completes modification of supports
November 1981	Mark II Owners Group	Issues Revision 4 of Dynamic Forcing Function Report
November 1981	NRC	Issues NUREG-0783, suppression pool temperature limits
February 1982	Limerick	Begins issuing to field design modifications of piping, HVAC and electrical supports
March 1982	Limerick	Issues Containment Design Analysis Report
March 1982	Limerick	Issues Containment Design Assessment Report
April 1982	Limerick	Discovers structural loads exceed design
April 17, 1982	LaSalle	Operating license issued
May 1982	Susquehanna	Hangers complete
July 17, 1982	Susquehanna	Operating license issued
August 1982-February 1983	Limerick	Constructs bracing in pool

October 1982	NRC	Issues NUREG-0802 approves SRV load definitions - approves SRV loads in NUREG-0487 Suppl. 1
August 1983	Limerick	NRC-SER says Mark II loads OK

Q. IR-OCA-4-47. Please state the delay in Unit 1 Commercial Operation that the Company attributes to the Mark II problem, and furnish all documents that form the basis of this quantity.

A. IR-OCA-4-47. There was no project delay specifically attributable to the Mark II containment problems other than that associated with a two-month hold on containment wall pours in 1975. The containment wall pour hold resulted in less of an impact on construction activities than that which would have occurred had no such construction hold been ordered to permit thorough evaluation of the problem. This hold period allowed for the inclusion of additonal rebar and embeds to accommodate the increased Mark II design loads.

The Mark II problem, by itself, did not cause project schedule extensions since the required design changes were accommodated within the established overall project schedule. The increased design loads attributable to the Mark II problem did increase the scope of the project along with many other externally imposed changes. The cumulative effect of these required changes was a significant increase in bulk commodities, increased manual and non-manual manhours, higher unit installation rates, and disruption of system completion and preoperational testing activities. These factors, which prevented acceleration of the project schedule beyond that which was achieved, are discussed in PECo Statement Nos. 2, 4, 7, 9 and PECo Exhibit 2.

The Monthly Project Status Reports and the Project Forecasts, which are the basis for the above statments' are available for inspection at the office of Morgan, Lewis and Bockius, 2000 One Logan Square, Philadelphia, PA 19103.

Responsible Witnesses: J. O. Love, TB&A

J. J. Clarey, Superintendent, Limerick Section

D. R. Helwig, Supervising Engineer, Nuclear Services Branch

Q. IR-OCA-4-18. With reference to page 30-31 of Mr. Clarey's testimony, please detail the new and changing regulatory requirements which affected Limerick construction completion which were not also applicable to other GE BWR Mark II nuclear plants.

A. IR-OCA-4-18. No new regulatory requirements affected Limerick which were not applicable to other GE BWR MK II plants. However, the schedule for implementation and/or the scope of the required changes was different in a number of cases at Limerick than at other plants. For example, Limerick was required to complete ATWS modifications, equipment qualification, control room modifications, and fire protection programs prior to receiving its operating license, whereas many other plants were able to address these issues on alternative implementation schedules.

Responsible Witnesses: J.J.Clarey, Superintendent, Limerick Sect.

D.R.Helwig, Supervising Eng.-Nuclear Services Branch

Q. OCA-4-45.

Please furnish all, "...available documentation and meetings and discussions..." as mentioned in the Company's response to DR - OCA 7-1 as the basis for the Company's cost allocation of $ 136.1 million to the Mark II changes.

A. OCA-4-45.

Quantitative cost documentation supporting the cost allocation of $ 136.1 million to Mark II changes, is provided in OCA-4-45 - Attachment 1. The information provided below is also contained in response to Interrogatory OCA-4-29. The costs associated with Mark II are attributable to the following items:

Item a) Engineering to evaluate the suppression pool swell phenomenon and associated costs to incorporate modifications as per NUREG-0487 and NUREG-0808.
Unit 1 & Common Cost: $ 5.1M.

Item b) Engineering to perform Mark II containment analysis and provide technical support to owner's group.
Unit 1 & Common Cost: $ 0.9M.

Item c) Plant unique analysis and support design due to Mark II new loads as per NUREG-0487 and NUREG-0808.
Unit 1 & Common Cost: $ 1.9M.

Item d) Re-evaluation of loads on passive safety-related equipment to assure seismic qualification accommodated Mark II loads.
Unit 1 & Common Cost: $ 0.1M.

Item e) Mark II analysis consisting of: development of a 3-dimensional finite element coupled model of containment and adjacent structures; Mark II containment adequacy assessment; development of reactor pressure vessel (RPV) models based on GE-refined RPV model; and, safety relief valve (SRV) analysis.
Unit 1 & Common Cost: $ 1.1M.

Item f) Functional capability analysis of safety-related large and small pipe due to new Mark II loads.
Unit 1 & Common Cost: $ 0.3M.

Item g) Generated revised spectra using Mark II load specifications for condensation oscillation chugging and re-analysis of piping, equipment and structures per NUREG-0487.
Unit 1 & Common Cost: $ 0.2M.

A. OCA-4-45 (continued).

Item h) Mark II analysis, including: fatigue evaluation of downcomer and SRV lines; analysis of wetwell to drywell vacuum breakers; analysis of shroud support legs; owner's group support; small pipe layout per NUREG-0487; unique and generic analyses; verification of acoustic model for Mark II loads; asymmetric SRV acceleration time histories and analysis of suppression pool temperature monitoring system in compliance with NUREG-0487.
Unit 1 & Common Cost: $ 2.1M.

Item i) Design and licensing of SRV T-quenchers.
Unit 1 & Common Cost: $ 0.7M.

Item j) Determination of final Mark II loads in accordance with NUREG-0487 and NUREG-0808 resulting in significant design changes.
Unit 1 & Common Cost: $ 37.8M.

Item k) Added installation costs for the main steam relief valve (MSRV) discharge line quencher and baseplate.
Unit 1 & Common Cost: $ 1.3M.

Item l) Additional manhours required for welding plates to containment liner for attaching hangers, etc., due to Mark II-related loads.
Unit 1 & Common Cost: $ 0.4M.

Item m) Engineering manhours and modifications to seismic structures adjacent to the containment to account for hydrodynamic loadings.
Unit 1 & Common Cost: $ 33.3M.

Item n) Addition of one redundant vacuum breaker valve to each MSRV discharge line.
Unit 1 & Common Cost: $ 0.2M.

Item o) Increase in snubbers to accomodate Mark II loads; and, additional costs for MSRV discharge line, quencher, and downcomer supports.
Unit 1 & Common Cost: $ 3.8M.

In addition, $ 46.9 million of indirects and distributables are prorated to Mark II, which, together with the above cited costs, provides a total figure of $ 136.1 million for costs associated with Mark II.

Responsible Witnesses:
B. P. Kononetz, TB&A
David R. Helwig, Supervising Engineer, Nuclear Services Branch, PECo

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Public Utility Commission

PECO EXHIBIT NO. 5

R-850152

2-7-86

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DOCUMENT FOLDER

Q. Provide a copy of all testimony, articles, analyses, or other documents prepared by Dr. Hanauer for the 8 plants listed in OCA St. No. 2, p. 2.

A. For the 8 plants, the following documents are available for review at the OCA.

Callaway - Report on Decision Analysis Related To Design and Construction, August 1984.

Surrebuttal Testimony on General Management Performance and Construction, November 1984.

Diablo Canyon -No documents have been published yet on this case. All documents furnished so far are confidential within the California Public Utility Commission Public Staff.

Marble Hill- Audit of July 1982 Cost Estimate for the Marble Hill Nuclear Project, May 23, 1983.

River Bend- No documents have yet been furnished.

San Onofre- Report - Preliminary Review of Construction and Management, May 1983.

Report - Preliminary Review of Construction and Management, February 1984.

Report - Review of Construction and Management, March 1985.

Report - Review of Construction and Management, Issue 5 March 1985.

Report - Review of Construction and Management, Issue 6 March 1985.

Report - Review of Construction and Management, Issue 11, March 1985.

Direct Testimony on - Senior Reactor Operator Licensing Failures Startup Delays and the Impact of High Seismicity on Licensing SONGS 2 and 3, December 1985.

Vogtle - No documents have yet been furnished.

Wolf Creek - Direct Testimony on Regulatory Change and the Health of the Nuclear Power Industry, June 1985.

Surrebuttal Testimony on Regulatory Change, September 1985.

Zimmer -

report Analysis of Possible Mismanagement and Correlated Cost, June 15, 1984.

Witness: Dr. Stephen H. Hanauer

Q. At what time did Dr. Hanauer first become familiar with the GE Mark II pressure suppression containment design? Also provide a complete description of Dr. Hanauer's involvement or association with the discovery and resolution of the Mark II "problem" prior to 1982. Provide references to and copies, if available, of all memoranda, studies, reports or other documents authored by, reviewed by while an employee of the NRC or whose preparation was managed on behalf of the NRC by Dr. Hanauer.

A. (a) As part of the ACRS review of the Shoreham application. This review was conducted during 1969 and culminated in the ACRS report dated December 18, 1969. A copy of the ACRS report is available for review at the OCA.

(b) No "complete description" exists of Dr. Hanauer's involvement or association with the discovery and resolution of the Mark II "problem" prior to 1982. He recalls discussions of the Wuergassen, Browns Ferry and Mark III testing problems but has no documents available to describe the extent of his involvement, and does not recall the details. He read a contemporary report, no longer available to him, regarding the Wuergassen failure. He discussed with TVA personnel their observations during the Browns

Ferry tests. No document is available regarding these discussions.

Starting with his appointment as Director of the NRC Unresolved Safety Issues Program in 1979, he had overall supervision of the resolution of the Mark II problem until his reassignment as Director, Division of Human Factors Safety, in 1980. In 1981, his reassignment as Director, Division of Safety Technology, included supervision of Mark II issue resolution until his departure from the NRC in December 1982. The only documents available now to him are the NUREG reports referenced in PECO and OCA testimony.

Witness: Dr. Stephen H. Hanauer

Q. At the bottom of page four Dr. Hanauer states that the technology was available in the 1960's to make the necessary measurements and specifications to discover the problems with the Mark II containment. Precisely what technology is Dr. Hanauer referring to? For each such technology identified, please give the status of that technology to the best of Dr. Hanauer's knowledge during the 1960's. For each technology identified, was the technology identified available as recently completed research, research validated by replication by other scientists, well-known techniques already in wide-spread common practice, or at some other stage of technological development.

A. The technology to measure loads, forces and pressures was well developed in the 1960's. Since the effects are transient, the use of strain gauges and fast recording systems would be indicated.

Witness: Dr. Stephen H. Hanauer

Q. Reference OCA Statement No. 2, page 6. Has Dr. Hanauer quantified in his testimony the cost "that would not have had to be spent if the plant had been designed correctly to start with." (i.e., Item 9)? If so, provide all such analyses or studies, a full explanation of how they were performed, and all workpapers and calculations produced or employed in analyses or studies.

A. No such quantification appears in Dr. Hanauer's testimony and no such analyses or studies have been performed.

Witness: Dr. Stephen H. Hanauer

Q. Reference OCA Statement No. 2, page 6, Item 11. Please describe all actions which Dr. Hanauer believes the Company would have needed to have taken "earlier" in order to achieve Mark II problem resolution by "early-to-mid-1982".

A. The necessary actions are similar to those which resulted in successful completion of LaSalle Unit 1 and Susquehanna Unit 1 in this time frame. They included pre-1978 decisions to increase the structural strength of the suppression pool--Limerick did the same. At LaSalle and Susquehanna, the utilities decided in about 1978 to begin design work and analysis based on the lead plant load specification, based on NRC's partial approval in NUREG-0487. Later, PECO would have had to make many, if not all of the same changes that were required in the other plants as additional load definition information became available.

Witness: Dr. Stephen H. Hanauer

Q. Reference OCA Statement No. 2, page 6, Item 12. Please explain the bases and provide all data which is believed to support Dr. Hanauer's view that the following non-Mark II regulatory requirements could have been resolved for LGS by early-to-mid-1982:

(a) ATWS Modifications
(b) Environmental Qualification Modifications
(c) Control Room Enhancements
(d) Fire Protection Modifications
(e) Emergency Response Facilities
(f) Post-Accident Instrumentation Enhancements
(g) Probabilistic Risk Assessment

A. All these issues were resolved on Susquehanna Unit 1 and LaSalle Unit 1, in time to permit 1982 licensing by the NRC. They could have been resolved on Limerick, also, although the resolution details might have been influenced by the higher population density at the Limerick site.

Items (b), (c), (d), and (f) involve no special considerations for the Limerick site. For Item (a), the resolution approved by the NRC staff for Limerick was the same as for other boiling water reactors of this vintage. (NRC, Limerick Safety Evaluation Report, pp. 15-20 and 15-21) Item (e) received special attention at Limerick because of the high population density and was a contested issue in the NRC

Operating License hearings for Limerick. In fact, the full-power Operating License for Limerick was delayed by these considerations. There is no way to tell how much earlier these contentions could have been resolved if the plant had been completed 2 years earlier and the licensing process had been correspondingly advanced in time. Item (g) was required for Limerick and a few other plants as a result of its high population density site. Its review certainly added to the complexity of the NRC licensing review process. There is no reason to suppose that, if the plant had been completed earlier, and the probabilistic risk assessment had been completed at an appropriately earlier time, the plant licensing safety review would have been impacted by this item.

Witness: Dr. Stephen H. Hanauer

Q. Reference OCA Statement No. 2, page 14. Provide all documentation or other support for the assertion by Dr. Hanauer that: "During some of the GE pressure suppression tests in the early 1960's, noises were heard indicating large pressure pulses, which could lead to large potential forces.

A. This statement is based on personal recollection by Dr. Hanauer of conversations with General Electric and Atomic Energy Commission employees in the late 1960's and early 1970's. No documentation is known to exist.

Witness: Dr. Stephen H. Hanauer

Q. Did the NRC regulations in existence at the time of the designing of the LGS containment specify hydrodynamic loading as a design criteria? Provide copies of and specific references to the regulation section so specifying, if any.

A. The conceptual design of the Limerick containment was performed by General Electric in the late 1960's. At that time, the NRC was not in existence. Its predecessor, the Atomic Energy Commission, had only general regulations requiring the plant to be constructed and operated so as not to endanger the health and safety of the public. Applicants for licensing were required to describe, in Construction Permit Applications, the proposed design criteria of the facility and, in Operating License applications, the design and safety analysis of the facility showing that it could be operated without undue risk to the public health and safety. (Code of Federal Regulations, Title 10, Part 50) Requirements for containment loads and the like were not included in the regulations at that time.

The structural design of the Limerick containment was developed during the mid 1970's. At that time, AEC and then NRC regulations required, "The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss of coolant accident. (Code of Federal Regulations, Title 10, Part 50, Appendix A, Criterion 50)

It was up to the license applicant and his design contractor to show that the correct loads had been specified and that the containment system was adequately strong to withstand these loads. Like all Mark II applicants, Limerick filed documents with the NRC purporting to show this, but as a result of the "Mark II problem" these documents did not, in fact, demonstrate the adequacy of the load specification and the containment design.

Witness: Dr. Stephen H. Hanauer

Q. On page 20 of his testimony, Dr. Hanauer mentions a lawsuit between the owner of Zimmer and General Electric. Please provide all information that he has regarding this lawsuit. If any documents which you have reviewed previously and relied upon for the statement in his testimony are no longer in his possession, please identify such documents to the best of his ability.

A. An Associate of Technical Analysis Corporation has been approached confidentially about the possibility of TAC representing one of the parties in this lawsuit. There are also newspaper articles. No special knowledge is available.

Witness: Dr. Stephen H. Hanauer

Q. Is it Dr. Hanauer's opinion that the owners of LaSalle, Shoreham and Zimmer were able to achieve lower costs of construction by participating in the lead plant program?

A. No. Information is not available regarding the cost of construction of the containment of LaSalle or Shoreham. Dr. Hanauer has not compared the costs of Mark II containment construction, depending on whether or not a plant participated in the lead plant program.

Witness: Dr. Stephen H. Hanauer

Q. Suppose hypothetically that Susquehanna and LaSalle had turned out to have much more costly containments due to subsequent regulatory changes after large expenditures had already been incurred on their fast-track solution to the Mark II problem. Under these circumstances, would Dr. Hanauer still believe that seeking fast-track solutions as LaSalle and Susquehanna did would have been the prudent course of action, while PECO's more cautious approach would still have been imprudent.

A. The "belief" stated in the question is not to be found in Dr. Hanauer's testimony. If implementing the Mark II lead plant program had, hypothetically, resulted in "much more costly containments due to subsequent regulatory changes after large expenditure had already been incurred on their fast-track solution to the Mark II problem," whether it still would have been prudent for PECO to finish Limerick much earlier in spite of incurring this expense is a cost-benefit analysis that has not been performed by Dr. Hanauer and is not available to him.

Witness: Dr. Stephen H. Hanauer

Q. If PECO had had perfect foresight as to the subsequently discovered Mark II problem and had included in the original design of Limerick Unit 1 a containment structure sufficient to withstand all of the loads that were eventually understood to be necessary, would that hypothetical strengthened the original design have been more expensive than the actual original design? Has Dr. Hanauer done any study of how much? Provide the results of all such studies.

A. No such studies have been performed by Dr. Hanauer.

Witness: Dr. Stephen H. Hanauer

Q. On page 37 of Dr. Hanauer's testimony, he describes his collaboration with Mr. J.J. O'Brien on the preparation of "an as-could-have-been schedule". Precisely what information or conclusions did Dr. Hanauer furnish to Mr. O'Brien to support his analysis?

A. The information transmitted from Dr. Hanauer to Mr. O'Brien is specified on page 37 of Dr. Hanauer's testimony: Dr. Hanauer's analysis of the LaSalle and Susquehanna schedules in Section 7 of Dr. Hanauer's testimony. Dr. Hanauer also had informal discussions with Mr. O'Brien and his colleagues regarding the basis for his analysis and conclusions. In addition, Dr. Hanauer and Mr. O'Brien and his colleagues discussed which activities would be impacted by the Mark II work, and when the information became available that would have allowed these activities to go forward. No documentation exists of these discussions.

Witness: Dr. Stephen H. Hanauer

14. Reference OCA Statement No. 2, pages 39-45, and Exhibit SHH-4. Provide for inspection and copying all documents or other sources of information reviewed by Dr. Hanauer, Mr. O'Brien, others at OKA and/or TAC or at the OCA in obtaining information as to the construction schedules, costs or other data respecting the LaSalle and Susquehanna projects, including especially, but not limited to, all documents or other sources of information related to the resolution of the Mark II problem or other non-Mark II regulatory requirement changes at said plants.

A. Documents available to Dr. Hanauer and his colleagues at TAC are available for review at the office of the OCA in Harrisburg, Pennsylvania. From these documents were obtained the information about Mark II problems and the other schedule and cost information used. It is no longer possible to reconstruct which of this voluminous collection of documents was the source for each number or bit of information.

Witness: Dr. Stephen H. Hanauer

Q. Reference OCA Statement No. 2, page 46. Provide all documentation or other support which Dr. Hanauer believes supports the assertion that: "Both LaSalle and Susquehanna complied with NRC requirements when they were licensed to operate. The Limerick items with different scopes and/or schedules could have been accommodated at Limerick similarly to their accommodation at LaSalle and Susquehanna."

A. This statement is based on Dr. Hanauer's general knowledge and on the voluminous LaSalle and Susquehanna material reviewed by him and made available in response to Item 14 of this set of responses.

Witness: Dr. Stephen H. Hanauer

Q. Provide all specific references relied upon by Dr. Hanauer to conclude that the NRC staff, NRC Commissioners, and/or ACRS were willing to treat the licensing of Limerick on the same basis as LaSalle and Susquehanna.

A. Dr. Hanauer has not concluded that the licensing of Limerick would have been "on the same basis" as LaSalle and Susquehanna. Some differences are discussed in the response to Data Request 6 of this series. The testimony states that the licensing of Limerick could have been resolved in mid 1982, and this statement is based partly on the fact that similar problems were resolved at LaSalle and Susquehanna in this time frame.

Witness: Dr. Stephen H. Hanauer

IR-PECO-OCA-4-17

Q. Has Dr. Hanauer compared Limerick's construction schedule with that of LaSalle and Susquehanna?

A. Some comparisons are given in Exhibit SHH-3 attached to Dr. Hanauer's testimony.

IR-PECO-OCA-4-17a

Q. Has Dr. Hanauer performed a similar comparison with the schedules of Nine Mile Point 2 and Shoreham, both of which have GE Mark II containment systems?

A. Some comparison is given with Shoreham in Exhibit SHH-3. No comparison has been made with Nine Mile Point Unit 2.

IR-PECO-OCA-4-17b

Q. If so, please provide such comparisons.

A. Such comparisons as have been made are included in Exhibit SHH-3.

IR-PECO-OCA-4-17c

Q. If not, what is his general expectation concerning what such a comparison would show.

A. He does not know.

Witness: Dr. Stephen H. Hanauer

Q. Provide all the documentation, workpapers and findings used or generated by Dr. Hanauer in his "sorting through and analyzing these regulatory impacts upon the Limerick schedule". (See OCA Statement No. 1, page 13).

A. This was done informally, using primarily the NRC Safety Evaluation Report, PECO Statement No. 9 and PECO Exhibit No. 2 as sources of information. No workpapers exist.

Witness: Dr. Stephen H. Hanauer

Q. Provide all the documentation, workpapers, findings and materials used or generated by Dr. Hanauer to produce his testimony "that none of the scope additions or rework occasioned by regulatory change has detailed at PECO's Exhibit No. 2 (dated 9/27/85) would have precluded Limerick from acquiring an operating license (with a low-power restriction) and loading fuel sometime in early-/mid-1982". (Reference OCA Statement No. 1, page 13)

A. See response to question 18 of this series.

Witness: Dr. Stephen H. Hanauer

Q. Reference OCA St. No. 2, p. 18. List and briefly describe all of the "essential loads" that in Dr. Hanauer's opinion, were "omitted . . . in the GE information furnished to PECO." Provide all documents and information relied upon by Dr. Hanauer in formulating this option.

A. The omitted and inadequate essential loads are those described and resolved in the NUREG reports on the Mark II program.

Witness: Dr. Stephen H. Hanauer

Q. Reference OCA Statement No. 2, page 12. Describe each of the "additional tests [which] were needed to determine the forces and loads for design. Describe the equipment, technologies, and analytical tools needed to conduct these tests and then to incorporate them into plant containment designs. Provide the dates where each of these equipment technologies and analytical tools become available.

A. Several years and many millions of dollars were spent in conducting the additional tests and developing the additional analyses. The equipment, technologies, analyses and dates are given in a variety of industry reports that are referenced in the NRC reports furnished. Some of these industry reports are available to OCA, but many of them are not and some are proprietary.

Witness: Dr. Stephen H. Hanauer

ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

R-850152

September 20, 1972

J. F. O'Leary, L
F. E. Kruesi, RO
L. Rogers, RS

Here is an idea to kick around. Please let me have your reactions.

S. H. Hanauer, DRTA

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Public Utility Commission

Enclosure

cc: E. G. Case, L
J. M. Hendrie, L
D. F. Knuth, L
R. L. Tedesco, L
V. Stello, L
G. Lainas, L

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1. Conclusions and Recommendations

Recent events have highlighted the safety disadvantages of pressure-suppression containments. While they also have some safety advantages, on balance I believe the disadvantages are preponderant. I recommend that the AEC adopt a policy of discouraging further use of pressure-suppression containments, and that such designs not be accepted for construction permits filed after a date to be decided (say two years after the policy is adopted).

2. Discussion

A pressure-suppression containment system has some means of absorbing the heat of vaporization of the steam in the fluid released to the containment volume. In all three GE models, the steam is forced to bubble through a pool of water and is condensed. In the Westinghouse design, the steam is condensed by flowing it over ice cubes. The objective is to reduce the pressure in the containment through "suppressing" the partial pressure of the steam by condensing it. To be effective, pressure suppression must take place concurrent with the flow of steam into the containment, and its effectiveness is therefore dependent on the rate at which steam is generated or released. If some unexpected event should result in steam generation or flow greater than the suppression capability, then the steam that is not condensed would add an increment of containment pressure. Since the objective of pressure suppression is to permit use of a smaller containment rated at lower pressure than would be required without suppression, then incomplete suppression would lead to overpressurizing a pressure-suppression containment so designed.

It may be noted that the Stone and Webster "subatmospheric" design has little effect on the initial containment pressure rise due to an accident, and is therefore not a "pressure-suppression containment" for the present discussion. In this design, chilled water sprays are used to reduce the containment pressure, and therefore the containment leakage, quickly after a postulated LOCA. The pressure capability and volume are designed to take the full accident, without credit for condensation.

Like all containments, the pressure-suppression designs are required to include margins in capability. Experiments have been conducted by GE and Westinghouse to establish the rate of steam generation that can be accommodated. The pressure-suppression pools, ice condenser, etc., are then sized for the double-ended break steam flow, with margins for unequal distribution of steam to the many modular units of which the condenser is composed. The rate and distribution margins are probably adequate.

More difficult to assess is the margin needed when applying the experimental data to the reactor design. Recently we have reevaluated the 10-year-old GE test results, and decided on a more conservative interpretation than has been used all these years by GE (and accepted by us). We now believe that the former interpretation was incorrect, using data from tests not applicable to accident conditions.

We are requiring an independent evaluation of the ice condenser design and its bases to make less probable any comparable misinterpretation of this design.

Since the pressure-suppression containments are smaller than conventional "dry" containments, the same amount of hydrogen, formed in a postulated accident, would constitute a higher volume or weight percentage of the containment atmosphere. Therefore, such hydrogen generation tends to be a more serious problem in pressure-suppression containments. The small GE designs (both the light-bulb-and-doughnut and the over-under configurations) have to be inerted because the hydrogen assumed (per Safety Guide 7) would immediately form an explosive mixture. The GE Mod 3 and the Westinghouse ice condenser designs (they have equal volumes) require high-flow circulation and mixing systems to ensure even dilution of the hydrogen to avoid flammable mixtures in one or more compartments (see following for an additional serious disadvantage of this needed recirculation and its valves). By contrast, the dry containments only require recombination or purging starting weeks after the accident.

All pressure-suppression containments are divided into two (or more) major volumes, the steam flowing from one to the other through the condensing water or ice. Any steam that flows from one of these volumes to the other without being condensed is a potential source of unsuppressed pressure. Neither the strength nor the leakage rate of the divider (between the volumes) is tested in the currently approved programs for initial or periodic inservice testing. Some effort is now underway to devise a leakage test, but none has so far been accomplished.

Because of limited strength against collapse, the "receiving" volume has to be provided with vacuum relief. In all designs except GE Mod III, this function is performed by a group of valves. Such a valve stuck open is a large bypass of the condensation scheme; the amount of steam that thus escapes condensation can overpressurize the containment.

Valves do not have a very good reliability record. Recently, five of the vacuum relief valves for the pressure-suppression containment of Quad Cities 2 were found stuck partly open. Moreover, these valves had been modified to include redundant "valve-closed" position indicators and testing devices, because of recent Reg concerns. The redundant position indicators were found not to indicate correctly the particular partly open situation that obtained on the five failed valves. We have only recently begun to pay serious attention to these valves, so previous surveillance programs have not generally included them. The GE Mod III design has an elegant water-leg seal that obviates the need for vacuum relief valves.

The high-capacity atmosphere recirculation systems provided for hydrogen mixing involve additional valves which, if open at the wrong time, would constitute a serious steam bypass and thus a potential source of containment over-pressurization. These valves are large, and must open quickly and reliably when recirculation is needed. In other engineered safety features, no single valve is relied on for such service, yet redundancy has not been provided even for single failures, open and closed, of these valves. This is a serious mission, since opening at the wrong time leads to over-pressurization, while failure to open when needed inhibits recirculation.

The smaller size of the pressure-suppression containment, plus the requirement for the primary system to be contained in one of the two volumes, has led to overcrowding and limitation of access to reactor and primary system components for surveillance and in-service testing. Separate shielding of components has tended to subdivide into compartments the volume occupied by the primary system. (Some compartmentation of dry containments also occurs.) A pipe break in one of these compartments creates a pressure differential; each compartment must be designed to withstand this pressure. A method of testing such designs has not been developed.

What are the safety advantages of pressure suppression, apart from the cost saving. GE people talk about a decontamination factor of 30,000 from scrubbing of iodine out of the steam by the water. This is hard to swallow, but some decontamination undoubtedly occurs. One wonders why GE doesn't do an experiment to measure it, and get credit for it. The ice condenser decontamination is measurable but not significant.

Recirculation of the containment atmosphere through the ice has the potential for rapidly reducing the containment pressure by cooling its atmosphere. But in the present design there's not enough ice for that, so containment sprays are furnished (in both volumes), just as in dry containments. Recirculation through the water in the GE designs seems not to have been tried, but may be necessary in Mod 111 for hydrogen control. We have no analysis whether any significant cooling will result.

It is by no means clear that the pressure-suppression containments are, overall, significantly cheaper than dry containments when all costs are included. Information on this point would be useful in evaluating costs and benefits, and should be obtained.

2-7-86

R-850152

Phila

BJS

NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

June 20, 1978

MEMORANDUM FOR: Chairman Hendrie

FROM: S. H. Hanauer, TA EDO

SUBJECT: PRESSURE SUPPRESSION CONTAINMENT

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Public Utility Commission

This note is in response to your telephoned request to me to state my current opinion as to whether designs including the concept of pressure suppression containment should still be licensed. The query is in Senator Hart's letter to you dated June 15, 1978, and refers to my old note of September 20, 1972.

My current opinion is that designs including pressure suppression containments can be licensed, because we have adequate assurance of their safety. This was also my opinion in 1972.

My point in 1972 was that the problems involved in such designs seemed, on balance, not worth the extra resources that were likely to be required to resolve them. I hoped to elicit a study of costs and benefits in sufficient depth to establish whether my intuition was correct. Since such information was not developed for us, I cannot say, then or today, whether I was right.

It now seems to have been naive, even in 1972, for me to suppose that scrapping a design which was adequately safe, though troublesome, would accomplish any significant saving to the taxpayer or the ratepayer. Even in 1972, there were enough pressure suppression containments already approved that resolution of their safety problems was required. Such resolution would apply, without additional work, to new plants using the same designs.

It would be more than naive today for anyone to propose eliminating pressure suppression containments. They work. There are still technical issues requiring further work, including some of the ones I discussed in 1972 and some that have arisen since that time. "Dry" (non pressure suppression) containments have technical issues, too. Both the industry and the NRC have programs to improve our quantitative knowledge of system behavior for all types of containment. Margins in design and available technology are the basis for the acceptability of present designs.

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The Reactor Safety Study included a detailed examination of a BWR with pressure suppression containment. The kind of problems discussed in my 1972 memo are treated in the Study. The risks so calculated, including the various containment failure modes, are acceptable in magnitude.

Thus while we may yearn for the greater simplicity of "dry" containments, the problems of both "dry" and pressure suppression containments are solvable, in my opinion, and the designs safe, therefore licensable.

Stephen H. Hanauer
Technical Advisor to
Executive Director for Operations

cc: Commissioner Kennedy
Commissioner Gilinsky
Commissioner Bradford
L. V. Gossick
E. G. Case
S. Levine

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Public Utility Commission

I. Introduction

A. Problem Definition

There are 11 BWR facilities in various stages of construction with the Mark II containment system that are being built in the United States. None of the domestic facilities with Mark II containments is currently in operation. However, facilities with the Mark II containment in Japan and Italy are currently undergoing initial operational tests. A listing of the domestic BWR facilities with the Mark II containment system is provided in Table I-1.

The original design of the Mark II containment system considered only those loads traditionally associated with design basis accidents. These included pressure and temperature loads associated with a loss-of-coolant accident (LOCA), seismic loads, dead loads, jet impingement loads, hydrostatic loads due to water in the suppression chamber, overload pressure test loads, and construction loads. However, since the establishment of the original design criteria, additional loading conditions have been identified that must be considered for the pressure suppression containment system design.

In the course of performing large scale testing of an advanced design pressure suppression containment (Mark III), and during in-plant testing of Mark I containments, new suppression pool hydrodynamic loads were identified that had not been included explicitly in the original Mark II containment design basis. These additional loads result from dynamic effects of drywell air and steam being rapidly forced into the suppression pool during a postulated LOCA and from suppression pool response to safety/relief valve (SRV) operation, which is generally associated with plant transient operating conditions. Since these new hydrodynamic loads had not been considered explicitly in the original design of the Mark II containment, the NRC staff determined that a detailed reevaluation of the Mark II containment system was required. A similar reevaluation is being conducted for the Mark I containment system design. The results of the short term Mark I reevaluation were documented in December, 1977.(1)

The Mark II containment design was based on the experimental technology obtained from testing performed on a pressure suppression concept for the Humboldt Bay Power Plant and from testing performed for the Bodega Bay Plant concept.(2,3) The purpose of these initial tests, performed during 1958 through 1962, was to demonstrate the viability of the pressure suppression concept for reactor containment design. Tests were designed to simulate a LOCA with various equivalent piping break sizes up to approximately twice the cross-sectional break size of the design basis LOCA. The tests were instrumented to obtain quantitative information for establishing containment design pressures. Data from these tests were the primary experimental bases for the design and the initial staff approval of the Mark II containment system.

During the large scale testing of the Mark III containment system design in the period 1972 through 1974, new suppression pool hydrodynamic loads were identified for the postulated LOCA event. GE tested the Mark III containment concept in its Pressure Suppression Test Facility (PSTF).[67] These tests were initiated for the Mark III concept because of the geometrical configuration differences between the previous containment concepts and the Mark III design, principally in the utilization of horizontal vents. (Steam had been ejected vertically downward into the suppression pool in the previous BWR containment designs, whereas the Mark III design ejects steam horizontally into the suppression pool). More sophisticated instrumentation was available for the Mark III tests as well as computerized methods for data processing.

It was from the PSTF testing that the short term dynamic effects of drywell air being forced into the pool in the initial stage of the postulated LOCA event were first clearly identified.

In addition to the information obtained from the PSTF data, other LOCA-related dynamic load information was obtained from foreign testing programs[5] for similar pressure suppression containments. It was from these foreign tests that oscillatory condensation loads on the vent system downcomers and suppression pool boundaries during the later stages of steam vent flow were identified.

Also, recent experience at operating plants indicated that the dynamic effects of SRV discharges to the suppression pool could be substantial. Although the SRV discharge and the design basis LOCA events may not be directly related, both events are characterized by an initial short period of air injection into the suppression pool followed by an extended period of steam blowdown.

The staff recently issued a report providing a technical update on pressure suppression type containment in U.S. light water reactor nuclea power plants.[66] This report includes additional information relative to the historical development of pressure suppression containment technology, a description of the various suppression containment designs, and a discussion of the review areas and technical bases for licensing suppression containments.

B. Program for Resolution

The NRC sent letters to each of the domestic utilities owning BWR facilities with Mark II containment system designs in April 1975*

*The significant events related to the staff review of the Mark II program are described in Appendix A.