OCA STATEMENT No. 2A
P.J. 3-14-86
Mbg

PENNSYLVANIA PUBLIC UTILITY COMMISSION

v.

PHILADELPHIA ELECTRIC COMPANY

DOCKET No. R-850152

Surrebuttal Testimony and Exhibits of

Stephen H. Hanauer

Limerick I and Common

EFFECT OF NRC LICENSING ON SCHEDULE

AND

COST INCREASE CAUSED BY

ERRORS IN MARK II CONTAINMENT DESIGN

March 1986

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EXHIBITS

SHH-1 through SHH-7 Were Attached to OCA Statement No. 2 filed in this Docket in December 1985.

SHH-8 Pacific Gas and Electric Company, Final Hazards Summary Report for Humboldt Bay Unit 3, September 1, 1961, (selected pages)

SHH-9 In the United States District Court for the Southern District of Ohio, Western Division, Case No. C-1-84-0988, Second Amended Complaint and Jury Demand, filed February 14, 1986, selected pages.

SHH-10 Company response to IR-OCA-4-18, New regulatory requirements.

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 Ave., McLean, Virginia.

Q. Are you the same Stephen H. Hanauer who gave direct testimony in this docket?

A. Yes, I am. My prefiled direct testimony on behalf of the Pennsylvania Office of Consumer Advocate is identified in this docket as OCA Statement No. 2.

Q. On whose behalf are you providing this surrebuttal testimony?

A. This surrebuttal testimony is also at the request of Office of Consumer Advocate.

Q. What is the purpose of your testimony.

A. In my surrebuttal testimony, I respond to information and arguments presented by several Company witnesses on two principal subjects: (1) the error made in the original design of the Limerick Mark II containment by not including the dynamic forces and loads to which this type of containment would be subjected in the quenching process, both from loss of coolant accidents and from safety/relief valve operation; and (2) the effect that Mark II and all other NRC licensing requirements would have had if construction of Limerick Unit 1 had been completed in mid 1982. In addition, I respond to other statements by Company witnesses related to my direct testimony. This surrebuttal testimony is directed principally toward responding to the testimony of Company witnesses Dr. Mattson (PECO Statement No. 9A and PECO Exhibit RJM-1), Mr. Vollmer (PECO Statement No. 31) and Dr. Levy (PECO Statement No. 34). To a lesser degree, I will also respond to statement made by Company witnesses Mr. Clarey (PECO Statement No. 4A), Mr. Helwig (PECO Statement No. 5A), and Messrs. Love and Kononetz (PECO Statement No. 8A).

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

A. My qualifications in these matters are given on pages 1-3 of my direct testimony, OCA Statement No. 2, and a resume is given in Exhibit SHH-1 attached to that testimony. I have 35 years of experience in nuclear technology, including 17 years in various capacities at the U.S. Atomic Energy Commission and its successor, the U.S. Nuclear Regulatory Commission.

1.2 Summary

Q. Dr. Hanauer, please summarize your surrebuttal testimony.

A. Following this introduction, which also contains my principal findings and conclusions, I discuss in Section 2 how the U.S. Nuclear Regulatory Commission would likely have approached the licensing for operation of Limerick Unit 1, if construction of that plant had been completed in mid 1982. In Section 2, I also consider some specific NRC licensing issues in this same context. Section 3 of my surrebuttal testimony discusses the original errors made in the design of Limerick's Mark II containments. Section 4 contains my rebuttal of certain other statements by Company witnesses.

Q. Please state the findings and conclusions of your surrebuttal testimony.

A. The findings and conclusions of my surrebuttal testimony are the following:

1. If construction of Limerick Unit 1 had been completed in mid 1982, the NRC would not have delayed its initial operation by two years as stated by Company witnesses.

2. In the post-TMI era, Limerick Unit 1 was actually scheduled for fuel load in 1983, then 1984, so it had low priority for NRC licensing review resources. If Limerick Unit 1 had been scheduled for fuel load in mid 1982, and if PECO had submitted its application in a timely way in mid 1978 or before, Limerick Unit 1 would have found its place in the NRC's priority scheme. The requirements and schedules of the NRC licensing review

would in all likelihood have been adjusted so as to succeed in making the required licensing decision by mid 1982.

3. The basic NRC licensing requirements for Limerick would probably have ended up looking pretty much like the requirements that Limerick actually had to meet. However, the implementation schedule for these requirements would not necessarily have been the same. Evidence from the other licensing reviews during this period shows that a large number of requirements for plants licensed in 1980-82 had their implementation postponed to a later time.

4. Because of the high population density surrounding the Limerick site, the NRC would have required PECO to perform the Probabilistic Risk Analysis and the Severe Accident Risk Analysis, and these reports would have been thoroughly reviewed by the NRC and its contractors, as actually occurred. However, some combination of preliminary work, staged work and reviews and license conditions would have been found so that the generation of these analyses and their reviews would not have impacted the mid 1982 completion and licensing date of Limerick Unit 1.

5. If Limerick construction had been completed by mid 1982, the heavily contested adjudicatory licensing proceeding would have begun much earlier than it

actually did, the schedule and implementation requirements of the NRC and the willingness of PECO to implement required changes before fuel loading would have been very different, taking into account what was possible and practicable on the mid 1982 schedule. This may have resulted in additional litigated contentions in the hearing, but I believe that favorable Board decisions related to low-power operating license issuance could have been made consistent with the mid 1982 hypothesized construction completion.

6. It is apparent from the 1961 Humboldt Bay Report that some of the dynamic phenomena later to surface in GE pressure suppression containments were or should have been known and measured during the early pressure suppression tests. It also appears from a recent pleading in U.S. District Court that the owners of another Mark II plant had sued GE and others and aver that additional knowledge of dynamic effects was or should have been available to General Electric.

7. Although the Limerick structures and equipment were modified as a result of the Mark II changes, major structural failure of the original design would not have occurred.

8. New or modified NRC requirements related to containments and a general tightening up of NRC requirements were applicable to all GE Mark II plants.

Q. How does this change your original findings and conclusions, as given in your direct testimony in this docket?

A. The findings and conclusions given in my direct testimony (OCA Statement No. 2 pages 4-6) have been substantially unchanged by consideration of the rebuttal testimony of the Company witnesses, as considered in the body of my sur-rebuttal testimony. In the following, I give my findings and conclusions related to Mark II containment and NRC licensing issues in Limerick Unit 1. I have indicated for each conclusion whether it is reaffirmed unchanged, or whether the information and arguments given by the Company rebuttal witnesses and other information, as discussed here in my surrebuttal testimony, have caused me to change my original conclusion.

My overall 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. (Original conclusion reaffirmed)

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 to which this type of containment would be subjected 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. (Original conclusion reaffirmed)

3. The omitted loads and forces were known to General Electric people, but the errors were not reported or correctly evaluated by General Electric, the plant owners or the regulatory authorities until actual plant damage occurred during transients, and additional tests confirmed that the problem would exist for both safety/relief valve discharge and accident circumstances. (Revised conclusion)

4. If the errors had not been identified and corrected, and an accident had occurred, the containment structures would not have failed, but the containment safety function had insufficient assurance, and changes in structures and equipment had to be made. (Revised conclusion)

5. The NRC required the errors to be corrected. (Original conclusion reaffirmed)

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. (Original conclusion reaffirmed)

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. (Original conclusion reaffirmed)

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. (Original conclusion reaffirmed)

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. (Original conclusion reaffirmed)

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

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. (Original conclusion reaffirmed)

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

2. NRC LICENSING APPROACH FOR A 1982 LIMERICK COMPLETION

2.1 Introduction

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

A. In this discussion, I give my views concerning what would have happened in the NRC licensing process if Limerick Unit 1 had been completed in mid 1982. This discussion is hypothetical, since Limerick Unit 1 construction was not completed until much later. However, much is known about how the NRC ran its business in the period we are considering, from 1979 through 1982.

Q. Has the Company presented such an analysis?

A. No, I don't think so. The analysis of Company witnesses has concentrated on what actually happened; that is, how plant construction actually proceeded and how NRC licensing actually proceeded. In fact, the construction of the plant was completed in the Fall of 1984 and initial licensing for operation was accomplished by the NRC in October 1984. Indeed, Company witness Dr. Mattson develops "an earliest possible schedule for completion of the low power licensing process for Limerick 1". (PECO Exhibit RJM-1, starting at page 27) However, in general this exercise assumes that all NRC requirements were immutable; that is, the NRC would have imposed the same requirements on Limerick Unit 1 if construction had been completed in 1982 as was the case for

Limerick Unit 1 whose construction was completed in late 1984.

Q. Do you agree?

A. I believe that, <u>at some time</u>, Limerick Unit 1 would have had to comply with the licensing requirements that, in fact, this plant has been and still is being required to comply with. However, I do not believe that the NRC would have required a plant completed in mid 1982 to delay the beginning of its operation until 1984 because of delays associated with the licensing process. Only the most drastic safety concerns, such as those associated with the discovery that the design process at Diablo Canyon Nuclear Station had been mismanaged and resulted in an unknown but probably large number of design errors, or the discovery that the construction quality assurance at Zimmer Nuclear Power Station could not justify the conclusion that the plant had been adequately built, would have resulted in the NRC delaying a completed plant for several years.

2.2 <u>NRC Licensing Requirements and Schedules</u>

Q. What is the influence of the Three Mile Island accident on the requirements imposed by the Nuclear Regulatory Commission on Limerick and the schedules for implementation of these requirements?

A. The TMI accident in March 1979 was a watershed in NRC licensing history. The accident showed serious deficiencies in industry design and operating programs and in NRC regulatory programs. For about a year after the accident, the NRC was pre-occupied in ensuring that existing, operating plants were adequately safe and in determining what additional design and operating requirements were shown to be necessary by the accident. While most operating plants were allowed to continue operation, no new plants were licensed for operation for about a year after the accident.

When licensing review was resumed, the combination of the many new requirements and the diversion of NRC resources to the operating plants created a log jam in the reviews of applications for operating licenses. Although the first post-TMI operating license was granted on the last day of February 1980, the NRC faced at that time 36 pending applications for operating licenses, and expected to receive another 20 applications representing 38 reactors by the end of 1982. (Energy and Water Development Appropriations for 1981, Hearing of the Sub-committee of the Committee on Appropriations, House of Representatives, April 17, 1980.) At that time, the Commission believed that a large number of plants would be delayed in licensing after their construction was complete.

The substantial licensing delays then expected for a large number of plants did not in fact occur as the months and years passed. There were three reasons for this: (1) construction on many plants was delayed; (2) licensing decisions on some plants were speeded up; (3) ways were found, such as separating zero-power issues, low-power issues, and high-power issues in the licensing review, to allow initial plant fuel loading, zero-power and low-power operation to go forward even though some issues were unresolved. In addition, implementation of a great many requirements, both TMI-related and other, was allowed to be postponed past the dates when low and full power licenses were issued. The reasons for this were the impossibility of completing many studies, engineering redesigns, and plant modifications on the over-optimistic schedules originally proposed, and the results of analyses that showed that the delay in implementation would not have an unduly adverse affect on public health and safety. The end result was that, except for plants in a mess of their own making like Diablo and Zimmer, few plants were actually delayed by licensing past the date when construction was complete.

Q. How were NRC resources allocated?

A. The NRC resources available for review of license applications were allocated in accordance with the predicted dates of construction completion of the various plants. The objective of this allocation was to complete the licensing review for each plant by the day that that plant was ready for a license. The progress of this effort can be followed in the monthly (much later, this was changed to quarterly) reports that the NRC was required to send to Mr. Bevill, the Chairman of the Appropriations Sub-Committee. Starting in

November 1980, each report contains a tabulation of the plants expected to be completed in the next two years at least, together with a comparison, for each plant, of the expected licensing decision date and the expected date of construction completion.

In the first few "Bevill Reports", the NRC gave two estimates of forecasted construction completion, the utility's and the NRC's. Starting in April 1980, only the utility's forecast date was used, at the direction of the Congressional Sub-committee, in order that a later NRC forecasted date of construction completion would not become a self-fulfilling prophecy of delay. Since the utilities' forecasts were often over-optimistic, many predicted delays due to the licensing process did not materialize when construction ended up taking longer to complete.

Q. What scheduled completion dates were forecasted for Limerick Unit 1?

A. During late 1980 and early 1981, Limerick appears briefly in the Bevill Reports as scheduled for completion in late 1983, then disappears from the predictions, which go only through the end of 1983. Limerick I reappears in the September 1981 Bevill Report. At that time, the utility's forecasted construction completion date is October 1984. As we now know, this turned out to be when construction was indeed completed and licensing accomplished.

Q. What is the significance of these dates?

A. Starting when licensing was first resumed after TMI, Limerick was a plant scheduled for what must have seemed at the time to be the distant future. NRC resources were focused on the plants needing operating licenses in 1980, then 1981, and so forth.

Q. Does this mean that the licensing of Limerick was entirely neglected during the period 1980-82?

A. No, it was not. The chronologies in the NRC Safety Evaluation Reports and the discussions and chronologies in the testimony of various Company witnesses attest to NRC activities during this period related to Limerick. In May 1980, the NRC requested that PECO prepare a probabilistic risk assessment (PRA) for Limerick, and in January 1981, the NRC Staff stated that review of the Limerick license application would not begin until a PRA had been submitted. PECO submitted its licensing application and PRA in March 1981.

Q. Would this have been consistent with a mid 1982 construction completion date?

A. Certainly not. Schedule 1 of PECO Exhibit RJM-1 shows that almost 39 months was required between the July 1981 docketing date and the October 1984 issuance of the low power license for Limerick, and that this is the shortest such time for a contested plant since TMI. The median time is 65 months. To maintain a mid 1982 fuel load date, the FSAR for Limerick Unit 1 would have had to have been docketed no later than April 1, 1979, even if the 39 month duration is used. Practically, this would have required submission of the FSAR in mid 1978 or earlier. Limerick I would therefore have been caught in the TMI log jam. As a mid 1982 plant, it would have been on the "short list" of operating license reviews of plants expected to be completed in 1981 and 1982. In fact, a Limerick Unit 1 license issued on July 1, 1982, would have been the tenth low power operating license issued after TMI.

Q. What difference would this have made in the licensing requirements and the licensing review schedule?

A. Both would have been substantially different for a mid 1982 plant than they actually were for a late 1984 plant. The NRC review would have started (or resumed) with the resumption of licensing review in late 1979 and early 1980, and would have been directed toward achieving a licensing decision in mid 1982. Whether sufficient licensing review resources could have been focused on the Limerick review, and whether the contested proceeding could have been resolved by mid 1982 requires additional consideration.

Q. Could the NRC review have been completed by mid 1982?

A. I believe it could. It wasn't attempted, because plant completion was still two years away.

Q. What would have been the licensing requirements?

A. The basic requirements for Limerick would probably have ended up looking pretty much like the requirements that Limerick actually had to meet. However, the implementation schedule for these requirements would not necessarily have been the same. Evidence from the other licensing reviews during this period shows that a large number of requirements for plants licensed in 1980-82 had their implementation postponed to a later time. The problem was that these plants were nearing completion when the additional TMI licensing requirements were specified and people realized just exactly what was necessary to implement them. Originally, such groups as the NRC Lessons Learned Task Force and the NRC group which developed the TMI Action Plan had developed deadlines for implementation of the various new requirements on operating plants, plants then in licensing review, and future new plants (if there ever are any). However, we can see now, and the utilities and the NRC soon began to see at the time, that these schedules had been developed without an adequate appreciation of what was required to implement the various new requirements. The result was a wholesale postponement of implementation of many of the requirements. That is, another priority scheme had to be devised, to make sure that the TMI changes most necessary to safety were made promptly, while others that might be important but less urgent were deferred. Even in

1986, TMI-related changes are still being implemented on some plants.

Q. Does that mean that Limerick Unit 1 could have been licensed for operation by mid 1982?

A. Yes, I believe it could. If the plant construction had been completed in mid 1982, and the Company had filed its application in a timely way in mid 1978 or before, Limerick Unit 1 would have found its place in the priority scheme, and the requirements and schedules of the NRC licensing review would in all likelihood have been adjusted so as to succeed in making the required licensing decision by this time.

Q. Do the Company witnesses agree with you?

A. No, they do not. Several Company witnesses have pointed out that Limerick was unusual (some witnesses say unique) in two important respects: (1) the Limerick site has a high population density surrounding it, higher than all but the two highest population plants in the United States, Indian Point and Zion; and (2) the Limerick licensing proceeding was heavily contested by knowledgeable, motivated intervenors. (PECO Statement No. 1A, pps. 11-12; PECO Statement No. 5A, p. 17; PECO Statement No. 8A, p. 11; PECO Statement No. 9A, pps. 27-36) In the following sections, I consider these two points further.

Q. What would have happened to a mid 1982 Limerick with respect to the requirements imposed by the NRC staff during or after 1982, for example, ATWS, environmental qualification of mechanical systems, and the severe accident risk assessment.

A. I believe that these requirements would have been imposed on Limerick, but I believe that their implementation would have been scheduled in such a way as not to impose major delays on the licensing of a plant which had already been completed. Specifically, if Limerick Unit 1 had been completed, ready for fuel loading, by mid 1982, and licensed at that time for low power operation, as I have stated could have been the case, these 1982, 1983 and 1984 requirements could have been implemented after fuel loading. Some of them (studies such as SARA, for example) could have been performed concurrent with initial operation; others (additional ATWS piping, for example) could have been installed at a refueling outage.

To summarize: I do not believe that the risks of operation without any of these safety improvements were so severe as to require a completed plant to sit around for a year or more for these changes to be made.

2.3 Limerick as a High Population Plant

Q. Just how high is the population around the Limerick site?

A. This is not a simple question, because there are a variety of ways to measure the population. Company witness Dr.

Mattson has reviewed this aspect of the Limerick site (PECO Exhibit RJM-1, pps. 17-19 and references 10, 11, and 15). Three U.S sites - Indian Point, Zion and Limerick - stand out by several measures as substantially more highly populated than the rest of the U.S. nuclear power plant sites.

Q. What is the significance of a high surrounding population density?

A. There are two principal effects on reactor safety: (1) areas of high population density make difficult the execution of some kinds of emergency preparedness measures, especially evacuation of people; and (2) any accident affecting public health and safety would affect more people in a more highly populated area; thus, the societal risk for a given plant is higher in a site of higher population density.

Q. How did TMI affect the consideration of high population sites?

A. TMI showed that evacuation might be a prudent protective measure, even though the area evacuated near TMI was so close to that plant that a comparable evacuation at Limerick would not have involved a large population. Furthermore, TMI showed that serious accidents could happen in a nuclear plant - a fact which had always been true, but which many people did not believe before TMI.

Shortly after TMI, intervenors petitioned for the shutdown of the Indian Point reactors then operating, and Congressional questions were asked regarding continued operation of reactors in highly populated sites. The utility owners of Indian Point and Zion offered to perform comprehensive probablistic risk assessments (PRAs) on those plants in order to enable a technical evaluation to be made of the risks of their continued operation. After considering various other sites, the NRC decided that only Limerick was in the same high population class as Indian Point and Zion. The resulting Limerick PRA and its review by the NRC Staff, including its expansion by adding the SARA, are reviewed in PECO Exhibit RJM-1.

Q. What would have happened in this area of review if Limerick Unit 1 had been a mid 1982 plant?

A. I believe that the course of events would have been substantially different, although the eventual outcome would have been the same. The mid 1980 request by NRC to PECO to perform a PRA on Limerick would very likely have come along about the same time as it did. This was the period during which the NRC staff and Commissioners were formulating their approach to high population plants in general, and the forthcoming contested hearing on continued operation of the Indian Point units in particular. However, since the application for operating licenses for Limerick would have already have been filed years before, and since licensing of Limerick Unit 1 was in mid 1980 expected within a couple of years, I do not believe that there could have been a requirement to delay review of the Limerick FSAR until the PRA had been submitted.

Q. What do you think would have happened?

A. I think the full-blown review of the PRA and the SARA would have taken several years, just as it did in fact. However, I do not believe that licensing of Limerick would have been delayed until the PRA and the SARA had been developed and fully reviewed by the NRC, and litigated in the contested hearing. Rather, I believe that PECO and the NRC would have devised a way to accomplish an abbreviated review of the essential features of the Limerick plant and the Limerick site, so as to enable a preliminary safety decision to be made on a schedule consistent with the mid 1982 construction completion date of Limerick Unit 1. Such a review was in fact performed on Indian Point and Zion, to enable the NRC to decide whether operation in the interim was acceptable pending completion and review of their PRAs, which also took several years. These interim measures included the "60-day studies" performed by the Indian Point and Zion owners (PECO Exhibit RJM-1, p. 18) and also, for Indian Point at least, an initial NRC staff study based on WASH 1400, the Reactor Safety Study, and the known facts about the Indian Point site.

Q. Dr. Hanauer, what do you conclude about the role of the PRA in a mid 1982 Limerick schedule.

A. I believe that because of the high population density surrounding the Limerick site, the NRC would have required PECO to perform the PRA and the SARA, and that these reports would have been thoroughly reviewed by the NRC and its contractors, as actually occurred. However, I believe that some combination of preliminary work, staged work and reviews and license conditions would have been found so that the generation of the PRA and its review would not have impacted the mid 1982 completion and licensing date of Limerick Unit 1.

Q. What were the consequences of the PRA and its review in the Limerick project?

A. The conclusion eventually reached by PECO, the NRC Staff, and the Hearing Board was that the risk, both individual and societal, of the operation of Limerick was comparable to the risks of other nuclear power plants and acceptable. However, because of the high population density, the NRC staff considered further whether the details of the PRA revealed accident sequences for which cost-effective measures in plant design or operation could significantly reduce the risks. They found some such items, and some features were in fact provided by PECO to decrease the risk at Limerick.

Q. Is this what you mean by the effect of the PRA?

A. Yes, it is. I believe that this "effect" would eventually have resulted, probably on a later time scale than actually resulted, even if Limerick had been completed in mid 1982. Some of the changes might in fact have been more difficult and expensive to implement on a plant that was already in operation. However, the cost of delay of a large plant is typically so high that the increased cost of this or that modification usually is insignificant compared to the cost of the delay which might otherwise be incurred.

2.4 Limerick as a Highly Contested Plant

Q. In what context are you calling Limerick a "highly contested plant"?

A. I am here using "contested" as the word is used in 10 CFR 2.4 (N): "contested proceeding" means ..."a proceeding in which a petition for leave to intervene in opposition to an application for a license has been granted or is pending before the Commission." In the Limerick proceeding, there were a large number of intervenors, who appeared to be strongly motivated to prevent the granting of the Limerick operating license or to substantially modify the design and operation of the facility. They were represented by counsel. Following a substantial period of discovery and refinement of contentions, 52 days of evidentiary hearings on the record were held on the issues proposed by the intervenors, accepted by the Board, and not later settled. (PECO Exhibit RJM-1, pps. 13-14)

Q. Besides the time taken by the proceeding itself, what other effect did the heavily contested proceeding have?

A. Dr. Mattson has pointed out at least 2 collateral effects of the heavily contested proceeding on the licensing of Limerick: (1) the fact that their work might be reviewed and heavily cross-examined served to make NRC Staff reviewers more demanding and more rigid in their application of NRC licensing requirements; and (2) in order to settle some contentions, PECO committed to implementing various new (TMI and other) licensing requirements before fuel loading, even though the NRC might have allowed implementation to have been postponed.

Q. Do you agree with Dr. Mattson?

A. Yes, he is describing what actually happened on the time scale of the actual Limerick project.

Q. What would have happened in this respect if Limerick construction had been completed by mid 1982?

A. I believe that in these circumstances the NRC reviewers and their management and the Company would have had a different attitude toward what could have been accomplished by the fuel load date and what could and should be postponed until after that time. This might have resulted in more issues being litigated in the hearing before the Board, since presumably the Company would not have caved in and agreed to have implemented items whose implementation was not possible or practicable before the mid 1982 fuel load date.

Q. Does that mean that you think that the NRC would have taken short cuts with Limerick's safety to maintain the schedule?

A. No, of course not. However, many improvements which are highly desirable for safety are not urgent or immediately required, and the risk in postponing them is slight. To put it another way, the risk - cost - benefit equation of early implementation of this or that safety improvement is very different for a plant whose construction is otherwise capable of being completed in 1982, compared to the same equation for such a plant whose construction is not scheduled for completion until late 1984.

Q. Dr. Hanauer, what do you conclude regarding the effect of the heavily contested proceeding on Limerick's licensing schedule if construction had been completed by mid 1982.

A. I believe that, if Limerick construction had been completed by mid 1982, the heavily contested proceeding would have begun much earlier than it actually did, and the schedule and implementation requirements of the NRC and the willingness of PECO to implement required changes before fuel loading would have been very different, taking into account what was possible and practicable on the mid 1982 schedule. This may have resulted in additional litigated contentions in the hearing, but I believe that favorable Board decisions related to low-power operating license issuance could have been made consistent with the mid 1982 hypothesized construction completion.

3. MARK II DESIGN ERROR

3.1 The Original Errors

Q. What have Company witnesses stated in response to your direct testimony that the original load specification for Mark II containments was incorrect?

A. My discussion of the original Mark II design errors (OCA Statement No. 2, pp. 9-19) has been discussed at length by several Company witnesses. (PECO Statement No. 9A, pp. 2-21; PECO Statement No. 31, pp. 5-9; PECO Statement No. 34, pp. 5-20) The gist of these Company statements is, first, a recital of the development program for pressure suppression containments; second, a statement that the dynamic loads could not have been foreseen; and third, statements that technology available during the time that these containments were being developed was inadequate to allow consideration of these loads.

Q. Do you agree with the statements of these Company witnesses?

A. No; while the Company witnesses' factual recitations are generally correct, I don't agree with their conclusions. Moreover, some additional facts and alleged facts have come to light since I wrote my direct testimony.

Q. What additional facts have come to light?

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Q. Do you agree with the statements of these Company witnesses?

A. No; while the Company witnesses' factual recitations are generally correct, I don't agree with their conclusions. Moreover, some additional facts and alleged facts have come to light since I wrote my direct testimony.

Q. What additional facts have come to light?

A. My colleagues and I went back into the recorded documents of the pressure suppression containment and found some relevant statements in the Humboldt Bay Final Hazards Summary Report issued in September 1961. Humboldt Bay was the first pressure suppression containment. Exhibit SHH-8 attached to this surrebuttal testimony contains selected pages from that report. In particular, Appendix 4, "Pressure Suppression Development Program", contains several revealing statements.

Q. How were the tests instrumented?

A. "The facility was equipped with instrumentation to obtain pressures and temperatures in the drywell, the pool and the air space above the pool". (Exhibit SHH-8, p. 4)

"Provision is made for the measurement of pressures and temperatures at points shown in Figure 9. A light beam oscillograph is used to record transient pressures in the simulated reactor vessel, the drywell and the suppression chamber." (Exhibit SHH-8, p. 7) Thus instruments were provided to observe the transient pressure. Reduction of such data would have been tedious without computers but it could be and was done in various applications at that time, using people instead of computers.

"The drywell pressure built up rapidly, reaching a peak in about .06 seconds and then fell back rapidly." (Exhibit SHH-8, p. 5) Thus, fast-response sensors and recording equipment were evidently provided and used, contrary to assertions in Company testimony.

Q. Were any dynamic effects observed?

A. Yes. The report states:

> "Test results show that tank vibration began when the water was 120-130° F or hotter. It was most severe at high steam flows."
> (Exhibit SHH-8, p. 4)

> "Under certain conditions, the condensation of steam becomes noisy and is accompanied by pressure oscillations. These were observed when the water temperature exceeded about 130° F."
> (Exhibit SHH-8, p. 6)

> "Although it was demonstrated that air carryover does not interfere with steam condensation, it is of interest to know what has been observed in the suppression chamber during some of the test runs.

> "It has been observed that sudden release of air from the submerged vent pipe accelerates the pool level upward resulting in water being thrown high in the chamber. It is believed that the

mass of water was moved upward by air in
the pool."
(Exhibit SHH-8, p. 8)

Thus at least condensation oscillation and pool swell, two of the phenomena to be "identified" and investigated many years later, in the mid 1970s, were observed in pre-1961 tests.

Q. Has any other information come to light on this early history?

A. In suit in the United States District Court for the Southern District of Ohio, Western Division, the owners of the Zimmer plant are attempting to recover from General Electric and others the costs of mistakes made in its Mark II design. Exhibit SHH-9 is part of the amended complaint in that case. In this complaint, the plaintiffs have evidently received GE documents not available to me at the present time that show other dynamic effects such as chugging (Exhibit SHH-9, p. 45), water hammer and unstable steam condensation, for which the description given resembles condensation oscillation phenomena, as they are now known in the Mark II world (Exhibit SHH-9, p. 46), large forces (Exhibit SHH-9, p. 47), and pool swell (Exhibit SHH-9, p. 49). All these dynamic phenomena were later part of the Mark II Owners' Group Program which consumed so much time and so many millions of dollars in the late 1970's.

Q. Dr. Hanauer, please state your conclusion.

A. It is apparent to me from the Humboldt Bay Report that some of the dynamic phenomena later to surface in GE pressure suppression containments were or should have been known and measured during the early pressure suppression tests. It also appears from the recent pleading in U.S. District Court that other Mark II owners have asserted that additional knowledge of dynamic effects was or should have been available to General Electric.

3.2 <u>Role of the AEC and NRC in Pressure Suppression Design</u>

Q. How do you view the AEC and NRC's role in nuclear system design?

A. In my direct testimony, I stated:

> "The 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. Do the Company witnesses agree?

A. Two Company witnesses have commented on this statement of mine. Company witness Dr. Mattson describes the interaction between the AEC and Pacific Gas and Electric and General Electric in the initial approval of the first pressure suppression containment. (PECO Statement No. 9A, pp. 2-5) The AEC (and the ACRS) did not accept the first proposed design or the original test program as adequately supporting the design and asked for more tests in a full scale configuration that better matched the Humboldt Bay design. (PECO Statement No. 9A, pp. 3-4) Additional specification of the test was given by the AEC and the ACRS. This is described by Dr. Mattson as, " How the AEC staff and ACRS were involved in the design of the test program for the Humboldt Bay plant." (PECO Statement No. 9A, p. 4) The ACRS also recommended that the suppression pool incorporate a vent header and also baffles to make it look more like the test facility. Dr. Mattson concludes:

> "These examples of design changes and additional testing shows there was direct involvement by both the AEC staff and the ACRS in the development program and initial design of the pressure suppression containment."
> (PECO Statement No. 9A, p. 5)

Dr. Mattson finds further AEC involvement in this activity:

> "The tests were run and the results were found acceptable by the AEC as a basis for designing the Bodega Bay containment."

Q. What do you think?

A. It was the AEC's (and the ACRS's) job to review the design of the containment and the tests offered in justification of the validity of the design and to approve them or to say where they fell short and require further information. Dr. Mattson can describe this as "involvement" if he wants to. I'll stick to my original statement, which I believe is entirely consistent with Dr. Mattson's description.

Q. Which other Company witness comments on this point?

A. Company witness Dr. Levy also comments on this point. He states that I "understate the AEC/NRC involvement in the development and design of the Mark II containment and in the resolution of the hydraulic loads issue." (PECO Statement No. 34, p. 4) He describes the AEC representatives at the tests, the extensive review of the test data and early containment design evaluations and the directed retesting of certain aspects of the design. Similarly, he states: "The AEC reviewed and approved the original Mark II designs including that of Limerick without raising any concerns respecting hydraulic loads." (PECO Statement No. 34, p. 5) Later on, he describes the NRC's role in the Mark II program to rectify the original errors. He describes it as an "iterative process of review, testing, evaluation, retesting and re-evaluation." Again, I believe this is entirely consistent with the statement in my direct testimony quoted earlier.

3.3 <u>Safety Significance of the Design Error</u>

Q. What did you say about the safety significance of the error?

A. I said,

> "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."
>
> (OCA Statement No. 2, pp. 20-21)

PECO witnesses have correctly stated that I have quoted the NRC concern, and that the facts as they finally emerged are not entirely consistent with the above quoted statement that I made. Company witnesses Mr. Vollmer and Dr. Levy have called my attention to this apparent error (PECO Statement No. 31, pp. 7-8; PECO Statement No. 34, pp. 30-31) The design margins were reduced by the additional forces, but in the end, major design modifications were not required in the Mark II containment structures. The structures of Limerick and other plants were modified to some extent during the pause in construction in 1975 for this purpose. (PECO Statement No. 8A, Appendix A, page A-3) In addition, the bracing added to the downcomers and other structural changes to piping, hangers, ventilating ducts, cable trays and other equipment inside and outside the drywell suggest an insufficiency in some structures before the changes were made.

3.4 New Requirements Related to Mark II

Q. Were there new requirements related to Mark II?

A. Company witnesses Dr. Mattson and Dr. Levy have described changes in NRC requirements related to Mark II containments during the 1970s (PECO Statement No. 9A, pp.18-19; PECO Statement No. 34, pp. 25-26) These include additional requirements relative to safety/relief valve operation, including postulation of multiple simultaneous valve openings, and more severe design basis combinations of loads. Dr. Mattson also describes, correctly, a general tightening up of NRC requirements for more detailed analysis, greater reliance on experimental verification, and more conservative acceptance criteria.

Q. Was any of this special to Limerick?

A. No. In response to an OCA data request, Company witnesses Mr. Clarey and Mr. Helwig state: "No new regulatory requirements were effected which were not applicable to other GE BWR MK II plants." (Exhibit SHH-10) These witnesses also commented on more severe implementation requirements for Limerick. In fact, the "related" requirements discussed by Drs. Mattson and Levy were applied to all BWR Mark II plants.

3.5 <u>Meaning of the Mark II Program</u>

Q. What does Dr. Levy say about the significance of the Mark II program.

A. Dr. Levy states:

> "The length and complexity of this program, its many iterations of testing and evaluation, clearly establish the error of Dr. Hanauer's position. Certainly, if the loads had been so obvious that their nonrecognition prior to 1974-1975 time period could constitute 'technical error', then it

would not have taken seven years for the combined resources of the NRC, the BWR utilities, GE and many other domestic and foreign contractors to define their nature and develop design solutions."
(PECO Statement No. 34, p.5)

Q. Do you agree?

A. Certainly not. There is a vast difference between the recognition of these dynamic loads and the necessity for including them in the program, and the technical work required to understand and adequately specify them. I agree that the latter took many years and many millions of dollars, as the Company witnesses state. However, I have shown that the existence of these phenomena could and should have been identified early. At that time, there was plenty of time to do the necessary experimental and theoretical development and get the design of these facilities right the first time.

3.6 Effect of Mark II Loads On Other Parts of the Plant

Q. Is much of the equipment in the reactor building essentially built using the suppression chamber as a foundation?

A. Yes. I stated this in my direct testimony. (OCA Statement No. 2, p. 14) Company witness, Mr. Vollmer says I was wrong.

> "The reactor building floor and foundation are separated from the containment by a "seismic" gap which is specifically provided such that no load is transmitted from the containment directly to the reactor building. As the Limerick containment is founded on bedrock, some hydrodynamic loading would however be transmitted through the rock to the adjacent structures."
>
> (PECO Statement No. 31, p. 15)

Mr. Vollmer describes this as "unique" to the Limerick design, then a few lines later he states that Susquehanna has a similar design.

Inspection of Mr. Vollmer's Schedule 3 will show that the reactor, all the equipment in the drywell, and some equipment outside the drywell but supported on the drywell is shaken as a result of forces in the suppression pool. Item M on page 2 of Mr. Vollmer's Schedule 1 is related to modifications "adjacent to the containment" to account for hydrodynamic loading. The backup material furnished by PECO in response to an OCA data request includes structures and equipment outside containment. Examples include the control rod drive hydraulic equipment frame steel, (Forecast 6, page 149), and control room hanger modifications.

4. OTHER REBUTTAL MATTERS

4.1 Does My Testimony Disagree With Mr. O'Brien's?

Q. Where is it stated that you two OCA witnesses disagree?

A. The apparent disagreement is highlighted by Company witnesses Messrs. Love and Kononetz. (PECO Statement No. 8A, p. 18) I don't think these Company witnesses are correctly interpreting our two quoted statements. My emphasis on "results" is not the same as Mr. O'Brien's emphasis on "the bottom line". I intended "results" to include what was done, rather than the organization charts and committee meetings which I was discussing in the testimony referenced by these Company witnesses.

Q. How do you use "results" in your analysis?

A. The results - cost and schedule components - are a very useful way to begin the process of analysis. Where the results are inconsistent with previous experience or apparently inconsistent with each other, you know it is worth looking into whether the Company behaved reasonably in achieving those results. However, it is the Company's actions, based on what they knew or should have known at the time, on which I base my evaluation.

4.2 Milestones Schedule SHH-3

Q. What is the problem here?

A. Company witnesses Love and Kononetz (PECO Statement No. 8A, pp.46-8) and Clarey (PECO Statement No. 4A, p. 17) both assert that there are errors in my Exhibit SHH-3.

Q. Have you checked these dates again?

A. I rechecked the date for Susquehanna Unit 1. For this date, the Company witnesses are right and my Exhibit SHH-3 is wrong. For LaSalle Unit 1 the problem is more difficult, since the June 1982 Yellow Book doesn't contain LaSalle Unit 1. I got the date in my Exhibit SHH-3 from an earlier Yellow Book which is no longer available to me, so I will accept the Company witnesses' correction on this date also. I regret that these errors crept into the table.

Q. Does that conclude your testimony?

A. Yes. My conclusions were given earlier in Section 1.3.

3 **PACIFIC GAS AND ELECTRIC COMPANY**

2 **FINAL HAZARDS SUMMARY REPORT**
1 Docket-50133-1 **HUMBOLDT BAY POWER PLANT**
UNIT NO. 3

SEPTEMBER 1, 1961

Appendix IV

PRESSURE SUPPRESSION DEVELOPMENT PROGRAM

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up to 150 pipes. Variables in this facility were primary system break area, dry well volume, number of vent pipes, total vent area, and the depth of submergence of the vents in the pressure suppression pool. The test parameters were scaled down from the Humboldt Bay design by a factor of 10 for length, 100 for area, and 1000 for volume. The time scale in the model is one tenth that of the Unit.

The facility was equipped with instrumentation to obtain data on pressures and temperatures in dry well, the pool, and the air space above the pool when the diaphragm was ruptured.

In some tests, substances were added to the water in the model vessel to simulate fission products, and after the tests, samples were taken from the air space above the water pool for analysis to determine the ability of the pool to retain fission products.

3. Test Results

a. Condensing Tests Without Compartment in Tank

The first series of tests with the Condensing Test Facility were without the box compartment in the tank.

The following range of parameters was tested:

Vents:	4, 6, 8, and 14 inch diameter single straight pipes, and a triple 4 inch diameter vent
Depth of submergence:	1 inch to 6 feet
Direction of discharge:	Vertically downward and horizontal
Steam flow:	10,000 to 93,000 lbs/hr
Tank water temperature.	50°F to 150°F

Steam was completely condensed in all but three out of forty tests in this series. The exceptions were: 6 inch diameter pipe discharging 60,000 lbs/hr horizontally with 6 inch nominal depth of submergence; 8 inch diameter pipe discharging 83,000 lbs/hr horizontally with 6 inch nominal depth of submergence; and 8 inch diameter pipe discharging 77,000 lbs/hr downward vertically with 1 inch depth of submergence.

At high steam flow rates, the water in the pool appeared to be mixing well, the surface was agitated, and vortices formed temporarily around the pipe.

b. Tank Vibration Tests

Test results showed that tank vibration began when the water was 120-130°F or hotter. It was most severe at high steam flows.

The dry well pressure built up rapidly, reaching a peak in about .06 seconds and then fell back rapidly. Within the range of parameters tested, the peak dry well pressure occurred at the instant the pool water was completely pushed out of the vent pipes by expanding air and steam in the dry well. In some tests, water was forced back into the dry well, by the suppression chamber air pressure, a few seconds after firing. The water then condensed the steam remaining in the dry well, creating a vacuum.

Pressure in the air space over the water pool reached a maximum shortly after the dry well peak pressure occurred. In all tests, the pressure over the water pool returned to essentially atmospheric pressure seconds after the firing.

e. Effectiveness of Water Pool as a Barrier to Escaping Fission Products

The specific objective of five tests was to determine qualitatively the effectiveness of the water pool as a barrier to fission products which might be discharged into the pool from the dry well. The Transient Test Facility was used with some special equipment for releasing simulated fission products and for measuring the quantity escaping through the pool to the enclosure.

The simulated fission products used in testing were xenon, krypton, sodium iodide, iodine crystals and zinc sulfide with a mean particle size of two microns. The xenon and krypton were placed inside the pressure vessel in glass bottles which broke when the test began. The other simulated fission products were placed directly in the water in the vessel. Air samples were drawn from the space over the water pool after a test and examined for the amount of "fission product". A mass spectrometer was used to detect noble gases, a scrubbing column and examination of the scrubbing agent was used for iodine and sodium iodide, and in the case of fluorescent zinc sulfide, the air was drawn through a filter paper and the particles counted.

Test results for simulated fission product entrainment indicate that the water pool retained a very high proportion of impurities entering from the dry well. The measured separation factor between the dry well and the enclosure was of the order of 10^{-8} for solid particles, and 10^{-5} to 10^{-6} for the halogen and soluble salt. More than half of the noble gases were retained initially, which is considerably more than expected on the basis of solubility.

4. Conclusions

A summary of conclusions from the intial test program follows:

1. Condensation of steam entering water from nozzles consisting of simple open ended pipes is generally very rapid and complete. In the initial test program, all the steam was condensed except with a horizontal jet six inches below and a vertical jet directed downwards and submerged one inch below the water surface. For all other conditions, condensation was complete with water temperatures up to 170°F (the highest obtained).

2. Analytical prediction of the transient facility test results and correlation between analysis and test results indicated that the behavior of a pressure suppression system can be calculated with sufficient accuracy.

3. There is a tendency for a jet to depress the water surface in its vicinity. If air is sucked into the jet because of a depressed water surface, condensation is not particularly affected.

4. Under certain conditions, the condensation of steam becomes noisy and is accompanied by pressure oscillations. These were observed when the water temperature exceeded about 130°F.

5. Air in the injection pipe does not interfere with condensation of the steam under the conditions of the tests.

6. Shortly after the simulated rupture accident, the pressures in the dry well and in the air space above the pool return to essentially atmospheric pressure.

7. If fission products are carried through the vent pipes, the pressure suppression pool would permit release to its air space of only minute fractions of the solids and halogens.

5. Design Parameters Obtained from Testing

A pressure suppression system must be desinged so that the dry well and the pool containment will withstand the forces to which they would be subjected following the maximum credible operating accident. Phase III of the initial development program developed the values of peak pressures and other parameters to be used in design.

a. Dry Well Pressure

The initial tests showed that the maximum dry well pressure could occur under three different conditions depending upon relative magnitudes of various parameters:

(1) If the vent area is very small compared to rupture flow, a peak pressure (corresponding to dry containment) is reached before any appreciable amount of steam can escape through the vents. For this condition maximum dry well pressure depends upon dry well volume alone.

(2) If the vent area is large enough that condition (1) does not occur, dry well pressure is a maximum when vent flow just balances rupture flow. For this condition dry well pressure depends on the pressure drop in the vents, and dry well volume and vent submergence have no effect.

(3) If the vent area is very large, maximum dry well pressure corresponds to the pressure buildup necessary to force the water out of the vents. For this condition maximum dry well pressure depends on the rate of

c. Dry Well and Internals

The test dry well, shown in Figure 13, is a cylindrical vessel 6.5 feet in diameter containing 230 cubic feet, including the vent pipe down to the water level, which is approximately 1/48th of the corresponding Humboldt volume. Since the Humboldt dry well will operate at a temperature of 150°F, provision was made for heating the Moss Landing dry well vessel to that temperature during some of the tests.

The piping carrying the steam and water from the simulated reactor vessel through the rupture discs is made oversize to direct rupture flow without introducing significant losses. It is terminated inside the dry well with different fittings for different tests. The fittings are shown in Figure 13. Arrangements A to E were used to produce different amounts of water carryover in the steam leaving the dry well and also accelerate or delay the time of air discharge.

d. Vent Piping

The vent piping consisted of 14-inch schedule 30 pipe with one capped tee, one long radius ell, and one short radius ell. The arrangement is shown in Figure 10.

e. Suppression Chamber

The Moss Landing suppression chamber shown in Figure 14 is contained in a tank 12 feet in diameter and 49 feet high. The suppression chamber is approximately trapezoidal in section, being 12 feet across, 2.35 feet on one base and 1.23 feet on the other. It is formed by two flat plates extending the height of the tank. The inner surfaces of the plates above the water pool are covered with ½ inch cork sheet to minimize condensation. The pool depth is 18 feet, the same as Humboldt.

f. Instrumentation

Provision is made for the measurement of pressures and temperatures at the points shown in Figure 9. A light beam oscillograph is used to record transient pressures in the simulated reactor vessel, the dry well, and the suppression chamber. A photograph inside the instrument shelter is shown in Figure 15.

4. Test Results

a. Maximum Credible Operating Accident

A plot of three representative test results for the simulated maximum credible operating accident is shown in Figure 16. Pressure, in pounds per square inch, is plotted against time in seconds for (1) the reactor vessel, (2) the dry well, and (3) the suppression chamber. By the use of internal piping Arrangements A, B, and C, Figure 13, in the test dry well, the following conditions were produced:

With an orifice corresponding to the maximum credible operating accident and the valve from the auxiliary air tank still closed the test was initiated in the usual manner. Five and one half seconds later the quick opening valve was operated. The results are shown in Figure 19. The first part of the pressure suppression curve has the expected initial peak of almost 7 psig, then it dips briefly followed by a slow increase until 5½ seconds after the test has started. For the next 2 seconds it continues but at a greater rate during which about 60% of the air in the auxiliary tank flowed into the dry well. Additional air in the auxiliary tank then discharged more slowly as the dry well pressure fell.

The maximum suppression chamber pressure for this test, which from an air standpoint, is much more severe than Humboldt conditions, was 12.5 psig. This pressure is entirely accounted for by the air carryover indicating prompt steam condensation in spite of the extra flow of air during the condensation period.

f. Suppression Chamber Observations

Although it was demonstrated that air carryover does not interfere with steam condensation, it is of interest to know what has been ovserved in the suppression chamber during some of the test runs.

It has been observed that the sudden release of air from the submerged vent pipe accelerates the pool level upward resulting in water being thrown high in the chamber. It is believed that the mass of water was moved upward by air in the pool. This water acting as a piston compresses the upper air in the suppression chamber until the air from the pool slips through the water which then falls back in the pool. The initial peak in the suppression chamber pressure-time curves is believed to be caused by this phenomena because no peak occurred in tests where the end of the vent pipe was located above the normal pool level.

g. Suppression Pool Temperature and Temperature Rise

In Test No. 26 with an orifice area corresponding to the maximum credible accident, the intitial pool water temperature was about 140°F. This was 60°F higher than the design value of 80°F. The maximum dry well pressure was 8.5 psig indicating satisfactory steam condensation at this pool temperature. No vibration of the suppression chamber occurred, probably due to its mass and the dampening effect of the earth.

The amount of temperature rise in pool water during a test does not appear to have any effect on operation of pressure suppression within the range tested. Average pool temperature rise for different tests corresponding to Humboldt conditions was 29°F; the calculated value is 33°F. The difference between the two is primarily due to heat absorption by air and metal in the test facility which was neglected in the calculation. To show that this heat absorption does not significantly affect pressure suppression, tests with 25% more water in the reactor vessel and 45% less water in the condensing pool resulted in pool temperature rises as high as 58°F without failure to promptly condense the steam.

FILED FEB 14 1986

IN THE UNITED STATES DISTRICT COURT
FOR THE SOUTHERN DISTRICT OF OHIO
WESTERN DIVISION

THE CINCINNATI GAS & ELECTRIC COMPANY, 139 E. Fourth Street, Cincinnati, Ohio 45202; THE DAYTON POWER AND LIGHT COMPANY, Courthouse Plaza, S.W., Dayton, Ohio 45401; and COLUMBUS AND SOUTHERN OHIO ELECTRIC COMPANY, 215 N. Front Street, Columbus, Ohio 43215, Plaintiffs, vs. GENERAL ELECTRIC COMPANY, 3135 Easton Turnpike, Fairfield, Connecticut 06431; SARGENT & LUNDY, 55 East Monroe Street, Chicago, Illinois 60603; and the individual partners of Sargent & Lundy being LOWELL E. ACKMANN, 200 Dover Circle, Palatine, Illinois 60067; WILLIAM A. CHITTENDEN, 339 Sturges Parkway, Elmhurst, Illinois 60126; DAVID C. McCLINTOCK, 92 Hart Road, Barrington Hills, Illinois 60010; WILBERT G. HEGENER, 1320 Hillside Road, Northbrook, Illinois 60062; RICHARD I. GAVIN, 1442 Ridge Road, Northbrook, Illinois 60062; GEORGE C. KUHLMAN, 636 E. Hillside Avenue, Barrington, Illinois 60010, EUGENE V. ABRAHAM, 6016 N. Louise, Chicago, Illinois 60646; ROBERT F. SCHEIBEL, 427 South Donald Avenue, Arlington Heights, Illinois 60004; RICHARD X. FRENCH, 14435 South Dearborn, Riverdale, Illinois 60627; DONALD L. LEONE, 178 Nuttall Road, Riverside, Illinois 60546; ROBERT J. MAZZA, 288 Shenstone Road, Riverside, Illinois 60546; JOHN M. McLAUGHLIN, 230 South Stone Avenue, LaGrange, Illinois 60525;	Case No. C-1-84-0988 J. Spiegel SECOND AMENDED COMPLAINT AND JURY DEMAND SHH-9 Use existing page nos.

HENRY M. SROKA, 1285 E. Westliegh Road, Lake Forest, Illinois 60045; KENNETH T. KOSTAL, 558 North Edgewood, LaGrange Park, Illinois 60525; JOHN A. WERHANE, 2314 Lincoln Park West, Chicago, Illinois 60614; CARMEN M. CHIAPPETTA, 4565 Franklin Avenue, Western Springs, Illinois 60558; PAUL L. WATTELET, 201 Longleaf Drive, Naperville, Illinois 60540; and DONALD E. WOLNIAK, 1212 Elizabeth Avenue, Naperville, Illinois 60540,

Defendants.

CONSOLIDATED WITH

SARGENT & LUNDY,

Plaintiff,

v.

THE CINCINNATI GAS & ELECTRIC COMPANY, et. al.,

Defendants.

Case No. C-1-85-1014

Plaintiffs The Cincinnati Gas & Electric Company, The Dayton Power and Light Company, and Columbus and Southern Ohio Electric Company, by their respective attorneys, for their Complaint against defendants General Electric Company, Sargent & Lundy, and the individual partners of Sargent & Lundy being Lowell E. Ackmann, William A. Chittenden, David C. McClintock, Wilbert G. Hegener, Richard I. Gavin, George C. Kuhlman, Eugene V. Abraham, Robert F. Scheibel, Richard X. French, Donald L. Leone, Robert J. Mazza, John M. McLaughlin, Henry M. Sroka, Kenneth T. Kostal, John A. Werhane, Carmen M. Chiappetta, Paul L. Wattelet, and Donald E. Wolniak, allege as follows:

THE PARTIES

1. Plaintiffs The Cincinnati Gas & Electric Company ("CG&E"), The Dayton Power and Light Company ("DP&L"), and Columbus and Southern Ohio Electric Company ("C&SOE") are utilities engaged in the business of generating and distributing electric power for use by their customers in homes, factories, offices and public areas. Plaintiffs are incorporated and have their principal places of business in Ohio.

2. Defendant General Electric Company ("GE") is incorporated in New York and has its principal place of business in a State other than Ohio. At all relevant times, GE has been engaged, among other things, in the development, design, marketing, construction and the provision of other services in connection with nuclear reactors and nuclear steam supply systems for electric generating plants.

3. Defendant Sargent & Lundy is a general partnership of engineers, none of whose partners is a citizen of Ohio, which was formed for the purpose of carrying on a trade or business in Ohio, which has carried on and presently carries on a trade or business in Ohio, and which holds property in Ohio. S&L is located in Chicago, Illinois and is engaged, among other things, in the provision of professional design engineering and

Plaintiffs' Fraud Allegations

Plaintiffs CG&E, DP&L and C&SOE (the "Owners") further allege as follows:

120. When GE first disclosed to the Zimmer Owners in late 1974 and 1975 the existence and potential design significance of hydrodynamic loads for the Mark II containment at the Zimmer Plant, GE represented to the Owners that those loads had been recently discovered. In fact, the existence of suppression pool hydrodynamic loads was recognized by GE at least as early as 1958 and was a concern to GE engineers and managers in the 1960s and early 1970s. Between 1958 and 1972, GE personnel knew of every hydrodynamic phenomenon that caused loads on pressure suppression containments and equipment. However, in order to promote the sale of its nuclear products in the face of stiff competition, GE concealed its knowledge of the deficiencies inherent in its containment concept for almost two decades and fraudulently induced the Zimmer Owners to buy its products, designs and services and to rely on its expertise. Once GE had firmly entrenched itself in the nuclear business, GE fraudulently disclaimed knowledge of those deficiencies and coerced and fraudulently induced its purchasers into financing a multi-million program to investigate and develop the fundamental technology of pressure suppression.

General Electric's Knowledge of Hydrodynamic Loads During Early Pressure Suppression Tests Between 1958 and 1963

121. In the late 1950s, GE commenced development work on the concept of using pressure suppression to gain a marketing advantage over Westinghouse and other competitors in the nuclear power industry. Specifically, GE hoped that pressure suppression containments would be less expensive to design and construct than the cylindrical or spherical pressure vessels (called "dry containments") then built around GE's boiling water NSSS and other suppliers' pressurized water reactors. By using a pool of water to condense highly pressurized steam released during certain operating conditions or during an accident, thereby preventing large increases in pressure inside the containment, GE hoped to offer utilities smaller and less costly containment structures. At that time, GE knew that such a concept involved difficult and untested assumptions about the reaction of high pressure steam and water and the response of structures to that process.

122. In 1958, GE performed initial small-scale pressure suppression tests at GE's Vallecitos Atomic Power Laboratory. Although the steam which was discharged into the suppression pool appeared to condense, GE witnessed

water hammer effects during the steam quenching process. In particular, GE engineers observed water being repeatedly drawn into the vent pipe and discharged in a pulsating fashion coinciding with an audible water hammer. Distinct bubble formations were expelled from the pipe mouth during the discharge portion of the pulsation cycle. In 1975, GE called this purportedly "new" hydrodynamic phenomenon "chugging." In 1958, GE decided that it would not be safe to conclude that full-scale steam discharges would not create shock waves or forces which could be damaging to structures on the basis of these small-scale tests.

123. GE compiled its observations and conclusions from these initial tests at Vallecitos in an internal report entitled, "GEAP-3013, Preliminary Tests of Direct Water Contact Condensation of Steam" ("GEAP-3013"), dated May 14, 1958. GEAP-3013 was explicitly restricted to use by GE employees only. On information and belief, GEAP 3013 was not made available to the Atomic Energy Commission ("AEC") or the Advisory Committee on Reactor Safeguards ("ACRS"), then the primary nuclear safety regulators for the United States Government. Nor was GEAP-3013 provided to the Zimmer Owners prior to discovery proceedings in this lawsuit.

124. In 1958, GE also proposed to Pacific Gas & Electric Company ("PG&E"), a utility based in Northern California, a program to test and develop the concept of

pressure suppression for possible application to PG&E's proposed nuclear power plant at Humboldt Bay. PG&E accepted GE's proposal and participated with GE in a research program that included a series of tests at the Condensing Test Facility ("CTF") at Moss Landing and the Transient Test Facility at San Jose, California.

125. During the tests at the CTF in 1958 and 1959, GE engineers reported observations of hydrodynamic phenomena that caused the entire test facility to shudder and bang severely, sounding like rapid fire from a rifle. GE also witnessed water hammer effects which, at one point, increased to the point where everything in the vicinity of the tank was shaking wildly. GE engineers recognized that the vibration and banging of the tank were caused by unstable steam condensation which occurred at suppression pool temperatures above 120°F to 130°F. In 1974, GE labeled this purportedly "new" hydrodynamic phenomenon "high temperature steam condensation instability." It is also referred to as the "Wurgassen effect," after a German nuclear facility accident in April 1972 in which the phenomenon tore bolts and ripped open the containment pool liner releasing radioactive water and fission products. During the 1958-59 tests, the shaking became so rough at times that the test tank appeared to be bouncing on its foundation. In fact, during one test run engineers recorded

an earthquake in a control room several hundred yards away. In another run, it was reported that the test facility sank into the ground about two inches.

126. GE recorded its tests in a confidential report entitled, "GEAP-3143, Test Report for the Pressure Suppression Development Program" ("GEAP-3143"), dated April 2, 1959. GEAP-3143 was explicitly limited to use by GE employees only and contained a warning to those employees to handle information concerning pressure suppression with discretion and not to discuss it outside GE. On information and belief, GEAP-3143 was submitted by PG&E to the AEC in April 1959 with the request that it be kept confidential. Besides this transmittal, GEAP-3143 was not made available to anyone besides GE and PG&E not connected with the program, including the Zimmer Owners, prior to discovery proceedings in this lawsuit. Despite observations of the effects of severe hydrodynamic phenomena, the conclusion of GEAP-3143 was that the pressure suppression concept -- to which GE had already devoted substantial engineering, marketing and financial resources -- was a viable method of protecting the public against a nuclear accident.

127. During 1959 and 1960, GE provided test results and other assistance to PG&E during licensing proceedings for the Humboldt Bay Plant. Certain amendments to PG&E's Preliminary Hazards Summary Report, filed with the

AEC by PG&E in 1959 after review and approval by GE, minimized the importance of GE's CTF test observations. Although GE acknowledged that condensation of steam at suppression pool temperatures above 120°F to 130°F was accompanied by considerable shock and vibration, it provided no explanation for this phenomenon and merely specified that the pressure suppression system would be designed to stay below this temperature. Further, in contemporaneous technical papers which GE prepared for certain engineering journals, GE indicated that tank vibration, though severe at high steam flows, was not likely to be a problem. In these public filings and meetings and technical papers, GE omitted any mention of actual tank bouncing, a recorded earthquake and other severe dynamic effects accompanying the steam quenching process.

128. In May and June of 1960, GE conducted pressure suppression tests at Moss Landing using a steam discharge vent sized to represent one of the 48 vertical vent pipes or downcomers planned for the Humboldt Bay containment (the "Humboldt Bay tests"). The suppression pool of the test facility was contained in a cylindrical pressure vessel which was partially buried in the ground. The large tank volume between the circular walls of the vessel and the thin test region that represented a small slice of the actual suppression pool was filled entirely

with concrete in an attempt to minimize any vibrations that would result from hydrodynamic phenomena in the test facility.

129. In the Humboldt Bay tests, GE engineers again observed the effects of powerful and distinct hydrodynamic forces. In particular, technicians filmed a violent upsurge of suppression pool water that occurred as the air leaving the vent pipe expanded rapidly in the pool and threw the mass of water above it upwards at a high velocity. In the early 1970s, GE named this purportedly "new" hydrodynamic phenomenon "pool swell." In fact, in the 1960 tests, the turbulent pool action resulted in water rushing out of the test tank when it was left uncovered during one test run. Pressure sensitive instrumentation recorded a sharp, initial rise in suppression chamber pressure accompanying pool swell.

130. In addition, during the Humboldt Bay tests in 1960, GE personnel again recorded significant pressure fluctuations and water hammer effects during the blowdown of steam into the suppression tank. For example, Charles Robbins, GE's test expert, stood on top of the tank and felt the entire concrete and steel structure shake beneath him. GE barely alluded to the tank shaking in an internal test report, entitled, "GEAP-3596, Tests of a Full Scale 1/48 Segment of the Humboldt Bay Pressure Suppression

Containment" ("GEAP-3596"), dated November 17, 1960, which dismissed the shaking tank as inconsequential. GE did not measure or calculate the structural response of the tank to those forces. On information and belief, GEAP-3596 was not submitted to the AEC or the ACRS, nor made available to anyone besides GE and PG&E not connected with the program when it was written.

131. The Final Hazards Summary Report for the Humboldt Bay Plant, filed with the AEC by PG&E in 1961 after review and approval by GE, obscured the potential significance of the hydrodynamic forces which GE engineers had regularly observed. Pool swell was described, but was dismissed as irrelevant simply because it apparently did not prevent steam from condensing. Further, the report falsely asserted that no tank vibration had occurred in the tests and attributed its absence to the tank's mass and the earth's dampening effect. In contemporaneous technical articles which GE published in certain engineering journals, GE provided only oblique references to the violent movement of water or pool swell and pressure oscillations observed during the Humboldt Bay tests to minimize the significance of such observations.

132. Between 1960 and 1963, GE conceived of and adopted the basic design and configuration of a pressure suppression containment system (later called the "Mark I"

containment) to house the larger boiling water NSSS which was planned by GE and PG&E for the proposed nuclear plant at Bodega Bay, California. The Mark I containment was divided between a drywell which resembled a huge lightbulb and a wetwell (the "torus") which was shaped like a donut and which sat below and around the drywell.

133. During 1962 and 1963, GE participated in pressure suppression tests at Moss Landing of one of 112 downcomers ("single vent tests") and fourteen of 112 downcomers ("multivent tests") planned for the Bodega Bay containment. The single vent tests were conducted in 1962 using the same cylindrical pressure vessel for the suppression pool that was used for the Humboldt Bay tests. Again, in an attempt to minimize any vibrations that would result from hydrodynamic phenomena in the test facility, the test vessel was partly buried in the ground, containing a narrow slot formed by steel plates on the side and bottom to represent a segment of the suppression chamber, and the rest of the tank volume was filled with concrete.

134. In the course of running the Bodega Bay single vent tests, GE and PG&E engineers again found evidence of significant hydrodynamic forces. For example, GE and PG&E recorded the sharp, initial rise in suppression chamber pressure which had accompanied the phenomenon of pool swell in the Humboldt Bay tests. Further, GE measured

significant water level oscillations within the downcomer caused by chugging in several tests. In certain runs, water reached up to five feet inside the downcomer before being expelled violently back into the pool. After reviewing the results from these tests, GE engineers recognized that the dynamic effect of oscillating water inside the downcomer had not been properly assessed. Nevertheless, GE again did not measure or calculate the structural response of the test facility to those forces.

135. The Preliminary Hazards Summary Report for the Bodega Bay Plant, filed with the AEC by PG&E in 1962 after review and approval by GE, again provided only limited information concerning the effects of severe hydrodynamic forces observed during the tests and again attempted to minimize the design significance of pool swell, chugging and other hydrodynamic forces. For instance, without analysis, GE attributed the sharp, initial rise in suppression chamber pressure to the upward movement of pool water observed in the Humboldt Bay tests. Vent water level oscillations due to chugging were not explicitly described or analyzed at all.

136. In 1963, GE and PG&E conducted tests of pressure suppression using multiple vents or downcomers. During those multivent tests, GE and PG&E gathered and analyzed additional evidence of potent hydrodynamic forces

during pressure suppression. Specifically, observers witnessed the test tank jump due to the violent upward surge of water immediately following the simulated pipe breaks. Water level oscillations within the downcomers due to chugging were also recorded. Further, after some test runs, technicians discovered that certain steel structures called baffles, which divided the suppression pool into sections, were bowed and twisted as the result of what GE believed to be a strong hydrodynamic force.

137. The results of the Bodega Bay multivent tests were compiled and analyzed by GE in a series of internal status reports and by PG&E in a report entitled, "Pressure Suppression Tests With Multiple Vents," dated October 25, 1963. None of these reports mentioned observations of tank jumping. Only one report - an internal summary of test results by Charles Robbins, GE's test expert - referenced the baffle damage. These reports were not made available to the AEC or the ACRS or to anyone besides GE or PG&E not connected with the program when they were written. When GE later used PG&E's test report as part of its effort between 1967 and 1969 to assist utilities in justifying to the AEC the deletion of the baffles from the suppression pools of the Mark I and Mark II containments, GE did not disclose to the regulatory authorities its awareness of tank jumping or baffle damage caused by strong hydrodynamic forces during the multivent tests.

General Electric's Concealment
of Hydrodynamic Loads During Development
and Marketing of Its Mark I Pressure
Suppression Concept From 1964 to 1968

138. Upon completion of the Bodega Bay tests, GE commenced marketing of its Mark I containment concept specifications and design. The Mark I design and its functional specifications were based upon the early pressure suppression test results, analytical models derived therefrom and GE decisions to ignore the effects of hydrodynamic loads. Those functional specifications and designs neither revealed nor accommodated the severe hydrodynamic forces that GE knew would be imposed on the Mark I containment. Rather, on information and belief, GE marketed its Mark I products and services by representing that:

(i) the Mark I was thoroughly tested, safe and reliable;

(ii) the early pressure suppression tests confirmed that the pressure suppression concept was technologically sound; and

(iii) further pressure suppression testing was unnecessary.

At the time GE made these representations to prospective purchasers, including the Zimmer Owners, GE knew them to be false or acted with reckless disregard for the truth or falsity of such representations.

139. For example, in an internal report entitled, "Some Thoughts on Condensation in Pressure Suppression Systems" (NUSA 86), dated October 1964, GE's analytical expert, Dr. Fred J. Moody, stated that the process of steam condensation was poorly understood. Moreover, Dr. Moody specifically discussed test observations of large, surging masses of suppression pool water or pool swell and recommended that the phenomenon be further investigated for structural reasons. Dr. Moody also suggested a broad array of tests to develop further knowledge of the dynamic behavior of steam bubbles during condensation. Dr. Moody's report and recommendations were approved by A.P. Bray, the man who later led GE's Projects Department in late 1974 and 1975 when GE first disclosed to the Zimmer Owners the existence and potential design significance of suppression pool hydrodynamic loads. However, in 1975, Bray intentionally misrepresented to the Owners on behalf of GE that hydrodynamic loads had just been discovered and demanded that GE be paid to test and analyze those loads to correct fundamental deficiencies in GE's own pressure suppression system which had been recognized by GE's own experts more than a decade before.

140. Moreover, shortly after the conclusion of the Bodega Bay tests, top-level GE managers issued a comprehensive set of Nuclear Safety Criteria. The criteria

were intended, in part, to govern the testing, development and arrangement of new pressure suppression containment configurations. Nuclear Safety Criterion No. 2.13, which GE promulgated in 1964, declared that the current state of pressure suppression technology did not permit prudent design based on analysis alone. This fundamental safety criterion was based on GE's internal recognition that there was insufficient understanding of the process of pressure suppression. Thus, the criterion required that design conditions and geometrical specifications (including the number, size and location of vent pipes and the layout of the suppression chamber) of containment configurations adhere to conditions established during the early pressure suppression tests.

General Electric's Concealment of Hydrodynamic Loads During Sales and Contractual Negotiations with the Zimmer Owners from 1968 to 1972

141. In 1968, GE responded to specifications H-2210 and H-2211 by submitting to the Zimmer Owners a proposal to provide a boiling water NSSS and related services and equipment. GE's proposal initially coupled its NSSS with the pressure suppression containment design then known as the torus/lightbulb configuration and later called the Mark I containment. In its contract proposals and marketing brochures and in contemporaneous meetings,

technical papers and correspondence, GE represented to the Owners that the torus/lightbulb containment was adequately tested, technically sound, safe and reliable. For example, GE provided the Owners with a marketing brochure entitled, "GE/BWR - For Safe, Simple, Dependable, Flexible Power Generation," which represented that the arrangement and design conditions of the torus/lightbulb containment had been substantiated by extensive testing and development programs and had been specified in a number of GE boiling water systems then on order and under construction.

142. Between 1965 and 1968, in an effort to develop a product that would make GE more competitive in the marketplace, GE undertook a study to determine the feasibility of designing and constructing a new containment configuration. By March 1968, GE concluded that an arrangement which placed the drywell directly over the wetwell (then known as the "over-under" design and later called the Mark II containment) might offer significant cost savings for utilities and containment designers in the balance of plant area. Specifically, GE found that the simplified design of the vent or downcomer system and the possible use of prestressed or reinforced concrete (rather than steel) made the Mark II easier to construct and less expensive than the torus/lightbulb configuration. GE believed that such cost savings were essential to its

efforts to compete with suppliers of pressurized water reactors and to market and sell to utilities its boiling water NSSS and that delay in introducing the Mark II containment concept would effectively cripple GE in the marketplace.

143. At the time GE offered concepts, specifications and designs for the Mark II containment to prospective purchasers of GE's boiling water NSSS, GE knew that the design of the Mark II containment departed from the configuration of the Mark I containment. Without any testing whatsoever and without any advance in fundamental understanding of the phenomena associated with steam condensation, GE managers and engineers approved of significant changes in GE's boiling water NSSS and in the structures and geometries of pressure suppression containments, including, but not limited to, the following:

(i) elimination of paired sets of vent pipes or downcomers in the suppression pool;

(ii) selection of long cantilevered downcomers more vulnerable to dynamic and vibratory forces;

(iii) decrease in the number of total downcomers in the suppression pool;

(iv) use of arbitrary criteria for determining the spacing between downcomers;

(v) deletion of the requirement for baffles in the suppression pool;

(vi) increase in the maximum allowable suppression pool temperature;

(vii) elimination of the toroidal shape of the suppression chamber;

(viii) introduction of a pedestal to hold the GE nuclear reactor high above the suppression pool, making the reactor more vulnerable to dynamic and vibratory forces;

(ix) the use of a basemat coupled to the pedestal and the containment itself; and

(x) increase in the size and power rating of the GE boiling water NSSS.

GE adopted each of these changes without prior thorough study and testing and in violation of GE's Nuclear Safety Criteria, including Criterion No. 2.13.

144. Moreover, Charles Robbins, the man who had run GE's pressure suppression development program between 1957 and 1963, warned internally in 1967 that GE's existing test data did not provide a complete understanding of the steam condensation process for the Mark I, let alone for the new Mark II design, and that, because extrapolation from such data was not prudent, geometrical similarity between the test facility configurations and any new containment

designs was essential. Further, in separate reports, other GE analysts warned that the height of the GE nuclear reactor made it particularly susceptible to damage from dynamic and vibratory forces. GE performed no tests of its Mark II containment prior to including it in its product line.

145. Despite these internal warnings and in direct contravention of GE's Nuclear Safety Criteria, in 1968, GE included the Mark II containment as part of its product package in an effort to convince prospective customers, including the Zimmer Owners, to purchase GE's boiling water NSSS.

146. During 1968 and 1969, GE submitted an amended proposal to the Owners to provide its most recent boiling water NSSS (the "Zimmer Class" or "BWR-5" NSSS) and related technical and supervisory services and coupled that proposal with conceptual and scope designs for the Mark II containment. In contract proposals and marketing brochures and in contemporaneous meetings, technical papers and correspondence, GE made several material misrepresentations to the Zimmer Owners and to S&L on behalf of the Owners regarding the safety, cost, and reliability of the Mark II containment concept, each of which was based in whole or in part upon information which was known to be misleading or incorrect at the time the statements were made, including, without limitation, the following:

(1) In October 1968, GE sales engineer R.W. Snyder recommended that the Zimmer Owners carefully consider the advantages of utilizing a concrete pressure suppression containment. Snyder stated that in GE's Mark II design the suppression pool was placed directly below the drywell and the torus was eliminated. Snyder represented that the same experimental data developed for the current design pressure suppression containment system (the Mark I) had been used in designs for concrete containments, resulting in no AEC licensing problems. Further, Snyder indicated that the Owners could expect significant savings in construction costs.

(2) In November 1968, GE proposal engineer Ted Brown made a presentation to S&L on behalf of the Zimmer Owners in which he stated that the Mark II containment had economic and safety advantages over other containment systems offered by GE's competitors.

(3) At the November 1968 meeting, Brown also presented to S&L on behalf of the Zimmer Owners a technical paper prepared by D.R. Miller, GE's containment specialist, entitled, "Pressure Suppression Containment Design - Current State of the Art," which was later published in the Journal of Engineering for Power (ASME Paper No. 68-WA/NE-1) in January 1969. Miller's paper reported that GE had developed an analytical model which purportedly could

reproduce accurately the Humboldt and Bodega Bay pressure suppression test data. According to Miller, these analytical tools allowed the containment designer to make major changes in containment design without the need for additional testing and had already led to development of the concrete Mark II containment. Further, Miller represented that sufficient information had been provided in his article to permit the formulation of pressure suppression designs which would insure proper and safe functioning of the containment system during a major accident.

(4) In sales meetings between October 1968 and August 1969, GE sales engineers, including C.W. Billings and R.W. Snyder, represented to the Zimmer Owners and to S&L on behalf of the Owners that the pressure supression concept had been validated by tests and that the Mark II containment was better than the Mark I and was an improvement in the state of the art for containment technology.

(5) Between October 1968 and August 1969, GE provided the Zimmer Owners and S&L on behalf of the Owners with a copy of its marketing brochure entitled "Another Step Forward In the General Electric Boiling Water Reactor Design." Prominently featured both on the cover and inside the brochure was the Mark II containment. GE stated that pressure suppression remained GE's recommended system and that the most significant development in this area had been

the evolution of the much simpler containment vessel shape embodied in the Mark II design. GE touted the economic advantages of the Mark II which included a reduction in containment cost, a more compact reactor building and an improved construction schedule. Further, GE referenced D.R. Miller's "State of the Art" paper and reminded customers of the advances made by GE in theoretical design aspects of pressure suppression in recent years and indicated that analysis of the consequences of a loss-of-coolant accident was well-established.

147. GE knew and intended that the Zimmer Owners would rely on GE's misrepresentations regarding the safety, cost, and reliability of its Mark II containment in deciding whether or not to purchase GE's boiling water NSSS, related technical and supervisory services and certain containment design information. For example, on August 20, 1969, shortly before the Owners sent GE a letter of intent, J. Williams, GE's Manager of Power Reactor Sales, advised J.R. Birle, GE's Manager of Marketing, that S&L had estimated that GE's Mark II containment was less than the cost of constructing a conventional dry containment and indicated to Birle the importance of this factor in GE's efforts to win the Zimmer contract.

148. In addition, and on information and belief as to S&L, during discussions and in communications with the

Zimmer Owners and with S&L on behalf of the Owners in 1968 and 1969 concerning GE's amended proposal to the Zimmer Owners, GE omitted to communicate the following material statements of fact:

(i) that hydrodynamic forces had caused significant structural vibrations during early tests of the pressure suppression concept;

(ii) that the BWR-5 NSSS and the Mark II containment were particularly susceptible to damage from dynamic forces;

(iii) that the Mark II containment configuration was a substantial departure from the Mark I configuration;

(iv) that GE had never performed any test of the functioning of pressure suppression during release of steam through safety valves, relief valves, or safety/relief valves;

(v) that GE had chosen to reach conclusions about the safety of pressure suppression without enlisting the expert assistance of qualified structural engineers or testing the structural response of the proposed containment; and

(vi) that GE had itself concluded that knowledge of the process of steam condensation was inadequate to allow any deviation from tested parameters.

149. In August 1969, the Owners, in reliance upon all of GE's statements, omissions, and reports and all inferences reasonably drawn therefrom regarding the safety, cost and reliability of the BWR-5 NSSS and Mark II containment, executed a letter of intent to enter into a contract with GE to purchase a boiling water NSSS, related technical and supervisory services and certain containment design information.

150. Beginning in 1969, under its contract, and in accordance with professional and industry practice and its own policies, GE furnished to the Owners and S&L on behalf of the Owners certain engineering requirements for and descriptive drawings of the Mark II containment and represented that GE's information would be sufficient to enable S&L to complete the design of the Zimmer containment.

151. For example, in November 1969, A.W. Thorson, GE's Manager of Project Engineering, transmitted to S&L (with a copy of the cover letter to the Zimmer Owners) certain GE drawings relating to the reactor building, including the reactor system outline and the reference Mark II containment definition, which Thorson described as

completely adequate for containment design purposes. In fact, GE knew or recklessly disregarded the substantial risk that those drawings and other containment information supplied by GE would not enable the Mark II containment to be designed to withstand safely the hydrodynamic loads exerted upon it by the BWR-5 NSSS during certain operating and accident conditions.

152. In 1969 and 1970, as the contract between GE and the Zimmer Owners was being negotiated, GE resumed tests to develop pressure suppression technology and to supplement the Humboldt and Bodega Bay test results under the auspices of its Nuclear Safety Development Department ("NSD"). Data gathered from these experiments and later tests laid the groundwork for design of a new containment configuration, later called the Mark III containment. In these small-scale tests, GE engineers observed the effects of severe hydrodynamic forces like those which it had witnessed in test runs at the CTF during 1958 and 1959.

153. In particular, GE NSD engineers observed during certain 1969 tests that the suppression tank jumped off of the ground as the test was initiated and that loud banging noises accompanied the blowdown of steam and water into the suppression pool. GE then carried out additional tests to measure pressure fluctuations and to observe the tank vibrations. In these tests, GE engineers again observed at least three distinct hydrodynamic phenomena:

(1) In several tests, the suppression tank jumped upwards just as air and water cleared the downcomer. During one test series, one of the feet holding down the suppression tank jumped upwards and came to rest on top of a five inch guide bolt. GE concluded that the forces causing the jumping action were associated with steam condensation which could occur during operation of relief valves piped directly to the suppression pool.

(2) Discharges of steam and water through downcomers produced pool pressure oscillations in certain tests which caused noise and vibration effects. GE engineers concluded that the vibrations were associated with certain mass flow velocities encountered during steam condensation, and that this was confirmed by the severe pressure fluctuations reported during the CTF tests in 1958 and 1959, and other investigations. GE decided that there was insufficient data available to permit drawing conclusions with regard to plant design as affected by the vibration phenomena.

(3) In certain tests, chugging was observed during steam flows at low velocities near the end of a simulated accident. GE concluded that chugging occurred when the rate of condensation exceeded the steam flow rate. At that point, the vapor bubble rapidly collapsed causing the process of steam condensation to move back inside the

downcomer pipe until the pressure increased sufficiently to force the water back into the pool. In several test runs, GE observed that chugging was accompanied by sharp steam hammer knocks and large pressure and flow fluctuations within the downcomer and that further investigation was required.

154. GE NSD compiled its observations and conclusions in an internal report entitled, "NEDM-13036-1, Pressure Suppression Pool Investigations, Report #1" ("NEDM-13036-1"), dated July 1970. NEDM-13036-1 was restricted to the use of GE employees only. On information and belief, NEDM-13036-1 was not then made available to the AEC or the ACRS. Nor was NEDM-13036-1 provided to the Zimmer Owners prior to discovery proceedings in this lawsuit.

155. In NEDM-13036-1, GE's safety department recognized the significance of chugging for containment design, but concluded that thorough evaluation of the phenomenon was impossible without further tests. GE NSD engineers concluded that additional pressure suppression testing with better instrumentation was needed because tank vibration and chugging could have serious design implications for containments then on order and under construction. Nevertheless, GE failed to conduct any pressure suppression tests of the Mark II containment

configuration or inform the Mark II utilities, including the Zimmer Owners, of the results of the tests GE had conducted. Instead, in contract negotiations between GE and the Zimmer Owners which took place after the issuance of NEDM-13036-1 and in contemporaneous meetings, technical papers, licensing presentations and correspondence, GE continued to adhere to its representations to the Owners that its NSSS and the Mark II containment were adequately tested, technically sound, safe and reliable.

156. In late July 1970, shortly after GE internally issued NEDM-13036-1, D.R. Miller, GE's containment specialist, appeared before the AEC on behalf of the Zimmer Owners to address containment licensing issues. Despite the abundant evidence of powerful hydrodynamic forces and their serious design implications described internally in NEDM-13036-1 and in all prior test reports, Miller represented to the AEC and the Zimmer Owners that the response of the Mark II containment to a LOCA was adequately and safely predicted by GE's containment analytical models.

157. Throughout the period between 1970 and 1972, GE continued to provide extensive licensing assistance, much of it bearing on containment issues, to the Zimmer Owners in their attempts to obtain a construction permit for the Zimmer Plant from the AEC. In the course of providing such

licensing assistance, GE made several representations to the Owners and to the AEC with respect to the capability of the Mark II containment to withstand the energy released to the containment during normal operations and accidents.

158. In particular, in 1970, the AEC expressed concern about possible vibration of the downcomers in the Mark II containment. The AEC asked whether the downcomers could withstand water hammer and other dynamic events that could occur during an accident. In response to such questions, GE provided information to the Owners which ignored GE's own conclusions that vibration of the downcomers due to steam bubble collapse during chugging and other hydrodynamic forces was a potential design concern. GE indicated to the Owners and the AEC that bracing of the downcomers was unnecessary because suppression pool water surrounding the submerged portion of the downcomers would dampen the lateral movement of the downcomers during an accident. Further, without any testing, GE stated that water hammer effects associated with condensing steam were localized in nature, geometry dependent and unimportant. In fact, GE had already decided that a system of downcomer bracing might be required and was chiefly concerned with how to "dump" the problem into the laps of its Mark II customers, including the Zimmer Owners, without imperiling the commercial viability of the pressure suppression concept.

159. Between 1970 and 1972, GE carried out a number of pressure suppression tests for possible application to a third generation of pressure suppression containment design, the Mark III containment. During those tests, GE engineers again witnessed the effects of violent forces at work in the suppression pool, including surging pool water and chugging. GE observed that the violence and rapidity of the chugging phenomenon could be likened to a rapidly recurring water hammer. GE knew that further study of these hydrodynamic forces was necessary prior to final design and regulatory approval of the Mark III containment.

160. Moreover, based on its previous small-scale tests and experiments conducted in 1969 and 1970 and reported in NEDM-13036-1, GE knew that chugging would occur in the Mark II containment and that its significance could not be properly evaluated absent new tests. Nevertheless, in contract negotiations between GE and the Zimmer Owners lasting until May 1972 and in contemporaneous meetings, technical papers, licensing presentations and correspondence, GE repeatedly represented to the Owners that its BWR-5 NSSS and the Mark II containment were adequately tested, safe and reliable.

161. Between 1971 and 1972, GE observed severe hydrodynamic forces resulting from the operation of certain safety or auxiliary systems, including the reactor core

isolation cooling ("RCIC") system, in suppression pools of operating GE NSSS plants with Mark I containments. These safety systems were designed to discharge exhaust steam into the torus or suppression pool. Specifically, GE witnessed "thunder" in the pool, water slug carry-over and pipe and torus vibration and movement.

162. GE knew that similar hydrodynamic forces would be caused by steam discharges during routine operations or in the event of an accident. In fact, as early as 1966, GE had debated internally the merits of designing such safety systems to vent exhaust steam into the suppression pool in light of the tank banging and shuddering observed in the CTF tests at Moss Landing during 1958 and 1959. Nevertheless, without disclosing that the forces generated by operation of the safety systems were part of a larger problem relating to the adequacy of the pressure suppression system itself, GE represented to the Zimmer Owners and to S&L on behalf of the Owners that operation of the safety systems, including the RCIC system, presented a minor and discrete problem requiring only simple design changes.

163. In 1971 and 1972, discharges of steam through safety/relief valves ("SRVs") caused structural damage in several operating GE plants with pressure suppression containments. Like the technicians who had observed baffle

damage due to strong hydrodynamic forces in the Bodega Bay multivent tests in 1963, plant operators found that baffles in the suppression pool had been torn from their supports. When GE learned about these baffle dislocations shortly after they occurred, GE knew that SRV discharges created significant pressure disturbances throughout the suppression pool as a result of the clearing of highly compressed air from the SRV discharge line (known as "SRV discharge loads" or "air-water clearing loads") and that such hydrodynamic loads would be exerted upon Mark II containments.

164. Moreover, GE knew that only accident conditions resulting from a major steam pipe break had been tested in GE's pressure suppression tests between 1957 and 1963 and again during 1969 and 1970. GE realized that the number of expected routine and extended SRV discharges into suppression pools had increased with the construction and operations of larger nuclear plants like the Zimmer Plant. Nevertheless, in contract negotiations between GE and the Zimmer Owners lasting until May 1972 and in contemporaneous meetings, technical papers, licensing presentations and correspondence, GE concealed from the Owners its knowledge of SRV-related testing deficiencies and its internal concerns about baffle dislocations.

165. In April 1972, an initial series of SRV tests at a BWR nuclear power plant in Wurgassen, West Germany

indicated that potent SRV discharge loads were exerted upon the pressure suppression containment. When they inspected the suppression pool, German engineers discovered indentations in the steel pool bottom, some with dimensions up to nearly half a foot, and decided to reinforce the pool liner with beams bolted directly to the bottom shell.

166. Shortly thereafter in April 1972, during another SRV test, an SRV could not be reclosed which resulted in the continual venting of pressurized steam into the suppression pool. In the course of the steam blow down, steam condensation became unstable when the pool temperature reached between 150°F and 170°F, thereby producing tremendous vibration in the suppression chamber. In fact, like the shaking and movement of the CTF test facility in tests conducted during 1958 and 1959, the whole Wurgassen containment sphere was in motion with waves appearing on the outside of the shell; the suppression pool bounced off of and onto concrete supports; and considerable vibration was felt in the control room. The vibrations ripped the pool liner away from the newly installed supporting structure and resulted in the leakage of radioactive suppression pool water into the control rod drive room under the BWR.

167. By as early as April 21, 1972, top-level GE managers and engineers, including A.P. Bray, were informed about the severe accident at the Wurgassen Plant. GE knew

that unstable condensation at high temperatures or the "Wurgassen effect" had been observed in the CTF tests between 1958 and 1959 and again in small-scale tests between 1969 and 1970. Further, GE knew or recklessly disregarded the substantial risk that the Wurgassen effect could cause structural damage in the Mark II containment at the Zimmer Plant. Nevertheless, in contract negotiations between GE and the Zimmer Owners lasting until May 1972 and in contemporaneous meetings, technical papers, licensing presentations and correspondence, GE concealed from the Owners its knowledge of the tremendous vibrations which ripped open the containment at the Wurgassen Plant. Moreover, GE failed to disclose to the Owners until late 1974 the occurrence of the Wurgassen accident and any of the hydrodynamic loads encountered there.

168. In May 1972, after prolonged contractual negotiations in which GE never informed the Zimmer Owners of its observations and conclusions concerning the existence and the design significance of suppression pool hydrodynamic loads, the contract between GE and the Zimmer Owners was fully executed.

General Electric's Concealment of Hydrodynamic Loads During the Course of Providing Containment Design Information to the Zimmer Owners from 1972 to 1975

169. Between May 1972 and 1975, GE continued to provide the Zimmer Owners and S&L on behalf of the Owners certain engineering requirements for and advice regarding the design of the Mark II containment at the Zimmer Plant. During this same period, GE continued to observe and gather evidence of severe hydrodynamic forces as the result of new incidents at operating BWR plants and new GE pressure suppression test programs. However, partly to guard against a loss of new sales of BWR systems to prospective purchasers, including a second nuclear unit to the Zimmer Owners, GE failed to disclose to the Owners and, in fact, obscured its knowledge of the existence and design significance of hydrodynamic loads until late 1974 and 1975.

170. Between May 1972 and 1975, GE understood but failed to disclose to the Zimmer Owners the design significance of suppression pool hydrodynamic loads observed in several incidents at operating BWR plants and new pressure suppression tests, including, but not limited to, the following:

(1) In May 1972, during a series of SRV tests at Quad Cities Unit 2, a GE BWR plant with a Mark I containment near Moline, Illinois, SRV discharge loads caused the torus to bulge and vibrate, thereby shearing off several bolts

supporting the ring or suction header which encircled and was attached to the outside of the torus. The torus ring header is part of the essential safety system that suctions and pumps water from the suppression pool to the reactor core in the event of an accident. In the Quad Cities incident, nearly a quarter of the massive ring header was dislodged by the SRV discharge loads and accompanying vibration. GE engineers participated in the tests at Quad Cities Unit 2 and witnessed the May 1972 incident.

(2) In July 1972, less than three months after the Wurgassen accident, start-up tests were conducted at the KKM Plant, a GE BWR plant with a Mark I containment in Muhleberg, Switzerland. During relief valve tests, engineers witnessed substantial hydrodynamic loads due to high temperature steam condensation instability. After a second valve was opened during the tests, the noise and vibration in the KKM torus increased dramatically until a test observer became frightened and ordered the testing stopped. When the torus was inspected by plant technicians and GE engineers, they found a broken one-inch pipe, evidence of torus movement and displaced gratings around and inside the torus.

(3) In the summer of 1972, GE hurriedly put together a series of generic SRV tests (the "Quad Cities tests") to demonstrate, in GE's own words, that there was

"no problem" with GE pressure suppression. GE engineers realized that accurate and meaningful analyses of SRV discharge forces would be impossible without pressure data from actual steam blow down tests. The SRV tests were conducted at Quad Cities Unit 2 in October 1972 and a report entitled, "NEDO-10859, Steam Vent Clearing Phenomena and Structural Response of the BWR Torus (Mark I Containment)" ("NEDO-10859") which analyzed some, but not all, of the test results, was completed by April 1973. The Quad Cities tests were deficient in several respects, in part, owing to the failure of 8 of 14 strain gauges and wide scatter in the test data, which made the results unsuitable for verification of the torus structural model and which provided the basis for the AEC's rejection in 1974 of NEDO-10859 and the torus structural model.

(4) In late 1972, GE continued to carry out small-scale pressure suppression tests related to its development of the Mark III containment. During these tests, GE engineers again witnessed severe hydrodynamic phenomena reminiscent of the forces observed in the early pressure suppression tests. For example, in December 1972, GE technicians saw severe splashing as water was blown clear of the test tank and heard "booming" sounds when steam was discharged into the test pool.

(5) In April and May of 1973, GE conducted additional pressure suppression tests dedicated to investigation of the pool swell phenomenon. Based on these tests, by August 1973, GE recognized that pool swell would undoubtedly occur in the Mark III containment and that more testing was necessary, including small-scale tests to delineate more fully the effects of multiple vents on pool swell and large-scale tests to confirm the scaling parameters established in the small-scale tests. Further, in an internal design review conducted in August 1973, GE concluded that its mathematical model would have to account for pool swell, vent clearing, chugging, steam condensation and other dynamic effects for satisfactory design and licensing. Nevertheless, on information and belief, GE did not provide the AEC or the ACRS with any of its pool swell test results until December 1973, and failed to inform the AEC or the ACRS of the conclusions reached by its internal review.

(6) In May and June 1973, during tests of certain safety or auxiliary systems, including the RCIC system, at Fukushima Unit 2, a GE BWR plant with a Mark I containment in Japan, plant operators observed severe water hammer due to the collapse of large steam bubbles when steam was exhausted into the torus suppression pool. In fact, in tests to verify the problem at Fukushima Unit 2, GE

engineers found that the water hammer was more than severe: "[I]t was thunder accompanied by violent pipe movement and even more violent personnel movement leaving the torus area!"

(7) In 1973, during a series of SRV tests at Browns Ferry Unit 1, a GE BWR plant with a Mark I containment, SRV discharge loads caused excessive torus wall and ring header vibration. Shortly after the tests, GE assisted plant operators in choosing certain corrective actions, including the jacking and shimming of the torus support cradle feet, so as not to delay further the startup of Browns Ferry Unit 1.

(8) Between 1972 and early 1974, GE actively followed the progress of certain large-scale pressure suppression tests using vertical vents or downcomers conducted at the Marviken Reactor Station in Studsvik, Sweden. Test results indicated significant structural vibrations, not unlike those encountered at the CTF between 1958 and 1959, during steam blow downs into the Marviken suppression pool.

171. Between May 1972 and late 1974 and 1975, GE also knew but did not inform the Zimmer Owners that hydrodynamic loads had never been accounted for in the design of the Mark II containment. For instance, in November 1972, GE licensing engineer A.J. Levine advised

top-level GE managers, including A.P. Bray, that because hydrodynamic effects had not been considered in any pressure suppression containment design, it would be necessary to review the Mark II containment to ensure its capability for withstanding safely these additional effects. Moreover, in January and March 1973, GE Design Safety Review Teams submitted reports to GE managers A.P. Bray and A. Rubio which concluded that no dynamic effects, regardless of source, had been considered in any suppression containment design. Further, the reports recommended that analyses of the dynamic effects of relief valve blow downs should be made for all containment designs, including the Mark II containment.

172. Throughout the period between May 1972 and 1975, GE continued to provide containment design information to the Zimmer Owners and to S&L on behalf of the Owners. Despite abundant evidence of hydrodynamic loads and their design significance for all pressure suppression containment designs for more than fifteen years, GE did not disclose to the Zimmer Owners any information concerning the hydrodynamic loads or their design significance prior to late 1974 and 1975. On information and belief, on only two occasions did GE provide materials to the Zimmer Owners which alluded to the matter of hydrodynamic loads and on both occasions GE falsely indicated that the data was not

applicable to the Mark II containment at the Zimmer Plant. In June 1972, GE transmitted to S&L a copy of its report entitled, "SCER-259, Design Guide for the Reactor Vessel Relief Valve Piping," which provided some information concerning the arrangement of relief valve piping in pressure suppression containments. In May 1973, GE transmitted to the Zimmer Owners a copy of NEDO-10859, GE's report on some of the Quad Cities tests conducted in October 1972, which was later rejected in 1974 by the AEC as a design basis for any domestic nuclear power plant. In each instance, GE's misleading transmittal letters indicated that much of the data and recommendations contained in the reports were irrelevant to the design of the Mark II containment at the Zimmer Plant.

General Electric's Concealment of Its Early Knowledge of Hydrodynamic Loads When It Coerced the Zimmer Owners to Sign Change Orders in 1975 and 1976

173. Rather than disclosing test results which had demonstrated the existence and design significance of severe hydrodynamic loads for more than fifteen years, GE deliberately and willfully in 1975 and thereafter sought to force owners of GE BWR plants with Mark II containments, including the Zimmer Owners, to pay for the costs of experimental and development work, as well as design changes required by analysis of hydrodynamic loads. To that end, GE

repeatedly misrepresented in several communications to the Zimmer Owners and regulatory authorities that such hydrodynamic forces had been discovered only in the Mark III tests in 1973 and 1974 and as the result of recent incidents involving structural damage at operating plants.

174. During late 1974 and 1975, shortly after the AEC's rejection of GE's report on the Quad Cities tests, GE released to the Zimmer Owners and to S&L on behalf of the Owners information and analyses which indicated that condensation of steam in the suppression pool would cause certain hydrodynamic loads of potential but undetermined design significance.

175. By early 1975, GE adopted a corporate policy designed to address concerns over suppression pool hydrodynamic loading conditions without cost to GE. Put simply, that policy shifted responsibility for solving the problem to GE's customers, including the Zimmer Owners, on the false pretense that hydrodynamic loads were newly discovered and previously unknown to GE.

176. Throughout 1975, GE implemented its corporate strategy of shifting responsibility for handling the problem of suppression pool hydrodynamic loads. For example, on April 9, 1975, A.P. Bray, the Manager of GE's Projects Department, wrote the Nuclear Regulatory Commission ("NRC") to provide some clarification regarding concerns with

pressure suppression containment designs. Bray represented that hydrodynamic phenomena, such as water movement effects due to vent clearing actions, were first diagnosed in the Mark III tests. Further, despite the abundant containment design information GE had provided to the Zimmer Owners, GE licensing engineers told the NRC in early 1975 that GE had no contractual responsibility for Mark II containment design. On October 28, 1975, as part of his effort to coerce utilities to fund GE's new development work, Bray wrote the Mark II utility executives, including the Zimmer Owners, and falsely asserted that the need for an evaluation of Mark II containment designs was brought about by the "discovery" of certain dynamic phenomena, primarily effects resulting from the purging of air through and from discharge vents into the suppression pool, which was "not available" when the original Mark II containment designs were completed.

177. In reliance on GE's false representations that hydrodynamic loads were newly discovered and easily accommodated, the Zimmer Owners shared the cost of programs recommended by GE to study these phenomena. By 1984, when the Zimmer Plant was terminated as a nuclear facility, those programs had gradually expanded to address virtually every fundamental aspect of pressure suppression, had cost millions of dollars to utilities with pressure suppression

containments and had led to design and construction costs at the Zimmer Plant that exceeded $360 million.

COUNT XVI

(Against GE for Fraud)

178. Plaintiffs CG&E, DP&L and C&SOE repeat and reallege paragraphs 1 through 41, 43 through 49, 54 through 55, 60 through 64, 67, 72 through 77, 80 through 81, 85 through 86, 91 through 92 and 120 through 177 of the Second Amended Complaint.

179. In GE's contract proposals and in contemporaneous meetings, technical papers, marketing brochures and correspondence between 1967 and 1969, GE knowingly misrepresented or represented with utter disregard and recklessness for truth or falsity to the Zimmer Owners that its BWR-5 NSSS and the Mark II pressure suppression containment were adequately tested, technically sound, reliable and capable of meeting all regulatory requirements and of operating in a safe manner under all conditions.

180. At the time it proposed to furnish a BWR-5 NSSS housed in a Mark II pressure suppression containment, GE knew but failed to disclose, despite its duty to do so, that:

a) no dynamic effects, regardless of source, had been considered in any pressure suppression containment design, including the Mark II containment;

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

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