UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter	)
LONG ISLAND LIGHTING COMPANY	)
	) Docket No. 50-322 (O.L.)
(Shoreham Nuclear Power Station, Unit 1)	)

PREPARED DIRECT TESTIMONY OF
DALE G. BRIDENBAUGH
ON BEHALF OF SUFFOLK COUNTY

REGARDING

SUFFOLK COUNTY CONTENTION 21

MARK II CONTAINMENT

May 25, 1982

SUMMARY OUTLINE OF SUFFOLK COUNTY
CONTENTION 21

The origin of the design deficiencies potentially affecting the Shoreham containment design (the Mark II) are well known and well documented in the industry literature.

The seriousness of the Mark II problem is well evidenced by the fact that investigation of the basic problems has been underway for eight years (1974-1982) but the program is still not fully completed and implemented.

The basic concerns are threefold:

- The program is still not completed; therefore how can it be demonstrated that all necessary modifications have been identified and implemented?
- LILCO, being in a quasi-lead plant position, has selected a combination of acceptance criteria for the Shoreham evaluation. Has this complex design and evaluation program introduced weaknesses into the Shoreham design and/or review process?
- The work carried out in investigation of the Mark II problems has been performed by numerous organizations, both with and without official status, over a period of eight years in at least five countries.

Has all of the work and data been controlled and reviewed in a manner consistent with the appropriate quality regulations?

LILCO and the Staff have not satisfactorily demonstrated that these deficiencies have not affected the capability of the containment to withstand the LOCA and transient conditions necessary for operating license approval.

Attachments:

1. Portions of NUREG-0808.
2. Background of the Mark II Design, D. G. Bridenbaugh.

PREPARED DIRECT TESTIMONY OF
DALE G. BRIDENBAUGH
REGARDING SUFFOLK COUNTY CONTENTION 21

I. INTRODUCTION

1. My name is Dale G. Bridenbaugh. A statement of my qualifications has been separately provided to this Board. In that qualification summary is one managerial position that is of particular relevance to this contention. During my assignment at General Electric's Nuclear Energy Division as Manager, Performance Evaluation and Improvement (1973-1976), I also served as Manager of the Mark I Containment Reassessment Program. This task began in early 1975 when the hydrodynamic and SRV load problems affecting all of the GE containment designs surfaced as a significant safety issue. I was asked to develop a plan for GE's response to the request by 16 Mark I owning utilities for support in resolution of this problem affecting 24 BWR's in the United States. This first response grew into the BWR Owners Group Short Term and Long Term Programs. My responsibilities in this included project management of the interfaces with the utilities, the NRC, EPRI, Bechtel, Teledyne, and other consultants. Since many of the Mark I and Mark II problems were commonly based, I worked closely with the Mark II

project manager and was kept appraised of the activities of that program. I left GE in February of 1976, but have kept in touch with the containment program work since then through my consulting activities.

II. STATEMENT OF CONTENTION

2. The purpose of this testimony is to address Suffolk County Contention 21 as admitted by the Board as follows:

> LILCO and the NRC Staff have not adequately demonstrated that Shoreham's primary containment, reactor pressure vessel supporting structure and attached and associated safety-related equipment meet the requirements of 10 CFR 50, Appendix A, GDC 4, 16, 50, 51 and 52. The specific concerns are as follows:
>
> (a) Forces generated during the suppression pool LOCA dynamics have not been completely and adequately determined and taken into account. Of the numerous Mark II Containment loads assessed on a generic and on a plant unique basis under the Mark II reassessment program underway for the past six years, several LOCA forces have not yet been shown to have been suitably handled in the design of the structures, systems and components important to plant safety. Included in this category are the forces due to Steam Condensation Downcomer Lateral Loads. (Loads I.B.1.a & b in Table 6-1, NUREG-0420, Supp. No. 1), Steam Condensation Oscillation Loads (Loads I.B.2.a in Table 6-1, NUREG-0420, Supp. No. 1), and Steam Condensation Chugging Loads (Loads I.B.2.c in Table 6-1, NUREG-0420, Supp. No. 1).
>
> (b) Forces generated during safety relief valve (SRV) actuation, continuing SRV blowdown, and those due to suppression pool heatup resulting from such

extended blowdowns have not been demonstrated to be adequately accommodated. Concern specifically remains for the Quencher Air Clearing Loads (Loads II.B in Table 6-1, NUREG-0420, Supp. No. 1), Steam Condensation Submerged Drag Loads (Loads III.C in Table 6-1, NUREG-0420, Supp. No. 1), and proper specification and accommodation of the suppression pool temperature limit (phenomenon II.A in Table 6-1, NUREG-0420, Supp. No. 1).

(c) The capability and adequacy of the test procedure to periodically demonstrate an acceptable leakage rate of the drywell floor seal and downcomer vacuum breakers and other leakage paths that could lead to excessive steam bypass of the suppression pool has not been demonstrated.

(d) Adequacy of the design to insure, with sufficient margin, that the primary containment and associated safety-related structure can accommodate the simultaneously applied loads of transient and LOCA events has not been demonstrated.

(e) Suffolk County further contends that the extent of the deficiencies resulting from the Mark II containment design program may be further exacerbated by the fact that an adequate and properly controlled experimental design verification program as required by 10 CFR 50, Appendix B, Sections III and XI has not been performed. The verification of the design adequacy of the primary containment reactor pressure vessel supporting structure, and associated safety-related systems and components is deficient with specific regard to testing under the most adverse design conditions, performance of tests under suitable environmental conditions, documentation and evaluation of test results, and use of test data developed under a non-controlled (foreign) test program. There is, therefore, lack of assurance that the acceptance criteria used by LILCO in evaluating the Shoreham design contains suitable conservatism.

III. DISCUSSION OF ISSUES

III.A. Introduction: Nature of Problem

3. The origin of the design deficiencies potentially affecting the Shoreham containment design (the Mark II) are well known and well documented in the industry literature. A good summary of the Mark II issues is found in the first issue of the SER, [1/] and the chronology of the Mark II program is contained in the NRC's 1981 "Unresolved Safety Issue" (A-8) report. [2/] To provide background and a context for my testimony I am appending a copy of Section 1 of the A-8 report (Attachment 1) and a copy of a background memorandum I have prepared (Attachment 2).

4. The seriousness of the Mark II problem is well evidenced by the fact that investigation of the basic problems has been underway for eight years (1974-1982) but the program is still not fully completed and implemented. The scope of effort that has been required to investigate these issues also is indicated by the list of 49 reference documents identified in the NRC's acceptance criteria report [3/] and by the eleven page list of more than 160 reports submitted to the NRC during the course of the generic program. [4/]

5. My concerns regarding the satisfactory resolution of this significant safety issue at Shoreham are not that the problems have not been extensively investigated. It is obvious

that LILCO, the Mark II Owners Group, the NRC, and numerous consultants have spent many millions of dollars worth of resources in the investigation, analysis, testing, and modification that have been required. My concerns, rather, as indicated by the contention wording, can be categorized into three basic issues:

a. The program is still not completed; therefore how can it be demonstrated that all necessary modifications have been identified and implemented?

b. LILCO, being in a quasi-lead plant position, has selected a combination of acceptance criteria for the Shoreham evaluation that include some from the Lead Plant Program (LPP), some from the Long Term Program (LTP), and some that are plant unique. Has this complex design and evaluation program introduced weaknesses into the Shoreham design and/or review process?

c. The work carried out in investigation of the Mark II problems has been performed by numerous organizations, both with and without official status, over a period of eight years in at least five countries. Has all of the work and data been controlled and reviewed in a quality manner commensurate with the importance of reliable containment design and in conformance with applicable quality assurance criteria?

6. With regard to work remaining on the program, the following issues have either not yet been formally closed out, or represent new concerns that have not yet been addressed in the three issues of the SER:

	Issue	Reference	Status
a.	Pool dynamic loads	SER, Suppl. 2 page 1-2	"Awaiting futher information"
b.	Downcomer fatigue analysis	"	"Resolved pending confirmation"
c.	Steam condensation oscillation and chugging loads	SER, Suppl. 2 page 1-3	"Resolved pending confirmation"
d.	Steam condensation submerged drag loads	"	"
e.	Suppression pool bypass	SER, Suppl. 2 page 1-3 & 6-1	"Staff position"(?)
f.	Seismic and LOCA loadings	SER, Suppl. 2 page 1-2	"Resolved pending confirmation"
g.	Consultants' concern re chugging load specification	Letters from F. Eltawila to W. Butler dated 3/9/82 & 4/21/82	Reported resolved subject to submittal of a reduced damping analysis
h.	Vacuum Breaker performance	5/13/82 Letter, F. Eltawila to W. Butler	?

7. Items g and h above represent new concerns which have been raised since the issuance of Supplement 2 of the SER. As indicated in the above table, issue g may be resolved, but requires yet another analysis. Items b, c, and d are issues believed by the Staff to be resolved but for which confirmatory data have not yet been received. Item e was an issue of disagreement between LILCO and the Staff but it may have since been resolved by a commitment made in LILCO's 4/23/82 letter SNRC-693. Item a remains an open item, presumably to indicate an open commitment made by LILCO to:

a. "revise the DAR to address the generic long-term program (LTP) condensation oscillation and chugging load methodologies" and

b. "committed to the generic LTP as a whole for a confirmatory check". 5/

It is not clear how, if at all, this future "commitment" to the LTP is to be implemented or enforced. Thus, in view of the open nature of so many Mark II issues, I cannot conclude that there has been satisfactory resolution of the issues considered in the generic program or a satisfactory basis for licensing.

III.B. Program Continuation

8. Of the eight incomplete items identified in the preceding paragraph, issues a, g, and h are of the greatest concern. Issue a, pool dynamic loads, relates to the need

to compare the Shoreham Mark II program to the generic LTP as a whole for a confirmatory check. This then would be followed by updating of the DAR (Design Assessment Report)[6/] to include the LTP condensation oscillation and chugging load methodologies. Issue g identified in the preceding paragraph relates to a concern expressed by an NRC consultant regarding the chugging load specification. This concern was formally communicated by a letter dated February 16, 1982 (Board Notification 82-09). Background information on this problem was transmitted via F. Eltawila's April 21, 1982, letter to W. Butler and a meeting was held on April 8, 1982, to further consider this issue. The outcome of the meeting as reported by Eltawila was the NRC Staff's belief that the Mark II load specifications for these particular loads would bound the experimental data reported when a damping value of seven percent is used. The Staff pointed out, however, that a damping value of four percent is recommended in Reg. Guide 1.61. The Mark II owners felt that additional calculations using the four percent figure would also demonstrate acceptability of the load specification. This requires additional calculations and review to demonstrate and therefore remains an open item.

9. Issue h from the above table, vacuum breaker performance, represents a new issue first formally communicated to me via

F. Eltawila's May 13, 1982, memorandum to W. R. Butler.[7] This memorandum advised of a forthcoming meeting to be held on May 20, 1982, to discuss vacuum breaker performance during the pool swell phase of a loss of coolant accident. The May 13 memorandum did not further identify the nature of the problem but in a telephone conversation with Mr. Eltawila, I was advised that the pressurization rate in the wetwell during the early phases of the design basis accident has been determined to be of such a magnitude that the integrity of the vacuum breaker valves may not be assured. The results of this meeting are not yet known to me at the time of preparation of this testimony. The continuing appearance of "new problems" some eight years after the identification of the Mark II design deficiencies provides a continuing concern that all such issues have not yet been discovered.

III.C. Shoreham Acceptance Criteria

10. The acceptance criteria used in LILCO's basic review of the Shoreham containment design are identified in Appendix A of the DAR.[8] A total of 35 criteria positions are stated in Appendix A, some of which are based on the lead plant program (NUREG-0487), some of which are based on the long term program, and some of which are unique to Shoreham. This scattered selection of acceptance criteria is a practice that I believe could affect the adequacy of the NRC Staff's review of the

Shoreham design. I attended a meeting in Bethesda, Maryland, on March 20, 1981, which was scheduled for a discussion by LILCO and the NRC of the Mark II closure program. LILCO's intent to utilize a combination of LPP, LTP, and plant unique acceptance criteria was discussed at that meeting. The Staff's position was that such a process would add to their design review work load since they had been working towards the review of the two generic programs. The primary concern expressed at that time was the potential impact on the completion of the review and the issuance of the subsequent Safety Evaluation Reports. However, it must be recognized that work load can also affect quality of work. The complexity of the review process necessary to evaluate the Shoreham submitted loads can be illustrated by looking only at the various load definition tables contained in some of the basic documents involving the Mark II program in general and Shoreham in particular. To review only in summary form the load specification tables and acceptance criteria for Shoreham, one must look at the following documents:

Design Assessment Report, Revision 5, Appendix A
NUREG-0487, Appendix D
NUREG-0487, Supplement 1, Table IV-1
NUREG-0487, Supplement 2, Sections 2 and 3
NUREG-0420, Table 6-2

NUREG-0420, Supplement 1, Table 6-1

NUREG-0420, Supplement 2, No summary table

NUREG-0808, Appendix A and Table C-1

The difficulty of reviewing the Shoreham Plant to this continuation of changing and diverse criteria is emphasized by merely pointing out that in all of the above tables, each uses a different alpha numeric identification system for the load or phenomenon discussed. This lack of consistency between the different programs and the different time frames makes review extremely difficult and gives rise to concern that reviews by the various organizations may not be accurate and complete.

III.D. Diverse QA Problems

11. With regard to administration of the quality assurance program efforts of the many and diverse organizations contributing to this program, a few examples are in order. I was personally involved in meetings with the NRC in the early days of these programs where we were not even permitted to state the "Country of Origin" for the test data on which the lateral downcomer loads were based. This testing had been done in Germany and was considered proprietary, and the basic data were not provided to the NRC for the initial work because of this proprietary designation. I find it interesting to see

that LILCO has used the same 8.8 kip static equivalent lateral downcomer load that came out of this pre-program foreign test data base. Other data were taken from tests performed in Sweden, Italy , and Japan. It may be that the data have since been independently verified and that the loads adopted are suitably conservative. The fact remains that 10 CFR 50, Appendix B requires that Design Control and Test Control of safety related equipment be carried out under a suitable quality assurance program and that tests verifying such design adequacy shall be conducted in accordance with written test procedures. This requires preapproval of the test program and performance of the test under "suitable environmental conditions." My specific concern on this point is that such a diverse group of resources has been employed in resolution of this problem. Many of the organizations were outside of NRC regulatory control and some had no legal existence. I believe that a thorough and independent quality assurance audit would reveal many cases of loosely defined interfaces. To my knowledge, such an audit has not been done.

IV. CONCLUSIONS & RECOMMENDATIONS

12. Based on the foregoing discussion I conclude:

a. The Mark II safety reassessment program which has been underway for eight years is still not complete. The meeting scheduled for May 20, 1982 to consider the newly discovered problem of a potentially destructive acceleration load to vacuum breaker internals because of a rapid wetwell pressurization rate is illustrative of this point.

b. LILCO and the Staff have not demonstrated that the acceptance criteria used for Shoreham have been consistently applied to assure that a thorough and effective review has been performed.

c. Implementation of all necessary modifications may not yet be complete at Shoreham.

d. The Mark II generic programs (LPP & LTP) were not conducted in compliance with applicable quality assurance requirements.

10. I recommend:

a. An operating license should not be issued for Shoreham before the LTP is fully complete.

b. LILCO and the Staff should verify and document that each Shoreham acceptance criteria position is at least as conservative as that approved for the generic LTP.

c. A clear definition should be made of LILCO's commitment to the LTP.

d. An independent audit should be conducted over the full range of the Mark II generic programs.

e. All of the above should be completed before Shoreham operation is permitted.

REFERENCES

1/ NUREG-0420, SER Related to the Operation of Shoreham Nuclear Power Station, Unit No. 1, April, 1981, pp. 6-1 through 6-41.

2/ NUREG-0808, Mark II Containment Program Load Evaluation and Acceptance Criteria, August 1981, pp. 1-1 through 1-4.

3/ Ibid 2/, pp. 4-1 through 4-4.

4/ Ibid 2/, pp. B-1 through B-12.

5/ NUREG-0420, Supplement No. 1, September 1981, p. 3-1.

6/ Plant Design Assessment for SRV and LOCA Loads, Revision 5, December 1981.

7/ Memorandum, F. Eltawila to W. R. Butler, Forthcoming Meeting ... to Discuss the Vacuum Breaker Performance .... May 13, 1982.

8/ Ibid 6/, pp. A-1 through A-14.

ATTACHMENT 1

PORTION OF NUREG-0808

NUREG-0808

OCT 1 4 1981

MARK II Containment Program Load Evaluation and Acceptance Criteria

Generic Technical Activity A-8

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

C. Anderson

1 INTRODUCTION

Pursuant to Section 210 of the Energy Reorganization Act of 1974, the capability of the boiling-water reactor (BWR) Mark II containment to withstand loss-of-coolant accident (LOCA)-related pool dynamic loads, which were not considered in the original containment design, was designated an "Unresolved Safety Issue" (Task Action Plan (TAP) A-8). This report, along with three previous NRC reports,[1,2,3] describes the generic hydrodynamic loads to be used to evaluate BWR/Mark II facilities. The NRC and its consultants have reviewed the applicable experimental and analytical programs, and have concluded that the proposed LOCA hydrodynamic loads referenced in Appendix C, as modified by the requirements set forth in Appendix A ("NRC Acceptance Criteria for Mark II Containment"), will provide a conservative evaluation of the containment structures, piping, and equipment for suppression pool hydrodynamic LOCA loading. These loads constitute the resolution of TAP A-8.

1.1 Problem Definition

In the United States there are 11 BWR facilities in various stages of construction which have the Mark II containment system. About half of these are currently scheduled for operation by the end of 1982. A listing of the domestic BWR facilities with the Mark II containment system is provided in Table 1.1-1.

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

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

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

Table 1.1-1 Listing of Domestic BWR Facilities with the Mark II Containment System

Plant Name	Applicant
Bailly 1	Northern Indiana Public Service Co. Chesterton, Indiana
WPPSS-2	Washington Public Power Supply System Richland, Washington
LaSalle 1 and 2	Commonwealth Edisor Company Chicago, Illinois
Limerick 1 and 2	Philadelphia Electric Company Philadelphia, Pennsylvania
Nine Mile Point 2	Niagara Mohawk Power Company Syracuse, New York
Shoreham	Long Island Lighting Company Hicksville, New York
Susquehanna 1 and 2	Pennsylvania Power and Light Company Allentown, Pennsylvania
Zimmer	Cincinnati Gas and Electric Company Cincinnati, Ohio

During the large-scale testing of the Mark III containment system design in the period 1972 through 1974, new suppression-pool hydrodynamic loads were identified for the postulated LOCA. General Electric (GE) tested the Mark III containment concept in its Pressure Suppression Test Facility (PSTF)[4]. These tests were initiated for the Mark III concept because of the geometrical configuration differences between the previous containment concepts and the Mark III design, principally in the utilization of horizontal vents. (Steam had been ejected vertically downward into the suppression pool in the previous BWR containment designs, whereas the Mark III design ejects steam horizontally into the suppresion pool.) More sophisticated instrumentation was available for the Mark III tests, as were computerized methods for data processing.

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

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

Consequently, in April 1975, the NRC sent letters to each of the domestic utilities having BWR facilities with Mark II containment system designs requesting that they provide information demonstrating the adequacy of their containment design. These letters reflected NRC concerns about the need to evaluate the containment response to the newly identified dynamic loads associated with postulated design-basis LOCA.[5]

The domestic Mark II containment owners formed an ad hoc Mark II Owners Group to develop responses to these NRC requests. They developed a two-part program consisting of the Lead-Plant Program (LPP) and the Long-Term Program (LTP) to accommodate the licensing needs of the lead and the following Mark II plants. These programs are described below.

1.2 Lead-Plant Program

Licensing activities for certain Mark II lead plants (Zimmer, Shoreham, and LaSalle) were originally scheduled to precede completion of the entire Mark II containment program. Consequently, the LPP was developed to demonstrate that sufficient information about and understanding of the pool dynamic phenomena of interest existed to establish conservative loads for the lead plants. Because of the LPP emphasis on developing loads consistent with the licensing requirements of the lead plants, a bounding interpretation of the available test data was utilized for many of the pool dynamic loads. This was done to ensure that conservative loads were available for the lead-plant evaluations.

The NRC staff reviewed the Mark II owners' lead-plant program and identified acceptable pool dynamic loads to be used in the evaluation of Mark II pressure-suppression containment designs for these plants. These loads were discussed in the staff report, "Mark II Containment Lead Plant Program Load Evaluation and Acceptance Criteria," (NUREG-0487)[1] issued in October 1978. Supplement 1 to this report[2] was issued in October 1980. It addressed acceptable alternative loads to the original NUREG-0487 lead-plant loads. Supplement 1 also identified problems relating to the lead-plant condensation-oscillation (CO) and chugging loads. Large-scale testing applicable to the Mark II design which was conducted in 1979 indicated deficiencies in the lead-plant CO and chugging loads. As a result, the Mark II owners developed interim CO and chugging loads to reflect the results of these tests. The staff's evaluation of these loads was provided in Supplement 2 to NUREG-0487[3]. Supplement 2 was issued in February 1981. It completed the Lead-Plant Program.

1.3 Long-Term Program

The objectives of the LTP were (1) to provide justification, by tests and analyses, for refinement of selected lead-plant bounding loads, and (2) to provide additional confirmation of certain loads used in the LPP.

The Mark II owners elected to adopt most of the pool-swell loads developed during the LPP[1,2]. An exception was the diaphragm-floor upload specification. The Mark II owners revised the LPP floor upload specification to reflect a deficiency in the earlier specification. Since completion of the LPP, the

Japan Atomic Energy Research Institute (JAERI) has planned an extensive series of full-scale tests using a 20° sector of a Mark II containment, which to date have been partially completed. The NRC and its consultants have utilized the results of these tests to confirm the Mark II LPP pool-swell acceptance criteria. The staff's evaluation of the Mark II owners' revised diaphragm-floor upload specification and the results of the staff pool-swell-loads confirmation study are provided in Section 2.1 of this report.

During the LPP, the Mark II owners concluded that steam condensation pool dynamic loads should be developed that were more rigorous than the bounding loads used in the LPP. These loads included the CO and chugging loads on the pool boundary and the chugging-induced lateral loads on the containment downcomers. Large-scale tests were conducted in 1979 to provide additional information related to these loads. These tests formed the basis for the LTP steam loads. The staff evaluation of these loads is provided in Sections 2.2 and 2.3 of this report.

The Mark II owners proposed an alternate load specification for the LPP LOCA-related submerged structure drag load in the form of a ring-vortex model. This ring-vortex model considers the drag loads that occur during the water-jet and the air bubble periods following a LOCA. The staff's evaluation of this model is provided in Section 2.4 of this report.

The LOCA-related suppression-pool hydrodynamic acceptance criteria for the Mark II containment design are provided in Appendix A. This appendix contains the applicable LOCA-related criteria from the LPP and the revised pool-swell and steam-load criteria resulting from the LTP. Appendix B contains a list of the reports transmitted to the NRC during the course of the lead-plant and long-term programs. A summary of the LOCA pool dynamic loads acceptable to the NRC is provided in Appendix C.

This report concludes the generic Mark II containment program. The loads referenced in Appendix C, as modified by the acceptance criteria provided in Appendix A, constitute the resolution of Unresolved Safety Issue A-8.

ATTACHMENT 2

BACKGROUND OF THE MARK II DESIGN

BACKGROUND

MARK II CONTAINMENT DESIGN

by

Dale G. Bridenbaugh

All U.S. nuclear power reactors are enclosed in leak-tight structures called the primary containment which is designed to contain and control the leakage of reactor coolant and radioactive materials that might be released by leaks or ruptures of the reactor primary system. Early power reactors and most pressurized water reactors (PWR's) are housed in dry containments similar in structure to the large petroleum gas storage tanks. PWR's have been able (with some exceptions) to use dry containment even for the larger sizes now being constructed. The boiling water reactor (BWR) built by General Electric however, has by nature of its design a larger primary system. This makes it more difficult (or costly) to use dry containment, so starting with the series of plants placed in service in the late 1960's, GE utilized a so-called "pressure suppression" containment system.

In the pressure suppression containment, the reactor system is housed in a chamber called the dry well. The dry well communicates with a second chamber, the wet well, by a series of large vents which terminate several feet beneath the surface of the large volume of water contained in the wet well. In the event of a system rupture, the dry well fills with steam from the reactor coolant and then vents the releasing vapor through the pool of water. The water condenses the released steam being forced through it and suppresses the pressure built up in the containment, hence the name pressure suppression. Some Westinghouse plants use a containment system which absorbs energy and suppresses the pressure by passing the escaping steam through a series of baskets containing ice. This system is called the ice condenser containment.

Water pressure suppression is a containment concept used exclusively (in the U.S.) by plants using GE's BWR. This concept was first used in the PG&E Humboldt Bay-3 nuclear unit in Eureka, California. Humboldt Bay-3 is a 65 MWe plant placed in service in 1963, but since indefinitely shutdown due to seismic design deficiencies. GE, with PG&E did a series of pressure suppression verification tests in the period between 1958 and 1962. The first series of tests performed were to support the licensing of the Humboldt Bay-3 unit and the later series of tests were for design verification for the proposed Bodega Bay plant which was never built. These tests, in 1958 through 1962, provided the design basis for both the Mark I and the Mark II containment designs, initially developed by GE and utilized in 36 BWR plants in operation or planned for the U.S. (25 Mark I and 11 Mark II). GE also utilized both containment designs in their overseas business, building Mark I plants in Japan, Switzerland, and Spain, and Mark II plants in Japan and Italy. It is also worthwhile mentioning that the Mark II concept was first utilized by ASEA-ATOM in the BWR's built in Sweden. GE had no direct responsibility for the Swedish plants, but did have a licensing arrangement with the Swedes and in all probability based some of the GE Mark II design on Swedish experience.

It is also of some relevance to keep in mind the contractual responsibilities of GE in the furnishing of the equipment for the Mark I and II plants. In the Mark I units first placed in service, GE acted as the turnkey supplier and was totally responsible for the design and construction of the containment systems. In the late 60's, partially as a result of significant cost overruns on the turnkey plants and partially as a result of believing that the market and design had stabilized, GE dropped out of the turnkey business. The scope of supply offered by GE was limited to the nuclear steam supply system (exclusive of containment) and an extensive package of engineering support services. The support services consisted of, among others, definition of the critical parameters of the containment such as volume, pressure, etc., but the detailed structural design of the containment system was made the responsibility of the utility. For the most part, the utility utilized the same suppliers that GE had used in the turnkey business. Detail structural design continued to be performed by the Bechtels and Sergeant and Lundys and other architect-engineers and the containment itself was fabricated in most plants by Chicago Bridge & Iron. There were some exceptions to this relationship, particularly in GE's overseas business and at TVA but these exceptions are not relevant for the discussion at hand.

The Mark II containment concept was proposed by GE in their marketing of reactors in the late 60's. It had been found that the Mark I systems were costly to design and build and involved difficult construction sequencing which extended the theoretical construction schedules of the plants. This resulted in a cost penalty to the BWR that GE found required elimination in order to maintain competitiveness with Westinghouse and other PWR suppliers and the all-concrete construction Mark II system was the interim step in resolution of this problem. The ultimate solution to the problem was development of the Mark III containment, a concept which used more simplified concrete construction methods and eliminated the high pressure design requirement on the wet well space. GE obtained orders for 11 BWR's with Mark II containments from U.S. utilities, the first of which was originally scheduled to be the Shoreham plant on Long Island. None of the domestic Mark II plants are in service and one of the 11, Bailly, in northern Indiana has been cancelled. In all of the Mark II's (and the Mark III's for that matter), GE's responsibility was supplier of the nuclear steam supply system hardware and provider of only the essential safety-related parameters to be used by the utility or the utility's designee in the detailed design of the containment.

In the early and mid-70's, after approximately 15 of the Mark I's were in service, it was discovered that neither the Mark I's or II's had properly taken into account a number of dynamic loading conditions that might be experienced during transient operation or during a design-basis loss of coolant accident (LOCA). These discoveries were made separately, through operating experience at several Mark I (and similar European) plants and at the GE San Jose Mark III Pool Swell Test Facility (PSTF), a one-third scale, segmental facility, built by GE to verify their new concept, the Mark III containment. The events experienced at the operating plants had to do with severe hydrodynamic loads experienced with the blowdown of steam into the suppression pools through safety relief valves (SRV's). The SRV's are piped to the suppression pool and open to vent steam from the reactor to the pool in the event of a turbine trip or other transient events requiring rapid shutdown of the plant. Such operation was found to cause heavy vibration of the pool structure and mechanical damage was experienced.

The first evidence that the SRV discharge loads were a problem occurred at the German Wuergassen plant (not built by GE) in about 1971 or 72. At the Wuergassen plant, blowdown of an SRV caused a small rupture of a plate in the wet well pool boundary. The significance of this event was in all probability overlooked by U.S. designers for several years. SRV openings were also experienced at the KKM plant (Muhleberg, Switzerland) during startup testing about 1972-73, at Quad Cities about 1972-73, and at Browns Ferry about 1974. At each of these plants, heavy vibration was noted and piping supports were damaged requiring subsequent repair. Because of this experience, GE and the NRC became concerned and GE and Commonwealth Edison instrumented one of the Quad Cities units and performed tests (about 1974) but these were somewhat inconclusive. Similar tests were subsequently performed at Browns Ferry.

Meanwhile, the Mark III PSTF was placed in operation about 1973 and tests there ultimately demonstrated a series of loads associated with LOCA events had been overlooked or underassessed at all pressure suppression plants. This was a most serious problem for the Mark I's and II's. The Mark I's were most significantly affected since they were in operation and suddenly facing a generic shutdown because the newly discovered problems represented an unreviewed safety question. These same problems represented for the Mark II's a serious scheduling difficulty because a number of the Mark II's were ready to begin concrete pours of the containment structures. Utilities recognized that changes would probably be required but did not know how extensive the changes would be.

PP&L, owner of the Susquehanna plant took the first overt action and halted construction in April, 1975 because of uncertainties about the pool dynamic loads. On April 17, 1975, the NRC issued letters to all Mark II licensees on SRV loads and then issued additional letters on LOCA loads on April 18, 1975.

The Mark II owners formed an Owners Group in May, 1975 to address these problems on a generic basis. The Mark II program is still underway. Modifications have been and are continuing to be made at the Mark II plants. As indicated earlier, no Mark II plant is in operation in the U.S. but LaSalle is presently scheduled to be the first in 1982.

The foregoing background information is also summarized in numerous documents. The NRC issued a Mark II lead plant acceptance criteria (NUREG-0487) in October, 1978. This document contains a fairly accurate and complete summary of the Mark II problems. In 1978, the NRC declared the Mark II problem an "unresolved safety issue." It was reported to Congress by the NRC's report documenting their program for resolution of generic issues in January, 1978. Pressure suppression problems have also been reported in each NRC Annual Report starting with 1976.

Other documents which contain descriptions of the containment (GE) issues are:

1. NUREG-0474, A Technical Update on Pressure Suppression Type Containments in Use in U.S. Light-Water Reactor Nuclear Power Plants, July, 1978. This report was issued by the NRC in order to inform Congress and the public of the NRC's position on pressure suppression containments. This was brought about by the release of internal NRC documents that appeared to question the licensing of such facilities.

2. NUREG-0420, Safety Evaluation Report Related to the Operation of Shoreham Nuclear Power Station, Unit 1, April, 1981. This report documents the latest detailed review by the NRC of a Mark II plant for licensing purposes. It contains in Section 6 a fairly detailed description of the NRC's current position on Mark II issues and is useful as general background information.

3. NUREG-0661, Safety Evaluation Report, Mark I Containment Long-Term Program, July, 1980. This report was issued by the NRC documenting their assessment of the analytical portion of the Mark I long-term program.

4. Licensing Documentation. Each plant under construction and being reviewed for operating license issuance by the NRC is required to submit detailed information on the plant. The basic document submitted is a Final Safety Analysis Report (FSAR), a multiple volume series describing the design features of the plant. These reports are available to the public through the Public Document Rooms and contain in Section 6 extensive descriptions of the containment systems. In addition, each Mark II owner has agreed to submit to the NRC for review a more detailed Design Assessment Report (DAR). These detailed reports describe how the specific licensee is responding to the different loads presently known which impact the adequacy of the containment.

5. Generic Mark II Reports. The above list of reports contain references to literally hundreds of other related reports that have been submitted to the NRC. NUREG-0487, for example, contains in Section V, a list of 67 referenced reports.