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Public Utility Commission

PENNSYLVANIA PUBLIC UTILITY COMMISSION

V.

PHILADELPHIA ELECTRIC COMPANY
DOCKET NO. R-850152

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MAR 24 1986

SUR-SURREBUTTAL TESTIMONY

OF

ROGER J. MATTSON

NRC LICENSING EFFECTS ON SCOPE AND SCHEDULE OF CONSTRUCTION OF LIMERICK 1; MARK II CONTAINMENT ISSUES

March 13, 1986

SUR-SURREBUTTAL TESTIMONY OF ROGER J. MATTSON

Q. Please state your name and business address for the record.

A. I am Roger Mattson, and my business address is: 2600 Virginia Avenue, N.W., Washington, D.C. 20037.

Q. Are you the same Roger Mattson who has previously presented testimony in this proceeding?

A. Yes, I am. My professional experience and other qualifications are stated in my prepared direct testimony, PECO Statement No. 9, including my extensive experience in the NRC licensing staff for most of the period from 1967 to 1984 and my senior management role in the licensing of Limerick 1 and many other plants between 1977 and 1984. I also prepared rebuttal testimony in this proceeding, PECO Statement No. 9A and PECO Exhibit RJM-1.

Q. What is the purpose of your sur-surrebuttal testimony?

A. I have reviewed all or portions of the surrebuttal testimony of Stephen H. Hanauer and James J. O'Brien filed on behalf of the Office of Consumer Advocate. The purpose of my testimony is to provide for consideration by the Commission and the Administrative Law Judge the opinions which I have formed as a result of my evaluation of the positions of these witnesses.

I. SUMMARY OF TESTIMONY

Q. Would you please summarize the OCA opinions that you have evaluated and the opinions you have formed about them?

A. Mr. O'Brien has testified (OCA Statement 1B, pp. 7-9) that it is inconceivable to him that, if Limerick 1 had been completed in mid-1982, the NRC would have forced the plant to remain idle for two years before issuing an operating license (OL). He also testified that my analysis was flawed in PECO Exhibit RJM-1, and that it did not support the conclusion that NRC requirements would have delayed the licensing of Limerick 1 until mid-1984. Mr. O'Brien presented no basis of his own for his opinions, he has no qualifications or experience in NRC licensing matters, and he failed to identify any faults in my analysis. Furthermore, what is inconceivable to him, i.e., that the inability to meet NRC requirements would force a completed plant to remain idle for a long time period before receiving an operating license, has happened to at least one third of the 30 plants that have received operating licenses since 1979. Five other plants are essentially complete and are, at this time, not in receipt of an OL because of unresolved NRC licensing issues. Also, since 1979 there have been 10 operating plants shut down for nonconformance to NRC requirements for an average time in excess of one year. These data show that, contrary to the unsupported incredulity of Mr. O'Brien, if Limerick 1 had been completed in mid-1982, it would have joined a host of other plants that have not been able to operate because of scope additions caused by the NRC and more stringent interpretations of NRC requirements than ever before. Furthermore, Limerick 1 was the highest population density and one of the most highly contested of all plants licensed since 1979. These factors would have combined to prohibit PECO from obtaining an OL for Limerick in mid-1982.

Dr. Hanauer also conjectures that NRC would not have delayed licensing if Limerick 1 had been completed in mid-1982. Although he concedes that the same licensing requirements would have applied in that earlier time frame and that they could not all have been met in mid-1982, he says that somehow PECO and the NRC would have found a way to license Limerick without meeting the requirements. He also concedes that the highly contested nature of the hearing would have presented problems, but again he conjectures that the hearing could have been started earlier so as to end earlier than it did. He fails to recognize that even if the hearings were started earlier, there were items that were incomplete in 1982 owing to NRC requirements uniquely applied to Limerick (e.g., environmental qualifications, fire protection, TMI backfits, ATWS, and other PRA matters), all of which were the subject of contentions in the hearing and important to the outcome of the Probabilistic Risk Assessment (PRA). Unlike LaSalle and Susquehanna, these items were all required to be completed at Limerick before operation because of the NRC risk criterion that was applied solely to Limerick and because of contentions in the NRC licensing hearing. Thus, these items could not have been resolved to the satisfaction of the Atomic Safety and Licensing Board without implementation of the changes prior to operation. Even though he has little or no experience in negotiating licensing issues in contested OL proceedings, Dr. Hanauer should realize that if the plant had been complete except for these issues in mid-1982, the resolution of contentions in the hearing, especially those rather sophisticated contentions surrounding ATWS, other aspects of the PRA and the level of acceptable societal risk at Limerick, would have delayed operation for many months. This would have occurred even if the NRC Staff had caved in to requests by PECO for special consideration. Based on my position as the senior staff manager responsible for resolution and negotiation of the bulk of these issues at the time, I know we would not have caved in to such a request because of the high population density of the Limerick site, the highly contested nature of the licensing hearing and the likelihood that the Commission would ultimately have denied such a basis for a full power operating license.

Dr. Hanauer also addressed the Mark II containment question in his rebuttal testimony (OCA Statement 2A). He concedes that the Mark II problem for Limerick was a question of design margin, not a question about the basic structural adequacy of the original design. However, he continues to cling to the notion that the omission of hydrodynamic loads and forces from the original pressure suppression containment design was either an error by General Electric or was known and not reported to early BWR plant owners or regulatory authorities until actual plant damage occurred. This is contrary to the evidence provided in this proceeding and cited by Dr. Hanauer himself. Appendix IV of the Final Hazards Summary Report submitted by Pacific Gas and Electric Company for the Humbolt Bay reactor (PECO Exhibit SL-3) shows that knowledgeable experts and responsible officials of General Electric Company, Pacific Gas and Electric Company, Bechtel, the Atomic Energy Commission, and the Advisory Committee for Reactor Safeguards, at a minimum, were aware of the early test results and agreed with the basis provided for discounting any indications they might have contained of hydrodynamic phenomena. As described in my rebuttal testimony, all of these parties endorsed the adequacy of the pressure suppression design for accommodating the safety issues perceived to be important at the time. Dr. Hanauer also fails to account for the fact that the technical basis for the first pressure suppression containment design was then used over and over for each BWR licensed in the U.S. between the early 1960s and the mid-1970s. Thus, the technical adequacy of the design was exposed to many opportunities for reexamination by a number of utilities, architect engineering companies, hearing boards, AEC license reviewers, ACRS members, national laboratory scientists, and qualified members of the technical community. As I testified in PECO Statement

No. 9A at page 14, until operating experience showed otherwise, this community of experts simply did not perceive that there were questions remaining to be answered about pressure suppression containment performance, so they did not look for the answers.

In the following sections of this testimony, I will address specific portions of the rebuttal testimony of Mr. O'Brien and Dr. Hanauer to demonstrate how they have not correctly accounted for the effects of the NRC licensing process on the scope and schedule of construction of Limerick 1 and to further demonstrate errors in Dr. Hanauer's conclusions regarding the Mark II containment design.

II. NRC LICENSING EFFECTS ON SCOPE AND SCHEDULE OF CONSTRUCTION OF LIMERICK 1

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

A. I wish to correct impressions left by the rebuttal testimony of Dr. Hanauer and Mr. O'Brien. They mischaracterize the analysis we provided in PECO Exhibit RJM-1 and mistakenly imply that the NRC never delays or forbids operation of nuclear power plants except in cases involving what Dr. Hanauer called "the most drastic safety concerns." I will also summarize the evidence we have assembled here and in my earlier testimony to demonstrate that Limerick 1 could not have been licensed before mid-1984, even if construction was essentially complete in mid-1982.

Q. Since you are testifying as an expert on the NRC licensing process in opposition to the OCA witnesses, can you summarize the qualifications of Mr. O'Brien and Dr. Hanauer with respect to NRC licensing proceedings?

A. As far as I can tell, Mr. O'Brien has never participated in any capacity in an NRC licensing proceeding. On the other hand, Dr. Hanauer was an observer of the

No. 9A at page 14, until operating experience showed otherwise, this community of experts simply did not perceive that there were questions remaining to be answered about pressure suppression containment performance, so they did not look for the answers.

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Q. Since you are testifying as an expert on the NRC licensing process in opposition to the OCA witnesses, can you summarize the qualifications of Mr. O'Brien and Dr. Hanauer with respect to NRC licensing proceedings?

A. As far as I can tell, Mr. O'Brien has never participated in any capacity in an NRC licensing proceeding. On the other hand, Dr. Hanauer was an observer of the licensing activities of the AEC and the NRC for a number of years. However, his participation in the licensing process was mainly limited to a few months in late 1978 and early 1979 just before the accident at TMI. Even during this period I do not believe that he actively participated in any licensing hearings, he was not responsible for development or approval of staff testimony in any particular licensing hearing, and he did not negotiate the deferral of utility licensing commitments in any contested proceedings. More importantly, experience from this period, as I have previously described, is not indicative of licensing conditions as they existed in the post-TMI period. During this later period, the NRC licensing process become considerably more rigid, and the necessity to comply prior to receipt of an OL with all risk-significant NRC requirements was much greater, as was the level and effect of public intervention. Dr. Hanauer, however, was not very much involved in the licensing process during this period, and in fact, left the NRC entirely in 1982.

Q. How did Dr. Hanauer and Mr. O'Brien misinterpret PECO Exhibit RJM-1?

A. Dr. Hanauer says at page 10 of OCA Statement 2A and Mr. O'Brien says at pages 1 and 8 of OCA Statement 1B that our analysis concentrated on what actually happened in the Limerick licensing process, not what could have happened in a pre-1982 licensing process. That is not the case. At page 27 of PECO Exhibit RJM-1, we made the following statement: "This section examines each component of the licensing process actually experienced by Limerick 1 and describes whether and how each might have been accelerated to support an earlier issuance of the low power operating license." The purpose of the analysis that was presented in PECO Exhibit RJM-1 was to see if completion of the plant in mid-1982 could have led to earlier operation than was actually experienced. More than one dozen specific components of the licensing process were examined, and three were found to be amenable to shortening. Some steps would certainly have lengthened if the plant were finished in mid-1982; e.g., there were scope changes required by the NRC that would have been unfinished in 1982 and these were the subject of proposed contentions in the hearing, so the hearing would have gone on past mid-1982. In our analysis, those components of the licensing process that would have lengthened the hearing process were conservatively assumed to take the same amount of time they took for a 1984 completion date (where they were largely dismissed from contention in the hearing because the design changes already were implemented in the plant).

Dr. Hanauer and Mr. O'Brien have not contested our conclusions on even one of the dozen components we analyzed in PECO Exhibit RJM-1, nor have they identified even one concrete way to shorten the time required to satisfy the PRA licensing requirements that were imposed by the NRC on Limerick. Relying totally on unsupported generic judgments about a licensing process in which they never directly participated, they say that the NRC staff and PECO would have worked it out. However, they do not say how this would have been done. Moreover, they do not recognize that the Commission or the Atomic Safety Licensing Board likely would have overturned any such deals between the staff and PECO, just as they did in comparable OL cases such as Shoreham, Diablo Canyon, Comanche Peak, Sequoyah, and Byron or in the restart licensing action on TMI-1.

Q. Has Dr. Hanauer also mistakenly portrayed the frequency with which unresolved NRC licensing issues keep plants from operating?

A. Yes. He said that only the most drastic safety concerns would have resulted in the NRC delaying a completed plant for several years, and he cited two examples, Diablo Canyon and Zimmer. In actual fact there have been a number of plants whose operations have been delayed by NRC or its licensing boards since the accident at Three Mile Island. By my estimate, more than 25 plants licensed for operation or awaiting licenses for operation since the accident at TMI have been kept from operation because of a variety of unresolved licensing issues. Delays of two or more years are not uncommon. The issues that would have kept Limerick 1 from operating if it had been finished in mid-1982 include the PRA, ATWS, fire protection, environmental qualifications, and numerous TMI backfits, none of which could have been completed in 1982 and all of which were subjects of contentions in what Dr. Hanauer has conceded was a "heavily contested proceeding" (OCA Statement 2A, p. 26).

Q. Tell us about the role in NRC decision-making of costs associated with a delayed startup. Dr. Hanauer implies at page 24 of OCA Statement 2A that they are so high that the NRC would let a plant go into operation even if the PRA showed that modifications were necessary. Is that correct?

A. No, it is not. The delay costs are not an element of the licensing decision. Rather, they provide pressure to assure that a licensing decision is timely and, if it is negative, that it is technically well-founded. That was the case at Limerick. Without fire protection, environmental qualification, ATWS and certain post-TMI modifications, the plant did not meet the "comparable risk" standard set by the NRC solely for this particular high population density site. Thus, the plant would not have been licensed until these modifications were completed, regardless of the cost of delay.

Q. Dr. Hanauer and Mr. O'Brien have also said that if PECO had submitted the Limerick OL application to the NRC in mid-1978, it would have found a place higher in the NRC's priority scheme. Do you agree that this could have led to earlier licensing?

A. No, I do not. In PECO Exhibit RJM-1, we examined the case of an early submission of the OL application in mid-1978. We found that the unique circumstances at Limerick would have prevented completion of several items that were prerequisites (for Limerick) to plant operation, such as ATWS modifications, litigation of the PRA contentions, and other non-Mark II items described in the rebuttal testimony of PECO witnesses Helwig and Sproat. Additionally, Mr. O'Brien and Dr. Hanauer fail to account for several resource constraints that were acting on the licensing staff in the 1979 to 1984 time period that would have precluded arbitrary elevation of Limerick in the licensing priority scheme of the NRC. For example, in 1981 and 1982 we had moved engineers from all over the NRC into the licensing staff to overcome the post-TMI backlog of license applications. We had reached diminishing returns with these efforts because we began to use people who were more suited to research or policy development than licensing reviews. Hence, it is not the case that one or two more plants could be worked into the schedule. I was managing a staff that had swollen from 150 to 200 people in a few months. Most of these people were involved in licensing reviews. Every time a construction schedule slipped at one of the near term OL plants, my top managers and I breathed a sigh of relief because we knew we could not meet the schedules required of us for the high priority plants without such slippages. This was really a period of self-fulfilling prophecy -- many utilities could not complete enough of the post-TMI backfits to satisfy the staff, the

Commission and the Boards to allow them to go into operation on the schedules the utilities were projecting. Thus, the staff had more time for its review and did not have to take the blame for schedule delays. Dr. Hanauer was not involved in many of the monthly schedule meetings with the Director of Licensing because he had left the NRC's Human Factors Division by the time the licensing momentum was reestablished after TMI. These meetings were where we reviewed the so-called "Bevill Reports" that Dr. Hanauer cites. Thus, Dr. Hanauer probably does not recall our close calls with licensing reviews nearly slipping onto the critical path for plant startups. We literally could not have managed one more high priority, accelerated review without slipping other plants in the queue. In addition, acceleration of the Limerick review would have been even more difficult because of unique aspects involving the PRA, severe accidents, and ATWS. Extra reviewers would have been needed to achieve an acceleration of the Limerick schedule, and in these technical specialties the reviewers were even more rare than the normal license reviewers.

Q. At page 16 of OCA Statement 2A, Dr. Hanauer refers to a "Short List" of operating license reviews. What is that?

A. This comment results from a misunderstanding by Dr. Hanauer of the assignment of resources and the juggling of licensing schedules that licensing managers were facing in the 1980 to 1983 time frame. There was no such thing as a "short list". Rather, we worked on all plants that had submitted OL applications, but we gave priority attention to those plants mentioned in the Bevill Report to see that they did not slip on our account. In many cases, this was not difficult because the bulk of the licensing review of a plant is normally complete by the time it is within one or two years of an OL, except for last minute additions like the TMI backfits. In other cases it was much more difficult. The troublesome plants were the ones with highly contested hearings like Diablo Canyon, Midland, Shoreham, McGuire, and Limerick that kept eating up our resources long after the normal licensing review was complete and where the hearing had become the critical path to licensing. Remember that the Indian Point special hearing was going on in this same time frame and taking first priority for the nation's limited resources in the PRA discipline. In addition, a new round of contentions introduced by the intervenors in such hearings could easily put us on report to Congressmen Bevill for that plant. So those were the plants that received special management and extra resources. There was no near term OL "short list" to make those determinations.

Q. Is it your conclusion that the 39 month OL schedule achieved by PECO for Limerick 1 could not have been achieved if the application had been submitted in mid-1978?

A. That is correct. The 39 month schedule could not have been achieved for the reasons we cited in PECO Exhibit RJM-1. The factors cited there, when added to the NRC resource constraints that I have described above, would have meant that the review schedule for a mid-1978 Limerick OL application would have been stretched from 39 to at least 60 months. The upper end of the schedule slip is established according to when the resolution of the non-Mark II issues, which were contended in the hearing and required to be implemented to conform to the risk standard for Limerick, would have been implemented in the plant. From Schedule 1 in PECO Exhibit RJM-1, it can be seen that a 60 month elapsed time for a completed OL review would have placed Limerick 1 as 12th out of 30 OLs issued after the accident at TMI, slightly behind the elapsed times for LaSalle 1 and

Susquehanna 1. That would have still been an excellent achievement in view of the differences in population density and licensing hearings that made the Limerick OL more difficult to obtain.

Q. OCA witness Hanauer concedes that had PECO deferred implementation of late regulatory requirements, additional litigation would result. In your opinion, could ASLB hearings on contentions related to such deferred requirements have been successfully completed by mid 1982?

A. Certainly not. Had Limerick submitted an operating license application in 1978 as hypothesized in Part IV of PECO Exhibit RJM-1, the publication of new NRC regulatory requirements in the 1979-1981 time frame would have resulted in the filing, by intervenors with standing in the hearing, of many proposed contentions based on those requirements. Examples of new or reinterpreted requirements are ATWS, Equipment Qualification, Emergency Planning, TMI Requirements, Fire Protection, PRA and others described in PECO Exhibit 2 and the testimony of PECO witnesses Helwig and Sproat. Litigation of these issues, in lieu of implementation prior to licensing, could not have resulted in the issuance of partial initial decisions by the ASLB until well after mid 1982. Furthermore, as demonstrated in RJM-1, litigation of the PRA could not have been completed until mid 1984.

Q. Can you do an analysis to show how soon after mid 1982 these non Mark II and non PRA matters could have been litigated?

A. Yes, we have done that. As described in PECO Exhibit RJM-1, the ASLB's hearing related activities begin upon receipt of petitions to intervene, which are accompanied by proposed contentions. If an intervenor has previously been admitted to a licensing proceeding, the receipt by the ASLB of newly proposed contentions, based on information just made available to the intervenor, would start the process anew. Thus, under the "earliest possible" licensing scenario from Exhibit RJM-1, each time the NRC issued a new requirement for Limerick in the 1979-1981 time frame, additional contentions would be proposed by intervenors admitted as parties to the Limerick OL proceeding in early 1979.

Following issuance of each new regulatory requirement and the propounding of related contentions, some period of time would be required by PECO to evaluate its effects on the design of the plant, and to perform preliminary engineering sufficient to evaluate alternative courses of action. Additional time would be required to develop any necessary design changes sufficiently to support the NRC staff's review. Because the Limerick construction schedule would not have provided enough time to implement such design changes prior to licensing, justification for plant operation for some interim period prior to implementation would also be necessary. Such justification must demonstrate that interim operation can be permitted while adequately protecting the health and safety of the public. At this point, a submittal to the NRC would be made, beginning the NRC staff's review process. Simultaneously, copies would be served on the ASLB and intervenors, informing them of PECO's deferral plans. Schedule 1 develops a "shortest possible" interval between publication of the late regulatory requirements described in the rebuttal testimony of PECO witnesses Helwig and Sproat and an NRC submittal for each. For purposes of this hypothetical analysis, which I have testified is an impossible situation, the 30 month average time actually experienced is conservatively cut in half, to 15 months, to reflect the urgency in making a representative submittal as soon as practicable to avoid licensing delay for a completed plant.

Following the completion of its review, the NRC would document in one or more Supplemental Safety Evaluation Reports (SSER) the results of its reviews and, in this hypothetical case, the explanations of why the health and safety of the public would be protected during Limerick's operation even though the NRC safety requirement would not be implemented prior to initial operation. The NRC would have to make this finding in light of the results of the Probabilistic Risk Assessment which showed that Limerick posed a disproportionately high risk without these features, such as ATWS, being implemented (see rebuttal testimony of PECO Witness Helwig). As described in Exhibit RJM-1, the fact that these would be contested issues would cause the NRC Staff to exercise great care in justifying and documenting its conclusions. For these reasons, the Staff's earliest possible review of these deferred regulatory requirements for Limerick would have taken an average of at least 12 months, compared to the 15 month average time from submittal to SSER which actually occurred, as shown on Schedule 1. Completion of the NRC Staff's review of an issue is a prerequisite to the filing of Staff testimony on related contentions in the hearing. On a highly expedited basis, the filing of direct testimony could occur as early as one month after the issuance of the SSER for each issue, providing a short period for discovery by parties in the hearing.

Also on a highly expedited basis, a hearing could begin about one week after filing of testimony, and last, conservatively, one week on each issue. An ASLB partial decision could, if necessary, be expected in another two weeks. Thus, on a highly accelerated schedule, a partial decision could issue about one month after the filing of testimony on a particular issue, although more time would be required where multiple issues were litigated in the same time frame.

Schedule 2 tabulates these shortest possible intervals leading to earliest possible ASLB partial decisions for each of the issues described in the rebuttal testimony of PECO witnesses Helwig and Sproat. While litigation of related activities would begin as soon as possible following the issuance of new regulatory requirements, litigation would have lasted until at least December, 1983 using this very conservative model. The NRC decision to issue an OL could be expected to follow the Board decision by about one month. So, even for this hypothetical case advanced by Dr. Hanauer where the NRC Staff and the Board agreed to defer all of the unmet requirements, which I have already testified would not have occurred, no substantial acceleration in licensing results.

In my opinion, the ASLB's partial decisions on these issues would, in any event, have required implementation of these requirements prior to licensing to assure that the health and safety of the public would be adequately protected and to assure that the Commission or Appeal Board would not overturn the decision to license the plant, as occurred at Shoreham, Diablo Canyon, and TMI-1.

Q. At page 4 of OCA Statement 2A, Dr. Hanauer says that evidence from other licensing reviews between 1980 and 1982 shows that a large number of requirements had their implementation postponed to a later time. Does that evidence exist, and, if so, does it apply to Limerick?

A. No, there was no systematic deferral of licensing requirements for the so-called near term OLs in that time period. Instead, the decisions on whether and what to defer were made on a plant specific basis. Some plants had very few things deferred. Those were the plants with highly contested hearings. The important factors that we weighed in those decisions were overall plant safety, competency of plant management, unique features of the site or the design, and contentions in the hearing (both proposed contentions and admitted contentions). That is not to say there were not some deferrals on every plant; in fact, there were. The important point is that Limerick had very few deferrals. My rebuttal testimony went to some lengths to compare how the deferrals at Limerick were fewer than at Susquehanna and LaSalle, for example, and to demonstrate how the unique features of Limerick's site and hearing led to less flexibility in interpretation of NRC licensing requirements and fewer deferrals. OCA witnesses have presented no contrary evidence on this point.

Q. Continuing with this same subject, does that mean that Dr. Hanauer in OCA Statement 2A is wrong in his assertion that "only the most drastic safety concerns for plants in a mess of their own making" (pg. 11 and 13) caused licensing delays for plants that were already completed?

A. Yes, he is wrong. The same Bevill Reports that Dr. Hanauer cites show that in the period from 1980 to 1983, nearly every construction project slipped beyond the completion dates projected by the utilities. That was in part due to stricter interpretations of quality assurance requirements of the NRC, to added scope of construction due to new NRC requirements, and to licensing hearings. I know that because of NRC manpower shortages and the backlog of NTOL applications in that time period, we on the NRC staff were hard pressed to meet any of the licensing schedules. As fast as the completion dates slipped because of added scope, we slipped our safety reviews to match. I can recall no single case where we finished our review in advance of the completion of construction.

Q. Dr. Hanauer cites Diablo Canyon and Zimmer as being the only plants in such "drastic" state that they required licensing delays. Were there others?

A. Yes, there were many plants in this situation and Limerick would certainly have been one of them in mid-1982. There are a variety of facts to demonstrate this, as follows:

1. Even as late as 1984, Limerick was delayed 6 months between low power and full power licensing by a single contention in the hearing process (it concerned emergency plans for the Graterford Prison, a responsibility of the Commonwealth of Pennsylvania).

2. Shoreham licensing has been delayed by more than two years because of intervention involving emergency preparedness.

3. Grand Gulf was denied a full power license for nearly a year because of too many open items in the license, including technical specifications, management, and training items.

4. Byron was delayed by many months because of the Licensing Board concern for quality assurance of some electrical equipment.

5. Comanche Peak is being delayed for several years because of highly contested hearings involving allegations of deficiencies in construction quality, structures, and electrical equipment.

These are but a few examples. Altogether, I count more than one third of the last 30 plants licensed by NRC as being delayed by licensing issues. Also, nearly all of the 30 were slipped in schedule because of scope additions to meet NRC requirements issued after TMI. I also count five unlicensed plants now being delayed by licensing issues, and I know of 10 operating plants that were shut down for nonconformance to NRC requirements for periods averaging in excess of one year. Thus, it is not correct to say, as do Dr. Hanauer and Mr. O'Brien, that it is drastic, inconceivable, etc. for the NRC to keep a completed plant from operating. There is a mountain of data to the contrary, and Limerick was in great jeopardy of becoming such a plant.

Q. Tell us more about Diablo Canyon. Does it illustrate what Dr. Hanauer claims, i.e., that only a drastic plant with a mess of its own making can lead to long licensing delays?

A. No, it does not. The Diablo Canyon reactors, in the last two years or so of their licensing delay, were held up by the same types of problems that would have delayed licensing of Limerick had its construction been completed in mid-1982, namely, unanswered intervention in the hearing process. A widespread allegation of incomplete or inadequate construction, brought by a determined group of intervenors, significantly delayed Diablo Canyon at the end of its licensing process. This was long after PG&E completed the redesign and strengthening of the plant to accommodate the increase in seismic design basis. My experience in managing the staff participation in both the Diablo Canyon and the Limerick hearings teaches me that if Limerick had attempted to obtain a license in mid-1982 with the large number of items that we know today would have been unresolved at that time, then, like Diablo Canyon, the Limerick hearing would have stretched on for the two years required by PECO to finish construction of ATWS, fire protection, environmental qualification and other modifications described by PECO witnesses Sproat and Helwig. In addition, PECO and the NRC would have required the same period of time to complete the performance, review and litigation of the Limerick PRA, as we demonstrated in PECO Exhibit RJM-1.

Q. But Dr. Hanauer says that special short term decisions were made on the basis of a partial PRA on Indian Point, and it was allowed to continue in operation while its full scope PRA was being litigated. Could this have been done at Limerick?

A. No, we tried that as NRC Staff reviewers and failed for several very good reasons. First, the burden of proof is different in the case of granting a new OL than it is in the case of revoking an existing OL. Second, the risk standard applied to Indian Point was less stringent than that applied to Limerick.

Generally, the regulations and the licensing history of NRC make it harder for a license applicant to demonstrate that a new plant is qualified to begin operation than it is for a licensee to demonstrate that there is reasonable assurance that an operating plant continues to pose no undue risk to public health and safety. For the two cases Dr. Hanauer compares, Indian Point and Limerick, the differences in the burdens of proof and the risk standards are striking. The NRC staff found no violations of NRC regulations in the 60 day risk study for Indian Point. As a result, this previously licensed plant was allowed to keep its operating license for the four years required to complete and review its more detailed PRA and the associated hearings to assure that Indian Point presented no disproportionate risk, including a comparison to NRC's proposed safety goal. The burden of proof was on the staff to use the PRA results to show why the Indian Point licenses should be revoked or modified. By contrast, the initial Limerick PRA showed that the plant did not have a risk level less than the WASH-1400 BWR (the risk standard for comparability with other plants) unless fire protection, environmental qualification, TMI amd ATWS backfits were fully implemented. (Note that Dr. Hanauer appears to concede this point on page 23 of OCA Statement 2A). Thus, Limerick was deprived of an OL until these items were satisfied and until the PRA litigation was essentially complete. These same implementation requirements would have prevailed at Limerick even if construction had been otherwise complete in mid-1982. At Limerick the burden of proof was on the applicant to show why the plant was safe enough to receive a license to operate.

Q. Specifically, why is Dr. Hanauer wrong when he says at page 4 and page 23 of OCA Statement 2A that some combination of preliminary work, staged work and reviews and license conditions would have been found to justify licensing of Limerick if construction had been complete in mid-1982?

A. The specific reasons are as follows:

1. Limerick was different than Indian Point and the PRA had to be litigated before starting operations, as I just explained.

2. Limerick was a plant with serious licensing issues, just like some others that were delayed in initial operation until almost all requirements were fully implemented. In Limerick's case, the serious issues stemmed from the high population density rather than allegations of poor design, poor QA, mismanagement, and other things that plagued other plants.

3. The Licensing Board and the Commission would have decided whether Limerick could operate with unfinished business. In this case, with a high population density, determined intervenors, and the same governor that was determined to stop the restart of TMI-1 because of loose ends, my judgment is that the Commission and the Board would not have negotiated such a deal as Dr. Hanauer describes.

Q. Can you summarize this portion of your testimony?

A. Yes. When Dr. Hanauer says at page 9 of OCA Statement 2A that the non-Mark II regulatory requirements cited by PECO could have been resolved by mid-1982 and when Mr. O'Brien in OCA Statement 1B relies on that assertion, they are both in error. Their assertions are based on unsupported generic arguments about licensing schedule negotiations at which they are inexperienced. Moreover, they have failed to account for the highly plant specific basis upon which NRC always has and continues today to conduct such decision making.

III. MARK II CONTAINMENT ISSUES

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

A. To correct impressions left by the surrebuttal testimony of Dr. Hanauer regarding the regulatory aspects of the Mark II containment issues.

Q. What aspects of Dr. Hanauer's surrebuttal testimony on the Mark II issues are incorrect?

A. There are two within my purview as an expert on NRC regulations, as follows:

1. A failure to account for all of the companies and institutions, including the federal safety regulators in the AEC and the ACRS, that participated in the initial testing, review and approval of the pressure suppression containment concept and thus share in any responsibility for failing to uncover certain hydrodynamic loads before they occurred in operating plants.

2. A failure to account for the numerous other firms, boards, ACRS members, AEC license reviewers, national laboratory scientists and others who reviewed pressure suppression systems on all BWRs licensed before the early 1970s. They also share in any responsibility for failing to appreciate the importance of hydrodynamic loads.

Q. Were two hydrodynamic phenomena identified in the pre-1961 Humbolt Bay tests as described by Dr. Hanauer at pages 28-30 of OCA Statement 2A?

A. Yes, but Dr. Hanauer fails to note that the tests were conducted by Pacific Gas and Electric Company, interpreted by PG&E, GE and Bechtel, reviewed by the ACRS and AEC and then eventually accepted as part of the licensing basis of the

Humbolt Bay containment. Appendix IV of the Humbolt Bay Final Hazards Summary Report, provided in PECO Exhibit SL-3 was a public document that was widely relied on for many years as a part of the licensing basis for many other plants. Since this is the only evidence provided by Dr. Hanauer and since it does not support his theories of technical error or coverup, then these theories are unsupported in this proceeding.

Q. Does this also mean that Dr. Hanauer's Statement that "NRC played no role in the development of the Mark II concept and load definitions" is also in error?

A. It certainly does. The NRC was created from the AEC, the ACRS is the same institution today that it was in 1960, and the Mark II containment evolved from the Humbolt Bay design. Therefore NRC/AEC played a role in the development of the Mark II, first without accounting for hydrodynamic loads and later by a very stringent accounting of those loads, as I described in my rebuttal testimony. Simply said, the government approved the development work and, along with others, shares in any imperfections.

Q. Can you summarize this section of your testimony?

A. The best summary is still the one provided by the Administrative Law Judge in the Susquehanna rate proceedings for PP&L and quoted at page 26 of my rebuttal testimony in PECO Statement 9A. The Humbolt Bay FHSR provided in PECO Exhibit SL-3 supports that statement by showing that others had a chance to detect the loads earlier and also failed. Dr. Hanauer has provided no other evidence. Thus, as the ALJ said:

> "There is no support for the $38.6 million adjustment because (1) it is undisputed that PP&L expenditures to solve the containment problem were necessary to build a safe and licensable plant; (2) PP&L's efforts to

evaluate and incorporate the new loads in the design of Susquehanna were prudent and essential to commercial operation of the plant; (3) the record does not demonstrate that the containment problem was reasonably discoverable at an earlier point in time; (4) the problem was not in identifying the loads, but determining the impact of the loads on the plant and how to design for the impact; and (5) there is no showing that General Electric was imprudent in the design of the containment."

Q. Does that conclude your testimony?

A. Yes, it does.

SCHEDULE 1

SHORTEST POSSIBLE INTERVALS BETWEEN LATE NRC REQUIREMENTS, NRC SUBMITTAL AND NRC SSER

REQUIREMENT	MAJOR REQUIREMENT ISSUED	SUBMITTAL TO NRC	INTERVAL SINCE REQUIREMENT (MONTHS)	NRC SSER ISSUED	INTERVAL SINCE SUBMITTAL MONTHS
ATWS	3/80 (NUREG-0460 VOL. 4)	5/83	26	8/83*	N.A.
CONTROL ROOM DESIGN REVIEW	11/80 (NUREG-0737)	8/83	33	10/84	14
ENVIRONMENTAL QUALIFICATION	2/80 (NUREG-0588)	10/83	32	10/84	12
FIRE PROTECTION	7/81 (CMEB 9.5-1 REV. 2)	10/83	27	10/84	12
SPDS	2/81 (NUREG-0696)	6/83	28	10/84	16
RMMS	11/80 (NUREG-0654)	6/83	31	10/84	16
TMI ELECTRICAL ITEMS	11/80 (NUREG-0737)	5/83	30	10/84	16
AVERAGE ACTUAL INTERVAL			30 MONTHS		15 MONTHS
SHORTEST POSSIBLE INTERVAL			15 MONTHS		12 MONTHS

*INTERIM APPROVAL OF ATWS ALTERNATE 3A; FORMAL APPROVAL UNDER 10 CFR 50.62, ISSUED ON JUNE 26, 1984, STILL PENDING.

SCHEDULE 2

EARLIEST POSSIBLE DATES LEADING TO ASLB PIDs ON CONTENTIONS RELATED TO LATE NRC REQUIREMENTS WITH DEFERRED IMPLEMENTATION

REQUIREMENT	MAJOR REQUIREMENT ISSUED	NRC SUBMITTAL	NRC SSER	FILE TESTIMONY	ASLB PID ISSUED
ATWS	3/80	9/81	9/82	10/82	11/82
CRDR	11/80	2/82	2/83	3/83	4/83
EQ	2/80	5/81	5/82	6/82	7/82
FIRE PROTECTION	7/81	10/82	10/83	11/83	12/83
SPDS	2/81	5/82	5/83	6/83	7/83
RMMS	11/80	2/82	2/83	3/83	4/83
ELECTRICAL ITEMS (TMI)	11/80	2/82	2/83	3/83	4/83

PECO STATEMENT NO. 35

PH 3-14-86
Nb1

PENNSYLVANIA PUBLIC UTILITY COMMISSION v.
PHILADELPHIA ELECTRIC COMPANY,

DOCKET No. R-850152

REBUTTAL TESTIMONY OF
FRANKLIN D. SANDERS

LIMERICK 1 AND COMMON PLANT
FINANCIAL MANAGEMENT

February 19, 1986

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February 28, 1986

Honorable Joseph H. Matuschak
97 East Main Street
Uniontown, PA 15401

Re: Pennsylvania Public Utility Commission v.
Philadelphia Electric Company
Docket No. R-850152

Dear Judge Matuschak:

Enclosed please find one original and two copies of the Errata of the Philadelphia Electric Company Rebuttal Testimony of David R. Helwig, (PECO Statement No. 5A), in the above-captioned proceeding.

In addition, copies of the schedules with the designated corrections are being provided.

Sincerely,

Jack Jerrett

Jack E. Jerrett
Counsel for Philadelphia
Electric Company

mab
Enclosures

cc: All Active Parties

BEFORE THE
PENNSYLVANIA PUBLIC UTILITY COMMISSION

PENNSYLVANIA PUBLIC UTILITY COMMISSION	:	
	:	
v.	:	Docket No. R-850152
	:	
PHILADELPHIA ELECTRIC COMPANY	:	

ERRATA
of
PHILADELPHIA ELECTRIC COMPANY
REBUTTAL TESTIMONY
OF DAVID R. HELWIG,
PECO STATEMENT NO. 5A

Page	Line	Errata
Schedule 2	28	Change "88,600" to "117,000"; Change "-57,100" to "-28,700"; Change "(-39%)" to "(-20%)"
Schedule 2	30	Change "59,600" to "53,200"; Change "-13,900" to "-20,300"; Change "(-19%)" to "(-28%)"
Schedule 2	32	Change "428,000" to "482,000"; Change "-29,000" to "-83,000"; Change "(-7%)" to "(-17%)"
Schedule 2	34	Change "14,300" to "20,700"; Change "8,800" to "15,200"; Change "(160%)" to "(276%)"
Schedule 2	36	Change "6,563,000" to "6,567,000"; Change "-978,000" to "-982,000";
Schedule 2	38	Change "162,500" to "163,400"
Schedule 3	24	Change "88,600" to "117,000"
Schedule 3	26	Change "59,600" to "53,200"; Change "-4,100" to "-10,500"; Change "(-6.4%)" to "(-16.5%)"
Schedule 3	30	Change "14,300" to "20,700"

SCHEDULE 2

COMPARISON OF UNIT 1 AND COMMON COMMODITIES FOR LIMERICK 1 AND SUSQUEHANNA 1

Commodity	Units	LGS	SSES	Differences	
Concrete	CY	203,000	185,000	18,000	(10%)
Structural Steel	Tons	16,341	15,937	404	(3%)
Large Pipe	LF	350,000	270,000	80,000	(30%)
Lg. Pipe Hangers	ea.	13,600	10,000	3,600	(36%)
Small Pipe	LF	213,000	180,000	33,000	(18%)
Instrument Pipe	LF	36,000	11,000	25,000	(227%)
Instrument Tubing	LF	117,000	145,700	-28,700	(-20%)
Cable Tray	LF	53,200	73,500	-20,300	(-28%)
Metal Conduit	LF	399,000	482,000	-83,000	(-17%)
Gutter	LF	20,700	5,500	15,200	(276%)
Wire & Cable	LF	5,585,000	6,567,000	-982,000	(-15%)
Connections	ea.	166,206	163,400	3,706	(2%)
Ductwork	lbs	2,335,422	1,289,800	1,045,622	(81%)
Duct Hangers	lbs	768,874	522,200	246,674	(47%)

SCHEDULE 3

COMPARISON OF UNIT 1 AND COMMON COMMODITIES FOR LASALLE 1 AND LIMERICK 1

Commodity	Units	LGS	LaSalle**	Difference	
Concrete	CY	203,000	303,600	-100,600	(-33.1%)
Structural Steel	Tons	16,341	N/A	—	
Large Pipe*	LF	249,000	182,300	6,700	(36.6%)
Lg. Pipe Hangers	ea.	13,600	12,900	700	(5.4%)
Small Pipe*	LF	203,000	130,000	73,000	(56.2%)
Instrument Piping	LF	36,000	N/A	—	
Instrument Tubing	LF	117,000	N/A	—	
Cable Tray	LF	53,200	63,700	-10,500	(- 16.5%)
Metallic Conduit	LF	399,000	342,300	56,700	(16.6%)
Gutter	LF	20,700	N/A	—	
Wire & Cable	LF	5,585,000	5,152,000	433,000	(8.4%)
Connections	ea.	166,206	174,500	-8,294	(- 4.8%)
Ductwork	Lbs	2,335,422	N/A	—	
Duct Hangers	Lbs	768,874	N/A	—	

N/A - Not Available
* - Excludes yard piping and CRD piping.
** - All commodities are estimated on the basis of 60% of total plant, except concrete which is an actual Unit 1 and Common accounting.

REBUTTAL TESTIMONY OF FRANKLIN D. SANDERS

Q. Please state your name and business address for the record.

A. Franklin D. Sanders, Park Avenue Plaza, New York, New York 10055.

Q. By whom are you employed, Mr. Sanders, and in what capacity?

A. I am employed by The First Boston Corporation, an investment banking firm, as a Managing Director in the Utility and Telecommunications Group in the Investment Banking Department.

Q. Will you please outline your educational and business background?

A. I attended the Longmeadow and Wellesley, Massachusetts, public schools and was graduated from Amherst College in 1957 with a Bachelor of Arts degree. I then attended the Harvard University Graduate School of Business Administration where I concentrated in finance. Following my graduation from the Business School with the degree of Master in Business Administration in 1959 and six months' active duty service with the Massachusetts National Guard, I entered The First Boston Corporation training program in January 1960. Since January 1961, I have been assigned to First Boston's Investment Banking (formerly Corporate Finance) Department. I was appointed an Assistant Vice President in 1964, elected a Vice President in 1967, and elected a Director of the Corporation in 1976. My position was redesignated as Managing Director in April 1978. I am a member of the New York Society of Security Analysts, Inc. and a lecturer before the semi-annual seminars for public utility executives and regulators conducted by Irving Trust Company.

Q. Will you state the nature of your firm's business?

A. The First Boston Corporation for many years has been a leading underwriter and distributor of securities, and a very large portion of its business has been identified with practically every phase of utility financing, including the marketing of new securities at competitive bidding and on a negotiated basis. The latter includes issues offered publicly and those placed directly with institutional investors. The business from time to time also includes financial advisory services to utility and other companies involving the design of financing programs.

For many years, The First Boston Corporation has been one of the several leading underwriters of corporate and utility securities according to the total dollar amount of public offerings of taxable securities managed. The First Boston Corporation, alone or jointly with others, has managed underwriting groups or acted as agent in direct placements in connection with the marketing of securities of public utilities and public utility holding companies, industrial companies, foreign corporations and governments, the International Bank for Reconstruction and Development, tax-exempt revenue entities and United States Government Agencies.

Q. What are your specific duties with The First Boston Corporation?

A. I am a member of the Utility and Telecommunications Group in the Investment Banking Department, responsible for financing and financial advisory activities for approximately 20 clients in the electric, gas and water utility industries. In connection with my work, I am extensively involved in the design of financing programs for public utilities, with the analysis and structuring of proposed issues with a view toward their being successfully marketed. This work requires close communication with our Capital Markets Group, securities dealers, insurance companies, pension funds, investment trusts and other institutional investors as to their requirements for securities to be purchased by them.

I have presented testimony in connection with utility matters before the

United States Environmental Protection Agency, Federal Energy Regulatory Commission, Connecticut Public Utilities Control Authority, Massachusetts Department of Public Utilities, New Jersey Board of Public Utility Commissioners, New York Public Service Commission, North Carolina Utilities Commission, Oklahoma Corporation Commission, Pennsylvania Public Utility Commission, Utah Public Service Commission, and Public Service Board of Vermont.

Q. What is the subject matter of your testimony?

A. My testimony addresses three topics. First, I will review and evaluate the ability of Philadelphia Electric Company (PECO) to finance the earlier completion of Limerick Unit 1 on the assumption that the revenue requirements necessary to maintain historic earnings per share could have been obtained. Second, I will assess the risks inherent in PECO's attempting to obtain the necessary capital in the securities markets in light of market conditions and the position and reputation of PECO. Finally, I will review and evaluate the ability of PECO to attract capital in the 1986-1990 period if the position of certain intervenors in this case were to be adopted by the Pennsylvania Public Utility Commission (Commission).

Q. What is the source of the financing plan supporting the earlier completion of Limerick Unit 1?

A. First, the plan for financing the earlier completion of Limerick Unit 1 is based upon the testimony of James J. O'Brien on behalf of the Office of the Consumer Advocate as to the revised direct construction expenditures necessary to meet a fuel load date of July 31, 1982. Second, PECO has prepared schedules: (a) showing financing designed to fund the revised direct construction expenditures and Allowance for Funds Used During Construction (AFUDC) amounts revised in relation thereto, (b) evaluating the ability of PECO to issue the necessary securities and calculating the revenue requirement to support maintenance of historic earnings per share, and (c) showing analytical ratios based upon financing with and without the additional revenues. The supplemental financing plan, as well as the parameters for its design, are explained by Joseph F. Paquette, Jr. in Statement No. 3A, and the analytical data is summarized in Tables 3 - 6 attached thereto.

Q. Please explain the importance of the assumption that revenue requirements be obtained to support maintaining historic earnings per share.

A. Testimony on behalf of the Office of Consumer Advocate claims that significant savings in construction cost of Limerick Unit 1 and Common Plant could have been achieved by pursuit of a more aggressive construction schedule. First, a more aggressive construction schedule and the associated acceleration of disbursement of direct costs would have required significant additional financing by PECO during the period of accelerated construction. (See Statement 3A, Table 4.) Second, the revised schedules of direct costs and associated financing would have resulted in a changed schedule of AFUDC for the years commencing with 1975. (See Statement 3A, Table 3.) Third, the combination of revised AFUDC and the capital costs of a new financing schedule would have resulted in changes in earnings, with significant and increasing reductions in earnings in the years commencing with 1979. Such reductions in earnings would have adversely affected the ability of PECO to obtain the necessary financing. The assumption of unchanged earnings per share is one method for establishing the minimum revenue required to support the revised financing program, as any financing program must be founded upon the ability to attract the necessary equity to support senior securities. However, as I will point out below, the maintenance of earnings per share historic dollar values does not, without more, maintain the quality of such earnings because of the substantial component of AFUDC.

Q. Is the revised financing plan developed by PECO the only means by which financing might have been obtained for the additional expenditures?

A. No. There are many alternative plans by which PECO might be hypothesized to have obtained the necessary additional financing. However, I believe the revised financing plan to be a reasonable plan, designed to obtain the necessary funds by intelligent choices among the alternatives available at each point in time when additional financing was indicated, while preserving the financing actually accomplished during the affected years together with conditions permitting such actual financing.

Q. Do you believe that any cheaper financing plan could have been devised consistent with maintaining PECO's ability to finance?

A. While it is difficult to provide a categorical answer because of the innumerable permutations which could be devised, I believe that the PECO plan minimizes financing cost consistent with maintaining the ability to finance.

Q. Please explain the principal parameters affecting the revised financing program, as you understand them.

A. The principal parameters include: (a) maintenance of the original financing program as a base, (b) financing in amounts considered appropriate in the marketplace at the time and for the security concerned, (c) the mortgage requirement for additional bonds that historic earnings (as therein defined) be twice interest requirements on bonds to be outstanding, (d) the charter requirement that earnings (as therein defined) be one and one-half times the sum of interest and preferred dividend requirements reflecting the financing, (e) limiting short term debt to appropriate levels and (f) maintaining appropriate capitalization ratios.

Q. Would PECO's financial standing have improved or deteriorated during the 1975-

1982 period if it had undertaken the additional financing shown in the PECO study and had obtained additional revenues sufficient to maintain historical earnings per share, and why do you reach your conclusion?

A. It is my judgement that PECO's financial standing would have deteriorated, even assuming it received the additional revenues required to maintain earnings per share at historical levels. Despite the conservative selections of financings, there would have been a reduction in interest coverages calculated according to rating agency methodology, particularly in coverages in which earnings available for interest is reduced by the amount of AFUDC included therein. Table 1 attached hereto presents such data for the eleven year period involved. Furthermore, as also shown in Table 1, the proportion of earnings comprised by AFUDC would have increased as a result of the accelerated construction and the proportion of construction expenditures generated from internal sources would also have been reduced. The deterioration in financial standing would have led, in my judgement, to earlier reductions in the ratings of PECO securities by the recognized rating agencies.

Q. Are there substantial uncertainties in PECO's ability to raise the additional capital required to meet the hypothesized July 31, 1982 fuel load date, based on the additional financing plan proposed by PECO and on the assumption that additional revenues were obtained sufficient to maintain historic earnings per share?

A. Yes. Given the nature of market conditions in the securities markets during this eight-year period, with highly volatile securities prices and increasing interest rates and the adverse investor reaction to developments in nuclear power, to more stringent regulation of the nuclear power industry and to the deteriorating credit of public utility companies, there can be no assurance that the substantial amounts of additional capital could have been raised as required without incurring substantial additional costs of capital. In other words, I see a very small margin of safety in the PECO additional financing plan. Each individual financing tests the limits of the securities markets and their ability to absorb PECO's securities. In particular, the repeated sales of 9 million shares of common stock in four issues in 1981 and 1982 would have strained the capacity of the common stock market for PECO's shares.

Q. Mr. Sanders, can you estimate the additional costs of financing which PECO would have incurred as a result of financing the increased construction expenditures required to advance fuel loading of Limerick Unit 1 if PECO were limited to its historic revenues?

A. Yes. I believe that PECO would have incurred higher costs for financing accomplished after 1978, as the cumulative impact of the additional funds requirement became measurable in its financial statements. It is probable that the rating agencies would have anticipated such deterioration in financial position and acted to reduce ratings earlier, with consequent increases in financing costs, as early as 1976.

Based upon my expectations of rating reductions, I would expect an increase in debt interest cost and preferred stock dividend costs of at least one-half of one percent of principal amount or par value, respectively, and that the realized price of PECO's common stock would have been reduced by at least 50¢ per share, so as to produce a somewhat higher yield to stock purchasers. It is difficult to be precise given the nearly limitless permutations of possible ratings and market conditions. Actual cost increases could have been over one percent and stock price reductions over $1 per share under poor market conditions or if PECO ratings were to have been reduced to below investment grade. Nevertheless, it is a certainty that the failure to obtain revenues sufficient to maintain earnings per share would have had a debilitating effect on PECO's financial standing and increased the cost of its financing.

Q. Mr. Sanders, have you reviewed the testimony of intervenors as to the appropriate PECO revenue requirement and analyzed the effect of Commission acceptance of their proposals upon PECO's financial standing and ability to attract capital in the years 1986-1989?

A. Yes, I have reviewed the testimony of the several intervenors which have offered phase-in or other revenue proposals, as well as analytical models prepared by PECO as to the proposals of the Commission Staff, Office of the Consumer Advocate (OCA), City of Philadelphia (City), and Governor's Energy Council (GEC).

Q. Please summarize your analysis and findings.

A. Based upon this review, particularly of the schedules showing the effects upon income, coverage, cash flow and capitalization, I believe that adoption of any one of the intervenor proposals by the Commission in its Order at this docket would be injurious to the financial standing of PECO and its ability to attract capital for the forseeable future. The proposals of the City, OCA and the Commission Staff are the most injurious in their effects upon PECO.

I am concerned about (a) the potential inability of PECO to obtain a full cash return upon its investment in Limerick Unit 1 and common facilities, (b) the predominant proportion of earnings which will be comprised of AFUDC and deferred revenues, both of which are non-cash items, (c) the potential write-off which will be required by Financial Accounting Standards Board Statement No. 71 (Exposure Draft) and which will severely reduce common stock equity, and (d) the deterioration of various analytical measurements of financial standing.

I expect that the rating agencies will respond to a Commission Order adopting any of the intervenor positions by downgrading various PECO securities. Based upon currently existing PECO ratings of Baa3 by Moody's Investor's Service and BBB- by Standard & Poor's Corporation, which are the lowest investment grade ratings, PECO securities would all be rated below investment grade. As a consequence, PECO would incur significant increases in its costs of capital and reduction in the breadth of market for its securities, with consequent reduction in its financing flexibility.

Q. Will you please explain your reasoning, with reference to the particular intervenor proposals?

A. Tables 1 and 2 appended to Statement No. 3A provide a useful frame-work for analysis of the four intervenor proposals. I shall refer to these Tables in responding to the question. Taking the proposals from the most injurious to the least injurious, I will analyze the proposals of the Commission Staff, OCA, City and GEC.

The Commission Staff position would result in a rate base disallowance in excess of $1 billion, deferral of rate base recognition of $452 million, and reduction in the allowed return on equity. Consequences for PECO's financial standing would include inadequate earnings, with returns on equity of only ±10% in 1987-1989, earnings per share below the prevailing $2.20 dividend rate per share in those years, a continuing high level of AFUDC in earnings, poor internal generation of funds for construction, and declining interest coverages. PECO's financial standing would continue to deteriorate over the four years.

The OCA proposal would have comparable results, with rate base disallowances and deferrals of almost $1.5 billion and a proposed return on equity of only 14%. Again, an average return on equity of less than 10% in the years 1987-1989 would produce earnings per share well below PECO's existing common stock dividend rate. Earnings quality would be poor, with a high component of AFUDC, and internal generation of construction funds would be extremely low, averaging a negative figure for the four years 1986-1990. Declining interest coverages over the four-year period portray a situation of continuing deterioration.

The position of the City, while appearing to support the full revenue request, would phase-in the revenues over so long a period as to have significant adverse effects upon PECO. If the phase-in is accepted on its face -- e.g., the deferral of revenues is accepted for financial statement reporting purposes -- PECO's return on equity and earnings per share will include substantial non-cash amounts of AFUDC and deferred revenues. Internal sources of funds for construction purposes will be inadequate, only a few percent or will be negative. As discussed below, the City's proposal is unlikely to be accepted for book reporting purposes and, consequently, will have other adverse effects.

The proposal of GEC is the most reasonable of the four analyzed in detail; however, it is founded upon an unsound principle, sinking fund depreciation, which has adverse effects upon cash flow. The proposal produces effects, as measured by the financial ratios of Table 2 to Statement 3A, which are comparable to PECO's own proposal, except as relates to the internal generation of funds for construction. The GEC proposal produces inadequate funds flow as a result of the lower depreciation dollars reflected in rates for service. Again, as with the City proposal, it is also unlikely to be accepted for financial reporting purposes. Moreover, the sinking fund depreciation method is unsound because of the long postponement of recovery of capital inherent in the method and the poor internal cash flow which results. While sinking fund depreciation was prevalent in the early years of the investor-owned utility industry, it has fallen into disrepute. Based upon my experience, the method is not used among investor-owned companies except for a very small number of special purpose project companies where depreciation is designed to match sinking fund requirements under level debt service type sinking funds.

Q. You have referred to possible problems with "book" or financial statement reporting for various proposals. Will you please explain?

A. Mr. David J. Farling has discussed the ramifications of FASB Exposure Draft No. 71. It is my understanding that, based upon the Exposure Draft, the intervenor proposals would either, in the case of the Commission Staff and OCA proposals, result in a write-down of assets or, in the case of the City and GEC phase-in proposals, result in denial of recording of deferred revenues, in the last case as a result of a slower than straight-line depreciation rate. Mr. Paquette has testified (Statement No. 3A) as to probable writedown amounts approaching $1 billion. A writedown of that amount and the probable reduction or omission of common and, possibly, preferred stock dividends would have drastic effects as discussed below. The inability to record deferred revenues would result in PECO having to report severely reduced earnings during the early, deferral years. It would be faced with the necessity of raising substantial amounts of capital to fund the deferrals and meet refunding requirements with inadequate earnings to meet the issuance tests of the mortgage indenture and charter and an inability to access the common stock market on reasonable terms.

Q. Can you quantify the increase in costs which you expect and illustrate the reduction in financing flexibility which you predict based upon Commission adoption of an intervenor position in its Order in this case?

A. Yes. Based upon market conditions in early 1986, I would estimate a minimum increase in cost of debt of one-half of one percent if ratings are reduced to the Ba2 (Moody's) and BB (S&P) rankings. Furthermore, I expect that PECO would find it difficult to raise debt with a maturity over 15 years and, if conventional 30 year debt were to be available, that its cost would be at least one percent higher than if ratings had not been reduced. Inability to issue mortgage bonds because of inadequate coverage under the indenture and the need to resort to debenture financing would result in a further increase in cost, additional to the above, of one-half percent to one percent.

PECO would also experience a marked increase in its cost of preferred stock financing, of at least one-half of one percent for intermediate maturity preferred stock and of at least one percent for perpetual preferred stock. Furthermore, PECO would find a significant narrowing of the preferred stock marketplace. Maximum issue sizes would be reduced and certain preferred stock structures probably would not be available. Moreover, if PECO's earnings are inadequate to meet the charter test, or it records a substantial writedown of assets, preferred stock financing may not be available.

If, as Mr. Paquette suggests, PECO were forced to reduce or omit its common stock dividend, PECO would find the market for new common stock prohibitively expensive, if even open for new issue financing. The market price of PECO common stock can be expected to fall drastically if the dividend is reduced or omitted. Potential common stock purchasers (which invariably include existing holders of a company's stock) will be leery of committing new funds to PECO common stock pending a demonstration of PECO's ability to continue the common stock dividend for several quarterly periods. Moreover, the conditions leading to reduction or omission of the common stock dividend are hardly conditions attractive to common stock investment, as investors will not expect the opportunity to receive a fair return upon investment.

Q. Can you offer an opinion concerning other intervenor proposals?

A. The proposal of Witness Chernick, on behalf of Utility Users Committee/University of Pennsylvania is an incompletely developed proposal to tie rates to benefits of Limerick Unit 1. It is not sufficiently developed for me to analyze.

The proposal of Witness Falkenberg, on behalf of Philadelphia Area Industrial Energy Users Group, would reduce common plant by 50%, adopt sinking fund depreciation, tie rates to "guaranteed fuel savings", and phase in revenues, without interest on the basis of one-third of additional revenues per year. While I have not been provided with a study of the effects of this proposal, its elements are sufficiently similar to other proposals for me to conclude that it would be severely adverse to PECO's financial standing by resulting in inadequate earnings and cash flow and the probability of disallowance of deferred revenue accounting.

I have not seen the testimony of Witness King on behalf of Philadelphia Business Utility Users Group, and do not have an opinion at this time.

Q. Mr. Sanders, what are your conclusions concerning PECO's ability to finance its capital requirements if, as a result of the adoption by the Commission of an Order in this case as proposed by one or more of the several intervenors, PECO should (a) incur a significant writedown of assets or (b) be unable to record deferred revenues according to FASB Statement 71 or (c) be denied a significant portion of the revenues related to commercial operation of Limerick Unit 1?

A. Under any of the three circumstances you cite, I believe PECO would be severely hampered in raising the substantial amount of capital it will require. Unfortunately, PECO will be required to raise substantial additional capital merely by reason of any significant deferral of revenues in this case. As testified to by Mr. Paquette, PECO studies indicate possible additional cash requirements related to intervenor phase-in proposals of $1.5 billion, on top of a base capital requirement of $1 billion during the same time period. Therefore, PECO faces the risk of enormous capital requirements under circumstances of severely reduced financial standing.

Q. Mr. Sanders, does that complete your prepared rebuttal testimony?

A. Yes, it does.

TABLE 1

Philadelphia Electric Company
Selected Analytical Ratios
1975 - 1985

(1) Year	(2) Pre-tax Interest Coverage Incl. AFUDC	(3) Pre-tax Interest Coverage Excl. AFUDC	(4) AFUDC As % Earnings for Common	(5) Overall Coverage of Interest and Preferred Dividend Reqts
[Ratio based on historic financial data]				
1975	2.44x	1.98x	62.0%	1.60x
1976	2.59	2.08	61.8	1.66
1977	2.50	1.97	64.9	1.65
1978	2.44	1.93	64.4	1.64
1979	2.20	1.63	75.7	1.61
1980	2.14	1.52	84.3	1.60
1981	2.15	1.52	84.4	1.63
1982	2.37	1.75	76.5	1.70
1983	2.37	1.62	85.8	1.74
1984	2.44	1.62	86.6	1.79
1985	2.31	1.35	99.7	1.80
[Ratios based upon hypothetical financing program with additional revenues (1)]				
1975	2.43x	1.95x	63.2%	1.60x
1976	2.57	2.04	63.8	1.66
1977	2.45	1.89	68.9	1.65
1978	2.36	1.80	71.0	1.93
1979	2.06	1.44	85.7	1.59
1980	1.99	1.31	95.9	1.58
1981	2.04	1.31	97.5	1.61
1982	2.21	1.48	91.5	1.66
1983	2.18	1.38	92.5	1.70
1984	2.98	2.79	20.7	2.17
1985	2.97	2.77	23.8	2.25

(1) Additional revenues as designed to maintain historic earnings per share.

PECO STATEMENT NO. 35A

P H 3-14-86

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PENNSYLVANIA PUBLIC UTILITY COMMISSION

v.

PHILADELPHIA ELECTRIC COMPANY

DOCKET NO. R-850152

DOCKETED
MAR 24 1986

SUR-SURREBUTTAL TESTIMONY OF

FRANKLIN D. SANDERS

DOCUM... FOLDER

RE: LIMERICK 1 AND COMMON PLANT

FINANCIAL MANAGEMENT

March 1986

SUR-SURREBUTTAL TESTIMONY OF FRANKLIN D. SANDERS

Q. Are you the same Franklin D. Sanders who previously submitted testimony in this proceeding?

A. Yes, I am. I previously submitted rebuttal testimony identified as PECO Statement No. 35.

Q. Mr. Sanders, what is the purpose of your sur-surrebuttal testimony?

A. I will respond to certain statements in the surrebuttal testimony of Gregory A. Palast.

Q. Mr. Palast states in his testimony, in response to a question which assumes (a) a Commission Order based upon the City proposal for a seven-year phase-in and ten-year recovery period for rates in this proceeding, (b) adoption, subsequent to the Order, of FASB Statement No. 71, (c) no amendment of the Order in response to Statement No. 71, and (d) PECO cannot record deferred revenues related to the phase-in of rates, that (i) "investors would know that the lower earnings, in comparison to other utilities, are the result of a change in the method of reporting income, not a change in financial condition" and (ii) that "[t]o the extent FASB-71 changes result in underreporting income, investors will take that into account in comparing PECO to other utilities." What, in fact, would be the investors' response to the reporting of earnings in such fashion?

A. First, one must distinguish between the principal classes of investors in PECO securities before conclusions can be drawn as to investor reaction to the sequence of events outlined in the question. PECO securities are owned by two general groups of investors, individuals which tend to own PECO common and preferred stock and institutions which own the substantial portion of PECO senior securities as well as significant amounts of PECO common and preferred stocks.

Second, one must draw attention to invalid premises. Mr. Palast offers no basis for his conclusion that earnings reported without the benefit of deferred revenues do result in "under-reporting income." It is plainly arrogant for Mr. Palast to hold himself out as an expert in accounting practice against financial statements prepared in accordance with the principles which have been or may be adopted by the Financial Accounting Standards Board, as the standards-setting entity for the accounting profession.

It is my experience that individual investors generally rely upon financial statements as presented to them in the issuer's annual and interim reports. Such investors do not have the background and experience to apply relatively sophisticated accounting techniques to the restatement of the financial statements which they use. The effect of the series of events postulated, capped by the denial of deferred revenue accounting, will be income statements of PECO which show earnings below its dividend, at least in the early years of the phase-in. Such investors will react to the failure of PECO to earn its dividend by reducing or eliminating their holdings of PECO common and preferred stocks, with consequent reduction of the market prices of the outstanding stocks and reduced salability of new issues of such securities.

Institutional investors may be expected to understand PECO's inability to record deferred revenues. However, it is totally incredible to expect that investors will make pro-forma adjustments to record such non-cash revenues or their effects upon earnings. If the series of events postulated occurs, with an extended phase-in plan <u>not</u> reflected in deferred revenues, financial statements prepared by PECO in accordance with FASB-71 will stand in mute testimony to the failure of regulation in Pennsylvania to provide a fair return to investors in PECO securities.

It is my experience that institutional investors can and do make adjustments to earnings to remove non-cash revenue items, such as unbilled revenues and AFUDC, and/or to restore deferred expenses, such as deferred fuel expense, in order to analyze earnings on an "acid-test" basis. The reverse seldom if ever occurs. Sophisticated investors tend to look at earnings on both a reported and an "acid-test" basis in making judgments leading to buy and sell decisions. Rating agencies follow similar procedures in analyses leading to ratings of a company's debt and preferred stock. In PECO's case, under the conditions assumed, they merely start with a realistic reported earnings figure. It is naive and unrealistic to expect investors to re-inflate earnings with non-cash items not to be received in cash for many years.

Q. Mr. Palast states "[i]f he did not subtract deferred revenue in analyzing my recommendation, Mr. Paquette would find that, as discussed in my original testimony, most major indicators are not seriously eroded by extending the phase-in period by four years." Is Mr. Palast correct in his analysis?

A. No. Mr. Palast again operates under the erroneous assumption that he can make accounting policy and that investors will accept that policy. Mr. Paquette's conclusions in PECO Statements 3A and 3B based upon his Table 2 are correct; PECO's ability to attract capital would be severely eroded by adoption of the City phase-in plan, a plan which extends the recovery period by a total of <u>seven</u> years, not the four years referred to by Mr. Palast.

Q. Mr. Sanders, does that complete your sur-surrebuttal testimony?

A. Yes, it does.

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

DOCKETED
MAR 2 4 1986

REBUTTAL TESTIMONY
OF
H. WILLIAM VOLLMER

MARK II CONTAINMENT ISSUES

FEBRUARY 19, 1986

Rebuttal Testimony Of
H. William Vollmer

Q. Please state your name and business employment.

A. My name is H. William Vollmer, and my business address is Philadelphia Electric Company, 2301 Market Street, Philadelphia, Pennsylvania, 19101. I am the Supervising Engineer of the Structural Branch of the Engineering and Research Department.

Q. What is your educational background?

A. I received my Bachelor of Science degree in Civil Engineering from Bucknell University in 1968. I have also attended continuing professional education courses in the engineering field including:

Steel Design, Current Practice

Non Destructive Testing of Welds

BWR Simulator Training

Engineering Economics

Decision Analysis Kepner-Tregoe

In addition, I am a registered Professional Engineer in Pennsylvania.

Q. Are you a member of any professional organizations?

A. I am a member of the American Concrete Institute and the American Welding Society.

Q. Please describe your work experience that would qualify you to testify on matters related to Mark II containment design.

A. I was first employed by the Boeing Company, Vertol Division, as a Flight Test Technical Support Technician in 1966 and later as a Technical Support Engineer in 1968. In this capacity, I was responsible for developing flight maneuvers that would demonstrate the adequacy of the test aircraft when subjected to forces during flight. It was also my responsibility to develop instrumentation requirements to verify that the planned maneuvers were met, which activities required knowledge of instrumentation sensitivity and data reduction methodology.

In 1970, I was first employed by PECO as a Transmission Design Engineer. In this capacity, I was responsible for designing the high voltage transmission towers and foundations. This position also required project planning, scheduling, and resolving field construction problems.

My next assignment within PECO was as a Design Engineer in the Structural Engineering Branch of which I am now the Supervising Engineer. In this position, I was responsible for providing structural engineering support for all Philadelphia Electric Company facilities. This work ranged from designs of office buildings to modifications at generating stations.

It was in this capacity that I began working with the industry group (Mark II) in evaluating and achieving resolution of the hydrodynamics load problem.

Q. Please describe your role in the resolution of the Mark II Containment Program.

A. In February 1975, when the Philadelphia Electric Company became aware of the questions concerning adequacy of the Mark II containment design, a joint task force was formed to develop an action plan for solving these problems. This task force was made up of representatives from General Electric Company, Bechtel Power Corporation, Pennsylvania Power and Light Company, and the Philadelphia Electric Company (PECO). I was the PECO representative on this task force. Later that year, this group was expanded to include representatives from the other utilities with Mark II containments and also to include the Engineering firms that were designing these plants. This group which was formed in April of 1975, was known as the Mark II Owners Group and functioned from this time until 1982, at which time all generic issues related to the Mark II program were resolved with the NRC. In 1978, when this group was restructured to better manage the expanding load definition and resolution program, I was selected by the Group to act as technical manager for all tasks related to the Loss-of-Coolant-Accident (LOCA) load development work.

I was also responsible for directing the work within PECO related to evaluating the ability of the Limerick Generating Station (LGS) containment and adjacent structures to withstand the Mark II loads. The final product of this work was the preparation and issuance of the three volume Limerick Containment Design Assessment Report which was submitted to the NRC as a licensing document. This document contained the summary analysis of the LGS containment based on the information developed through the Mark II Program.

Q. Have you previously provided expert testimony?

A. Yes. I was the lead witness in Gas Pipeline Explosion contention hearings before the Atomic Safety and Licensing Board (ASLB) of the Nuclear Regulatory Commission (NRC). In addition, I provided testimony on the welding contention before the same Board. I have also testified before the Advisory Committee on Reactor Safeguards (ACRS).

Q. Have you had any technical papers published on the containment problems related to Mark II or the hydrodynamic load phenomenon?

A. Yes. This summer, I presented a paper titled "In-Plant SRV Discharge Tests and Their Use in Peach Bottom Structural Requalification" at the Sixth International Conference on Structural Mechanics in Reactor Technology in Brussels, Belgium.

This paper was based on testing and analysis related to loads in containment which result from discharges of the safety relief valve (SRV).

Q. What is the purpose of your rebuttal testimony?

A. I will provide rebuttal to testimony presented by Dr. Stephen Hanauer on matters related to the Mark II containment. Specifically, my testimony addresses the following subjects:

1) The reasons why PECO selected a Mark II containment for LGS;

2) The Hanauer allegations that GE made a technical error in Mark II containment design;

3) The Hanauer allegation that the Mark II containment design was severely impacted by the discovery of the previously unappreciated hydrodynamic loads;

4) The effects of the Mark II loads issue on the Limerick completion schedule;

5) The incremental effects of the Mark II loads issue on Limerick cost; and

6) The comparability of the Limerick, Susquehanna and LaSalle Mark II experiences.

Q. Why did the Company choose a Mark II pressure suppression containment?

A. The Philadelphia Electric Company chose a Mark II containment structure because it was believed to be superior in both design and construction to the previous GE containment, i.e. the Mark I. The design was selected after a thorough evaluation which included consideration of recommendations from both GE and Bechtel that the Mark II concept be adopted. Our studies indicated that a $2 million savings in construction cost per unit could be expected from this design when compared to the Mark I design. A separate and important basis of this choice was the fact that the Mark II design provided more space inside containment than earlier designs.

PECO was not alone in choosing the pressure suppression containment concept. Approximately 40% of the nuclear reactors on order, under construction, or on line in 1975, were supplied by GE and included the pressure suppression containment.

Q. Did GE, in developing the Mark II containment design, fail to adequately measure, predict, or specify the loads and forces even though the technology was available?

A. No. In the early 1960's when the load basis for the pressure suppression containment system was developed, design was focused on providing a containment that was capable of withstanding the peak pressure and temperature resulting from loss of coolant accidents. It was felt that by applying appropriate safety factors to these loads, other accident-related conditions would be covered.

Furthermore, the data acquisition systems, computer modeling capability and supporting technology used to characterize the new hydrodynamic loads in the mid-1970's and later were not available in the late 1950's and early 1960's when the basic tests of pressure suppression containment were performed. The data acquisition systems available when the design basis was developed consisted of pen type recorders (Visicorders). These systems included instruments electrically connected to galvanometers with light reflecting mirrors to produce traces on light sensitive paper that moves by as the test takes place. Prior to initiating a test, it was necessary to speed the light sensitive paper up to a speed that would provide resolution of the test results. The magnitude of the load or force was then determined by manually measuring the amplitude of the trace on the paper with a scale. Since the time period for some of these forces (later identified as hydrodynamic loads) is measured in milliseconds, it was difficult to accurately determine the magnitude or frequency of these loads. Later technology, on the other hand, utilizes high speed recording of the results on magnetic tape. This information can be played back later at slower speeds, electronically filtered and enhanced to provide clear representation of the data. In addition, instrumentation available at the time of the original tests had limited ability to respond to these high speed phenomena. The analytical and modeling techniques employed to characterize the new hydrodynamic loads additionally requires the use of large, high speed computers which also were not available when the basic testing was performed.

The load definitions developed in the absence of the later available sophisticated equipment by GE, PG&E and Bechtel for the early designs was reviewed and approved by the AEC and the ACRS.

Q. How were the design problems detected?

A. Design concerns related to hydrodynamic loads were first noted during an event at the Wuergassen power plant in Germany in 1972 when the plant operators allowed a stuck open safety relief valve to continue to discharge to the suppression pool without initiating suppression pool cooling or shutting down the plant. Unstable condensation occurred in the suppression pool as the temperature in the pool approached boiling. This unstable condensation created pressure waves which caused damage to the concave shaped bottom of the steel suppression chamber.

The likelihood of having a similar situation in operating plants in the U.S. was extremely small, because limits were specified for the maximum suppression pool temperature. Procedures required scramming the reactor if these limits were exceeded. In addition, safety relief valve discharge lines in the U.S. plants were equipped with ramshead shaped discharge devices, not employed at Wuergassen, which further reduced the potential for this problem. Also the suppression pool shape and structural design of Wuergassen were unique.

During GE development testing on its latest containment product line (Mark III) the importance of dynamic loads in the suppression pool resulting from the simulated loss-of coolant-accident was discovered. An evaluation of the data from these tests indicated the need for a reassessment of the earlier containment designs since these designs had not explicitly considered the pool hydrodynamic loads.

Q. If the errors had not been detected, could the containment have failed?

A. No. There were no significant modifications to the containment structure required as a result of the Mark II Program. When the loads were identified, construction of the containment was suspended for approximately two months while the situation was evaluated. When construction resumed in June of 1975, the concrete walls of the suppression chamber were completed, followed by diaphram slab and upper containment walls, with only minor changes being made to the structure.

It was never established that hydrodynamic loads posed any significant threat to the integrity of the Mark II containment. In fact, operating BWRs with Mark I pressure suppression containments were allowed by the NRC to continue in operation for years while the new hydrodynamic loads were being addressed. Thus, the NRC had concluded that hydrodynamic loads did not threaten safety. The analysis undertaken as a part of the Limerick Probability Risk Assessment (PRA) has shown that the containment will maintain integrity up to 140 psi which is approximately 3.5 times the post accident pressure used in the original design. This fact alone substantiates the inherent strength of the Mark II reinforced concrete containment design. The Mark II containment was obviously superior in design because no major modifications, similar to those eventually required for the Mark I containment structures, were performed.

Q. Did the NRC require that actions be taken to address these loads?

A. Yes. In April of 1975, the NRC sent letters to each utility with a pressure suppression containment. These letters specified the loads to be considered in containment reanalysis and required that each utility propose a program to address the hydrodynamic loads concern. These requests from the NRC focused the attention of the industry and culminated in the formation of the owners groups to address these concerns in a cost-effective, generic program.

Q. How long did it take the utility group to develop the generic loads necessary for licensing the Mark II plants?

A. The technical problem was ultimately solved through a massive research and development program which cost over $40 million dollars. This was the most complex issue to face the BWR owners. The Mark II Owners Group was formed in April of 1975 and continued to function until 1982 to satisfy NRC's generic licensing requirements. By contrast, plants in Japan and Italy, which were not subjected to NRC reviews, addressed these issues in a different manner and were licensed in the late 1970's.

Throughout the course of the program, the NRC continually reviewed the progress of these programs. Because of the complexity of the problem, it was necessary for the NRC to employ scientists from Brookhaven National Laboratory and university professors from Princeton University and Massachusetts Institute of Technology.

Methods for addressing the hydrodynamic loads were approved by the NRC in a number of NUREG reports (NUREGs 0487, 0763, 0783, 0802, 0808, etc.). Final

NRC acceptance of the particularities of the Limerick containment design did not occur until the Limerick Design Assessment Report (DAR) was reviewed by the NRC and documented in Supplemental Safety Evaluation Report No. 3 of October 1984.

Q. Did the Mark II hydrodynamic loads issue delay the completion of LGS Unit 1 and Common?

A. As previously stated in PECO Exhibit 2, the containment problem by itself was not a cause for the delay of fuel loading. PECO always maintained the option to adopt the design basis and schedule of the lead plants (LaSalle and Susquehanna) if the Limerick schedule had indicated that final design information was needed at a particular time to support construction.

Q. Could the Mark II problem have been resolved at LGS by early to mid-1982?

A. Yes. However, this would have been undertaken at increased risk to the consumer because larger, more conservative loads would have had to be used which could have required extensive rework like that experienced by lead plants. As the contemporaneous plant construction schedules did not necessitate such risk taking, this would not have been the most efficient solution to the problem. PECO saved substantial costs by waiting to perform Mark II modifications at Limerick until near final loads had been adopted and until modifications were required to maintain the schedule dictated by other concerns.

Q. Did GE provide any sharing of the cost associated with the Mark II program?

A. GE provided approximately 35% of the total $43,505,000 Mark II Program cost, while the PECO share of the cost amounted to less than 10% or approximately $4,080,000.

Q. Did you participate in the preparation of the cost reconciliation for the impact of

Mark II as identified in response to IR-OCA-4-29?

A. Yes, I did. As the responsible engineer on the Mark II program, I was the person most familiar with the expenditures during the program and, as such, reviewed all costs identified to be attributable to this cause.

Q. Would most of the costs of the Mark II work have had to be spent if the load definition eventually used (i.e. including hydrodynamic loads) had been available prior to initial design?

A. Yes. Most of the costs would have been necessary even if the loads had been fully characterized prior to construction. Since GE passed all research and development costs through to its customers, some proportionate share of the Mark II development cost would have been billed to PECO regardless of when these costs were incurred.

Only limited costs associated with the base slab excavation for quencher base plate anchors, and the pedestal excavation for the downcomer bracing system would have been avoided.

Q. Have you quantified the potentially avoidable costs of rework resulting from the Mark II program?

A. Yes. I considered each item in the Mark II portion of the cost reconciliation (IR-OCA-4-29) and determined which of these costs would have been incurred even if the Mark II loads had been available prior to starting construction. In support of this evaluation, I labeled each category of the cost summary titled "RECONCILIATION ITEM--IMPACT OF MARK II" with the letters "A through "O". The following, which is based on this summary (see attached Schedule 1) describes the basis used to develop the estimate:

A. Not all early work was used in the final analysis of the plant. Some of the work supported the Lead Plant program exclusively, some of the lead plant work also was used by later plants such as Limerick, and some served primarily as the basis for the development of the final loads. Therefore, to be conservative, I determined that only 50% of the identified engineering work, or $2.55 million, would have been spent if the loads were available at the beginning of the project.

B. Using the same basis as above, I determined that 50% of this additional engineering work, or $0.45 million, would have been spent if the loads had been available at the beginning of the project.

C. This item covers the GE analysis of the nuclear steam supply system using the new loads. Limerick was the first plant with defined loads available at the time when it was necessary to perform this analysis. Therefore, I determined that all of the $1.9 million dollars would have been spent even if the loads had been identified at the beginning of the project.

D. Since this item is a repeat of work that would have been performed only one time had the final loads been available at the start of construction, none of this $0.1 million cost would have been incurred had these loads been identified at the beginning of the Project.

E. Since this item covers the development cost for computer models used in the Limerick analysis, all of the $1.1 million would have been required regardless of the timing of Mark II load definition.

F. This item covers the cost to incorporate "functional capability analysis" into the design. Since this requirement resulted from increased NRC requirements, the work would have been required no matter when the Mark II loads were identified and therefore, the entire $0.3 million would have been incurred.

G. This item covers sensitivity analyses based on lead plant loads. Since none of this work would have been required if loads had been available earlier, none of the $0.2 million would have been incurred if the Mark II loads had been identified earlier.

H.&I. Both of these items contributed directly to final Limerick design and therefore, the full $2.1 million and the $0.7 million would have been incurred regardless of the timing of load definition.

J. The first part of this item covers 16 items of work, of which 12 consist of additions that would have been required even had the Mark II loads been known at the start of construction. I estimate this work represents approximately 75% of the total, or $19.05 million. The two other items are associated with piping material changes and the addition of snubbers. I have conservatively determined that 50% of these remaining costs, or $6.2 million, would have been avoided had these loads been identified at the start of the project.

K. All of this item is attributable to rework and as such would not have been incurred if the loads had been identified earlier.

L. This work would have been required regardless of when the loads were identified and thus the full $0.4 million would have been incurred.

M. This work is related to piping and equipment in the reactor building (i.e., adjacent to containment). A review of construction records shows that only 1000 of the 5000 hangers were reworked primarily due to the Mark II loads. Therefore, 80%, or $26.6 million of this cost would have been incurred regardless of the timing of Mark II load definition.

N. This cost covers the addition of one vacuum breaker on each discharge line. These costs would have been incurred regardless of when the Mark II loads were defined. Thus, all of the $0.2 million would have been incurred.

O. Approximately $1.0 million of this cost was for excavation of the containment basemat in order to anchor quencher baseplates which could have been avoided had the Mark II loads been identified earlier. This leaves $2.8 million as being incurred regardless of the timing of Mark II load definition.

The above analysis indicates that approximately $24.8 million of the $136.1 million Mark II dollars could have been avoided had the loads been defined prior to the start of construction. An additional $8.5 million of distributable costs could also be considered potentially avoidable due to the manner in which these costs were handled in the project cost reconciliation.

Q. Did you compare your estimate with assessments made for any other construction projects?

A. Yes. I found that the Consumer Advocate's witness (D. Bridenbaugh) in the PP&L rate case computed the cost of such rework for Susquehanna at $38.6 million (final resolution by PaPUC rejected Consumer Advocate estimate and allowed full rate base inclusion for Susquehanna). My determination of the incremental cost at Limerick of $24.8 million of the Mark II loads resolution effort is consistent with this Susquehanna value. As I explained above, by waiting until nearly final loads were developed before implementing Mark II changes, savings were obtained in both engineering and construction costs over that experienced at Susquehanna. It is my understanding that OCA witness Hanauer recommends that only this incremental cost, i.e., $24.8 million, should be at issue in this proceeding.

I should emphasize that, as I explain above, the Company believes that all expenditures related to Mark II containment modifications should be permitted. These costs were necessary to build a licensable plant and were not incurred due to any PECO or GE error.

Q. Are there basic differences in the designs of Limerick, Susquehanna and LaSalle which would significantly affect the way the structures respond to Mark II and other loads?

A. Yes. The plants in question have major design differences. Two of the most significant differences are the seismic design basis and the designs of the plant foundations.

Q. How do the seismic design bases for Limerick, Susquehanna, and LaSalle compare?

A. As discussed in PECO Statement No. 5A, the design seismic accelerations for these plants are as follows:

	Limerick	Susquehanna	LaSalle
OBE horizontal	0.075g	0.05g	0.100g
vertical	0.05g	0.033g	0.067g
SSE horizontal	0.15g	0.10g	0.20g
vertical	0.10g	0.067g	0.13g

Q. Does this mean that the effect of the seismic load used in the analysis of Limerick amounted to only 75% of the seismic load used in the LaSalle analysis and 150% of the seismic load used in the Susquehanna analysis as implied by the differences in SSE horizontal accelerations?

A. No. The impact of the seismic load on the various plants depends on many factors. Such things as foundation material and ground response spectra significantly influence the effect of the seismic and hydrodynamic loads on the plant structure. For example, as shown on Schedule 2, the Limerick ground response spectra are broader in the frequency range close to the natural frequency of the plant. As a result of this condition, design amplitudes in the frequency range from 6.5 to 33 Hz are as much as 2.25 times comparative values used in the Susquehanna design, rather than the 1.5 times suggested by a simple comparison of the SSE horizontal ground accelerations.

Q. What is the significance of these differences in terms of the costs and complexity of the three plants?

A. The higher acceleration and the particular foundation conditions at Limerick influenced our decision to wait for the lower Mark II loads of the long term program. Had we used the lead plant loads, combining them with the unique seismic loads at Limerick would have significantly increased the complexity of the plant and its costs because of the need for additional and more complex pipe hangers and snubbers.

Q. Is Limerick designed such that much of the equipment in the reactor building is built using the suppression chamber as a foundation, as stated by Dr. Hanauer?

A. No. As is obvious from the attached diagram (Schedule 3), the Limerick design is unique in that 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 will, however, be transmitted through the rock to the adjacent structures.

Q. Are either of the other two comparison plants constructed in this manner?

A. Susquehanna has a design similar to that of Limerick. However, because of the lower seismic loading at Susquehanna, the overall effect of designing for the earthquake and hydrodynamic load at Susquehanna is smaller than at Limerick.

At LaSalle, on the other hand, the plant is built on a common basemat foundation. Since the plant is situated on a soil base up to 120 feet thick, the effect of the higher seismic loading has a smaller impact on the overall plant. As a result of the damping effect from this thick soil base, the LaSalle designers had to consider only minimal effects from the hydrodynamic loads in the evaluation of the piping, ducts and cables outside of the containment. In addition, the design pressure of the containment is only 45 psig, compared to the 55 psig design pressure at Limerick and Susquehanna. The net result of these differences is a containment wall thickness of 4'-0 "at LaSalle compared to the 6' 2" for Limerick and Susquehanna.

Q. Would you please summarize the conclusions in your testimony?

A. Philadelphia Electric Company chose the Mark II containment structure because it was superior in both its design and construction to alternatives then available. In the mid-1970's, employing technology and computer analytical systems not available in the early 1960's, the importance of certain hydrodynamic loads not previously appreciated was discovered. An extensive and iterative testing and evaluation program was required to characterize and assure design resolution of concerns related to these loads. The costs associated with this Program, most of which would have been incurred even had the loads been discovered prior to the start of Limerick construction, were reasonable and necessary to obtain a safe and licensable plant. All such costs should be included in the rate base value of Limerick permitted in this proceeding.

Q. Mr. Vollmer, does this conclude your rebuttal testimony?

A. Yes, it does.

SECTION ONE: REGULATORY AND OTHER EXTERNALLY-IMPOSED CONDITIONS

	RECONCILIATION ITEM	TOTAL COST ($MM)	UNITIZATION %	UNIT 1 & COMMON COST ($MM)	FORECA BASIS
	IMPACT OF MARK II				
	NUREG's-0487, 0808, 0783, 0763, 0802				
A	Engineering to evaluate the suppression pool swell phenomenon and associated costs to incorporate modifications as per NUREG-0487 and NUREG-0808.	1.7 1.5 0.1 4.0 2.6	50% 50% 50% 50% 50%	0.9 0.8 0.1 2.0 1.3	2(P.III-1) 3(P.1-5) 2(P.III-1) PECo(8332) 5(P.22)
B	Engineering to perform Mark II containment analysis and provide technical support to owner's group.	1.5 0.1	50% 50%	0.8 0.1	4(P.14) PECo(8332)
C	Plant unique analysis and support design due to Mark II new loads as per NUREG-0487 and NUREG-0808.	3.8	50%	1.9	PECo(8332)
D	Re-evaluation of loads on passive safety-related equipment to assure seismic qualification accommodated Mark II loads.	0.1	50%	0.1	5(P.18)
E	Mark II analysis consisting of: development of a 3-dimensional finite element coupled model of containment and adjacent structures; Mark II containment adequacy assessment; development of reactor pressure vessel (RPV) models based on GE-refined RPV model; and, safety relief valve (SRV) analysis.	0.9 0.2 1.0	50% 50% 50%	0.5 0.1 0.5	5(P.18) 5(P.18) 5(P.18)
F	Functional capability analysis of safety-related large and small pipe due to new Mark II loads.	0.5	50%	0.3	5(P.18)
G	Generated revised spectra using Mark II load specifications for condensation oscillation chugging and reanalysis of piping, equipment and structures per NUREG-0487.	0.3	50%	0.2	5(P.18)

SECTION ONE: REGULATORY AND OTHER EXTERNALLY-IMPOSED CONDITIONS

	RECONCILIATION ITEM	TOTAL COST ($MM)	UNITIZATION %	UNIT 1 & COMMON COST ($MM)	FORECAST BASIS
	IMPACT OF MARK II (cont'd)				
H	Mark II analysis, including: fatigue evaluation of downcomer and SRV lines; analysis of wetwell to drywell vacuum breakers; analysis of shroud support legs; owner's group support; small pipe layout per NUREG-0487; unique and generic analyses; verification of acoustic model for Mark II loads; asymmetric SRV acceleration time histories and response spectra analysis; and, analysis of suppression pool temperature monitoring system in compliance with NUREG-0487.	0.8 0.9 2.3	50% 50% 50%	0.4 0.5 1.2	5(P.18) 5(P.22) 5(P.22)
I	Design and licensing of SRV T-quenchers.	0.4 1.0	50% 50%	0.2 0.5	PECo(8332) PECo(8332)
J	Determination of final Mark II loads in accordance with NUREG-0487 and NUREG-0808 resulting in significant design changes.	25.4 24.6 0.1	100% 50% 50%	25.4 12.3 0.1	7(P.53) 6(P.149) 6(P.148)
K	Added installation costs for the main steam relief valve (MSRV) discharge line quencher and baseplate.	2.5	50%	1.3	5(P.19)
L	Additional manhours required for welding plates to containment liner for attaching hangers, etc., due to Mark II-related loads.	0.7	50%	0.4	5(P.19)
M	Engineering manhours and modifications to seismic structures adjacent to the containment to account for hydrodynamic loadings.	20.0 0.1 46.3	50% 50% 50%	10.0 0.1 23.2	5(P.17) 5(P.18) 6(P.149)

SECTION ONE: REGULATORY AND OTHER EXTERNALLY-IMPOSED CONDITIONS

	RECONCILIATION ITEM	TOTAL COST ($MM)	UNITIZATION %	UNIT 1 & COMMON COST ($MM)	FORE BAS
	IMPACT OF MARK II (cont'd)				
N	Addition of one redundant vacuum breaker valve to each MSRV discharge line.	0.3	50%	0.2	4(P.1
O	Increase in snubbers to accommodate Mark II loads; and, additional costs for MSRV discharge line, quencher, and downcomer supports.	4.5 3.0	50% 50%	2.3 1.5	4(P.1 4(P.1

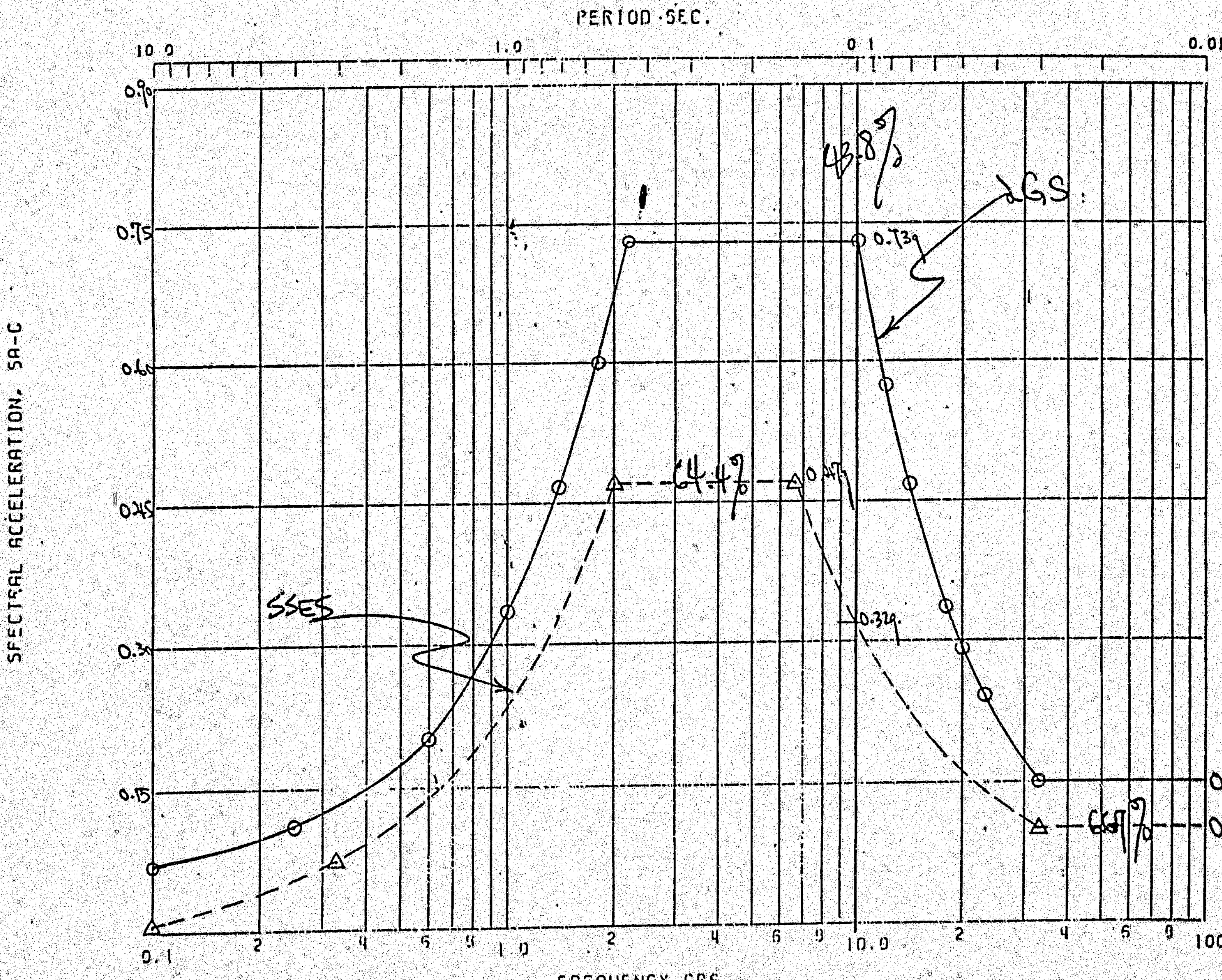
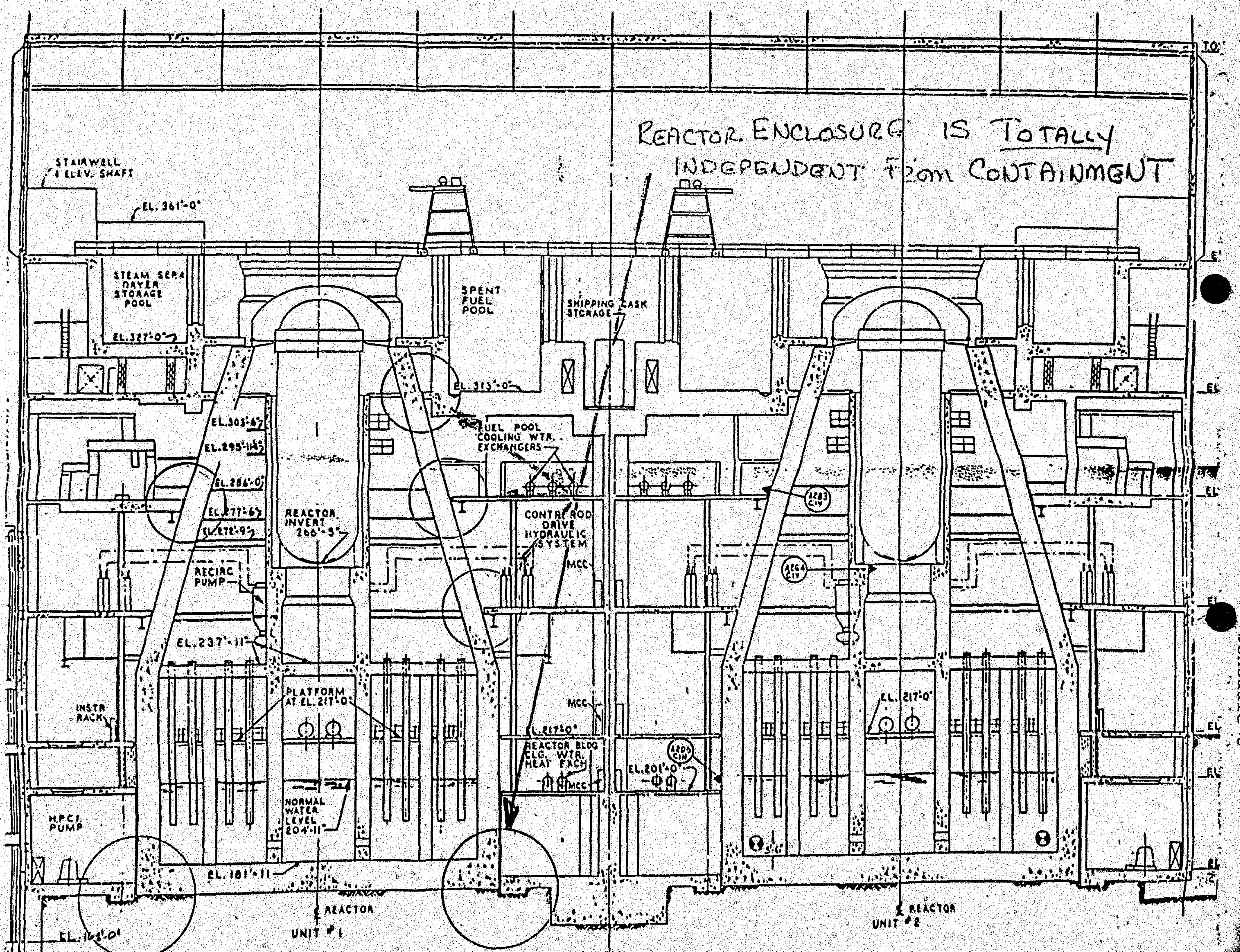


Figure 1 Comparison of SSES & LGS Design Spectra (SSE) at 0.5% Damping


REACTOR ENCLOSURE IS TOTALLY INDEPENDENT FROM CONTAINMENT
STAIRWELL & ELEV. SHAFT
EL. 361'-0"
STEAM SEP & DRYER STORAGE POOL
EL. 327'-0"
SPENT FUEL POOL
SHIPPING CASK STORAGE
EL. 313'-0"
EL. 303'-6"
EL. 295'-11"
EL. 286'-0"
EL. 277'-6"
EL. 272'-9"
REACTOR INVERT 266'-5"
FUEL POOL COOLING WTR. EXCHANGERS
CONTROL ROD DRIVE HYDRAULIC SYSTEM
MCC
RECIRC PUMP
EL. 237'-11"
PLATFORM AT EL. 217'-0"
INSTR RACK
EL. 217'-0"
REACTOR BLDG CLG. WTR. HEAT EXCH
MCC
EL. 201'-0"
NORMAL WATER LEVEL 204'-11"
HPCI PUMP
EL. 181'-11"
EL. 163'-0"
℄ REACTOR UNIT #1
℄ REACTOR UNIT #2
EL. 217'-0"

PECO STATEMENT NO. 31A

R-850152 Hbg
3/14/86 PG

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MAR 17 1986
SECRETARY'S OFFICE
Public Utility Commissio

PENNSYLVANIA PUBLIC UTILITY COMMISSION

V.

PHILADELPHIA ELECTRIC COMPANY
DOCKET NO. R-850152

SUR-SURREBUTTAL TESTIMONY

OF

H. WILLIAM VOLLMER

DOCKETED

MAR 24 1986

MARK II CONTAINMENT ISSUES

March 13, 1986

SUR-SURREBUTTAL TESTIMONY OF H. WILLIAM VOLLMER

Q. Please state your name and business address.

A. My name is H. William Vollmer, and my business address is Philadelphia Electric Company, 2301 Market Street, Philadelphia, Pennsylvania, 19101.

Q. Are you the same H. William Vollmer who provided rebuttal testimony at Docket No. R-850152 for the Philadelphia Electric Company?

A. Yes, my rebuttal testimony was previously filed as PECO Statement No. 31.

Q. Mr. Vollmer, have you reviewed the surrebuttal testimony of Dr. Hanauer and Mr. O'Brien?

A. Yes I have.

Q. What is the purpose of your sur-surrebuttal testimony?

A. I provide rebuttal to statements made by Dr. Hanauer concerning the state of technology at the time the pressure suppression containment was developed and also show that the AEC evaluated and accepted the test and analytical data which underlay Mark II pressure suppression design. In addition, I will rebut statements by Mr. O'Brien concerning schedule and cost of Mark II containment work at Limerick.

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

A. My qualifications to testify on these subjects are based on experience over the past 20 years related to structural design and testing, data acquisition, and Mark II program management, all of which are covered in detail on pages 1 through 4 of my rebuttal testimony (PECO Statement No. 31).

Q. Mr. Vollmer, Dr. Hanauer refers to certain tests described in the Humboldt Bay Hazards Report. Could you describe the reasons why these tests were conducted and their results?

A. The tests cited by Dr. Hanauer were conducted to demonstrate that steam discharged into water at high flow rates, could in fact be condensed in water. These tests were further directed at determining the maximum drywell and suppression pool pressure and temperature that would result from this process, as well as other effects of containing a LOCA with pressure suppression. In all cases, the effectiveness of the pressure suppression concept was confirmed. In fact, it was further demonstrated at the full scale Bodega Bay tests that steam flow up to four times the projected maximum credible LOCA event could be condensed. (Bodega Appendix IV, p. 34, see PECO Exhibit SL-4). In addition, a test was run for the design steam flow, with the pool temperature elevated to 60 degrees F. above the design value (i.e. actual temperature of 140°), and no vibration was identified (see Bodega Appendix IV, p. 32, see PECO Exhibit SL-4). Although not mentioned by Dr. Hanauer, these tests established that there was substantial conservatism and, thus, safety margin in the pressure suppression design. Moreover, they provided strong evidence that the concept was safe for use at other proposed nuclear generating facilities.

Q. Mr. Vollmer what test results and analytical information were referenced in the containment section of the Limerick Preliminary Safety Evaluation Report (PSAR)?

A. Schedule 1 attached to this testimony is a copy of the reference page from the Limerick Generating Station PSAR. All of the identified information was either submitted to the AEC or was available to regulators in the open literature. The data provided included the Bodega Bay Preliminary Hazards Summary Report, which describes the numerous single vent and multi-vent tests conducted at the Bodega Bay scale test facility. It is my understanding that internal, proprietary

GE Reports containing additional information on these tests were provided to the AEC as part of the Mark II containment licensing process. References 3, 4 and 5 describe computer model and analytical procedures employed in evaluating the Mark II containment concept, and how that concept was properly supported by the Humboldt Bay and Bodega Bay test data. Similar presentations were made to the AEC by other utilities planning to construct Mark II containment BWRs, including principally the Shoreham and Zimmer plants. The AEC reviewed this data extensively in licensing proceedings for those plants and issued construction permits in each instance. This was also the case for Limerick.

Q. Mr. Vollmer, could the Mark II hydrodynamic loads and associated design specifications have been developed with available testing equipment and analytical techniques in the early 1960's?

A. Not in my opinion. It would have been very difficult in the early 1960s to produce an effort similar to what actually took place from 1975 to 1982. A program to develop loads to the level of accuracy used in the Mark II analyses required computer capability available only in the 1970's. One of the most difficult problems faced by the Mark II program was to develop methods for removing the effects of the test structure (or resonance) from the load measured by the gage. Many man years of research were required to develop these methods. This work was referred to as FLUID STRUCTURE INTERACTION or FSI. The industry computer codes used for this work did not exist prior to the mid-1970's.

Q. Does Dr. Hanauer continue to err in his understanding of the Limerick containment system?

A. Yes. On pages 37 and 38 of his surrebuttal testimony and on page 14 of his direct testimony he refers to various equipment being directly attached to the containment. In response to my rebuttal testimony, he comprehends for the first time that the base mat of the reactor enclosure is separate from the containment, however, he continues to imply that certain reactor enclosure equipment is still in direct contact with the containment. A correct analysis of the drawing provided in my rebuttal testimony (PECO St. 31, Schedule 3) demonstrates that each floor of the reactor building, is separated from the containment structure by the seismic gap I have previously described. This gap provides substantial isolation from the containment for all reactor enclosure equipment. In response to Dr. Hanauer's quibbling over whether the Limerick design is "unique" or not, the Limerick design was the prototype which Susquehanna followed.

Q. Has Mr. O'Brien correctly characterized your statements on the complexity of the Mark II issue (OCA St. 1B, p. 6, lines 13-26)?

A. No, he has not. In the discussion that he referenced, PECO Statement 31, pp. 8-9, it was pointed out that this very complex issue was made more difficult to solve because of the more stringent regulatory requirements present in the U.S. At no time in this discussion did I imply that Mark II was a critical path issue. In fact, I specifically testified that Mark II was not a critical path activity and would not have prevented a July 1982 completion of the plant. Rebuttal testimony provided by PECO witnesses Sproat, Mattson, TB&A and Helwig provide further information and documentation of the actual critical path which consisted of other externally imposed regulatory requirements.

Q. Does Mr. O'Brien's testimony indicate a proper understanding of the cost basis of the Mark II issue at Limerick?

A. No. From page 28, line 22 through page 33, line 26 of his testimony, it is apparent that Mr. O'Brien does not understand the cost information provided and has erroneously interpreted these facts.

Q. Would you please clarify this matter?

A. Certainly. First, the $136.1 million identified by the cost reconciliation analysis is the total cost of the Mark II work determined to be applicable to Unit 1. This total is comprised of $89.2 million of costs specifically identified as Mark II related in the project forecasts, and the $46.9 million of project costs distributed to the Mark II project as described in PECO Statement 8. Further, in response to IR-OCA-14-1 (see Schedule 2), a calculation of the AFUDC associated with this work was provided. The $162 million figure cited by Mr. O'Brien as inconsistent with the cost reconciliation identified cost, was a preliminary estimate of Mark II costs for <u>Limerick Units 1 and 2</u> provided to the OCA well before the completion of the cost reconciliation. It was identified to the OCA as a very preliminary estimate (refer to DR-OCA-II-8, see Schedule 3). Thus, our more refined Mark II cost effect analysis increased rather than decreased costs associated with this program in direct contradiction of Mr. O'Brien's implication. This fact should have been known to Mr. O'Brien at the time his surrebuttal testimony was prepared.

Q. In what other respect does Mr. O'Brien distort the facts related to the Mark II costs?

A. On page 28, line 1 of OCA St. 1B, he refers to the $24.8 million figure stated in my testimony as "...the Mark II cost at Limerick (which) would have been incurred if the plant loads had been specified correctly to start with." In my rebuttal testimony, (PECO St. 31, page 10, line 31 and page 13, lines 39 and 49), as well as IR-OCA-29-2 (see Schedule 4), and IR-OCA-25-6 (see Schedule 5), it is clearly stated that the $24.8 million is instead the incremental cost of <u>not</u> having the Mark II hydrodynamic loads characterized at the start of construction. When a proportionate share of the distributable costs is included (see IR-OCA-25-6), this figure is on the same order of magnitude as the costs presented by the Consumer Advocate witnesses on the Susquehanna case. It is our position, however, that the total cost was prudent since the new information was not available until after the original design and in fact, PECO minimized adverse cost effects by prudently scheduling the work as late as possible while meeting the project completion schedule established by other factors.

Q. Do you believe that Mr. O'Brien understands the basis of your cost breakdown?

A. Yes, he acknowledges that the Mark II incremental cost quantification is based on project cost forecasts; however, he incorrectly states that because of overall project cost increases (OCA St. 1B, p. 30, lines 6-8), these figures may be incorrect.

Q. Are they incorrect due to project cost increases?

A. No. The sum of all of the forecast costs and distributable costs equal the total cost of the project. Increases over initial cost estimates for the Mark II or other regulatory effects are accounted for by subsequent forecast identified cost increases and by allocation of distributables. These latter costs are the "unallocated by cause" cost increases which Mr. O'Brien correctly notes arise in the Bechtel forecast system. As described above, our development of Mark II incremental costs correctly accounts for each of the factors described by Mr. O'Brien.

Q. Does Mr. O'Brien's logic on page 31 correctly infer that PECO has given improper credit for the cost of original work when rework is taking place?

A. No. The Company is entitled to recover the cost of constructing the Limerick containment, including provision for the Mark II hydrodynamic loads. However, if there has been imprudence in failing to recognize and design for the Mark II loads in the Limerick containment's initial design, then I am advised that the Company is not entitled to recover the costs of essentially designing and constructing the containment twice, i.e. once prior to discovery of the hydrodynamic loads and once for their resolution. My quantification properly eliminates all costs associated with the second design/construction effort, thus permitting the Company to recover the costs of design and construction of a Mark II containment which resolves all hydrodynamic loads as if those loads had been known from the beginning of design. The original design costs which I have not excluded from my incremental cost quantification are costs which would have been incurred even had the Mark II loads been recognized earlier. I would like to emphasize, as stated elsewhere in my testimony, that I perceive no imprudence in the identification of the Mark II hydrodynamic loads phenomenon and thus believe that no cost disallowance is appropriate.

Q. Would you address Mr. O'Brien's statement that the "installation of about 1000 pipe hangers was wasted and had to be done entirely over"?

A. I state on page 12, lines 44 and 45, of my rebuttal testimony that "a review of construction records shows that only 1000 of the 5000 hangers were reworked". This statement does imply that they were "thrown away". As explained in my rebuttal testimony, I have included as an incremental Mark II loads related cost, the cost of reworking these hangers. Thus, consistent with my treatment of engineering costs discussed above, my incremental cost calculation permits the Company to recover the cost of installing these hangers only once.

Q. In your estimate of Mark II loads incremental cost effect have you properly accounted for the fact that Unit 2 will not require as much rework as performed on Unit 1?

A. Yes. A review of my backup data, provided to the OCA as an interrogatory response, documents that the Unit 1 and 2 costs have been properly apportioned. Redesign efforts, which involve engineering time, are equally applicable and will be employed at each Unit (see items A to I). Thus, the 50-50 split of these costs employed is appropriate. Since the structural concrete already exists for Unit 2, work like excavating the suppression chamber foundation to anchor the quencher baseplates, and drilling through the reactor pedestal in the suppression chamber to anchor the downcomber bracing will also be equally split. Therefore, items K and O in PECO St. 31 are properly allocated. In addition, items L and N would maintain the splits as previously shown because these items were not previously considered as rework.

The review of the cost components for item J also supports not changing the previously identified allocations. The 4 items considered rework under category j.1 would apply equally to both units. Furthermore, item j.2 would also maintain the split previously identified because the change applies equally to both Unit 1 and 2 due to the state of construction of Unit 2. The last item, item j.3, would maintain the 50-50 split because the In Plant Test covered by this item provided methodology for analysis of both plants.

The cost components of item M have been reviewed in detail utilizing the supporting documentation provided in response to IR-OCA-4-45. The three components of this cost are identified there as items m.1, m.2 and m.3. Item m.1 includes engineering and field work associated with Mark II loads on pipe hangers, conduit, and cable tray in the adjacent structures. The cost trend which supports this entry specifically indicates the amount of work associated with Unit 1, Unit 2, and Common Facilities. The cost of all Unit 1 and Common field work is indicated to be $3.7 million. Since 20% of this was rework, as previously discussed in my rebuttal testimony, the cost of Mark II related rework in Unit 1 and Common was approximately $740,000.

Item m.2 is entirely comprised of engineering costs which are appropriately shared equally between the Units as described above.

Item m.3 includes engineering and field work associated with the consideration of Mark II loads on pipe hangers in the adjacent structures. Again, the cost trend which supports this entry specifically indicates the amount of work associated with Unit 1, Unit 2, and Common Facilities. The cost of all Unit 1 and Common field work was $33.5 million of which $5.1 million was rework.

Thus, detailed analysis of the supporting documentation indicates that rework associated with Item M was $5.8 million which is actually somewhat less than the amount indicated by the more approximate analysis described in my previous statement.

Q. Mr. Vollmer, would you please summarize your sur-surrebuttal testimony?

A. Although looking back on early test data, small hydrodynamic phenomenon were present, they were not viewed as significant to containment design by the responsible engineers, the regulators or the scientific community. The level of awareness of these containment phenomena increased as a result of field experience, the development of technology and increased regulatory pressures, forces not present prior to the mid-1970s. Additional research was conducted under the sponsorship of the Mark II utilities and the load specification necessary to license the plants was developed. Costs of the program reflected the level of effort necessary to solve the problem. Because the Mark II program was not critical path for Limerick, Limerick was able to benefit from solutions developed by earlier scheduled plants. As I have demonstrated above, the several criticisms by Mr. O'Brien of my prior analyses are not valid and should be rejected.

Q. Does this conclude your sur-surrebuttal testimony?

A. Yes, it does.

SECTION 5

CONTAINMENT

REFERENCES

1. Bodega Bay Preliminary Hazards Summary Report, Appendix 1, Docket 50-205, December 28, 1962.

2. Ghosh, S. and Wilson, E., Dynamic Stress Analysis of Axisymmetric Structures Under Arbitrary Loading, University of California, Berkeley, Earthquake Engineering Research Center, Report EERC 69-10, September, 1969.

3. Moody, F. J., Maximum Flow Rate of a Single Component, Two Phase Mixture, ASME 64-HT-35.

4. Miller, D. R., "Pressure Suppression Containment Design - Current State of the Art", Transactions of the ASME, Journal of Engineering for Power, Vol 91, Series A, No. 1, p. 13, January, 1969.

5. General Electric Company Topical Report, The General Electric Pressure Suppression Containment Analytical Model", NEDO-10320, April 1971.

Q. IR-OCA-14-1. Please provide an annual breakdown, by January-June and July-December, of the incurrence of the $136.1 million related to Mark II changes. OCA-4-45 does not readily provide these costs by such periods, only by Forecast number.

A. IR-OCA-14-1. Attachment IR-OCA-14-1 contains the annual breakdown of estimated expenditures related to Mark II changes.

Responsible Witnesses: V.S.Boyer, Sr.Vice President-Nuclear Power

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

ESTIMATED MARK II EXPEDITURES

Year		$ in Thousands
1975	1st Half	60
	2nd Half	200
1976	1st Half	420
	2nd Half	520
1977	1st Half	850
	2nd Half	850
1978	1st Half	1,700
	2nd Half	2,100
1979	1st Half	2,200
	2nd Half	3,100
1980	1st Half	4,500
	2nd Half	6,400
1981	1st Half	11,100
	2nd Half	11,200
1982	1st Half	13,300
	2nd Half	23,600
1983	1st Half	26,700
	2nd Half	18,800
1984		8,500
		$ 136,100

Q: DR-OCA-II-8: Please provide PECO's best approximation of the cost incurred as a result of the GE Mark II redesign.

A: DR-OCA-II-8: Total direct costs incurred as a result of the Mark II redesign approximate $164.2 million.

Q.IR-OCA-29-2. Please provide the total cost of Mark II, including AFUDC, overhead and taxes, you believe could have been avoided had the loads been defined prior to the start of construction.

A.IR-OCA-29-2. The costs for Mark II that could have been avoided had the loads been defined prior to the start of construction are contained in PECo Statement #31 and in response to IR-OCA-25-6.

The potentially avoidable costs for Mark II are $24.8 million and for distributables are $13.0 million. AFUDC accruals on these direct costs, from the year of incurrence through February 1986, based on actual AFUDC rates in effect, are $11.3 million for Mark II and $5.9 million for distributables. There would have been no significant reduction in overheads and no reduction in capitalized taxes had these costs been avoided.

Responsible Witness: H. W. Vollmer, Supervising Engineer - Structural Branch

Q.IR-OCA-25-6. Please show the derivation of the $8.5 million of distributable costs noted as potentially on page 13, line 21.

A.IR-OCA-25-6. The $8.5 million figure related to the distributable costs was derived by dividing the $24.8 million potentially avoidable costs as discussed in items A through O (pages 10-13) of PECo statement 31 by the total Unit 1 reconciled Mark II cost of $136.1 million and multiplying this number by the distributed cost of $46.9 million.

The above calculation however, is in error and correctly should have been calculated as follows; the $24.8 million potentially avoidable costs should have been divided by the $89.2 million total Unit I allocated Mark II (exclusive of the $46.9 million distributable costs), multiplied by the $46.9 million distributable costs to give a potentially avoidable distributable cost of $13.0 million.

It should be kept in mind however, that the $46.9 million distributable cost was calculated by spreading the total costs not attributable to specific causes, over each major category (as previously discussed in PECo Statement 8). Thus these costs may or may not have been directly associated with this rework; however, to be conservative, we have identified this number.

Responsible Witness: H. William Vollmer, Supervising Engineer, Structural Branch

PECO EXHIBIT SL-1

PH 3-14-86

Nb.

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

SELECTED INTERROGATORY RESPONSES
TO THE
OFFICE OF CONSUMER ADVOCATE

March 13, 1986

Q. IR-OCA-28-1

At page 2, lines 1-3, Dr. Levy states that he has been involved in the characterization and design resolution of containment hydrodynamic loads at various nuclear power plants. Please state the plants referred to, Dr. Levy's role at each plant, and the year or years of his involvement.

A. IR-OCA-28-1

Since leaving GE in 1977, Dr. Levy has consulted on and reviewed the work associated with defining the condensation oscillation and chugging loads for two of the lead plants, i.e., LaSalle and Zimmer. His involvement began in late 1979 and lasted through 1980. Dr. Levy has also consulted with Bechtel in developing the improved chugging model IWEGS discussed in NUREG-0808. His participation in this activity extended through the first half of 1980. This model has been applied to several nuclear power plants. Finally, Dr. Levy was a consultant in 1978 and 1979 to the Mark II program, Task A.17-Condensation Oscillation Frequency Study.

Responsible Witness: Salomon Levy
S. Levy Incorporated

Q. IR-OCA-28-2 Please describe Dr. Levy's role in the original creation of the GE pressure suppression concept and design.

A. IR-OCA-28-2 Dr. Levy had no role in the original creation of the GE pressure suppression concept and design.

Responsible Witness: Salomon Levy
S. Levy Incorporated

Q. IR-OCA-28-3

Please describe Dr. Levy's role in the testing and design of the GE pressure suppression system in the

a) 1950's
b) 1960's

A. IR-OCA-28-3

In the late 1950's, Dr. Levy performed a transient pressure analysis of the Humboldt Bay containment utilizing a different flow model as requested by the AEC/ACRS. After becoming Manager of Systems Engineering in 1968 and through 1971, Dr. Levy was responsible for a group which performed the safety analysis and, in particular, calculated transient pressures and temperatures in General Electric pressure suppression containments. As a result of these activities and Dr. Levy's later study of the hydrodynamic load phenomena and containment design, Dr. Levy has detailed knowledge of the results of testing and evaluation of the GE pressure suppression concept and of the design of the Mark II pressure suppression system.

Responsible Witness: Salomon Levy
S. Levy Incorporated

Q. IR-OCA-28-8 Please state the year in which Dr. Levy believes GE first became aware of the existence of hydrodynamioc loads in pressure suppression systems.

A. IR-OCA-28-8 As pointed out in Dr. Levy's testimony, the concern about hydrodynamic loads associated with relief valve actuation started in 1972. Their potential design significance was not recognized until 1975. The significance of LOCA hydrodynamic loads was recognized in late 1974 and early 1975.

Responsible Witness: Salomon Levy
S. Levy Incorporated

Q. IR-OCA-28-9 Please state what information became available in the year which is the answer to question 8 above.

A. IR-OCA-28-9 The significance of LOCA hydrodynamic loads was established in late 1974 and early 1975 after the first large scale tests of the Mark III pressure suppression configuration were completed. The potential effects of SRV loads were confirmed after testing at operating plants in 1972. However, as described in Dr. Levy's testimony, prior to 1975, these loads were perceived as having been fully corrected for by technical specification directives for plant shut down should suppression pool temperature exceed a level well below that at which SRV related loads occur. However, in 1975 in connection with its examination of the recently discovered possible significance of LOCA loads, the NRC determined to review whether additional steps for resolution of these loads should be taken.

Responsible Witness: Salomon Levy
S. Levy Incorporated

Q. IR-OCA-28-11 Is Dr. Levy aware of, or familiar with, GE tests at the Vallecitos Atomic Power laboratory in the 1950's. If yes, please state his involvement or the basis of his knowledge.

A. IR-OCA-28-11 Dr. Levy is aware of and is generally familiar with the GE tests of the pressure suppression concept performed at the Vallecitos Atomic Power Laboratory in the 1950's. Dr. Levy had no direct involvement in those tests, but participated in discussions of test results.

These tests were very early efforts to begin the data collection process to determine the preliminary feasibility of the pressure suppression concept. The tests are identified in a published document, i.e. ASME Paper 59-A-215; a copy of which is attached to this interrogatory response. The tests and associated analyses involved a literature search to determine whether others had performed tests and analyses respecting the concept and the conduct of "preliminary small scale tests of steam condensation" at the Vallectios facility.

Responsible Witness: Salomon Levy
S. Levy Incorporated

PAPER NUMBER

59-A-215

ATTACHMENT IR-OCA-28-11

MERICAN SOCIETY OF MECHANICAL ENGINEERS
29 West 39th Street, New York 18, N. Y.

Pressure Suppression Containment for Nuclear Power Plants

C. C. WHELCHEL

Chief Mechanical Engineer,
Pacific Gas and Electric
Company, San Francisco, Cal.

C. H. ROBBINS

Manager, Mechanical and Process
Development, Atomic Power
Equipment Department, General
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Assoc. Mem. ASME

Pressure-suppression containment provides for venting into a water pool the steam-water mixture that would be released from a water-moderated reactor in the very unlikely event of a break in the primary system. The steam would be condensed and most of the entrained fission products would be retained in the pool. This paper describes a development program which was performed to determine feasibility of the concept and how a practical system could be designed. Test work included direct condensation of large-diameter steam jets in water, and a scale model of a pressure-suppression system for investigating transient behavior. Test results show that large steam flows can be completely condensed in small volumes of water, system pressure transients can be predicted, and that the water pool can be extremely effective in limiting release of fission products. It is concluded that the pressure-suppression concept has important potential advantages over the more usual dry-capsule type of reactor containment.

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Pressure Suppression Containment for Nuclear Power Plants

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Through the years station designers and equipment manufacturers have made significant cost reductions in conventional steam power plants by simplifying design, by lowering manufacturing costs, by increasing the size of individual generating units, and by learning to use materials more effectively. In terms of constant dollars, without inflation, the results are impressive.

It was only natural that as designers with this background began working on atomic plants, they were impressed by the relatively high cost of reactor containment, and began thinking of ways and means by which it might be reduced. Most water-cooled reactors have been enclosed in large spherical or cylindrical pressure vessels designed to withstand the resulting pressure from escaping steam in the event of a major failure.

A concept of reactor containment which showed considerable promise of reducing space and plant cost was suggested by General Electric Company. With it the escaping steam is quenched in a water pool, thereby limiting pressure rise and permitting much simpler plant design. Pacific Gas and Electric Company became interested in the potentialities of this scheme and about a year ago decided to finance a research and development program to determine its feasibility and design parameters. This paper describes the results of that program and how a practical pressure-suppression reactor-containment system may be designed. Judgments as to what kinds of major failures could credibly occur in any particular reactor system are outside the scope of this paper.

Fig.1 is a schematic arrangement illustrating the basic concept. The reactor vessel, V(1), containing the core, is in a dry well, V(2), filled with air at about atmospheric pressure. The dry well vents to a water pool. The air space, V(3), above the water pool is enclosed by a vapor-tight container. In the very improbable incident of a break in the primary system, steam and water would first enter the dry well. As pressure increased in the dry well, the air, steam, and water would vent into the water pool where the steam would be condensed. Pressure build-up within the enclosure would be relatively small because the water pool would act as a heat sink for the steam and water released from the reactor vessel.

The pressure-suppression concept has poten-1 safety advantages over dry-capsule containment. The water pool would tend to retain fission products entering it and only a very small fraction of any radioactive contaminants (other than noble gases) released in the dry well, V(2), would escape to the containment volume, V(3). Leakage from the container would be low because of the small and short-lived driving pressure between the inside and the outside of the container; the container would return essentially to atmospheric pressure shortly after the event. Further, the water in the pool lends itself to designs making it available for post-accident core cooling.

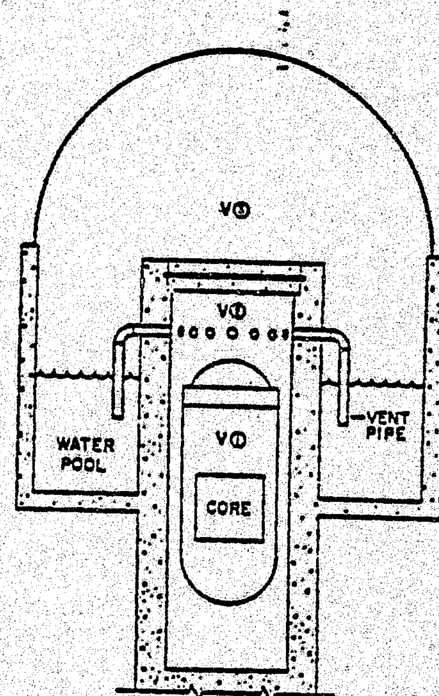


Fig. 1 Simplified arrangement of the pressure-suppression system for reactor containment

A development program conducted by General Electric Company concerning pressure suppression has been completed. The program was planned with the aim of obtaining information useful to a wide range of plant sizes, and with the expectation that it might be used first on Humboldt Unit Number 3 of Pacific Gas and Electric Company at Eureka, Calif. The Containment Section of the Preliminary Hazards Summary Report for that unit is being prepared at this time (August, 1959) for

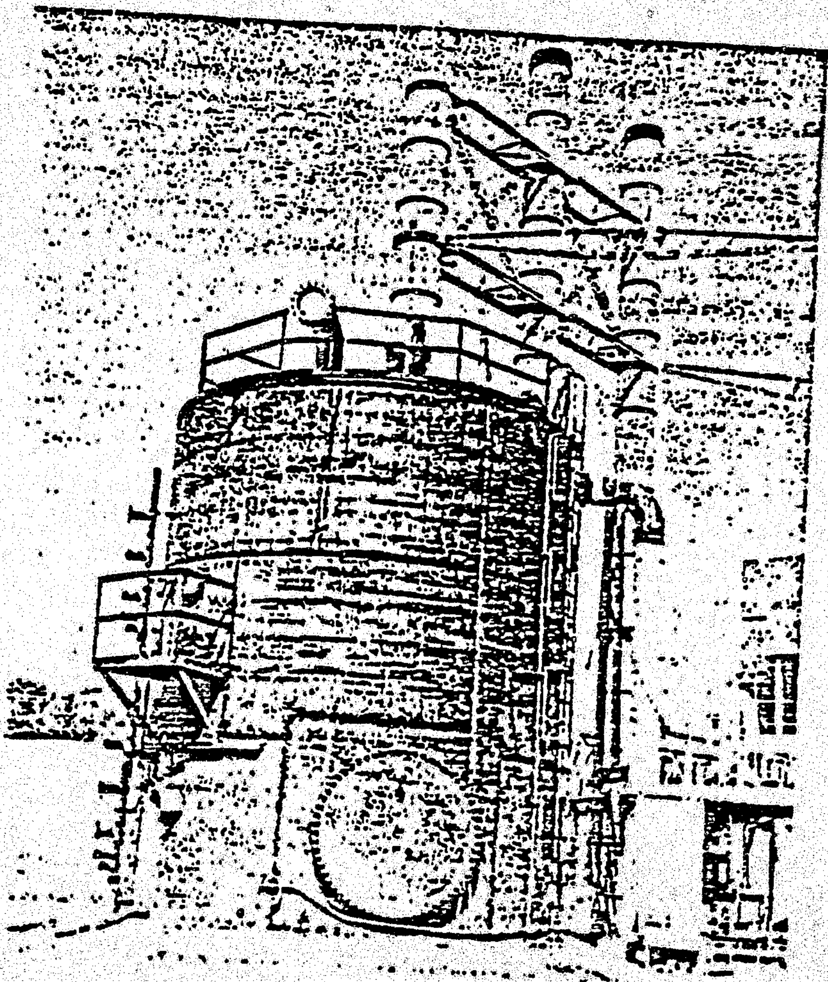


Fig. 2 Condensing test facility

submission to the Atomic Energy Commission.

DEVELOPMENT PROGRAM

Tests and analyses were necessary to make it possible to predict with confidence how pressure-suppression containment would work if ever called on. The following specific questions required answers before such a system could be designed and built.

1 How should the dry well be designed so it would not rupture and allow the steam to by-pass the condensing pool?

2 What is required to make sure that steam introduced to the water pool would be condensed and would not enter the vapor space and rupture enclosure?

3 How effective would the water pool be in removing entrained fission products?

The development program was performed in three phases. Phase I was concerned with determining preliminary feasibility, Phase II consisted of testing in two major facilities, and Phase III involved interpretation of test results in terms of design considerations.

Phase I

Phase I work included review of applicable work by others, development of methods of analysis, preliminary small-scale tests of steam condensation, evaluation of the need for further work, and planning of the rest of the program. On the basis of the favorable information obtained in Phase I, it was decided to perform Phases II and III.

Phase II - Testing

Two test facilities were built and operated to determine complementary information.

The condensing test tank was designed to test full-scale vent units with the volume of water an individual vent might require. Steady flow of steam was available in amounts as high as the maximum expected through a vent during the operating transient of the pressure-suppression system. Full scale vent operation was desired since scaling up of results from small-diameter jets and water pools might be subject to excessive doubts. Further, it seems reasonable to assume that a transient flow of steam can be condensed if steady flow can be condensed with the same maximum flow, pool and jet geometry.

The Transient Test Facility was constructed to simulate transient operation of a complete pressure-suppression system on a small scale. Use of a complete system helps insure that no important factors are overlooked in analyses predicting behavior. A full-scale pressure-suppression system for development would have been too costly and cumbersome for convenient testing.

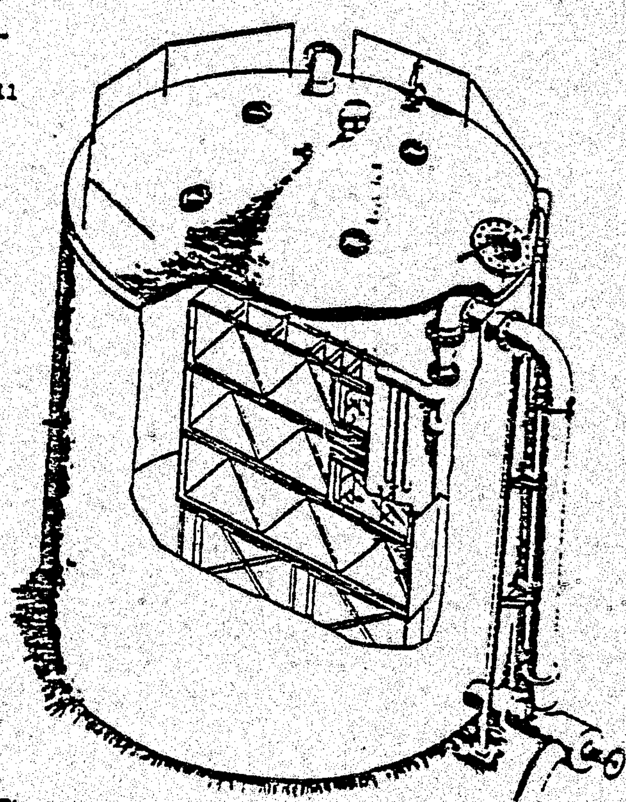


Fig. 3 Condensing test facility showing test compartment with vents

Condensing Tests

a) Objectives. The specific objectives of the condensing tests were:

1 Demonstrate, with steady stream flows, that rapid and effective condensation may be obtained by a simple straight pipe discharging below the surface of a pool of water.

2 Investigate the effect on condensation effectiveness of vent diameter, steam-flow rate, depth of submergence of the open end of the vent, direction of discharge, and effect of neighboring jets.

3 Evaluate the effect of pool volume and geometry on effectiveness of condensation.

b) Description of Test Equipment. The Condensing Test Facility, located at the Moss Landing Power Plant of Pacific Gas and Electric Company, is shown in Fig.2, and a cutaway drawing, Fig.3. The large water-storage tank was 20 ft diam and 24 ft high with dished heads. Steam at 100 psig, approximately saturated, was available from the power station at flow rates up to about 100,000 lb per hr. Steam entered the water tank through either a flange near the bottom or another flange at the top. Various experimental vents were connected to these flanges inside the tank to discharge the steam into the water. Test vents included straight pipes from 4 to 14 in. diam.

To simulate the volume and shape of water associated with an individual vent, an open top box was used in some tests. The box is shown in Fig.3 with three 4-in-diam vents. Dimensions of the box could be varied.

Instrumentation was provided to measure steam-flow rates, pressure, and temperature; water-pool depth; water temperature at several points in the tank; and pressure and temperature in the vapor space. Glass ports in the tank and windows in the box permitted viewing the performance within the tank and taking both still photographs and movies. Vapor-space instrumentation was provided for the purpose of detecting incomplete condensation of steam with a closed tank.

c) Test Program. (1) Condensing Tests Without Compartment in Tank. The first series of tests with the Condensing Test Facility utilized the tank without the box compartment. The volume of water for the vents was considerably larger than would be desirable for a pressure-suppression system of the type pictured in Fig.1. The following range of parameters was tested.

Vents	4, 6, 8, and 14-in-diam single straight pipes, and a triple 4-in-diam vent
Depth of submergence. . .	1 in. to 6 ft
Direction of discharge. .	Vertically downward and horizontal
Steam flow.	10,000 to 93,000 lb/hr
Tank water temperature. .	50 to 150 F

Steam was completely condensed in all but 3

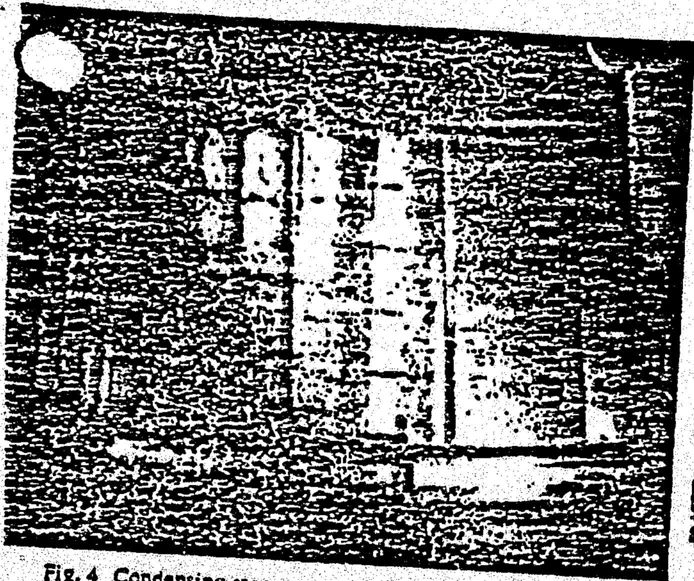


Fig. 4 Condensing steam jets. 37,000 lb per hr steam, compartment 6 in. x 12 ft x 12 ft

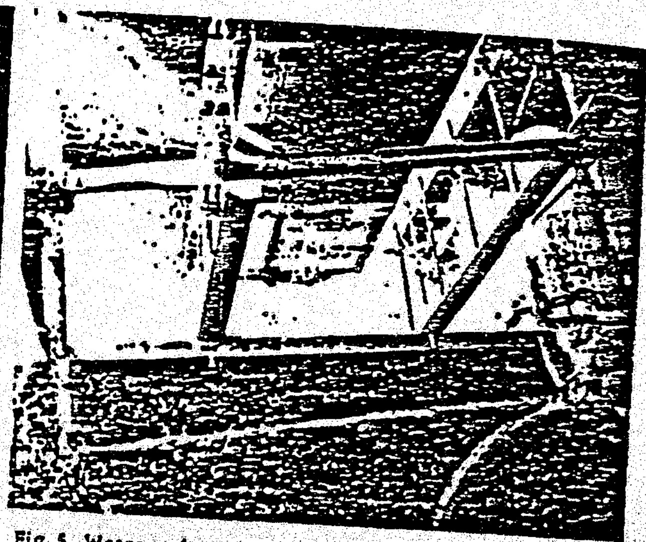


Fig. 5 Water surface during discharge of submerged steam jets. 52,000 lb per hr steam, compartment 18 in. x 8 ft x 8 ft

out of 40 tests in this series. The exceptions were: 6-in-diam pipe discharging 60,000 lb per hr horizontally with 6 in. nominal depth of submergence; 8-in-diam pipe discharging 83,000 lb per hr horizontally with 6 in. nominal depth of submergence: and 8-in-diam pipe discharging 77,000 lb per hr downward vertically with 1 in. 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.

2 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.

From the standpoint of design of pressure-suppression pools, tank vibration is not likely to be a problem. The pool water could be kept below 120-130 F without seriously hampering the design, operating time of the pool and vent system would be short if ever, and hot pool water would occur at a time when steam flow through the vents would have fallen off from the maximum.

3 Tests With Compartments. A third series of tests was conducted to investigate the interaction of vents discharging vertically into small volumes of water. An open top box within the tank, as shown in Fig.3, was used with windows in the sides which permitted viewing the surface and the discharging jet. Tests were of short duration, sometimes as short as 20 sec, because of rapid heating of the water in the compartment.

Ninety-one tests were conducted to investigate the effect of steam-flow rate, vent diameter, and width, length, and depth of the compartment.

The surface of the water was observed to be depressed near the vent pipe because of the momentum of the jet. The depression was observed to be as much as 5 ft in some cases. However, complete steam condensation in the water pool was always obtained with the compartment tests.

Mixing of the water in the compartments was excellent, based on observation of the flow and measurement of the water temperature.

Fig.4 is a view of three parallel steam jets discharging downward into the compartment. Numbers on the photograph are inches below the initial water surface. Note the short length required for condensation of the jets and that the jet condenses without forming large steam bubbles.

Fig.5 is a view of the surface of the water in the compartment during steam discharge through pipes with the open end 6 ft below the top of the box. The manifold pipe for the vents is shown on the extreme right of the picture. Only the right side of the compartment is shown. On the left of the photograph is an abrupt change in elevation of the surface similar to a hydraulic jump. Bracing shown in the picture was necessary to maintain the shape and position of the box.

2 Transient Tests for Pressure Response.

a) Objectives. The specific objective of the transient tests was to obtain data on pressure transients during operation of pressure suppression containment. This information was necessary

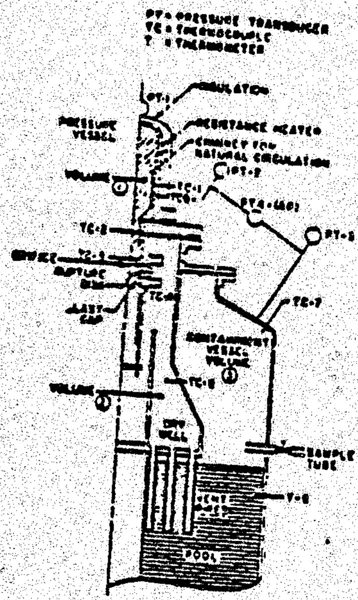


Fig. 6 Arrangement of transient test facility

to either confirm the analytical method of calculating pressure transients or to show how the analysis should be modified.

b) Description of Test Facility. The transient test facility consisted basically of three interconnected pressure vessels that simulated a reactor vessel, dry well, and outer enclosure over a water pool. The arrangment is shown in Fig.6, and Fig.7 is a photograph of the equipment. The relative position of the reactor vessel differs from the schematic of pressure suppression, Fig.1, to provide better accessibility for testing. Water in the reactor vessel was heated to high pressure and then discharged through an orifice plate into the dry well by breaking a rupture disk. The air and steam were discharged through vent pipes into the water pool where the steam was condensed.

Primary instrumentation consisted of pressure transducers connected to a four-channel recorder, which measured pressure in the reactor vessel, in the dry well, in the space over the water pool, and differential pressure between the dry well and the air space over the water.

The volumetric scale size of the test facility compared with the design of the Humboldt Unit Number 3, is about 1:1000; the area is 1:100, and the linear scale 1:10. The time scale is then 10:1; that is, events happen ten times as fast in the test as they would in the reactor plant.

c) Test Parameters. Parameters varied in the test program included the following:

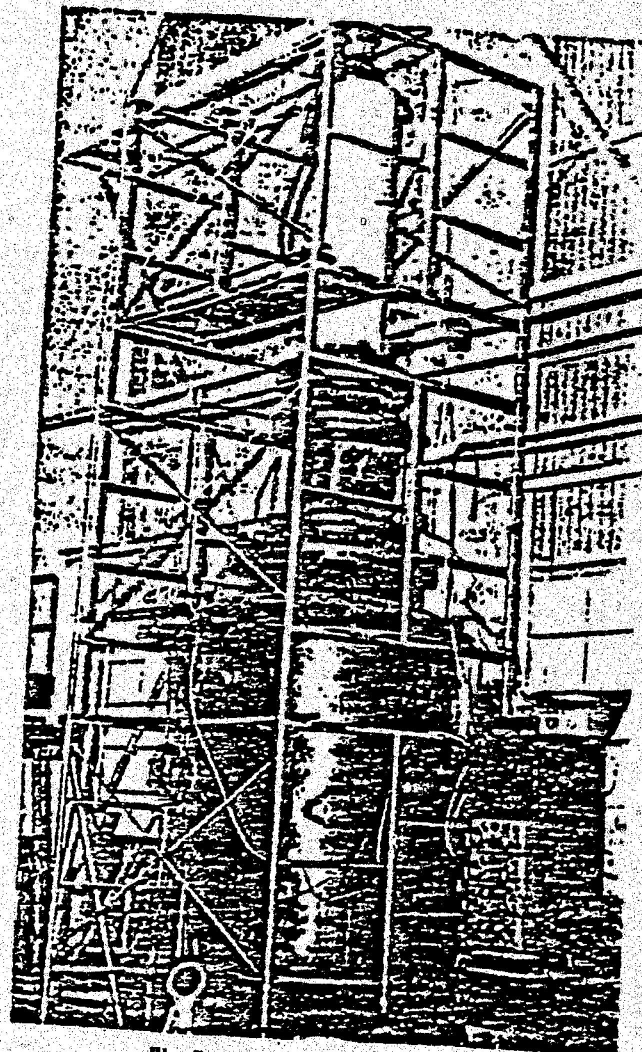


Fig. 7 Transient test facility

Reactor pressure before firing	600, 1000 psi
Fraction of reactor vessel filled with liquid before firing. . . .	0.66, 0.80
Orifice diameter.	0.6, 1.2, 1.6 in.
Dry-well volume	17.5, 20.3, 26.0 cu ft
Vent area	0.5, 0.9, 2.1 sq ft
Vent depth of submergence	0.75, 1.0, 1.5 ft

d) Test Results. A typical set of pressure traces is shown in Fig.8. The reactor-vessel pressure dropped off rapidly in the first 60 millisec, then fell at a slower rate up until about 2.6 sec, and finally decayed exponentially. The three periods probably corresponded to (1) expulsion of water before flashing occurs, (2) flow out of the vessel of a two-phase flashing

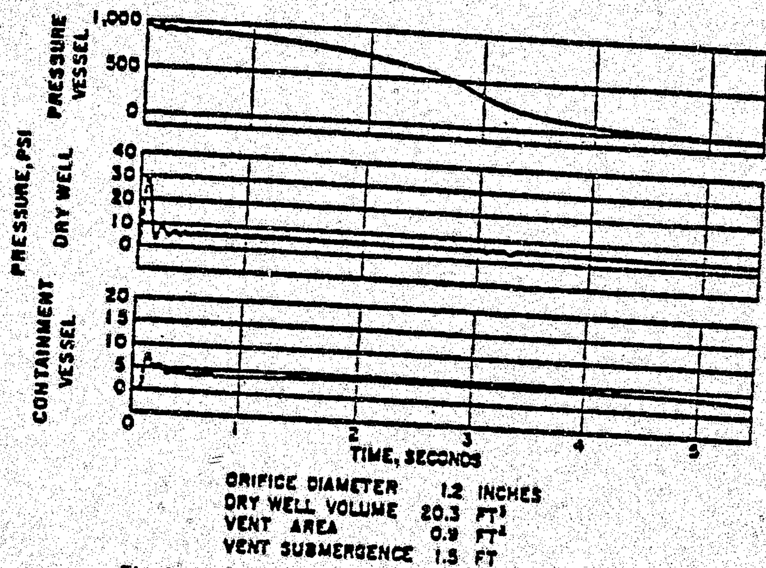


Fig. 8 Pressures recorded during a transient test

mixture, and (3) flow of steam only after all the liquid has been expelled from the vessel.

The dry well pressure built up rapidly, reaching a peak in about 60 millisec and then ...l back rapidly. Within the range of parame...rs 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 container 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 (containment volume) reached a maximum shortly after the dry-well peak pressure occurred. Calculations substantiate the hypothesis that the air in the container was compressed adiabatically and then cooled by water surging into the space carried up by the air coming from the dry well. In all tests, the pressure over the water pool returned to essentially atmospheric pressure seconds after the firing.

e) Comparison of Analytical with Experimental Peak Dry Well Pressures. Peak dry-well pressure transients were calculated mathematically on the basis of certain assumptions. These were:

1 Flow from the reactor vessel up to the ... of peak dry well pressure is saturated liq... at reactor pressure. This assumes that steam produced by flashing in the vessel is not drawn through the orifice by flow.

2 Heat transfer to the dry-well walls is negligible.

3 Flow friction is negligible.

4 No mixing takes place between the pool water, the air, or the steam-water mixture. This enables the air volume to be treated as a perfect gas, undergoing reversible, isentropic changes before the pool water is completely expelled from the vent.

5 Uniform pressure exists in the dry well.

6 The steam-water mixture in the dry well is homogeneous and in thermodynamic equilibrium.

The equations prepared using these assumptions were programmed for a computer to determine dry-well pressure transients.

The experimental results indicate that the analytical method of predicting peak dry-well pressure is satisfactory and conservative. The effect of varying orifice size, dry-well volume, depth of submergence, and vent area was predictable by the analysis. In all tests the experimental peak dry-well pressure was less than and occurred slightly later than the analytical prediction. Experimental pressure peaks were about 60-80 per cent of the analytical values and the experimental peaks took about 20-30 per cent longer to occur. Analysis of the test results indicates that discrepancies between experiment and analysis are most likely caused by flashing in the flow from the vessel, contrary to assumption (1).

Within the ranges tested, some parameters had a greater influence than others on the magnitude of the maximum dry well pressure. In relative order of importance, orifice size had the greatest weight, followed by depth of submergence, dry well volume, and vent area.

3 Fission Product Entrainment Tests. a) Objective. The specific objective of these tests was to determine in preliminary fashion the effectiveness of the water pool as a barrier to fission products. In other words, find out what fraction of fission products carried from the dry well by the steam and air manage to get through the pool to the enclosure space over the water.

b) Test Equipment. The Transient Test Facility was used with some special equipment for releasing the simulated fission products and for measuring the quantity escaping through the pool to the enclosure air space. Operation of the facility was essentially the same as with the tests to investigate pressure transients.

c) Test Program. Samples of both krypton and xenon were used as tracers to simulate fission gases. The gas, contained initially in glass bottles placed in the reactor vessel, was released rapidly when the bottles shattered as a result of the sudden drop in reactor-vessel pressure during the test. Air samples were taken immediately after firing from the enclosure over the water pool and the quantity of noble gas present determined by a mass spectrometer. The gas sample volume initially in the reactor corresponded to about six times the maximum amounts of these gases expected in all the fuel elements of a reactor, considering the volumetric scale of the tests of 1:1000. Even so, the amount of krypton in the air space was close to the lower limits of detection, and the amount of xenon in the air was too low to be measured by the mass spectrometer.

Soluble Salt. Sodium iodide was placed directly in the reactor-vessel water. Right after firing, a sample of air from the space above the water pool was drawn through a scrubbing column. The water used as the scrubbing agent was analyzed for iodine content using the colorimetric method.

Solid Particles. Fluorescent zinc sulfide with a mean particle size of 2 microns was placed directly in the reactor-vessel water. This test was run at the same time as the sodium-iodide test. An air sample from above the water pool was taken about 1/2 hr after firing and passed through filter paper. The number of particles on the filter paper were counted using a microscope under ultraviolet illumination.

Halogen. Elemental iodine crystals were [illegible]ced in the reactor-vessel water in the last [illegible]t. During heating up, the rupture-disk holder developed a severe wire-drawing leak, so the transient was initiated at a reactor pressure of 575 psig instead of the planned pressure of 1000 psig used for all other entrainment tests. Air samples were drawn through the scrubbing column right after firing, and again 5 hr after firing. The scrubbing agent, water, was analyzed for iodine content.

TABLE 1 RESULTS OF TESTS ON ENTRAINMENT OF SIMULATED FISSION PRODUCTS IN WATER

Element	Initial Amount in Pressure Vessel	Percent Released to Containment Vol.	Concentration in Containment Vol. [illegible]
Krypton	250 cc. (std)	[illegible]	[illegible]
Xenon	250 cc. (std)	Less than 5%	Less than .00[illegible]%
Iodine	1 lb.	5.8×10^{-3}	1.83×10^{-8}
		$7.2 \times 10^{-[illegible]}$ @ 5 hrs.	2.3×10^{-7} @ 5 hrs.
Sodium Iodide	110 gm	$2.7 \times 10^{-[illegible]}$	1.4×10^{-6}
Zinc Sulfide	110 gm	2.1×10^{-6}	2.7×10^{-10}

Containment Volume (after firing) = 78.5 ft^3 (air)
Pool Volume = 175 ft^3
Pressure Vessel Volume = 1.12 ft^3

d) Test Results. Test results for simulated fission product entrainment are presented in Table 1. It is evident 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.

Results of the test are sufficient to show qualitatively that the water pool is an effective barrier. However, the precise numerical separation factors obtained with the tests cannot now be claimed for a full-scale pressure-suppression containment pool because of the unknown effect of factors such as scale, geometry, and chemical species.

DESIGN CONSIDERATIONS IN PRESSURE SUPPRESSION SYSTEMS

The pressure-suppression system used as a general frame of reference for the following section is that shown in Fig.1.

Dry Well

The dry well should maintain its integrity

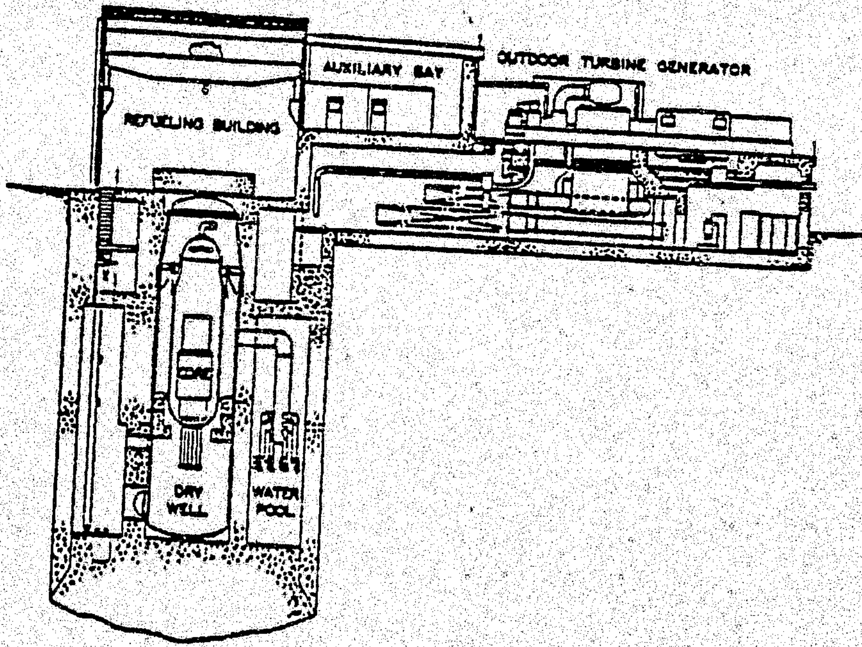


Fig. 9 Elevation of a pressure-suppression-containment system

following the unlikely occurrence of a break in the primary system. If the dry well were to break, the water pool could be by-passed, the en-closure could rupture, and any fission products released to the dry well might escape to the environs and be a health hazard. Threats to dry-well integrity are blast, dynamic force, missiles, and static pressure.

1 Blast. If a rupture of the primary system were to occur suddenly, a shock wave would form in advance of the interface of the air and steam-water issuing from the break. The dry well should withstand the resulting blast. Simplified calculations were made, treating the shock wave as being plane rather than spherical. The results compared with other factors to be considered, showed that the blast pressure on the dry well wall would not be a controlling design condition. The analysis gave a blast pressure of 248 psi at the dry-well wall for an initial reactor pressure of 1000 psia and an initial pressure of 14.7 psia for the air in the dry well.

2 Dynamic Force. Dynamic force from a jet issuing from a break in the primary system is an important consideration in design of the dry well. The dynamic force of the jet on the wall can be calculated readily by assuming flow through the break to be equivalent to an orifice, and by assuming that the direction of the fluid after striking the wall is 90 deg from the initial direction. The dry-well wall under the circumstances should be able to withstand the momentum of the jet. Because of the directional character of the jet, its high velocity, and the short distance between dry-well wall and break, the jet would not be likely to expand very much by the time it struck the dry-well wall. Consequently, the force of the jet would be concentrated in an area not much bigger than the break in the primary system. With the assumptions listed, the total dynamic force would be directly proportional to the break area and to the initial reactor-vessel pressure. If the initial reactor pressure is 1000 psia and the break area is 0.785 sq ft, the dynamic force on the dry-well wall is calculated to be 145,000 lb on the basis of the assumptions described.

3 Missiles. The dry well must also be designed so it would not be breached by any missiles which could result from parts being torn loose and accelerated by a jet of steam-water. The reactor vessel itself should be braced so that it could not become a missile from the reaction force of a jet.

4 Static Pressure. If a rupture were to occur in the primary system, static pressure would rise in the dry well, reach a maximum, and then drop off. A most important requirement of a dry well is that it be capable of withstanding the peak static pressure. The following factors affect peak static pressure, and the importance of each depends on the design conditions:

Volume of the reactor vessel.

Initial state of vessel contents such as

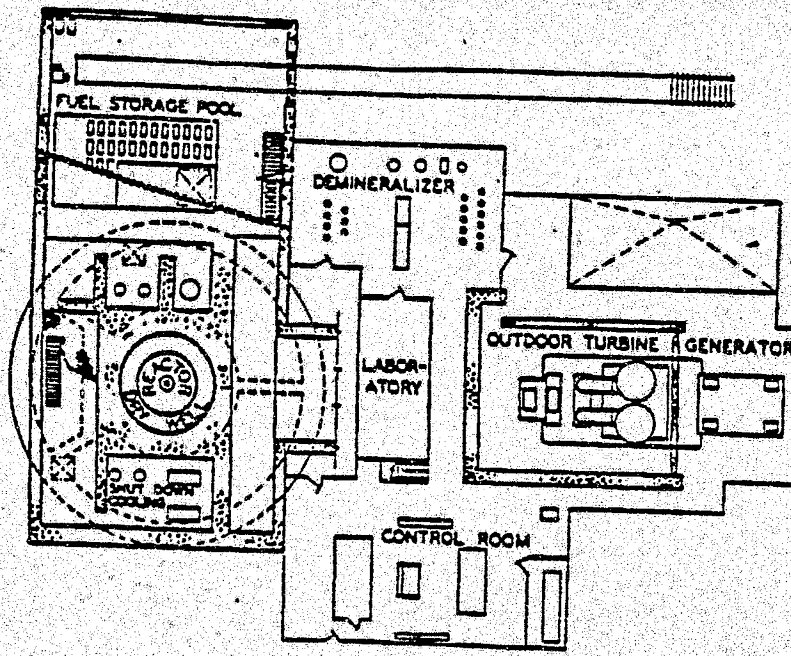


Fig. 10 Plan of a pressure-suppression-containment system

pressure and amount of steam and water.

Size and location of the break.

Volume of the dry well.

Vent flow area.

Vent depth of submergence.

If the vent area was large enough, an increase in it would have a small effect on reducing the peak dry-well pressure. Under these circumstances, after a break, pressure would build up in the dry well, accelerating the water down out of the vents. Peak dry-well pressure would occur when the water has just been expelled and the pressure would drop rapidly as soon as the air and steam in the dry well could blow directly in to the water pool. The peak pressure would be determined by the time required to move the mass of water out of the vents. With these conditions, depth of submergence would be an important factor in determining peak static pressure in the dry well.

If the vent area was too small compared with the size of the break, the water pool would not be effective as a heat sink; peak pressure would have been reached before a significant amount of air and steam from the dry well could be forced through the vents in to the water. In this situation, the volume of the reactor vessel, initial state of the reactor-vessel contents, and the dry-well volume would be the most important factors in determining peak static pressure.

If the vents were designed to utilize the water pool effectively (large vent area and small depth of submergence) the relative amounts of water and steam in the reactor would be unimportant. This is true because the peak static pressure in the dry well would be determined by the flow from the vessel in the first short period of time. This flow would be a small fraction of the total contents of the vessel.

The location of the break in the primary system of a boiling water reactor is also important in determining peak dry-well pressure for an effectively designed pressure-suppression containment system. Analysis shows that for a given break and an initial pressure of 100 psia the energy release rate to the dry well is about five times as great for an all-liquid flow as for all steam flow. Consequently, a break at a bottom connection to a boiling-water reactor vessel will create a higher peak dry-well pressure than will one at the top.

Injector Equipment or Vents

The tests indicated steam can be condensed in the water pool very satisfactorily by discharging through simple straight pipes. Other steam injectors would probably operate satisfactorily also, but there is now no known advantage in using more elaborate equipment.

Within the range of the tests, no significant diameter effect was found. Tests included pipe diameters of 4, 6, 8, and 14 in., although the steam supply was inadequate to establish jet flow from the 14-in-diameter pipe.

The open end of the vents should be submerged sufficiently so that the jet momentum does not depress the water surface below the end of the pipe. Depth of submergence should be small to keep peak dry-well pressure low, but the submergence should be large enough to insure that discharge is always far enough below the surface of the water to obtain complete condensation.

Condensing Pool

The main functions of the condensing pool would be to condense steam released from a reactor accident and to retain radioactive fission products that might be released with the escaping steam. Depending on the arrangement, the pool may also serve as a biological radiation shield, source of water for post-accident dry-well flooding, heat sink for emergency pressure-relief system, heat sink for emergency load-dump system, and storage for fuel.

The volume of pool water should be large enough to absorb the potential energy release from the reactor water with a maximum water temperature of about 120-130 F. Tests indicated that pressure waves in the water during high steam-discharge rates may be large if the water temperature exceeds this value. With this limitation, the volume of water depends upon the total energy released, and the initial water temperature. It is important that the vent pipes and water pool be so arranged that good mixing will take place.

Containment Vessel

The primary function of the containment vessel over the water pool would be to retain radioactive isotopes that might be released. It is, in effect, a back-up on the water pool and dry well.

The pressure rise in the containment vessel would be created largely by the air purged from the dry well. With proper design, the design pressure for the enclosure may be only a few psig. The magnitude of the pressure rise in the containment vessel is a function of system parameters, such as, ratio of containment volume to dry-well volume, ratio of initial and containment pressures, and ratio of initial dry well and containment temperatures. Vapor pressure of the water pool is not apt to be very significant if the water-pool temperature does not exceed 120-130 F, recommended as a limit for other reasons.

PRESSURE SUPPRESSION SYSTEM DESIGN

The results of the development program are now being applied in further work aimed at designing a pressure-suppression containment scheme for Humboldt Bay Unit No. 3.

The reactor system including the primary vessel, dry well, and pressure-suppression pool, together with shielding and containment provisions, will be designed to minimize environmental hazards from leakage of fission products and radiation shine under the most severe credible accident conditions.

Different arrangements of the containment system are being studied. Sketches showing what a cross section and plan of the Humboldt Unit may look like are given in Figs.9 and 10. With this concentric pool layout the reactor and pressure-suppression system, which are closed at all times when the reactor is operating, are below ground level. An above-ground building is provided over the reactor to enclose all refueling operations.

CONCLUSIONS

Pressure suppression offers important safety advantages and shows considerable promise of being less expensive than dry-capsule-type containment for boiling-water or pressurized-water reactors. The development program conducted substantiates these conclusions and provides a basis for design of pressure-suppression systems which, if ever needed, would operate in a safe, predictable manner.

Acknowledgements

The information in this paper is a result of the efforts of many individuals. Special mention must be made of the important contributions of W. L. Flock, E. Janssen, and A. G. Steamer of General Electric Company and J. O. Schuyler and D. B. Barton of Pacific Gas and Electric Company.

Q. IR-OCA-28-12

Would Dr. Levy agree that during tests at Vallecitos in 1958 or 1959, GE witnessed a) waterhammer effects during the steam generating process; b) water being repeatedly drawn into the vent pipe and discharged in a pulsating fashion coinciding with an audible water hammer, c) distinct bubble formations were expelled from the pipe mouths during the discharge portion of the pulsation cycle.

A. IR-OCA-28-12

As described in his direct testimony, Dr. Levy would agree that noises due to bubble formation and condensation were experienced at the Vallectios tests. Some of the noises may have been labeled "water hammer" effects. Water hammers are associated with large pressure changes, but the pressure transients measured at Vallectios were in fact quite small, if not negligible. Similar effects were described in a published paper, ASME Paper No. 59-A215, a copy of which is attached to IR-OCA-28-11.

These limited hydrodynamic effects were not then and are not now perceived to be significant. As noted above, these small scale tests had as their purpose the testing of the possibility of steam condensation in a pool of water. It was not their purpose to identify or evaluate hydrodynamic effects associated with the transient quenching operation during a LOCA. Indeed, it was not feasible to do so based upon these tests. These were steady state tests and the configuration and size of the test tank and downcomer pipes employed could not be conformed to actual or planned plant containments as these had not yet been designed. Such conformance is necessary to attain useful data as respects the design significance, if any, of hydrodynamic loads. As described below, later tests were conducted jointly by GE and PG&E at the latter's Moss Landing facility employing the proposed Humboldt Bay containment configuration and size to obtain this data.

A steady state test involves the continuous injection of steam into the water pool. By contrast, actual LOCA or transient conditions involve initial large steam flow for a very brief period which then continuously decreases thereafter. Hydrodynamic effects of these two conditions on the suppression pool are substantially different.

In Dr. Levy's opinion, the Vallecitos tests provided little useful data as to hydrodynamic effects of the steam generating process. This is in part due to the small size of the facility and its difference in configuration, as well as the fact that it lacked various specific features included in actual power plants to mitigate hydrodynamic effects. An example of such a feature is vacuum breakers. These breakers return air to the drywell specifically to reduce pressure in the wetwell and thereby mitigate the capability to drawback water into the vents.

The principal conclusion derived from these tests was that steam condensation could be successful and that further tests were needed to validate the pressure suppression concept. The test results fully supported this conclusion. I should note that the test tank was only the size of an office desk with a one or two inch downcomer pipe, as compared to the much larger suppression pools and 24" downcomers employed in actual plant configurations.

Responsible Witness: Salomon Levy
S. Levy Incorporated

Q. IR-OCA-28-13 If Dr. Levy agrees with any part of Question 12 above, does he believe these events or effects were related or unrelated to hydrodynamic loads known as chugging. Please explain.

A. IR-OCA-28-13 These effects were unrelated to the hydrodynamic load subsequently identified as "chugging". Waterhammer effects involving vacuum conditions in the steam chamber, such as occurred at Vallecitos, are different from chugging. When chugging occurs, the drywell pressure always remains above the pressure in the suppression pool; minor pressure oscillations develop in the vent piping where condensation takes place intermittently inside the vent. Such pressure oscillations were noted in my testimony (line 15, page 11) and were reported in a paper published by General Electric and Pacific Gas and Electric in August 1962 in Nuclear Engineering. A copy of this paper is attached to this interrogatory.

Responsible Witness: Salomon Levy
S. Levy Incorporated

ATTACHMENT IR-OCA-28-13

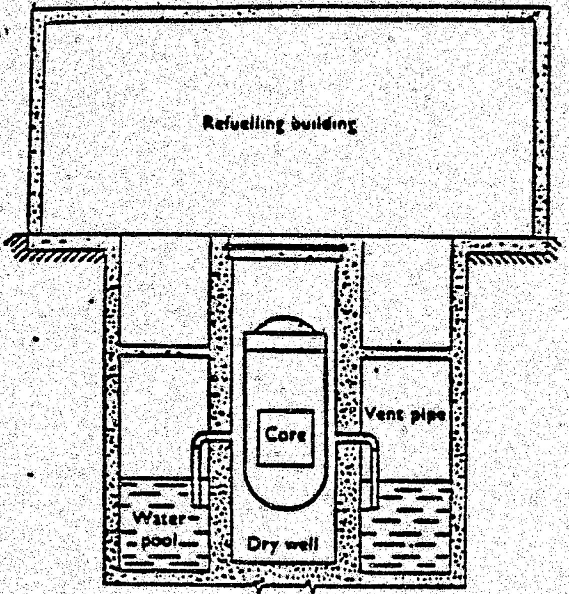


Fig. 1.—Simplified arrangement of containment.

In early 1958 Pacific Gas and Electric contracted for the construction of a nuclear unit at their Humboldt Bay Power Plant near Eureka, California. General Electric's Atomic Power Equipment Department will supply the reactor, instrumentation, controls and nuclear fuel. The reactor has a rated capacity of 50 MW, is light water cooled and moderated with natural circulation and internal steam separation. Output may be increased to 70 MW. The original containment design included a large steel cylinder. In order to reduce containment size and cost without reducing safety, Pacific Gas and Electric investigated a different type of containment known as pressure suppression. This development programme—carried out with the assistance of the General Electric Company—was successful and, following receipt of a construction permit from the U.S.AEC, pressure suppression containment was applied to Humboldt Bay. A description of this type of containment, its development, application and advantages are the subjects of this article.

Pressure Suppression

by C. P. ASHWORTH
D. B. BARTON
(Pacific Gas and Electric Co., San Francisco)
and C. H. ROBBINS
(APED, General Electric Co., San Jose)

THE basic concept of pressure suppression involves quenching the steam issuing from a break in the reactor system in a water pool—as opposed to containing the steam under pressure in a very large vessel. Fig. 1 shows the type of pressure suppression containment used at Humboldt Bay and described in this article.

The boiling water reactor and primary coolant system—up to double isolation valves—is completely contained in the dry well (including the guard piping for the penetrations). The dry well is a pressure vessel designed for the maximum credible internal pressure, the determination of which is described later. The dry well is vented to a pool of water by large flow area vent piping. The water pool is located in the vapour-tight suppression chamber which contains a large air volume. Fig. 1 shows a refuelling building which may be used in conjunction with the basic suppression system to collect system leakage and to provide containment during refuelling.

In the unlikely event of a primary system rupture within the isolation valves, reactor water and steam would discharge into the dry well. This might be accompanied by jet forces, a shock wave and possibly missiles. The resulting dry well pressure would force the pool water out of the vents and a mixture of air, steam and water then could flow from the dry well through the vents into the pool. The steam from the rupture would be condensed rapidly and completely in the suppression pool resulting in relatively low dry well and chamber pressures.

RELATION OF PRESSURE SUPPRESSION TO PLANT SAFETY

Pressure suppression containment is one of several barriers which can act to prevent the escape of fission products from the reactor to the environs. Other possible barriers include: (1) the fuel cladding; (2) the reactor vessel and its associated piping circuits, and (3) a refuelling building which encloses the pressure suppression containment.

The refuelling building has a limited volume gas exhaust processed through a gas treatment system and then released via a tall stack. Emergency core spray cooling systems prevent fuel cladding melting or the core can be kept immersed by flooding the dry well.

Where pressure suppression can be used, certain safety features and benefits are available if needed.

1. If through failure of protective barriers, fission products do escape from the fuel, the rapid reduction of primary containment pressure removes driving force for significant leakage from the primary containment. Pressure build-up in the primary containment vessel (dry well) following an accident would be rapidly suppressed by condensing of steam in the suppression pool. Well before fission products would be released from the fuel in the event of core heat-up, the driving force necessary to cause leakage from the primary containment would be reduced to near atmospheric pressure.
2. If fission products are carried to the suppression pool by steam during the initial pressure build-up, they would be scrubbed by and largely retained in the pool water. Non-condensible gases would be retained in the vapour space above.
3. The plant may be designed to have substantial shielding of the primary system containment by placing it below ground level. This would virtually eliminate the significance of any direct radiation from the containment following a fission product release incident.
4. The compact dry well and suppression chamber can be readily built to have low leak rates—less than 0.1% of the volume in 24h. The back-up of the dry well by the concrete biological shield further reduces the chance of fission products escaping to the environs.
5. The pressure suppression system with a refuelling building can be highly resistant to forces of external origin such as falling aircraft or unusual weather conditions.
6. The dry well surrounding the reactor vessel may be readily filled with water to provide post-accident core cooling. This feature is beneficial if some types of major reactor repair become necessary.

DEVELOPMENT PROGRAMMES

A. General

To provide a basis for design of pressure suppression, development programmes were conducted by Pacific Gas and Electric Company and General Electric's Atomic Power Equipment Department. Many of the results, such as blow-down rates and jet condensation, are not restricted in application to pressure suppression.

Prior to the design of the Humboldt Bay containment, testing was performed principally in two specially-built facilities. The first was a large tank in which steady flow of steam was used to investigate condensation. The second facility was built as a reduced scale model of a reactor and pressure suppression system to explore the transient behaviour of the entire system and the effects of various design parameters. A third facility was later constructed and operated to proof test and demonstrate the specific Humboldt containment design.

B. *Condensing Test Facility*

The objective of the tests was to determine the effect on condensing steam jets of various parameters including injector pipe diameter, depth of submersion, pool geometry, water temperature, steam flow rate, direction of discharge and effect of neighbouring jets.

The equipment, shown in Fig. 2, consisted of a 20 ft diameter tank containing water with means for discharging steady steam flows up to 100 000 lb/h at a supply line pressure of 100 psi. Injector pipes ranged from 4 to 14 in d. In many tests, an open top box (as shown in Fig. 2) restricted the volume of water surrounding the discharge pipes.

The tests demonstrated the extreme rapidity of steam jet condensation. The jets stirred up the water vigorously and complete condensation was obtained in all but a few tests where the injector pipe end was barely below the water surface. No large steam bubbles were observed. The degree of water surface depression in the vicinity of the jet varied with the pool

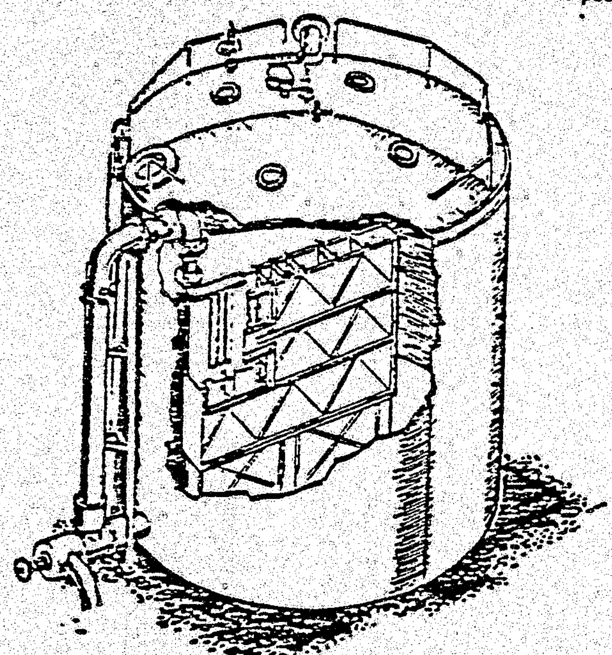


Fig. 2.—Condensing test facility.

geometry as well as steam flow rate. The condensing tests proved the effectiveness of straight pipes discharging downward and all subsequent work was done with this basic arrangement.

C. *Transient Test Facility*

The tests with the first small-scale facility had three objectives.

1. To observe the operation of a system simulating an accident and thereby gain insight into the behaviour.
2. To measure transient pressures to compare with analytical predictions.
3. To obtain preliminary information on the ability of the water pool to retain fission products.

The facility consisted basically of three interconnected pressure vessels simulating a reactor vessel, dry well and outer enclosure over a water pool. The schematic arrangement and instrument location is shown in Fig. 3. The relative position of the reactor vessel was different from a pressure suppression system to facilitate test changes. Also the vent pipes, compared with likely pressure suppression designs, were free of bends and of large area, resulting in a low flow resistance.

Instrumentation was provided to measure temperatures and transient pressures during a run. The system was chosen to have a volume scale about 1/1000 of the corresponding parts of the Humboldt design as it was then anticipated. With corresponding area and linear scales, the time scale was 1/10 of Humboldt. The pressure vessel simulating the reactor had a volume of $3{\cdot}12\ ft^3$.

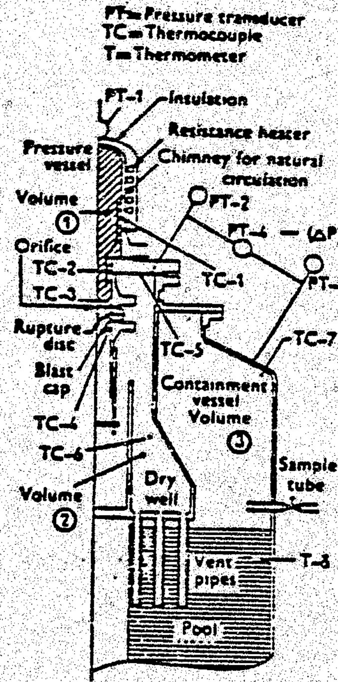


Fig. 3.—Arrangement of transient test facility.

The test procedure was to fill the vessel partially full of water, heat it to a pressure of about 1 000 psi, and rupture a disc to allow the water to flow through the orifice into the dry well and then into the suppression pool. The orifices were changed to simulate different primary system break sizes. In addition the dry well volume, vent area, and vent depth of submergence were varied.

A typical set of transient test pressures is shown in Fig. 4. Test results were in substantial agreement with the analytical predictions, except for the flow rate from the reactor vessel. Lower flow rates were obtained through the orifice than had been calculated using an orifice equation, a coefficient of 0·61 and the density of saturated water. The maximum suppression chamber pressure measured indicated that condensation was complete and effective and that for this facility the air from the dry well was not completely transferred to the suppression chamber.

In one set of tests, simulated fission products were placed in the reactor vessel before the test rupture and the amounts present in the air above the water pool were measured after the

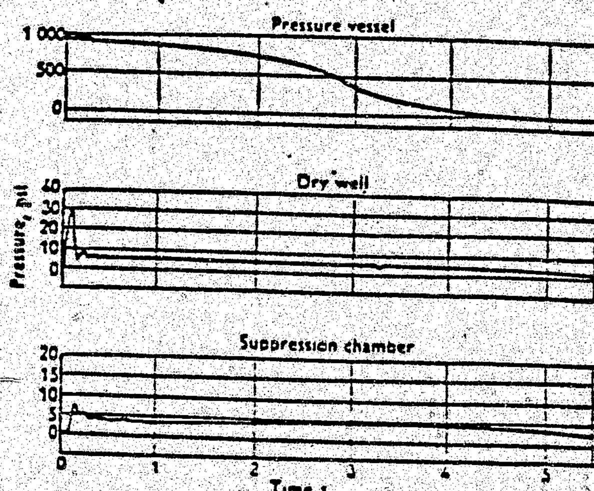


Fig. 4.—Typical pressure traces of initial transient tests.

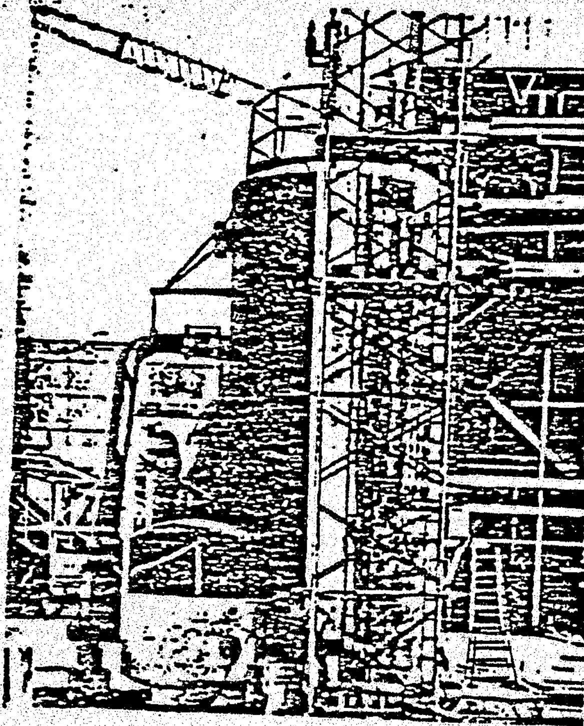


Fig. 5.—Humboldt test facility.

test. Xenon, krypton, sodium iodide, elemental iodine and zinc sulphide particles with a mean size of 2 microns were used. Test results must be considered as preliminary but encouraging.

The measured separation factor between the reactor vessel and the enclosure was of the order of 10^{-4} for the small solid particles and about 10^{-3} to 10^{-4} for the halogen and soluble salt. More than half of the noble gases were retained initially; this is considerably more than expected on the basis of solubility.

D. *Humboldt Full Scale 1:48 Segment Demonstration Tests*

A larger transient test facility was constructed to proof test and demonstrate the Humboldt design. The equipment simulated, as nearly as possible, one of the 48 Humboldt Bay vent pipes, the portion of the suppression chamber and pool associated with it and an appropriately sized dry well and simulated reactor vessel. Fig. 5 is a photograph of the facility, and Fig. 6 shows a schematic diagram. The segment of the Humboldt suppression chamber which the test chamber represents in full size and shape is shown in heavy outline on Fig. 7. The time scale is the same in these tests as it is in Humboldt and is not shortened as was done in the previous small transient tests.

The simulated reactor vessel was a vertical 32 ft length of 20 in o.d. Schedule 80 pipe with a volume of 55·8 ft³. The transient tests were initiated by breaking rupture discs, causing flow to pass from the reactor vessel through different sized sharp edge orifices ranging from 0·14 to 3·28 in d. A 1·64 in d orifice equals 1/48 of the area taken as the maximum credible rupture for Humboldt. The test dry well volume was approximately 1/48 of the corresponding Humboldt volume. The piping which carried steam and water into the dry well was made oversize to direct rupture flow without introducing significant losses. It was terminated inside the dry well with different fittings for different tests, as shown in Fig. 8. Arrangements A to E were used to produce different amounts of water carry-over in the steam leaving the dry well and also to influence the time of air discharge. The piping connecting the dry well with the suppression chamber was 14 in d. and was designed to have the same volume and flow resistance as the Humboldt design.

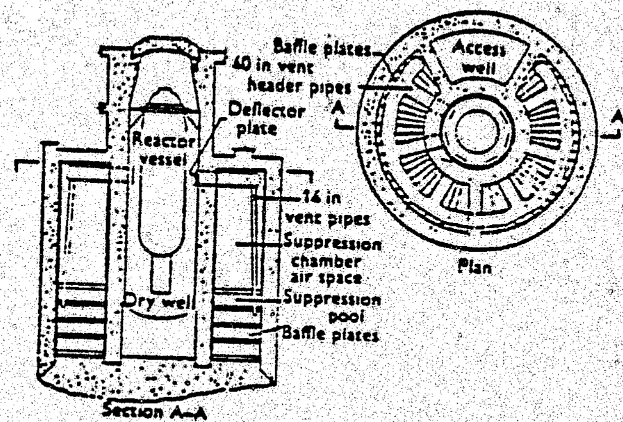


Fig. 7.—Section of Humboldt suppression chamber represented by tests.

The trapezoidal shaped suppression chamber was 12 ft by 2·35 ft by 1·23 ft in horizontal cross-section. The entire chamber was 49 ft high and normally was filled with water to a depth of 18 ft. The walls were insulated to reduce condensation. The suppression chamber and the 14 in vent pipe were a full-scale model of a Humboldt vent and the portion of the suppression chamber associated with it. Provisions were made for transient measurements at the points shown on Fig. 6. Transient pressures were measured by strain gauge transducers and were recorded by a light beam oscillograph.

The performance of the system was investigated over a wide range of conditions simulating accidents both more and less severe than the maximum rupture assumed for the Humboldt design. The following parameters were varied.

(a) Flow rate was controlled by nine different size orifices ranging from 1/137 to four times the area of the 1·64 in orifice representing the assumed Humboldt maximum rupture.

(b) The effect of air in the dry well was investigated by varying the time air entered the vent and by adding air to the dry well during a run.

(c) The amount of water left in the dry well and not carried with the steam into the vents was varied between about 10% and 90% of the maximum possible.

(d) The initial vent depth of submergence was varied by raising or lowering the water level in the pool. The maximum submergence was 12·5 ft and the minimum occurred with the

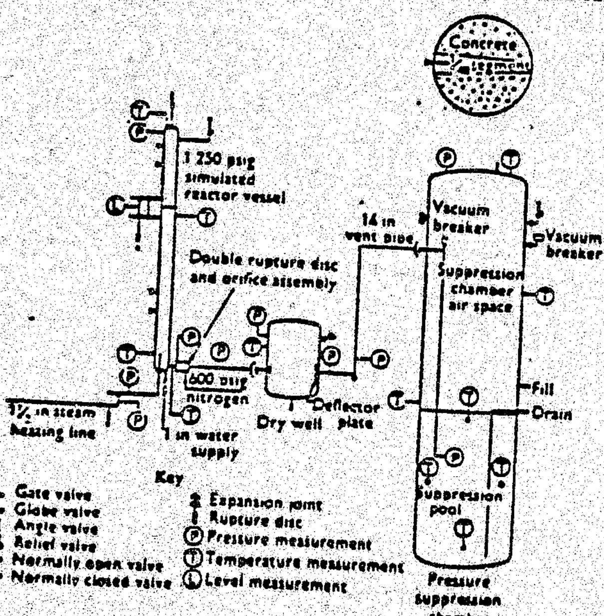


Fig. 6.—Schematic diagram of Humboldt test facility.

vent 3 ft above the water. The Humboldt design has 6 ft submergence. Varying the depth of submergence also varied the volume of pool water from 50% to 170% of the design value.

(e) The initial temperature of the pool varied between 60°F to 140°F and the final temperature was as high as 160°F. (The Humboldt pool design is based on an initial temperature of 80°F and a final temperature of 126°F.)

(f) The initial reactor vessel pressure was 1 250 psig in all tests except one when the pressure was 1 000 psig.

Fig. 9 is a plot of three representative test results using a 1·64 in orifice in the rupture assembly. The differences in these tests are due to the use of different dry well internal arrangements. Test No. 44 used Arrangement A (Fig. 8) which gave high water carry-over to the suppression chamber, indicated by the amount of water in the dry well after a test. Test No. 25 used Arrangement C which gave low water carry-over. Test No. 27 used Arrangement B which gave medium water carry-over and also resulted in a delayed venting of dry well air.

Fig. 10 is a plot of three tests using large orifices with diameters of 2·32, 2·84 and 3·28 in. Table 1 lists the relevant test data.

The top plots on Figs. 9 and 10 show that flow from the simulated reactor vessel proceeds in the same manner as in the small transient tests. After an initial sharp drop and partial recovery, the pressure declines while the water is being expelled. Then, at about 17 s in the case of Fig. 9, the pressure starts to fall more rapidly, indicating that the water is gone from the reactor vessel, and the decay curve corresponds to a blow-down of a steam-filled vessel.

Dry well pressure rises rapidly during the time water is being forced down out of the end of the vent and then pressure remains fairly constant when the flow resistance in the vents becomes the dominant factor.

The suppression chamber pressure curves differ in shape mainly because of the dry well internals, as is illustrated by Fig. 9. Internals affect timing of air transfer, hence the manner of pressure in the suppression chamber.

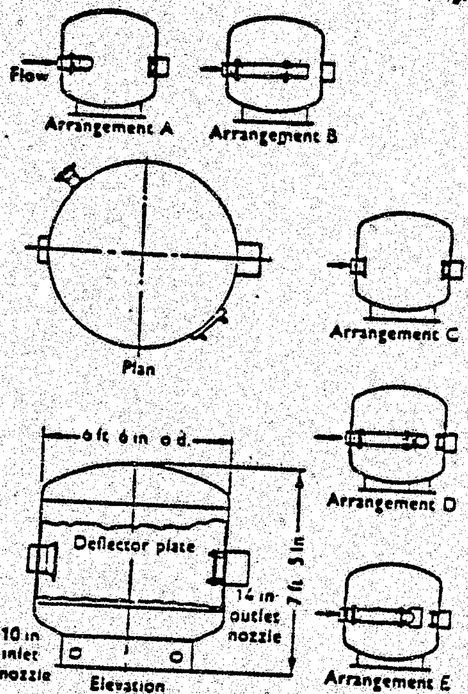


Fig. 8.—Dry well internals.

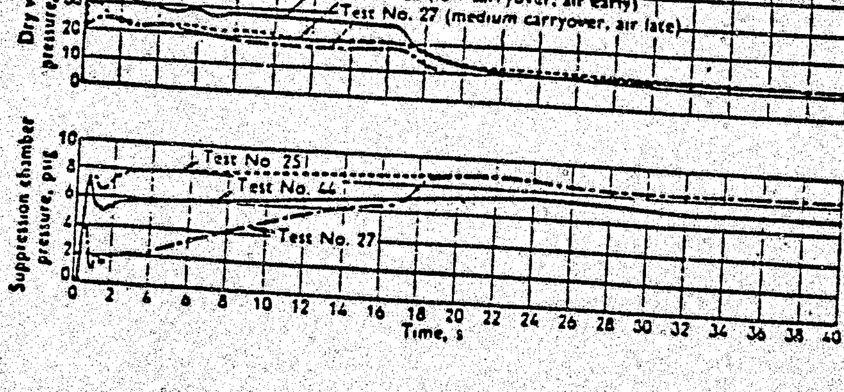


Fig. 9.—Representative test results for 1·64 in orifice.

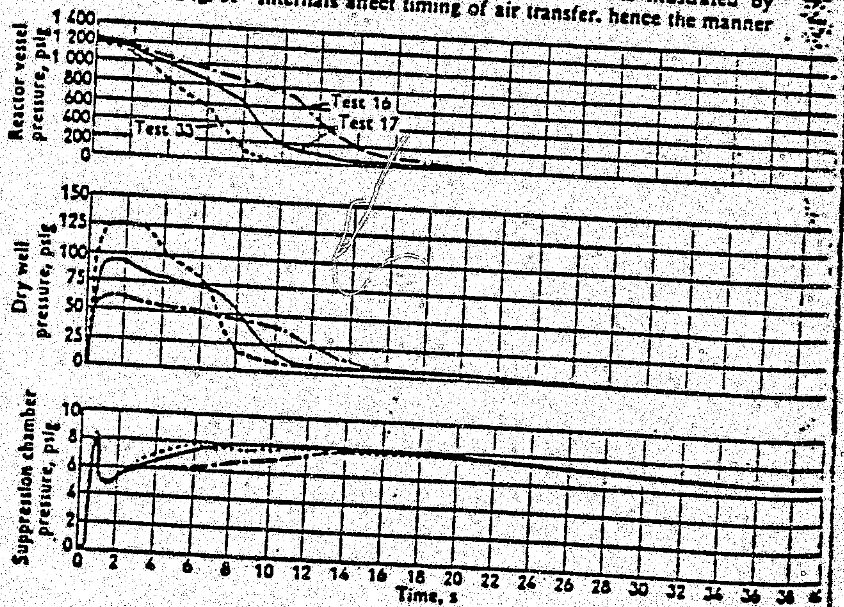


Fig. 10.—Test results for large orifices.

At the beginning of the test, chamber pressure is zero until the pool water is cleared from the vents at about 0·3 s on Fig. 9. The transient chamber pressure which peaks during the first second may be explained by the sudden expansion of air from the vents into the pool. This expansion throws some of the pool water up into the chamber air space compressing the air. As the water breaks up and falls back to the pool, the chamber pressure falls off. This initial peak did not occur on any tests where the vent was initially out of the water.

Rupture Flow Rate

The average flow rate during the period of water expulsion can be calculated knowing the initial mass in

TABLE 1
TEST DATA

	REACTOR VESSEL			DRY WELL				SUPPRESSION CHAMBER		
	Initial pressure psig	Water volume ft³	Water discharge time s	Orifice diameter in	Internal arrangement	Max. pressure psig	Residual water ft³	Submergence ft	Max. pressure psig	
									Measured	Calculated
5	1 250	38·3	17·2	1·64	A	40	8·8	6	7·0	9·4
6	1 250	39·3	17·1	1·64	B	23	6·7	6	—	—
7	1 250	39·3	17·1	1·64	B	26	5·9	6	8·0	8·8
8	1 250	39·2	16·8	1·64	C	28	15·4	6	8·5	8·6
9	1 250	39·1	12·2	2·10	C	45	14·2	6	7·5	8·3
10	1 250	38·5	10·9	2·32	C	57	13·4	6	6·5	8·8
11	1 250	38·7	—	1·64	C	—	—	3	—	—
12	1 250	38·7	16·1	1·64	C	38	—	12·5	8·0	9·8
13	1 250	38·3	16·4	1·64	D	32	11·8	6	8·1	7·2
14	1 250	39·6	—	1·64	E	32	10·4	6	7·5	7·4
15	1 250	39·6	17·0	1·64	A	33	4·2	6	8·8	8·7
16	1 250	39·4	10·3	2·32	A	64	3·1	6	8·0	8·4
17	1 250	39·4	7·9	2·84	A	90	2·4	6	8·5	7·9
20	1 250	38·7	16·9	1·64	A	33	3·6	−0·5 (3)	6·5	6·6
24	1 250	38·2	16·8	1·64	A	32	2·7	6	9·3	9·4
25	1 250	39·2	16·8	1·64	C	32	18·1	6	8·5	9·1
26	1 250	39·2	16·7	1·64	C	33	—	6	8·5	9·0
27	1 250	38·7	17·0	1·64	B	25	7·1	6	8·0	8·0
28	1 250	38·7	17·0	1·64	B	25	6·8	6	7·8	8·0
29	1 250	49·8	21·8	1·64	A	33	3·7	−1·5 (3)	7·0	6·9
30	1 250	48·9	21·2	1·64	A	35	4·3	−3·0 (3)	14·5	10·2
31	1 250	48·9	1 840·0	0·14	A	5·5	26·2	6	6·0	7·6
32	1 250	39·2	10·7	2·32	A	59	3·7	−2·0 (3)	7·2	6·9
33	1 250	38·7	6·0	3·28	A	128	1·9	6	8·5	7·9
34	1 250	38·7	81·0	0·60	A	10	—	6	6·5	7·5
35	1 000	39·2	31·7	1·10	A	18	9	6	8·0	7·9
36	1 250	39·2	16·9	1·64	A	35	3·9	6	8·2	8·5
37	1 250	39·2	17·3	1·64	F (1)	33	4·8	6	8·0	8·3
38	1 250	39·2	16·3	1·64	F	32	3·5	6	—	—
39	1 250	39·2	16·8	1·64	F	32	—	6	—	—
40	1 250	37·8	16·5	1·64	A	34	4·3	6	11·3 (2)	11·2
41	1 250	38·7	16·9	1·64	A	34	4·2	6	12·5 (2)	12·9
42	1 250	38·7	16·7	1·64	A	37	—	6	13·6 (2)	14·0
43	1 250	37·8	3100·0	0·30	A	8	—	6	7·0	6·4
44	1 250	37·8	16·6	1·64	A	36	4·1	6	7·5	7·2
45	1 250	39·6	10·7	2·32	A	65	3·5	6	7·8	7·6

1. F is arrangement A without vent entrance plate.
2. Extra air injected in dry well during venting period.
3. Negative submergence indicates vent was initially out of the water.
(Tests 5–12 do not represent Humboldt because of large dry well.)

the reactor vessel and the water expulsion time as determined from the pressure trace. For every orifice size tested, the flow rate measured in this fashion was significantly less than the value predicted by calculation methods used for the Humboldt design. A comparison between measured and calculated flows is shown in Table 2. The calculations used the orifice equation with a flow coefficient of 0·61 and the density of saturated liquid.

The assumption of saturated liquid density in the calculation conservatively neglects the effect of steam. As saturated water is expelled some flashed steam may be drawn into the orifice with the water. This effect would be greatest in tests with large orifices and consequent large water velocities inside the vessel. In addition, the flashing to steam during flow through the orifice may be expected to influence the flow.

The maximum dry well pressure was in the range of 25 to 36 psig for tests simulating the Humboldt maximum credible rupture accident. By comparison, the Humboldt dry well is designed for 72 psig. The range of pressures was caused by differences in dry well internals which influence the amount of

TABLE 2
COMPARISON BETWEEN MEASURED AND CALCULATED VESSEL

Orifice, in (d)	Relation of test flow rate to predicted (%)
0·14	80
0·30	80
0·60	80
1·10	62
1·64	54
2·10	46
2·32	44
2·84	41
3·28	39

The flow rate per unit area of orifice was about the same for similar vessel and orifice configurations for the large tests and the small tests described earlier.

water carry-over to the vent and the time air leaves the dry well. The dry well pressure at the end of the test is approximately 8 psig. This pressure represents equilibrium between the hot saturated steam and water remaining in the dry well, and the suppression chamber pressure.

Rupture flow rate significantly affected the maximum dry well pressure. The highest dry well pressure of 128 psig occurred with the 3·28 in orifice while the lowest maximum dry well pressure of 5·5 psig resulted in a test with the 0·14 in orifice.

The initial depth of submergence affected the time required to clear the pool water out of the vents. Increasing initial submergence from 6 ft to about 12½ ft lengthened vent clearing time and clearing pressure by about one-third. Reducing submergence made the vent clearing dry well pressure insignificant in comparison with the quasi-steady flow pressure which occurs during the 1 to 17 s period in Fig. 9.

Between tests 12 and 13 the dry well volume including vents was reduced from 308 to 230 ft³. The earlier tests had about 20% lower vent clearing transient pressures than the later tests, as would be expected, but the change did not appear to have any other effect on dry well pressure.

Heat transfer to the walls of the dry well apparently does not affect pressure in the well significantly. Initial dry well temperatures varied as much as 70°F on similar tests.

Pressure Drop in the Vent Pipe

Calculated pressures along the pipe based on average flow rates determined from test data and the homogeneous model of two-phase flow agree very well with the test measurements. It was not possible to determine with any assurance whether critical pressure occurred at the vent exit because of instrumentation difficulties at this point. However, error in the exit pressure has a small effect on calculated dry well pressure.

Condensation in Water Pool

Condensation was rapid and complete in all tests where the vent pipe was initially submerged regardless of flow rate, added air, submergence and pool temperature. This conclusion is based on comparison between measured and calculated maximum suppression chamber pressures. The calculations made use of measured values of initial and final temperatures and volumes, assumed complete condensation, postulated complete transfer of air from the dry well, and neglected minor effects of heat transfer and relative humidity. As the measured and calculated pressures agree well within the limits of the data, steam must have been condensed effectively. The design pressure for the Humboldt suppression chamber is 10 psig but the maximum pressure with tests simulating the maximum credible accident was 9·3 psig. Flow rate did not have any detectable effect on condensation within the wide range tested.

The presence of substantial quantities of air did not interfere significantly with condensation. Three tests with the 1·64 in orifice were run with an auxiliary pressurized air tank connected through a quick-opening valve to the dry well. The valve was opened five or six seconds after the beginning of the test when steam flow rate in the vents is high. In spite of the extra air, which was nearly equal to the quantity initially in the dry well, steam condensation was rapid and complete.

Steam condensation in the suppression system was shown to be relatively insensitive to vent pipe submergence. Four tests were conducted with the pool water level lowered so that the discharge end of the vent pipe was completely out of the water at the start of the tests. The most extreme test with complete condensation was run 32 with the vent initially 2 ft above the water and an orifice corresponding to twice the Humboldt maximum credible accident. Incomplete condensation was observed only in run 30 with the vent initially 3 ft above the water. The observed suppression chamber pressure rose after the water was expelled from the reactor vessel and reached a value of 4·5 psig higher than predicted for complete condensation. For complete condensation the jet momentum must be sufficient to bring the steam and pool water into intimate contact.

The pool water can be hot and can experience a large temperature rise and still condense the steam effectively. The Humboldt pool is normally at 80°F and would rise 46°F following the assumed maximum credible accident. By comparison, complete condensation was achieved in run 26 when the initial water temperature was about 140°F. In another run a temperature rise of 58°F did *not* have any effect on condensation.

DESIGN CONSIDERATIONS

Major considerations in pressure suppression design are rupture flow rates, jet and other dynamic forces, and static pressure rise in the dry well and suppression chamber. These considerations are discussed in this section with particular reference to the Humboldt Bay containment.

Rupture Flow Rate

A pressure suppression system is designed to protect against primary system ruptures of any size up to some maximum. For Humboldt, the maximum was assumed to be a complete severance of the largest pipe, a 12 in schedule 80 line with an area, A, of 0·705 ft². A rupture of this magnitude would be very unlikely in a carefully designed and built primary cooling system like Humboldt.

The effect of rupture size on maximum dry well pressure is illustrated by Fig. 11. For very large ruptures, venting would be ineffective and the pressure represents the pressure of the dry well as a closed vessel. For rupture sizes in the middle of the curve, dry well pressure depends on rupture flow rate and the flow resistance in the vent piping. For small ruptures (or very large vent areas), the initial pressure build-up to clear water from the vents may predominate, and the pressure is a function of the initial depth of water inside the vents. This depth would equal the submergence of the vents if dry well and chamber pressures were equal.

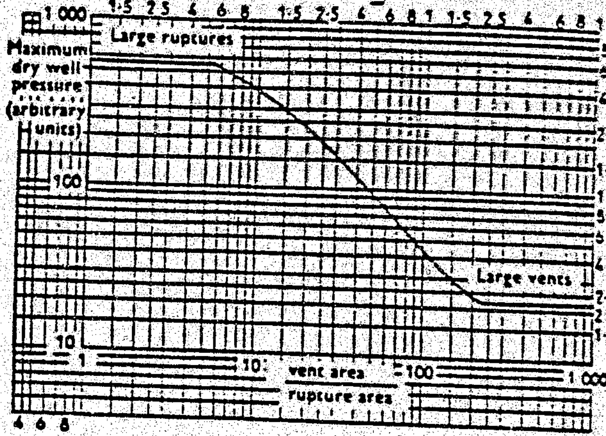


Fig. 11.—The effect of rupture size on maximum dry well pressure.

The rupture flow rate for Humboldt was calculated from the conventional orifice flow equation assuming liquid flow through a squared-edged orifice with a discharge coefficient, C, of 0·61. For dry well pressure in the range of 100 to 0 psig and 1 250 psig reactor pressure (water density, ρ, of 44·4 lb/ft³), the calculated flow is as follows (gravity constant, g, is 32·2 ft/s²):

$$
\begin{aligned}
w &= 12\, CA\sqrt{2g\rho \Delta P} \\
&= 12 \times 0{\cdot}61 \times 0{\cdot}705\sqrt{64{\cdot}4 \times 44{\cdot}4 \times (1\,250-100 \text{ to } 0)} \qquad (1) \\
&= 9\,350 \text{ to } 9\,750 \text{ lb/s}
\end{aligned}
$$

The demonstration tests indicated that actual flow rate would be significantly less than the result of the orifice equation as indicated in Table 2. The actual flow rate for the Humboldt design condition would be only about 5 000 lb/s which introduces a significant margin in the Humboldt dry well design pressure. At the time Humboldt was designed it did not appear that actual flow rate could be justified as a design basis because of insufficient experimental data.

Dynamic Effects

The dry well should be designed to withstand the dynamic effects that might accompany a primary system rupture. Probably the most significant dynamic effect in a pressure suppression containment would be the impact of the jet from the rupture on the wall of the dry well and the reaction on the primary system. The jet impact force can be conservatively calculated on the basis of perpendicular impingement of a liquid jet. For the Humboldt design rupture:

$$
\begin{aligned}
F &= 2CA\,(\Delta P) \\
&= 2 \times 0{\cdot}61 \times 0{\cdot}705 \times 144\,(1\,250-0) \qquad (2) \\
&= 155\,000 \text{ lb}
\end{aligned}
$$

The biological shield may be used to advantage in backing up the dry well wall to withstand this force.

Other dynamic effects which should be considered are possible shock waves and missiles, but for Humboldt these were determined to be of secondary importance.

Dry Well Pressure

The initial dry well pressure build-up to clear water from the vents can be minimized by allowing only a small amount of water inside the vents. Thus, the dry well pressure during the vent clearing condition need not be the maximum design pressure and will not be discussed further. Shortly after the rupture, the dry well pressure levels off at a maximum. Thereafter, dry well pressure drops as the flow rate from the rupture diminishes.

The maximum dry well pressure under flow conditions is the sum of the pressure at the discharge end of the vent pipe (which may or may not be critical pressure) and the vent pipe pressure drop. For a given maximum rupture size, dry well pressure is a function only of vent area and vent flow resistance if the end conditions of Fig. 11 are not limiting. Total energy to be released and the volume of the dry well does not affect dry well pressure directly.

Vent pipe pressure drop depends on the amount of water carried into the vents with the steam. To be conservative, a design should assume that all the water in the mixture from the rupture is carried with the steam into the vent pipe. Conditions approaching 100% carry-over were observed on some of the tests.

The following equation is the basis of this calculation:

$$dP = \left[\frac{fu^2dL}{D2g} - \frac{u\,du}{g} - dH\right]\frac{\rho}{144} \qquad (3)$$

where dP = incremental pressure drop, psi
f = friction factor
dL = incremental equivalent length of pipe, ft
D = inside diameter of pipe, ft
u = mixture velocity in the pipe, ft/s
g = acceleration due to gravity, 32·2 ft/s²
ρ = mixture density, lb/ft³
dH = incremental increase in elevation, ft

The first term on the right side of equation (3) represents the familiar Darcy formula for friction; the second term on the right is the drop due to acceleration ($F = ma$); and the right-hand term is the drop in elevation head.

In order to solve equation (3), the density of the mixture for given pressures must be found. The assumption that water and steam flow as a homogeneous mixture (equal velocities) gives good correlation with test results. The density can be calculated from steam table data for homogeneous flow making use of the energy equation:

$$h = h_o - \frac{u^2}{2gJ} \qquad (4)$$

where h = enthalpy of the mixture in the pipe at pressure P, Btu/lb
h_o = stagnation enthalpy of the flowing mixture, or enthalpy of water leaving the primary system
J = 778 ft lb/Btu

Where the flow encounters a reduction in area such as at a pipe junction or entrance, an additional pressure drop occurs and it is necessary to evaluate equation (3) without the friction term to determine the velocity head change.

A critical end-of-pipe condition could exist if critical pressure is greater than the back pressure on the piping (pool pressure). The pressure at which a critical condition could exist is where $dP = \rho u du/144g$ or where entropy becomes constant, either of which can be evaluated by trial and error for a given flow.

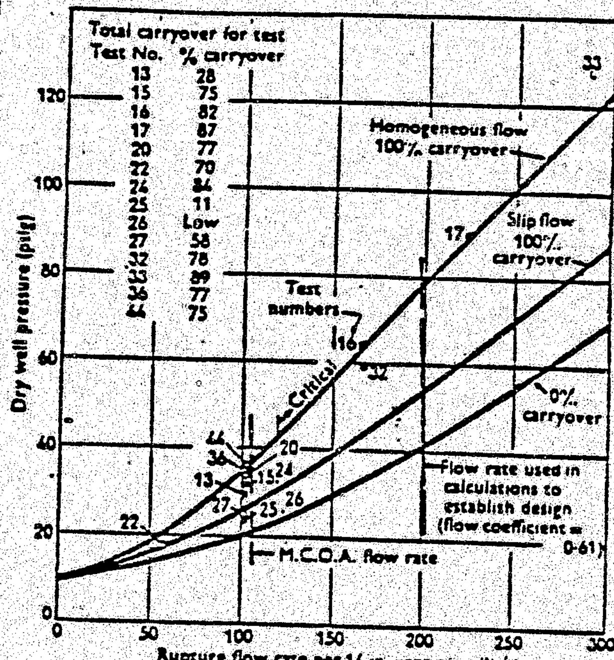


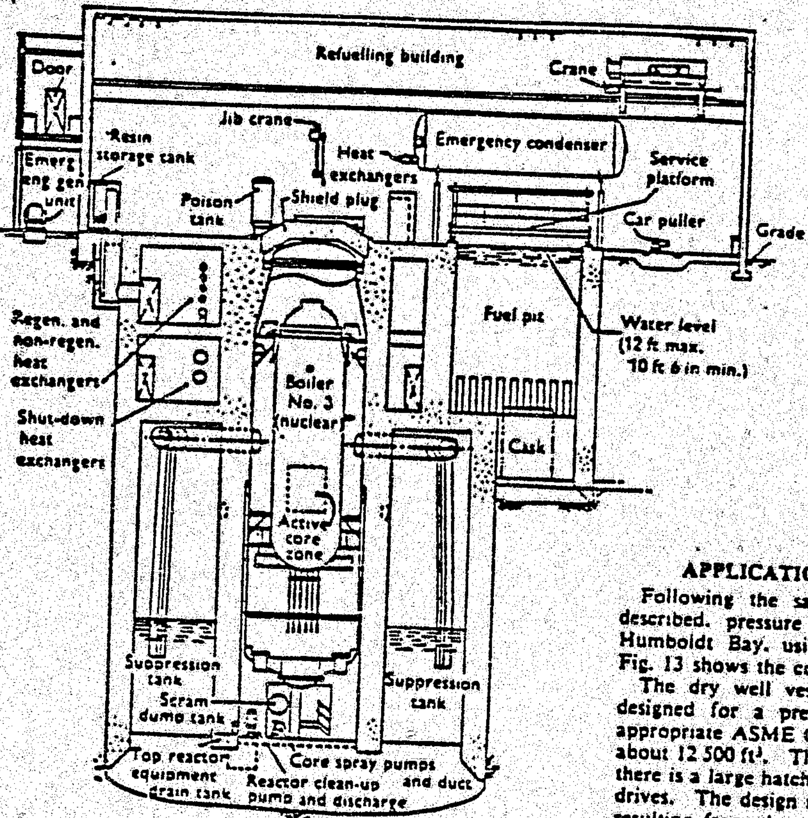
Fig. 12.—Comparison of calculated dry well pressure with test results.

Fig. 12 shows calculated dry well pressure as a function of rupture flow rate. The upper line is for the method just described. The middle line shows results of similar calculations assuming slip flow instead of homogeneous flow. Slip flow is a method which assumes that steam and water velocities are not equal. The lower line shows calculated results assuming all the water from the rupture flow falls out in the dry well and steam only flows in the vents. The dashed line on the right represents the calculated maximum credible rupture flow, and the other dashed line represents maximum credible rupture flow based on tests. The results plotted on this figure indicate that the homogeneous method is a good and fairly accurate method for predicting steady flow dry well pressure. Maximum calculated dry well pressure during the vent flow period is the pressure at the intersection of the solid and dashed line. For Humboldt,

TABLE 3
HUMBOLDT SUPPRESSION CHAMBER DESIGN

	psia	°F	ft³	ft³/lb	lb	Btu/lb	1 000 Btu
Before							
Reactor vessel	—	573·94	—	—	400,000	—	—
Reactor water	1,265.00	573.94	1,790	0·0226	79,375	575.42	45,674
Reactor steam	1,265.00	573.94	860	0·3400	2,529	1,100.70	2,784
Dry well air	14.70	150.00	12,500	15·3805	813	20.35	17
Chamber air	14.70	100.00	33,400	14.1192	2,366	11.73	28
Pool water	—	80.00	20,900	0.0161	1,300,479	48.02	62,448
Dry well totals	—	—	15,150	—	482,717	—	48,475
Chamber totals	—	—	54,300	—	1,302,844	—	62,475
System totals	—	—	69,450	—	1,785,561	—	110,950
Change							
Decay heat	—	—	—	—	—	—	4,000
Feedwater	—	182.53	—	—	27,000	150.45	4,062
Dry well totals	—	—	15,150	—	509,717	—	56,537
Chamber totals	—	—	54,300	—	1,302,844	—	62,475
System totals	—	—	69,450	—	1,812,561	—	119,012
After							
Reactor vessel	—	237.75	—	—	400,000	−36.98	−14,793
Dry well steam	23.97	237.75	15,150	16.9429	894	1,084.64	970
Chamber air	21·96	126.25	31,449	9.8950	3,178	16.26	52
Chamber vapour	2.01	126.25	31,449	172.8279	182	1,052.04	191
Pool water	—	126.25	22,851	0.0162	1,408,307	94.15	132,592
Dry well totals	23.97	—	15,150	—	400,894	—	−13,823
Chamber totals	23.97	—	54,300	—	1,411,667	—	132,835
System totals	—	—	69,450	—	1,812,561	—	119,012

Dry well pressure = 9.27 psig
Chamber pressure = 9.27 psig.



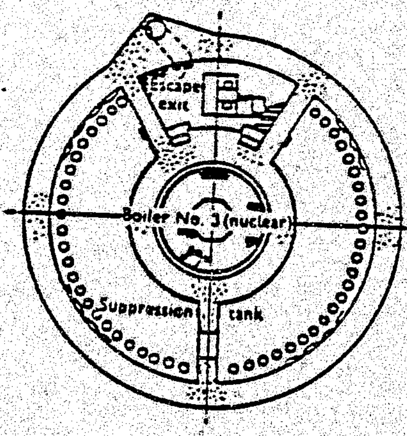


Fig. 13.—Layout of Humboldt containment.

this is 72 psig for the design rupture flow rate. For the actual rupture flow rate, maximum calculated dry well pressure would be 36 psig, just half the design pressure.

Suppression Chamber and Pool

The design pressure of the chamber can be determined by a weight and energy balance and the assumption of complete condensation. The maximum pressure occurs at the end of the venting period when all the dry well air has been transferred and temperatures in the chamber are a maximum. To be conservative, it should be assumed that all of the air initially in the dry well is transferred to the suppression chamber during the venting period.

The computed weight and energy balance shown in Table 3 for the Humboldt design condition indicates the relative importance of various parameters. Most of the accident energy comes from the reactor water and is absorbed in the pool water, and the final temperatures are related to the initial amount of water in the pool. The pressure is principally determined by the quantity of air transferred from the dry well to the chamber. The design pressure for Humboldt is 10 psig.

After the initial venting of steam, dry well pressure would immediately drop to balance chamber pressure. The pressure in the pressure suppression system as a function of time is found by additional weight and energy balances accounting for decay heat and post-incident cooling. The Humboldt post-incident cooling system would take water from the chamber, cool it, return it to the dry well, and inject it into the reactor vessel over the core to avoid core melting. This cooling in addition to continuation of feedwater flow would cause the dry well pressure to tend to go negative relative to chamber pressure. Vacuum breakers protect the dry well from excessive external pressure by returning air from the chamber to the dry well. Subsequent dry well flooding would displace excess air with any contained fission products to the chamber. In about 2 h enough water would be added to the dry well to flood up to the bottom of the vent pipe entrances. Water would then spill back into the chamber completing the water circuit. The discharge ends of the vent pipes would still be under water.

APPLICATION OF PRESSURE SUPPRESSION

Following the satisfactory results of the test programme described, pressure suppression containment was applied to Humboldt Bay, using the design consideration given above. Fig. 13 shows the containment arrangement used.

The dry well vessel is about 68 ft long and 17 ft 6 in d., designed for a pressure of 72 psig in accordance with the appropriate ASME Codes. The net volume of contained air is about 12 500 ft^3. The top head is removable for refuelling, and there is a large hatch in the bottom for access to the control rod drives. The design is adequate to resist the calculated pressure resulting from the unlikely event of the rupture of the largest pipe connected to the reactor vessel. As mentioned before, the actual pressure to be expected is about half this value. The dry well is encased in a concrete structure support and biological shield which also provides the back-up needed to resist jet forces and missiles. The dry well heads are lined with concrete for this purpose.

The vent pipe system connecting the dry well to the suppression pool consists of six 40 in pipes radiating out from the dry well connecting to a 40 in header going around the top of the suppression chamber. Connected to this header are 48 14 in pipes extending down to 6 ft below the pool surface. The vent pipe system has a design pressure equal to that of the dry well.

The suppression chamber surrounds the dry well in a partial annulus. It contains 20 900 ft^3 of water and 33 400 ft^3 of air. Its design pressure is 10 psig in the air space and 10 psig plus hydrostatic head in the lower portion. The strength to resist this pressure is provided by the concrete. A thin steel liner provides a water-tight seal. A concrete refuelling building provides containment for accidents during fuel handling or reactor fuel loading. Additionally it collects any possible leakage from the pressure suppression system in the event of a reactor system rupture accident.

Hold up and decay in the refuelling building would reduce by a significant factor the release of active material to the atmosphere during the first few hours following such an accident. Leakage into the refuelling building would be discharged through a scrubber which is designed to remove at least 95% of the halogens and solids. Then the leakage would be discharged to the atmosphere through a 250 ft stack. The combined effect of leakage hold-up, scrubbing, and discharge at high elevation would be to reduce maximum off-site ground concentrations during the first 2 or 3 h by something of the order of 10^{-3} or 10^{-4} for noble gases and 10^{-4} or 10^{-5} for halogens and solids compared to direct leakage from dry well to atmosphere at ground level.

The result of all these containment features is that the probability of anyone receiving a dose approaching the emergency limit of 25 rem after a gross core melt-down in the dry well of a Humboldt type plant is infinitesimal.

The physical arrangement of pressure suppression at

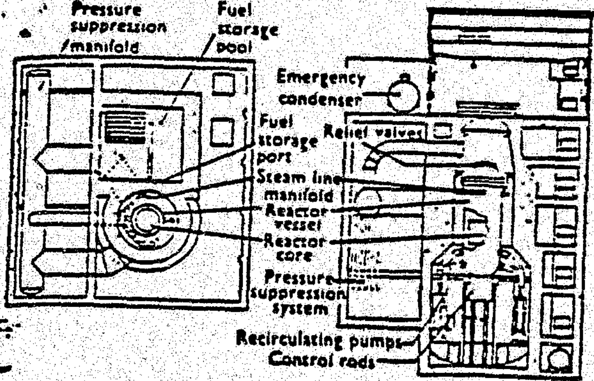
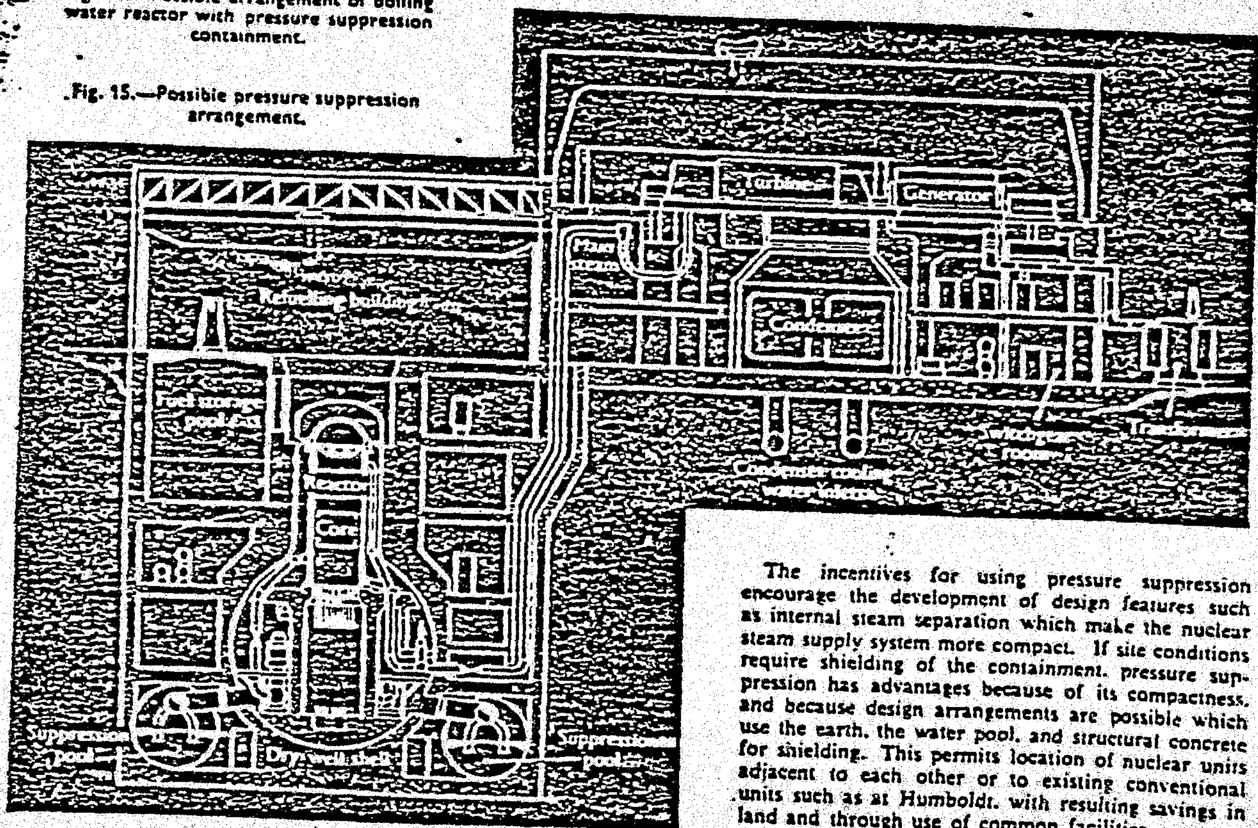


Fig. 14.—Possible arrangement of boiling water reactor with pressure suppression containment.

Fig. 15.—Possible pressure suppression arrangement.



Humboldt Bay was to a large extent dictated by local conditions. Because of the proximity of the nuclear unit to an existing power plant and because of soil characteristics, it was decided to sink the reactor and containment structure as a caisson. Other arrangements might be selected at other sites.

Figs. 14 and 15 show two arrangements that have been considered by the authors' companies. In both cases the dry well is relatively large as the reactor is a forced circulation design with recirculating pumps within the dry well. The arrangements shown here do not explore all the possibilities open in adapting pressure suppression to various reactor configurations and nuclear power plant sites.

Pressure suppression offers economical containment for water moderated reactors of compact design. At Humboldt Bay, this type of containment has resulted in a saving, exclusive of development costs, of about $650 000 when compared with the originally contemplated large pressure vessel type of containment—a saving in the vicinity of $10/kW installed capacity. Substantial benefits may also be expected for plants much larger than Humboldt Bay, as illustrated by Pacific Gas and Electric Company's decision to use pressure suppression containment for the 325 MW Bodega Bay nuclear plant.

Conclusions

In general, as plant size increases, total dollar savings tend to increase but the savings in dollars per kilowatt decrease. The cost difference between pressure suppression and dry containment will depend on details of specific designs and on site requirements. Pressure suppression is most advantageous when used with nuclear steam supply systems which can be readily fitted into a compact dry well. A natural circulation direct cycle boiling water reactor such as Humboldt with no recirculating pumps and steam separation inside the reactor vessel is particularly adapted to this method of containment. External pumps, heat exchangers, or steam drums require a larger dry well which would tend to increase the cost of containment.

The incentives for using pressure suppression encourage the development of design features such as internal steam separation which make the nuclear steam supply system more compact. If site conditions require shielding of the containment, pressure suppression has advantages because of its compactness, and because design arrangements are possible which use the earth, the water pool, and structural concrete for shielding. This permits location of nuclear units adjacent to each other or to existing conventional units such as at Humboldt, with resulting savings in land and through use of common facilities.

The methods of analysis described in this article were used to predict the maximum pressures in a pressure suppression containment design should a break in the primary system occur. Tests have been described which showed that the analysis for Humboldt is quite conservative and that condensation in the water pool would certainly be rapid and complete.

Pressure suppression offers the reactor power plant builder a method of containment which may have distinct advantages in cost and safety for certain reactor designs and is an important contribution toward making nuclear power plants independent of siting restrictions. It is a major step toward economic nuclear power.

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1. WHELCHEL, C. C., ROBBINS, C. H., "Pressure Suppression Containment for Nuclear Reactors," ASME Paper 59-A-215.
2. ASHWORTH, C. P., BARTON, D. B., JANSSEN, E., ROBBINS, C. H., "Predicting Maximum Pressures in Pressure Suppression Containment," ASME Paper 61-WA-222.
3. Final Hazards Summary Report, Humboldt Bay Power Plant Unit No. 3, Pacific Gas and Electric Co., September 1, 1961.

Q. IR-OCA-28-18 Is Dr. Levy familiar with GE tests at the Condensing Test Facility (CTF) at Moss Landing and Transient Test Facility at San Jose, California in 1958-1959? If so, please state the basis of his involvement or familiarity.

A. IR-OCA-28-18 Yes, Dr. Levy is familiar with GE tests at the Condensing Test Facility and the Transient Test Facility. His familiarity comes from recollections of the test and from the two published papers dealing with such tests and their discussion in available AEC/NRC submittals. The publications are ASME Paper 59-A-215 and ASME Paper 61-WA-222. A copy of 61-WA-222 is attached to this interrogatory; a copy of 59-A-215 is attached to IR-OCA-28-11.

Responsible Witness: Salomon Levy
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AN ASME PUBLICATION

Predicting Maximum Pressures in Pressure Suppression Reactor Containment

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This paper describes the analytical and experimental bases for pressure-suppression reactor containment which is used in the Pacific Gas and Electric Company Humboldt Bay plant now under construction. Analytical means of predicting maximum pressures in the containment system are described with their application to the Humboldt design. Experiments supporting the analysis and the design are discussed including transient tests of a full-scale 1/48 segment of the Humboldt system. The tests demonstrated that the methods of analysis presented can be used to predict maximum pressures in the pressure suppression containment for the Humboldt Bay plant should a rupture accident occur.

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Predicting Maximum Pressures in Pressure Suppression Reactor Containment

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NOMENCLATURE

- A = area, sq ft
- C = discharge coefficient
- D = diameter, ft
- E = internal energy, Btu
- e = specific internal energy, Btu/lb
- F = force, lb
- f = friction factor
- g = acceleration of gravity, ft/sec^2
- h = specific enthalpy, Btu/lb
- J = mech. equiv. of heat, ft-lb/Btu
- K = loss coefficient
- L = vent submergence (initial), ft
- P = pressure
- T = temperature, deg F
- t = time, sec
- u = velocity, fps
- V = volume, cu ft
- v = specific volume, ft^3/lb
- W = mass, lb
- w = mass rate, lb/sec
- X = quality
- x = displacement of water in vent, ft
- y = position along axis of vent, ft
- z = elevation (positive in upward direction), ft
- γ = ratio specific heats
- ρ = density, pcf

Subscripts

- A = air
- B = break
- C = suppression chamber
- F = liquid phase, saturated
- G = vapor phase, saturated
- H = steam-water
- P = pool
- V = pressure vessel
- W = water vapor

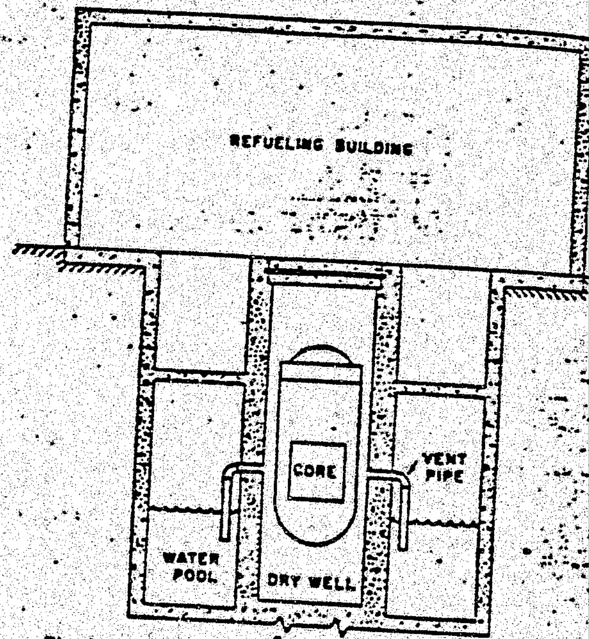

Fig. 1 Schematic diagram of pressure-suppression containment

- o = initial value
- 1 = dry well
- i = interface between air and water in vent
- 3' = conditions just inside vent discharge
- 3 = condition just outside vent discharge

Note: Dot notation is used to indicate derivative with respect to time.

1 INTRODUCTION

In November 1960, a construction permit was issued by the AEC for Humboldt Bay Power Plant Unit No. 3, the design of which provides for a pressure-suppression system of reactor containment. This paper presents the experimental and analytical bases for designing the Humboldt Bay pressure-suppression containment.

An initial development program was conducted for the Pacific Gas and Electric Company by the General Electric Company, Atomic Power Equipment Department, to study the feasibility of pressure-suppression containment, to conduct tests for the purpose of obtaining design information, and to perform analytical work to obtain the necessary design parameters and to demonstrate the adequacy of the design. This initial testing and analytical work was performed in a three-part program which is described in an earlier ASME paper (1).[1]

[1] Underlined numbers in parentheses designate References at the end of the paper.

Section 4 shows some results of those early tests. Following the initial development program the results were used by the Bechtel Corporation in the design of Humboldt containment.

In the Spring of 1960 the Pacific Gas & Electric Company designed, constructed, and operated a new test facility which represented a full-scale 1/48th segment of the Humboldt Bay pressure-suppression system. A report on these tests is included in Section 5.

Methods for predicting maximum pressures in pressure-suppression reactor containment are described in Section 3. Section 5 presents the application of the experimental and analytical work to Humboldt Bay Unit No. 3.

2 DESCRIPTION AND ADVANTAGES

The type of pressure-suppression containment considered in this paper is shown schematically in Fig.1. The boiling-water reactor and primary coolant system up to double isolation valves is completely contained in the dry well (including the guard piping for the penetrations which is an extension of the dry well). The dry well is a pressure vessel designed for the maximum credible internal pressure. The dry well is vented to a pool of water by large flow-area vent piping. The water pool is located in the vaportight suppression chamber which contains a large air volume. Fig.1 shows a refueling building which may be used in conjunction with the basic pressure-suppression system to collect system leakage and to provide containment during refueling.

In the unlikely event of a primary-system rupture within the isolation valves, reactor water and steam would discharge into the dry well. This could be accompanied by jet forces, a shock wave, and possibly missiles. The resulting dry-well pressure would push the pool water out of the vents and then a mixture of air, steam, and water could flow from the dry well through the vents into the pool. The steam from the rupture would be condensed rapidly and completely in the suppression pool, resulting in relatively low dry well and chamber pressures.

Pressure suppression appears most suitable for plants with compact nuclear steam-supply systems. For such reactors pressure-suppression containment may offer a number of advantages over conventional dry-type (steel spheres or cylinders) containment. In addition to simplification of reactor-building layout and lower containment cost, it is believed that the following advantages related to safety apply:

1 In the event of a maximum credible rupture, fission products would largely remain in the small volume of the dry well. Halogen or solid fission products that might be vented from the dry well would be retained in the water pool, thus reducing the possibility of leakage to the atmosphere.

2 A major portion of the accident energy would be absorbed rapidly in the water pool.

3 The small physical size of the dry well and the concrete backup by the biological shield result in an enclosure with fewer welds, lower probability of imperfections and less chance of the fission products escaping if they should leak from the steel vessel into the concrete enveloping it.

4 The small physical size of the pressure-suppression system may make it economically feasible to locate the reactor operating containment below ground.

5 System leakage could be collected in a refueling building. The leakage would be vented through clean-up equipment and discharged out a high stack.

As a result of the advantages noted, post-accident radiation levels outside the containment system would be lower than with conventional dry containment.

3 ANALYSIS

The system and events can be considered in three parts, each with its appropriate model and analysis. These parts are identified as follows: (A) Dynamic effects; (B) venting period; (c) termination of venting period. The significant dynamic effects would be the shock wave preceding the interface between steam-water and air, and the reaction and impact of the steam-water jet from the break, the latter continuing during the venting period. Although not a part of predicting pressures the dynamic effects of possible missiles should be considered.

The venting period would begin with a pressure transient followed by a quasi-steady flow condition. It would be during the venting period that the dry well would experience its highest static pressure.

At the termination of the venting period the temperature of the pool water would be at a maximum, and, except for the effect of heat transfer to the surroundings, the suppression-chamber pressure also would be maximum.

Dynamic Effects

Consider first the shock wave. An analytical model is used for which the flow is one-dimensional, and the pressure behind a shock reflected from a normal surface is evaluated. This model is conservative because it postulates a plane shock wave. The shock would actually tend to advance on a spherical front, the pressure being thus attenuated with distance.

The escaping mixture of steam and water driving the air ahead of it is assumed to expand reversibly and isentropically to the pressure at the mixture-air interface. This pressure is equated to the pressure behind the advancing shock. The applicable equations of continuity, energy, state, and momentum, and the Rankine-Hugoniot equation (2), are employed to evaluate the pressure behind both the advancing shock and behind the reflected shock.

Consider next the jet of steam-water issuing from the postulated break, and impinging on the dry-well wall. The model for this analysis has the axis of the jet normal to the wall, the axial component of velocity of the impinging fluid being brought just to zero at the wall. The force with which the jet acts is given by (3)

$$F = 2CA_B\ (P_V - P_1) \qquad (1)$$

where C is the discharge coefficient for flow through the break, A_B is the area of the break, P_V is the reactor-vessel pressure, and P_1 is the dry-well pressure.

The force is relatively concentrated. However, the highest pressure experienced by the wall due to the impinging jet cannot exceed the dynamic pressure of the jet, which is equal to or less than $P_V - P_1$. As a general rule, the pressure exerted by the jet will greatly exceed that of the shock wave.

Venting Period

Rupture Flow Rate. The reactor vessel blows down via the break throughout the entire venting period. Complete separation of phases within the vessel is assumed at all times. It is also assumed that only liquid phase flows through the break, and that no flashing occurs prior to passage through the break. Test results indicate that either or both of these latter two assumptions may be in error, but they are defensible in that they are conservative; i.e., yield larger mass rates than would actually occur. On the basis of these assumptions, the mass rate through the break is given by

$$\dot{W} = CA_B \left[\frac{P_V - P_1}{\nu_{fV}}\right] \qquad (2)$$

It may conservatively be assumed to facilitate calculations that the reactor-vessel pressure remains at its initial value during the venting period. However, the pressure in the reactor vessel as a function of time may be determined simply by keeping an inventory of the mass and energy of its two-phase fluid contents. Thus, the addition or removal of any mass via any of the pipelines still intact, or the addition of energy due to the heat capacity of the reactor-vessel walls or to delay neutrons and decay heating, can be taken into account. The pressure, assuming thermodynamic equilibrium between liquid and vapor phases, is a function of specific volume and specific internal energy.

Eventually all of the liquid phase will have left the vessel. Some modified form of equation (2) can be used if it is important to know the mass rate of the vapor remaining.

In terms of events taking place in the dry well, vent, and pool, the venting period may logically be divided into two parts, an initial transient and a quasi-steady state. The initial transient is very brief, and ends when the last of the pool water initially in the vent pipes is blown clear. The quasi-steady state exists during the time that first air and then steam and water are vented into the pool. Depending upon the relative physical proportions of the pressure-suppression system, the maximum pressure which the dry well will experience occurs either at the end of the vent clearing transient, or during the quasi-steady discharge of steam and water into the pool. Each part is analyzed separately to determine the maximum pressure for that part. Whichever pressure is the higher determines the dry well design pressure.

Vent Clearing Transient. The analytical model employed for this part has the following features:

1 The steam-water mixture in the dry well is in thermodynamic equilibrium at every instant, the state being characterized by quality X_H and dry-well pressure P_1.

2 There is no mixing of the steam-water with the dry-well air.

3 The air is compressed reversibly and adiabatically. All the air is essentially at zero velocity in relating dry-well pressure and air volume.

4 Air friction in the vent is negligible.

5 Pressure in the pool at the elevation of the vent discharge is everywhere uniform, and depends only on the depth and acceleration of the pool water over it, plus pressure at the pool surface.

The following relationships apply to the steam-water mixture in the dry well. From the first law of thermodynamics (4),

$$E_H = \dot{W}_H h_{fV} - \frac{P_1 \dot{V}_H}{J} \qquad (3)$$

The volume of a two-phase mixture is

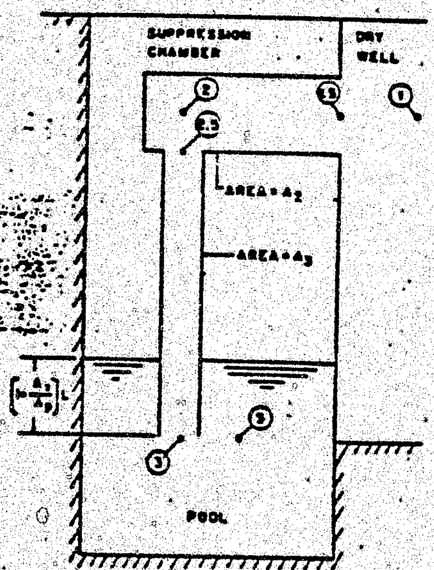

Fig. 2 Schematic diagram of a vent

$$\dot{V}_H = W_H \left[x_H v_{fg1} - v_{f1} \right] \qquad (4)$$

The steam quality is given by

$$x_H = \frac{e_H - e_{f1}}{e_{fg1}} = \frac{E_H}{W_H e_{fg1}} - \frac{e_{f1}}{e_{fg1}} \qquad (5)$$

Another group of relationships apply to the air in the dry well and vent. A constant mass of air undergoing reversible and adiabatic changes of state is governed by

$$P_1 = P_{10} \left[\frac{V_{Ao}}{V_A} \right]^{\gamma} \qquad (6)$$

The volume of the dry-well air is increased by the volume of water displaced from the vent, but decreased by the volume occupied by the steam-water.

$$\dot{V}_A = A_3 u_1 - \dot{V}_H \qquad (7)$$

The pressure acting at the interface between the air and the pool water in the vent may be equated to the dry-well pressure. This neglects friction, kinetic energy, and inertia of the air in the vent. These should be small at the velocities which exist in the vent during the vent clearing transient.

$$P_i = P_1 \qquad (8)$$

The dynamic behavior of the pool water remaining in the vent at any instant is described by

$$\dot{u}_1 = \frac{g}{\rho_p} \frac{P_3' - P_1}{L - x} + g - \frac{f}{D_3} \frac{u_1^2}{2} \qquad (9)$$

It should be noted here that better agreement with test results was obtained by assuming the mass of the water involved in equation (9) to be slightly greater than that contained within the vent. This increase in mass corresponded to an increase in vent length of approximately one diameter.

The displacement of pool water in the vent is given by

$$x = \int_0^t u_1 \, dt \qquad (10)$$

The dynamic behavior of the pool water outside of the vent and above the elevation of the vent discharge is described by

$$P_3' = P_3 = P_C + \rho_p \left[L + \frac{A_3}{A_p} x \right] + \frac{\rho_p}{g} \left[L + \frac{A_3}{A_p} x \right] \frac{A_3}{A_p} \dot{u}_1 \qquad (11)$$

The suppression chamber pressure P_c changes only slightly during the vent clearing transient and may be treated as a constant.

The vent-clearing transient ends when $x = L$.

Quasi-Steady Venting of Air and Steam-Water. The dry-well pressure may be equated to the vent pressure drop plus the pressure at the discharge end of the vent pipes. Reflecting briefly about the relative losses of air and of steam-water (about 30 per cent quality), flowing through the same vent at roughly the same volume rate, leads to the conclusion that dry-well pressure during the venting of steam-water will exceed dry-well pressure during the venting of air. Only the venting of steam-water need be considered for the purposes of this analysis.

The model employed for the analysis of this part has the following features:

1. The process is quasi-steady state.
2. All of the steam-water flowing through the reactor-vessel break also flows through the vent; no water remains in the dry well.
3. The steam-water mixture is homogeneous and in thermodynamic equilibrium at every position along its path. This is conservative, though it is probably a good approximation in view of the high velocities encountered in the vent. The homogeneous assumption is supported by test results.
4. The pressure in the pool at the elevation

of the vent discharge is equal to hydrostatic pressure plus surface pressure.

5 All of the steam is condensed in the pool.

6 The suppression-chamber air temperature is the same as the pool temperature.

The mass rate is given by equation (2), and the state in the dry well is defined by the specific enthalpy $h_1 = h_{fv}$ and the pressure P_1. Another relationship is needed which relates mass rate and pressure drop in the vent. Then, given the suppression-chamber pressure, both the mass rate and the dry-well pressure may be determined.

A schematic diagram of the vent is shown in Fig.2. The flow path may be divided into segments such that for each segment the flow area is constant, except for transitions from one flow area to another. Consider the transition from dry well to vent inlet. Referring to Fig.2 for notation, Euler's equation (5) may be integrated from position 1 (dry well) to position 1.5 (vent inlet) to yield

$$\frac{\dot{W}^2}{A_2^2} = 2\,g\rho_{1.5}^{\,2} \int_{1.5}^{1} \frac{dP}{\rho} \qquad (12)$$

It may be noted that

$$\frac{\dot{W}^2}{A_2^2}\,\frac{1}{2g\rho_{1.5}^2} = \frac{u_{1.5}^2}{2g}$$

which is simply the velocity head at the vent inlet.

Consider next the constant-area segment from position 1.5 to position 2. For constant area

$$d\,(\rho u) = 0$$

This plus loss terms combined with Euler's equation and integrated between the indicated limits yields

$$\frac{\dot{W}^2}{A_2^2} = 2g\,\frac{\int_2^{1.5} \rho\, dP + \int_2^{1.5} \rho^2\, dz}{2\ln\dfrac{\rho_{1.5}}{\rho_2} + \dfrac{fL_2}{D_2} + \sum K_2} \qquad (13)$$

$\sum K_2$ includes the entrance loss and any bend losses between positions 1.5 and 2.

Consider next the short segment from position 2 to position 2.5. If there is a change in the flow area, then a variation of equation (12) applies.

$$\frac{\dot{W}^2}{A_3^2} = \frac{2g\rho_{2.5}^2}{1 - \left(\dfrac{\rho_{2.5}A_3}{\rho_2\,A_2}\right)^2} \int_{2.5}^{2} \frac{dP}{\rho} \qquad (14)$$

If the change in area is small, this calculation may be omitted.

Finally, consider the segment from position 2.5 to position 3'. Equation (13) is again used, except for the appropriate change of subscripts and integration limits.

The calculation procedure involves first the determination of whether critical flow exists at vent discharge. Critical flow is given by

$$\frac{\dot{W}_{crit}^{\,2}}{A_3^2} = g\rho^2 \left(\frac{\partial P}{\partial \rho}\right) = -g\left(\frac{\partial P}{\partial v}\right) \qquad (15)$$

The derivative should be evaluated at constant entropy but evaluation at constant enthalpy results in only a small error on the conservative side. Further, it permits simpler calculation procedures and ultimately affects the calculated dry well pressure only slightly.

Another simplification introduced at this time to facilitate calculations is to impose the condition that the vent flow process also be isenthalpic (as opposed to adiabatic). This again results in a small conservative error.

Knowing the enthalpy at vent discharge, equation (15) gives the relationship between the pressure at vent discharge $P_3{}'$ and the mass rate for critical flow. Equations (2), (12), (13), and (15) may be solved together for the dry well pressure as well as mass rate and vent discharge pressure $P_3{}'$.

A check is next made on whether critical flow does exist at vent discharge. If the calculated $P_3{}' \geq P_3$ the flow is critical, and the mass rate and dry-well pressure as determined in the preceding paragraph hold. But if the calculated $P_3{}' < P_3$ the flow is subcritical, and then the actual $P_3{}'$ is equal to P_3.

$$P_3{}' = P_3 = P_C + \rho_P\, L\left[1 + \frac{A_3}{A_P}\right] \qquad (16)$$

The calculations are repeated using P_3 as determined from equation (16) rather than $P_3{}'$ as determined from equation (15).

Suppression-Chamber Pressure. The suppression chamber pressure P_C is equated to the sum of the partial pressures of water-vapor and air.

$$P_C = P_{CV} + P_{CA} \qquad (17)$$

P_{CV} is the vapor pressure of water at temperature $T_C = T_P$. P_{CA} is given by

$$P_{CA} = \frac{W_{CA}}{V_C}\, R_A T_C \qquad (18)$$

$W_{CA} = W_{AO} + W_{CAO}$, the total mass of air in the

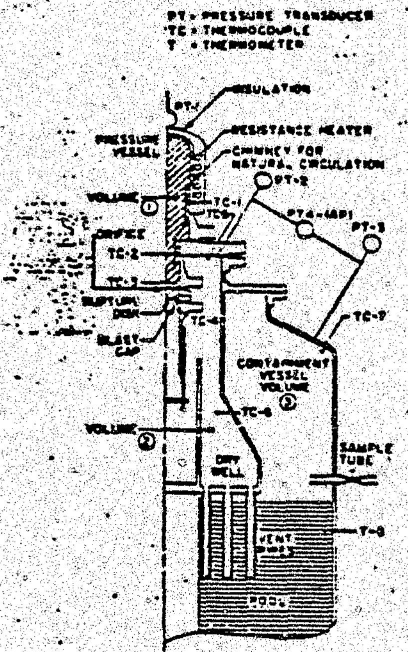

Fig. 3 Arrangement of transient test facility

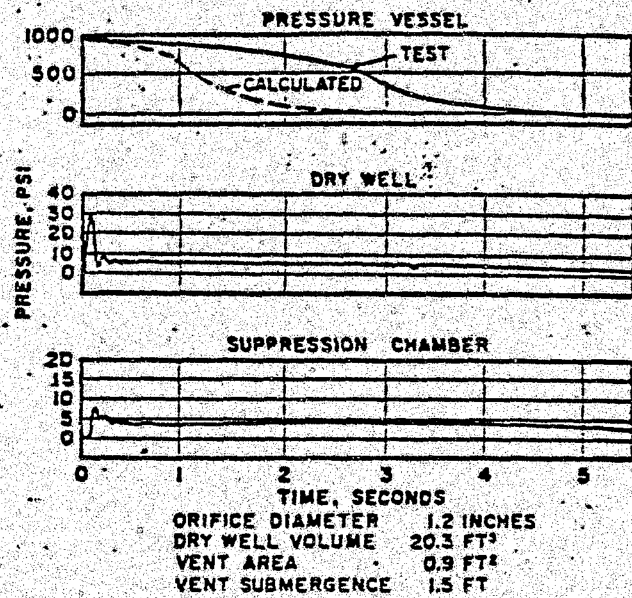

Fig. 4 Pressure traces, initial transient test

pressure-suppression system. The suppression-chamber volume V_c is the initial volume minus the increase in volume of the pool water:

$$V_c = V_{co} - LA_3 - \left[\frac{W_P}{\rho_P} - \frac{W_{Po}}{\rho_{Po}}\right] \quad (19)$$

The temperature of the pool is determined from a mass and energy balance:

$$\dot{E}_P = \dot{W}_H h_{fV} \quad (20)$$

$$\dot{W}_P = \dot{W}_H \quad (21)$$

Integration of (20) and (21) permits ready determination of the average specific internal energy of the pool water. The mean pool temperature T_P is a function of the internal energy.

Termination of Venting Period

At the termination of the venting period the dry well pressure will have decreased from its maximum, but the suppression chamber pressure will have just reached its maximum. The only significant considerations for the termination of venting period relate to suppression chamber pressure.

Equations (17), (18) and (19) may be used for the calculation of the terminal value of the suppression chamber pressure P_c, by simply substituting terminal values for T_P, W_P, and ρ_P.

An alternate scheme for determining the suppression-chamber pressure at termination of the venting period is to:

1 Calculate the total energy in the reactor vessel and pressure-suppression system before the postulated primary-system rupture. This calculation may be simplified by neglecting internal energy in the reactor vessel and dry-well walls; or it may be refined to include these plus delay neutron and decay heating, and energy added via pipelines after the accident.

2 Calculate the total air mass and the total steam and water mass in the reactor vessel and pressure suppression system before the rupture. This calculation may also be refined to include any mass transported via pipelines after the accident.

3 Equate the total energy at termination of the venting period to the total energy before the rupture.

4 Equate the total mass of water and steam at termination of the venting period to total mass of water and steam before the rupture; do likewise for the total mass of air.

5 Determine, based upon the model being used, the energy and mass remaining in the reactor vessel and dry well at termination of the venting period. The remainder must be in the pool and suppression chamber.

6 Impose the condition of thermodynamic equilibrium in the pool and suppression chamber.

This means that all components and phases are at the same temperature and pressure. Then, knowing for the pool and suppression chamber the total volume, total energy, total mass of water and steam (water vapor), and total mass of air, the pressure may be determined.

Events following the termination of the venting period are not considered here but were considered in the design of Humboldt.

4 INITIAL TRANSIENT TESTS

An initial test program was carried out to assist in establishing the method of analysis.

Preliminary considerations showed that a scaled-down model of a reactor and pressure-suppression system should undergo the same pressure transients as the prototype in the event of a large rupture of the primary system, with a time scale which varies directly as the linear dimension, except for some minor gravitational effects. Such a model was designed and built, and is hereinafter referred to as the transient test facility.

The transient test facility has been described in an earlier paper (1). It consisted basically of three interconnected pressure vessels that simulated a reactor vessel, dry well, and suppression chamber. The pool was located in the bottom of, and was integral with, the suppression chamber. The arrangement is shown in Fig.3. The relative position of the reactor vessel differed from the position shown in the schematic arrangement for pressure suppression, Fig. 1, to provide better accessibility for testing. Also, the vent pipes for the transient facility were free of bends and were generally of relatively large area.

Twenty-two tests were run. The procedure for any given test was to heat the water in the reactor vessel until the desired initial pressure was reached, then to break a rupture disk by means of a blast cap and prima-cord. Immediately the contents of the reactor vessel discharged through an orifice (located directly upstream of the rupture disk) into the dry well, and finally passed through the vent pipes into the pool where essentially all the steam condensed.

Instrumentation included four pressure transducers connected to a multichannel recorder, which measured pressure in the reactor vessel, in the dry well, and the suppression chamber. A typical set of the resulting pressure traces appears in Fig.4.

Reactor-vessel, dry-well, and suppression-chamber pressures were calculated by the methods given in the preceding section, for the same set of parameter values as the traces in Fig.4 using a discharge coefficient for the break of 0.65. The calculated reactor vessel pressure is superposed on the top trace of Fig.4 for comparison with experiment. Calculated dry-well and suppression-chamber maximum pressures are given in the following for comparison with the experimental maximums.

	(Pressure, psig) Analysis	Test
Dry well	42	30
Suppression chamber . . .	7.7	8

Referring to the top trace in Fig.4, reactor-vessel pressure, the "knee" of the curve occurs when the last water has left the reactor vessel. Comparing the times at which the knees for calculated and experimental curves occur, it is evident that the average mass rate for the calculated case is about 3 times the average mass rate for the experimental during the period of water expulsion.

The calculated dry-well peak pressure is 40 per cent greater than the experimental value. This difference can be attributed primarily to a higher mass rate than for the experimental.

The calculated suppression-chamber pressure maximum is within 4 per cent of its experimental maximum. The experimental maximum occurred 140 millisec after the beginning of the transient, at the peak of the first oscillations. On the other hand, the calculated value occurs at the termination of the transient. The experimental oscillations are attributed to a spring-mass action involving the lifting of a large fraction of the pool water mass early in the transient, with first compression of the air above, then pressure reduction as the water broke up and fell back. The experimental suppression-chamber pressure for comparison with calculated should be taken later in the transient. The peak value of 4.2 psig at $3^1/2$ sec is chosen. This is 45 per cent below the calculated value. The difference can be attributed to two things:

1. Incomplete transfer of dry well air.
2. Nonadiabatic mixing and compressing.

Both must have played a role. Temperature measurements before and after the transient show no temperature rise in the suppression chamber air. But even isothermal compression would have resulted in a higher pressure than 4.2 psig if all the air had been transferred.

No cases were run with the transient facility for which pressure drop in the vent was significant in determining pressure. Hence the analytical model was not checked for such cases.

TABLE 1

ENERGY BALANCES FOR HUMBOLDT VENT CLEARING TRANSIENT

	Pressure psia	Temp. °F	Weight lbs.	cu.ft/lb	Btu/lb	Volume cu.ft.	Energy Btu
1. Time = 0 seconds							
Dry well air	14.70	150	816.2	15.3	Base	12,500	-
2. Energy added in first .02 seconds (Displacement = .004 ft							
Water from rupture	1265.	574	195	.02256	580.6	-	113,217
Work done on water in vents	-	-	-	-	-	.2	-.6
Total (1 and 2)	-		1011.2			[illegible]	[illegible]
3. Time = .02 seconds (Displacement = .004 ft							
Air	18.14	188.0	816.2	13.2	-	10,760	5,260
Steam		222.8	79.5	21.9	1080.5	1,738	85,896
Water	-	222.8	115.5	.0168	191.0	2	22,060
Totals (3)	18.14		1011.2			12,500	[illegible]
4. Energy added in next .02 seconds (Displacement = .026 ft							
Water from rupture	1265.	574	195	.02256	580.6	-	113,217
Work done on water in vents	-	-	-	-	-	1.3	-5
Totals (3 and 4			1206.2			12,501	226,428
5. Time = .04 seconds (Total displacement = .03 ft.)							
Air	21.43	219.3	816.2	11.7	-	9,540	9,680
Steam		231.6	157.2	18.8	1082.9	2,946	170,211
Water	-	231.6	232.8	.0169	199.9	[illegible]	46,537
Totals (5)	21.43		1206.2			12,501	226,428

The following conclusions may be drawn from this first series of tests:

1 The actual mass flow rate out of the reactor vessel is much less than indicated by the analysis, which assumes complete separation of phases inside the vessel and no flashing prior to clearing the break.

2 Except for differences which affect mass rate, the analytical model gives an adequate quantitative picture of events inside the reactor vessel.

3 The considerable difference by which the calculated exceeded the experimental dry-well pressure peak must be due largely to the lower mass rate for the experimental case.

4 The suppression-chamber pressure and temperature measurements indicate that only about 45 per cent of the dry-well air was transferred for the particular run in question. Inspection of all the pressure traces shows the assumption that all of the air transferred is conservative. (The tests described in Section VI indicate that the oscillations which occurred early in the suppression chamber pressure transient are not significant for the Humboldt design.)

In summary, the transient facility tests showed that the method of analysis yielded values of maximum pressure which were conservative in a system where the vent offered relatively little resistance to flow.

5 APPLICATION OF ANALYSIS TO HUMBOLDT

The application of the analytical methods of Section 3 to the Humboldt Bay pressure-suppression system is described here.

The starting point in applying the analysis is to select a maximum credible rupture size. For Humboldt the postulated maximum credible accident assumes a near instantaneous rupture of the 12-in. schedule 80 main steam line at a reactor overpressure condition of 1250 psig.

Dynamic Effects. The dynamic force of the jet on the dry-well wall (which would be equal to the reaction on the reactor system) is calculated from equation (1) for a 12-in. schedule 80 pipe break assuming a discharge coefficient of 0.61:

$$F = 2 \times 0.61 \times 0.705 \times 144 \, (1250-0) = 155{,}000 \text{ lb}$$

The dynamic pressure of this jet on the dry-well wall cannot exceed 1250 psig. The calculated shock pressure on the portion of the dry-well wall nearest the rupture is about 238 psig which

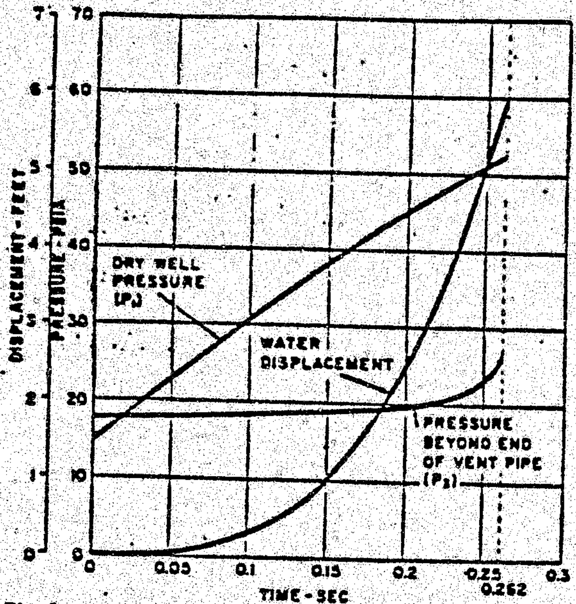

Fig. 5 Calculated vent clearing transient for Humboldt

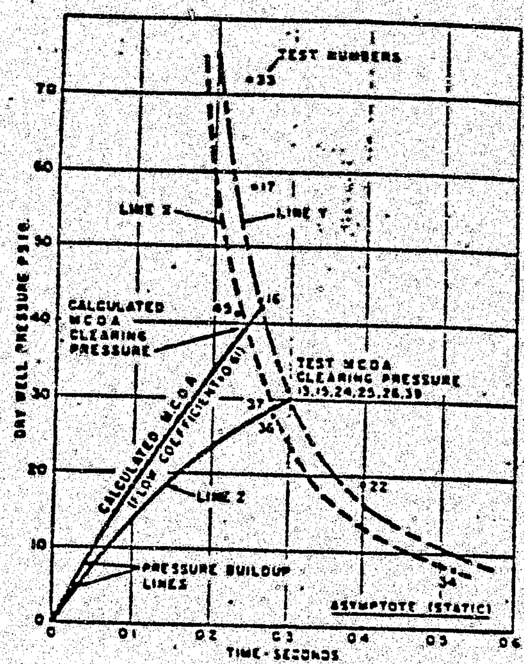

Fig. 6 Comparison of test and calculated vent-clearing transients

is insignificant compared to the jet pressure.

Rupture Flow Rate. From equation (2) the rupture flow rate for water flow from a 12-in. schedule 80 pipe break assuming a dry-well pressure in the range of 0 to 100 psig would be in the range of

$$12 \times 0.61 \times 0.705 \left[2 \times 32.2 \times 44.4 \,(1250\text{-}100 \text{ to } 0)\right]^{1/2}$$

$$= 9350 \text{ to } 9750 \text{ lb/sec}$$

On the basis of the demonstration tests described in the next section actual flow rate for the Humboldt maximum credible accident would be only about 5000 lb/sec. The difference between calculated and actual rupture flow rate introduces a significant margin in the Humboldt dry well design pressure.

Dry-Well Pressure to Clear the Vent Pipes. Equations (3) through (11) can be solved by finite differences using small time steps of say 0.02 sec. Table 1 shows typical balances [results of equations (3) through (8)] for the first few hundredths of a second for Humboldt. The air energy shown is the work done on it by the steam-water less the work done by air on the water in the vents. The work is $144P\Delta V/778$ where P is the average air pressure during the time increment and ΔV the volume change of steam-water, air or water displacement. Notice that equation (3) says that the total change in internal energy of the steam-water mixture in the dry well is equal to the enthalpy added (from the rupture) less the work done on the air which is represented by a steam-water volume change. The displacement in the energy balance corresponds with the trial value of u.

Fig.5 shows the calculated results of the vent clearing transient for Humboldt. The vent clearing transient calculations end when the water displacement equals the initial vent submergence (6 ft). The calculations include the mass of pool water down to 1 ft below the end of the vent pipe. Fig.5 shows a considerable pressure unbalance in the pool at the time the vents clear which would cause a significant upward acceleration of pool water momentarily.

The pressure drop of air in the vents has been neglected in this method. The air pressure drop can be checked by estimating the vent entrance flow required in a given time increment to fill the vent volume to the new pressure. Air flow rate diminishes going down the pipe. If the entrance velocity corresponding to the entrance flow rate represents a velocity head of less than 1 or 2 psi, air pressure drop could be considered negligible.

Fig.6 shows a comparison of calculated vent clearing pressures for various rupture flow rates with results of tests described in Section 6. Line X was calculated neglecting the pool water mass below the end of the vent pipe. Line Y includes the mass of pool water down to 1 ft below the end of the vent pipe. There is generally

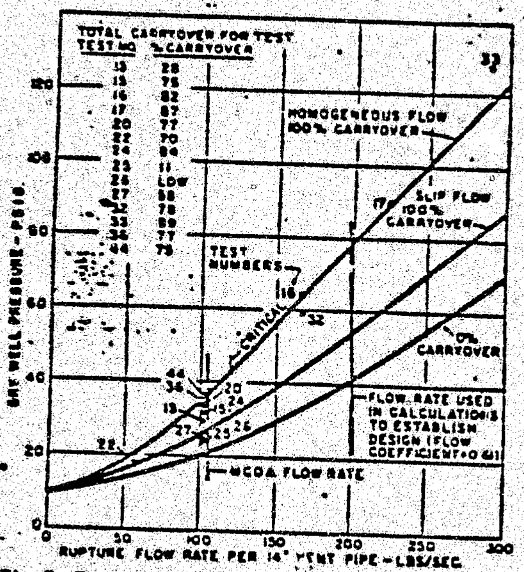

Fig. 7 Comparison of calculated dry-well pressure with test results

good agreement between line Y and test results which indicates that a water mass of about one extra pipe diameter of depth should be included in the calculations.

Line Z shows calculated dry-well pressure build-up as a function of time. The dry-well pressure build-up for the tests is slower which indicates that the actual rupture flow rate is only 50 to 70 per cent of the rupture flow rate used in the calculations.

Dry-Well Pressure During Vent Flow Period. The maximum dry-well pressure under the quasi-steady vent flow condition determines the design pressure for the Humboldt dry well because the transient pressure at the beginning of the venting period is less for the Humboldt design proportions.

Fig.7 is a comparison of calculated dry-well pressure during the vent flow period with results of the tests described in Section 6. On some of the tests with low water carry-over the steady flow dry-well pressure is not the maximum dry well pressure because the steady flow pressure is less than the vent clearing pressure.

The three solid lines in Fig.7 are the calculations for (a) 100 per cent carry-over homogeneous flow of steam-water in the vents; (b) 100 per cent carry-over slip (or Martinelli type) flow; and (c) 0 per cent carry-over or steam flow. Representative tests points plotted on this figure indicate that the homogeneous method is a good and fairly accurate method for predicting steady flow dry-well pressure. Because of the poor showing of the slip method, only the homogeneous method is considered in this paper.

Table 2 gives a summary of the vent-pipe friction equivalent lengths for the Humboldt vent piping.

TABLE 2

	Equivalent length of straight pipe, ft
14-in. straight pipe (vertical)	39
14-in. entrance (or branch flow tee). 0.59 > K > 0.5, say 0.55 . . .	61
Total equivalent length of 14-in. piping	100
40-in. ring header	0
40-in. tee; divergent flow, K = 0.7	229
40-in. straight pipe	14
40-in. entrance and entrance plate, K = 0.841	275
Total equivalent length of 40-in. piping	518

A friction factor of .01 for both pipe sizes has been used.

Table 3 gives flow properties of homogeneous steam-water mixtures at given pressures using equations (6) and (7) and letting $h = h_{f_o} = 580.6$ Btu/lb. The last column is the higher pressure less the lower pressure of two consecutive lines in the table divided by the corresponding difference in specific volumes. This is the difference approximation to the partial derivative of equation (15) taken at constant enthalpy.

Two trial vent flow rates (Table 4) can be assumed which straddle the 9350 to 9750 lb/sec range of rupture flow and a point of balanced flow and dry-well pressure found. For trial vent flows these calculations will use 8616 and 9768 lb/hr instead of rounded numbers because these particular values were used in establishing the design. Areas of 14 and 40-in. pipes are 0.958 and 8.40 sq ft (let $G = (W/A)^2/144g$).

Comparing the values of $(W/A)^2/144g$ for 14-in. pipes with values of $\Delta P/\Delta v$ in Table 3 as indicated by equation (17), it is apparent that critical end-of-line pressure should exist for homogeneous flow at about 35 psia for Trial A and about 40 psia for Trial B. By plotting the values of $\Delta P/\Delta v$ as a function of P, the critical points are found to be more nearly 35.5 and 40.2 psia. Since the back pressure of the pool water

TABLE 3

FLOW PROPERTIES FOR HOMOGENEOUS FLOW
(ENTHALPY = 580.6 Btu/lb)

P psia	Quality X	ρ	$\Delta P/\Delta v$
24	.3926	.1502	
27	.3875	.1698	3.89
30	.3827	.1897	4.85
33	.3783	.2098	5.95
36	.3742	.2300	7.14
39	.3703	.2505	8.45
42	.3666	.2711	9.86
45	.3632	.2920	11.4
48	.3598	.3130	13.0
51	.3567	.3341	14.8
54	.3536	.3555	
57	.3507	.3771	
60	.3479	.3987	
63	.3452	.4207	
66	.3426	.4427	
69	.3400	.4649	
72	.3376	.4873	
75	.3352	.5099	

TABLE 4

	Trial A	Trial B
Total vent flow, lb/sec . . .	.8616	9768
40-in. vent pipe flow, lb/sec	.1436	1628
14-in. vent pipe flow, lb/sec	. 179.5	203.5
For 14-in. pipes:		
$(\dot{W}/A)^2/144g = G$ = ...	7.57	9.73
For 40-in. pipes:		
$(\dot{W}/A)^2/144g = G$ = ...	6.30	8.10

on the vents would be only about 24 psia during the vent flow period (sum of chamber pressure after venting dry-well air and static head of water) critical pressure would occur if vent discharge flow were truly homogeneous. Table 3 data can be used directly in solving equation (13) for equivalent lengths. Since most of the piping is horizontal it is convenient to rearrange equation (13) and write it in terms of equivalent length and finite differences as follows:

$$\Delta L\left(1+\frac{2D}{f}\,\frac{\rho^2\Delta Z/\Delta L}{144\,G}\right)=\frac{2D}{f}\left(\frac{\rho\Delta P}{G}-\ln\frac{\rho_{1.5}}{\rho_2}\right) \quad (24)$$

The ρ without subscript is the average of $\rho_{1.5}$ and ρ_2.

For horizontal pipes $\Delta Z/\Delta L = 0$ and the left side reduces to ΔL.

Table 5 shows the numerical integration of equation (24) for Trial A. A similar calculation made for Trial B gives a dry-well pressure of

TABLE 5

CALCULATED RESULTS FOR HOMOGENEOUS FLOW

Trial A -- Solutions of equation (24) for ΔL:
Starting with the critical end pressure and taking various ΔP's the ΔL's are found and then summed for each length of pipe.

Piece	ΔP, psi	$\rho\Delta P/G$	$\ln \rho_1/\rho_2$	$\frac{2D}{f}\frac{\rho^2}{144G}$	ΔL, ft.
$P_1 = 35.5$ psia (critical end pressure)					
(1)	36 to 35.5	.0151	.0131	.0105	0.4
	39 to 36	.0953	.0694	.0117	2.2
	42 to 39	.1034	.0790	.0138	5.5
	45 to 42	.1115	.0743	.0161	8.4
	48 to 45	.1200	.0695	.0185	11.4
	51 to 48	.1280	.0652	.0212	14.2(x.780)
$P_{2.5} = 50.4$ psia ($\sum\Delta L = 39$ ft.)					
(2)	51 to 48	.1280	.0652	-	13.9(x.220)
	54 to 51	.1367	.0621	-	16.5
	57 to 54	.1451	.0590	-	19.0
	60 to 57	.1534	.0556	-	21.5
	63 to 60	.1620	.0537	-	23.9(x.025)
$P_2 = 60.2$ psia ($\sum\Delta L = 60.7$ ft.)					
(3)	63 to 60	.1950	.0537	-	92.5(x.975)
	66 to 63	.2053	.0510	-	101.0
	69 to 66	.2160	.0489	-	109.3(x.346)
$P_{1.5} = 67.0$ psia ($\sum\Delta L = 229$ ft.)					
(4)	69 to 66	.2160	.0489	-	109.3(x.654)
	72 to 69	.2270	.0475	-	117.5
	75 to 72	.2379	.0453	-	125.6(x.794)
$P_1 = 74.4$ psia ($\sum\Delta L = 209$ ft.)					

The mixture velocity at the entrance plate (flow area = 8.8 sq.ft.) is:

$u = \dot{W}/\rho A = 1436/.506 \times 8.8 = 323$ ft./sec

The velocity head at this point is:

$\Delta P = \rho u^2/2g\,144 = .506 \times 323^2/64.4 \times 144 = 5.7$ psi

Allowing for the velocity head in the dry well (about 0.8 psi), the dry well pressure is 79.3 psia or 64.6 psig.

TABLE 6

ENERGY BALANCE FOR CHAMBER PRESSURE

	Pressure PSIA	Temp. °F	Volume cu.ft.	Weight lbs	Btu/lb	Energy 1000 Btu
1. Before Rupture						
Reactor water	1265	574	1,730	79,555	575.4	45,776
Reactor steam	1265	574	880	2,527	1100.7	2,781
Dry well air	14.7	150	12,500	814	9	7
Chamber air	14.7	100	33,400	2,370	Base	
Pool water	14.7	80	20,900	1,305,000	48.02	62,666
Totals			69,450	1,390,200		111,230
2. Termination of Venting Period						
Steam, R & DW	(22.50)	234.28	15,150	842	1163.6	912
Chamber air	21.14	111.55	31,866	3,184	2	6
Moisture in air	1.36	111.55	31,866	127	1067.3	135
Pool water	-	111.55	22,414	1,386,113	79.49	110,175
Totals	22.5 psia 7.8 psig		69,450	1,390,266		111,230

75.0 psig. Plotting these results against the rupture flow rate versus dry-well pressure relation gives 72.0 psig as the dry-well pressure at the balanced flow condition. This is the dry well design pressure for Humboldt.

Maximum Suppression-Chamber Pressure. Table 6 is a simplified weight and energy balance for Humboldt which neglects decay heat, feedwater flow, and heat transfer from the reactor vessel and walls. These were considered in establishing the design pressure of 10 psig.

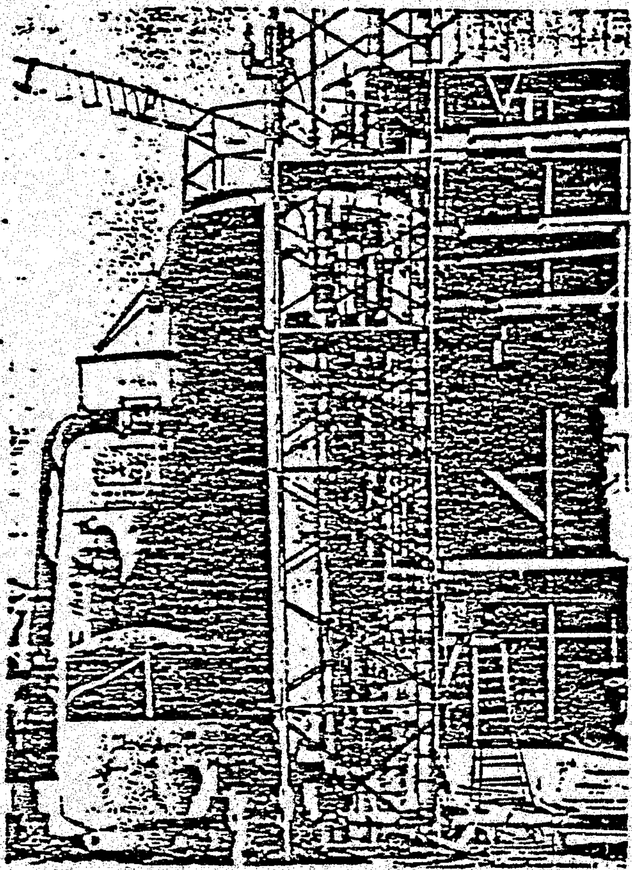

Fig. 8 Humboldt test facility

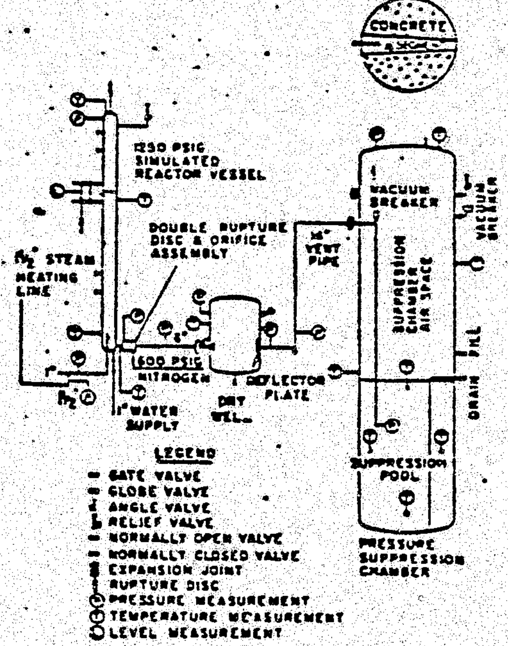

Fig. 9 Schematic diagram of Humboldt test facility

HUMBOLDT FULL-SCALE 1/48 SEGMENT DEMONSTRATION TESTS

A transient test facility was constructed and operated at P.G.&E. Moss Landing power plant in the Spring of 1960 to proof test and demonstrate the Humboldt design. These tests confirmed the Humboldt design with considerable margin and also resulted in obtaining valuable information for future designs.

Description of Test Facility

The facility was made to represent as nearly as possible one of the 48 Humboldt Bay vent pipes, the portion of the suppression chamber and pool associated with it and an appropriately sized dry well and simulated reactor vessel. Fig. 8 shows the test facility, and Fig.9 shows a schematic diagram of the equipment. The segment of the Humboldt suppression chamber which the test chamber represents in full size and shape is shown in heavy outline in Fig.10. The time scale is the same in these tests as it is in Humboldt.

Reactor Vessel. The simulated reactor vessel was a vertical 32-ft length of 20-in-OD Schedule 80 pipe with a volume of 55.8 cu ft.

Rupture Disks and Orifices. The transient tests were initiated by breaking rupture disks, causing flow to pass from the reactor vessel through different sized sharp-edge orifices ranging from 0.14 to 3.28 in. diam. A 1.64-in-diam orifice equals 1/48 of the area taken as the maximum credible rupture for Humboldt.

Dry Well. The test dry-well volume for runs 13 and on was 230 cu ft, including the vent pipe down to the water level, which is approximately 1/48 of the corresponding Humboldt volume. The volume was 308 cu ft for the first 12 tests.

The piping which carried steam and water into the dry well was made oversize to direct rupture flow without introducing significant losses. It was terminated inside the dry well with different fittings for different tests, as shown in Fig.11. Arrangements A to E were used to produce different amounts of water carry-over in the steam leaving the dry well and also to influence the time of air discharge.

Vent Piping. The piping connecting the dry well with the suppression chamber was 14 in. diam and was designed to have the same volume and flow resistance as the Humboldt design.

Suppression Chamber. The trapezoidal shaped suppression chamber was 12 ft by 2.35 ft by 1.23

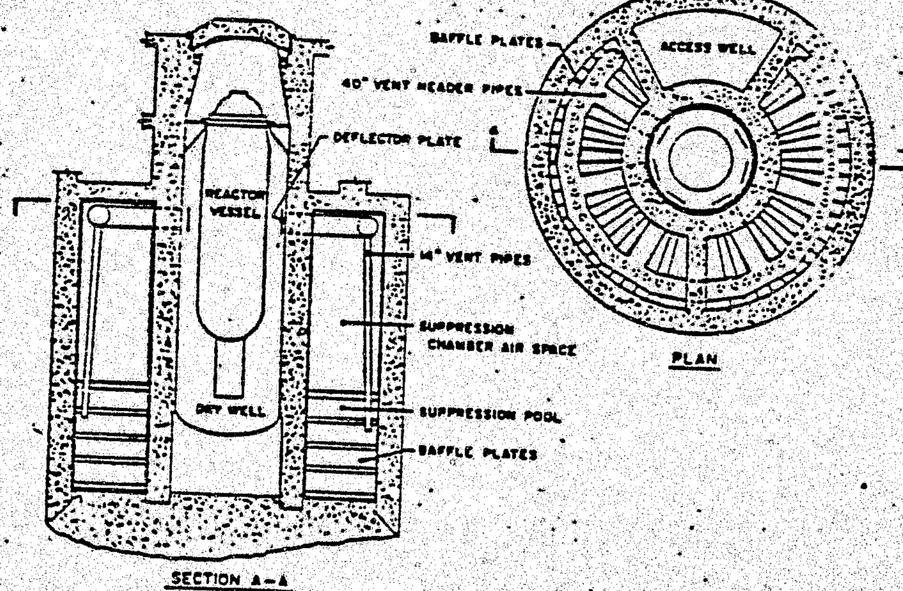

Fig. 10 Humboldt suppression chamber

ft in horizontal cross section. The entire chamber was 49 ft high and normally was filled with water to a depth of 18 ft. The walls were insulated to reduce condensation. The suppression chamber and the 14-in. vent pipe were a full-scale model of one of the Humboldt vents and the portion of the suppression chamber associated with it.

<u>Instrumentation</u>. Provisions were made for transient measurements at the points shown in Fig.9. Transient pressures were measured by strain-gage transducers and were recorded by a light-beam oscillograph.

<u>Test Program</u>

The performance of the system was investigated over a wide range of conditions simulating accidents both more and less severe than the maximum rupture assumed for the Humboldt design. The following parameters were varied.

1 Flow rate was controlled by nine different size orifices ranging from 1/137 to 4 times the area of the 1.64-in. orifice representing the assumed Humboldt maximum rupture.

2 The effect of air in the dry well was investigated by varying the time air entered the vent and by adding air to the dry well during a run.

3 The amount of water left in the dry well and not carried with the steam into the vents was varied between about 10 and 90 per cent of the maximum possible.

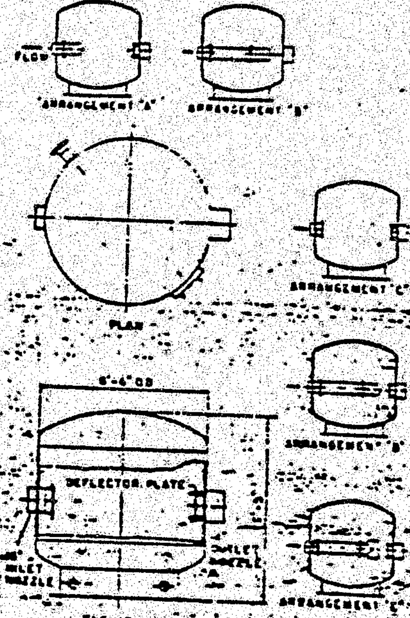

Fig. 11 Test dry well and internals

4 The initial vent depth of submergence was varied by raising or lowering the water level in the pool. The maximum submergence was 12.5 ft and the minimum occurred with the vent 3 ft above the water. The Humboldt design has 6 ft submergence. Varying the depth of submergence also

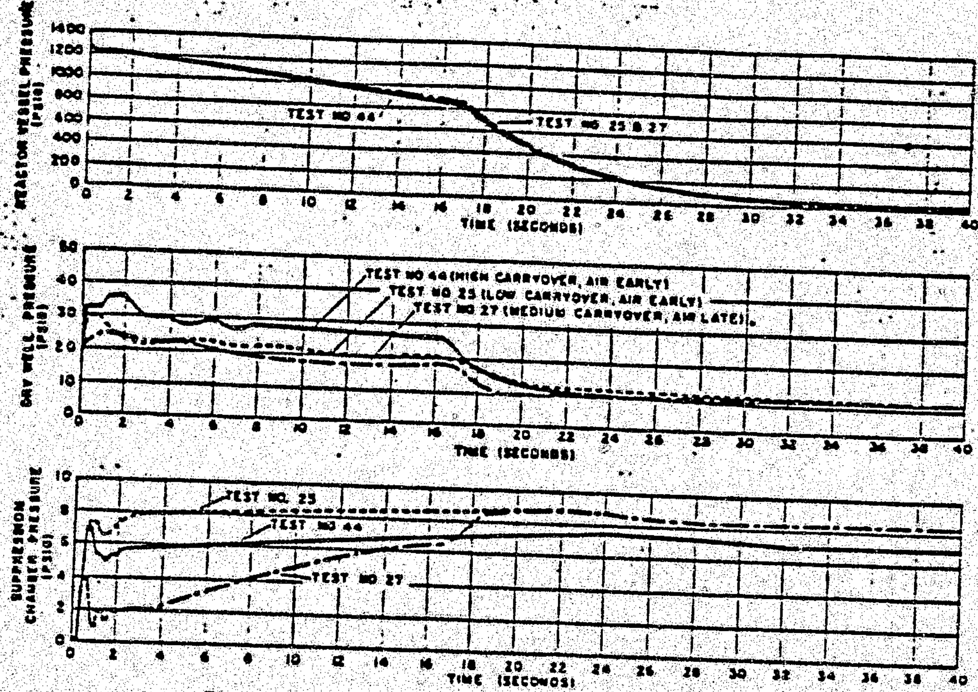

Fig. 12 Representative test results for 1.64-in. orifice

varied the volume of pool water from 50 to 170 per cent of the design value.

5 The initial temperature of the pool varied between 60 and 140 F and the final temperature was as high as 160 F. The Humboldt pool design is based on an initial temperature of 80 F and a final temperature of 113 F.

6 The initial reactor-vessel pressure was 1250 psig in all tests but one which had a pressure of 1000 psig.

Test Results

Summary. The suppression pool was completely effective in condensing steam in all tests except one where the vent was initially out of the water. The design pressure for the Humboldt suppression chamber is 10 psig but the maximum pressure with tests simulating the maximum credible accident was 9.3 psig.

The maximum dry well pressure for conditions simulating the Humboldt maximum credible rupture was between 25 and 36 psig as compared with the Humboldt design value of 72 psig. The difference between test and design is largely caused by the difference between the actual and calculated flow rates from the reactor vessel.

Test Results. Fig.12 is a plot of three representative test results using a 1.64-in. orifice in the rupture assembly. The difference in these tests are due to the use of different dry-well internal arrangements. Test No. 44 used arrangement A, Fig.11, which gave high water carry-over to the suppression chamber, as indicated by the amount of water in the dry well after a test. Test No. 25 used arrangement C which gave low water carry over. Test No. 27 used arrangement B which gave medium water carry-over and also resulted in a delayed venting of dry-well air.

Fig.13 is a plot of three tests with large orifices. Test 16 was twice, test 17 was three times, and test 33 was four times the rupture-orifice area of the tests shown in Fig.12. Table 7 is a tabulation of test data.

Reactor Vessel Pressure Trace. The top plots in Figs.12 and 13 show that flow from the simulated reactor vessel proceeds in the same manner as in the small transient tests described in Section 4. After an initial sharp drop and partial recovery, the pressure declines while the water is being expelled. Then, at about 17 sec in the case of Fig.12, the pressure starts to fall more rapidly, indicating that the water is gone from the reactor vessel, and the decay curve corresponds to blowdown of a steam-filled vessel.

Rupture Flow Rate. The average flow rate during the period of water expulsion can be calculated knowing the initial mass in the reactor vessel and the water-expulsion time as determined from the pressure trace. For every orifice size tested, the flow rate measured in this fashion

TABLE -
TEST DATA

	REACTOR VESSEL			DRY WELL				SUPPRESSION CHAMBER		
	Initial Pressure psig	Water Volume cu.ft.	Water Discharge Time, Sec.	Orifice Diam. Inches	Internal Arrangement	Max. Pressure psig	Residual Water cu.ft.	Submergence, ft.	Max. Pressure psig Measured	Max. Pressure psig Calculated
5	1250	38.3	17.2	1.64	A	40	0.8	6	7.0	9.4
6	1250	39.3	17.1	1.64	B	22	6.7	6	-	-
7	1250	39.3	17.1	1.64	B	26	5.8	6	8.2	8.8
8	1250	39.2	16.8	1.64	C	28	15.4	6	8.5	8.6
9	1250	39.1	12.2	2.10	C	45	14.2	6	7.5	8.3
10	1250	38.5	10.9	2.32	C	57	13.4	6	8.5	8.8
11	1250	38.7	-	1.64	C	-	-	3	-	-
12	1250	38.7	16.1	1.64	C	38	-	12.5	8.0	9.8
13	1250	38.3	16.4	1.64	D	32	11.8	6	8.1	7.2
14	1250	39.6	-	1.64	E	32	10.4	6	7.5	7.4
15	1250	39.6	17.0	1.64	A	33	4.2	6	8.8	8.7
16	1250	39.4	10.3	2.32	A	64	3.1	6	8.0	8.4
17	1250	39.4	7.9	2.84	A	90	2.4	6	8.5	7.5
20	1250	38.7	16.9	1.64	A	33	3.6	-0.5(3)	6.5	6.6
24	1250	38.2	16.8	1.64	A	32	2.7	6	9.3	9.4
25	1250	39.2	16.5	1.64	C	32	15.1	6	8.5	9.1
26	1250	39.2	16.7	1.64	C	33	-	6	8.5	9.0
27	1250	38.7	17.0	1.64	B	25	7.1	6	8.0	8.0
28	1250	38.7	17.0	1.64	B	25	6.8	6	7.8	8.0
29	1250	40.8	21.8	1.64	A	33	3.7	-1.5(3)	7.0	6.9
30	1250	48.9	21.2	1.64	A	35	4.3	-3.0(3)	14.5	10.2
31	1250	48.9	1860.	0.14	A	5.5	26.2	6	6.0	7.6
32	1250	39.2	10.7	2.32	A	59	1.7	-2.0(3)	7.3	6.9
33	1250	38.7	6.0	3.28	A	129	1.9	6	8.5	7.9
34	1250	38.7	81.0	0.64	A	10	-	6	6.5	7.5
35	1000	39.2	31.7	1.10	A	18	9.	6	8.0	7.9
36	1250	39.2	16.9	1.64	A	35	3.9	6	8.2	8.5
37	1250	39.2	17.3	1.64	F(1)	33	4.8	6	8.0	8.3
38	1250	39.2	16.5	1.64	F	32	3.5	6	-	-
39	1250	39.2	16.8	1.64	F	32	-	6	-	-
40	1250	37.8	16.5	1.64	A	34	4.3	6	11.3(2)	11.2
41	1250	38.7	16.9	1.64	A	34	4.2	6	12.5(2)	12.9
42	1250	38.7	16.7	1.64	A	37	-	6	13.4(2)	14.0
43	1250	37.8	310.	0.30	A	8	-	6	7.0	6.4
44	1250	37.8	16.6	1.64	A	34	4.1	6	7.5	7.2
45	1250	39.6	10.7	2.32	A	65	3.5	6	7.8	7.6

1. F is arrangement A without vent entrance plate.
2. Extra air injected in dry well during venting period.
3. Negative submergence indicates vent was initially out of the water.

(Tests 5-12 do not represent Humboldt because of large dry well.)

Table 8
Flow Rate
Comparison Between Measured and Calculated Vessel

Orifice diam, in.	Relation of test flow rate to predicted, per cent
.14	80
.30	80
.60	80
1.10	82
1.64	54
2.10	46
2.32	44
2.84	41
3.28	39

was significantly less than the value predicted by calculation methods used for the Humboldt design. A comparison between measured and calculated flows is shown in Table 8. The calculations used equation (2) with a flow coefficient of 0.61, and the density of saturated liquid.

The assumption of saturated liquid-density in the calculation conservatively neglects the effect of steam. As saturated water is expelled, some flashed steam may be drawn into the orifice with the water. This effect would be greatest in tests with large orifices and consequent large water velocities inside the vessel. In addition, the flashing to steam during flow through the orifice may be expected to influence the flow.

The flow rate per unit area of orifice was about the same for similar vessel and orifice configurations for the large tests and the small tests of Section 4.

Dry-Well Pressure. The maximum dry-well pressure was in the range of 25 to 36 psig for tests simulating the Humboldt maximum credible rupture accident. By comparison, the Humboldt dry well is designed for 72 psig. The range of pressures was caused by differences in dry well

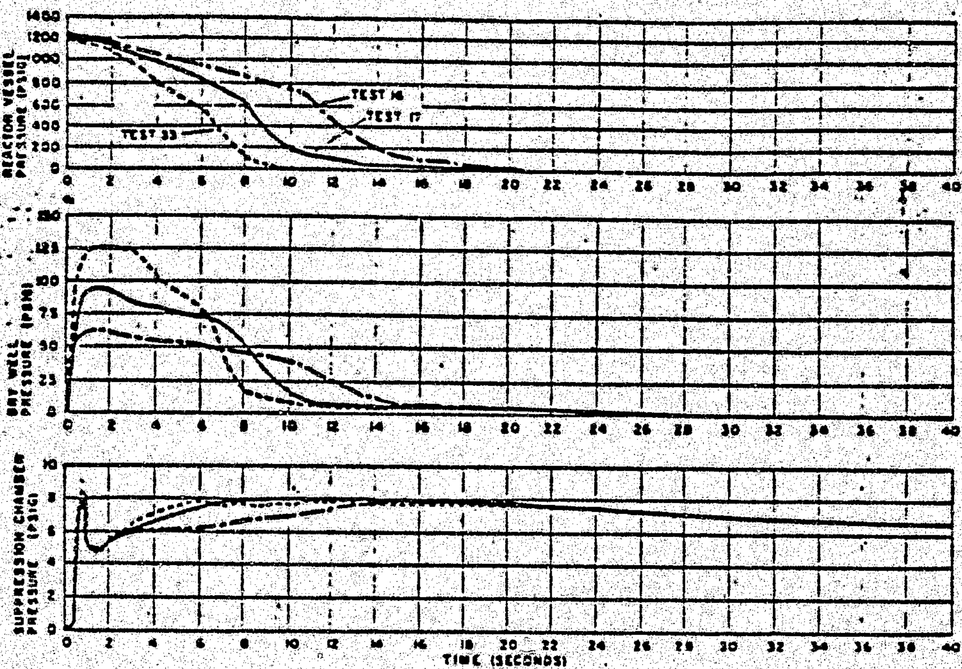

Fig. 13 Test results for large orifices

internals which influence the amount of water carry-over to the vent and the time air leaves the dry well. The dry-well pressure at the end of the test is approximately 8 psig. This pressure represents equilibrium between the hot saturated steam and water remaining in the dry well, and the suppression-chamber pressure.

Rupture flow rate significantly affected the maximum dry-well pressure. The highest dry-well pressure of 128 psig occurred with the 3.28-in. orifice while the lowest maximum dry well pressure of 5.5 psig resulted in a test with the 0.14-in. orifice.

Air did not appear to have any significant effect on dry-well pressure.

The initial depth of submergence affected the time required to clear the pool water out of the vents. Increasing initial submergence from 6 to about 12 1/2 ft increased vent clearing time and clearing pressure by about one third. Reducing submergence made the vent clearing dry-well pressure insignificant in comparison with the quasi-steady flow pressure which occurs during the 1 to 27-sec period in Fig. 6.

Between tests 12 and 13 the dry-well volume including vents was reduced from 308 to 230 cu ft. The earlier tests had about 20 per cent lower vent clearing transient pressures than the later tests as would be expected, but the change did not appear to have any other effect on dry-well pressure.

Heat transfer to the walls of the dry well apparently does not affect dry-well pressure significantly. Initial dry-well temperatures varied as much as 70°F on similar tests and dry well pressure appeared to be unaffected.

Pressure Drop in the Vent Pipe. Calculated pressures along the pipe based on average flow rates determined from test data and the homogeneous model of two-phase flow agree very well with test measurements. Unfortunately it was not possible to determine with any assurance whether critical pressure occurred at the vent exit because of instrumentation difficulties at this point. However, error in the exit pressure has a small effect on calculated dry well pressure.

Condensation in Water Pool. Condensation was rapid and complete in all tests where the vent pipe was initially submerged regardless of flow rate, added air, submergence, and pool temperature. This conclusion is based on comparison between measured and calculated maximum suppression-chamber pressures, shown in Table 7. The calculations used measured values of initial and final temperatures and volumes, assumed complete condensation, postulated complete transfer of air from the dry well, and neglected minor effects of heat transfer and relative humidity. Since the measured and calculated pressures agree well within the limits of the data, steam must have been condensed effectively. The design pressure for the Humboldt suppression chamber is 10 psig but the maximum pressure with tests simulating the maximum credible accident was 9.3 psig.

(a) Effect of flow rate. Flow rate did not have any detectable effect on condensation within the wide range tested.

(b) Effect of air. The presence of substantial quantities of air did not interfere significantly with condensation. Three tests with the 1.64-in. orifice were run with an auxiliary pressurized air tank connected through a quick-opening valve to the dry well. The valve was opened 5 or 6 sec after the beginning of the test when steam flow rate in the vents is high. In spite of the extra air, which was nearly equal to the quantity initially in the dry well, steam condensation was rapid and complete.

(c) Effect of submergence. Steam condensation in the suppression system was shown to be relatively insensitive to vent-pipe submergence. Four tests were conducted with the pool water level lowered so that the discharge end of the vent pipe was completely out of the water at the start of the tests. The most extreme test with complete condensation was run No. 32 with the vent initially 2 ft above the water and an orifice corresponding to twice the Humboldt maximum credible accident. Incomplete condensation was observed only in run No. 30 with the vent initially 3 ft above the water. The observed suppression-chamber pressure rose after the water was expelled from the reactor vessel and reached a value 4.5 psig higher than predicted for complete condensation. Apparently the jet momentum must be sufficient to bring the steam and pool water into intimate contact.

(d) Effect of pool temperature. The pool water can be hot and can experience a large temperature rise and still condense the steam effectively. The Humboldt pool is normally at 80 F and would rise 35 deg F following the assumed maximum credible accident. By comparison, complete condensation was achieved in run No. 26 when the initial water temperature was about 140 F. In another run a temperature rise of 58 deg F did not have any effect on condensation.

Suppression Chamber Pressure. At the beginning of the test, chamber pressure is zero until the pool water is cleared from the vents at about 0.3 sec in Fig.12. The transient chamber pressure which peaks during the first second may be explained by the sudden expansion of air from the vents into the pool. As the water breaks up and falls back to the pool, the chamber pressure falls off. This initial peak did not occur on any tests where the vent was initially out of the water.

The suppression-chamber pressure curves differ in shape mainly because of the dry-well internals, as is illustrated in Fig.12. Internals affect timing of air transfer and hence the manner of pressure rise in the suppression chamber.

Table 9

	Humboldt design conditions	Successful test conditions
Orifice size, in.	1.64[a]	0.14 to 3.28
Vent submergence, ft	6	12.5 submerged to 2 ft out of water
Initial pool temp, deg F	80	55 to 138
Final pool temp, deg F	115	84 to 161

[a] Test equivalent.

Margins in Humboldt Bay Pressure-Suppression Design

Large margins of safety are shown by the tests to exist in the Humboldt dry-well design pressure and the suppression-chamber design pressure. At least twice the maximum credible break area can be handled without exceeding either the suppression chamber or the dry-well design pressures. The suppression-chamber design pressure would not be exceeded with a break more than four times the maximum credible break but the dry-well design pressure would be exceeded. Table 9 gives a comparison of Humboldt design conditions with test conditions in which the suppression chamber did not go above the Humboldt design value.

Conclusions

The methods of analysis presented in this paper were used to predict the maximum pressures in a pressure-suppression containment design should a rupture accident occur. Tests showed that the analysis predicted higher dry-well pressures than were experienced and that condensation in the water pool was rapid and complete. The tests also showed that the Humboldt pressure-suppression containment design, for which the analysis was developed, has large margins of safety.

References

1 "Pressure Suppression Containment for Nuclear Reactors," by C.C. Whelchel and C.H. Robbins, ASME Paper 59-A-215.

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... p. 101.

4 "Mechanical Engineers Handbook," by L.S. Marks (editor), fifth edition, McGraw-Hill, New York, 1951, p. 285.

5 "Fundamentals of Hydro- and Aerodynamics," by L. Prandtl and O.G. Tietjens, McGraw-Hill, New York, 1934.

Q. IR-OCA-28-19 Would Dr. Levy agree that during tests in 1958 and 1959 at the CTF, GE engineers reported observations of hydrodynamic phenomena that caused the test facility to shudder and bang.

A. IR-OCA-28-19 A purpose of these tests was to obtain data respecting the limits of steam quenching ability of a suppression pool. In other words, the tests were designed to determine what would occur if a suppression pool's steam quenching ability was exceeded. To measure this effect, tests were performed imposing conditions upon the test facility which would not be expected to occur in normal operation or during credible accident conditions.

Thus, during the 1958 and 1959 tests at the CTF, tests were performed by injecting a steady flow of steam into the suppression pool until it reached a temperature of 120-130°. As reported in the published ASME Paper 59-A-215, a copy of which is attached to this interrogatory response: "Test results showed that tank vibration began when the water was 120-130°F or hotter. It was most severe at high steam flows. From the standpoint of design of pressure suppression pools, tank vibration is not likely to be a problem. The pool water could be kept below 120-130° F without seriously hampering the design, operating time of the pool and vent system would be short if ever, and hot pool water would occur at a time when steam flow through the vents would have fallen off from the maximum." My recollection is that some of these tests were allowed to produce severe tank vibration. However, as noted in my testimony, such steady flow condensation tests are not relevant to LOCA conditions where the steam flow rate decreases with time and includes air in the early stages of the accident. Furthermore, subsequent transient tests typical of LOCA reported in ASME Paper No. 61-WA-222 and the Bodega Bay test report showed no condensation effects with suppression pool temperatures up to 163°F as reported in my testimony. Finally, restrictions on initial pool temperature are provided in plant Technical Specifications to avoid reaching such temperatures during a LOCA or relief valve actuation. Accordingly, these vibrations were not viewed as having design significance until 1974, since this occurrence was anticipated to be prevented by alternative means.

Responsible Witness: Salomon Levy
S. Levy Incorporated

Q. IR-OCA-28-20 Would Dr. Levy agree that during the 1958-1959 tests at CTF, water hammer effects were observed?

A. IR-OCA-28-20 No water hammer effects involving large pressure changes occurred at the Condensing Test Facility.

Responsible Witness: Salomon Levy
S. Levy Incorporated

Q. IR-OCA-28-21 Would Dr.Levy agree that during the 1958-1959 CTF test, shaking made the test tank appear to bounce in its foundation.

Q. IR-OCA-28-22 Would Dr. Levy agree that during the 1958-1959 CTF tests, one test engineer recorded an earthquake in a control room several hundred yards away.

A. IR-OCA-28-21 & 22 Although Dr. Levy would agree that, as the specific result of the nature and purpose of the tests performed, severe tank shaking was observed during the tank vibration tests, Dr. Levy cannot confirm the assertions made in the questions.

Responsible Witness: Salomon Levy
S. Levy Incorporated

IR-OCA-28-23
IR-OCA-28-24
IR-OCA-28-25

Q. IR-OCA-28-23 Would Dr. Levy agree that GE recorded its CTF test results in a document titled "GEAP-3143, Test Report for the Pressure Suppression Development Program," dated April 2, 1959. Please state Dr. Levy's familiarity with this document and when he first reviewed or became aware of it.

Q. IR-OCA-28-24 Would Dr. Levy agree that GEAP-3143 was initially restricted to GE employees.

Q. IR-OCA-28-25 Please provide a copy of GEAP-3143. The OCA will sign an appropriate protective agreement, if necessary.

A. IR-OCA-28-23, 24 & 25 Dr. Levy agrees that GE recorded its CTF test results in a document numbered GEAP 3143. Dr. Levy received and reviewed GEAP 3143 shortly after its preparation, and had occasion to rereview it on several occasions while still at GE. GEAP 3143 was provided to the AEC shortly after its preparation as part of the application for licensing of the Humboldt Bay facility.

As described in Dr. Levy's testimony, the CTF tests were a larger and more comprehensive test series to determine the feasibility of the pressure suppression concept. As such these results are of substantially greater significance than the Vallecitos tests. However, the CTF test facility was neither sized nor configured to equate to an operating BWR's containment. Also, the test conditions which produced the vibrations which are the subject of these interrogatories are, as explained above, not credible occurrences under nuclear plant operating or accident conditions.

Neither PECO nor Dr. Levy has a copy of GEAP-3143.

Responsible Witness: Salomon Levy
S. Levy Incorporated

IR-OCA-28-26
IR-OCA-28-27
IR-OCA-28-28

Q. IR-OCA-28-26 With respect to the 1960 Humboldt Bay tests referred to at pages 10-11 of St. 34, would Dr. Levy agree that during a test at Moss Landing in 1960, technicians observed an upsurge of suppression pool water that occurred as the air leaving the vent pipe expanded rapidly in the pool and threw a mass of water upwards at high velocity.

Q. IR-OCA-28-27 Would Dr. Levy agree that the May and June 1960 Humboldt Bay test effects decribed in question 26 above are now referred to as pool swell.

Q. IR-OCA-28-28 Please state Dr. Levy's familiarity with the Humboldt Bay tests. Please state his involvement.

A. IR-OCA-28-26, 27 & 28 The 1960 Humboldt Bay tests were performed at the request of the AEC in a test facility configured to equate to 1/48th of the proposed Humboldt Bay containment suppression pool. Various parameters of the test were specified by the AEC.

Dr. Levy would agree that technicians did observe an upsurge of suppression pool water during the tests. This upsurge was reported and described in ASME Paper No. 61-WA-222, a copy of which is attached to this interrogatory response, as follows: "The (suppression chamber) pressure which peaks during the first second may be explained by the sudden expansion of air from the vents into the pool. This expansion throws some of the pool water up into the chamber air space compressing the air. As the water breaks up and falls back to the pool, the chamber pressure falls off." However, this observed upsurge of water did not constitute "pool swell". As pointed out in NUREG-0487, the term pool swell is used to describe the coalesence of air bubbles formed at the end of each vent and the growth of this coalesced air until it reaches across the entire suppression pool and lifts a water ligament above it. No observations of an unbroken water ligament covering the entire pool was made during the 1960 test, nor were associated impact load measurements made prior to 1974. The results of the 1960 Humboldt Bay tests were reported to the AEC. Dr. Levy had no involvement in the actual conduct of the 1960 Humboldt Bay tests.

Responsible Witness: Salomon Levy
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Q. IR-OCA-28-29

Is Dr. Levy familiar with a Mr. Charles Robbins, then of GE. Does Dr. Levy know if Mr. Robbins reported that the Humboldt Bay Tank structure shook beneath him during the tests.

A. IR-OCA-28-29

Dr. Levy is familiar with Mr. Robbins. However, Dr. Levy is not familiar with the statement attributed to Mr. Robbins. Nc such statement was made by Mr. Robbins to Dr. Levy.

Responsible Witness: Salomon Levy
S. Levy Incorporated

Q. IR-OCA-28-30 To Dr. Levy's knowledge, did GE measure or calculate the structural response of the Humboldt test tank to the forces described in question 26 above.

A. IR-OCA-28-30 General Electric measured and calculated the transient pressures in the simulated reactor vessel, drywell, the space over the suppression pool, and the differential pressure between the drywell and the air space over the water. These are important parameters to determine the completeness of condensation.

Responsible Witness: Salomon Levy
S. Levy Incorporated

Q. IR-OCA-28-31 With respect to the Bodega Bay tests referred to at page 11 of St. 34, is Dr. Levy aware whether pool swell occurred during tests at that facility in the early 1960's. If yes, please state your knowledge of these events.

A. IR-OCA-28-31 As reported in the Bodega Bay Unit I Preliminary Hazards Summary Report, Appendix I, Pressure Suppression Test Program, visual observation was not practical during the Bodega Bay testing. However, "a brief upward surge of pool water" was deduced to have occurred during the Bodega Bay tests because they exhibited similar pressure behavior to that observed during the Humboldt Bay tests. However, it was not concluded as the result of this pressure surge observation that "pool swell", as identified and defined as the result of the mid and late 1970s test program, had occurred. As described above, no containment design significance was attached to the pressure surge or its associated water upsurge.

Responsible Witness: Salomon Levy
S. Levy Incorporated

Q. IR-OCA-28-32 Is Dr. Levy aware whether GE, in 1963 conducted tests of pressure suppression in which pool swell, chugging, or other hydrodynamic forces were recorded. If yes, please state your knowledge of these tests.

Q. IR-OCA-28-32 The tests referred to in both this interrogatory and IR-OCA-28-31 are the Bodega Bay single vent and multi-vent tests. The purpose of these tests was to reconfirm the pressure suppression concept employing the anticipated Bodega Bay downcomer and suppression pool configurations. Bodega Bay employed a larger diameter downcomer pipe than was used at the Humboldt Bay. The results of these tests were provided to the NRC as part of the Bodega Bay licensing proceeding. Ultimately, the facility was not constructed due to seismic concern with the site.

Pool swell, chugging or other significant hydrodynamic forces such as those identified in 1974 and 1975 were not observed during these tests. Pressure surges and vent condensation similar to that described in IR-OCA-28-26 were observed, but were not judged to be significant. These phenomena were reported as part of the test results provided to the AEC.

These tests have been repeatedly cited and their results provided to the AEC or NRC as part of the construction permit applications for Mark I and Mark II BWR plants.

Responsible Witness: Salomon Levy
S. Levy Incorporated

Q. IR-OCA-28-33 Is Dr. Levy aware whether certain baffles in the 1963 GE tests were bowed and twisted due to hydrodynamic forces. If yes, please state Dr. Levy's knowledge of these occurrences.

Q. IR-OCA-28-34 Is Dr. Levy familiar with an internal GE report by Mr. Charles Robbins referencing the baffle damage mentioned above. If yes, please state his familiarity and understanding of the report.

A. IR-OCA-28-33&34 Dr. Levy is not aware of any baffle damage due to hydrodynamic forces in the 1963 GE tests.

Dr. Levy is also not aware of any internal GE report by a Mr. Charles Robbins referencing any such damage.

Responsible Witness: Salomon Levy
S. Levy Incorporated

Q. IR-OCA-28-35 — Is Dr. Levy familiar with a report by Dr. Fred J. Moody dated October 1964, titled "Some Thoughts on Condensation in Pressure Suppression Systems." If yes, please state Dr. Levy's familiarity and understanding of the contents of the report.

Q. IR-OCA-28-36 — Is it Dr. Levy's understanding that Dr. Moody's report discussed test observations of pool swell and recommended further investigation.

Q. IR-OCA-28-37 — Please provide a copy of Dr. Moody's report. An appropriate protective agreement will be signed by OCA, if necessary.

A. IR-OCA-28-35, 36&37 — Dr. Levy is not familiar with and has never reviewed the cited report by Dr. Moody. Neither Dr. Levy nor PECO possesses a copy of the report.

Responsible Witness: Salomon Levy
S. Levy Incorporated

Q. IR-OCA-28-38 Is Dr. Levy aware whether in 1969, GE resumed pressure suppression tests during which GE engineers observed hydrodynamic forces? If yes, please state his understanding of the tests and forces and the basis of his knowledge.

Q. IR-OCA-28-39 Is Dr. Levy aware whether GE prepared a report entitled "NEDM-13036-1, Pressure Supression Pool Investigations Report #1" dated July, 1970 which discussed pressure suppression tests results. If yes, please state Dr. Levy's understanding of what the report discussed and concluded and the basis for that understanding.

Q. IR-OCA-28-40 Please provide a copy of NEDM-13036-1. An appropriate protective agreement will be signed by OCA, if necessary.

Q. IR-OCA-28-41 Does Dr. Levy know whether NEDM-13036-1 was made available by GE to the AEC/NRC. If so, when?

A. IR-OCA-28-38, 39,40&41 Dr. Levy is aware that, starting in the late 1960s, the Development Engineering organization of General Electric Company initiated studies of advanced containment designs which led to the development of the Mark III concept. These studies were complemented by tests of the Mark III pressure suppression concept "at a very small scale (about 1/2000)" which "lacked geometrical similitude to the Mark III configuration." These preliminary feasibility tests were, in many respects, very similar to those at the Vallecitos facility in the late-1950s. These very small-scale tests were not designed to determine or measure hydrodynamic effects. Dr. Levy was advised of these tests and their results as they occurred due to his position as Manager of Design Engineering. Dr. Levy is aware of NEDM-13036-1, but does not recall ever having received that document. Neither

Dr. Levy nor PECO has a copy of that document. Dr. Levy does not specifically know whether NEDM-13036-1 was provided to the AEC/NRC, but results of these tests were provided to the AEC/NRC as part of its evaluation process described at Schedule 5 of Dr. Levy's testimony.

Responsible Witness: Salomon Levy
S. Levy Incorporated

Q. IR-OCA-28-42

Is Dr. Levy aware whether a number of pressure suppression tests were conducted by GE between 1970 and 1972, which tests showed pool swell and chugging. If yes, what is Dr. Levy's understanding of these tests, and the phenomena observed. What is Dr. Levy's basis for his understanding?

A. IR-OCA-28-42

Dr. Levy is aware that testing of the Mark III concept was continued beyond 1970. This testing was not in full-sized facilities, but rather in small scale tanks. Moreover, the Mark III concept has a configuration much different from the Mark II. The Mark II design utilizes vertical vents submerged in the suppression pool, in contrast to the horizontal vents in the Mark III pool. Chugging was not observed in these tests. Pool swell, however, was observed. Also, a form of condensation oscillation attributable to the special configuration of the Mark III and not present in the Mark II was observed. These observations led to the decision to build a large scale test facility to determine whether the observed phenomenon would be present in actual plant configurations. Immediately following these large scale tests, at which similar phenomena were observed, the NRC letters of 1975 were issued.

Dr. Levy is familiar with these events due to his several positions at GE during this period and his general study and knowledge of these hydrodynamic phenomena.

Responsible Witness: Salomon Levy
S. Levy Incorporated

Q. IR-OCA-28-43

Is Dr. Levy aware whether any operating GE designed plants experienced baffle and/or other structural damage due to SRV discharges during 1971-1972. If yes, please state the plants, the year, the damage, and the reasons for the damage. Please state Dr. Levy's basis for his knowledge.

A. IR-OCA-28-43

Dr. Levy is aware of baffle damage to operating plants produced by relief valve discharge. These baffles were provided at the request of AEC/ACRS to conform the Humboldt and subsequent pressure suppression designs to the Humboldt Bay test configuration. Baffle damage was observed in 1971 and it involved several plants, e.g., Monticello, Oyster Creek, and Nine Mile Point. The baffle damage was produced by the proximity of the baffles to the relief valve discharge points. To alleviate this problem, baffle removal was proposed and approved by the NRC. As Manager of Design Engineering, Dr. Levy was responsible for the work which led to the removal of the baffles.

Responsible Witness: Salomon Levy
S. Levy Incorporated

Q. IR-OCA-28-44 Is Dr. Levy aware whether in November 1972, GE engineer A.J. Levine advised GE managers that hydrodynamic loads should be considered for Mark II containments. If yes, please state Dr. Levy's understanding of Mr. Levine's advice, and the basis for his understanding.

A. IR-OCA-28-44 Dr. Levy received no such advice from Mr. Levine, and is not aware of such advice having been provided to any other GE Manager.

Responsible Witness: Salomon Levy
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