File 17

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

APR 2 9 1975

Frank Schroeder, Acting Director for Technical Review
THRU: Robert L. Tedesco, Assistant Director for Containment Safety, TR

POOL DYNAMIC LOADS ON MARK I CONTAINMENTS

In response to your request, the following is a brief summary of the background and current status of our understanding of pool dynamic loads on Mark I pressure suppression type containments.

Recent developments have indicated that pool dynamic loads may not have been fully considered in the structural design of BWR plants utilizing Mark I and Mark II type containments. In response to this situation we have sent letters to all licensees/applicants for these types of plants requesting that they report on the potential magnitudes of pool dynamic loads and the structural capability of their suppression chamber design to tolerate such loads. We have requested that operating plants provide this information within 60 days and to notify us within 15 days as to their ability to meet such a schedule. For plants under construction we have requested that a schedule for response be submitted within 30 days. We have established an interim time period for assessment of each operating Mark I plant based on our conclusion that pool dynamic loads do not represent an immediate safety concern for these plants. This conclusion was reached on the basis of the information provided below which describes the background and current status of our understanding of the problem.

In March of 1974, GE performed a series of "air tests" to scope the range and magnitude of pool dynamic loads for the Mark III design. It was recognized that more definitive tests were required and therefore comprehensive tests in 1/3 scale were initiated in the summer of 1974 and are currently still in progress. Parallel efforts to develop analytical methods for the various pool dynamic phenomena have been undertaken by GE, the NRC's consultants, and by several A/E's.

We have maintained periodic contact with GE regarding the planning and progress of pool dynamics testing and associated analyses. Although the emphasis has been placed on resolution of these concerns for the Mark III design, our discussions with GE have noted that parallel efforts should be directed at evaluation of the Mark I and II containments designs.

At a meeting held in April, 1974, during which the results of the air tests were discussed with us by General Electric, we noted to them that it was apparent to us that this phenomenon did not readily lend itself to analysis and that not only additional testing was required to measure these forces but also that additional work should be done to determine the significance of these forces with respect to the Mark I and Mark II containments.

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General Electric stated that these forces would be less significant to the Mark I and Mark II containments but that they could not at that time quantify these forces.

In their letters of April 9 and 15, 1975 (A. P. Bray to E. G. Case), and in a meeting on April 10, 1975, GE provided a summary of their further actions in this regard. GE has performed a preliminary generic evaluation of pool swell loads for a typical internal structure arrangement of a Mark I containment torus (see Figure 1). The structural response analysis was based on pool swell loads extrapolated from Mark III test data, which is currently the only available data base. The resultant load profile, as shown in Figure 2, is a pressure pulse of approximately 40 psi and a duration of about 60 milliseconds. Some of the assumptions that GE used to arrive at this profile are listed in the enclosed Figure 3.

Although the pool swell loads have been derived from Mark III data, differences in the Mark I design suggest that the loads are conservative. These are the small free air volume of the torus and the lower submergence of the vent piping, compared to Mark III plants. The Humboldt and Bodega Bay tests had shown dramatically reduced pool motion with the suppression chamber closed (as during plant operation) as with it open. It is thought that this was the result of air compression effects dampening the pool swell phenomena. The Mark III test program has shown that the range and magnitude of pool swell effects are directly proportional to the submergence of the vents. The vent submergence for Mark I plants is four feet compared to a value of 7-1/2 feet for Mark III. Therefore, extrapolation of Mark III data, which is representative of deep submergences and a large (open) suppression chamber, to the Mark I design would indicate a degree of conservatism.

GE has completed a preliminary structural analysis for a typical Mark I configuration (Browns Ferry) and arrived at a preliminary conclusion that the loss of containment integrity will not occur. However, the preliminary analysis indicated that local yielding will occur in the clevis support of the ring header. Accordingly, GE has started a detailed structural analysis of the ring header and its clevis supports to determine the extent of any yielding, its potential effects on containment integrity, and the need for any backfitting.

GE also noted that the Mark I configuration was tested in full scale during the Humboldt and Bodega Bay tests performed between 1958 and 1965, and that the results of these tests verified the adequacy of the primary containment boundary. However, specific measurements of pool dynamics was not included as part of the test program.

Recognizing that the GE analysis is preliminary and that we have not reviewed it in detail, this analysis still represents the best estimate of the problem at this time. We believe the conclusions expressed by GE to be reasonable

on an interim basis. We have advised each licensee by letter requesting detailed analysis with respect to this matter and have also advised GE, by letter dated April 17, 1975, that the extrapolation of Mark III data may be acceptable on an interim basis but that we currently believe that additional testing may be required.

Gus C. Lainas, Chief
Containment Systems Branch
Division of Technical Review

Enclosures:
As stated

cc: B. Rusche
E. Case
TR A/D's

FIGURE 1

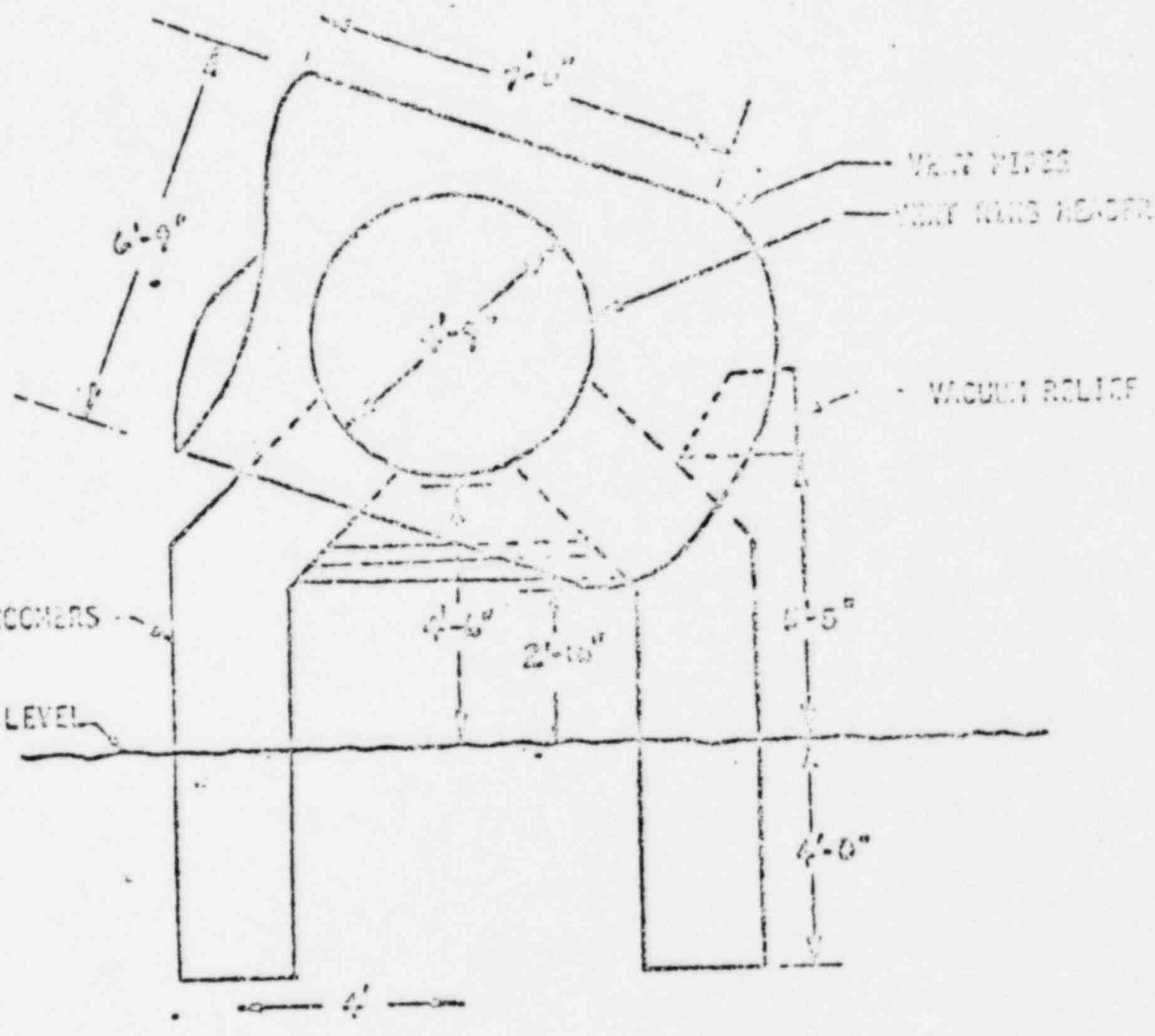
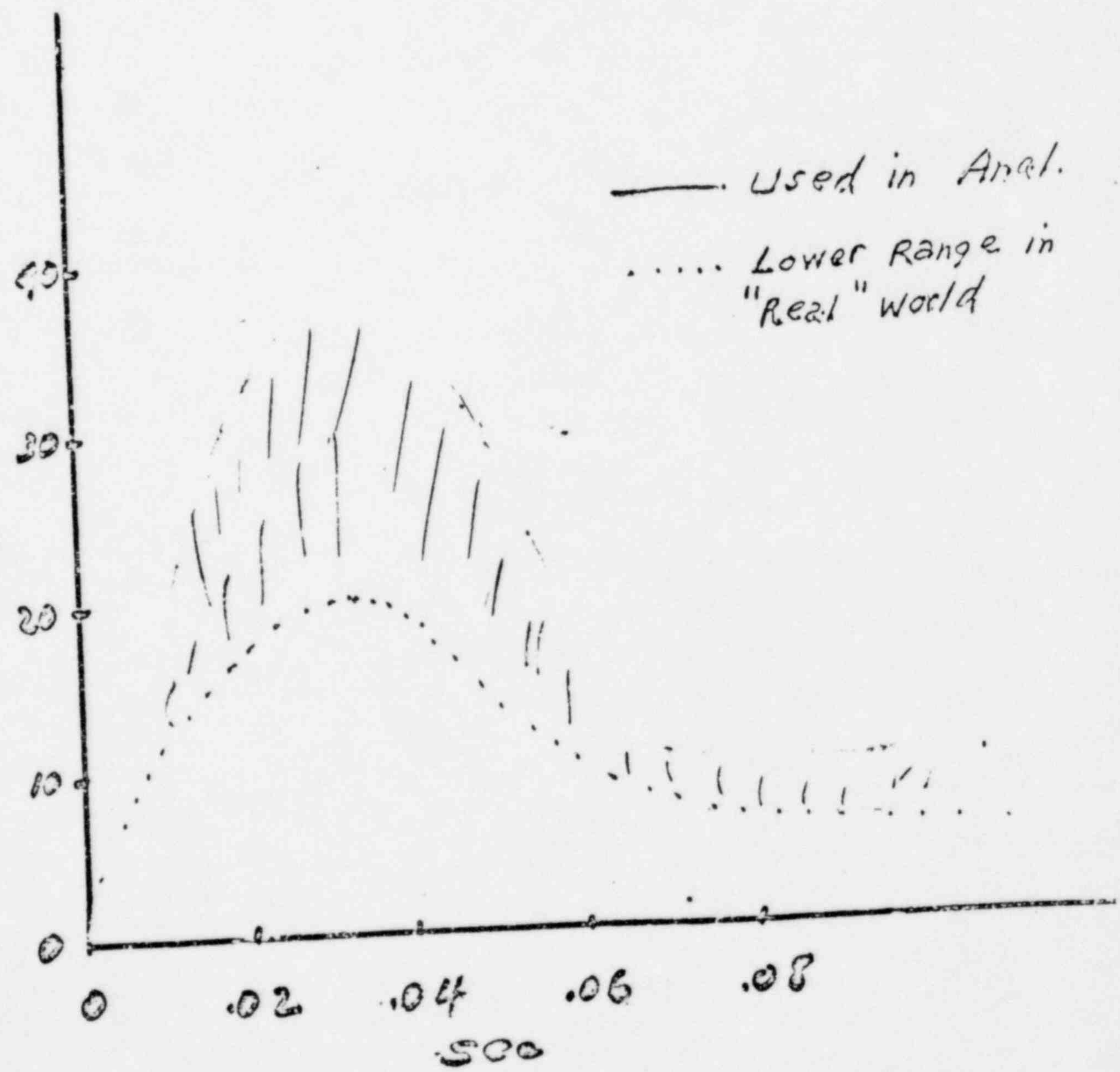


FIGURE 2. MARK I TYPICAL VENT SYSTEM (BROWNS FERRY)

FIGURE 2



Used in Anal.
Lower Range in "Real" world
Avg PSI
40
30
20
10
0
0
.02
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.08
SCO

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General Electric
Topical Reports Related to Ramshead

*NEDE-21062-P	Comparison of Safety-Relief Valve Model Predictions with Test Data
NEDO-20942 & *NEDE-20942-1P	Safety-Relief Valve Discharge Analytical Models (Amendment No. 1)
NEDO-10859	Steam Vent Clearing Phenomena and Structural Response of the BWR Torus (Mark I Containment)
NEDC-21581	Final Report In-Plant Safety/Relief Valve Discharge Load Test - Monticello Plant
*NEDC-21465-P	Preliminary Report In-Plant Safety/Relief Valve Discharge Load Test - Monticello Plant
*NEDE-20942-P	Safety-Relief Valve Discharge Analytical Model

*Proprietary Information

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Consultant's Report
(Proprietary Information)

INEL SRD-120-76	"Critique of General Electric Safety-Relief Valve Discharge Analytical Models (Amendment No. 1)"
INEL PG-R-76-002	"Review of Safety Relief-Valve Analytical Models."
INEL SRD-71-76	"Analysis of the General Electric Safety-Relief Valve Discharge Analytical Model"
INEL SRD-79-76	"Critique of the General Electric Safety-Relief Valve Discharge Analytical Model"
INEL RE-S-76-6	"Review of Safety-Relief Valve Analytical Models"

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	DATE	COORDINATION
2	INITIALS	FILE
	DATE	INFORMATION
3	INITIALS	NOTE AND RETURN
	DATE	PER CONVERSATION
4	INITIALS	SEE ME
	DATE	SIGNATURE

REMARKS

Any Comments or anything that you would like to add?

Urgent

Do NOT use this form as a RECORD of approvals, concurrences, disapprovals, clearances, and similar actions

FROM (Name, office symbol or location): L. Ahao

DATE: 12/27/74

PHONE

OPTIONAL FORM 41
AUGUST 1967
GSA FPMR (41CFR) 100-11.206

GPO 043-16-81418-1 419-015 5041-101

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DRAFT/SLEGERS/law/11 27-74

Dr. Ivan Stuart, Manager
Safety and Licensing
Nuclear Energy Division
General Electric Company
175 Curtner Avenue
San Jose, California 95114

Mr
Dear Dr. Stuart:

We reference your presentation of November 1, 1974, to the regulatory staff and your letter of November 6, 1974 on the same subject. The recent occurrences in several BWR plants that have prompted the letter have again pointed to the complications that can arise with the discharging of steam from relief valve vents. At the same time the adequacy of the suppression pool structure to withstand the resulting pressure oscillations over the life of the plant is still a largely open item. Furthermore, your letter points to the importance of imposing stricter specifications on the maximum temperature that is operationally maintained in the pool.

This letter addresses the overall problem of pressure oscillations in the suppression pool caused by relief valve action. We consider the problem to be generic to all BWR's with pressure suppression currently operating or in the construction or licensing phase which includes all three containment designs MARK I, II and III. The procedure that will be followed within the regulatory staff is that the Operating Reactors Branch will concern itself with all plants using the MARK I containment in separate correspondence. Therefore the information requested in this letter pertains to Mark II and Mark III containments on a generic basis.

The phenomena that have caused attention to be focused on the suppression pool can generally be classified in two categories. First there is the vent clearing phenomena that causes pressure oscillations in the suppression pool. Damage caused by these oscillations have been observed in a number of instances such as the spray header in Browns Ferry 1, the RHR header of Quad Cities 2 and earlier the suppression pool baffles in a number of plants. These oscillations have also been the subject of test programs performed in Quad Cities 2 and Browns Ferry 1.

There is also the problem of steam quenching at elevated pool temperature, which was first brought to our attention by General Electric at a presentation in Bethesda on November 1, 1974. You pointed out that pressure oscillation in the suppression pool also arises when pool temperatures of about 160 degrees Fahrenheit are exceeded, and the massflow rate through the vent is a significant fraction of the rated flow rate. These observations were based on actual plant experience in two European BWR plants.

Relief Valve Vent Clearing

Experimental information on the pressure oscillation in MARK I plants following vent clearing has been reported.(1)(2) A theoretical model for

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(1) NEDO 10859 "Steam Vent Clearing Phenomena and Structural Response to the BWR Tarus (MARK I Containment)" G. E. Company.

(2) "1973 Browns Ferry Unit 1 Tarus Experience" submitted by the Tennessee Valley Authority to the Office of Regulation, May 7, 1974.

predicting the amplitude of the oscillation is also included in (1) which was submitted as a topical report. This report was considered unacceptable mainly because of the small number of tests that supported the data. With respect to the applicability of these test results to the different pool designs of the MARK II and MARK III containments further questions present themselves. To date a specification of pressure amplitudes on several MARK III dockets of +25/-10 psi is based on MARK I test information. To determine the adequacy of this specification the following points should be addressed:

1. The need for further testing to verify the adequacy of the pressure specifications in the suppression pool for single and multiple vent opening events. and to obtain data for structural response
2. The need to verify the theoretical model in the range of vent parameters that apply to the MARK II and MARK III containments.
3. The need to perform testing in a suppression pool configuration that more nearly corresponds to the rigid structure of the MARK II and MARK III suppression pool.
4. Identification of components in the suppression pool other than bounding walls of the structure, and the location of such components relative to the vent exit.
5. The predicted maximum number of single and multiple relief valve openings over the life of the plant.
6. The loading of the suppression pool structure during a combination of small break LOCA and relief valve actuation.

Steam Quenching Vibrations

The steam quenching vibrations are critically dependent on the local pool temperature in the vicinity of the vent exit. These vibrations will set in when a temperature of about 160 degrees Fahrenheit is exceeded. The average pool temperature in the course of a reactor transient or accident is dependent on the course of events following the occurrence. In this context you are requested to address the following points:

1. The maximum temperature limits that will be specified for the suppression pool under normal operating and reactor transient conditions.
2. The temperature transient starting from these specified limits for:
   a. Containment Isolation
   b. Semi-automatic blowdown
   c. Stuck open relief valve

   covering all plant variations assuming the minimum amount of water in the suppression pool.
3. The type of operator action that will be required when specified temperature limits are exceeded.
4. The level of temperature instrumentation that will be installed in the pool and the sampling or averaging technique that will be applied to arrive at a definitive pool temperature.

It is requested that you submit within 30 days after receipt of this letter your time schedule for submitting this information.

Robert L. Tedesco, Assistant Director
for Containment Safety
Directorate of Licensing