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STRUCTURAL EVALUATION OF THE VACUUM BREAKERS
(MARK I CONTAINMENT PROGRAM)

NEBRASKA PUBLIC POWER DISTRICT
COOPER NUCLEAR STATION

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FOREWORD

This Technical Evaluation Report was prepared by Franklin Research Center under a contract with the U.S. Nuclear Regulatory Commission (Office of Nuclear Reactor Regulation, Division of Operating Reactors) for technical assistance in support of NRC operating reactor licensing actions. The technical evaluation was conducted in accordance with criteria established by the NRC.

1. INTRODUCTION

In a latter state of the generic resolution of the suppression pool dynamic load definition of the Mark I Containment Long-Term Program, a potential failure mode of the vacuum breakers was identified during the chugging and condensation phases of hydrodynamic loadings. To resolve this issue, two vacuum breaker owner groups were formed, one for those with General Precision Engineering (GPE) vacuum breakers, the other for those with Atwood-Morrill (AM) vacuum breakers.

The issue was not part of the original scope of the Mark I Containment Long-Term Program as described in NUREG-0661 [1]. However, vacuum breakers have the function of maintaining containment integrity and, therefore, are subject to Nuclear Regulatory Commission (NRC) review. In a generic letter dated February 2, 1983 [2], the NRC requested all affected plants either to submit the results of the plant-unique calculations which formed the bases for modifications to the vacuum breakers or to provide the justification for the as-built acceptability of the vacuum breakers.

Franklin Research Center (FRC) has been retained by the NRC to evaluate the acceptability of the structural analysis techniques and design criteria used in the plant-unique analysis (PUA) reports of 16 plants. As a part of this review, the structural analysis of the vacuum breakers has been reviewed and documented in this report.

The first part of this report (Sections 1 through 4) consists of generic information that is applicable to all affected plants. The second part of the report (Sections 5 and 6) provides a plant-specific review, which pertains to the Cooper plant.

1.1 GENERIC BACKGROUND

In 1980, the Mark I owners and the NRC became aware of the vacuum breaker damage during full-scale test facility testing and of the potential for damage during actual LOCAs. Two vacuum breaker owner groups, General Precision Engineering (GPE) and Atwood-Morrill (AM), were formed to develop action plan for resolving this issue. In February 1983, the NRC issued Generic Letter 83-08 [2], requesting commitments from affected utilities to provide analytical results. The licensees responded to the NRC request by developing appropriate force functions simulating the anticipated hydrodynamic loads, and then performing stress analyses that used these loads. With respect to loading, the NRC has reviewed and issued a staff position as indicated in Section 3. FRC's function is to review the stress analysis submitted by a licensee.

1.2 VACUUM BREAKER FUNCTION

During steam condensation tests on BWR Mark I containments, the wetwell-to-drywell vacuum breakers cycled repeatedly during the transient phase of steam blowdown. This load was not included in the original load combinations used in the design of the vacuum breakers. Consequently, the repeated impact of the pallet on the valve seat and body created stresses that may impair its capability to remain functional.

A vacuum breaker is a check valve installed between the wetwell and the drywell. Its primary function is to prevent the formation of a negative pressure on the drywell containment during rapid condensation of steam in the drywell and in the final stages of a LOCA. The vacuum breaker maintains a wetwell pressure less than or equal to the drywell pressure by permitting air flow from the wetwell to the drywell when the wetwell is pressurized and the drywell is depressurized slowly.

A vacuum breaker can be internally or externally mounted. Figures 1 and 2 illustrate locations of vacuum breakers.

Schematics of typical GPE and AM vacuum breakers are illustrated in Figures 3 and 4.

A typical pressure differential vacuum breaker during a LOCA is provided in Figure 5.

Table 1 lists the various vacuum breaker types and the plants affected by them.

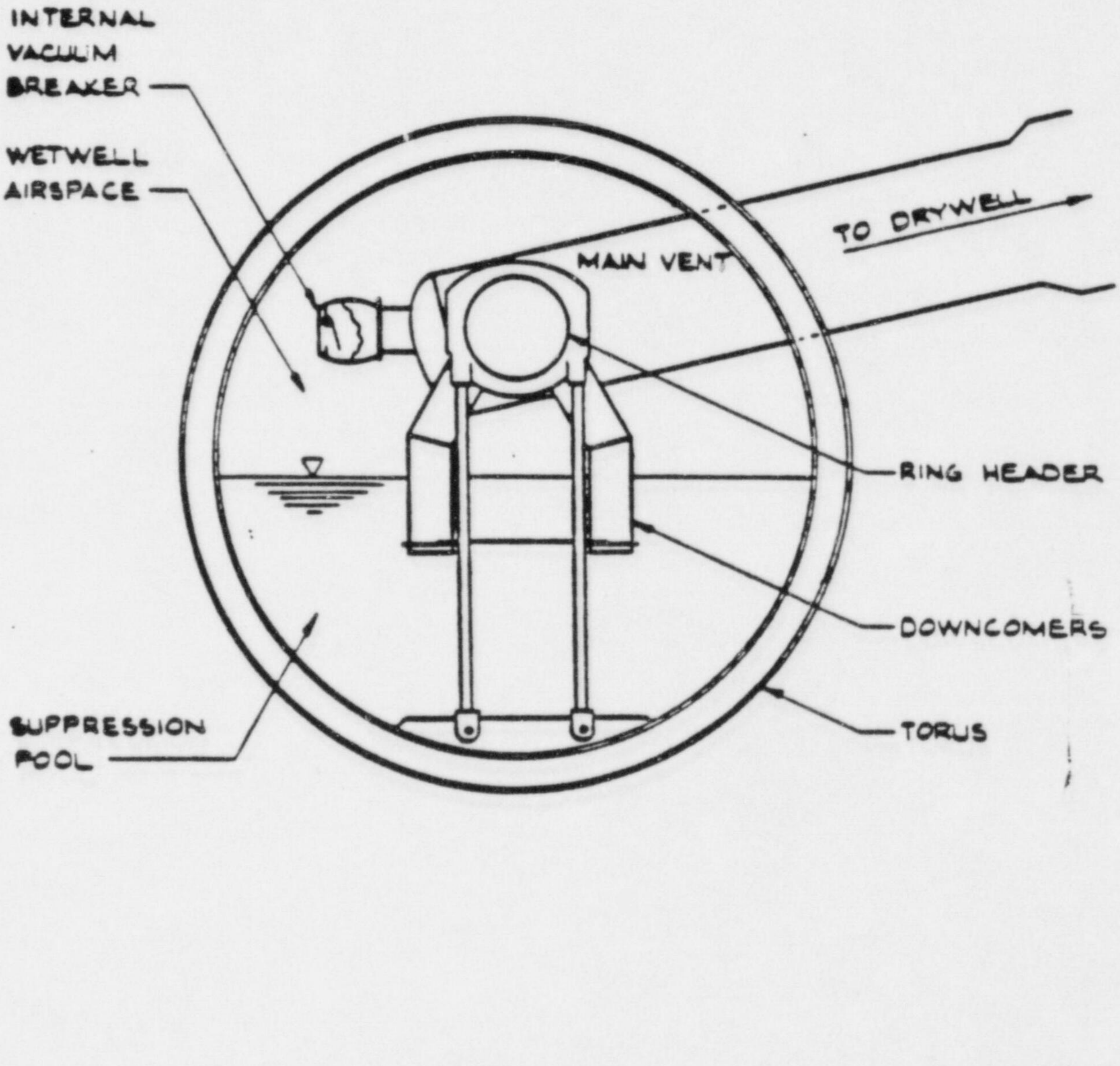

Figure 1. Internal Vacuum Breaker

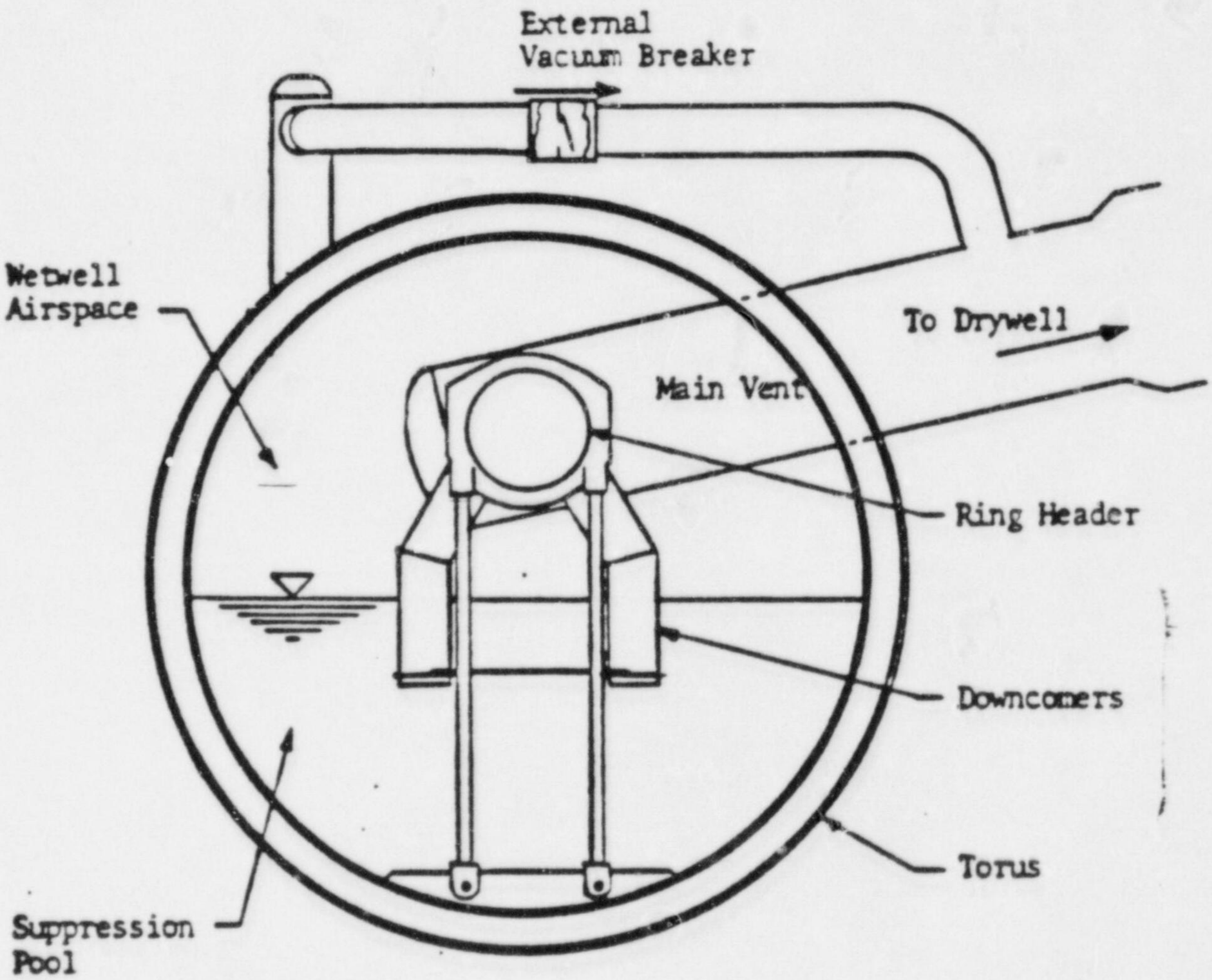

Figure 2. External Vacuum Breaker

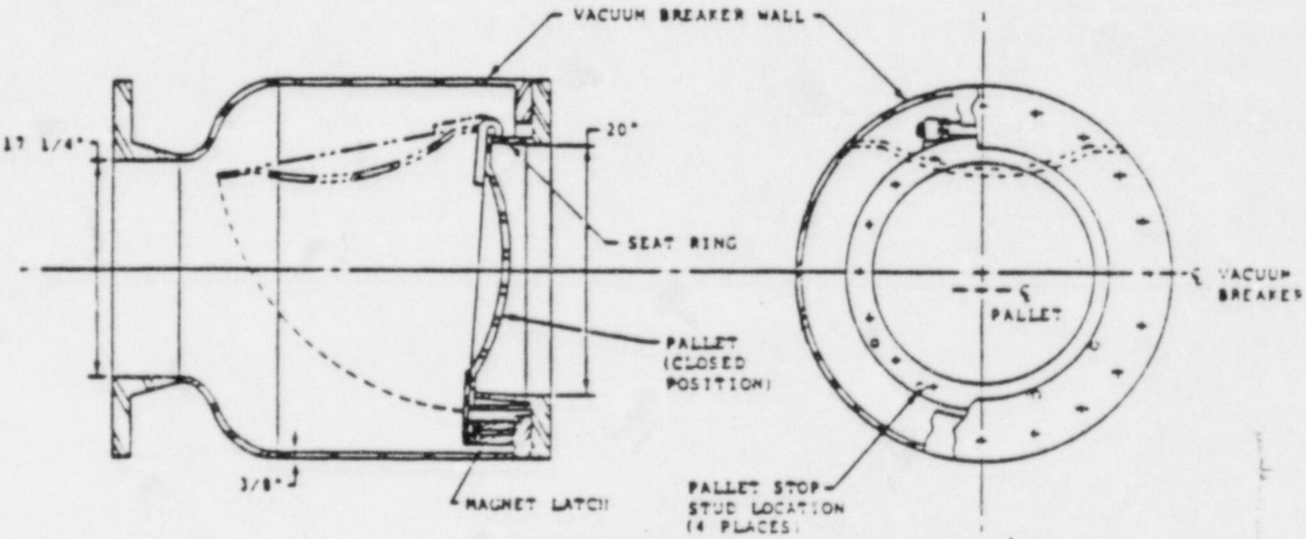

Figure 3. GPE Vacuum Breaker

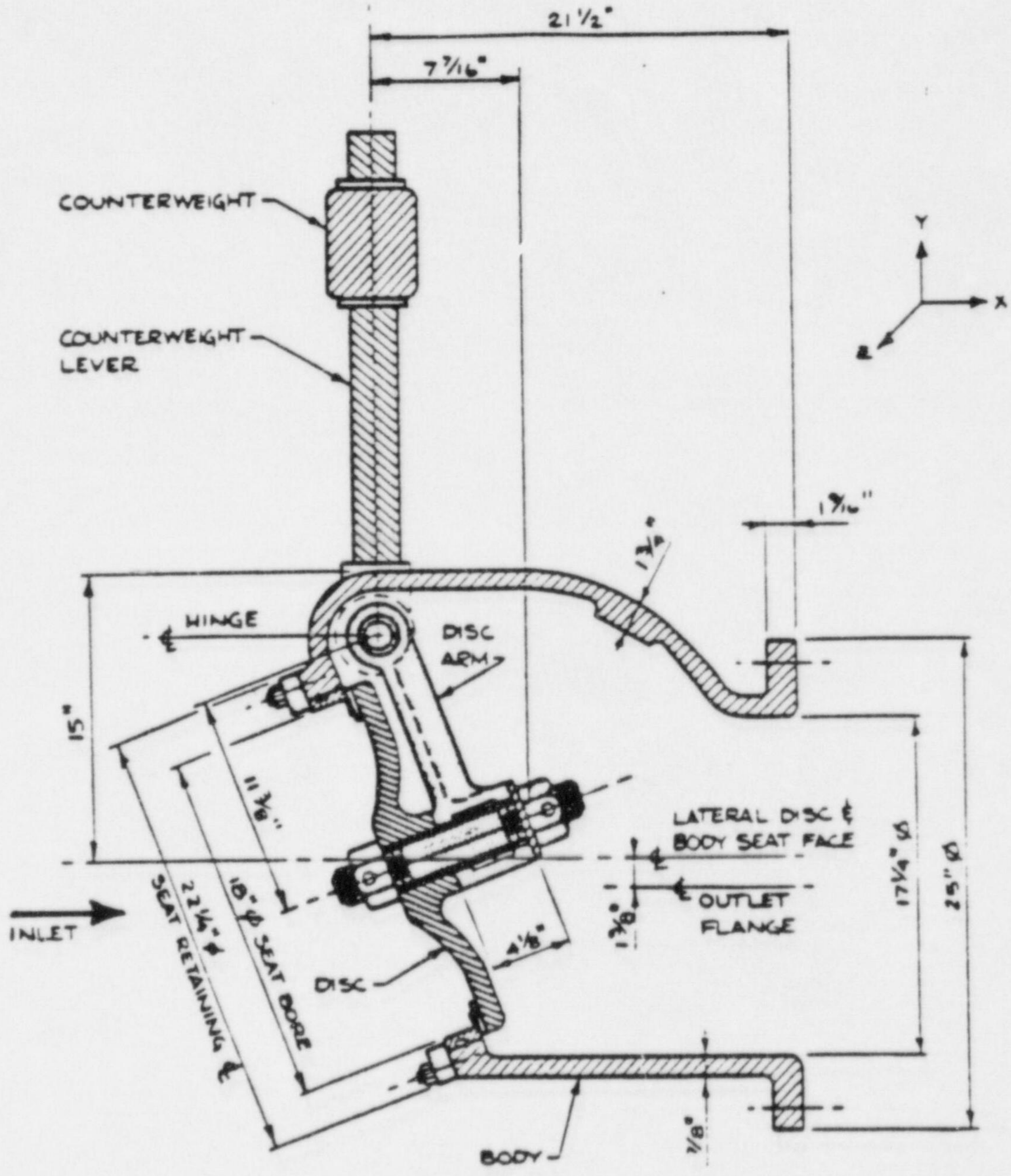

Figure 4. Atwood-Morrill Vacuum Breaker

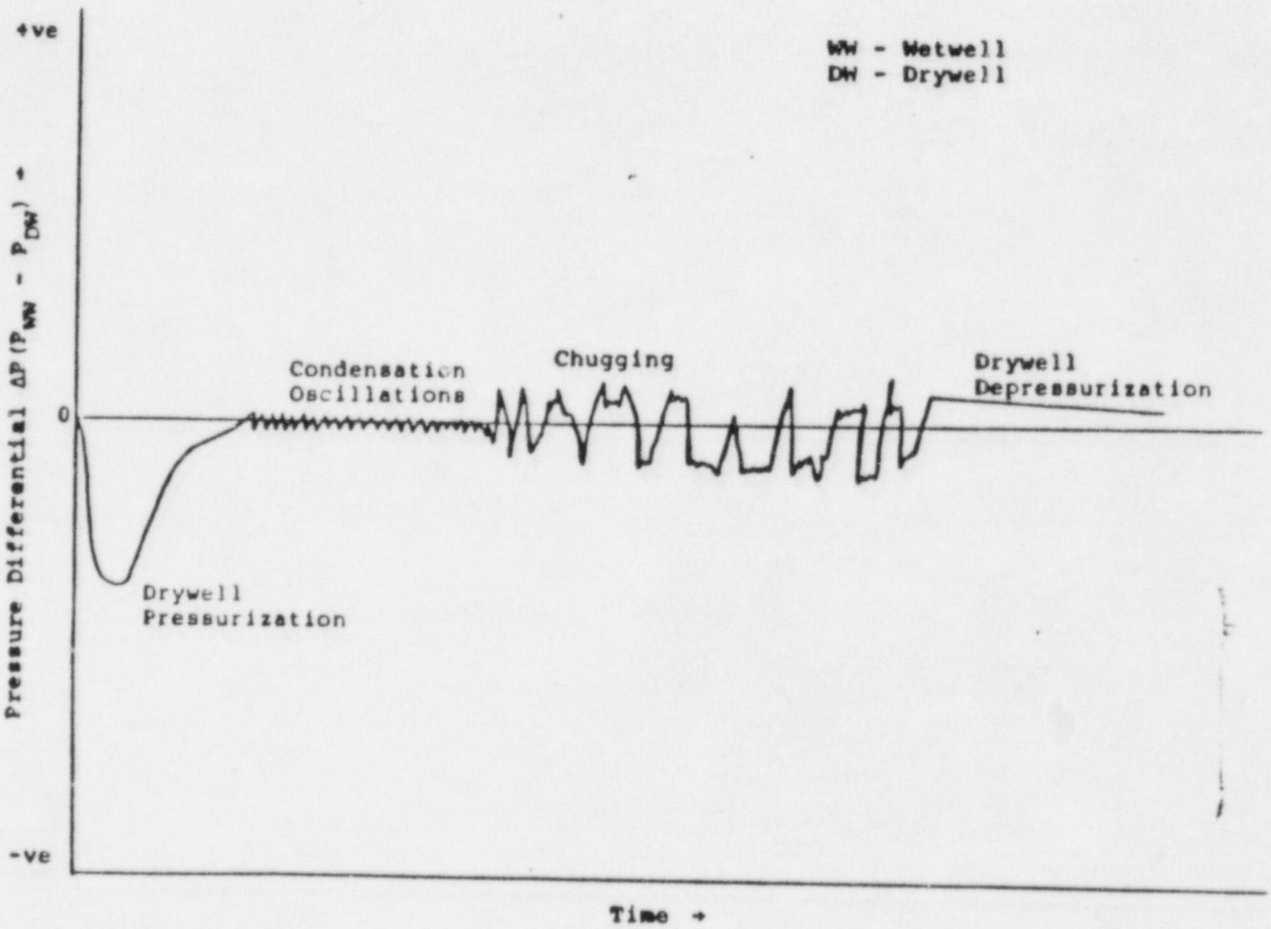

Figure 5. Typical DW/WW Vacuum Breaker Pressure Differential Due to LOCA

Table 1. Vacuum Breaker Types and Affected Plants

Vacuum Breaker	Plant
GPE 18 In (Internal)	Brown Ferry Units 1, 2, and 3 Pilgrim Unit 1 Brunswick Units 1 and 2 Cooper Hatch Units 1 and 2 Peach Bottom Units 2 and 3 Duane Arnold Fermi Unit 2
GPE 24 in (Internal)	Hope Creek
AM 18 in (Internal)	Monticello Quad Cities Units 1 and 2
AM 18 in (External)	Dresden Units 2 and 3 Millstone Unit 1 Oyster Creek Vermont Yankee
AM 18 in (External)	FitzPatrick Nine Mile Point Unit 1

2. EVALUATION CRITERIA

To evaluate the design of the vacuum breakers, the affected licensees follow the general requirements of NUREG-0661 [1] and those of "Mark I Containment Program Structural Acceptance Criteria Plant Unique Analysis Application Guide" [3]. Specifically, the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NC for Class 2 Components, 1977 Edition, including the summer 1977 addenda [4], have been used to evaluate the structural integrity of the vacuum breakers.

3. DESIGN LOADS

The loads acting on the Mark I structures and on the vacuum breaker are based upon the Mark I Program Load Definition Report [5] and the NRC Acceptance Criteria [1]. The loads acting on the vacuum breaker include gravity, seismic, and hydrodynamic loads. The hydrodynamic forcing functions were developed by Continuum Dynamics, Inc, (CDI). CDI used a dynamic model of a Mark I pressure suppression system, which was capable of predicting pressure transients at specified locations in the vent system. With this dynamic model and the full-scale test facility data, load definition resulting in pressure differential across the vacuum breaker disc was quantified as a function of time. This issue has been reviewed and addressed by the NRC [6].

4. STRESS EVALUATION

To determine structural integrity of the vacuum breaker, the licensees have employed standard analytical techniques, including the finite element method, to calculate stresses of critical components of the vacuum breaker under various design loadings. Loads resulting from the hydrodynamic phenomenon were compared with those values specified in the ASME Codes [4].

For illustration purposes, a schematic drawing of the moving parts of all components other than the actual disc of the Atwood-Morrill valve and of the corresponding finite element model are shown in Figures 6 and 7, respectively. The model in Figure 7 was used to investigate the dynamic response following impact.

A typical model for stress analysis of the vacuum breaker disc is shown in Figure 8. Loading inputs to this model are the displacement time histories that were obtained from the impact model analysis.

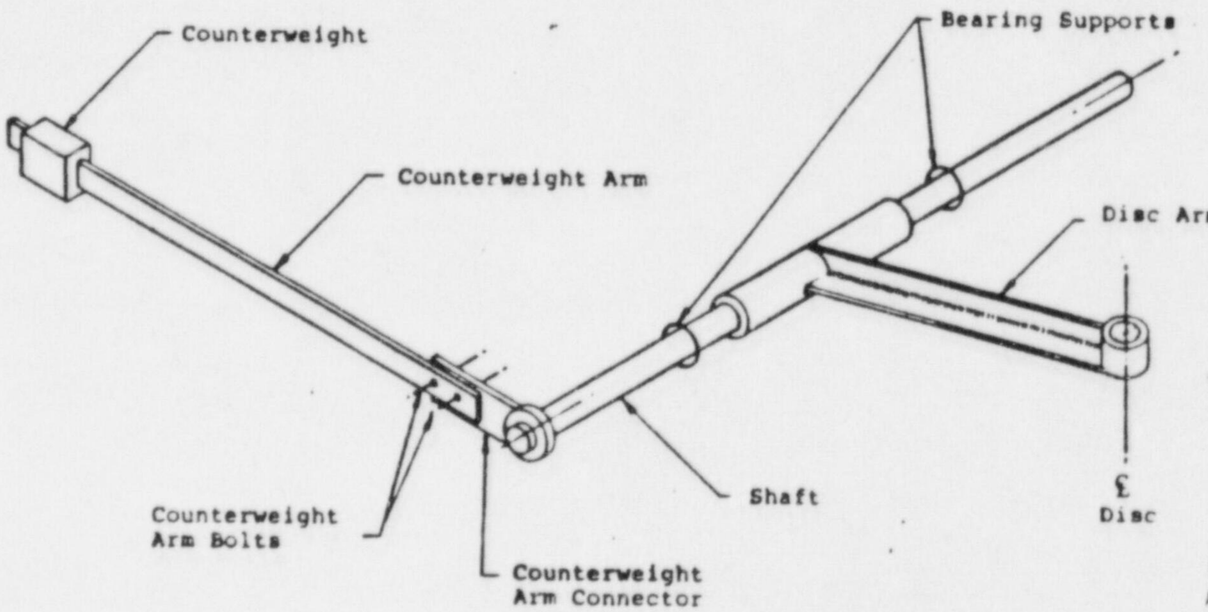

Figure 6. Detailed Valve Internal Model

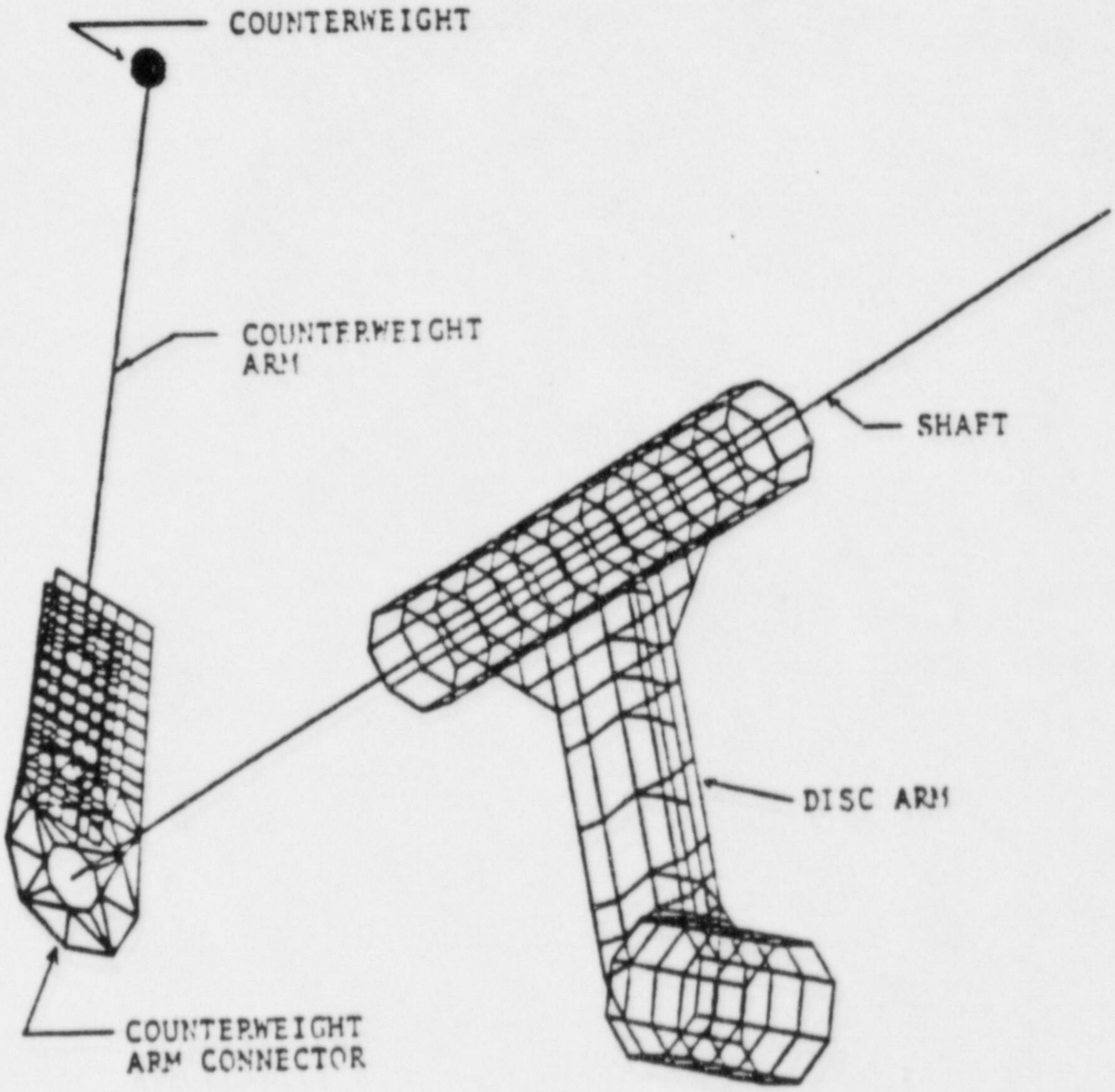

Figure 7. Finite Element Model of Valve Internals

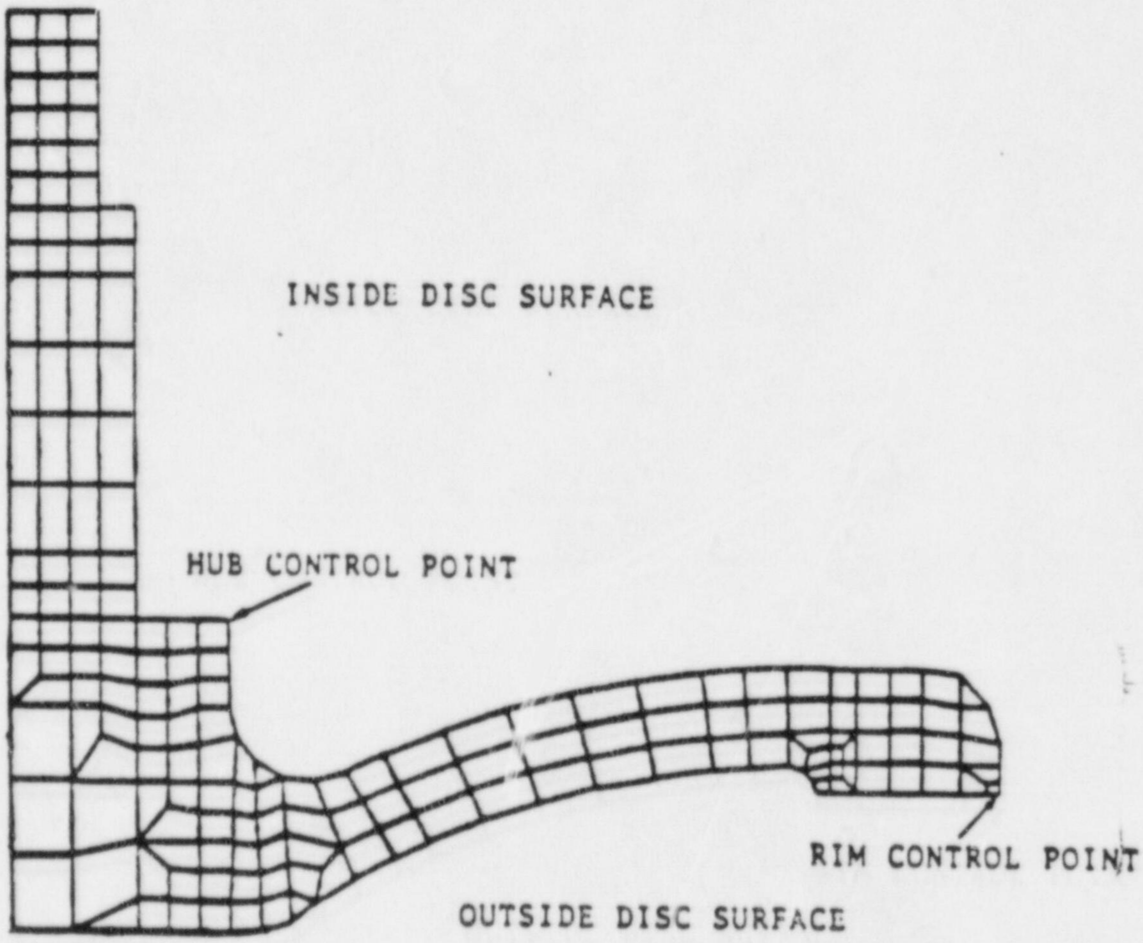

Figure 8. Valve Detailed Disc Model Geometry

5. PLANT-SPECIFIC REVIEW: COOPER NUCLEAR STATION

5.1 BACKGROUND INFORMATION

- o Vacuum breaker type: 18-in GPE (internal)
- o There are 12 vacuum breakers within the wetwell.
- o Vacuum breakers are located on the main vent/vent header intersection: two on each of six intersections.

5.2 STRESS ANALYSIS RESULTS

Stresses in critical vacuum breaker components were analyzed using an ANSYS finite element model. The pallet, hinge shaft, and hinge arm were analyzed for hydrodynamic loading due to the chugging transient, including pallet impact loads based on pallet impact velocities determined in Reference 7. Stresses in the hinge arm studs and valve seat bolts were based on the response of the pallet and hinge arm to the induced loadings. Table 2 provides a summary of the critical streses in the vacuum breaker components [8]. Allowable stresses were based on References 9 and 10. It is evident from Table 2 that vacuum breaker stresses were within allowable limits. Therefore, no modifications were necessary.

Table 2. Critical Stresses in Vacuum Breaker Components

	Stresses (psi)				
	Pallet	Hinge Arm	Hinge Shaft	Hinge Arm Studs	Pallet Seat Bolts
Material	SA-516 Gr 70	SA-516 Gr 70	SA-479 MX19	SA-516 Gr 70	SA-320 B8
Allowable S_h*	17,500	17,500	25,000	17,500	15,000
Service Level A	22,260	15,549	26,959	14,220	15,403
Allowable (1.5 x S_h)*	26,250	26,250	37,500	26,250	22,500
Stress Ratio	0.85	0.59	0.72	0.54	0.68
Service Level B	22,260	15,632	26,155	14,371	15,403
Allowable (1.65 x S_h)*	28,875	28,875	41,250	28,875	24,750
Stress Ratio	0.77	0.54	0.66	0.50	0.62
Service Level C	22,260	15,714	27,351	14,522	15,403
Allowable (1.8 x S_h)*	31,500	31,500	45,000	31,500	27,000
Stress Ratio	0.71	0.50	0.61	0.46	0.57

*Allowable stresses are based on References 9 and 10.

6. CONCLUSIONS

A review has been conducted to determine the structural integrity of the vacuum breakers at the Cooper plant. The design loads associated with the hydrodynamic phenomena have been reviewed and addressed by the NRC in Reference 6. This review covered only the structural analysis of the vacuum breaker, and the following conclusion is drawn from the review:

- The analytical methods used to evaluate stresses of critical components have been reviewed and judged to be adequate; the stress results are within the allowables as shown in Table 2 and, therefore, the existing design is structurally adequate.

7. REFERENCES

1. NUREG-0661
"Safety Evaluation Report, Mark I Containment Long-Term Program Resolution of Generic Technical Activity A-7," Office of Nuclear Reactor Regulation, USNRC
July 1980

2. D. G. Eisenhut
"USNRC Generic Letter 83-80, Modification of Vacuum Breakers on Mark I Containment"
February 2, 1983

3. NEDO-24583-1
"Mark I Containment Program Structural Acceptance Criteria Plant Unique Analysis Application Guide," General Electric Co., San Jose, CA
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4. American Society of Mechanical Engineers
Boiler and Pressure Vessel Code, Section III, Division 1, "Nuclear Power Plant Components," New York, 1977 Edition and Addenda up to Summer 1977

5. NEDO-21888 Revision 2
"Mark I Containment Program Load Definition Report," General Electric Co., San Jose, CA
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6. D. B. Vassallo, NRC
Letter with Attachment to H. C. Pfefferlen, BWR Licensing Programs, GE
"Evaluation of Model for Predicting Drywell to Wetwell Vacuum Breaker Valve Dynamics"
December 24, 1984

7. "Improved Dynamic Vacuum Breaker Valve Response for the Cooper Nuclear Station, Revision 0," C.D.I. Technical Note 82-31
Continuum Dynamics, Inc., Princeton, New Jersey
September 1982

8. J.M. Pilant
Letter with Attachment to D. B. Vassallo (NRC)
Subject: Mark I Containment Program Plant Unique Analysis Report
Nebraska Public Power District
April 29, 1983

9. ASME Code Case N-62-2 (1621-2), "Internal and External Valve Items, Section III, Division I, Class I, 2, and 3 Line Valves," Approved May 15, 1980

10. ASME Code, Section III, Subsection NC-3500, up to and including the Summer 1977 Addenda