NUREG-0474

A TECHNICAL UPDATE ON PRESSURE SUPPRESSION TYPE CONTAINMENTS IN USE IN U.S. LIGHT WATER REACTOR NUCLEAR POWER PLANTS

July 1978

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Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission

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Division of Systems Safety
Division of Operating Reactors
Office of Nuclear Reactor Regulation

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

EXECUTIVE SUMMARY

In 1972, Dr. S. H. Hanauer (Technical Advisor to the NRC's Executive Director for Operations) wrote a memorandum that raised several questions on the viability of pressure suppression containment concepts. The concerns raised by Dr. Hanauer have recently become the subject of considerable discussion by several members of the U. S. Congress and public. This report provides a response to these expressed concerns and a status summary for various technical matters that relate to the safety of pressure suppression type containments for light water cooled reactor plants.

Pressure suppression type containments utilize either large masses of water or ice as the principal heat sink to condense the steam and absorb the energy that might be released from a light water cooled reactor plant in the unlikely event of a postulated failure of certain pipes in the primary system. The absorption of energy by the stored water or ice reduces the potential buildup of pressure inside the containment which in turn reduces the driving force against the release of fission products to the environment that may have been released from the reactor core.

The concept of pressure suppression with water was first developed by the General Electric Company (GE) for the Humboldt Bay Nuclear Plant in the period 1958-1962. Since that time, GE has designed many BWR plants, including three distinctively different pressure suppression contain-

ment designs. The first generation of BWR containment designs was designated as the Mark I containment system. The Mark I design consists of a drywell shaped like an inverted light bulb and connected by a system of vent pipes to a toroidal shaped wetwell. Twenty-two facilities using the Mark I containment system have been licensed for power operation and three additional facilities are currently under construction.

The second generation of BWR containment designs was designed as the Mark II containment system. The Mark II design consists of a drywell shaped like a truncated cone separated from a cylindrical wetwell by a floor containing a number of straight and vertical vent pipes.

Facilities using the Mark II containment system have not yet been licensed for power operation; however, eleven such facilities are currently under construction. Based on current construction and licensed review schedules, licensed operation of the first BWR with the Mark II containment system is not anticipated before mid-1979.

The third and most recent generation of BWR containment designs was designated as the Mark III containment system. The Mark III design consists of a drywell volume surrounded by a cylindrical wetwell. The drywell volume is connected to the wetwell through the suppression pool by a weir annulus and a series of straight and horizontal vents located near the base of the drywell. Facilities using the Mark III containment system have not yet been licensed for power operation; however, twelve such

facilities are currently under construction and approximately fourteen additional facilities have been proposed. Based on current construction and licensing review schedules, licensed operation of the first BWR with the Mark III containment system is not anticipated before late 1980.

The introduction of the containment concept whereby large quantities of ice are used to achieve rapid steam condensation in the event of a LOCA came about during the 1970's and was developed by the Westinghouse Electric Corporation for application to PWR plant types. The D.C. Cook, Units 1 and 2 station is the first operational nuclear plant to utilize this type of containment.

The various pressure suppression containment designs (both water and ice) are discussed below including their a) design features; b) safety issues that have been identified with each design concepts; c) applicable safety criteria and procedures for assuring that each containment related safety issue has been addressed and conservatively resolved; and d) specific containment-related technical programs in the U. S. and abroad that have added significantly to the data base needed to confirm the adequacy of these designs.

A. BWR PRESSURE SUPPRESSION CONTAINMENTS

As discussed previously the development of the pressure suppression containment technology for boiling water reactor plants began with design of the Humboldt Bay Nuclear Power Plant, which was licensed for operation on August 28, 1962. Scaled tests conducted in 1959 demonstrated the basic feasibility of the pressure suppression containment and provided

data for use in initiating the design. In 1960, full scale tests of the Humboldt Bay design were conducted at Moss Landing. The objectives of the program were to demonstrate the condensation capability of the water pressure suppression containment concept and to verify the adequacy of the calculational techniques used in the analytical model for predicting the containment pressure and temperature response.

In 1962, a similar test facility was constructed at the Moss Landing Plant to proof test the Bodega Bay containment design. The facility consisted of a full scale model of the Bodega Bay containment which became the prototype for the Mark I generation of containments for BWR plants. Some pool swell hydrodynamic affects were observed in the tests as discontinuities in the pressure response measured in the wetwell airspace.

In early 1970, the most recent generation of pressure suppression containment design, known as the Mark III design was developed. As in the case of the Mark I and II designs, small scale tests were conducted in 1971 to provide a data base for the development of analytical models.

A large scale test program was initiated in the fall of 1973 to confirm the analytical models used in development of the Mark III design. In the first test series, it was observed that wetwell internal structures located above the suppression pool could be subjected to significant hydrodynamic loads during the pool swell process. Additional tests were conducted to determine the loads for the design of the affected structures.

An evaluation of the Mark III pool swell data indicated the need for a reassessment of both Mark I and Mark II containment designs. These facilities had not explicitly considered the pool hydrodynamic loads associated with a postulated loss-of-coolant accident in their designs. The affected utilities formed Mark I and II "ad hoc" owners groups in 1975 with the General Electric Company (GE) acting as program manager. A series of test programs aimed toward a better definition of the pool dynamic loads associated with each design followed.

These testing programs formed the basis for the Mark I Owners Groups Short Term Program (STP). The objective was to demonstrate that the licensed Mark I BWR facilities could continue to operate without undue risk to the health and safety of the public, during an interim period of approximately two years while a methodical and comprehensive Long Term Program (LTP) is conducted. The NRC has reviewed the results of the STP and concurs with the conclusion made by Mark I owners. In the owners group's LTP, the key element is a full scale testing program (FSTP) which will investigate the hydrodynamic loads and dynamic structural response from chugging phenomena on a representative torus sector.

In addition to the owners group's program, the NRC has funded a major confirmatory experimental research program at Lawrence Livermore Laboratory (LLL). The program involves the construction and operation of a 1/5 scale, 90° sector of a typical Mark I BWR containment system.

The purpose of this program was to obtain data regarding the magnitude and character of hydrodynamic LOCA-related air clearing loads on the Mark I containment system in order to confirm the results obtained from the testing programs sponsored by the Mark I Owner's Groups.

Preliminary indications from the LLL work have revealed that the anticipated load reductions due to three dimensional effects may not be realized. Early comparisons of two-dimensional data with three dimensional data show slightly higher equivalent loads for the three dimensional sector. Final evaluation of this phenomenon as well as all other aspects of the LOCA pool dynamic loads will be conducted as part of the staff's generic Task Action Plan A-7. A similar testing program plan was established by the Mark II Owners Group in early 1975 to determine the LOCA related pool dynamic loads for Mark II containment designs. The program plan consists of pool dynamic tests involving air clearing and steam chugging tests.

Until the early 1970's, the only significant hydrodynamic loads considered in the containment design were those with a postulated LOCA. During this time period, safety relief valve induced loads were considered to be small and secondary. Safety relief valve (SRV) discharges were experienced during the 1960's. They did not cause any serious damage to any of the structures, although there was some damage to the SRV line supports. However, in 1972 two German BWR plants with

water suppression containments experienced severe vibratory loads on the containment structure during extended SRV operation.

Following these incidents, extensive experiments were conducted to investigate various SRV downcomer discharge configurations. The objective of the investigation was to develop a discharge device which would reduce the hydrodynamic loads during SRV line air clearing and provide stable steam condensation.

The General Electric Company began developing an analytical model to predict the loads associated with the steam vent clearing phenomenon during SRV discharges. To support the model predictions, additional in-plant tests were conducted. Further efforts have led to the development of a new type of discharge device to reduce the suppression pool loads.

B. PWR ICE CONDENSER PRESSURE SUPPRESSION CONTAINMENT

During 1967 and 1968, the Westinghouse Electric Corporation conducted an intensive design and testing effort to demonstrate the feasibility of the ice condenser containment concept as applied to a nuclear power plant containment. The feasibility of the concept was demonstrated by the full-scale ice condenser tests conducted at the Westinghouse Waltz Mill test facility.

After these early full-scale demonstration tests, a number of design changes were made to ice condenser components to improve the structural

capability of some components. Moreover the need for improvements in fabrication and installation methods became apparent during the detailed design of the ice condenser plants. Westinghouse conducted additional test programs in 1973 and 1974 to qualify the ice condenser containment design as currently designed for use in a nuclear power plant.

In addition, operating experience with the ice condenser at the D.C. Cook Nuclear Plant Unit 1 has indicated the need to alter the design of a number of ice condenser components and certain equipment operating and maintenance procedures such as: (1) minor modifications to the ice condenser air handling unit components, change in control temperatures of the chiller packages and increased maintenance frequency for the air handling equipment to improve the ice condenser cooling performance, and; 2) minor modifications to the ice condenser lower inlet door assemblies and personnel access doors to reduce cold air leakage from the ice condenser. These design improvements, developed by the operating experience at the D. C. Cook plants, are being carried forward into the later ice condenser plants, e.g., McGuire and Sequoyah.

C. MAJOR REVIEW AREAS FOR PRESSURE SUPPRESSION CONTAINMENTS

The major review areas during licensing reviews of pressure suppression containments include:

1) containment pressure and temperature response;

2) subcompartment pressure analysis;

3) steam bypass;

4) suppression pool hydrodynamic loads (which includes LOCA pool dynamics and safety relief valve loads);

5) containment heat removal systems;

6) containment isolation system;

7) combustible gas control;

8) containment leak testing; and

9) ice maintenance for the ice condenser system.

D. TECHNICAL BASES FOR LICENSING OF PRESSURE SUPPPESSION CONTAINMENTS

The detailed acceptance criteria and review procedures used by the staff in reviewing BWR containment pressure response are found in Section 6 of the staff's Standard Review Plan, NUREG-75/087.

A discussion of the review procedures for verification of the acceptance criteria is given in Section IV of the report.

E. ONGOING PRESSURE SUPPRESSION CONTAINMENT PROGRAMS

The NRC staff has a number of ongoing programs which are designed to; 1) reconfirm the adequacy of the various aspects of the pressure suppression type of containments; and 2) identify areas of the designs that can or should be improved.

These ongoing efforts include: 1) The staff's Systematic Evaluation Program ; 2) The staff's Technical Activities Program; 3) The staff's research programs and its cooperative research programs with foreign agencies; 4) The staff's reviews of operating experience obtained from currently operating U. S. and foreign nuclear power plants; and 5) review of GE's assumption for its BWR-NSSS equipment adequacy evaluation program.

F. SUMMARY AND CONCLUSIONS

The objective of the NRC staff's review of the applicant's Safety Analysis Reports is to ensure that the proposed designs have adequate safety margins to protect the public health and safety. The process whereby the staff performs its safety review in the area of containment design is described in Section 6 of NUREG-75/087, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," September 1975 (currently being revised). The staff also has technical assistance programs in effect at various national laboratories to further increase its review capability in the area of containment design matters.

In addition to the efforts now underway in the United States with regard to containment design aspects, a number of tests programs are underway in foreign countries. These include tests in Sweden, Germany, Italy, and Japan.

In addition to the reviews by the staff and its consultants of the technical aspects of the pressure suppression type of containments, the Advisory Committee on Reactor Safeguards and its consultants also perform in-depth reviews both on a case basis and in connection with the generic programs.

The technical concerns identified by Dr. Hanauer in his 1972 memorandum are discussed in detail in the report, along with all the other major review areas. It is shown that the staff has given proper consideration to each of these concerns. In addition, as new safety concerns are identified, the staff gives careful consideration to their significance in ensuring the health and safety of the general public.

The NRC has completed its review of the generic Mark I containment Short Term Program (STP) conducted by the Mark I Owners Group and the associated plant-unique information provided by the licensees of operating Mark I BWR facilities.

Based upon its review, the NRC has concluded that licensed Mark I BWR facilities can continue to operate safely, pending completion of the comprehensive LTP evaluation. However, the NRC further concluded that the demonstrated safety margin of the containment systems for operating Mark I BWR facilities does not comply fully with the current interpretation of "sufficient margin" as prescribed in General Design Criterion 50, of Appendix A to 10 CFR Part 50, and, therefore, should be improved for long term reactor operation.

On February 28, 1978 the NRC granted the licensees of operating Mark I BWR facilities exemptions from GDC-50, with respect to LOCA-related hydrodynamic suppression pool loads, for an interim period until completion of the LTP (approximately two years). These exemptions provide for continued operation under the conditions specified in NUREG-0408 and under any resulting Technical Specification requirements. The generic Mark I LTP program commenced in June 1976 and is scheduled for completion in 1979.

Facilities utilizing the Mark II containment system design are currently under construction. Although there are several applications for an operating license pending, there are no domestic Mark II facilities in operation. In April 1975, it was determined by the staff that a complete reassessment of the Mark II facilities should be conducted. At that time, the staff sent letters to the owners of each domestic facility with a Mark II containment requesting that they provide information demonstrating the adequacy of their containment. As a result, the Mark II containment owners, formed an "ad-hoc" Owners Group to respond to the NRC requests. Utilities are making modifications to their Mark II plants on the basis of information already derived from the ongoing Mark II pool dynamic programs.

We consider the basic design and performance of the Mark III containment system to be well established based on our review of the analytical models and the available margins incorporated in the design. It is our view that those phenomena that are being addressed

in the future Mark III tests merit additional evaluation, but that they do not represent design governing conditions at this time. Nor, in our judgement, will they escalate into new design basis considerations as a result of these tests. In summary, we consider the remaining Mark III testing to be confirmatory nature and will require that the tests and our evaluation of the test results be completed prior to our issuance of the first operating license for a Mark III plant.

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A TECHNICAL UPDATE ON

PRESSURE SUPPRESSION TYPE CONTAINMENTS

IN USE IN U.S. LIGHT WATER REACTOR NUCLEAR POWER PLANTS

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I. GENERAL

A. Introduction

The primary reactor system of nuclear power plants is enclosed in a large structure called the containment. The purpose of the containment is to contain the radiological materials that would be released from the high energy fluids in the primary reactor system and reactor core in the unlikely event of a postulated rupture of the primary system piping (i.e., a loss-of-coolant accident [LOCA]) thereby mitigating the consequences of the accident. The containment structure and associated filtration systems serve as a low leakage barrier against the release of radiological materials to the environment in the unlikely event of a LOCA and is considered to be the final barrier in the defense-in-depth approach in the design of nuclear power plants.

The containment system designs generally utilized in commercial nuclear power plants are generally provided with a containment that is either a dry type or a pressure suppression type. The dry containment is a large volume structure designed to absorb the energy released from a LOCA and to maintain a low leakage barrier against release of fission products to the environment.

Pressure suppression containments are designed to reduce the buildup of steam pressure inside the containment following a LOCA by condensing the steam by passing it through either a large pool of water or a large ice-filled chamber.

The concept of pressure suppression with water was first developed by the General Electric Company (GE) for the Humboldt Bay Nuclear plant in the period 1958-1962. Since that time, GE has designed many BWR plants and has developed three distinctively different pressure suppression containment designs i.e., the Mark I, II, and III designs.

The ice-containment concept in which large quantities of ice are used to achieve the rapid steam condensation in the event of a LOCA was developed during the 1970's by the Westinghouse Electric Corporation for application to PWR plant types. The D.C. Cook, Units 1 and 2, nuclear plant is the first operational nuclear plant to utilize this type of containment.

The purpose of this report is to trace the evolution of these various pressure suppression containment designs (both water and ice) and to describe in detail: (a) the safety issues that

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have been identified with each design concept; (b) the applicable safety criteria and procedures for assuring that each containment related safety issue has been addressed and conservatively resolved,; and (c) specific containment related technical programs in the U.S. and abroad that have significantly added to the data base needed to confirm the efficacy of these designs. Extensive referencing of NUREG reports, Regulatory Guides and industry reports is used in these discussions in order that the detailed technical supporting analyses can be readily located.

In addition to the aforementioned objectives, this report also addresses in Enclosure A each of the technical concerns that were stated in Dr. S. H. Hanauer's (Technical Advisor to the Executive Director For Operations) September 20, 1972 memorandum.[1/*] Dr. Hanauer's concerns, which questioned the viability of pressure suppression containment systems design, have recently been the subject of considerable discussions culminating in a number of inquiries by members of the U.S. Congress and members of the public that the NRC demonstrate, with particular emphasis on Dr. Hanauer's concerns, that

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*/ References are denoted by superscripts and are listed by number at the end of each major section to this report.

pressure suppression containments will indeed protect the health and safety of the public in the event of a LOCA. A recent memorandum, dated June 20, 1978, was issued by Dr. Hanauer on this same subject that reflects his current views on pressure suppression type containments.[2/]

B. Status and Description of Pressure Suppression Designs

WATER PRESSURE SUPPRESSION - The basis for defining the containment design parameters is the assumed instantaneous rupture of the largest coolant pipe in the reactor primary coolant system. In a pressure suppression containment, the steam released due to this postulated accident would cause a rise in the pressure in the containment or drywell causing steam to flow through the vents into the water pressure suppression chamber. There are about 2500-5000 tons of water in the suppression pool depending upon the plant type. The steam is condensed in the suppression pool water, and the entrained non-condensibles, mainly air, bubble to the surface and collect in the suppression chamber air space. After the steam blowdown from the

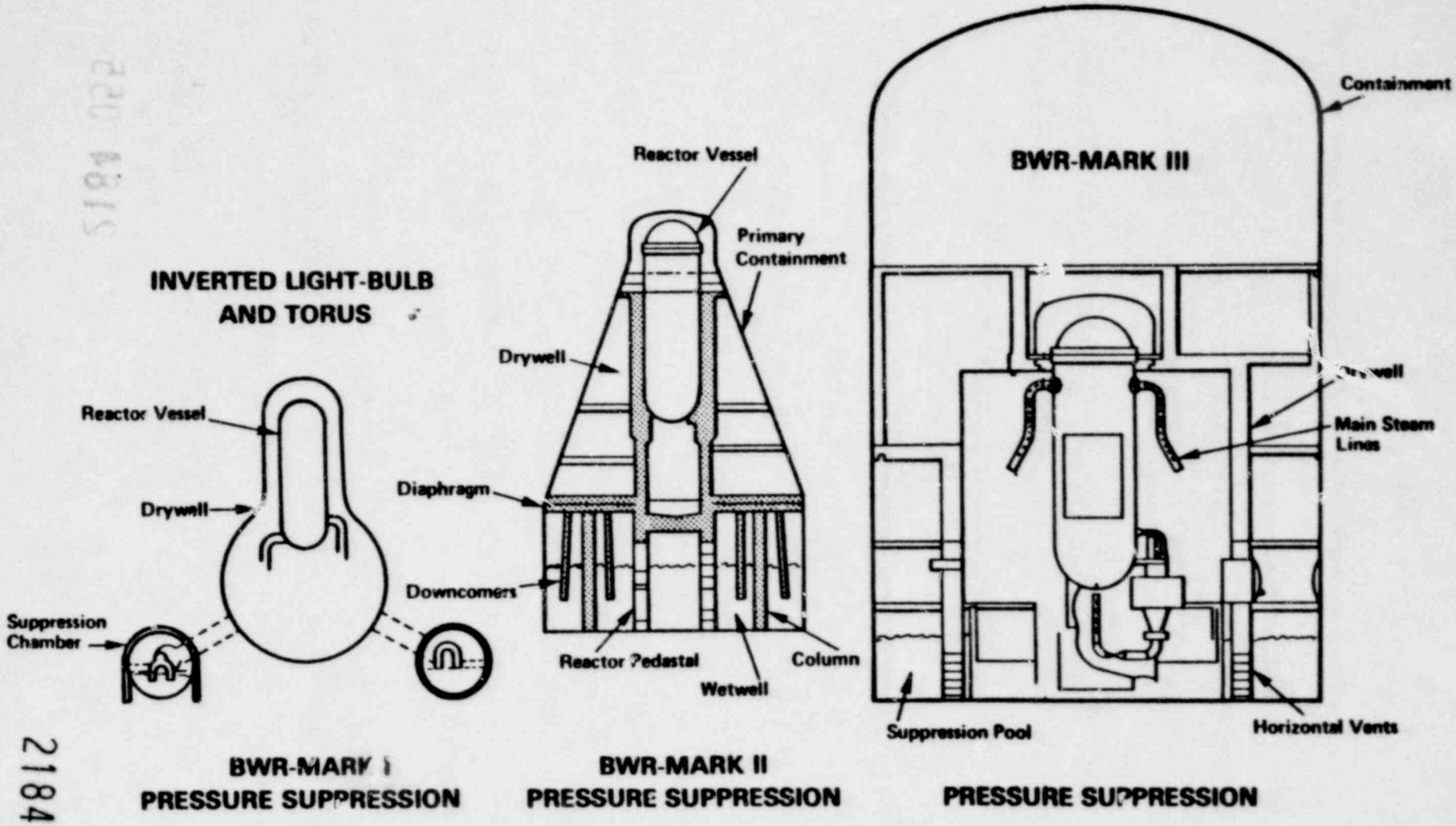

Figure I.1

ruptured pipe ceases, flow through the vents diminishes, and the drywell and suppression chamber pressure reach an equilibrium state.

Figure I.1 shows the general configuration and design features of the three types of BWR pressure-suppression containment designs. The first generation of BWR containments designs was designated as the Mark I containment system. The Mark I design consists of a drywell shaped like an inverted light bulb connected by a vent system to a toroidal shaped wetwell. Twenty-two facilities using the Mark I containment system have been licensed for power operation and three additional facilities are currently under construction in the United States.

The second generation of BWR containment designs was designated as the Mark II containment system. The Mark II design consists of a drywell shaped like a truncated cone separated from a cylindrical wetwell by a floor containing a number of vertical vent pipes.

Facilities using the Mark II containment system have not yet been licensed for power operation in the U.S.; however, eleven such

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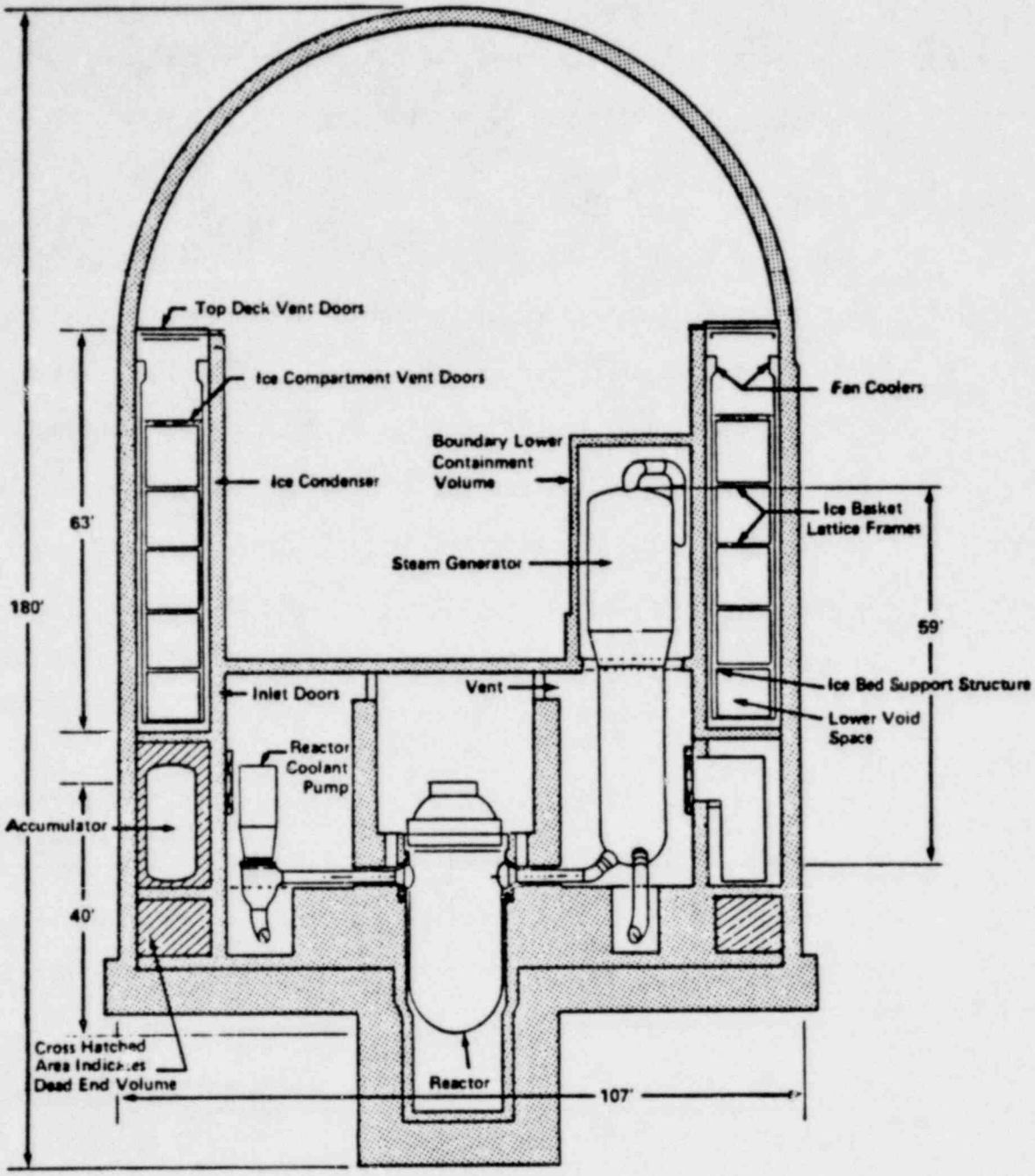
facilities are currently under construction. Based on current construction and licensing review schedules, licensed operation of the first BWR with the Mark II containment system is not anticipated before mid-1979.

The third, and most recent generation of BWR containment designs is designated as the Mark III containment system. The Mark III design consists of a drywell volume surrounded near its base by a cylindrical wetwell. The drywell volume is connected to the wetwell through the suppression pool by a weir annulus and a series of horizontal vents located near the base of the drywell. Facilities using the Mark III containment system have not yet been licensed for power operation; however, twelve such facilities are currently under construction and approximately fourteen additional facilities have been proposed. Based on current construction and licensing review schedules, licensed operation of the first BWR with the Mark III containment system in the U.S. is not anticipated before late 1980.

ICE CONDENSER PRESSURE SUPPRESSION - Ice condenser pressure suppression containments (see Figure I.2) are designed so that the

steam released from a postulated rupture of the reactor primary coolant system is rapidly condensed by being brought into intimate contact with large quantities of ice. The ice condenser structure is a vertical cylinder divided by structural barriers into three major subvolumes; a lower compartment enclosing the reactor system, an intermediate volume housing an ice bed in which the steam is condensed during the LOCA, and an upper compartment to accommodate the air displaced from the other two compartments during the LOCA. The ice condenser compartment, which performs the same function as the BWR wetwell, is an enclosed annular volume encompassing most of the perimeter of the intermediate portion of the containment structure and contains 120 tons of ice, which is continually refrigerated. During a LOCA, the rising pressure in the reactor or lower compartment pushes open the inlet doors located at the bottom of the ice condenser compartment providing a direct path into the ice beds. At present, there are two nuclear power plants in operation utilizing this design, while sixteen others are in various stages of construction.

A summary of pertinent data regarding pressure suppression plants is included in Tables I.1 and I.2. The cumulative reactor years



ICE CONDENSER CONTAINMENT

Figure I.2

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for plants using pressure suppression containments is 130 reactor years out of a total of 350 reactor years of operation.

C. Summary and Conclusions

Appendix A to 10 CFR Part 50 contains the General Design Criteria that establish minimum requirements for the principal design criteria for water-cooled nuclear power plants. The specific criteria that apply to the containment design are given in Table I.3. The NRC staff reviews the applicant's Safety Analysis Report to ensure that the proposed design meets the Commission's regulations or that an acceptable alternative (i.e., an exception) is provided. In either case, the staff's safety review ensures that the design is adequate to assure the public health and safety.

The process used by the staff in performing its safety review of containment designs for current plants is described in Section 6 of NUREG-75/087, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," September 1975 (currently being revised). Each principal design feature is reviewed by the staff according to the review guidelines specified in the review plan.

This report specifically addresses the particular containment type classified as pressure suppression containments as used in boiling water reactor plants as well as in some pressurized water reactor plants. The boiling water reactor plants use water suppression whereas the pressurized water reactor plants use ice suppression. Each technical review area is identified; and for each area the technical basis for finding the design under consideration acceptable is provided.

The staff has contracted for technical assistance programs at various DOE national laboratories to further increase its review capability and provide additional expert assistance in its reviews of containment design matters. Such programs are underway at the Brookhaven National Laboratories in New York, including special consultants at the Massachusetts Institute of Technology (MIT) and Princeton University; at the Los Alamos Laboratories in New Mexico, and the Idaho Nuclear Engineering Laboratories in Idaho. In particular, Brookhaven along with the MIT and Princeton consultants have been heavily involved in such matters as pool dynamics and safety relief valve loads in BWR plant reviews.

There are a number of generic safety issues directly related to pressure suppression containment designs. These are identified and discussed in the NRC program for generic resolution of safety issues.[3/]

In addition to the reviews by the staff and its consultants of the technical aspects of the pressure suppression types of containments, the NRC Advisory Committee on Reactor Safeguards including its own consultants, also performs in-depth reviews both on individual applications and on a generic basis. The staff has met with the Committee on many occasions to discuss containment-related matters. The Committee has reported favorably on the safety of pressure suppression containments.[4,5/]

In addition to the extensive review performed by the NRC staff and its consultants to ensure adequate safety, there are a number of ongoing analyses and test programs designed to further investigate and ensure the adequacy of the pressure suppression containment design. For example, as part of an ongoing test program in which the U.S. is a participant, the full-scale Marviken water pressure suppression containment located in Sweden has been subjected to a total of twenty-five non-nuclear blowdowns without failure of the containment system.[6/]

This report summarizes the major areas of safety review associated with pressure suppression containments including each of the technical concerns identified by Dr. Hanauer in his 1972 memorandum. Dr. Hanauer's concerns are addressed explicitly in Enclosure A. The staff has given consideration to each of these concerns. In addition, as new safety concerns have been identified, the staff has given careful consideration to their significance in ensuring the health and safety of the general public, for plants under construction as well as the operating reactor plants.

Recently identified concerns regarding pool dynamic loads for the BWR pressure suppression containment designs are being evaluated for all the designs. Significant actions are being implemented by the various licensees to restore the originally intended design safety margins. Based upon the reviews that have been performed, the staff concludes that the pressure suppression concept for containment design is acceptable and safe.

A corollary to this report is the recent expression of Dr. Hanauer's opinions, dated June 20, 1978,[2/] concerning the problems of pressure suppression containments where he stated, "Thus while we may yearn for the greater simplicity of "dry" containments, the problems of both "dry" and pressure suppression containments are solvable, in my opinion, and the design safe, therefore licensable."

REFERENCES

1. S. H. Hanauer to J. F. O'Leary, et. al., on Pressure Suppression Containments, dated September 20, 1972.

2. Memorandum, S. H. Hanauer to Chairman Hendrie, Pressure Suppression Containment, dated June 20, 1978.

3. NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants; NUREG-0410; Report to Congress, dated January 1, 1978.

4. ACRS Report on Ice Condenser Pressure Suppression Concept, November 9, 1967.

5. ACRS Report on Boiling Water Reactor Mark III Containment, Concept, January 17, 1973.

6. Full-Scale Reactor Safety Experiments In The Marviken Power Station Sweden; N. G. Thoren and L. Ericson; A. B. Atomenergi; Studsvik, Sweden; Paper given at IAEA Conference in Salzburg, Austria, May 1977 (IAEA-CN-36/284).

TABLE I.1

Nuclear Power Plants Using Pressure Suppression Type Containments

A. Licensed to Operate

BWR Mark I

Browns Ferry, Unit Nos. 1, 2 & 3
Brunswick, Unit Nos. 1 & 2
Cooper
Dresden Unit, Nos. 2 & 3
Duane Arnold
FitzPatrick
Hatch, Unit Nos. 1 & 2
Millstone, Unit No. 1
Monticello
Nine Mile Point, Unit No. 1
Oyster Creek
Peach Bottom, Unit Nos. 2 & 3
Pilgrim, Unit No. 1
Quad Cities, Unit Nos. 1 & 2
Vermont Yankee

BWR Mark II

None

BWR Mark III

None

PWR Ice Condenser

D.C. Cook, Unit Nos. 1 & 2

Other

Humboldt Bay

B. Construction Permits Granted

1. Under Operating License Review

BWR Mark I

Fermi, Unit No. 2

BWR Mark II

WPPSS, Unit No. 2
LaSalle, Unit Nos. 1 & 2
Shoreham
Susquehanna, Unit Nos. 1 & 2
Zimmer

BWR Mark III

Grand Gulf, Unit Nos. 1 & 2

PWR Ice Condenser

McGuire, Unit Nos. 1 & 2
Sequoyah, Unit Nos. 1 & 2
Watts Bar, Unit Nos. 1 & 2

2. Under Construction

BWR Mark I

Hope Creek, Unit Nos. 1 & 2

BWR Mark II

Bailly, Unit No. 1
Limerick, Unit Nos. 1 & 2
Nine Mile Point, Unit No. 2

BWR Mark III

Clinton, Unit Nos. 1 & 2
Hartsville, Unit Nos. 1,2,3 & 4
Perry, Unit Nos. 1 & 2
Phipps Bend, Unit Nos. 1 & 2
River Bend, Unit Nos. 1 & 2

PWR Ice Condenser

Catawba, Unit Nos. 1 & 2

C. Under Construction Permit Review

BWR Mark I

None

BWR Mark II

None

BWR Mark III

Allens Creek, Unit No. 1
Black Fox, Unit Nos. 1 & 2
Douglas Point, Unit No. 1
Montague, Unit Nos. 1 & 2
Skagit, Unit Nos. 1 & 2

PWR Ice Condenser

Floating Nuclear Plants 1 thru 8 (Manufacturing License)

NUMBER OF CONTAINMENTS

CONTAINMENT TYPE	NUMBER IN OPERATION	NUMBER PROPOSED
DRY (SINGLE)	32	52
DRY (DUAL)	7	20
SUBATMOSPHERIC	4	7
PRESSURE SUPPRESSION		
ICE CONDENSER	2	16
MARK I	22	3
MARK II	0	11
MARK III	0	22
OTHERS	2*	
TOTAL	69	131

* FT. ST. VRAIN, HUMBOLDT

TABLE I.2

TYPICAL U.S. CONTAINMENTS

CONTAINMENT TYPE	DESIGNER	MATERIAL	DESIGN PRESSURE (PSIA)	APPROX. VOLUME (FT^3)
DRY	BECHTEL, EBASCO, PIONEER, DUKE PWR. AND OTHERS	STEEL	55-62	2.5×10^6
	BECHTEL, EBASCO, AND OTHERS	CONCRETE (R)	70	2.5×10^6
	BECHTEL, EBASCO, AND OTHERS	CONCRETE (P)	70	2.0×10^6
SUBATMOSPHERIC	STONE & WEBSTER	CONCRETE (R)	60	2.3×10^6
ICE CONDENSER	WESTINGHOUSE	CONCRETE (R)	27	1.25×10^6
PRESSURE SUPPRESSION				
MARK I (DRYWELL)	GE	STEEL	74	159,000
(WETWELL)		STEEL	74	204,000
MARK II (DRYWELL)	GE	CONCRETE (R)	65	184,000
(WETWELL)		W/STEEL LINER	65	209,000
MARK III (DRYWELL)	GE	CONCRETE (R)	45	280,000
(WETWELL)		CONCRETE (R) W/STEEL LINER	30	1.5×10^6

TABLE I.3

GENERAL DESIGN CRITERIA
(10 CFR PART 50 APPENDIX A)

NO. 1	-	QUALITY STANDARDS AND RECORDS
NO. 2	-	DESIGN BASIS FOR PROTECTION AGAINST NATURAL PHENOMENA
NO. 16	-	CONTAINMENT DESIGN
NO. 38	-	CONTAINMENT HEAT REMOVAL
NO. 39	-	INSPECTION OF CONTAINMENT HEAT REMOVAL SYSTEM
NO. 40	-	TESTING OF CONTAINMENT HEAT REMOVAL SYSTEM
NO. 41	-	CONTAINMENT ATMOSPHERE CLEANUP
NO. 42	-	INSPECTION OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS
NO. 43	-	TESTING OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS
NO. 50	-	CONTAINMENT DESIGN BASIS
NO. 52	-	CAPABILITY FOR CONTAINMENT LEAKAGE RATE TESTING
NO. 53	-	PROVISIONS FOR CONTAINMENT TESTING AND INSPECTION
NO. 54	-	PIPING SYSTEMS PENETRATING CONTAINMENT
NO. 55	-	REACTOR COOLANT PRESSURE BOUNDARY PENETRATING CONTAINMENT
NO. 56	-	PRIMARY CONTAINMENT ISOLATION
NO. 57	-	CLOSED SYSTEM ISOLATION VALVES

II. HISTORICAL DEVELOPMENT OF PRESSURE SUPPRESSION CONTAINMENT TECHNOLOGY

A. BWR Pressure Suppression Containment

1. General Development Testing

The development of the pressure suppression containment technology began with design of the Humboldt Bay Nuclear Power Plant, which was licensed for operation on August 28, 1962. Scaled tests conducted in 1959 demonstrated the basic feasibility of the pressure suppression containment and provided data for use in initiating the design. In 1960, full scale [1/48 segment] tests of the Humboldt Bay containment design were conducted by GE at the Moss Landing Fossil Fuelled Plant. The objectives of the program were to demonstrate the condensation capability of the water pressure suppression containment concept and to verify the adequacy of the calculational techniques used in the analytical model for predicting the containment pressure and temperature response.

In 1962, a similar test facility was constructed at the Moss Landing Plant to proof test the containment design then proposed for the Bodega Bay Nuclear Plant. The facility consisted of a full scale [1/112 segment] of the Bodega containment which later became the prototype for the Mark I generation of BWR plants. A complete description of both test facilities along with the test

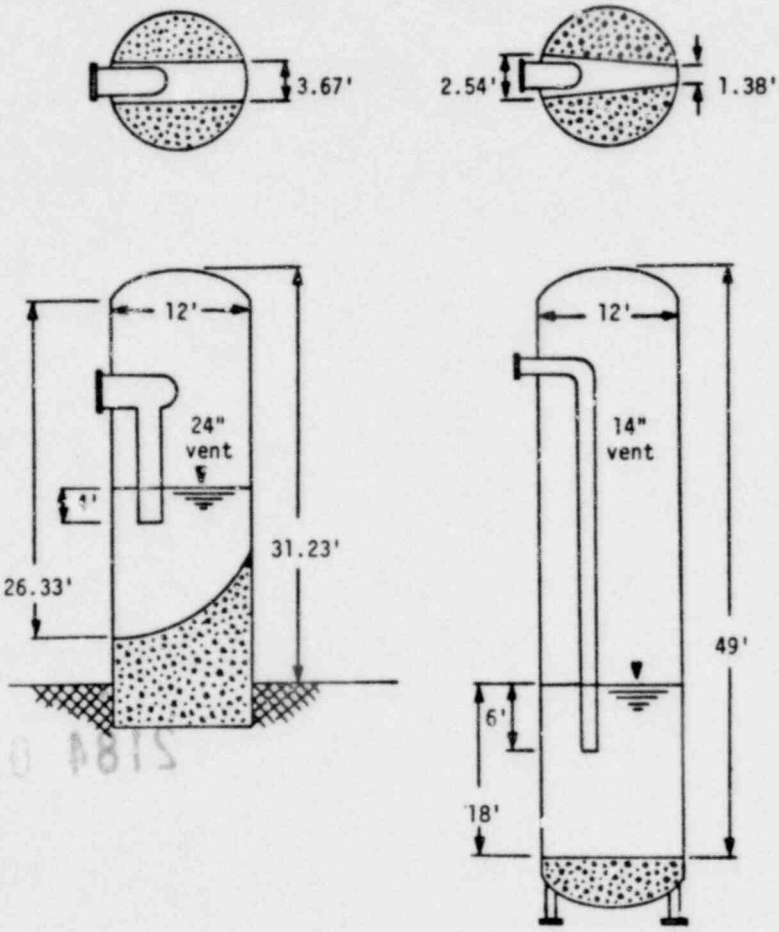
results is given in a General Electric Topical Report, NEDE-10182.[1/] A diagram of the test facility is shown in Figure II A-1. Selected design information for the two test facilities is listed in Table II-1.

Pool swell hydrodynamics and bubble breakthrough which were found to be significant factors years later, were observed in many of the tests as a discontinuity in the pressure profile response measured in the wetwell airspace. The hydrodynamic loads appeared small in comparison to the long term wetwell response. Based on this observation, it was concluded by GE that the pool hydrodynamic loads were insignificant. In retrospect, this observation was probably the result of limited instrumentation capability.

The Humboldt Bay and Bodega Bay test data were subsequently used as the basis for the design and analysis of other pressure suppression containments, including the Mark I and II containments. The data base was used to substantiate the adequacy of the General Electric containment analytical model.[2/]

In early 1970, GE developed the most recent generation of pressure suppression containment designs, known as the Mark III design. As in the case of the Mark I and II designs, small scale tests were

FIGURE II.A-1. SCHEMATIC OF WETWELL SEGMENTS FOR HUMBOLDT BAY AND BODEGA BAY TEST FACILITIES.



BODEGA BAY

HUMBOLDT BAY

TABLE II-A-1

HUMBOLDT BAY AND BODEGA BAY
TEST FACILITY DESIGN INFORMATION

	Humboldt Bay	Bodega Bay
Full Scale Segment	1/48	1/112
Downcomer diameter (in.)	14	24
Wetwell air volume (ft^3)	650	670
Wetwell water volume (ft^3)	378	340
Pool surface area (ft^2)	20.5	40.9
Pool area/vent area	19	31.4
Air volume/pool area (ft)	31.8	16.4
Drywell volume (ft^3)	277	1100

conducted in 1971 to provide a data base for the development of analytical models. Investigation of vent clearing phenomena was emphasized in this program since the major differences of Mark III from the earlier pressure suppression containments is the vent configuration and the way the vents are cleared.

A large scale test program was initiated in the fall of 1973[3/] to confirm the analytical models[4/] used in the Mark III design. The pressure suppression test facility (PSTF) located in San Jose, California was constructed by GE and used to conduct the test series. This facility was initially tested with an 8-degree sector of a full-scale Mark III suppression pool with a single column of horizontal vents.[5/] Later testing phases included one-third scale three vent tests.

The primary objective of the test program was to confirm the conservatism in the analytical models being used to predict the Mark III containment pressure response. In the first test series, it was observed that wetwell internal structures located above the suppression pool could be subjected to significant hydrodynamic loads during the pool swell process. Therefore, additional test series were conducted to determine the loads for the design of the affected

structures. A number of different target assemblies were used in the tests, including "I" beams and pipes in various sizes, gratings, and platforms. From the test data obtained for each type of target, correlations were developed to provide the impulse imparted to the target object as a function of object size and pool surface velocity at the time of impact. Our review of the PSTF test program is discussed further in the NRC staff's Safety Evaluation Report for GESSAR Standard Design (Docket Number STN 50-477), issued in December 1975.

At about the same time period as the initial PSTF test series (1972-1974), a joint testing program was initiated at the Marviken Power Station located in Sweden.[6/] A number of foreign countries, and the United States sponsored this program. The program consisted of full scale reactor containment response tests comprising sixteen vessel blowdowns into the pressure-suppression containment in the Marviken Power Station. Although the Marviken containment design differs from the U.S. designs, the basic principles of pressure suppression are the same. As a result, these full scale tests provided an additional data base to evaluate the significance of the newly identified pool swell loads from the PSTF. A second series of nine blowdown tests was conducted during 1976 to investigate steam condensation oscillations in the wetwell. The U.S. also participated in this test program.

2. Pool Dynamic Loads

An evaluation of the Mark III pool swell data indicated the need for a reassessment of both Mark I and Mark II containment designs. The designs of these facilities had not explicitly considered the pool hydrodynamic loads associated with a postulated loss-of-coolant accident (LOCA). The affected utilities formed Mark I and II "ad hoc" owners groups in 1975 with the General Electric Company (GE) acting as program manager. A series of testing programs aimed toward a better definition of the pool dynamic loads associated with each design followed.

Mark I - The 1/10 scale Mark I tests[7/] were performed by the Electric Power Research Institute (EPRI) to qualitatively determine, within a short time period (June 1975), the Mark I suppression pool surface response during the initial phase (0 to about 0.6 seconds) of a simulated loss-of-coolant accident. The facility consisted of a 1/10 scale segment of Mark I torus, a single pair of downcomers, a tank simulating the drywell, and a source of air for drywell pressurization. The scaled Mark I torus segment was constructed of clear plastic for photographic recording of the pool response during the tests. The tests were performed with different drywell pressurization rates and the bubble formation and pool surface swell were observed and recorded on film. In spite of the attempt to model flow resistance,

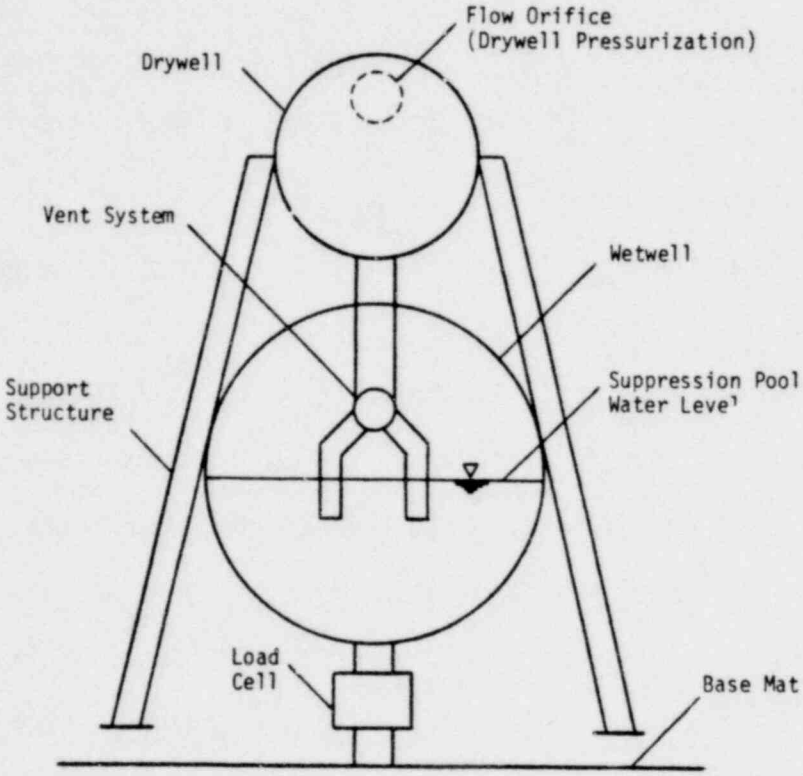
subsequent scaling analysis has indicated that flow rates into the wetwell were not well simulated in these tests.

A 1/12 scale test facility was constructed by GE to obtain additional and improved test data for use in the definition of torus hydrodynamic loads for a Mark I containment.[8/] The dimensions of the test facility and the test conditions were based on scaling relationships developed by GE. The facility, as shown in Figure II.A-2, consisted of a wetwell segment, a single downcomer pair and its associated vent system, and a drywell volume.

The transient was measured and recorded with the fast response instrumentation systems. The enthalpy flux into the vent system, the differential pressure between the drywell and the wetwell airspace, the temperature responses of the wetwell airspace and the suppression pool, the pressure profile in the wetwell, the pool swell inpulse on the vent, the vertical reaction of the system, and the pool swell velocity were determined from instrumentation readings.

A total of 42 tests were conducted from December 1975 through January 1976 in which the influence of downcomer submergence, initial system pressure, initial drywell/wetwell differential pressure, and drywell pressurization rate were studied.

FIGURE II.A-2. GENERAL ELECTRIC MARK I 1/12 SCALE TEST FACILITY



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The results from six of the 42 test runs were used to establish the torus pressure loads and the pool swell velocity for the base case conditions of a reference plant. The technique used to calculate the torus pressure force was confirmed by comparing the calculated results with measurements taken from the load cell on the base mat of the facility. The pool swell velocities were obtained from the test films.

Horizontal shearing forces and transverse loads, generated at the ends of vertical vent pipes in suppression pools during a LOCA, were investigated in a test program at a foreign test facility.[9/] The primary objective of this program was to quantify lateral downcomer loads over the range of blowdown conditions. The test facility consisted of a wetwell, simulated by a cylindrical tank, 10 feet in diameter and 60 feet high, and a single downcomer, two feet in diameter and with nine feet of submergence into the wetwell pool. The lateral loads were determined from measurements with strain gauges and linear displacement transducers located on the submerged sections of the downcomer and the wetwell wall. The test results indicated that the maximum lateral loads occur during the later stages of the blowdown (approximately 25 sec.) under conditions of low mass flux, high pool temperature, and pure steam flow.

The above testing programs formed the basis of the Mark I Owners Group's short term program (STP). The objective was to demonstrate that the licensed Mark I BWR facilities could continue to operate without undue risk to the health and safety of the public, during an interim period of approximately two years while a methodical, comprehensive long term program (LTP) is conducted. The NRC staff has reviewed the results of the STP and concurs with the conclusion made by Mark I owners. Our review is discussed further in the Mark I containment short term program safety evaluation report.[10/]

In the owners group LTP, the key element is a full scale testing program (FSTF) which will investigate the hydrodynamic loads and dynamic structural response from chugging phenomena on a representative torus sector. The facility consists of a wetwell, drywell, steam supply, vent system, instrumentation, data acquistion, and other auxiliary equipment, and testing is currently in progress.

The wetwell will house four downcomer pairs in a 22-1/2 degree sector. The drywell, steam supply and vent system are scaled to produce representative chugging conditions for a loss-of-coolant accident situation. Instrumentation is installed to measure steam flow, drywell temperature and pressure, vent flow, vent pressure, wetwell wall pressures, accelerations, strains and displacements, downcomer lateral loads, water level and temperature, and torus support column strains. The test

program will investigate the containment chugging loads and structural response for a typical geometry over a range of representative Mark I parameters. The facility has recently been completed and testing will begin in the summer of 1978.

In addition to the owners group program, the NRC has funded a major confirmatory experimental research program at Lawrence Livermore Laboratory. The program involves the construction and operation of a 1/5 scale, 7.5° and a 90° sector of a typical Mark I BWR containment system. The purpose of this program was to obtain data regarding the magnitude and character (i.e., two dimensional versus three dimensional geometry) of hydrodynamic LOCA-related air clearing loads on the Mark I containment system in order to confirm the results obtained from the testing programs sponsored by the Mark I Owners Group.[11/] The testing program has been completed, a report issued,[12/] however, the final evaluation of the data has not been completed.

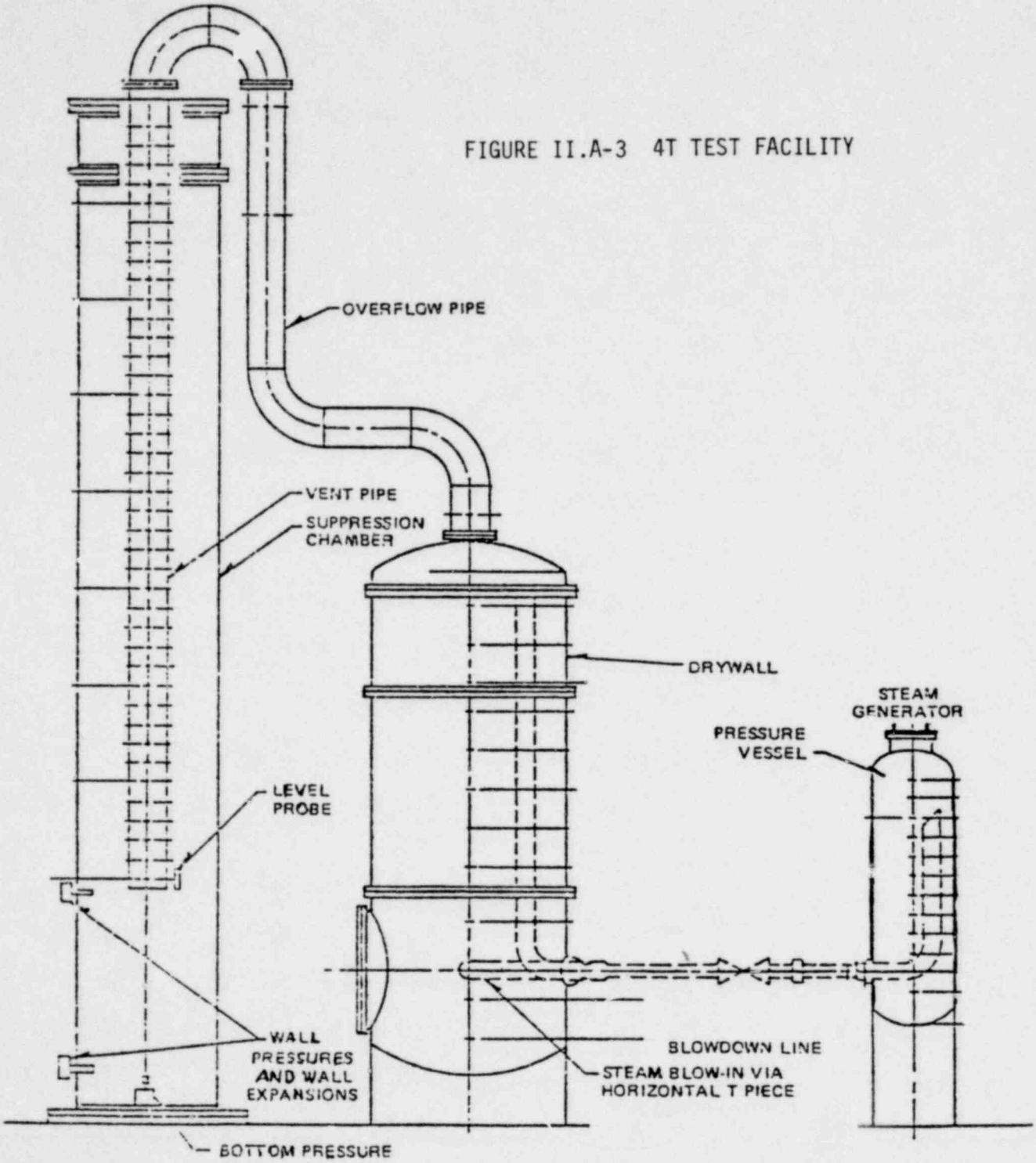
Preliminary indications have revealed that the anticipated load reductions due to three dimensional effects may not be realized. Early comparisons of 2-dimensional data (single downcomer pair) to the 90° sector data (12 downcomer pairs), in fact, show slightly higher equivalent loads for the 90° sector. Nevertheless, using the 90° sector data, the loads are still below those used for the STP

evaluation. Final evaluation of this phenomenon as well as other aspects of the LOCA pool dynamic loads will be conducted as part of the staff's generic Task Action Plan A-7, Mark I Long Term Program.[13/] Completion of this task is scheduled for late 1979.

Mark II - A similar testing program plan was established by the Mark II Owners Group in eary 1975 to determine the LOCA related pool dynamic loads for Mark II containment designs. The program plan evolved into a multi-phase plan. To establish the initial loads, several testing programs were established; the most notable being the Mark II pressure suppression test program which was initiated in the fall of 1975. The facility was constructed to simulate in full scale, a single downcomer vent. A schematic diagram of the 4T facility is shown in Figure II.A-3. The test program consisted of approximately 60 blowdown events. Parameters varied included downcomer submergence, diameter, and bracing configuration, wetwell pool temperature and drywell initial temperature.

The measurements which were taken yielded data on pool swell hydrodynamic loads and pool swell behavior, froth and breakthrough for the Mark II design. In addition, data were obtained on downcomer lateral loads and pressure fluctuation loads on the submerged wetwell during steam condensation and chugging. The data from these tests form the basis for qualification of the pool swell analytical model and specification of diaphragm floor loads and submerged wetwell pressure loads during steam condensation.[14/]

FIGURE II.A-3 4T TEST FACILITY



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In addition to the efforts undertaken by the Mark II Owners Group, there is further testing ongoing in Japan to investigate LOCA pool dynamic loads for BWR Mark II type containments. The testing consists of two phases: a series of 1/6 scale single and multi-vent tests and a full scale single and multi-vent test series. The 1/6 scale tests conducted in 1977 were sponsored by the Japanese BWR utilities and manufacturers. Based on the results of these tests, a full-scale segment of a Mark II containment was designed. Construction was initiated in the summer of 1978, with testing scheduled to begin in April 1979.

3. Safety-Relief Valve Testing

Until the early 1970's, the ony significant hydrodynamic loads considered in the containment design were those associated with a postulated LOCA. Safety relief valve (SRV) discharges experienced during the 1960's did not cause serious damage to any of the structures with only minor damage to the SRV line supports. During this time period, SRV induced loads were considered to be small and secondary.

However, in 1972, two German BWR plants with water suppression containments experienced severe vibratory loads on the containment structure during extended SRV operation. Both plants used straight down discharge pipes. In one case, the loads were severe enough to cause damage on the containment shell and components and resulted in water leakage from the suppression pool.

Following these incidents, extensive experiments were conducted by the Germans to investigate various SRV downcomer discharge configurations. The objective of the investigation was to develop a discharge device which would reduce the hydrodynamic loads during SRV line air clearing and provide stable steam condensation at elevated pool temperatures. Varied configurations of the discharge device considering more than 20 design parameters were investigated. Results of the investigation concluded that the quencher type device yielded superior performance. Data from the development and testing of the device is provided in a GE topical report.[15/]

In addition to the testing program discussed above, GE was developing an analytical model to predict the loads associated with the steam vent clearing phenomenon. To support the model predictions, in-plant tests were conducted at the Quad Cities Nuclear Power Plant.[16/] The test comparisons indicated a need for model improvements. In 1975, improvements to the model were completed and results from its use were compared with the available in-plant test data.[17,18/]

During this time period of model development, a review of plant operating practices was made by GE in light of the German experiences. Unlike the German reactors, the U.S. plants had ramshead discharge devices rather than the straight down pipe. Prolonged SRV blowdowns

had been experienced but without the severe vibrations.[15/] As a result of this review, GE recommended[19/] reduction in certain allowable pool temperatures to ensure avoidance of the steam quenching vibration phenomenon. Shortly thereafter, the Technical Specifications in licenses for operating plants were revised in light of the GE recommendation.

The staff also reviewed the revised analytical model along with the supporting data base. It was indicated that additional in-plant data were needed to verify the model. As a result, tests were performed at the Monticello power plant in 1976.[20/] These additional data are currently under review by the staff as part of its generic Task Action Plan A-39, "Determination of Safety Relief Valve (SRV) Pool Dynamic Loads and Temperature Limits for BWR Containment."[13/]

In view of the superior performance of the quencher device,[15/] GE proposed the cross quencher for the standard Mark III design (GESSAR-238 Nuclear Island, Docket Number STN 50-477). The staff reviewed and found the quencher acceptable for Mark III containment applications. Our review is discussed further in the NRC staff's Supplement No. 1 to the Safety Evaluation Report for GESSAR (Docket No. STN 50-477), which was issued in September 1976. Since that time, all Mark III applications have reflected the use of the GE cross quencher.

With respect to the quencher application for Mark I and II designs, both owners groups have included tasks in their overall program to develop a quencher. The Mark II Owners Group has included the GE cross quencher. To verify the quencher performance for a Mark II application, the owners have proposed an in-plant testing program at the Caorso Nuclear Power Plant in Italy.[21/] These tests are scheduled to be completed in the summer of 1978. The staff's review will be conducted as part of its generic Task Action Plan A-39.[13/]

Not all Mark II applicants, however, have selected the GE cross type quencher. The applicants for the three lead Mark II plants (Zimmer, Shoreham and LaSalle) have recently advised the staff that they are considering a T-Quencher designed by the Kraftwerk Union (KWU) of Germany. Documentation of the design and supporting test data will be filed with the NRC in October 1978.

In addition to the in-plant testing described above, the Japanese are also performing in-plant quencher tests. Tests are currently in progress at the Tokai-2 Nuclear Power Plant. Tokai has a Mark II containment design

using a GE cross quencher device. Discussions with the Japanese have indicated that preliminary results of the early tests show greatly reduced containment loads.

The Mark I Owners Group has also pursued the developments of a quencher device to replace the ramshead device. Because of the space limitations within the torus, a unique Mark I T-Quencher has been developed as part of the LTP. In 1977, small scale tests were performed to optimize the quencher design. These tests were followed by in-plant tests at Monticello in 1977. The documentation of these tests will be filed with the NRC as part of the LTP in the near future.

B. ICE CONDENSER CONTAINMENT

During 1967 and 1968, the Westinghouse Electric Corporation (Westinghouse) conducted an intensive design and testing effort to demonstrate the feasibility of the ice condenser containment concept as applied to a nuclear power plant containment. The feasibility of the concept was demonstrated by the full-scale ice condenser tests conducted at the Westinghouse Waltz Mill test facility.[22/] In these tests, the response of the ice condenser containment was tested by release of high energy steam and water into a test facility which simulated the ice condenser containment compartments, including a full-scale representation of a section through the ice bed. The results of these efforts led to the ice condenser containment preliminary design for a number of nuclear power plants, e.g., D.C. Cook, McGuire and Sequoyah. The test facility is shown in Fig, II.B-1.

After these early full-scale demonstration tests, a number of design changes were made to ice condenser components. These changes were prompted by the need to improve the structural capability of some components and the need for improvements in fabrication or installation methods which became apparent during the detailed design of the ice condenser plants. Recognizing the significance of the

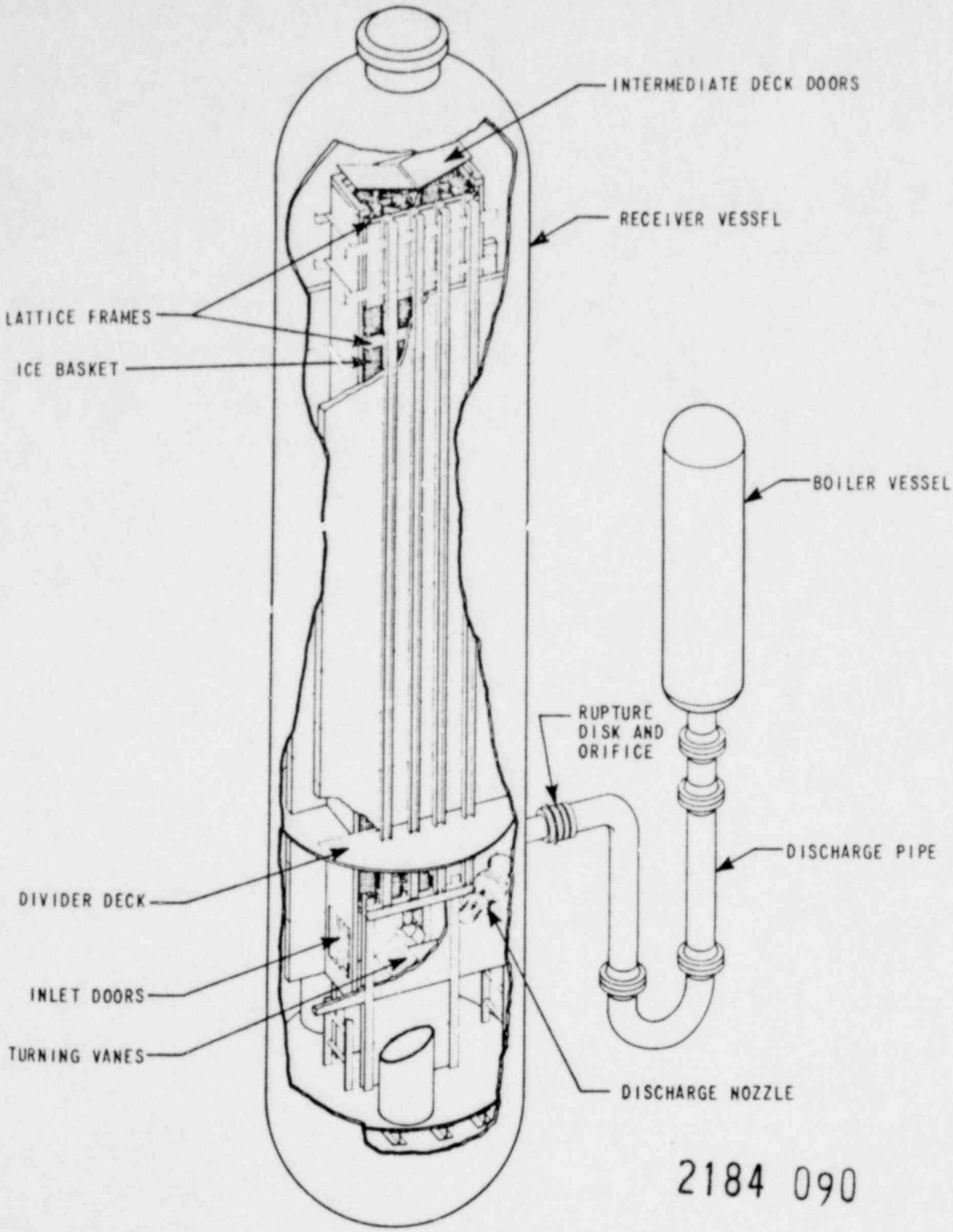

FIGURE II.B-1 Isometric View of Boiler and Receiver Vessels at the Waltz Mill Ice Condenser Test Facility

changes in ice condenser components during the detailed plant design evaluation, Westinghouse conducted additional test programs in 1973 and 1974 to qualify the ice condenser as currently designed for use in a nuclear power plant. These tests ranged from component tests through full-scale system tests, and are reported in the documents given in the reference list at the end of this section.[22-37]

Component tests were conducted on the ice condenser wall panels, ice baskets, lower inlet doors and shock absorbers, intermediate deck doors and top deck doors. One-quarter-scale air flow tests were conducted for models of a section of the ice condenser. Full-scale steam blowdown tests were performed for a section of the ice condenser containment. Vibration tests were performed to determine the minimum time which must be allowed for ice loaded into the ice baskets to age in order to achieve sufficient fusion to retain the ice within the baskets when subjected to design basis earthquake vibrations.

These test programs were reviewed by the NRC staff. The results of our review were published in safety evaluation reports.[22,23,24,25]

In addition, operating experience with the ice condenser at the D.C. Cook Nuclear Plant Unit 1 (licensed to operate in December of 1974) has indicated the need to alter a number of ice condenser components and equipment operating and maintenance procedures such as: (1) minor modifications to the ice condenser air handling unit components, changes in control temperatures of the chiller packages and increased maintenance frequency for the air handling equipment to improve the ice condenser cooling performance; and (2) minor modifications to the ice condenser lower inlet door assemblies and personnel access doors to reduce cold air leakage from the ice condenser. These design improvements developed by the operating experience at the D.C. Cook plants are being carried forward into the later ice condenser plants, e.g., McGuire and Sequoyah.

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24. "Topical Report Evaluation - Supplement 9 to WCAP-8110," September 1974 - Appendix C.

25. "Topical Report Evaluation - WCAP-8336 and Supplement 9 to WCAP-8110," November 3, 1975 - Appendix D.

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III. MAJOR REVIEW AREAS FOR PRESSURE SUPPRESSION CONTAINMENTS

A. BWR Pressure Suppression Containment Design

1. Containment Functional Design

a. Containment Pressure and Temperature Response

The design and sizing of the containment system are largely based on the calculations made for pressure and temperature conditions which are postulated to result from releases of the reactor coolant in the event of certain postulated loss-of-coolant accidents (LOCA). All of the BWR pressure suppression containment designs experience a short term and a long term pressure response in the drywell and in the wetwell volumes of the containment. The short term pressure peak is the design controlling peak for both the drywell and the wetwell for each Mark I, II, and III containment designs with one exception. For the Mark III containment, the wetwell design is determined by the long term peak pressure. A brief discussion of the review considerations follows.

1) Short Term Pressure Response

The peak short term pressure occurs within the first few minutes following a LOCA. The magnitude and time of the peak pressure vary for the different containment designs, depending primarily on the

largest break size that can be postulated for the primary system piping and the total vent area characteristic of the design.

The Mark I containment is typically designed to a short term peak drywell and wetwell pressure of about 60 psig while the Mark II containment is designed to a short-term peak drywell and wetwell pressure of about 45 psig. For both the Mark I and the Mark II designs, the short term pressure peak occurs in a time period of about 10 to 100 seconds. This is substantially after clearing of the vents and near the end of blowdown of the reactor vessel. The Mark III containment drywell is designed to a short term peak drywell pressure of 30 psid. This peak occurs in a 1 to 2 second time period near the time of vent clearing.

2) Long Term Pressure and Temperature Response

Following the initial phase of a LOCA, the drywell and wetwell pressures reduce substantially to about 5 psig as the steam in the drywell is condensed by ECCS water spilling out of the postulated primary

system break. Thereafter, the wetwell pressure and temperature continues to rise due to the input of the core decay heat and sensible (stored) heat to the suppression pool. The peak long-term pressure and temperature in the wetwell is reached when the suppression pool heat removal rate by the heat removal system is equal to the rate of heat input to the pool from the core decay heat and sensible heat.

The Mark III wetwell design pressure is determined by the long term pressure peak. A design pressure of 15 psig is typical for the Mark III wetwell. The long term pressure peak occurs several hours after a LOCA. For the Mark I and II designs, the long term pressure peak is substantially below the short term pressure peak. The relatively large wetwell volume for the Mark III containment is the reason that the long term pressure peak is controlling.

The peak wetwell temperature is reached at the same time as the peak long term pressure for the Mark I, II, and III designs.

The containment drywell temperature response associated with a small primary system steam line break may exceed that calculated for the large break LOCA. Although the initial temperature rise may be less rapid for the small break, the potential maximum temperature and in particular the duration can exceed the large break LOCA environment. In the case of a small steam line break the rupture size and location can be such that saturated steam, but no liquid, escapes to the drywell. For this condition, the throttling of the reactor system's high pressure steam to the low pressure drywell results in superheating of the steam.

The temperature transient in the drywell due to a small break does not cause reactor system depressurization or automatic operation of emergency core cooling systems. Following a break of this size, the reactor operators will initiate an orderly shutdown and cooldown of the plant. The large break produces the most rapid rise of containment temperature but the duration of the high temperature condition is less than that of the small break

because the large break rapidly depressurizes the reactor coolant system and causes the automatic initiation of the emergency core cooling systems which cool the reactor system and ultimately the containment. Thus the drywell environmental temperature envelope must take into consideration the rapid rise in pressure and temperature due to a large break LOCA and the maximum potential temperature and duration associated with a small primary system steam line break.

b. Suppression Pool Makeup System (Mark III)

A unique feature of the Mark III containment design is the suppression pool makeup system. This system provides water from the upper containment pool to the suppression pool following a loss-of-coolant accident. This increase in long term suppression pool inventory will provide additional pool heat capacitance, a minimum long term drywell vent coverage of two feet and will account for any post-accident entrapment of water in the drywell unique to Mark III because of the horizontal vent configuration.

The system is reviewed by the staff to assure: (1) that the system does not dump the upper pool volume during a LOCA so as to aggravate pool dynamic loads; and (2) that the system will dump when necessary to assure an adequate long term suppression pool inventory of water.

c. Reverse Pressure Response

In addition to designing the drywell and wetwell volumes to the internal pressure condition resulting from the release of reactor coolant, these containment volumes must also be designed to withstand reverse pressures. This is to assure the functional integrity of the containment over all operating conditions.

Those events common to all of the BWR containment designs which may result in reverse pressures for the drywell and wetwell include the following:

1) The operator inadvertently actuates containment spray during normal plant operation;

2) The containment spray is initiated following a loss-of-coolant accident; and

3) The containment spray is initiated following a small steam line break.

A potential reverse pressure situation also exists following a LOCA at the time ECCS spillage condenses the steam in the drywell. This is due to the carryover of drywell air to the wetwell immediately following the LOCA. This reverse pressure is exerted on the vent system in the Mark I design, the containment diaphragm in the Mark II design and the drywell walls in the Mark III design.

Pool swell related pool dynamic loads also may yield a momentary reverse pressure on the containment diaphragm for the Mark II design.

To accommodate the reverse pressure loads, one of the following design procedures is followed: (1) the containment structure is designed to the peak reverse pressure; (2) vacuum breakers are utilized to provide pressure equalization across the containment boundary; or (3) restrictions are placed on the temperature and humidity conditions in the containment atmosphere.

d. Subcompartment Pressure Analysis

A subcompartment is a fully or partially enclosed volume within the containment that houses high energy

piping systems and would limit the flow of fluid to the main containment volume in the event of a postulated pipe rupture within this volume.

Containment subcompartment pressure analysis is the evaluation of the consequences of a postulated pipe rupture in these compartmentalized regions of the containment. This analysis is concerned with the short term (less than one second) pressure response of the subcompartments which enclose critical components of the reactor system. The safety concern is that localized pressure gradients, due to a postulated pipe rupture, in small subcompartments could result in the damage of subcompartment walls and failure of certain major components (e.g., reactor vessel) supports. The damage of walls adjacent to major equipment or the failure of equipment supports could jeopardize the ability to bring the reactor to a safe condition in the event of a reactor system pipe rupture.

In the Mark I and II containment system design, there are essentially two areas in the drywell where the

arrangement of internal structures forms subcompartments or restricted volumes which may be subjected to differential pressure loadings following certain postulated pipe ruptures: the annulus formed by the reactor vessel and the sacrificial shield, and the drywell head region which is a cavity surrounding the reactor pressure vessel head. Since the suppression chamber is virtually an open space for these containment designs, no restricted volume exists.

The Mark III containment design has the same two drywell subcompartments as the Mark I and II containment described above. In addition, the Mark III containment has subcompartments located within the suppression chamber. These compartments consist of rooms where components of the reactor water cleanup system are located. Postulated coolant breaks in each region are included in the evaluation.

e. Steam Bypass of the Suppression Pool

For each of the BWR pressure suppression containment designs, steam released from the primary system during

a postulated LOCA is collected in the drywell and directed through the vent system to the suppression pool water where it is condensed. The potential exists for steam to bypass the suppression pool through a number of paths. If a sufficient amount of the steam bypasses the suppression pool, the resultant pressure could exceed the design pressure of the containment. Two major bypass paths exist.

One major bypass path results from high energy lines which pass through the vapor space above the suppression pool in the wetwell. This potential for steam bypass is unique to the Mark III containment since the Mark III wetwell surrounds the drywell and the high energy lines must penetrate the wetwell volume.

The second major bypass path consists of certain direct paths from the drywell to the vapor space above the suppression pool. They include such bypass paths as those through vacuum breakers, vent pipes, the Mark II diaphragm, wall seals around penetrations in the drywell-wetwell divider, cracks in the concrete drywell

walls of a Mark III containment, hydrogen mixing penetrations, and recombiner penetrations. Each of the BWR pressure suppression containment designs has a potential for some steam bypass through one or more of the above paths. However, the specific paths differ for each of the designs. The above described potential for pool bypass is accounted for in the design of the containment structure, system design margins or through a combination of both in order to maintain the containment pressure below the design pressure.

f. Suppression Pool Hydrodynamic Loads

1) LOCA Pool Dynamics

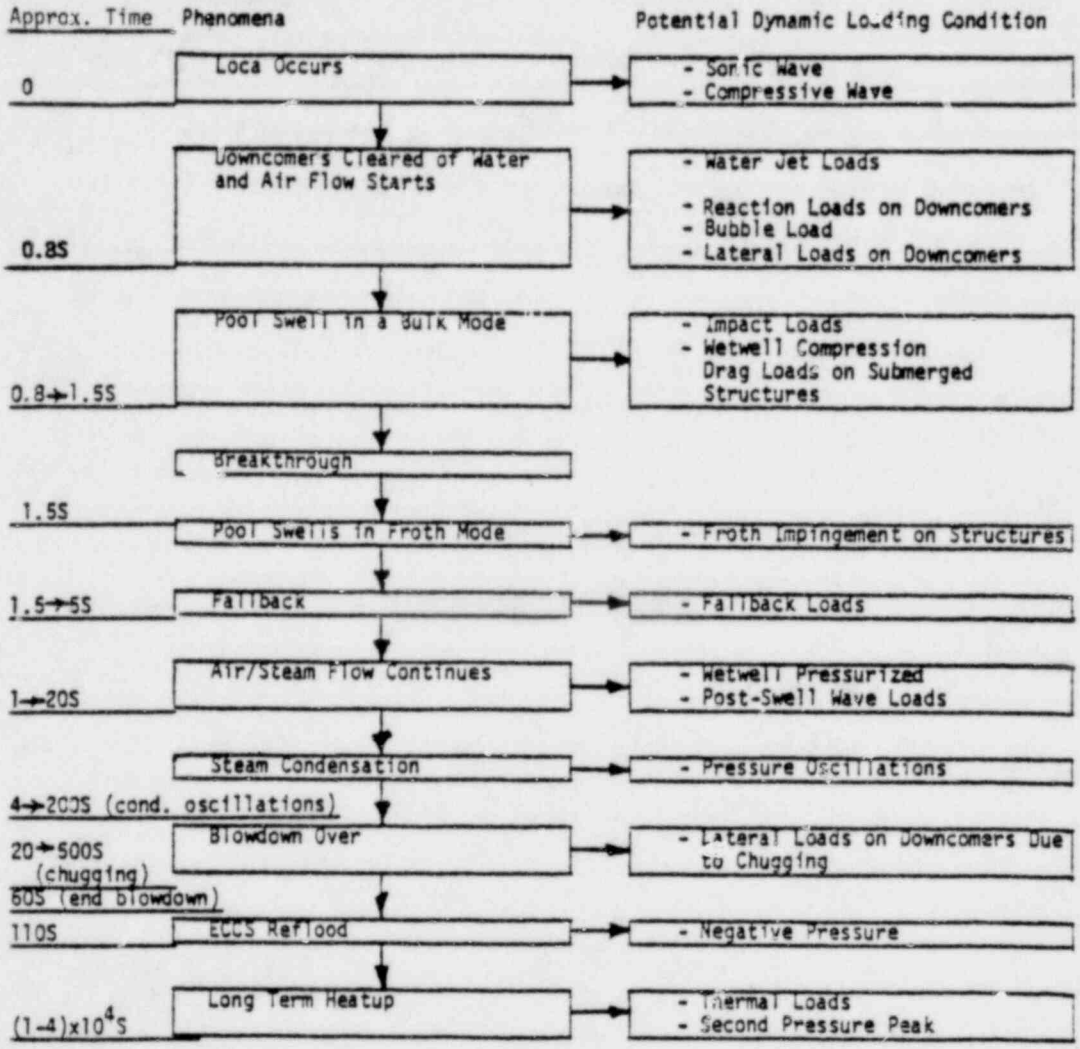
During the conduct of a large scale testing program in 1974 for the Mark III containment design, a number of suppression pool hydrodynamic loads associated with a postulated LOCA were identified which had not been explicitly included in the original design of the Mark I, II, or III BWR suppression containment systems. These additional loads result from the dynamic effects of drywell air and steam being rapidly forced into the suppression pool during a postulated LOCA event.

In view of the potential significance of these loads, it was determined that a reassessment of the Mark I and II containment system designs was required by the staff. A letter was sent by the staff to each domestic Mark I and II owner in April 1975 notifying them of the need for this reassessment. A General Electric program to review these loads for the Mark III design was already in progress. The significant events associated with this concern are discussed in Section II.A of this report.

Figure III.A.1 shows the sequence of events following a postulated LOCA and the potential hydrodynamic conditions associated with these events. Following initiation of the postulated LOCA, the drywell pressure increases due to blowdown of the reactor system. Pressurization of the drywell causes the water initially in the vent system to be accelerated out through the vents. During this water expulsion process, the resulting water jets cause impingement loads on local containment structures.

Following vent clearing, an air/steam bubble forms at the vent exits which causes a hydrostatic

FIGURE III.A-1
TYPICAL SEQUENCE OF EVENTS FOR LOCAS



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pressure increase in the pool water resulting in a loading condition on the pool boundaries. The steam condenses in the pool. However, the continued addition and expansion of the drywell air causes the pool volume to swell, resulting in a rise of the pool surface. Upward motion of this slug of water creates a drag load on structures submerged in the pool and impact loads on structures located above the initial pool surface.

After the pool surface rises to about 1.5-2.0 times the initial submergence of the vents, the rising slug of water breaks apart as the air bubble breaks through the surface of the water slug thus relieving the driving force for pool swell. Subsequent pool swell involves a two-phase air water froth which produces further structural-impingement loads. A gravity induced fallback of the pool returns the surface to the post LOCA elevation.

Following the pool swell transients, there will be a period of high steam flow through the main vent

system. At these high steam flow conditions, the water/steam condensation interface oscillates due to bubble growth and collapse. These condensation oscillations result in an oscillatory load on the pool boundary. At low vent flow rates, the water/steam condensation interface can oscillate back and forth in the vents causing "chugging." The chugging action results in loads on both the vents and the containment boundaries.

All of the above LOCA related hydrodynamic loads are potential loading conditions for the three BWR suppression containment designs (i.e., Mark I, II, and III). However, not all of these loads have the same significance for each design. The significant LOCA hydrodynamic loads for each design are summarized below.

a) Mark I Containment

The major concern addressed by the Mark I Short Term Program was the pool swell loading on the containment torus structures. The loads used

were the maximum probable loads defined in NUREG-0408 and they consisted of a down load followed by an up load on the torus. They occur during the air clearing phase of the vents and the wetwell compression phase at the time of bubble breakthrough. The load on the torus is transmitted to the torus support and to certain piping systems attached to the torus.

Other significant concerns of the Mark I Short Term Program related to the loads on structures that are internal to the torus structure are as follows: (1) the drag loads on components submerged in the pool; (2) pool swell impact and drag loads on structures above the pool surface; and (3) vent system lateral loads occurring during the "chugging" period.

b) Mark II Containment

Those LOCA related hydrodynamic loads which received primary attention in the Mark II evaluation program include: (1) the drag

loads on components submerged in the pool;

(2) pool swell impact loads and drag loads on structures above the pool surface;

(3) the upward load on the diaphragm separating the drywell and wetwell occurring at about the time of maximum pool swell height; and

(4) the "chugging" loads on the vents and the pool boundary.

c) Mark III Containment

Those LOCA related hydrodynamic loads which received primary attention in the Mark III evaluation program include:

1) the drag loads on components submerged in the pool; and

2) pool swell impact and drag loads on structures above the pool surface.

2) Safety/Relief Valve Pool Dynamics

BWR plants utilizing the Mark I, II and III containment designs are all equipped with safety/relief valves (SRV) that discharge into the suppression pool where the

steam is condensed. Upon SRV actuation, the initial air column within the SRV discharge line is accelerated by the high pressure steam flow; the air expands as it is released into the pool as high pressure air bubbles. The high rate of air and steam injection into the pool followed by expansion and contraction of the bubbles as they rise to the pool surface produces oscillatory loads on the containment structures and on components located inside the containment. This effect is referred to as the air-clearing phenomenon.

There is another phenomenon associated with SRV discharge, but it occurs at a later time in the transient. This is the phenomenon of steam condensation instability. Following the air-clearing phase, pure steam is injected into the pool. Condensation oscillations occur during this time period. However, the amplitudes of these vibrations are relatively small at low pool temperatures. Continued blowdown into the pool will increase the pool temperature and for certain discharge designs a threshold temperature can be reached. At this point, steam condensation becomes unstable. Vibration and the associated forces increase

sharply and become very severe. Current practice for BWR plants is to restrict the allowable operating temperature envelope via Technical Specifications such that the threshold temperature is not reached. This is referred to as the pool temperature limit.

Extensive experiments have been conducted to investigate various discharge device configurations (see Figure III.A.2). The objective of these investigations was to develop a discharge device which would reduce the hydrodynamic loads during SRV air clearing and provide stable steam condensation. Varied configurations of the discharge device, considering more than 20 design parameters, were investigated. Results of the investigation concluded that the quencher type of device yielded superior performance.

As part of the continuing GE program to better define the relief valve discharge pool dynamic loads, GE determined that a valve actuation sequence can occur which results in a new load combination. This resulted from a recent study by GE of the primary system pressure

[redacted]

response following isolation of the boiling water reactor by rapid closure of the main steamline isolation valves with the reactor at full power. This concern is referred to as multiple-sequential relief valve actuations.

A variety of Safety Relief Valve devices have been proposed to mitigate the loads associated with the air-clearing phenomena and to raise the pool temperature limits. These devices vary somewhat for the three containment designs. The devices proposed for use in each containment design are described briefly below.

a) Mark I Containment

Currently, all Mark I containments with exception of the Oyster Creek plant, which uses a quencher type device, use a ramshead device. However, the Mark I Long Term Program includes a number of tasks which will lead to the development of a quencher type device specifically designed for the Mark I containments. This device is referred to as the "T-quencher." The results of testing to date indicate that the loads are decreased and

the temperature concern diminishes.

b) Mark II Containment

Originally, the Mark II facilities proposed using the ramshead device. Recently, most of the Mark II facilities have proposed use of a quencher device. Two types of quencher devices have been proposed. One type is the GE cross quencher which is shown in Figure III.A.2. The other device is a T-quencher designed by Kraftwerk Union (KWU) of Germany. The KWU T-Quencher is designed for use in Mark II containments.

c) Mark III Containment

The cross quencher is currently proposed for use in all Mark III containments. It has four arms and each is perforated with several rows of small holes. Steam flows through the hub and is distributed among the four arms and discharged into the pool. It is unlike a ramshead device where air or steam forms large bubbles. The quencher device produces a cloud of air or steam mist. As a result, the magnitude of the air clearing load is reduced by

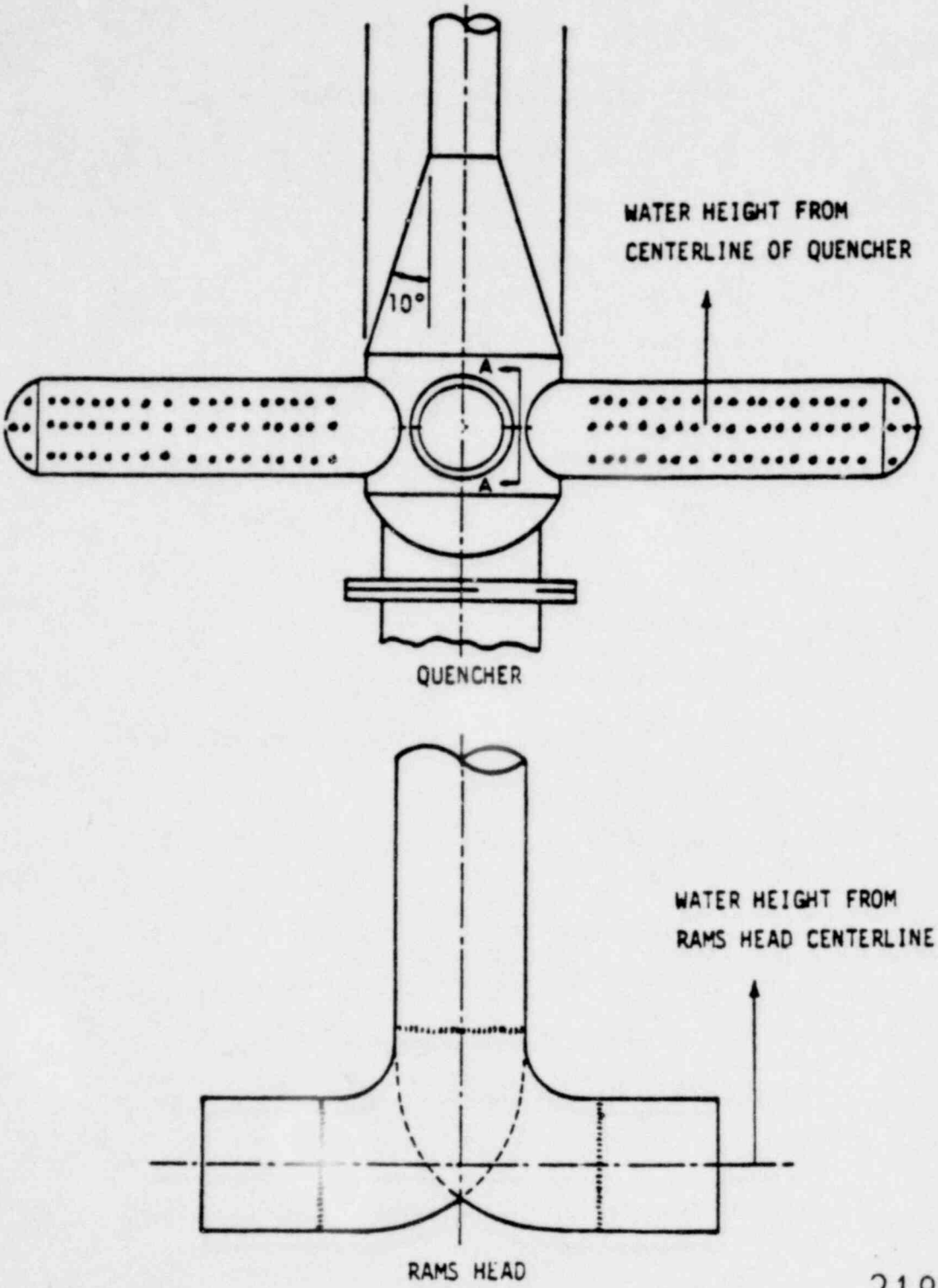

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Figure III.A.2 Quencher and Rams Head Schematic

a factor of two to four in comparison with the ramshead device. Steam condensation instability was found not to occur in tests where the pool temperature reached 210°F.

g. Secondary Containment Functionability.

The secondary containment system includes the structures and systems used to control and treat radioactive leakage from the primary containment in the event of a loss-of-coolant accident. All of the BWR suppression containment designs include this additional barrier to the release of fission products. In contrast, a secondary containment is not utilized in conjunction with all of the conventional dry containment designs.

The secondary containment, typically a cylindrical reinforced concrete wall and a domed roof, completely surrounds the primary containment. The secondary containment will be maintained at a negative pressure following a loss-of-coolant accident so that any radioactive material that may leak from the primary

containment following an accident will be collected and processed prior to release to the environment.

The staff review of the functional capability of the secondary containment system designs includes the following points:

1) Analyses of the pressure and temperature response of the secondary containment to a loss-of-coolant accident within the primary containment.

2) Analyses of the effect of openings in the secondary containment on the capability of the depressurization and filtration system to accomplish its design objective of establishing a negative pressure in a prescribed time.

3) Analyses of the pressure and temperature response of the secondary containment to a high energy line rupture within the secondary containment.

4) The functional design criteria applied to guard pipes surrounding high energy lines within the secondary containment.

5) Analyses of any primary containment leakage paths that bypass the secondary containment.

6) The design provisions for periodic leakage testing of secondary containment bypass leakage paths.

7) The proposed Technical Specifications pertaining to the functional capability of the secondary containment system and the leakage testing of bypass leakage paths.

III.A.2 Containment Heat Removal Systems

The containment heat removal system includes the piping, valves, and mechanical components which will be used to remove energy from the containment and limit the atmospheric temperature and pressure in the containment following a postulated loss-of-coolant accident. In a BWR containment, the primary function of the containment heat removal system (CHRS) is to control the long term pressure and temperature response of the containment atmosphere. This is accomplished by removing heat from the water in the suppression pool. The containment spray mode of the residual heat removal (RHR) system can be used to condense the steam released to the containment atmosphere following a small line break.

For a BWR pressure suppression containment design, this system (the CHRS) is referred to as the Residual Heat Removal (RHR) System. The system consists of two complete loops including heat exchangers and pumps. Operating in the containment cooling mode, the RHR pumps take suction from the suppression pool. Flow is then directed through the RHR heat exchangers to the suppression pool, the reactor vessel or the containment spray headers. The location of system return lines in the suppression pool facilitates mixing of the return water with the total pool inventory before the return water becomes available to the suction lines. Strainers are provided on the suction line inlet. Significant areas of the staff's review of the system include:

1) Analyses of the Net Positive Suction Head (NPSH) available to the containment heat removal pumps;

2) The effects of debris, including insulation, on the heat removal system;

3) Analyses of the consequences of single component malfunction in each system;

4) The quality group and seismic classification for each system;

5) The design of the sumps or inlet strainers on the RHR inlet lines; and

6) The placement of the RHR system's inlet and return lines to assure adequate mixing of the return water in the pool.

III.A.3 Containment Isolation System

The immediate result of a major reactor coolant pipe rupture inside containment would be the release of fission products to the atmosphere within the containment. To prevent release of these fission products to the outside environment, all penetrations in the containment must be isolated. In order to accomplish this goal, a system to isolate the containment is required. The design objective of this containment isolation system is to allow the normal or emergency passage of fluids through the containment boundary while preserving the integrity of the containment boundary.

Typically, a containment isolation system includes the containment isolation valves and associated piping and penetrations necessary to isolate the primary containment in the event of a loss-of-coolant accident.

III.A.4 Combustible Gas Control

Combustible gases may accumulate in the containment atmosphere following a LOCA. These gases include hydrogen and oxygen. The major sources of hydrogen and oxygen in the event of a postulated LOCA are: a chemical reaction between the fuel rod cladding and steam, the radiolytic decomposition of the water in the reactor core and the water in the containment, and possibly the corrosion of aluminum and other materials by the spray solution. If excessive hydrogen is generated, it may accumulate with the oxygen in the containment atmosphere leading to a flammable mixture which if ignited could affect the integrity of the containment.

Criterion 41 of the General Design Criteria (Appendix A to 10 CFR Part 50) requires that all containments be provided with systems to control the combustible gases following postulated accidents to ensure that containment integrity is maintained. The combustible gas control systems include the piping, valves, components, and instrumentation necessary to detect the presence of combustible gases within the primary containment and to control the concentrations of these gases. The staff's review of these systems includes the potential sources of combustible gases and their yields, the accumulation of gases within each volume of the

containment system, the capability to monitor the concentration of the gases, and the capability to control and reduce the combustible gas concentrations by suitable means.

A characteristic feature of pressure suppression containments is their relatively small containment volumes. As a result of these small volumes, calculations indicate that combustible gas limits may be reached in relatively short time periods.

The combustible gas control system for BWR pressure suppression containments are designed to accommodate the potential rapid production of combustible gases following a LOCA. The systems vary for each containment type, i.e., Mark I, II, and III, to reflect the specific characteristics of each containment. A description of the types of systems used in each type of BWR pressure suppression containment is provided below.

a) Mark I Containment

With the exception of Vermont Yankee and Hatch 2, all Mark I containments are currently required to be inerted during operation to assure control of the combustible gases in the event of a LOCA. This requirement resulted from the staff's

conservative assumption at the time of licensing for the amount of hydrogen generated by metal water reactor of the fuel cladding in the core immediately following a LOCA.

The staff's assumptions regarding this potential hydrogen source has since been revised based upon the development of a proposed change to the regulation, i.e., 10 CFR 50.44, "Standard for Combustible Gas Control Systems in Light Water Cooled Power Reactors." The revised assumptions in this proposed rule and those specified in the Branch Technical Position CSB 6-2, in SRP 6.2.5, Combustible Gas Control In Containment, would permit plants to de-inert where it can be demonstrated that the hydrogen concentration can be maintained below a combustible mixture. The analyses for Vermont Yankee indicate that most, if not all, plants could be de-inerted using the assumptions in the proposed 10 CFR 50.44.

Radiolytic decomposition of the water in the core and the containment leads to a continuous though slow increase in the hydrogen and oxygen in the containment which if not controlled could reach combustible limits even in an inerted containment. Containment Atmospheric Dilution (CAD) systems and containment

purge systems are used in Mark I containments to control this long term buildup of combustible gases in the containment.

Most Mark I containments have CAD systems. These systems reduce the concentration of combustible gases in the containment atmosphere by the addition of an inert gas (nitrogen) diluant. This system is started as combustible gas limits are approached which is calculated to occur about one week following a LOCA. Use of this system results in a continuous but slow pressurization of the containment. Containment atmosphere is periodically released to limit the amount of containment repressurization to acceptable limits.

Several of the older Mark I containments utilize a containment purge system to control the long term buildup of combustible gases. One Mark I facility (Hatch-2) which was recently licensed for operation utilizes recombiners to recombine the hydrogen and oxygen combustible gases to form water.

b. Mark II Containments

Those facilities with Mark II containment designs have not yet been licensed for operation; however, in light of the proposed change to the regulations and the staff's Branch Technical Position CSB 6-2, we do not expect that inerting will be

required for these facilities. Revised metal water hydrogen calculations based on the proposed change to the regulations (10 CFR 50.44) indicate that combustible limits can be maintained below the combustible limits due to this source. As a result, Mark II facilities are proposing the use of hydrogen recombiners to control the slow buildup of combustible gases due to the radiolysis of water in the core and containment. The recombiner need not be placed in operation for Mark II containments before several hours following a LOCA. In addition to redundant recombiners, a backup purge system provides an additional means of controlling the long term buildup of hydrogen in the Mark II containment.

c. Mark III Containments

Mark III containments have relatively larger containment volumes than Mark I and II designs and do not require inerting for combustible gas control. However, mixing systems are provided to take advantage of the large wetwell volume for controlling the hydrogen concentration in the small volume drywell. This system mixes the drywell and wetwell volumes. The system is placed in operation about 15 hours following a LOCA. The mixing system acting alone is inadequate to

control the long term buildup of hydrogen in the containment to acceptable limits. The long term buildup of containment hydrogen is controlled by hydrogen recombiners. Mark III recombiners usually need not be placed in operation before a week following a LOCA.

III.A.5 Containment Leak Testing

The capability of a containment structure to perform its safety function is dependent on its leak tightness. Therefore, it is necessary to maintain the leak tightness of the containment structure within permissible limits; i.e., the limits assumed in the calculations to predict the potential radiological consequences of postulated accidents. An effective containment leak testing program, then is essential to providing assurance of continuing containment integrity (i.e., a low leakage barrier) throughout a plant's lifetime. During the course of the staff's review of utility applications for nuclear power plant construction permits or operating licenses, the proposed containment leak testing programs are reviewed for compliance with the containment leak testing requirements specified in Appendix J to 10 CFR Part 50, Primary Reactor Containment Leakage Testing for Water Cooled Power Reactors. The basic elements of a containment leak testing

program include the containment integrated leak rate test (CILRT) and the local leak rate tests (LLRTs). The CILRT is conducted three times within a ten year period with the containment configuration closely approximating its post-accident condition to verify the leak tightness of the containment boundary. The LLRTs include the leak testing of containment penetrations, such as the equipment hatch, drywell head of boiling water reactor containments, personnel airlocks, piping expansion bellows and electrical penetrations, and the containment isolation valves provided to isolate system piping that penetrates containment. The LLRTs are conducted more frequently, i.e., during refueling outages, than the CILRTs since containment penetrations and isolation valves represent the more likely containment leakage paths. The proposed methods for CILRTs and LLRTs, the associated acceptance criteria and frequencies, and the identification of containment penetrations and isolation valves included in the LLRT program, are reviewed by the staff for acceptability. Proposed exceptions to the requirements of Appendix J to 10 CFR Part 50 are also reviewed by the staff, on a case-by-case basis, for acceptability.

III.B. PWR Ice Condenser Containment Designs

III.B.1. Containment Functional Design

The following issues constitute the major review areas for the ice

condenser containment functional design.

a. Short Term Containment Response

In the event of a high energy line rupture within the containment lower compartment, high energy fluid is released and transported throughout the various regions of the ice condenser containment. This flow of high energy fluid causes local pressure buildups within the subcompartments of the lower compartment, the ice condenser compartment and the upper compartment and short term differential pressures across the structures and equipment forming the boundaries of the compartments and subcompartments. The pressure magnitudes depend upon the volumes of the subcompartments, geometry of the interconnecting vent flow paths, mass flow behavior and the thermodynamic behavior within the ice condenser. During the early phase of the transient, flow through the ice condenser to the upper containment compartment is insignificant, and the upper containment compartment pressure remains near its initial pressure. It is during this time that the peak operating deck and crane wall differential pressures and peak lower compartment subcompartment differential pressures would be experienced. As the blowdown continues, the pressure in the upper compartment rises; and about 10 seconds after the start of blowdown, the upper compartment reaches a peak pressure approximately

equal to the lower compartment pressure, i.e., about 80 pounds per square inch gauge. The primary factor in producing this upper compartment pressure peak (called the compression peak pressure) is the displacement of air from the lower compartment through the ice columns into the upper compartment. These transient peak differential pressures are the controlling pressure loads for many of the containment internal structures and as such must be conservatively predicted and used as appropriate for the design pressures of the affected internal structures.

b. Long Term Containment Response

Following the rapid depressurization of a high energy system in the event of a failure of its pressure boundary, the containment response can be anayzed by relatively simple analytical methods, i.e., quasi steady state analysis. For the ice condenser containment design, the long term containment response period is characterized by low mass and energy release rates to the containment, suppression of the containment pressure by the ice condenser, and eventually melt out of the ice baskets and development of the maximum uniform containment pressure.

It is during this long term transient period that: 1) the maximum primary containment pressure; 2) the maximum reverse pressure across the internal operating deck (which provides the barrier which forces accident fluids through the ice condenser); 3) the minimum containment pressures during the recovery of the reactor core; and 4) the maximum containment temperatures are developed. Each of these items comprise major design concerns for the containment functional design of ice condenser containments. The significance of each of these items is discussed below.

1) For a loss-of-coolant accident, the containment spray system is activated after the completion of blowdown, i.e., about 30 seconds after accident initiation. After about 10 minutes, the return air fans are started and the containment pressure is reduced to approximately 6.0 pounds per square inch gauge as air is returned from the upper volume to the lower volumes. Steam from the reactor coolant system is still being removed, i.e., condensed almost entirely by the stored ice at this time. After ice meltout, which occurs a little over an

hour after initiation of the accident, steam from the reactor coolant system is removed by the containment spray system. The containment pressure will again rise to a peak about 1.5 hours after initiation of the accident, at which time the energy input to the containment atmosphere equals the minimum heat removal capability of the sprays. The magnitude of this pressure peak is determined by the heat input rate to the containment and the heat removal rate of the containment spray system. Figure III.B.1 shows a typical long term pressure profile for an ice condenser containment. It is this second peak pressure which establishes the design pressure for the primary containment building.

2) Maximum Reverse Differential Pressure

In an ice condenser plant, if the steam condensation rate in the lower compartment exceeds the steam release rate, the lower compartment will depressurize at a greater rate than the ice condenser and the upper compartments. This would result in a differential pressure between the upper compartment (higher pressure) and the lower compartment (lower pressure). This is referred to as a reverse

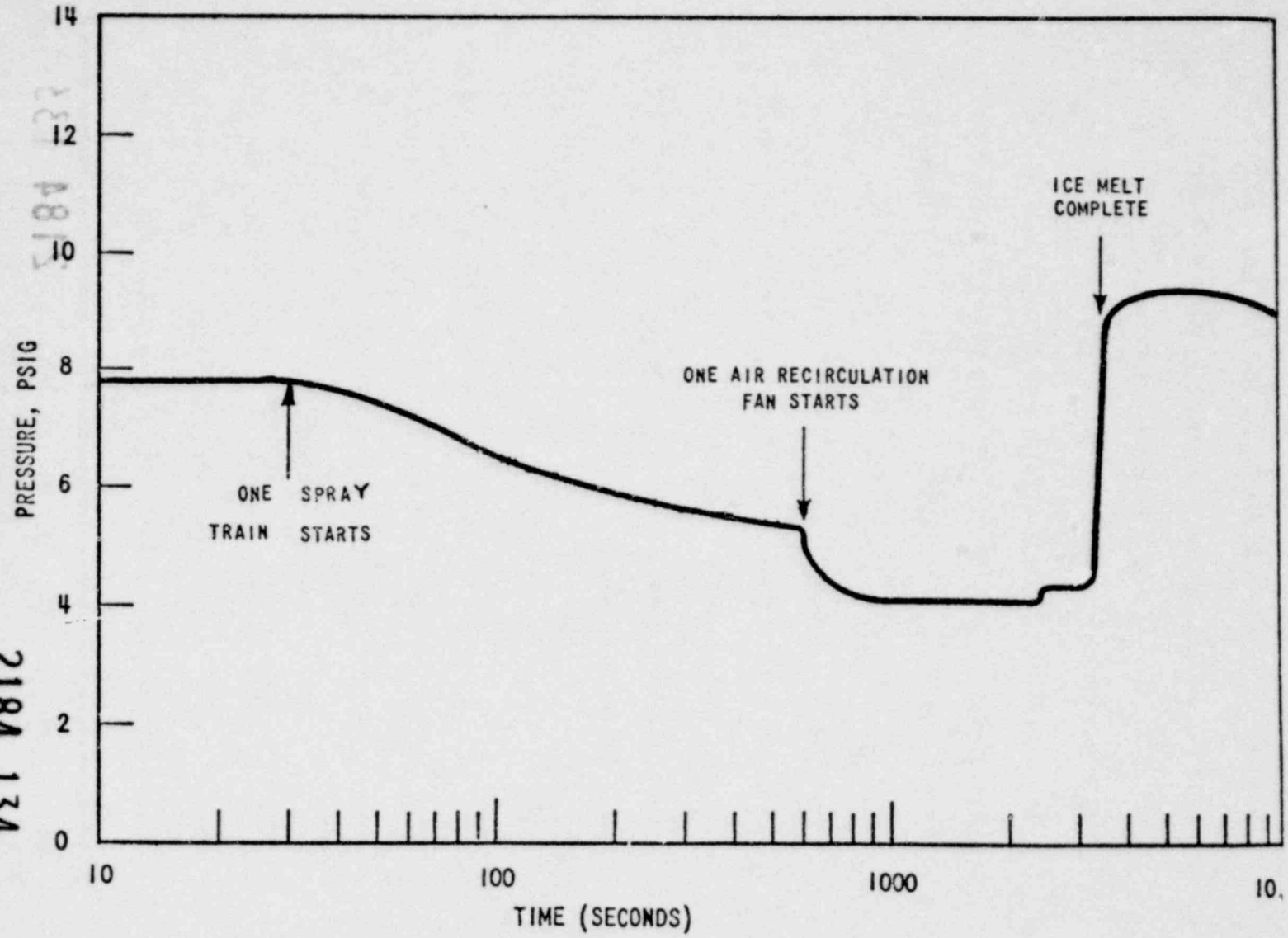

FIGURE III.B.1 - CONTAINMENT PRESSURE ANALYSIS (LONG TERM)

differential pressure, since a differential pressure between lower compartment (higher pressure) and the upper compartment (lower pressure) of varying magnitude would exist during all other containment flow conditions. The design of the operating deck and the ice condenser lower inlet doors must consider this maximum reverse differential pressure obtained from the short term response analyses.

3) Minimum Containment Pressure

In an ice condenser plant, if the steam condensation rate in the lower compartment exceeds the steam release rate into the compartment, the lower compartment pressure minimum would be reached. The potential for the steam condensation rate to exceed the release rate into the lower compartment exists during the "refill" period of a loss-of-coolant accident when the steam release rates from the reactor coolant system are very low. The accuracy of the prediction of lower compartment pressure during the refill period does not appear to be important because the ECCS evaluation model assumes adiabatic heating

of the core during this period. However, the reduction of the lower compartment pressure which could be experienced during the refill period will establish the backpressure available to the reactor coolant system at the start of "reflooding" of the core. The backpressure available to the reactor coolant system is important during the reflooding period because the performance of the ECCS is directly affected by the available backpressure. The lower the backpressure during the reflooding period the less effective will be the ECCS performance. Therefore, the calculation of the minimum containment lower compartment pressure response is important in determining the design for the ECCS.

4) Maximum Containment-Temperature

For an ice condenser containment the lower compartment temperature response for a main steam lines break may exceed that calculated for the design basis LOCA. In those cases where the combination of break size, location, and operating conditions result in a dry steam release, superheating of the containment environment is predicted to occur. The duration of the superheat temperature

transient is on the order of 600 seconds and reaches a peak temperature of about 320°F. Therefore, the equipment and instrumentation located in the lower compartment which is necessary to mitigate the consequences of a steam line break must be qualified for service in more severe environmental conditions than the design basis LOCA environment.

c. Subcompartment Design Consideration

Subcompartment design evaluation is independent of the type of containment design; therefore, the discussion given in Section III.A.1.d is also applicable here.

d. Ice Condenser Steam Bypass

In a pressure suppression containment steam is condensed in the pressure suppression medium thereby limiting the peak containment pressure. In the ice condenser design, steam released into the lower compartment is forced to flow into the ice condenser where it is condensed by the barrier formed by the operating deck and its associated seals. Since the ice in the ice condenser would ultimately be melted in the course of an accident, provisions for long-term steam condensation are required. This requirement is satisfied by redundant-

safety grade containment spray systems. The employment of sprays for long term cooling reduces the steam bypass problem to one which is of concern only until containment spray is initiated. Therefore, the potential for steam bypass of the ice condenser must be limited to a sufficiently small bypass flow area that during the period from initiation of an accident until containment spray flow is commenced, the total amount of steam which might flow to the upper compartment, bypassing the ice condenser, will not cause sufficient pressure buildup to exceed the containment design pressure.

e. Ice Condenser Component Design

It is important for each succeeding ice condenser plant that the ice condenser components be the same as those designed for the lead plant (D.C. Cook) and reviewed by the NRC during the D.C. Cook licensing review.

f. Ice Maintenance

For the ice condenser design it is important that the ice condenser, at all times during plant operation, contain at least the minimum amount of ice assumed for the analysis of the plant response to certain postulated accidents. If the amount of

ice contained in the ice condenser is less than that assumed in the plant analysis, adequate pressure suppression in the event of an accident is not assured and the resulting containment pressure may exceed the containment design value.

g. Ice Condenser Inspection

In order to provide continued assurance of proper operation of the ice condenser in the event of an accident, the ice condenser design must incorporate; 1) sufficient instrumentation to assess the performance of the ice condenser cooling system and its moving components; and 2) design provisions to allow for periodic inspections inside the ice condenser to assure proper status of passive components.

h. Containment External Pressure

In the event of the inadvertent operation of containment cooling systems (i.e., containment sprays and in the case of the ice condenser the ice condenser return airfans), the containment atmosphere would be cooled causing the containment internal pressure to be reduced and resulting in generation of an external pressure on the containment building. This must be considered in the design of the containment in order to preclude containment failure.

2. Containment Heat Removal

The review areas for the containment heat removal system are the same as discussed in Section III.A.2. However, heat removal in an ice condenser is achieved by the ice condenser and later, after ice melt-out, by containment sprays. The ice condenser design review is performed as part of the containment functional design review. The design and function of the containment spray systems for ice condenser plants are similar to other types of containment and therefore generic in nature.

3. Containment Isolation Systems

The review areas for the containment isolation systems are the same as discussed in Section III.A.3.

4. Combustible Gas Control

Due to the design of the pressure suppression containments the potential for post accident hydrogen buildup and uncontrolled burning inside the containment has been increased. The relatively small volume of the compartment containing the reactor system and the multicompartment design of the containment has necessitated the inclusion of post accident mixing systems to assure that local pockets of hydrogen in excess of the lower limit of flammability

are not formed. This has necessitated the incorporation of another safety system to assure continued containment safety.

5. Containment Leakage Testing

Containment leakage testing requirements are the same as those discussed in Section III.A.5.

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IV. TECHNICAL BASES FOR LICENSING OF PRESSURE SUPPRESSION CONTAINMENTS

A. BWR Pressure Suppression Containments

1. Containment Functional Design

a. Containment Pressure and Temperature Response

The detailed acceptance criteria and review procedures used by the staff in reviewing BWR containment pressure response are found in Sections 6.2.1.1.c. (II) and (III) of the staff's Standard Review Plan, NUREG-75/087. A brief discussion of the review procedures for verification of the acceptance criteria follows.

1) Short Term Pressure Response

For the Mark I and Mark II containments, the staff's acceptance of the short term pressure response in the drywell and wetwell is based on the results of calculations performed by the applicant and when necessary by confirmatory calculations by the staff. Calculations are performed for a spectrum of primary coolant system breaks in the reactor coolant system. The staff has determined from previous reviews that the recirculation line break constitutes the design basis loss-of-coolant accident.

The analyses by the applicant of the short term pressure response for Mark I and II containments is based on the

analytical model described in General Electric's topical report NEDO-10320, "The General Electric Pressure Suppression Containment Analytical Model" (April, 1971). This model has been reviewed by the staff and found to be acceptable. Confirmatory pressure analyses by the staff are performed using the CONTEMPT-LT computer code.

Comparisons between the staff's analytical results and GE's calculational techniques have demonstrated that the containment response analyses are conservative. Over the years, both the GE and the staff's independent analytical models have been continually improved as additional test data and new calculational procedures have been developed. As these improvements have been made, the staff determined that the changes to the calculated results were not significant with respect to the analyses for previously licensed Mark I plants.

For Mark III containments, the steam line break has been determined to be the design basis loss-of-coolant accident. However, the drywell response for the recirculation line break is also analyzed for each Mark III plant to ensure that this conclusion remains valid.

The Mark III drywell short term pressure response is based on the analytical model described in topical report NEDO-20533, "The General Electric Mark III Pressure Suppression Containment System Analytical Model." This model is similar to the model used for Mark I and Mark II containments with the exception that it contains submodels to account for unique features of the Mark III containment, such as vent clearing and vent coolant interaction. These submodels include a horizontal vent clearing model and a vent back pressure model.

Verification of the Mark III analytical models has been established by a test verification program as discussed in Section IIA to this report.

Confirmatory analyses of the short term pressure response by the staff for the Mark III drywell are performed using the CONTEMPT-LT code.

If analytical models other than the approved General Electric model identified above for Mark I, II or III containments are used, then the models must be demonstrated to be conservative. In addition, it is necessary to demonstrate the model performance with suitable test data.

2) Long Term Pressure and Temperature Response

The basis for acceptance of the long term pressure and temperature response for the wetwell of BWR pressure suppression containment designs is the review by the staff of the applicant's long term model.

This model consists of a straightforward mass and energy balance. The model accounts for potential post-accident energy sources including core decay heat, sensible heat, pump energy and metal-water reaction energy. The model also assumes that the suppression chamber atmosphere is saturated and equal to the suppression pool temperature at any time. Therefore, the suppression chamber pressure is equal to the sum of the partial pressure of air and the saturation pressure of water corresponding to the pool temperature.

The above described conservatisms in the long term pressure response analyses along with the assumption of minimum pool cooling with the Residual Heat Removal System (RHR) operating in the limiting pool cooling mode result in a conservative prediction of the peak long term wetwell pressure and temperature.

The staff has evaluated the potential for pool thermal stratification since this could result in an increased wetwell pressure and temperature. The evaluation includes consideration of the PSTF and Marviken test results. We find that "chugging" phenomenon of steam condensation agitates the suppression pool water creating good mixing. In addition, the RHR system operating in the pool cooling mode will also help pool mixing. GE is conducting a series of small break tests to investigate pool stratification. These tests are regarded by the staff as confirmatory.

Based on our review of information provided by applicants and our own confirmatory analyses for the long term response, we have developed an acceptable equipment qualification temperature envelope for BWR drywell response during a LOCA. A rapid rise to a saturation temperature of 340°F and a duration in excess of five hours is acceptable. Methods which we have found acceptable for calculating individual plant response are described in Appendix 3B to GESSAR-238 Nuclear Island PDA application and in topical report NEDO-20533, "The General Electric Mark III Pressure Suppression Containment System Analtyical Model." In addition, the CONTEMPT-LT computer code which is used by the staff is capable of performing this evaluation in a conservative manner. These

methods are appropriate for use on Mark I, II, or III containment systems.

b. Suppression Pool Makeup System (Mark III)

The staff's licensing basis for this system is our review of the system's ability to perform its required function when necessary. We have determined that this will be accomplished if the system includes the following features:

1) Two lines connect the upper pool located on top of the drywell to the suppression pool. Each line contains two normally closed valves in series which will open on a low-low suppression pool level signal in coincidence with a loss-of-coolant signal permissive. Following a loss-of-coolant accident, dumping of the upper pool water will start a few minutes following the beginning of emergency core cooling system flow and will require about five minutes for completion. The makeup system dump lines will be sized so that flow from one line exceeds the maximum emergency core cooling system pump flow.

2) The suppression pool makeup system dump valves will also be signaled to open by a loss-of-coolant accident signal in series with a 30-minute timer where the timer will be started by the loss-of-coolant accident signal. The

initiation logic will be independent of suppression pool level and will be directed at ensuring that the combined upper and lower pool volumes is available for small primary system breaks which do not lower the suppression pool to the trip level.

3) The suppression pool makeup system is designed to seismic Category I criteria and Quality Group B requirements in accordance with Regulatory Guide 1.29, "Seismic Design Classification," and 1.26, "Quality Group Classifications and Standards," respectively, and will consist of redundant dump lines and valves.

c. Reverse Pressure Response

The detailed acceptance criteria and review procedures used by the staff in reviewing BWR containment pressure responses are found in Sections 6.2.1.1.c. (II.7) and 6.2.1.1.c. (III.5) of the Standard Review Plan, NUREG-75/087. A summary of the acceptance bases is provided below.

The staff requires that analyses of BWR containment drywell and wetwell reverse pressure responses be provided by the applicant; and when necessary, confirmatory calculations are performed by the staff. The staff reviews the assumptions and initial conditions used by the applicant to assure a conservative calculation of the peak reverse pressure.

Confirmatory calculations by the staff frequently consist of boundary or end-point calculations based on conservative initial temperature and humidity conditions for the atmosphere and maximum containment spray capacity.

When vacuum breakers are utilized in the design to equalize pressure, the staff uses a more realistic model for its confirmatory calculations, including the CONTEMPT-LT and the CRYSTAL computer codes. Conservatism is introduced in the reverse pressure analyses by assuming a failure of one of the vacuum breakers. In addition, the staff reviews the proposed Technical Specifications to assure that appropriate surveillance and administrative controls will be maintained over the vacuum relief valves. This includes routine testing requirements to demonstrate the capability of these valves to perform their vacuum-relief function and the use of redundant position indicators to assure that excessive bypass leakage would not occur.

In addition to the above, the following reverse pressure considerations apply to the Mark III containment: 1) the drywell structure should be designed for an upper bound external pressure differential of 21 psid to reflect the conservative assumption of complete depressurization of the drywell to zero psig at the time of ECCS spillage following

a LOCA; and 2) for some Mark III wetwell designs which include vacuum breakers, we require that during normal plant operation an appropriate technical specification be imposed on the containment atmosphere temperature and relative humidity to reflect the vacuum breaker sizing analyses.

Dynamic pool swell considerations may be limiting for the reverse pressure design of the Mark II diaphragm.

Our basis for acceptance of the pool dynamic reverse pressure specification is described in the July 1978 Revision 44 to the Zimmer FSAR in Section I.2.3.9.

d. Subcompartment Pressure Response

Applicants are required to thoroughly describe the analytical methodology and to provide the results of the analysis in the form of subcompartment volume pressure transients and resolved loads on the major components. We review the applicant's analysis for conformance to the acceptance criteria of staff's Standard Review Plan Section 6.2.1.2, "Subcompartment Analysis." We also perform confirmatory subcompartment pressure analysis with the COMPARE computer code. The COMPARE code was developed to conservatively predict the pressurization of subcompartment volumes.

Comparisons with test data show that the COMPARE code results compare favorably with the results of subcompartment pressurization tests. There are ongoing programs to review and evaluate subcompartment pressure analysis computer codes against available test data.

The staff has a generic technical activity underway to develop detailed acceptance criteria for the calculation of asymmetric loads on reactor system components. The generic program designated Task Action Plan A-2, "Asymmetric Blowdown Loads on the Reactor Vessel" will result in acceptance criteria to address all aspects of the asymmetric loads analysis of which subcompartment pressurization is only one part. Other aspects of the asymmetric loads analysis include the addition of seismic loads and loads on the components due to system reaction to the pipe rupture. The Task Action Plan is scheduled to be completed near the end of this year.

Based on the analyses completed to date for plants licensed since 1972, the staff has determined that the Mark I BWR subcompartment loading conditions are of less safety significance than those for the PWR dry containments. However, the staff has concluded that these potential loading conditions

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should be reevaluated for all operating Mark I BWRs.

For facilities not yet licensed for operation, applicants are being required to perform analyses using the methodology and criteria currently available for these designs and to implement any design modifications which are necessary prior to the issuance of an operating license. Applicants are also required to commit to perform an evaluation of their facility following staff approval of generally applicable analysis methods should the methodology or criteria warrant such reevaluation.

For operating facilities, this issue will be addressed by plant-unique analyses utilizing the above-mentioned generic analysis methods. It is expected that this reevaluation will be completed within the next two years for all operating Mark I BWR facilities.

Acceptance criteria for subcompartment pressure analysis are being developed with the aid of our consultants at the Los Alamos Laboratories in New Mexico. Areas under review include sensitivity of the various calculated input parameters to the COMPARE code and sensitivity of results to the modeling of the subcompartment volume which may affect the calculated pressure distribution within the subcompartment.

Pending development of the final criteria for asymmetric loads analysis, continued operation and licensing of plants are justified on the basis that the probability of a severe pipe rupture resulting in substantial loads on the reactor vessel or the structures is acceptably small. Our conclusion that the probability of a severe pipe rupture is acceptably small is supported by the results of analysis performed by the reactor vendors and by the WASH-1400 study.

In summary, pending development of final acceptance criteria, assurance of design adequacy with regard to subcompartment pressure loads can be justified by:

1) Review of the applicant's subcompartment pressure analysis for conformance with current acceptance criteria;

2) NRC confirmatory analysis of the subcompartment pressure response; and,

3) The results of probabilistic analysis which demonstrate the likelihood of a severe pipe rupture to be acceptably small.

e. Steam Bypass of the Suppression Pool

The detailed acceptance criteria and review procedures used by the staff in reviewing steam bypass of the suppression pool are described in Sections 6.2.1.1.C. (II.4), 6.2.1.1.C. (II.5),

6.2.1.1.C. (III.3), and 6.2.1.1.C. (III.4) of the Standard Review Plan, NUREG-75/087 and in our proposed Branch Technical Position, "Steam Bypass for Mark II Containments." A summary of our acceptance bases for steam bypass for each of the BWR suppression containment designs is described below.

1) Mark I Containments

Vacuum breakers located in the vent header of the Mark I containments have been identified as the primary contributor to potential bypass leakage. To minimize this potential for bypass leakage, the staff requires the following for each Mark I facility: 1) operational testing of the vacuum breakers once each month; 2) performance of a leakage test of the drywell to torus vent system at the end of each refueling outage; and 3) provision of redundant position indication in the main control room for each drywell to torus vacuum breaker. The above described testing requirements are incorporated in the facility Technical Specifications.

In addition, operating BWR plants with Mark I containments have been operating with a positive pressure differential between the drywell and suppression chamber since February 1976. Maintaining this pressure differential provides a mechanism for continuously monitoring the

amount of bypass leakage and a verification of the status of the drywell to suppression chamber vacuum breakers.

2) Mark II Containment

The potential paths for steam bypass in a Mark II containment include the following: cracks in concrete diaphragm, around the diaphragm seal-ring, drywell floor vacuum breakers, around penetrations in the diaphragm and through the hydrogen recombiner.

Our bases for evaluating Mark II pool bypass are described in the proposed Branch Technical Position, "Steam Bypass for Mark II Containments." These bases include the following:

a) Transient bypass analyses are performed by the applicant to show a minimum small break capability of the order of .05 ft^2. Confirmatory calculations are performed by the staff using CONTEMPT-LT when necessary.

b) A preoperational high pressure leakage test should be performed at a pressure near the drywell-wetwell pressure differential.

c) A low pressure operational leakage test should be performed at each refueling outage.

d) The acceptance criterion for the leakage tests is that the measured leakage should be less than ten percent of design capability.

e) Redundant position indicators should be placed on all vacuum breakers with indication and redundant alarms in the control room.

f) The vacuum relief valves should be tested for operability at monthly intervals.

g) The wetwell spray system should be automatically actuated ten minutes following a LOCA.

Recombiners used in Mark II containments may introduce another potential path for steam bypass of the suppression pool, since they take suction from the drywell and return the processed drywell atmosphere to the wetwell. The recombiner design and operation are reviewed by the staff to ensure that the potential for bypass through the recombiner is minimized and within our acceptance criteria.

3) Mark III Containment

There are two potential paths for steam bypass of the suppression pool in the Mark III containment. These are

a postulated break of high certain energy lines that pass through the wetwell steam volume above the suppression pool and postulated cracking of the concrete in the drywell structure. Our acceptance criteria for high energy lines were developed for guarded and unguarded lines.

We find acceptable the use of guard pipes around high energy lines or other types of protective structures to assure that the suppression pool is not bypassed. If guard pipes are used, they should be designed in accordance with the acceptance criteria described in Section 3.6.2 of the Standard Review Plan, NUREG-75/087.

For unguarded high energy lines, we review the analyses of the consequences of postulated ruptures in these lines. Our acceptance is based on the conservatism of the methods and the assumptions used. If leakage detection and isolation are provided, as is the case for the Reactor Water Cleanup System, we evaluate the effectiveness of the detection instrumentation and isolation devices to mitigate the consequences of a pipe rupture.

In regard to the bypass leakage associated with the potential

for cracking of the concrete in the drywell structure or other leakage paths around penetrations, we find that the Mark III containment should have an allowable bypass area of approximately one square foot for the complete spectrum of reactor coolant system pipe breaks.

To mitigate the effects of steam bypass, a heat removal system is necessary. Currently, there are two types of heat removal systems proposed by the applicants. One is the containment spray; the other is air coolers. Either type of heat removal system is required to have sufficient cooling capacity to meet our specified containment bypass capability. The containment spray should be actuated automatically ten minutes following a LOCA.

In addition, the drywell will be leak tested at about its design pressure and at a low pressure prior to plant operation. Moreover, low pressure leak tests of the drywell will be performed periodically during plant lifetime. The acceptance criteria for the tests should be based on the measured leakage being less than ten percent of the capability of the containment to accommodate bypass leakage at the test pressure.

Other potential steam bypass paths for the Mark III containment consist of the mixing system penetrations in the drywell. To limit bypass through these penetrations, the mixing system design is required to have the following features: small drywell penetrations; two valves in series on each line to assure isolation capability and the capability to accommodate steam bypass through an inadvertently opened recirculation line.

f. Suppression Pool Hydrodynamic Loads

1) LOCA Pool Dynamics

Programs to resolve the LOCA pool dynamic concerns were developed for each BWR suppression containment. These programs are summarized below.

a) Mark I Containment

After the NRC staff had concluded that there was no immediate safety problem for the Mark I facilities, it was determined in April 1975 that a complete reassessment of the Mark I containment system design by the affected utilities should be conducted. This reassessment was divided in two phases, (1) a short-term program (STP), designed to confirm the adequacy

of the containment system for each operating Mark I BWR facility to maintain its integrity and functional capability during a postulated LOCA event; and, (2) a long-term program (LTP), designed to establish design basis loads appropriate for the intended life of each Mark I BWR facility and to restore the originally intended design safety margins for each Mark I containment system.

In order to evaluate the magnitude and significance of these newly identified loads, affected facilities formed an "ad hoc" Mark I Owners Group. Task Action Plans (A-6 and A-7) which describe and document the NRC staff's programs for the conduct of the Mark I STP and LTP evaluations were developed in June 1977 and included in NUREG-0410, "NRC Program for the Resolution of Generic Issues to Nuclear Power Plants."

During the STP review, whenever structural safety margins were found to be less than a factor of two at an operating Mark I BWR facility the safety margins were required to be increased. One of the methods used to accomplish this has been the use of drywell to torus differential pressure control, which reduces the dynamic

pressure loads on the pressure-suppression chamber and its supports. In addition, several utilities have voluntarily performed modifications to their torus support system to provide additional design safety margin. A list of these modifications is provided in Table IV A.1.

The NRC has completed its review of the generic Mark I containment Short Term Program (STP) conducted by the Mark I Owners Group and the associated plant-unique information provided by the licensees of operating Mark I BWR facilities. The results of this review are documented in the staff's "Mark I Containment Short Term Program Safety Evaluation Report," NUREG-0408, December 1977.

Based upon its review, the NRC has concluded that licensed Mark I BWR facilities can continue to operate safely, without undue risk to the health and safety of the public, during an interim period of approximately two years while a methodical, comprehensive LTP evaluation is conducted. This conclusion was based on the determination that: (1) the magnitude and character of each of the hydrodynamic loads resulting from a postulated design basis loss-of-coolant accident (LOCA)

have been adequately defined for use in the STP structural assessment of the Mark I containment system; and (2) for the most probable loads induced by a postulated design basis LOCA, a safety factor of at least two exists for the weakest structural or mechanical component in the containment system for each operating Mark I BWR facility.

Although the structural and mechanical components of the containment system for each operating Mark I BWR facility meet the STP structural acceptance criteria (i.e., a safety factor of at least two), certain components in each facility's containment system do not meet the ASME Code allowable stress limits. Consequently, the NRC concluded that the demonstrated safety margin of the containment systems for operating Mark I BWR facilities does not comply with the current interpretation of "sufficient margin" as prescribed in General Design Criterion (GDC) 50, "Containment Design Basis," of Appendix A to 10 CFR Part 50 and, therefore, is not sufficient for long term reactor operation.

However, since (1) Mark I BWR containment systems still retain sufficient margin under present conditions

to preclude failure under LOCA-related hydrodynamic suppression pool loads and thus provide reasonable assurance of no undue risk to the health and safety of the public, (2) the objective of the LTP, i.e., to restore the originally intended design safety margins for each Mark I containment system, is acceptable, (3) the Mark I Owners Program Action Plan for the LTP is reasonably designed to satisfy the LTP objective, and (4) there appeared to be no safety problem or public interest consideration favoring restriction of the operation of Mark I BWR facilities, the NRC on February 28, 1978 granted the licensees of operating Mark I BWR facilities exemptions from GDC-50, with respect to LOCA-related hydrodynamic suppression pool loads, for an interim period until completion of the LTP (approximately two years). These exemptions provide for continued operation under the conditions specified in NUREG-0408 and under any resulting Technical Specification requirements.

The generic Mark I LTP program commenced in June 1976 and is scheduled for completion in 1979. Revision 3 of the Mark I Containment Program Action Plan, which was submitted to the NRC on February 15, 1978 by GE on behalf of the Mark I Owners Group, describes

the objectives and schedules for each of the LTP analytical and testing programs. The LTP includes activities to obtain improved load definition for the current Mark I containment system design and to identify and qualify potential load mitigating devices and/or procedures which may be used to supplement the existing design.

The NRC continues to closely follow the progress of the LTP to assure that appropriate actions are taken in a timely manner. In addition, the NRC has sponsored several testing and analytical programs to provide independent confirmatory load definition and structural response information for use in its evaluation of the results of the Mark I Owners LTP. These include the test program recently performed at the Lawrence Livermore Laboratories in California.

TABLE IV A.1

LISTING OF MODIFICATIONS TO
MARK I TORUS SUPPORT SYSTEM COMPONENTS

Plant	Proposed Modification	Status
DUANE ARNOLD	1. Uplift anchor brackets were added during summer 1976	1. Complete
BROWNS FERRY UNITS NOS. 1, 2, and 3	1. Single vent header columns changed to twin columns	1. Complete
	2. SRV line restraints strengthened	2. Complete
COOPER	1. May 1976 - Web reinforcing plates added to each side of the torus support column at the column-torus shell weld	1. Complete
DRESDEN UNITS NOS. 2, 3	1. Jackets added to inner torus support column and bearing blocks installed in inside pin connections - completed at Unit No. 2 on 9/7/76 and at Unit No. 3 8/20/76	1. Complete
FITZPATRICK	1. Checkered plate catwalks to be removed	1. Complete
HATCH UNIT NO. 1	1. Strengthened the torus support column to torus shell welds and reinforced the torus support column connection to the torus by adding gusset plates.	1. Complete
	2. Installed anchor bolts in the base plate of each torus support column	2. Complete
MILLSTONE UNIT NO. 1	1. Two new anchor bolts added on both inner and outer torus support columns	1. Complete
	2. Jackets added to inner torus support column and bearing blocks installed in inside and outside pin connections	2. Complete
	3. Replacement of two pipe supports with spring hangers on atmospheric control line	3. Complete
MONTICELLO	1. Jackets added to inside and outside torus support column, bearing blocks installed in inside and outside pin connections, additional anchor bolts installed	1. Complete

Plant	Proposed Modification	Status
	2. Strengthened torus support column to torus shell welds and reinforced the upper torus support column connection to the torus	2. Complete
NINE MILE POINT UNIT NO. 1	1. Checkered plate catwalks removed	1. Complete
PILGRIM UNIT NO. 1	1. Gusset plates have been added to the ring girder web in the area of the outer torus column to shell attachment	1. Complete
QUAD CITIES UNITS NOS. 1, 2	1. Addition of weld material to existing web and flange welds at the torus column to torus shell connection completed at Unit No. 1 on 1/3/76 and at Unit No. 2 during the 9/13/76 refueling outage	1. Complete
VERMONT YANKEE	1. Modifications were made in March 1976 to provide tie-down to the torus support columns	1. Complete
PEACH BOTTOM UNITS NOS. 2, 3	1. Saddle supports added to Unit No. 2 in April-May 1976	1. Complete
	2. Similar modifications were made at Unit No. 3 during the refueling outage at the end of 1976	2. Complete
HATCH UNIT NO. 2	1. Saddle supports added to the torus in January 1977.	1. Complete
	2. Vent header support column connections strengthened	2. Complete
FERMI UNIT NO. 2	1. Checkered plate catwalks to be removed	1. Will be completed prior to issuance of Operating License
	2. Several other design modifications are currently under consideration by applicant	2. Will be completed prior to issuance of Operating License

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b) Mark II Containment

Facilities utilizing the Mark II containment system design are currently under construction; several have submitted applications for an operating license. There are no domestic Mark II facilities in operation.

In April 1975, it was determined, by the staff, that a complete reassessment of the Mark II facilities should be conducted. At that time, the staff sent letters to each domestic facility with a Mark II containment requesting that they provide information demonstrating the adequacy of their containment. As a result, the Mark II containment owners, formed an "ad hoc" Owners Group to respond to the NRC requests. The utility owners recognized that the additional evaluation work would be very similar for all plants. The formation of an Owners Group established a uniform program to respond to the NRC inquiries as quickly as possible.

Their generic program consisted of a Lead Plant Licensing Program (LPP) and a Long-Term Program (LTP).

The objective of the LPP is to establish design basis

(conservative) loads appropriate for the anticipated life (40 years) of each Mark II BWR facility. The licensing activities for certain Mark II lead plants (Zimmer, Shoreham and LaSalle) precede completion of the entire Mark II containment supporting program. The LPP was developed so as to demonstrate that a sufficient understanding of the pool dynamic phenomena and principles of interest exists to establish conservative loads for the lead plants. Because of the LPP emphasis on developing loads consistent with the licensing requirements of the lead plants, for many of the pool dynamic loads, a bounding interpretation of the test data was utilized. This assures that conservative loads are used for the lead plants.

The Mark II owners specification of the LPP loads are documented in the "Mark II Containment Dynamic Forcing Function Information Report (DFFR)", NEDE-21061 and NEDE-21061-P and its revisions.

The objectives of the LTP are: (1) to provide justification through tests and analyses for a reduction in selected design basis boundary loads of the LPP; (2) establish design basis loads for the GE quencher for non-lead plants; and, (3) provide

additional confirmation of certain loads utilized in the LPP.

The analytical and test programs supporting the loads specified in the DFFR are described in the "Mark II Containment Supporting Program Report", NEDO-2129.

In addition to the Mark II owner's generic programs to establish pool dynamic loads, each Mark II owner provides to the NRC a plant-unique Design Assessment Report (DAR).

The functions of the individual DARs are to:

1) Describe the plant-unique application of the generic Mark II pool dynamic loads methodology;

2) Establish pool dynamic loads excluded from the Mark II owners' generic programs; and,

3) Provide an evaluation of the response of the structures, piping and equipment in each Mark II plant to pool dynamic loads to demonstrate that the facility can safely accommodate these loads.

Individual Mark II owners issued these reports late in 1975 and the first quarter of 1976. These reports will be updated and revised as additional information becomes available from the Supporting Program and at the completion of the LPP for the lead Mark II plants and the completion of the LTP for the remaining Mark II plants.

The staff evaluation of the Mark II owners' generic LPP and LTP is being conducted under the NRC's Technical Activities Program titled, "Mark II Containment Pool Dynamic Load," (A-8) as described in NUREG-0410

The staff's current schedule for review of the loads proposed by the Mark II owners includes issuance of LPP acceptance criteria in August 1978 and issuance of the LPP Safety Evaluation Report in October 1978.

The Design Assessment Reports are being reviewed as part of the staff's review of the individual Final Safety Analysis Reports for each Mark II facility. These reviews are scheduled to be completed prior to the conclusion of the staff's reviews of the individual applications for operating licenses. Initial operation of the first domestic Mark II facility, Zimmer, is currently scheduled for mid-1979.

Our review schedule associated with the Mark II owners' Long-Term Program calls for issuance of a generic safety evaluation report in mid-1980.

Utilities are making modifications to their Mark II plants on the basis of information already derived from the ongoing Mark II pool dynamic programs. Typical modifications are given in Table IV.A.2.

c) Mark III Containments

General Electric conducted tests and performed analyses to establish LOCA pool dynamic loads for the Mark III containment design. These loads are described in NEDO-11314-08, "Information Report Mark III Containment Dynamic Loading Conditions."

The staff reviewed these loads as a part of its review of the GESSAR-238 Nuclear Island, preliminary design application. Based on this review, the staff issued a set of design criteria which we found to represent acceptable bases for issuance of a preliminary design approval or construction permit. These design criteria are described in Section 6.2.1.9 of NUREG-75/110, "GESSAR-238 Nuclear Island Standard Design Safety Evaluation Report" and the modified criteria for impact loads described in Supplement 2, Section 6.2.1.9 to the GESSAR Safety Evaluation Report.

Table IV.A.2

LISTING OF TYPICAL POOL DYNAMIC RELATED MODIFICATIONS RELATED TO MARK II FACILITIES

1. Ring stiffeners welded to interior walls of steel containment, additional reinforcement added to concrete containment walls and additional reinforcement bars added to drywell floor.

2. Safety/relief valve (SRV) lines rearranged symmetrically around the suppression chamber, horizontal runs of SRV lines rerouted up close to the drywell floor, steel framing system redesigned for support of the SRV lines in the pool and increased thickness of SRV lines.

3. SRV discharge devices changed from ramshead to quencher device.

4. Reactor support pedestal modified by filling inner core with concrete and reinforcing bars. For some designs, large holes in pedestal eliminated.

5. Modifications in the vent and vent support system. Vent bracing system redesigned, and flanges removed, vents shortened and vent wall thickness increased.

6. Equipment end piping in the suppression chamber removed, rerouted and redesigned. Includes gallery platforms, HVAC ducting, vacuum breakers.

7. Drywell steel framing stiffened or replaced, steel framing support modifications.

8. Pipe restraints in the suppression chamber redesigned.

9. Snubbers for primary system and BOP equipment in the drywell and reactor building upgraded or relocated in a number of locations.

10. Additional suppression chamber instrumentation for SRV in plant testing and pool temperature monitoring.

Additional large-scale tests are planned for the Mark III design as discussed below:

1) A series of liquid blowdown tests will be conducted to indicate comparability to steam blowdowns;

2) A series of small break tests will be conducted to investigate pool stratification and vent chugging effects;

3) Tests will be performed with the suppression pool at an initial elevated temperature to determine steam condensation characteristics under such conditions; and,

4) A multi-vent series will be run employing a test section of three columns of three, nine-inch vent (one-ninth scale by area), to consider possible vent interactions.

We consider the basic design and performance of the Mark III containment system to be well established based on our review of the analytical models and the available margins incorporated in the design. It is our view that those phenomena that are being addressed in the future Mark III tests merit the additional evaluation,

but that they do not represent design governing conditions at this time nor, in our judgment, will they escalate into design basis considerations as a result of these tests. In summary, we consider the remaining Mark III testing to be confirmatory in nature and will require that the tests and our evaluation of the test results be completed prior to concluding our review on the first operating license for a Mark III plant.

The Advisory Committee on Reactor Safeguards (ACRS) has commented on the progress of the confirmatory test program. In particular, the ACRS emphasized the importance of developing analytical models based on a first principles approach which can be used in conjunction with empirical test results. We are following this matter with GE on a generic basis.

2) Safety/Relief Valve Pool Dynamics

Numerous test programs have been and are being conducted to determine the magnitude and frequency of relief valve induced loads for the various relief valve discharge devices to be used on the Mark I, II and III containments. The NRC staff has these programs under active review.

A generic Task Action Plan, A-39, has been initiated to investigate the SRV pool dynamic loads and allowable pool temperatures for SRV discharges for BWR designs. This plan is included in NUREG-0410.

The status of our review of the Mark I, II and III containments in the SRV related areas is summarized below.

a) Mark I Containments

Currently, all Mark I containments, with the exception of the Oyster Creek Plant, which uses a quencher type device, use the ramshead discharge device. However, the Mark I owner's Long-Term Program Task (LTP) includes the development of a quencher type device, which will be installed as necessary pursuant to the results of the Mark I containment LTP.

In-plant tests of a "T-quencher" device have been performed in the Monticello plant as a part of this program. Preliminary results from these tests indicate that the peak positive pressures for the T-quencher are about 25% of the ramshead pressure, while the peak negative pressures are about 50% of the corresponding ramshead pressure. Parametric testing is currently being conducted in a small scale (1/4) test facility. The results of these tests will be used

to validate an analytical model for T-quencher load definition.

The relief valve loads are cyclical in nature, creating the potential for fatigue degradation of the containment. For the operating Mark I plants, we have determined that there is sufficient fatigue margin to permit continued plant operation while a new discharge device is being developed, or until the plant's capability to withstand the loadings from the existing discharge device for the anticipated (40 year) life of the facility can be demonstrated. In addition, SRV operating experience has shown that in all but a few instances, SRV discharge devices have performed satisfactorily without any evidence of damage either due to the hydrodynamic loads or the pool temperature effect. In those isolated cases where localized damage has been experienced, the damage did not result in either a loss of the containment function or any release of radioactivity. In those cases, repairs were made and additional margin was included in the capability of the structures.

The operating Mark I facilities currently have Technical Specifications which limit the temperature of the

suppression pool during normal plant operation. In addition, the bases for these Technical Specifications describe the administrative controls and corrective actions to be taken to minimize the temperature response of the pool in the event that an SRV opens and remains open.

In December 1977, each Mark I licensee was requested to provide plant-specific analyses of the temperature response of the suppression pool for a series of design basis operational transients. These analyses were requested to determine the adequacy of the existing Technical Specification requirements. The results of all of these analyses are expected to be submitted by August 1978.

With respect to postulated multiple-sequential relief valve actuations, a preliminary assessment of this loading combination for a typical Mark I plant has been performed. It indicated that the containment designs are able to accommodate this new load combination. In addition, plant-unique analyses are currently being performed to verify that each plant has the capability to withstand this new loading combination. In the interim, continued operation of the Mark I plants is

supported by the preliminary assessment and by operating experience, i.e., very few multiple-sequential SRV actuations have occurred and, even in those few cases, structural damage has not resulted.

b) Mark II Containments

Plants with Mark II containments, of which there are none in operation, will be required to demonstrate the capability of satisfying the acceptance criteria currently being developed by the NRC staff. The applicants for all the Mark II plants currently under OL review have committed to install the quencher design. The load criteria will be verified by scaled testing programs and in-plant tests. These testing programs are discussed in Section II.A.1 of this report.

The Mark II owners have indicated that they will demonstrate the containment capability to accommodate the effects of multiple-sequential relief valve actuations.

c) Mark III Containments

All applicants for the Mark III designs propose use of a quencher type discharge device which has been tested to justify the loads used in the containment structural design. We have issued acceptance criteria

for evaluation of SRVs with this quencher discharge device as a result of our evaluation of the supporting data base. Although we believe that the loads criteria are conservative, we will require in-plant tests for confirmation.

Our evaluation of the loads for the Mark III quencher device is documented in Section 6.2.1.9, Supplement 1 to "GESSAR-238 Nuclear Island Standard Design Safety Evaluation Report", NUREG-75/110.

With regard to potential multiple-sequential relief valve actuations, General Electric has developed a program for resolving this concern for the Mark III containment design. This program involves a modification to the relief valve control logic such that the current load criteria can be maintained. We are actively reviewing the approached proposed by GE. Although we have not completed our review, we believe that such an approach is technically feasible. Since all Mark III containments will be equipped with quencher devices, the pool temperature limits will not be an area of concern on the basis of current Mark III designs.

g. Secondary Containment Functionability

The acceptance criteria and review procedures used by the staff in its review of secondary containment system designs are described in Section 6.2.3 (II) and 6.2.3 (III) of the Standard Review Plan, NUREG-75/087. A summary of our basis for review is provided below.

The secondary containment systems are considered to be engineered safety features. Staff acceptance of these systems is based on a review of their design provisions for redundancy, power source, single failure protection, Quality Group B and seismic Category I criteria similar to those of other engineered safety features.

Following a postulated loss-of-coolant accident, the pressure in the secondary containment could increase due to inleakage of air, air expansion due to the heat generated by operating equipment, and the starting time specified for the secondary containment gas treatment system. We review the applicant's analysis and assumptions of the secondary containment pressure transient. If necessary, the staff performs confirmatory analyses of the pressure transient.

We also review the applicant's proposed secondary containment system testing program and the surveillance requirements in the proposed Technical Specifications to assure that tests will be periodically conducted to verify that the prescribed negative pressure can be uniformly maintained through the secondary containment.

Although the primary containment is enclosed by the secondary containment, there are systems which penetrate both the primary and secondary containment boundaries creating potential paths through which radioactivity in the primary containment could bypass the leakage collection and filtration systems associated with the secondary containment. A number of these lines contain physical barriers or design provisions which can effectively eliminate leakage, such as water seals, closed seismic Category I piping systems, or vent return lines to controlled regions. The criteria by which potential bypass leakage paths are evaluated has been set forth in Branch Technical Position CSB 6-3, "Determination of Bypass Leakage Paths in Dual Containment Plants."

2. Containment Heat Removal System

The acceptance criteria and review procedures used by the staff in its review of containment heat removal system designs are described in Section 6.2.2 (II) and 6.2.2 (III) of the Standard Review Plan, NUREG-75/087 and the General Design Criteria 38, 39 and 40 of Appendix A to 10 CFR Part 50. A summary of our basis for review is described below for significant review areas.

The containment heat removal system is an engineered safety feature. Staff acceptance is based on a review of the design provisions for redundancy, power source, single failure protection, Quality Group B and seismic Category I criteria similar to those of other engineered safety features.

The staff reviews the applicant's analysis of the Net Positive Suction Head (NPSH) for the containment heat removal pumps. These calculations are compared with the provisions of Regulatory Guide 1.1 for staff acceptance.

The type of insulation, placement of suction lines and design of the suction strainer assembly are reviewed and evaluated by the staff to assess the potential for clogging the RHR system suction line.

3. Containment Isolation System

The containment isolation system is normally designed to automatically isolate the containment atmosphere from the outside environment under accident conditions. Double barrier protection, in the form of two isolation valves in series or a closed system and isolation valves, are provided to assure that no single active failure will result in the loss of containment integrity.

Our review of this system includes consideration of the number and location of isolation valves, the valve actuation signals and valve control features, the positions of the valves under various plant conditions, and the environmental design conditions specified for the design of the components. In addition, the containment isolation system components, including valves, controls, piping and penetrations, must be protected from internally or externally generated missiles, water jets, and pipe whip. Finally, we assure that these systems are designed to the ASME Boiler and Pressure Vessel Code, Section III, Class 1 or 2, and are classified as seismic Category I systems.

The containment isolation provisions, relative to valve number and location, for most lines penetrating containment must conform to the requirements of the General Design Criteria (GDC) 55, 56, or 57.

However, there are systems, which either because of their performance or design constraints, have isolation provisions which do not explicitly conform with the valving arrangement outlined in GDC 55 and 56. But, as provided for in GDC 55 and 56, other alternative design arrangements can be found acceptable. Alternative isolation designs that we find acceptable are documented in Section 6.2.4 (II) of the staff's Standard Review Plan. Several of these alternative acceptance criteria are summarized below:

1) Lines that must remain in service following an accident and lines which should remain in service during normal operation for safety reasons are provided with at least one isolation valve. A second isolation boundary is formed by a closed system outside the containment.

2) Where a closed system outside the containment forms the second isolation boundary, each of the systems and all components which form its boundary are designed to Quality Group B and seismic Category I criteria. Valves which isolate the branch lines of these closed systems outside containment are normally closed and under strict administrative control.

3) On some ESF or ESF related system, remote manual valves

are used in lieu of automatic valves since these lines must remain in service following an accident. Where remote manual valves are used, leakage detection capabilities are provided.

4) On some penetrations the containment isolation provisions consist of two valves in series, both of which are outside the containment. The location of a valve inside containment would subject it to more severe environmental conditions (including suppression pool dynamic loads) and it would not be readily accessible for inspection.

Finally, most plants have found a need to purge the containment atmosphere during normal plant operation. However, because of the size of these direct paths to the outside environment, special attention is given to these lines. The bases for our acceptance of the design of these lines are outlined in Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operations," given in SRP 6.2.4, "Containment Isolation System."

4. Combustible Gas Control

The detailed acceptance criteria and review procedures used by the staff in reviewing the combustible gas control systems are found in Sections 6.2.5 (II) and 6.2.5 (III) of the Standard Review Plan, NUREG-75/087, Branch Technical Position CSB 6-2, and in the proposed change to the regulations; i.e., 10 CFR 50.44, "Standards for Combustible Gas Control Systems in Light Water Cooled Power Reactors." A brief discussion of the review procedures for verification of the acceptance criteria follows.

Combustible gas control systems are considered to be engineered safety features. Staff acceptance of these systems is based on a review of their design in terms of the provisions for redundancy, power source, single failure protection, and Quality Group B and seismic Category I criteria similar to those of other engineered safety features.

An additional basis for the staff's acceptance of the combustible gas control systems is the staff's review of the applicant's analyses of the production of oxygen and hydrogen in the containment during a LOCA, to see that the recommendations and guidelines of BTP CSB 6-2 have been followed. The Branch Technical Position describes acceptable methods for predicting the generation of hydrogen and oxygen in the containment during a LOCA. With regard

to the extent of metal-water reaction that should be considered, the analysis should be done in accordance with 10 CFR 50.44 and the results of ECCS evaluations performed in accordance with Appendix K, 10 CFR 50, "ECCS Evaluation Model." As necessary, the staff performs confirmatory analyses of combustible gas productions and accumulations. These confirmatory analyses are done using the COGAP computer code. A description of the COGAP code is attached as Appendix A to Section 6.2.5 of the Standard Review Plan.

These analyses of the combustible gas production are performed to establish the operational requirements (i.e., start time, capacity etc.) for the combustible gas control systems.

a. Mark I Containments

In addition to the above described acceptance bases, the following additional acceptance bases apply to our review of Mark I containments.

If a purge system is used as the primary method for controlling combustible gas concentrations in the containment, purge doses are calculated by the applicant and reviewed by the staff for consistency with the guidelines provided in BTP CSB 6-2. If a containment atmosphere dilution (CAD) system is used, the staff reviews the resulting containment repressurization to assure that the resulting peak containment

pressure is limited to a fraction of the containment design pressure.

b. Mark II and III Containments

Mark II and III containment combustible gas control systems include features which could result in potential steam bypass of the suppression pool. In the Mark II containment, recombiners may introduce a potential path for steam bypass. The Mark III mixing system may also introduce a bypass path. Our acceptance criteria related to these steam bypass concerns are discussed in Section IV.A.1.e of this report.

5. Containment Leak Testing

The containment leak testing programs of nuclear power plants are reviewed in sufficient detail by the staff to permit a conclusion that the proposed program complies with the requirements of Appendix J to 10 CFR Part 50, Primary Reactor Containment Leakage Testing for Water Cooled Power Reactors, or that any proposed exceptions to the requirements of Appendix J are fully justified and acceptable. The staff's program of reviews in this area is the same for all the containments of licensed nuclear power plants since they all have leakage limits and they all have containment penetrations.

Compliance with or the adoption of acceptable alternatives as exceptions to the requirements of Appendix J provides adequate assurance that the containment leak tightness can be verified throughout service lifetime and that the leakage rates will be periodically checked during service on a timely basis to maintain such leakage within specified limits. Maintaining containment leakage within such limits provides reasonable assurance that, in the event of any radioactivity release within the containment, the leakage of the containment atmosphere through leak paths will not be excess of the limits specified for the site.

There is an ongoing effort under generic Task A-24, "Containment Leak Testing" to revise Appendix J to provide better guidance to the industry in developing effective containment leak testing programs. The outcome of this task will be applicable to all nuclear power plants depending on their licensing status and design.

B. Ice Condenser Containments

1. Containment Functional Design

The staff's technical licensing basis for each of the major review areas for the ice condenser containments are addressed in this section.

a. Short Term Containment Response

We require applicants to use the Westinghouse Electric Corporation's TMD computer program to calculate the short term pressures, temperatures, heat transfer rates, and mass flow rates as a function of time and location throughout the ice condenser containment, including the containment compartments, following both a postulated loss-of-coolant and a postulated main steam line break accident. The containment model used includes a nodalization scheme of about 50 elements representing the containment to analyze the pressure response of each of the subcompartments within the lower compartment, including dead-ended compartments, the ice condenser compartment and the upper compartment.

The TMD code was developed to analyze specifically the short term pressure response of the ice condenser containment. The mathematical modeling in TMD is similar to that of the SATAN-V blowdown code in that the analytical solution is developed by considering the conservation equations of mass, momentum and energy, and the equation of state, and

uses the control volume technique for simulating spatial variation. The governing equations used in TMD are somewhat different from those in SATAN-V in that a two-phase (liquid water droplets and steam-air vapor), two-component (air-water) system is considered. TMD calculates the critical flow of a two-component, two-phase fluid (air, steam, and water) assuming thermal equilibrium conditions. However, a correction factor, which was determined by Westinghouse to account for experimental data on applicable flow regimes, is then applied to the calculated critical flow. The correction factor as used in the code increases the critical flow up to 20 percent through the compartments as the quality of the fluid decreases. The applicant refers to this increased critical flow as "augmented" flow. The net effect results in a lower compartment differential pressure when compared to a nonaugmented flow regime. The use of the augmented flow factor results in less conservatisms than use of the thermal equilibrium correlation.

During our review of the Ice Condenser Containment Test Programs and the design of the lead ice condenser plant (D. C. Cook), we reviewed the TMD code described in a Westinghouse topical report (WCAP-8077) and concluded that

the TMD code is acceptable for containment pressure response calculations if the nonaugmented critical flow correlation is employed. We have therefore required that short term ice condenser containment pressure response analyses be performed using the latest version of the TMD code with a nonaugmented or unity flow correlation. The latest version of the TMD code uses the new heat transfer correlation derived from the 1974 full scale tests with the one-inch by one-inch perforated metal ice basket, and a compressibility factor which is used with the subsonic incompressible flow equations to include the effects of compressible fluid flow.

Short term ice condenser response analyses (TMD analyses) are required for the postulated double-ended rupture of each reactor coolant system pipe within the loop compartment (i.e., the volume containing the reactor coolant pumps, steam generators, and pressurizer and bounded by the crane wall, reactor cavity wall, operating deck and containment floor). In addition, TMD analyses are required for high energy line breaks within any subcompartment which contains a high energy fluid line and a postulated pipe rupture is not precluded by conservative piping design (i.e., guard pipes or super pipe). Maximum pressures and

differential pressure acting upon containment internal structures are selected from these analyses to determine design pressures and/or capability of those structures. In addition, the short term ice condenser response analyses are used to determine the maximum nonuniform containment internal pressure time history which is used in performing dynamic buckling analyses of the primary containment building when the primary containment is a free standing steel design.

Our basis for accepting ice condenser short term transient analysis is further amplified in SRP 6.2.1.1.A.

b. Long Term Containment Response

1) Maximum Primary Containment Pressure

We require that applicants use the Westinghouse Electric Corporation's LOTIC-1 computer program to calculate the long term containment pressure response. LOTIC-1 is a computer program similar to the Westinghouse COCO which has been used to analyze the containment pressure transients for PWR dry and subatmospheric containments. The main difference between these computer codes lies in the methods by which the heat removal systems are modeled. LOTIC-1 includes features for modeling the heat removal capabilities of the ice condenser and has provisions to calculate the pressure response of the

containment. The containment upper and lower compartments and the ice condenser are modeled as control volumes in the code to represent the physical geometry of the containment. Conservation of mass and energy are applied and the equations are solved by appropriate numerical procedures.

The LOTIC-1 code is described in a Westinghouse topical report, WCAP-8354, which we reviewed during the course of our review of the lead ice condenser plant, D. C. Cook, and reported on through the NRC topical report evaluation program. We have concluded that the LOTIC-1 code is conservative for the calculation of ice condenser containment long term pressure and temperature responses for loss-of-coolant accidents.

In accordance with the provisions of SRP 6.2.1.1.A, we review the applicant's long term loss-of-coolant accident analyses to assure that the design basis events have been identified, the LOTIC-1 code has been used, that conservative input assumptions have been used and that the peak calculated containment pressure does not exceed the design pressure of the containment.

2) Maximum Reverse Differential Pressure

We require that applicants use the Westinghouse LOTIC-2 computer program to determine the maximum reverse differential pressure which the operating deck and ice condenser lower inlet doors may experience during an accident. The LOTIC-2 code is a modified version of the original LOTIC code (LOTIC-1). The LOTIC-2 code uses the conservation of momentum equations to calculate flow rates and pressure differentials between the containment compartments. The LOTIC-2 code includes the capability to model the cooling effect of water from the ice condenser draining to the lower compartment, and condensation on the sump pool surface, as well as the check valve type action of the ice condenser lower inlet doors. We have reviewed the LOTIC-2 code and have found the code acceptable for the calculation of the maximum differential pressure across the operating deck and lower inlet doors. The LOTIC-2 code is described in WCAP-8354 which we have reviewed through the NRC topical report evaluation program. We require that the maximum calculated reverse differential pressure be less than the design values for the operating deck and ice condenser lower inlet doors. Typical ice condenser values are a maximum calculated reverse differential pressure of 1 to 2 psi as compared to design reverse differential pressures of about 15 psi and 8.6 psi

for the operating deck and lower inlet doors, respectively.

3) Minimum Containment Pressure

We require that applicants also use the Westinghouse LOTIC-2 code for evaluation of the minimum expected lower compartment pressure response to loss-of-coolant accidents in plants with ice condenser containments. As indicated above we have reviewed the LOTIC-2 code through the NRC topical report program. We also found the LOTIC-2 code to be an acceptable code for the calculation of the minimum containment pressure response for ice condenser containment designs. We review the applicant's analyses of the minimum containment pressure response to assure that 1) the analysis was performed using the LOTIC-2 code; 2) conservative plant particular containment parameters were used in the analysis (see SRP 6.2.1.1.B and Branch Technical Position 6.1 attached to SRP 6.2.1.1.A for additional details); and, 3) the calculated lower compartment pressure transient or an arbitrarily assumed pressure value conservatively lower than that calculated transient were input to the ECCS calculations. These ECCS calculations must demonstrate acceptable core cooling capability.

4) Maximum Containment Temperatures

We review the methods and assumptions by which the lower compartment atmosphere temperature is calculated for assumed main steam line breaks. We have reviewed the applicant's analytical methods as described in WCAP-8354. For the purpose of calculating the lower compartment and response of an ice condenser plant to the main steam line break, the Westinghouse LOTIC-3 code is used. We have also reviewed the LOTIC-3 code through the NRC topical report evaluation program and have found it to be acceptably conservative for the purpose of calculating ice condenser containment response to assumed steam line breaks. We require that applicants analyze the containment response to a spectrum of main steam line break sizes over the complete range of operating conditions to determine the most severe environment which might exist in the ice condenser containment during a main steam line break. This most severe calculated lower compartment environment is then used as the basis for evaluating the acceptability of the qualification programs for equipment and instrumentation required to mitigate the consequences of a steam line break inside the containment.

c. Subcompartment Design Considerations

The evaluation for the subcompartment design follow that described in Section IV.A.1.d except that the Westinghouse TMD code is used for the response analysis.

d. Ice Condenser Steam Bypass

We require that applicants perform conservative analysis of the effects of steam bypass for a spectrum of loss-of-coolant accident break sizes to determine the minimum size bypass flow area which may exist in an ice condenser containment without resulting in a calculated containment pressure exceeding the containment design pressure. We review the methods of analysis, input assumptions, and results. Typical results have indicated that for the ice condenser design the minimum bypass flow area which could result in a maximum containment pressure equal to the containment design pressure is about 40 ft^2. Since the allowable bypass area is large, we have instituted the following to assure that bypass areas in the plant during operation will not exceed the design limit value (about 40 ft^2). A pre-operational flow test is required to indicate the presence of bypass areas unknowingly created by design or construction errors. Technical Specifications governing the operation of the plant require that prior to operating the plant and at each refueling

shutdown, the divider barrier (operating deck and its associated seals) be visually inspected for any signs of damage to the seals or seal surfaces. In addition, the Technical Specifications require that seal material sample coupons placed in the approximate locations of the seals be tested to demonstrate continued integrity of the seal materials. In this manner through continued surveillance of the ice condenser, bypass flow area is maintained at approximately 2.2 ft^2, which is the flow area provided for the spray system drains. These drains are required to return containment spray water to the containment sump for recirculation.

e. Ice Condenser Component Design

For any particular application under review, we evaluate the design of the ice condenser by comparing it with the design information presented in Appendices M and N to the D. C. Cook FSAR, and discussed in our safety evaluation report for Unit 1 of the D. C. Cook Nuclear Plant. We have reviewed the design of ice condenser as reported in these documents and have found that it satisfies our acceptance criteria which are expressed in Section II of SRP 6.2.1.1.B, Ice Condenser Containments. Any differences from the designs reported in the D. C. Cook documents are evaluated. We require that all design changes be justified and changed components be requalified for use in the

ice condenser by the same methods originally used to qualify them, i.e., for simple structures which were qualified by analytical methods a reanalysis will be acceptable, for components originally qualified by test programs, the tests must be repeated on the revised design.

f. Ice Maintenance

We require that, following the initial ice loading, the applicant weigh the ice in a sample population of greater than 50 percent of the ice baskets. The sample includes all the baskets (81 baskets) in four of the twenty-four ice condenser bags. These four bags should be selected on the basis of location where the highest ice loss might be expected based on operating experience at the D. C. Cook plant. In addition, all baskets in rows 1, 2, 4, 6, 8 and 9 with azimuthal locations 2, 3, 4, 6, 7 and 8 in the remaining 20 bags will be weighed. This will result in a sample of 1044 of the 1944 baskets being weighed to determine the amount of ice initially loaded into the ice condenser. Using the above sampling program the applicant is required to determine the minimum amount of ice loaded into the ice condenser at a 95% level of confidence by statistical analysis and the initial distribution of ice within the ice condenser. The minimum amount of ice thus determined must be sufficient to assure that considering a 10% margin

for loss during the plant operating period and a 1% margin for measurement error that the ice condenser contains sufficient ice to assure that if an accident were to occur during the following operating period the plant would not experience a containment pressure greater than design. In addition, Technical Specifications have been developed for the ice condenser which require the periodic weighing of a sample of the ice baskets to provide continuing assurance of the existence of the minimum amount of ice necessary to provide for continued safe operation of the plant.

g. Ice Condenser Inspection

We review the design provisions for monitoring the status of the ice condenser during plant operation to assure that the ice condenser retains its minimum required cooling capabilities. We also review the aspects of the containment design which allow inspection and functional testing of ice condenser components during the various modes of plant operation. Specific areas evaluated are the ice condenser temperature instrumentation system, lower inlet door position monitoring system, return air fan system, proposed inspection and test programs for the lower, intermediate, and top deck doors, floor drains, ice condenser flow passages, divider barrier seals and access hatches. As stated in SRP 6.2.1.1.B, we determine that the proposed surveillance

programs, Technical Specifications for plant operation and the attendant design provisions fulfill the requirements of General Design Criteria 39 and 40.

h. Containment External Pressure

In the ice condenser containment design, failure of the containment as a result of the inadvertent operation of the containment cooling systems is precluded by: 1) designing the containment building to withstand the maximum containment external pressure (D. C. Cook); 2) providing vacuum relief valves to preclude the depressurization of the containment building (Sequoyah); or 3) use of a pressure control system which would terminate the operation of containment cooling systems prior to achieving an external pressure in excess of design (McGuire). Our acceptance criteria and review procedures for the review of the above approaches to the control of the external pressure transient are defined in Sections II and III of SRP 6.2.1.1.B.

2. Containment Heat Removal

The review of the Containment Heat Removal System is based upon the short term ice heat sink and the longer term containment spray heat removal system. It follows the discussion given in Section III B.2 and IV A.2.

3. Containment Isolation Systems

The staff's review of the isolation system follows the review aspects described in Section IV A.3.

4. Combustible Gas Control

We require that the applicant provide the results of analyses of the post-accident production and accumulation of hydrogen in the containment in accordance with the requirements of containment Branch Technical Position CSB 6-2, "SRP 6.2.5 'Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident' ". We require that the containment design include redundant safety grade post-accident hydrogen mixing systems and hydrogen recombiners. The post-accident mixing system provided for the ice condensers are redundant safety grade systems and provide connections to the major subcompartments within the containment (i.e., steam generator enclosures, reactor cavity, pressurizer enclosure, dead-ended compartments and the dome of the containment building. The systems are designed to operate as a part of or in conjunction with the ice condenser return air fan systems which are automatically operated ten minutes after an accident.

All ice condenser plants use redundant Westinghouse Thermal Electric recombiners. These recombiners have been reviewed and approved for use following our review of the design features of these recombiners.

If it is assumed that no hydrogen is transported from the containment lower compartment before the operation of the hydrogen mixing systems (i.e., 10 minutes after a LOCA), the ice condenser design can accommodate a release of hydrogen consistent with the reaction of about 0.5% of the core cladding without exceeding the 4% lower limit of flammability in the lower compartment. ECCS performance evaluations predict less than 0.3% metal water in the worst case. Typical ice condenser hydrogen accumulation analysis predict that following operation of the mixing system, the 4% limit would not be reached in the containment building for about two weeks if the hydrogen recombiners were not operated. The recombiners are designed to operate 24 hours after the accident, hence would be operated well in advance of this time and thence limit the hydrogen concentration to well below the 4% limit.

Further details regarding the review procedures and acceptance criteria for post-LOCA combustible gas control in ice condenser plants may be found in SRPs 6.2.1.1.B and 6.2.5.

5. Containment Leakage Testing

Containment leakage testing follows the review as discussed in Section IV A.5.

V. ONGOING PRESSURE SUPPRESSION CONTAINMENT PROGRAMS

The technical bases for continued operation and continued licensing of nuclear power plants that rely on the pressure suppression type of containment systems are addressed in Sections III and IV of this report. There are also a number of ongoing programs that are designed to:

1) reconfirm the adequacy of various aspects of the pressure suppression type of containments; and

2) identify areas of the designs that can or should be improved.

These ongoing efforts include: 1) the staff's Systematic Evaluation Program; 2) the staff's Generic Technical Activities Program; 3) the staff's research programs and its cooperative research programs with foreign agencies; 4) the staff's reviews of operating experience obtained from currently operating U.S. and foreign nuclear power plants; and 5) BWR-NSSS Equipment Adequacy Evaluation to be performed for each affected plant.

A. Systematic Evaluation Program

During fiscal year 1977, the NRC developed a program for the systematic evaluation of certain nuclear power plants licensed for

operation before 1972. Since these facilities first began operation, many new licensing criteria have come to be applied in the review process for construction permit and operating license applications. Modifications of earlier licensed plants to assure their continued safe operation have taken place during the intervening years, but these improvements have been made generally on the basis of individual operating experience or particular isolated systems within the plants.

The systematic evaluation of these older plants was started in fiscal year 1978. The NRC is evaluating the safety of individual systems in the context of overall plant safety and will reassess the safety margins prevailing in selected facilities licensed before 1972 to determine the degree to which each one meets current licensing requirements. Areas in which a facility falls short of meeting the requirements for a contemporary plant will be appraised, taking into account the unit's operating history, the probability of potential accident, and the probable consequences thereof. Any changes required in a plant's equipment or operating procedures will be based on a balanced overall safety assessment. Containment design aspects of several Mark I

plants will be included in this evaluation in conjunction with the staff's work on the Technical Activities Programs, which are described in the next section.

B. Generic Technical Activities Programs

The staff established its Generic Technical Activities Program in early 1977, as described in NUREG-0410, to develop resolutions to generic issues. The generic issues have been categorized in accordance with a prescribed set of NRR priorities. Work is well underway on the Category A (highest priority) technical activities. The Category A technical activities in the area of the pressure suppression type containment systems include:

A-2 Asymmetric Blowdown Loads on the Reactor Vessel

A-6 Mark I Short Term Program

A-7 Mark I Long Term Program

A-8 Mark II Program

A-23 Containment Leak Testing

A-24 Qualification of Class 1E Safety Related Equipment

A-34 Instruments for Monitoring Radiation and Process Variables During Accidents

A-39 Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits for BWR Containment

C. Research Programs

The staff's Office of Nuclear Regulatory Research is sponsoring a number of confirmatory research and computer code development programs that relate to the pressure suppression type of containments. There are also a number of cooperative ventures between the NRC staff and selected foreign agencies which were addressed in Section II of this report.

The programs underway by the Office of Nuclear Regulatory Research are described in the NUREG 0135, Water Reactor Safety Research. The areas under investigation in these programs include the development of a multidimensional transient compressible flow program (BEACON) at EG&G Idaho, Inc. The task also includes review and evaluation of containment data from the Marviken (MX-11) and Battelle-Frankfurt programs. In addition efforts at Lawrence Livermore Laboratory are underway to develop a pressure suppression model for a best estimate code for BWR containment analysis. Additional studies are underway at the Massachusetts Institute of Technology to develop the basis for small-scale modeling of LOCA induced flows in BWR pressure suppression containments. The University of California in Los Angeles is engaged in studies to provide basic experimental data on transient thermal hydraulic phenomena induced by air and steam injection into water for use in confirmation of numerical modeling, establishing scaling relations for steam venting, and for increased understanding of free-surface heat transfer in BWR suppression pools.

D. Operating Experience

The staff also has an ongoing program to evaluate the operating experience from currently operating nuclear power plants. The Technical Specifications of licensed nuclear plants define those events or occurrences that must be reported to the NRC staff in the form of Licensee Event Reports (LER). These LERs and other reports in operating experience are reviewed by the NRC staff to determine whether the design of currently operating plants and current licensing requirements need to be modified.

The NRC staff, through its Office of International Programs, also has a cooperative program with foreign agencies for the purpose of exchanging data on operating experience and for discussion of their impact on plant safety.

E. BWR-NSSS Equipment Adequacy Evaluation

In March 1978, General Electric Company (GE), the NSSS-vendor for BWR facilities, met with the NRC staff to discuss a systematic program being pursued by GE to reevaluate the adequacy of the NSSS-equipment (i.e., the reactor vessel, reactor vessel intervals, steam piping, reactor piping and mounted equipment) utilized in Mark II and III BWR facilities. This program will demonstrate through analysis that NSSS hardware is adequately designed to accommodate original design loads plus additional

hydrodynamic containment response loads. Each utility would als` do a similar evaluation of the BOP equipment at the same ti~ These loads include 1) LOCA air clearing, 2) LOCA steam condensation and chugging, 3) SRV actuations, and 4) reactor vessel annulus pressurization. GE has proposed to perform this reanalysis for all Mark II and III plants and to have it completed before the OL is issued. There are five principal evaluation assumptions that are under consideration for the GE evaluation program; e.g., 1) LOCA load definition, 2) SRV load definition, 3) load combination acceptance criteria, 4) load combination methods, and 5) mass and energy release data for annulus pressurization.

The integrated evaluation program is primarily applicable to facilities under construction and not yet licensed for operation. Similar reevaluations of the effects of LOCA and SRV related pressure-suppression pool dynamic loadings will be performed for Mark I BWR operating facilities if significant hydrodynamic structural responses at the NSSS equipment are found to occur. Reactor vessel annulus pressurization dynamic loading effects will also be considered separately for the Mark I BWR facilities.

ENCLOSURE A

Summary of NRC Staff Actions Related To The Technical Issues Identified In Dr. Hanauer's Memorandum of September 20, 1972

A. Concern:

"Like all containments the pressure suppression designs are required to include margins in capability. Experiments have been conducted by GE and Westinghouse to establish the rate of steam generation that can be accommodated. The pressure-suppression pools, ice condenser, etc., are then sized for the double-ended break steam flow, with margins for unequal distribution of steam to the many modular units of which the condenser is composed. The rate and distribution margins are probably adequate.

More difficult to assess is the margin needed when applying the experimental data to the reactor design. Recently, we have reevaluated the 10-year old GE test results, and decided on a more conservative interpretation than has been used all these years by GE (and accepted by us). We now believe that the former interpretation was incorrect, using data from tests not applicable to accident conditions.

We are requiring an independent evaluation of the ice condenser design and its bases to make less probable any comparable misinterpretation of this design."

Responses:

Since this concern was expressed, additional tests, both domestic and foreign, of BWR pressure systems have been conducted, e.g., 4-T, PSTF, and Marviken.[1] Computer codes which have been and are being used to predict the containment pressure and temperature response of the BWR pressure suppression containment systems have been used to calculate the pressure response for these test facilities. The calculated values when compared to the test results have confirmed the adequacy of the computer models. These comparisons have been made by both the vendor and the NRC.[2]

Consequently, the viability of the pressure suppression concept which was originally demonstrated by testing performed in 1958 through 1962 has been confirmed.

With respect to ice condenser containments, the NRC has developed computer codes which are used to predict the containment's pressure response

during postulated LOCAs. These codes are now being compared to test data and the preliminary results of such comparisons are that the adequacy of the current models has been confirmed. Vendor's codes (Westinghouse) have been compared to tests and have been confirmed.[3]

The margins applied for pressure suppression containment design have also been confirmed by the additional test data that has become available since 1972. These margins exist both in the basic design of the containment structure and in the analytical models used to predict the containment response. The experimental data are no longer applied directly to determine the containment design requirements.

B. Concern:

"Since the pressure-suppression containments are smaller than conventional "dry" containments, the same amount of hydrogen, formed in a postulated accident, would constitute a higher volume or weight percentage of the containment atmosphere. Therefore, such hydrogen generation tends to be a more serious problem in pressure-suppression containments. The small GE designs (both the light-bulb-and-doughnut and the over-under configurations) have to be inerted because the hydrogen assumed (per Safety Guide 7) would immediately form an explosive mixture. The GE Mod 3 and the Westinghouse ice condenser designs (they have equal volumes) require high-flow circulation and mixing systems to ensure even dilution of the hydrogen to avoid flammable mixtures in one or more compartments (see following for an additional serious disadvantage of this needed recirculation and its valves). By contrast, the dry containments only require recombination or purging starting weeks after the accident."

Response

Most Mark I BWR pressure suppression containments are currently required to be inerted as part of the measures for combustible gas (i.e., hydrogen) control following a postulated loss-of-coolant accident. This requirement resulted from the staff's assumptions regarding the amount of hydrogen generated and the magnitude of the lower limit of hydrogen flammability. However, in 1974 the Commission ruled that the technical issues related to inerting requirements should be resolved by way of rulemaking. Subsequently, a rulemaking proceeding was initiated which led to the development of a proposed change to the regulations, i.e., 10 CFR 50.44, "Standards for Combustible Gas Control System in Light Water Cooled Power Reactors."[4]

The revised assumptions in this proposed rule and those specified in the Branch Technical Position, CSB 6-2,[5] would permit plants to de-inert where it can be demonstrated that the hydrogen concentration

can be maintained below a combustible mixture. The analyses for Vermont Yankee indicate that most, if not all, plants could de-inert using the assumptions in the proposed 10 CFR 50.44.[6]

Those facilities with the Mark II pressure-suppression containment system design have not yet been licensed for operation. However, in light of the staff requirements specified in Branch Technical Position, CSB 6.2, we do not expect that inerting will be required for these facilities.

The Mark III BWR containment system and the PWR ice condenser containment system have relatively larger volumes and do not require inerting for combustible gas control. However, mixing systems are provided to take advantage of the total containment volume for dilution of hydrogen. In the ice condenser containment design, the primary function of the mixing system is to assure long-term condensation within the ice bed. Staff positions were developed during the course of the review of the first Mark III plant application (i.e., Grand Gulf) which set forth the minimum design requirements for the mixing systems[5] and to preclude the potential for excessive steam bypass.[7]

Although the time frame within which combustible gas control must be initiated is much shorter for a pressure suppression containment than for dry containments, it is still long enough to permit manual operation and it occurs well after the initial blowdown transient.

C. Concern:

"All pressure-suppression containments are divided into two (or more) major volumes, the steam flowing from one to the other through the condensing water or ice. Any steam that flows from one of these volumes to the other without being condensed is a potential source of unsuppressed pressure. Neither the strength nor the leakage rate of the divider (between the volumes) is tested in the currently approved programs for initial or periodic inservice testing. Some effort is now underway to devise a leakage test, but none has so far been accomplished."

Response:

With respect to the BWR pressure suppression containment systems, the leakage of steam from the drywell directly to the suppression chamber airspace bypasses the suppression pool and could potentially result in an overpressurization of the containment. The maximum allowable bypass leakage rate is a function of the size of the postulated loss-of-coolant accident. Facility Technical Specifications[8] include periodic

(approximately every eighteen months) testing requirements to monitor the bypass leakage rate. The tests are performed by pressurizing the drywell to one to two pounds per square inch greater than the suppression chamber and monitoring the rate of pressure decay. The pressure decay rate is then correlatable to an equivalent bypass leakage area. This test is conservative since all drywell leakage paths are inherently included in the test results while only a small portion of these paths contribute to bypass leakage.

In addition, most BWR operating plants with pressure suppression containments have been operating with a positive pressure differential between the drywell and suppression chamber since February 1976.[9] Maintaining this pressure differential provides a continuous monitor of bypass leakage and a verification of the status of the drywell to suppression chamber vacuum breakers.

With respect to the ice condenser containment design, a substantive amount of bypassing can be tolerated without exceeding design conditions. Analysis indicates that bypass areas of about 35 to 50 square feet can be tolerated.[10] This is a large area when compared to the bypass area which can be tolerated for water pressure-suppression systems (which varies between about .02 and 1 square feet) and, therefore, less testing has been required. However, we do require both pre- and post-operational testing to confirm the bypass capability of each ice condenser plant.[11]

The strength capacity of the "divider" in the Mark I design is demonstrated by structural analysis of the vent system. The strength capacity of the "divider" floor in the Mark II design will be confirmed by preoperational testing.

D. Concern:

"Because of limited strength against collapse, the "receiving" volume has to be provided with vacuum relief. In all designs except GE Mod 111, this function is performed by a group of valves. Such a valve stuck open is a large bypass of the condensation scheme; the amount of steam that thus escapes condensation can overpressurize the containment.

Valves do not have a very good reliability record. Recently, five of the vacuum relief valves for the pressure-suppression containment of Quad Cities 2 were found stuck partly open. Moreover, these valves had been modified to include redundant "valve-closed" position indicators and testing devices, because of recent Reg concerns. The redundant position indicators were found not to indicate correctly the particular partly open situation that obtained on the five failed valves. We have only recently begun to pay serious attention to these valves, so pre-

vious surveillance programs have not generally included them. The GE Mod 111 design has an elegant water-leg seal that obviates the need for vacuum relief valves."

Response:

Vacuum breakers are provided between the drywell and the suppression chamber to allow reverse flow back to the drywell following the initial blowdown transient. These valves are normally closed; however should any of these valves be open at the time of the accident, steam bypass could potentially result in an overpressurization of the containment. Since 1972, staff positions were developed which required periodic testing and redundant position indication to assure that excessive bypass leakage through the vacuum breakers would not occur.[12/]

Continuous monitoring of these valves is provided by the positive pressure differential between the drywell and suppression chamber. Additional testing requirements also exist to demonstrate the capability of these valves to perform their vacuum-relief functions.[13/] All of these testing requirements are included in the surveillance requirements contained in the Technical Specifications for each plant.

These testing requirements have also served as a basis for the development of maintenance programs to correct deficiencies in the valve position indicators. As a result of these independent maintenance programs, failures of the position indicators have been very infrequent over the past several years.

E. Concern:

"The high capacity atmosphere recirculation systems provided for hydrogen mixing involve additional valves which, if open at the wrong time, would constitute a serious steam bypass and thus a potential source of containment overpressurization. These valves are large, and must open quickly and reliably when recirculation is needed. In other engineered safety features, no single valve is relied on for such service, yet redundancy has not been provided even for single failures, open and closed, of these valves. This is a serious mission, since opening at the wrong time leads to overpressurization, while failure to open when needed inhibits recirculation."

Response:

This issue relates to the BWR Mark III containment system design. In 1974, the AEC developed a position in conjunction with the review of the first Mark III BWR (i.e., Grand Gulf) which addresses the concern of large mixing system penetrations in the drywell.[7] This position included the following features:

1. Alternate mixing system designs were to be developed to limit the potential for bypass through large drywell penetrations.

2. Containment bypass capability was to be increased to accommodate single failures of the valves in the lines.

As a result of this position each Mark III applicant provided a mixing system design consistent with our position. The designs included the following features: small drywell penetrations; redundant inlet and exhaust penetrations to assure a recirculation path; the use of two valves in series on each line to assure isolation capability; and an evaluation of the containment capability to accommodate bypass through an inadvertantly open recirculation line.[14]

F. Concern:

"The smaller size of the pressure-suppression containment, plus the requirement for the primary system to be contained in one of the two volumes, has led to overcrowding and limitation of access to reactor and primary system components for surveillance and in-service testing."

Response:

Although pressure-suppression containment system designs are generally more crowded and less accessible than dry containment system designs, based upon the experience gained through our reviews of the Inservice Inspection and Inservice Testing (ISI/IST) programs which have been submitted by licensees in accordance with the requirements of 10 CFR 50.55.a, only one significant BWR inspection-related accessibility problem has been identified, i.e., the beltline region of the reactor pressure vessel. This inaccessibility is a result of the vessel design, not the containment design.

The beltline region of PWR vessels can be inspected from the inside of the vessel because the core internals can be removed whereas this is not possible for BWRs. Augmented inspection of accessible

areas of BWR reactor pressure vessels and operating limits on reactor pressure and temperature compensate for this inability to perform ISI.

With the exception of the above-mentioned area, no other significant inspection-related accessibility differences between PWR and BWR containments have been identified.

G. Concern:

"Separate shielding of components has tended to subdivide into compartments the volume occupied by the primary system. (Some compartmentation of dry containment also occurs.) A pipe break in one of these compartments creates a pressure differential; each compartment must be designed to withstand this pressure. A method of testing such designs has not been developed."

Response:

The arrangement of structures internal to the containment differ between the Mark I/II containment system design and the Mark III containment design. The Mark I/IIs have fewer compartments than PWR dry containments because there is less need for radiation shielding. The Mark I/IIs are essentially inaccessible during normal plant operations, thereby requiring fewer structures for shielding. The Mark III design for internal structures is generally comparable to the PWR dry containment design.

For all designs, both dry and pressure suppression containments, we analyze the pressure response within compartments for postulated pipe breaks to ensure the adequacy of the design pressure differential for compartments.[15], [16], There are ongoing foreign tests being conducted to verify analytical methods.[17] NRC and vendor codes are part of this program.

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