DRAFT SAFETY EVALUATION
(CONTAINMENT SYSTEMS)
GRAND GULF UNITS 1 & 2
DOCKET NOS. 50-416 and 50-417

6.2 Containment Systems

The containment systems for the Grand Gulf Nuclear Station include a reactor containment structure, containment heat removal systems, containment isolation system, combustible gas control system, an enclosure building surrounding the primary containment, and a standby gas treatment system. Most notably Grand Gulf will be the first nuclear plant to utilize the Mark III containment design, a design which includes the basic water pressure suppression feature of previous BWR containments but incorporate a large containment structure instead of a small torus section inherent in Mark I and II designs.

The basic performance and design evaluation of the Mark III containment system has been the subject of both ongoing analyses and experimental programs. These efforts are described below and provide the basis upon which our evaluation was performed. Full-scale containment performance tests are currently in progress at the General Electric Company's test facility in San Jose, California.

The Mark III containment concept was the subject of preliminary reviews by both the Regulatory staff and the ACRS. The results of these reviews were published in a staff Safety Evaluation dated

October 5, 1972 and an ACRS Report to the then Chairman, Dr. J. R. Schlesinger, dated January 17, 1973. The Safety Evaluation presented in this report is the result of an ongoing review effort that includes the proposed design for the Grand Gulf facility, the GE test program, the analytical studies and model development and the matters identified by the ACRS in its report; i.e., the vacuum relief system, the recirculation system between the drywell and the containment, and the system for coping with hydrogen.

6.2.1 Containment Functional Design

The containment system is divided into two major subvolumes, a drywell enclosing the reactor system, and the primary containment surrounding the drywell and containing the suppression pool. (See Figure 1) The containment and the drywell volumes are connected, through the suppression pool by an array of horizontal vents in the drywell wall.

The primary containment is a steel-lined reinforced concrete structure consisting of a vertical cylinder, domed top, and a flat base. The net free volume of the primary containment is 1.4×10^6 ft^3 and the design pressure is 15 psig. To satisfy its design basis as a fission product leakage barrier the primary containment is designed for a leakage rate of 0.1% of the volume per day at 15 psig.

An additional structure called the enclosure building, surrounds

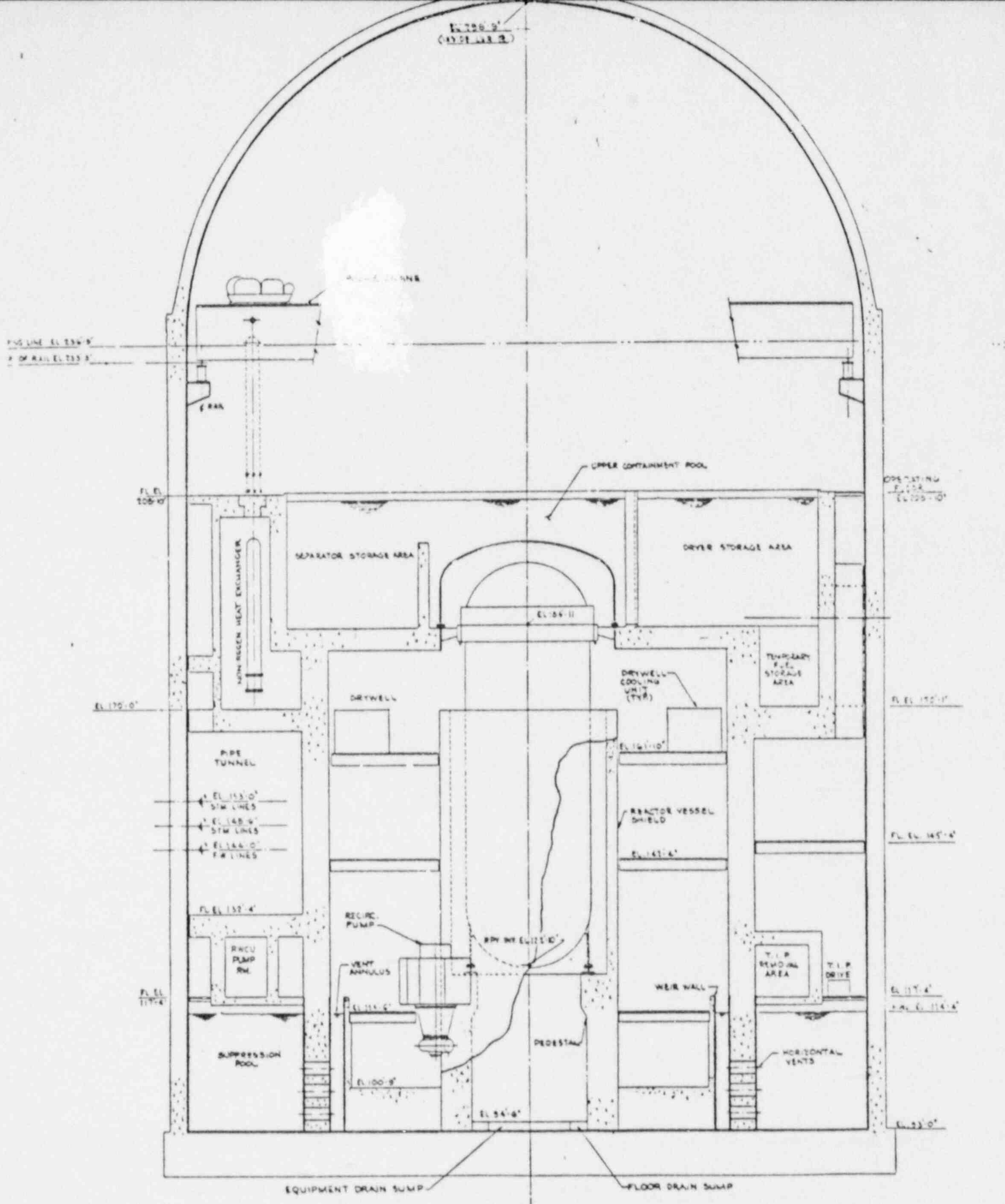

FIGURE 1
CONTAINMENT ARRANGEMENT

the primary containment. Its purpose in conjunction with the auxiliary building, is to provide a volume in which fission product leakage from the primary containment following a postulated loss-of-coolant accident can be diluted and held up prior to release to the environment. Our evaluation of the enclosure building design is included in Section 6.2.3 of this report.

Located within the primary containment is a substructure, called the drywell, which encloses the reactor and reactor coolant system. The drywell is an unlined concrete structure, enclosing a net free volume of 280,000 ft^3 and designed for a differential pressure of 30 psi. The purpose of the drywell is to channel steam released during an unlikely loss-of-coolant accident through the vent matrix system to the suppression pool for condensation. While not a fission product barrier the drywell must be free of gross leakage for adequate performance of the pressure suppression feature.

Since, for the Mark III design, the containment completely surrounds the drywell, high energy lines penetrating the drywell must pass through the containment volume. In the unlikely event that one of these lines was postulated to rupture inside the containment but outside the drywell, the leak tight integrity of the containment could be violated because the pressure suppression feature of the design would be bypassed. Therefore, the applicant has provided guard pipes on all high energy lines between the drywell and containment with the exception of several small lines which utilize break detection and isolation systems.

Because the pressure suppression concept relies upon a controlled channeling of steam through the suppression system, the possibility of bypass paths must be minimized. Our evaluation of potential bypass sources and containment bypass capability is discussed in Section 6.2.1.6 of this report.

The suppression pool is a 360-degree annular pool located in the bottom of the containment and retained between the containment wall and the drywell weir wall. The weir wall is a 360-degree, reinforced concrete wall located inside the drywell and 30 inches from the drywell wall. At a minimum water height, 24'4", the volume of water in the suppression pool is 156,800 ft^3. The suppression pool serves both as a heat sink for postulated transients and accidents and as the source of cooling water for the emergency core cooling systems. In the case of transients that result in a loss of the main heat sink, energy would be transferred to the pool by the discharge piping from the reactor pressure relief or safety valves. In the event of a loss-of-coolant accident within the drywell, the horizontal vent system in the drywell wall would provide the energy transfer path.

Located in the vertical section of the drywell wall and below the suppression pool water level are 135 vent holes of 28" diameter and arranged in 45 circumferential columns of three vents. The vent centerlines are located at 11 feet four inches, seven feet two inches, and three feet above the bottom of the suppression pool. In the event of a loss-of-coolant accident the pressure will rise in the drywell due to the release of reactor coolant, and force the level of water down in the weir annulus. When the water level has been depressed to the level of the first row of vents, the differential pressure will cause air, steam, and entrained water to flow from the drywell into the suppression pool. The steam will be condensed in the pool and the air driven from the drywell will be compressed in the primary containment. The net effect could result in approximately a 4 psi rise in containment pressure. Peak drywell differential pressure is calculated by the applicant to be 22.6 psi. Figure 2 illustrates the drywell and containment pressure response as a function of time following a design basis loss-of-coolant accident.

Following the initial phase of the accident, containment and drywell pressure will continue to rise due to the input of core decay and sensible heat to the suppression pool. The long-term pressure rise will be limited to 12.6 psig by operation of the

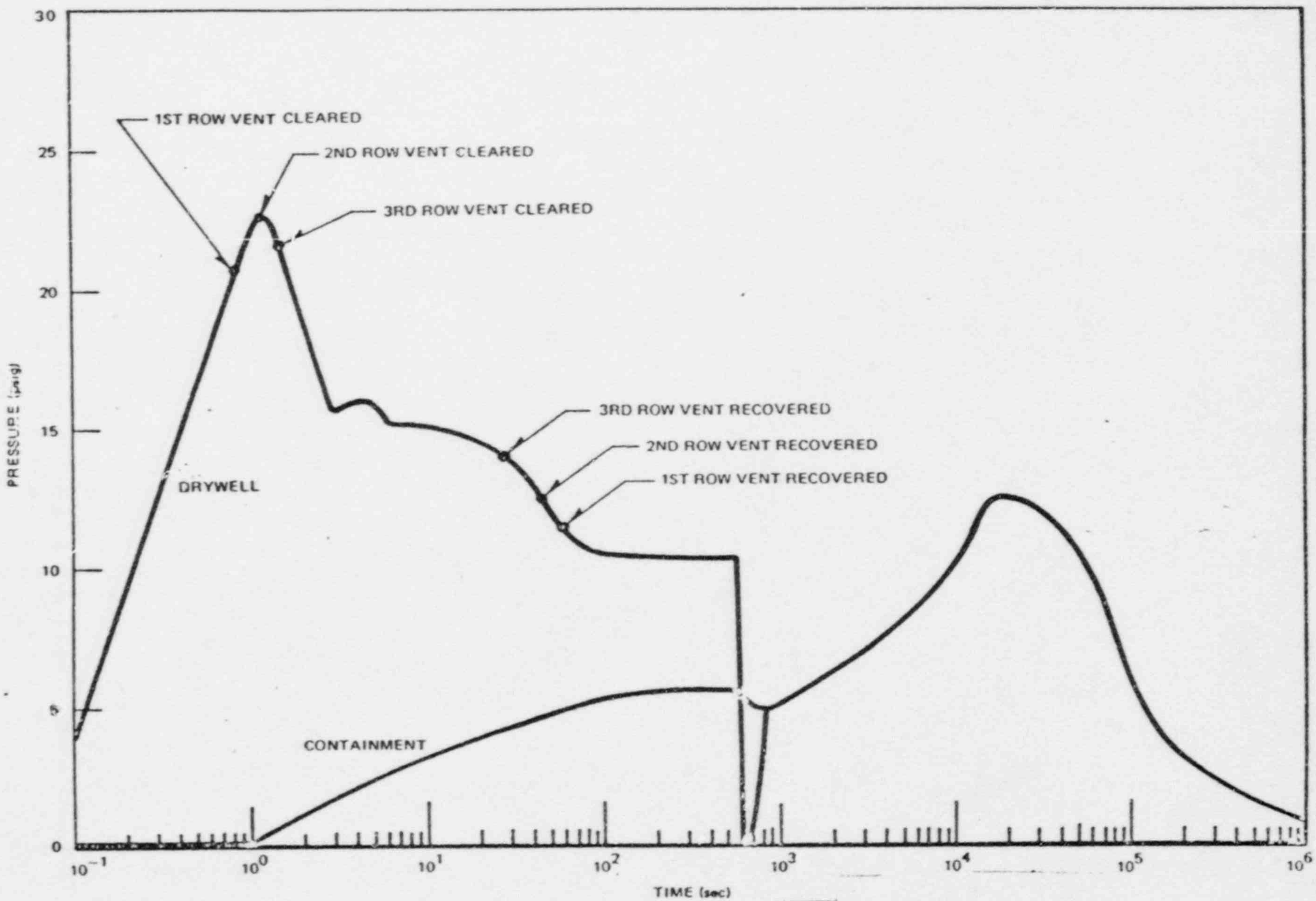

FIGURE 2- CONTAINMENT PRESSURE ANALYSIS - MAIN STEAM LINE BREAK

redundant containment heat removal system. Therefore, in the pressure response analysis of this type of containment two limiting conditions must be considered; the short-term drywell differential pressure and the long-term containment shell pressure. Our evaluation of the applicant's analytical methods for each of these time periods (i.e., both long and short term) is discussed in Sections 6.2.1.2 and 6.2.1.3 of this report. The General Electric Company has also completed small-scale tests and is performing full-scale tests to support the Mark III short-term analytical model. Our review of these test programs is also discussed below.

Both the drywell and containment are divided into a number of subcompartments by internal structures. The pressure responses within these subcompartments were analyzed by the applicant using the COPRA computer code developed by Bechtel Corporation. Our evaluation of the subcompartment designs is discussed in Section 6.2.1.5 of this report.

6.2.1.1 Review of BWR Containment Technology

Two basic pressure suppression designs have preceded the Mark III containment; i.e., the Mark I, or "lightbulb-torus" and the Mark II, or "over-under". A comparison of design parameters for the three containment types is provided in Table 1. The wetwell and

TABLE 1

COMPARISON OF BWR CONTAINMENT DESIGNS

DRYWELL	MARK I (Brown's Ferry)	MARK II (Zimmer)	MARK III (Grand Gulf)
type of construction	steel shell	steel-lined reinforced concrete	reinforced concrete
air volume	159,000 ft^3	184,000 ft^3	280,000 ft^3
design pressure	56 psig	45 psig	30 psid
leak rate	0.5%/day	0.5%/day	NA
WETWELL			
type of construction	steel shell	steel-lined reinforced concrete	steel-lined reinforced concrete
air volume	119,000 ft^3	103,000 ft^3	1.4×10^6 ft^3
pool volume	85,000 ft^3	106,000 ft^3	157,000 ft^3
design pressure	56 psig	48 psig	15 psig
leak rate	0.5%/day	0.5%/day	0.1%/day
thermal power	3293 MWt	2436 MWt	4025 MWt
break area	4.8 ft^2	2.243	4.46 ft^2
vent area	302 ft^2	274 ft^2	552 ft^2
break area/vent area	.017	.008	.008

and drywell of Mark I and II were connected by a vent system which entered the suppression pool vertically and was at a constant submergence. For both designs, the design basis loss-of-coolant accident for containment response was a recirculation line break. In both Mark I and II containments the peak drywell pressure occurred at about 10 seconds following the accident, which was after vent clearing, and during the vent flow part of the transient. Wetwell peak pressures occurred in about 10-20 seconds due primarily to the compression of drywell air in the wetwell.

Mark II containments also experienced a short-term drywell deck differential pressure which could occur either at the time of vent clearing or later in the vent flow transient. Generally those plants with relatively large vent areas had vent clearing controlled peak deck differential pressures. In the long term both the drywell and wetwell reached a secondary peak pressure due to continued decay heat generation; however this transient was less severe than the short term and therefore was not controlling for establishing containment design pressures.

For containment analysis "The General Electric Pressure Suppression Containment Analytical Model" as described in NEDO-10320 and its supplements was used. This model consists of five separate sub-models; blowdown, drywell, wetwell, vent clearing and vent flow.

Based on a review of the analytical methods employed in the model, correlation with Humboldt and Bodega Bay test results, and comparison with CONTEMPT-PS results, the staff has previously concluded that the GE model was conservative and therefore acceptable for containment analysis.

The Mark III type containment proposed for the Grand Gulf plant is different from the Mark I and II types of containments in three basic ways. First the BWR/6 type reactor system which is proposed for Grand Gulf has relatively larger steam lines than the previous BWR core designs. The effect of this is that the postulated loss-of-coolant accidents associated with the main steam line rupture and recirculation line rupture result in very nearly equivalent peak drywell pressures. Therefore both of these postulated pipe breaks must be considered in determining the DBA-LOCA for Mark III containment pressure response.

Second, the vent system connecting the drywell and containment utilizes a circumferential arrangement of horizontal vents at three different elevations which leads to an additional functional dependence on vent clearing and vent flow phenomena than the Mark I and II types. In addition, because of the relatively large vent areas provided, the peak drywell differential pressure is vent clearing controlled; i.e., the highest differential pressure across the drywell occurs during vent clearing. This places added emphasis on the dynamics of vent clearing but reduces the impact of vent flow assumptions on drywell pressure.

Third, as the volume of the containment is about five times that of the drywell, the compression of drywell air into the containment during vent flow results in only a small, about four psi, rise in containment pressure. This small effect leads to a long-term containment peak pressure which is not specifically related to the size of the reactor coolant break or the short-term pressure response.

Because of the above, the staff has concentrated its review in those areas where previous analytical models or testing cannot be extrapolated to the Mark III design. These items are covered in detail in the following sections.

6.2.1.2 Short-Term Pressure Response

As discussed above both the main steam line break and recirculation line break result in very nearly equal peak drywell pressures.

For the postulated double-ended rupture of a 28" main steam line (break area of about 4.5 sq. ft.) the applicant has assumed a blowdown profile which is separated into an initial two-second period of steam only blowdown followed by two-phase, liquid water and steam blowdown due to liquid level swell in the reactor vessel. During steam blowdown, the mass and energy input rates to the containment were calculated assuming critical flow of an ideal gas. The two-phase blowdown rate was based on the frictionless Moody critical flow model and the average density of fluid inventory within the reactor vessel. The staff has previously reviewed and found acceptable these assumptions for determining blowdown rates.

The time at which the liquid level in the reactor vessel swells to the elevation of the steam line nozzles following the break, determines the time at which the model changes from a steam to two-phase blowdown assumptions. Peak drywell differential pressure can be sensitive to the level rise time since two-phase blowdown yields a greater rate of steam addition to the drywell than steam only blowdown, and also introduces liquid water into the vent flow. Both of these effects increase drywell pressures.

In the Grand Gulf containment analysis the applicant has assumed a level rise time of two seconds and based on this assumption has calculated that the peak drywell differential pressure would be 22.6 psid. (Figure 2). The applicant has also provided studies of level rise time as a function of operating conditions and the sensitivity of peak drywell differential to level rise time. These studies indicate that the most rapid level rise would be about one second assuming a hot standby condition and would result in an increase in the peak calculated drywell pressure by about 0.5 psid.

Following the postulated design basis loss-of-coolant accident, the drywell pressure will rise and accelerate the water in the vent annulus. At about 0.9 seconds, the first row of vents will be cleared of water and a mixture of air, steam, and water will flow into the suppression pool. The water in the vent annulus will continue to accelerate downward resulting in clearing of the second row of vents at about 1.1 seconds and the third row at about 1.4 seconds. The peak drywell differential pressure occurs at the time the second row of vents is cleared (main steam line break) and is a result of sufficient vent area being uncovered to reverse the pressure transient in the drywell. Due to this phenomenon the peak pressure is predominantly controlled by the dynamics of vent clearing and only partially influenced by vent flow assumptions.

In the analysis of the vent clearing transient, General Electric used the vent clearing model described in the "Mark III Analytical Investigation of Small-Scale Tests Progress Report", NEDM-10976. In this model the vent system is nodalized into six control volumes representing the vertical weir annulus and horizontal vents. Conservation of mass and momentum is applied to each control volume to determine fluid accelerations and vent clearing times. An effective vent length is used to simulate the effects of suppression pool inertia and turning loss coefficients are applied to account for changes in flow path direction and area. The loss coefficients currently used in the model are derived from generally accepted data and General Electric intends to verify these coefficients during the large-scale Mark III testing program.

The General Electric vent flow model has also been revised to consider the more complex Mark III horizontal vent geometry; however, the basic thermodynamic flow assumptions used for previous water-pressure suppression designs, remain unchanged. For Grand Gulf the vent flow was computed on the basis of parallel path flow splits which are a function of the number of uncovered vents and geometric loss coefficients. These loss coefficients will also be confirmed experimentally on the large-scale facility.

Based on these analytical models, the applicant has determined that the postulated rupture of a main steam line would result in the highest drywell differential pressure and has calculated this pressure to be 22.6 psid. The applicant has stated in the PSAR that the drywell will be designed for a pressure of 30 psid which provides a margin of 33% above the peak calculated value.

Both the Regulatory Staff and our consultants have reviewed the analytical methods employed in the drywell pressure response calculation. Based on our review and our consultants' recommendations we believe that this margin should be adequate to account for uncertainties in the GE vent clearing and vent flow models.

The applicant has also provided analyses of the containment pressure response for a postulated rupture of a recirculation line (break area of 3.1 ft^2) based on the blowdown model described in "The General Electric Pressure Suppression Containment Analytical Model", NEDO-10320. The applicant has calculated that the peak drywell differential pressure for this break is 21.8 psid which is about one psi less than that calculated for the steam line break.

We are continuing our review of the blowdown aspects of a recirculation line break as an ongoing effort. Our consultants, the Aerojet Nuclear Company, have indicated that the General Electric blowdown model may not conservatively predict the short-term; (i.e., less than one second) blowdown rates. The short-term blowdown rate is a sensitive parameter for a Mark III containment since drywell pressure peaks very early in the transient. GE uses a single node model to represent the primary system and considers fluid properties on a mass average basis. A multinode representation of the primary system, including the recirculation lines, could yield higher short-term blowdown rates. Based on discussions with our consultants, we believe that revised modeling; i.e., a multinode reactor coolant system model may be necessary to conservatively predict this blowdown. This could result in a calculated drywell peak pressure higher than that currently calculated by the applicant (22.6 psig) in establishing the drywell design differential pressure of 30 psid. This matter has been cited by our consultant and needs to be dealt with for resolution. We plan to treat resolution of the concerns on early blowdown rates in a Supplement to this report.

In the course of our review we have also identified the potential for flow maldistribution effects within the drywell and vent system. These effects could occur in the following two ways. First, blockage of part of the vent annulus or vent holes would result in limited flow through the restricted part of the vent system with consequently higher flows in the remainder of the vents. Second, the blowdown from a postulated pipe break in one area of the drywell could preferentially clear those vents nearest the break and delay clearing of those vents furthest away. The effect of flow maldistribution on drywell pressure has been discussed with General Electric on a generic basis and GE has stated that they will provide additional information to us by December 21, 1973. We will treat the resolution of this concern in a Supplement to the Safety Evaluation.

6.2.1.3 Long-Term Pressure Response

Following the short-term blowdown phase of the accident, suppression pool temperature and containment pressure will increase due to the continued input of decay and sensible heat into the containment.

Referring to Figure 1, at about 100 seconds after the accident the drywell pressure has stabilized to approximately 4 psi above the containment pressure. This differential pressure corresponds to the submergence of the first row of vents. At some later time the drywell and containment pressures will equalize due to the return of air from the containment. The thermodynamic conditions at this point in the transient are drywell and containment pressure of abou. 5 psig and suppression pool temperature of 135°F.

During this time period the ECCS pumps, taking suction from the suppression pool, have reflooded the reactor pressure vessel up to the level of the main steam line nozzles. Subsequently ECCS water will overflow out the break and fill the drywell up to the top of the weir wall, establishing a recirculation flow path for the ECCS coolant.

At about 30 minutes following the accident, the containment cooling mode of the Residual Heat Removal (RHR) System is activated and suppression pool water is circulated through the RHR heat exchangers, establishing an energy transfer path to the service water system and ultimate heat sink.

In the long-term analysis, the applicant has conservatively accounted for potential post-accident energy sources. These include decay heat, sensible heat, ECCS pump heat, metal-water reaction energy, and feedwater energy. The applicant has assumed that the only heat sink available in the containment is the suppression pool and the only mechanism for heat rejection is the RHR heat exchangers.

The long-term model also assumed that the containment atmosphere is saturated and equal to the suppression pool temperature at any time. Therefore, the containment pressure is calculated from the partial pressure of air and the saturation pressure of water at the pool temperature.

Based on the above assumptions the applicant has calculated the peak suppression pool temperature to be 183°F and the peak containment pressure to be 12.4 psig. The design pressure of the containment is 15 psig which allows a 21% margin above the peak calculated value. On the basis of our review of the applicant's analysis and the pressure margin we conclude that the containment design pressure for this plant is adequate.

The drywell structure is designed for an external pressure of 21 psid. A drywell vacuum breaker system is provided to control suppression pool water level in the weir annulus and prevent inadvertent flooding of the drywell. The system is not required to operate to maintain the structural integrity of the drywell. We have reviewed the drywell design external pressure and find that it is reasonably conservative and acceptable.

The inclusion of a drywell vacuum breaker system, however, poses several concerns. The vacuum relief function is performed by the same valves and penetrations as the hydrogen recirculation system, with differential pressure signals initiating opening and closing of the valves. The staff believes that this design does not provide adequate assurance to preclude inadvertent opening of the valves thereby resulting in bypassing of the suppression pool. The staff believes that the applicant should provide a check valve in series with the power actuated valves or other equivalent means to ensure isolation of these lines during an accident. The resolution of this issue will be reported in a Supplement to the Safety Evaluation.

6.2.1.4 Test Program

The General Electric Company is presently conducting a full-scale test program to experimentally establish the vent clearing and vent flow performance of the Mark III containment concept. Full-scale testing was started in November, 1973 following completion of a two-year small-scale test program. The staff and its consultants, the Aerojet Nuclear Company, have discussed the test program with General Electric and visited the test facility on several ocassions.

A total of 67 small-scale tests of the Mark III concept have been performed by GE since June 1971. The test arrangement simulates the containment at a scale of approximately 1:2000 on a volume basis. Small-scale test data has been reported in "Mark III Confirmatory Test Program Progress Report", NEDM-10848 and "Mark III Analytical Investigations of Small-Scale Test Progress Report", NEDM-10976. Correlations between test data and analytical predictions for vent clearing times indicate reasonable agreement in this scale.

The full-scale test facility has been volumetrically scaled to mockup an 8° sector of a Mark III containment including one column of three full size vents. The initial test series began in November 1973 and will continue through January 1974 and will consider steam blowdowns for one, two, and three 28-inch vent configurations. The objectives of this series will be to establish a correlation between experimental data and the General Electric vent clearing model and to verify the values of loss coefficients used in the vent clearing and flow models. In addition, potential suppression pool impact forces during vent clearing on structures above the pool are being investigated. Following this initial phase of testing a general test schedule has been planned by GE through 1975. These test series will consider, in sequence, the following phenomena:

1. a series of liquid blowdown tests to indicate comparability to the vent clearing and vent flow performance as determined in the steam blowdown series,
2. a series of small break tests to investigate pool stratification and vent chugging effects,
3. tests performed with the suppression pool at an initial elevated temperature to determine steam condensation capability,
4. a test series with a column of five 16-inch vents, where selection of various combinations of 1, 2, and 3 vents will be used to investigate the effects of vent spacing, vent interaction and extended blowdowns and

5. a multi-vent test series which will employ a vent test section of three columns of three, nine-inch vents to consider vent interactions, vent clearing, and pool maldistribution for a sub-scale mockup of a 24° sector of the containment.

Based on our review of the Mark III containment concept and the associated test program we consider the present testing to be a continuation of previous pressure suppression testing done by GE in support of past BWR containment designs. The emphasis of the large-scale tests has been directed at those aspects of a Mark III containment which are innovative and which therefore must be demonstrated experimentally and correlated to analytical predictions. We believe that the design of the large-scale facility and the scheduled tests to be performed will provide a sufficient data base to establish the performance characteristics of the Mark III and to validate the analytical approach taken by GE in its accident analysis of Mark III containments. Our consultants also concur in this evaluation.

6.2.1.5 Subcompartment Pressure Analyses

Within both the drywell and containment, internal structures form subcompartments or restricted volumes which are subject to differential pressures following postulated pipe ruptures. In the drywell there are two such volumes; the annulus formed by the reactor vessel and the biological shield, and the drywell head region which is a cavity surrounding the reactor pressure vessel head. In the containment the various components of the Reactor Water Cleanup (RWCU) System are located in individual compartments.

The applicant has presented the results of calculations of pressure differentials across the walls of these subcompartments. The analyses of the short-term pressure transients were performed using the Bechtel computer code, COPRA, (Compartment Pressure Analysis). Based on these calculated values the applicant has applied a margin of 40% to establish the design pressure of each compartment. At this time, we have performed similar analyses for several compartments which are in reasonable agreement with the applicant's calculated values. However, the model used to determine blowdown rate from a circumferential break of a pipe located in the biological shield annulus has not been adequately justified as discussed in Section 6.2.1.2. We will complete our evaluation of the subcompartment design pressures and report our conclusions in a Supplement to the Safety Evaluation.

6.2.1.6 Steam Bypass of the Suppression Pool

There are three potential sources of steam bypass of the suppression pool associated with the Mark III containment design used for Grand

Gulf. First, since the drywell is of reinforced concrete construction, the potential exists for cracking of the drywell structure under accident loading conditions. This will allow direct leakage of blowdown steam to the containment volume. Second, the design of the combustible gas control systems allow the opening of direct flow paths between the drywell and containment for the dilution of hydrogen. Although the system is designed to operate after blowdown is complete, residual steaming in the reactor vessel continues after blowdown due to the addition of decay and sensible heat to the ECCS coolant. This energy would be added directly to the containment atmosphere. Also, if the hydrogen mixing system was inadvertently actuated before the end of blowdown, a bypass condition would exist. Thirdly, for the Grand Gulf plant the Reactor Water Cleanup (RWCU) System is located within the primary containment but outside the drywell. This system has high energy pipe lines, connected to the reactor primary system, which do not have guard pipes. Therefore postulated ruptures in these lines would result in blowdown of reactor coolant directly to the containment atmosphere without benefit of energy absorption in the suppression pool.

In the case of postulated RWCU System pipe breaks the applicant has provided design features to terminate the blowdown prior to exceeding the design limits of the containment. Two isolation valves in series are provided on both the RWCU suction and return lines which will automatically isolate the RWCU System from the primary reactor system. Isolation signals will be generated by two leakage detection systems; one based on RWCU System flow comparisons and another based on compartment temperatures. In addition, a flow limiter is provided in the suction line to limit the rate of blowdown prior to isolation. The applicant has calculated that the containment pressure response assuming a RWCU pipe rupture would be less than 4 psig, which is below the containment design pressure of 15 psig.

To minimize the potential for inadvertent opening of the hydrogen recirculation lines the applicant has proposed interlocking the valves on these lines until full signal coincidence indication of time delay of 10 minutes, drywell hydrogen concentration and containment spray operation. The staff has informed the applicant that additional interlocks may be required; e.g., reactor vessel pressure, however resolution of this issue will be a function of the final design of the combustible gas control recirculation system (Section 6.2.5) and containment spray system.

Possible bypass leakage paths from the drywell to the outer containment have been considered in our review of the Mark III containment. The control of such bypass paths are important to ensure that the design pressure of the containment is not exceeded for postulated design basis accidents. In regard to bypass leakage associated with potential cracking of the drywell or other sources around penetrations, we believe that the Grand Gulf containment should have an allowable bypass area of approximately one square foot ($A/\sqrt{K}$) for the spectrum of reactor coolant system breaks. The applicant had proposed a minimum capability of only 0.043 square foot. The allowable bypass area is considered to be that leakage area between the drywell and containment which would result in containment pressurization to design pressure following a postulated loss-of-coolant accident. To mitigate the effects of bypass a heat removal system is necessary. On the Mark III containment such a system is a containment spray system. In fulfilling this requirement we have informed the applicant that the containment spray system design currently proposed in the PSAR, would be unacceptable because of its delayed availability. We will require that the system be capable of automatic actuation at times which will provide about one square foot of allowable bypass area for the spectrum of loss-of-coolant accidents.

In addition, we will require the applicant to perform a leakage test of the drywell at approximately design pressure prior to plant operation, and low pressure leakage tests of the drywell periodically during plant lifetime. The full pressure test will impose loads on the drywell which are a substantial fraction of the accident loads and will provide added assurance that the drywell, as constructed, conforms to its design bases. Drywell leakage measured at high pressure will also be correlated to leak rates at low pressure and will be required to be less than one tenth of the allowable bypass leakage. Resolution of the above items will be reported in a Supplement to the Safety Evaluation.

6.2.1.7 Summary and Conclusions

The applicant has calculated the short and long-term containment pressures and subcompartment pressures as described above. Based on our review of the applicant's analytical methods we conclude that the ntainment design pressure is adequate. Due to the uncertainties involved in the modeling of the short-term reactor system blowdown rates we will consider the effect of a possible increase in early blowdown rate regarding the drywell design pressure and its associated margin. Our review of the subcompartment design pressures is also incompl e due, in part, to the short-term blowdown issue. We have reviewed the large-scale test facility design and the scheduled tests to be performed and find that this program should provide sufficient data to confirm the containment analytical vent models used in the short-term analysis.

Our review of other aspects of the containment functional design has resulted in the following positions:

1. The applicant will be required to provide a containment spray system which can be automatically actuated at times which provide a bypass leakage capability equivalent to about one square foot ($A/\sqrt{K}$) for the spectrum of loss-of-coolant accidents.
2. The applicant will be required to perform a full-pressure leakage test of the drywell prior to plant startup and low pressure leakage tests periodically thereafter to ensure that the measured drywell leakage is less than one tenth the allowable bypass leakage.
3. The applicant will be required to provide check valves in the hydrogen recirculation lines or other suitable means to ensure isolation of the flow paths during an accident.

These matters will be addressed in a Supplement to the Safety Evaluation.

6.2.2 Containment Heat Removal

The containment cooling mode of the Residual Heat Removal (RHR) System is used to remove heat from the suppression pool and to limit long-term containment post-accident temperatures and pressures. The RHR System consists of two heat exchangers and three pumps. One heat exchanger and one pump form an independent loop and each loop is physically separated and protected to minimize the potential for single failures causing the loss of function of the entire system. The RHR System is designed to Category I seismic criteria.

Operating in the containment cooling mode, the RHR pumps take suction from the suppression pool, pass it through the RHR heat exchangers, and direct the cooled water either back to the suppression pool or to the reactor vessel. The locations of suction and return lines in the suppression pool facilitate 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 inlets.

The applicant has stated in the PSAR that adequate net positive suction head is available at the RHR pump inlets assuming the suppression pool is at its post-drawdown level and maximum temperature, and with no credit taken for any increase in containment pressure. These assumptions are consistent with the requirements of Regulatory Guide 1.1 and therefore acceptable. Provisions are made in the containment heat removal system to permit in-service inspection of system components and functional testing of active components.

We conclude that the containment heat removal system can be operated in such a manner as to provide adequate cooling to the containment following a loss-of-coolant accident and meets the intent of General Design Criteria 38, 39 and 40.

6.2.3 <u>Containment Atmosphere Cleanup</u>

The Grand Gulf plant is provided with a secondary containment volume which encloses the primary containment and is maintained at a negative pressure by the Standby Gas Treatment System (SGTS) following a loss-of-coolant accident. The boundary of the secondary containment is defined by the walls of the enclosure building, the auxiliary building, and the steam tunnel.

The Standby Gas Treatment System consists of redundant exhaust fans, recirculation fans, and charcoal filter trains. Upon receipt of an accident signal, the SGTS will start and operate to reduce and maintain the secondary containment pressure at -0.25 inch water gauge. Therefore potential leakage through the primary containment will be collected and processed by the filtration train prior to release to the atmosphere through the SGTS vent. The recirculation fans will operate to provide mixing of the secondary containment volume.

Due to recent design modifications in this area the applicant has not provided details of the completed design. However, based on the commitments made by the applicant at this time and the preliminary information available we anticipate that this issue can be resolved. When the detailed information is submitted we will complete our review and report our conclusions in a Supplement to the Safety Evaluation.

6.2.4 Containment Isolation Systems

The purpose of the containment isolation system is to provide the necessary containment integrity between the primary coolant system pressure boundary or the containment and the environs in the event of a loss-of-coolant accident. The applicant has specified the design criteria and isolation valve arrangements used for isolation of primary containment penetrations.

No manual operation is required for immediate isolation of the containment. Automatic trip valves are provided in those lines which must be isolated immediately following an accident. Lines which must remain in service following an accident for safety reasons are provided with at least one remote manual valve. The containment isolation systems have been designed to the ASME Section III, Class 1 or 2, code and have been classified as Category I seismic design systems.

Based on our review we conclude that the design of the containment isolation systems is acceptable and meets the intent of General Design Criteria 54, 55, 56 and 57.

Instrument lines that penetrate the containment were assessed in accordance with the guidelines of Regulatory Guide 1.11. These lines have an isolation valve located outside containment as close as possible to containment. The applicant has installed a 1/4" diameter orifice in each of these lines inside the primary containment. Each instrument line has the capability to be tested for leakage. Based on our review we conclude that the instrument lines are designed to meet the intent of Regulatory Guide 1.11.

6.2.5 Combustible Gas Control

Following a loss-of-coolant accident, hydrogen may accumulate within the containment as a result of (1) metal-water reaction between the fuel cladding and the reactor coolant, (2) radiolytic decomposition of the post-accident emergency cooling water. The applicant has analyzed the production and accumulation of hydrogen within containment from the above sources using the guidelines of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident". The applicant will provide a redundant hydrogen mixing system and redundant hydrogen recombiners to limit the hydrogen concentration within the containment to below the Regulatory Guide 1.7 limit of 4 volume percent.

Both the hydrogen recirculation system and hydrogen recombiners are completely redundant systems, designed to seismic Category I criteria and containment post-accident environmental conditions, and will be protected from postulated missiles and pipe whip.

The recirculation system has four, 20,000 cfm fans (two required) and three inlet penetrations, each arranged in parallel to ensure opening capability when required. Each of the two recombiners is of 100% capacity and located within primary containment. In accordance with Regulatory Guide 1.7, the applicant will also provide a controlled purge system as a backup to the recombiner system.

The applicant has used the same assumptions as Regulatory Guide 1.7 to calculate the rate of hydrogen released by radiolysis and consistent with the Regulatory Guide has assumed that hydrogen is released as a result of 5% metal-water reaction of the fuel cladding although the analysis of the emergency core cooling system indicates that the metal-water reaction will be limited to much less than 1%. For the determination of containment hydrogen concentration, the applicant calculates the rate at which hydrogen is released by metal-water reaction by arbitrarily maintaining the cladding peak temperature at 2300°F using a core power distribution more conservative than the ECCS/IAC power distribution (which is the acceptable limit for emergency core cooling analysis) until the equivalent of 5% of the cladding is reacted. With these assumptions the Regulatory Guide 1.7 hydrogen flammability limit (4%) would be reached in the drywell at approximately one hour following the accident.

Although the calculated metal-water reaction rate is based on on conservative assumptions, we will require the hydrogen recirculation system, which provides mixing of the drywell and containment volumes, to have an earlier starting capability to provide additional margin. This design capability would assure an adequate margin to cope with hydrogen that may be evaluated very early in a LOCA situation.

In response to our requirements the applicant has proposed that the operator manually start the hydrogen recirculation system following an accident upon clearance of certain interlocks. The proposed interlocks would be a 10-minute time delay, hydrogen concentration near 4% by volume, and containment spray system in operation. The staff has reviewed the design and has determined that the system should be automatically actuated and that additional interlocks may be required; however, resolution of the specific design will depend on the containment spray system design which is selected as discussed in Section 6.2.1.6. Therefore when this information is available we will complete our review and report our conclusions in a Supplement to the Safety Evaluation.

The applicant has calculated that the Regulatory Guide 1.7 flammability limit (4%) would not be reached in the drywell and containment until nine days following the accident. Early actuation of the recirculation systems provides adequate and rapid mixing.

The applicant can therefore operate the hydrogen recombiners well in advance of this time to limit hydrogen concentrations to below this limit. We find the applicant's method of analysis to be acceptable.