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October 26, 1982
EF2 - 60,107

Mr. L. L. Kintner
U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Division of Licensing
Washington, D. C. 20555

Dear Mr. Kintner:

References: (1) Enrico Fermi Atomic Power Plant, Unit 2 NRC Docket No. 50-341

(2) NRC to Detroit Edison letter, "Concerns Regarding the Adequacy of the Design Margins of the Mark I and II Containment Systems", July 8, 1982

(3) GE to NRC letter, "Mark I Containment Program Humphrey Containment Concerns", MFN 138-82, September 24, 1982

Subject: Mark I Containment - Humphrey's Concerns

In response to Reference 2 and a similar letter sent generically to the BWR Mark I Containment Owner's Group, the Owners Group has prepared a generic response. Detroit Edison has reviewed the Reference 3 letter and attachments and determined that they are applicable to Fermi 2. This response was previously submitted to the NRC by GE as indicated in Reference 3. Forty copies of the response are also attached to this letter for your use.

If you should have any questions, please contact Mr. L. E. Schuerman (313) 649-7562.

Sincerely,

Harry Tauber

D005

Attachment

cc: B. Little

GENERAL ELECTRIC

NUCLEAR POWER

SYSTEMS DIVISION

GENERAL ELECTRIC COMPANY, 175 CURTNER AVE., SAN JOSE, CALIFORNIA 95125
MC 682, (408) 925-5040

MFN 138-82

September 24, 1982

Darrel G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: MARK I CONTAINMENT PROGRAM
HUMPHREY CONTAINMENT CONCERNS

Reference: Letter D. G. Eisenhut to R. Logue, Chairman, Mark I Owners Group, Same Subject, dated July 15, 1982.

Your July 15, 1982 letter was sent to Mr. Logue as Chairman of the Mark I Owners Group. It requested the Mark I Owners Group to address the Humphrey Concerns which the Nuclear Regulatory Commission (NRC) has identified as being potentially applicable to the Mark I Containment. The Mark I Owners Group requested the General Electric Company, as Program Manager for the Mark I Containment Program, to submit the Mark I response to the Humphrey Containment Concerns in behalf of the Mark I Owners. The enclosure to this letter provides the requested Mark I Owners Group response.

The Mark I Owners have elected to respond to the Humphrey Concerns generically by grouping the concerns into fourteen technical areas, similar to the area groupings presented to the Advisory Committee on Reactor Safeguards (ACRS) Fluid Dynamic Subcommittee on July 29, 1982 by Mr. M. Fields of the NRC. The Mark I Owners consider that the NRC request has been responded to in a responsible manner and that the the response is satisfactory.

In summary, the responses indicate that all of the concerns fall into the following categories:

1. Not applicable to the Mark I Program.

2. Previously resolved by earlier Mark I or other programs.

3. Insignificant to the design and safety of the Mark I Containment.

The enclosed response supports the summary remarks of the ACRS Fluid Dynamic Subcommittee on July 30, 1982 in that most of the concerns are either inapplicable or insignificant with regards to Mark I safety margins.

On behalf of the Mark I Owners, General Electric Company trusts that the NRC will find the enclosed responses acceptable as the Mark I Owners Group is anxious to terminate the effort it has applied to respond to the Humphrey Containment Concerns.

Very truly yours,

Glenn Sherwood

G. G. Sherwood, Manager
Nuclear Safety & Licensing Operation

Enclosure

cc: Mark I Owners Group
L. S. Gifford (GE Liaison Office)

NSEO-88-0982
DRF# T23-00452

MARK I OWNERS

RESPONSE TO

HUMPHREY CONTAINMENT CONCERNS

September, 1982

Prepared by: PFStanwage for
P.F. Billig
Plant Performance Engineering

Approved by: [signature] for
A.E. Rogers, Manager
Plant Performance Engineering

8210010224

ABSTRACT

Responses have been generated for all Humphrey containment concerns as they relate to Mark I. These responses indicate that all of the concerns fall into one of the the following categories:

1. Not applicable to Mark I.

2. Previously addressed in Mark I or other programs.

3. Insignificant to the design and safety of Mark I containment.

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INTRODUCTION

On May 8, 1982 a number of concerns regarding the adequacy of the General Electric (GE) Mark III containment design were raised by a former GE employee, J.M. Humphrey. Although these concerns were specifically raised for the Mark III containment, the Nuclear Regulatory Commission (NRC) felt that some of the issues may apply to the Mark I containment design.

On July 15, 1982 the NRC requested that the Mark I Owners Group address those concerns which the NRC had identified as being potentially applicable to the Mark I containment. It was decided by the Mark I Owners that a generic assessment would be appropriate to address concerns.

The Advisory Committee on Reactor Safeguards (ACRS) Fluid Dynamics sub-committee met in San Jose on July 29 and 30, 1982 to review the Humphrey Containment Concerns. The NRC at that meeting grouped the concerns into 21 technical areas. Of those 21 areas 14 contain concerns which were raised for Mark I.

The Mark I Owners Group decided to respond to the concerns grouped into the fourteen technical areas. GE is responding on behalf of the Owners Group.

The generic assessment of these concerns for the Mark I containment design has shown that this assessment resolves each of the concerns. Therefore no further plant-unique analyses are required.

The responses indicate that all of the concerns fall into one of the following categories:

1. Not applicable to Mark I.
2. Previously addressed in Mark I or other programs.
3. Insignificant to the design and safety of the Mark I containment.

Index Between Technical Areas and Humphrey Concerns

Area	Description	Humphrey Concerns
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AREA I

ECCS RELIEF LINE DISCHARGE LOADS

3.1 Concern

The design of the STRIDE plant apparently did not consider vent clearing, condensation oscillation and chugging loads which might be produced by the actuation of these relief valves.

3.2 Concern

The STRIDE design provided only nine inches of submergence above the RHR relief valve discharge lines at minimum drawdown suppression pool levels.

3.3 Concern

Discharge from the RHR relief valves may produce bubble discharge or other submerged structure loads on equipment in the suppression pool.

3.4 Concern

The RHR heat exchanger relief valve discharge lines are only provided with small vacuum breakers to prevent negative pressure in the lines when the valves close. If the valves experience repeated actuation, the vacuum breaker sizing may not be adequate to prevent drawing slugs of water back through the discharge piping. These slugs of water may apply impact loads to the relief valve or be discharged back into the pool at the next relief valve actuation and cause higher pipe pressures, clearing loads and potential RHR HX overpressurization.

3.6 Concern

If the RHR heat exchanger relief valves discharge steam to the upper levels of the suppression pool following a design basis accident, they may significantly aggravate suppression pool temperature stratification and discharge line condensation loads.

3.7 Concern

The concerns related to the RHR heat exchanger relief valve discharge lines should also be addressed for all other relief lines that exhaust into pool. (p. 132 of 5/27/82 transcript)

RESPONSE TO AREA I CONCERNS

A survey of Mark I ECCS relief lines indicates that except for the BWR/4 they do not discharge into the suppression pool. Later BWR/4 plants have relief lines on the RHR heat exchangers to protect the heat exchangers when operating in the steam condensing mode; however, these plants have positive procedures to prevent the use of this mode of RHR operation during normal shutdown.

AREA I (Continued)

Operating experience has confirmed that the steam condensing mode of the RHR is used less frequently than once every five years. The procedures ensure that the pressure in the heat exchanger is maintained below the relief valve setpoint. There has been no recorded instance of a heat exchange relief valve opening.

Even if this relief valve were to open during operation in the steam condensing mode, the consequent loads are expected to be within the capability of the Mark I containment. The maximum steam flow rate is approximately 100 lb/sec, less than half the flow rate from the main steam safety-relief valves (S/RV). Since loads are proportional to flow rate, it is expected that the loads from the heat exchanger relief lines would be less than half the load from the main steam S/RV lines with ramshead devices. Mark I plants have operated with ramshead devices on the mainsteam S/RV lines for several years before strengthening the containment without adverse effects on the containment. The infrequent actuation of the heat exchanger relief valve which could cause loads less than half of loads already accomodated by several Mark I plants is not significant to the design and safety of the containment.

Conclusion:

This concern is not applicable to most Mark I plants. For BWR/4 plants, the concern is insignificant to the design and safety of the containment.

AREA II

ISOLATION OF WATER IN DRYWELL

4.1 Concern

The present containment response analyses for drywell break accidents assume that the ECCS systems transfer a significant quantity of water from the suppression pool to the lower regions of the drywell through the break. This results in a pool in the drywell which is essentially isolated from the suppression pool at a temperature of approximately 135°F. The containment response analysis assumes that the drywell pool is thoroughly mixed with the suppression pool. If the inventory in the drywell is assumed to be isolated and the remainder of the heat is discharged to the suppression pool, an increase in bulk pool temperatures of 10°F may occur. This concern is related to the trapping of water in the drywell.

4.2 Concern

The existence of the drywell pool is predicated upon continuous operation of the ECCS. The current emergency procedure guidelines require the operators to throttle ECCS operation to maintain vessel level below level 8. Consequently the drywell pool may never be formed. Not applicable to Mark II facilities.

RESPONSE TO AREA II CONCERNS

The drywell liquid holdup volume in Mark I plants is part of the flowpath from drywell tp torus and is very shallow because the bottom of the drywell is filled with concrete. A typical liquid holdup volume is less than 5% of the total suppression pool volume. The increase in suppression pool temperature caused by a 10% reduction in suppression pool mass is less than 5°F. This possible increase is not significant and is well bounded by conservatisms in the analysis. Therefore, irrespective of operator action with ECCS, the volume of the suppression pool will not be significantly reduced.

In addition, the pool temperature is not the controlling factor for Mark I containment pressure. The peak presure occurs in the short-term, before a suppression pool mass reductiondue to holdup could occur, while the peak suppression pool temperature occurs in the long-term.

Conclusion:

This concern is insignificant to the design and safety of Mark I plants.

AREA III

BULK POOL TEMPERATURE IN DBA ANALYSIS

4.3 Concern

All Mark III analyses presently assume a perfectly mixed uniform suppression pool. These analyses assume that the temperature of the suction to the RHR heat exchangers is the same as the bulk pool temperature. In actuality, the temperature in the lower part of the pool where the suction is located will be as much as 7-1/2 $^{\circ}$F cooler than the bulk pool temperature. Thus, the heat transfer through the RHR heat exchanger will be less than expected.

4.4 Concern

The long term analysis of containment pressure/temperature response assumes that the wetwell airspace is in thermal equilibrium with the suppression pool water at all times. The calculated bulk pool temperature is used to determine the airspace temperature. If pool thermal stratification were considered, the surface temperature, which is in direct contact with the airspace, would be higher. Therefore the airspace temperature (and pressure) would be higher.

4.5 Concern

A number of factors may aggravate suppression pool thermal stratification. The chugging produced through the first row of horizontal vents will not produce any mixing from the suppression pool layers below the vent row. An upper pool dump may contribute to additional suppression pool temperature stratification. The large volume of water from the upper pool further submerges RHR heat exhanger effluent discharge which will decrease mixing of the hotter, upper regions of the pool. Finally, operation of the containment spray eliminates the heat exhanger effluent discharge jet which contributres to mixing. For Mark I and II facilities, confine your response on this issue to those concerns which can lead to pool stratification (e.g., operation of the containment spray).

7.1 Concern

The wetwell is assumed to be in thermal equilibrium with a perfectly mixed, uniform temperature suppression pool. As noted under topic 4, the surface temperature of the pool will be higher than the bulk pool temperature. This may produce higher than expected containment temperatures and pressures.

7.3 Concern

The analysis assumes that the wetwell airspace is in thermal equilibrium with the suppression pool. In the short term this is non-conservative for Mark III due to adiabatic compression effects and finite time required for heat and mass to be transferred between the pool and containment volumes.

RESPONSE TO AREA III CONCERNS

The Mark I program has addressed the issue of suppression pool stratification, and the resolution is contained in NUREG-0661.

AREA III (continued)

During the short term response to a postulated LOCA, testing in the Mark I Full Scale Test Facility (FSTF) demonstrated effective suppression pool mixing during condensation oscillation and chugging. In addition, the FSTF demonstrated the conservatism of the current assumptions and predictions regarding pool temperature, wetwell air temperature, and any adiabatic compression effects which would occur for LOCA's in the Mark I containment geometry.

During the long term response to a LOCA, initiation of the RHR system in either the pool cooling mode or the spray mode would provide effective suppression pool mixing to limit or eliminate stratification.

Containment spray will not cause stratification in Mark I plants. The wetwell spray diverts only 5% of the RHR flow; 95% of the flow is still available for suppression pool mixing, via the pool cooling mode or the drywell spray mode.

If the drywell spray is used the water return to the suppression pool is through the vent system. This system provides a uniform circumferential return which is submerged 3 to 4 feet in the pool. Consequently adequate suppression pool mixing to eliminate thermal stratification is assured during drywell spray operation.

Conclusion:

This issue has already been addressed in the Mark I program. NUREG-0661 summarizes the resolution. The RHR provides effective pool mixing; therefore, the concerns are not applicable.

AREA IV

ASPECTS OF THE RHR SYSTEM

4.6 Concern

The initial suppression pool temperature is assumed to be 95°F for all GGNS accident analyses as noted in FSAR table 6.2-50. If the service water temperature is consistently higher than expected, as occurred at Kuosheng, the RHR system may be required to operate nearly continuously in order to maintain suppression pool temperature at or below the maximum permissible value.

4.7 Concern

All analyses completed for the Mark III are generic in nature and do not consider plant specific interactions of the RHR suppression pool suction and discharge.

4.8 Concern

Operation of the RHR system in the containment spray mode will decrease the heat transfer coefficient through the RHR heat exchangers due to decreased system flow. The FSAR analysis assumes a constant heat transfer rate for the suppression pool even with operation of the containment spray.

4.9 Concern

The effect on the long term containment response and the operability of the spray system due to cycling the containment sprays on and off to maximize pool cooling needs to be addressed. Also provide and justify the criteria used by the operator for switching from the containment spray mode to pool cooling mode, and back again. (pp. 147-148 of 5/27/82 transcript).

4.10 Concern

Justify that the current arrangement of the discharge and suction points of the pool cooling system maximizes pool mixing. (pp. 150-155 of 5/27/82 transcript)

5.3 Concern

Leakage from the drywell to containment will increase the temperature and pressure in the containment. The operators will have to use the containment spray in order to maintain containment temperature and pressure control. Given the decreased effectiveness of the RHR system in accomplishing this objective in the containment spray mode, the bypass leakage may increase the cyclical duty of the containment sprays.

14. Concern

A failure in the check valve in the LPCI line to the reactor vessel could result in direct leakage from the pressure vessel to the containment atmosphere. This leakage might occur as the LPCI motor operated isolation valve is closing and the motor operated isolation valve in the containment spray line is opening. This could produce unanticipated increases in the containment spray.

AREA IV (continued)

RESPONSE TO AREA IV CONCERNS

For Mark I plants the RHR heat exchangers operate as effectively in the spray mode as the pool cooling mode. In either the spray or the pool cooling configuration, water is drawn from the suppression pool, passed through the heat exchangers, and returned to the pool at the same flow rate. The heat removal rate is the same for both modes of RHR operation. Therefore, with pool cooling not affected by either mode of operation there is no need to cycle from the spray mode to the pool cooling mode and the Mark I RHR design ensures effective heat removal in either the spray or pool cooling mode.

All suppression pool temperature analyses begin with the pool temperature at the high technical specification value for normal operation. Mark I plants are required by NUREG-0661 to monitor the suppression pool temperature and to operate within the technical specification limits.

Plant-specific interactions of the RHR suction and discharge locations have been addressed by the Mark I Long Term Program. The NRC in its Safety Evaluation Report (NUREG-0661, p.A-42) requires:

> The local to bulk pool temperature difference shall consider the plant-specific quencher discharge geometry and RHR suction and discharge geometry.

During normal plant operation there is no path from the vessel to the wetwell spray header because there are three normally closed valves and an interlock to prevent flow in addition to the check valve. The LPCI line will only following a LOCA and after the reactor has significantly depressurized. Mark I plants have an LPCI injection valve vessel pressure permissive signal which prevents opening the LPCI line at high pressure. In addition, there must be a failure of the LPCI pump because if the pump is running, it has sufficient discharge pressure to maintain the flow direction into the reactor. Therefore, the scenario postulated would require a line break, failure of the check valve, and LPCI pump failure. Postulating two specific independent failures in addition to the initiating event is beyond the licensing design basis for Mark I plants and is therefore not considered.

For Mark I plants with BWR/2 reactors this concern does not apply as they do not have an LPCI.

Conclusion:

These issues have either been addressed in previous Mark I programs, or they are not applicable to Mark I plants.

AREA V

STEAM BYPASS

5.1 Concern

The worst case of drywell to containment bypass leakage has been established as a small break accident. An intermediate break accident will actually produce the most significant drywell to containment leakage prior to initiation of containment sprays.

5.2 Concern

Under Technical specification limits, bypass leakage corresponding to $A/\sqrt{K}$=0.1 ft² constitute acceptable operating conditions. Smaller-than-IBA-sized breaks can maintain break flow into the drywell for long time periods, however, because the RPV would be depressurized over a 6 hour period. Given, for example, an SBA with $A/\sqrt{K}$ = 0.1, projected time period for containment pressure to reach 15 psig is 2 hours. In the latter 4 hours of the depressurization the containment would presumably experience ever-increasing overpressurization. For Mark I and II facilities, refer to Appendix I to Section 6.2.1.1C of the Standard Review Plan (SRP).

5.5 Concern

Equipment may be exposed to local conditions which exceed the environmental qualification envelope as a result of direct drywell to containment bypass leakage.

5.8 Concern

The possibility of high temperatures in the drywell without reaching the 2 psig high pressure scram level because of bypass leakage through the drywell wall should be addressed. (pp. 168-174 of 5/27/82 transcript)

8.3 Concern

If the containment is maintained at -2 psig, the top row of vents could admit blowdown to the suppression pool during an SBA without a LOCA signal being developed. Not applicable to Mark II facilities.

9.2 Concern

The continuous steaming produced by throttling the ECCS flow will cause increased direct leakage from the drywell to the wetwell. This could result in increased wetwell pressures.

RESPONSE TO AREA V CONCERNS

The only pathway from drywell to wetwell in a Mark I containment is the vent system. The vent system is of all-welded construction with the exception of wetwell to drywell vacuum breakers. The Mark I design initiates scram and LOCA signals at 2 psig containment pressure, rather than a differential between wetwell

AREA V (Continued)

and drywell. In Mark I's the wetwell is limited to a maximum negative pressure of -.5 psig by reactor building to wetwell vacuum breakers, and limited to a maximum negative drywell pressure, relative to wetwell, of -.2 psi by wetwell to drywell vacuum breakers.

Regardless of initial conditions or extent of bypass leakage, scram and LOCA signals are generated by 2 psig drywell pressure, and therefore the concerns expressed in 5.8 and 8.3 are not applicable to Mark I.

The integrity of wetwell to drywell vacuum breakers with respect to bypass leakage is maintained by tech spec required surveillance tests.

In Mark I containment plants, the bypass leakage permitted is not of concern because it is only a long term containment responce phenomenon, and the operator has the availability of drywell spray, vessel reflood, and ADS to respond to the full spectrum of SBA and IBA LOCAS as directed by EPG's.

Conclusion:

This issue is not applicable to Mark I plants.

AREA VI

HYDROGEN CONTROL SYSTEM

5.4 Concern

Direct leakage from the drywell to the containment may dissipate hydrogen outside the region where the hydrogen recombiners take suction. The anticipated leakage exceeds the capacity of the drywell purge compressors. This could lead to pocketing of hydrogen which exceeds the concentration limit of the 4% by volume. This concern applies to those facilities at which hydrogen recombiners can be used.

6.1 Concern

We understand that GE has recommended for Mark III containments that the combustible gas control systems be activated if the reactor vessel water level drops to within one foot of the top of the active fuel. Indicate what your facility is doing in regard to this recommendation.

6.2 Concern

General Electric has recommended that an interlock be provided to require containment spray prior to starting the recombiners because of the large quantities of heat input to the containment. Incorrect implementation of this interlock could result in inability to operate the recombiners without containment spray. This concern applies to those facilities at which hydrogen recomobiners can be used.

6.3 Concern

The recombiners may produce ''hot spots'', near the recombiner exhausts which might exceed the environmental qualification envelope or the containment design temperature. This concern applies to those facilities at which hydrogen recombiners can be used.

6.4 Concern

For the containment air monitoring system furnished by General Electric, the analyzers are not capable of measuring hydrogen concentration at volumetric steam concentrations above 60%. Effective measurement is precluded by condensation of steam in the equipment.

6.5 Concern

Discuss the possibility of local temperatures due to recombiner operation being higher than the temperature qualification profiles for equipment in the region around and above the recombiners. State what instructions, if any, are available to the operator to actuate containment sprays to keep this temperature below design values. This concern applies to those facilities at which hydrogen recombiners can be used.

AREA VI (Continued)

21. Concern

Regulatory Guide 1.7 requires a backup purge hydrogen removal capability. This backup purge for Mark III is via the drywell purge line which discharges to the shield annulus which in turn is exhausted through the standy gas treatment system (SGTS). The containment air is blown into the drywell via the drywell purge compressor to provide a positive purge. The compressors draw from the containment, however, without hydrogen lean air makeup to the containment, no reduction in containment hydrogen concentration occurs. It is necessary to assure that the shield annulus volume contains a hydrogen lean mixture of air to be admitted to the containment via containment vacuum breakers. For Mark I and II facilities, discuss the possibility of purge exhaust being mixed with the intake air which replenishes the containment air mass.

RESPONSE TO AREA VI CONCERNS

Regulation 10CFR50.44 requires inerting of all Mark I containments, including both wetwell and drywell. This Regulation also prohibits venting the containment for hydrogen control.

With an inerted containment hydrogen pocketing does not produce a flammable condition and concern 5.4 is not applicable.

For those Mark I units installing recombiners or provisions for recombiners, the matters of gas monitoring instrumentation and recombiner discharge hot spots are specifically addressed.

Conclusion:

The issues have been addressed in design and licensing of Mark I plants.

AREA VII

EMERGENCY PROCEDURE GUIDELINES

17. Concern

The EPGs contain a curve which specifies limitations on suppression pool level and reactor pressure vessel pressure. The curve presently does not adequately account for upper pool dump. At present, the operator would be required to initiate automatic depressurization when the only action required is the opening of one additional SRV. This issue as phrased applies only to a Mark III facility. However, the concern can be generalized. Accordingly, discuss what actions the reactor operator would take in the event that the limitations on the suppression pool level and the pressure in the reactor vessel are violated.

22. Concern

The EPGs currently in existence have been prepared with the intent of coping with degraded core accidents. They may contain requirements conflicting with design basis accident conditions. Someone needs to carefully review the EPGs to assure that they do not conflict with the expected course of the design basis accident.

RESPONSE TO AREA VII CONCERNS

The concern, that the operator must initiate the automatic depressurization system (ADS) to keep the plant below the suppression pool load limit curve, is unfounded. The EPG states that if the suppression pool water level cannot be maintained below the suppression pool load limit curve, then the RPV pressure must be maintained below the corresponding pressure limit. Only if other available methods fail to control pool water level and RPV pressure below the suppression pool load limit curve is ADS required.

A broad spectrum of events including postulated design basis accidents has been considered in developing the Guidelines. The Guidelines have been carefully reviewed by General Electric, the BWR Owners' Group, and the NRC.

Conclusion:

These issues are not applicable to Mark I.

AREA VIII

CONTAINMENT ATMOSPHERE RESPONSE

7.2 Concern

The computer code used by General Electric to calculate environomental qualification parameters considers heat transfer from the suppression pool surface to the containment atmosphere. This is not in accordance with the existing licensing basis for Mark III environmental qualification. Additionally, the bulk suppression pool temperature was used in the analysis instead of the suppression pool surface temperature. This issue as phrased applies only to a Mark III facility. However, the concern can be generalized and applied to the earlier containment types. For Mark I and II facilities, indicate what methodology was used to calculate the environmental qualification parameters including a discussion of heat transfer between the atmosphere in the wetwell and the suppression pool.

9.1 Concern

The current FSAR analysis is based upon continuous injection of relatively cool ECCS water into the drywell through a broken pipe following a design basis accident. Since the operator is directed to throttle ECCS operation to maintain the reactor vessel water level to about the level of the steam lines, the break will be releasing saturated steam instead of releasing relatively cool ECCS water. Therefore, the drywell air which would have been purged and then drawn back into the drywell, will remain in the wetwell and higher pressures than anticipated will result in both the wetwell and the drywell.

RESPONSE TO AREA VIII CONCERNS

The concern relates to the environmental qualification of equipment in the wetwell. In most Mark I plants there is no Class 1E equipment in the wetwell airspace which is sensitive to elevated temperature so heat transfer between the pool and the airspace is not relevant. Those plants which have equipment in the wetwell airspace have qualified the equipment to peak drywell temperatures in excess of 300°F, which are well above the temperature peak expected in the wetwell.

In Mark I plants long term high pressures following a line break are not a concern. The operator has the availability of drywell spray and vessel reflood to respond to the LOCA and to effect the wetwell to drywell air transfer.

Finally, it should be noted that the containment peak pressure in Mark I plants is governed by short-term response to a pipe break rather than the long-term response. The long-term effect of containment atmosphere response does not affect Mark I design.

Conclusion:

These issues do not apply to Mark I plants.

AREA IX

TECH. SPECS. VS DBA ASSUMPTIONS

11. Concern

Mark III load definitions are based upon the levels in the suppression pool and the drywell weir annulus being the same. The GGNS technical specifications permit elevation differences between these pools. This may effect load definition for vent clearing. For Mark I and II facilities, consider the water in the downcomers.

RESPONSE TO AREA IX CONCERNS

For the water elevation in the downcomers to be different than the suppression pool level there must exist a pressure difference between the drywell and the wetwell.

For Mark I plants with differential pressure control, there are technical specifications on the differential pressure which limit the drywell pressure to a minimum value, typically 1 psi, above the wetwell pressure. These tech specs effectively control the relative elevation of the water in the downcomers with respect to the pool water elevation. LOCA air claring loads are less severe under this differential pressure condition.

In Mark I plants without differential pressure control, normal plant operation would tend to limit drywell pressure to values no lower than wetwell pressure. In typical operation, the water level in the downcomers would be below or the same as pool levels. This condition would result in loads lower than or equal to those specified for design.

In the event of a pressure in the drywell lower than the pressure in the wetwell, the vacuum breakers between the wetwell and the drywell limit the wetwell pressure from increasing above the drywell pressure by more than the vacuum breaker setpoint. This opening pressure is typically 0.2 psi which would translate into an additional 0.46 ft. of water in the downcomers. The effect of this additional downcomer water leg during a postulated DBA can be estimated from the Mark I Program Quarter Scale Test Facility pool swell tests. The up loads would increase by less than 5% and the down loads by less than 10% over the condition where wetwell and drywell pressures were equal. These small increases are bounded by conservatisms in the development of the initial conditions for the Mark I Program load definition for DBAs, for example, drywell pressurization rate is typically 11% higher than expected due to margins in the initial conditions and calculations. The simulation of compressibility effects in the pool swell tests also gives a margin of approximately 15% in the pool swell loads. There are a number of other conservatisms such as an instantaneous break, no condensation in the drywell, and the use of air as the blowdown fluid which, while not quantified, make the design loads higher than expected.

Conclusion:

The effect of nominal variations in drywell and wetwell pressure are bounded by margins included in the Mark I Program.

AREA X

CONTAINMENT NEGATIVE PRESSURE

8.1 Concern

This issue is based on consideration that some Tech Specs allow operation at parameter values that differ from the values used in assumptions for FSAR transient analyses. Normally analyses are done assuming a nominal containment pressure equal to ambient (0 psig) a temperature near maximum operating (90°F) and do not limit the drywell pressure equal to the containment pressure. The Tech Specs permit operation under conditions such as a positive containment pressure (1.5 psig), temperatures less than maximum (60 of 70°F) and drywell pressure can be negative with respect to the containment (-0.5 psid). All of these differences would result in transient response different than the FSAR descriptions.

8.2 Concern

The draft GGNS technical specifications permit operation of the plant with containment pressure ranging between 0 and -2 psig. Initiation of containment spray at a presure of -2 psig may reduce the containment pressure by an additional 2 psig which could lead to buckling and failures in the containment liner plate.

8.4 Concern

Describe all of the possible methods both before and after an accident of creating a condition of low air mass inside the containment. Discuss the effects on the containment design external pressure of actuating the containment sprays. (pp. 190-195 of 5/27/82 transcript).

RESPONSE TO AREA X CONCERNS

The Mark I Long Term Program considered plant operation at parameter values which accounted for Tech Specs and operating experience, with the objective of selecting a set of conditions which would produce conservative transient responses. The NRC reviewed the assumptions made in the Mark I analyses and approved them in NUREG-0661.

Technical specifications for the Mark I reactor building to wetwell vacuum breakers limit operation to wetwell pressures above -0.5 psig. The vacuum breaker setpoints and size have been established to limit the minimum pressure to values well within the containment capability. If the containment pressure were reduced then the effect of evaporative cooling due to inadvertent spray would be less severe because the rate of depressurization would be reduced relative to normal pressure conditions. For the case of inadvertent spray in a high humidity environment, the vacuum breakers would open at the same conditions as during a transient from normal pressure. After vacuum breaker opening, the transient pressure response would be similar to the response from normal initial conditions. The vacuum breakers are sized to limit the maximum external pressure to within design values.

Mark I containments are isolated during normal plant operation and after accidents. There is no feasible method that would reduce the containment air mass.

Conclusion:

This issue was resolved in the Mark I program.

AREA XI

TREATMENT OF SRV ACCIDENTS AND SBAs

9.3 Concern

It appears that some confusion exists as to whether SBA's and stuck open SRV accidents are treated as transients or design basis accidents. Clarify how they are treated and indicate whether the initial conditions were set at nominal or licensing values.

RESPONSE TO AREA XI CONCERN

For Mark I plants, SBA's and stuck open SRV transients are addressed in the NRC Acceptance Criteria for the Mark I Program (NUREG-0661). In analyzing these accidents the initial conditions are set at licensing values.

Conclusion:

This issue was resolved in the Mark I program.

AREA XII

SECONDARY CONTAINMENT NEGATIVE PRESSURE

15. Concern

The STRIDE plants had vacuum breakers between the containment and the secondary containment. With sufficiently high flows through the vacuum breakers to containment, vacuum could be created in the secondary containment. There were no requirements to design for the vacuum that will be created in the annulus when the vacuum breakers open.

RESPONSE TO AREA XII CONCERN

Conservative analysis has been done which shows that the vacuum created in the containment buildings of typical Mark I plants because of wetwell and drywell sprays is less than 0.5 psi. This small reduction in the containment building pressure is a result of the large volume for the containment building relative to the wetwell and drywell volume space. Mark I plants have containment building volumes which are 6 to 12 times as large as the wetwell and drywell free space volumes.

The reactor buildings in Mark I plants are already evaluated for certain external loads due to tornados which are substantially higher than the expected load due to spray operation.

Conclusion:

This concern has an insignificant effect when compared to the design load for the reactor building.

AREA XIII

POOL TEMPERATURE SENSOR LOCATIONS

16. Concern

Some of the suppression pool temperature sensors are located (by GE recommendation) 3'' to 12'' below the pool surface to provide early warning of high pool temperature. However, if the suppression pool is drawn down below the level of the temperature sensors, the operator could be misled by erroneous readings and required safety action could be delayed.

RESPONSE TO AREA XIII CONCERN

The requirements and recommendations for suppression pool temperature sensor locations are contained in NUREG-0661 and NUREG-0783. The suppression pool temperature monitoring system (SPTMS) is required to ensure that the suppression pool temperature is within the allowable operating technical specification limits. The SPTMS sensors are located below the post-LOCA drawdown suppression pool level such that they will always monitor water and not air temperature. Therefore, it is not justified to assume that the operator could be misled by reading values of air temperature rather than water temperature.

Conclusion:

This concern has been addressed in the Mark I Program.

AREA XIV

INSULATION DEBRIS

18.1 Concern

Failure of reflective insultation in the drywell may lead to blockage of the gratings above the weir annulus. This may increase the pressure required in the drywell to clear the first row of drywell vents and perturb the existing load definitions.

This issue as phrased applies only to a Mark III facility. However, the concern can be generalized. Accordingly, discuss how the effects of insulation debris could perturb existing load definitions or could block suction strainers. In responding to this issue, you may refer to existing generic studies; e.g., the study done for the Cooper facility.

18.2 Concern

Insulation debris may be transported through the vents in the drywell wall into the suppression pool. This debris could then cause blockage of the suction strainers.

RESPONSE TO AREA XIV CONCERNS

The effects of insulation debris on existing load definitions are negligible. Mark I plants have vent deflectors protecting the entrances of the main vents. The narrow entrance of the deflectors will prevent massive pieces of insulation from entering the vent system and blocking flow. The probability of the entrance being blocked at the deflectors is small due to the large entrance area around the deflector. This entrance area is equal to the flow area of the main vent.

Even if there is some blockage near the entrance to the main vents the containment loads would not increase. Chugging and condensation oscillation loads will remain the same due to the vent ring header equalizing downcomer pressures. Tests have also shown that complete blockage of a main vent will not cause asymmetric pool swell and that the effect on the peak loads is insignificant (NCID-17539, ''Lawrence Livermore Laboratory - Mark I 1/5- Scale Boiling Water Reactor Pressure Suppression Experiment, Summary of Effects due to Vent Line Orifice Variations - Air Test Series'', July 28, 1977).

If some insulation works its way into the vent pipe, it would still face a tortuous path to the torus. Each vent pipe terminates at the ring header. Insulation debris must negotiate a 90° turn upon reaching the ring header after sliding along a shallow angle in the vent pipe. Downcomers off the ring headers limit the insulation size still further. A ''lip'' exists where the downcomers connect to the ring header, further adding to the difficulty for debris to enter a downcomer.

The ECCS suction strainers are designed for large amounts of clogging. The strainers can handle approximately 50% blockage and still maintain the required rated flow. The location of each intake screen in a different area of the wetwell

AREA XIV (Continued)

along with its excess screening capacity makes it unlikely that more than one core cooling pump suction would be impacted by a postulated pipe break. Even then, it is doubtful that the debris would be in sufficient quantity to block the flow area required for safe operation of the pump involved.

Studies including the Cooper plant have been conducted to determine the potential amount of debris expected following an accident (NUREG-CR2403, ''Survey of Insulation in Nuclear Plants and Potential for Debris Generation,'' October 1981). These studies conclude that insulation debris will not significantly block the ECCS suction strainers.

Conclusion:

This issue has been addressed in the plant design and reviewed by other programs. It is not applicable to Mark I plants.