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November 1, 1978

United States Nuclear Regulatory Commission
Washington, D.C.
attn: Mr. L. L. Kintner

Reference: Enrico Fermi Atomic Power Plant, Unit 2, NRC Docket No. 50-341

Subject: FSAR Amendment 16

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Appended is an errata to the Fermi FSAR Amendment 16 which you recently received.

Yours truly,

Frank M Coté

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Assistant General Manager
Publications Division

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FERMI AMENDMENT 16 ERRATA

Recipients of the Enrico Fermi Atomic Power Plant, Unit 2, Final Safety Analysis Report, Amendment 16, may have deleted too many pages from their copy(ies) because of a mistake in the Amendment 16 Instruction Sheet.

The instruction on page 9 reads as follows:

Remove Page	Insert Amendment 16 pages
6.2-5/ 6.2-6 thru 6.2-31/6.2-32	6.2-5/6.2-6 thru 6.2-31/6.2-32

In both cases, "thru" should be replaced with "and". Many recipients threw away 32 pages, which left only 4 pages for replacement. To correct this, pages 6.2-7/6.2-8 through 6.2-30a/6.2-30b are attached. Please reinsert these pages in your volume if you have deleted them.

One penetration is used for analog signals, to be used for vibration tests and miscellaneous primary signals.

One group of two penetrations for low voltage shielded instrumentation thermocouple extension lead wire is used to transmit RPV and other equipment temperature signals to recording and readout equipment.

One group of four penetrations is used for neutron monitoring. The penetrations include the following coaxial and triaxial cables per penetration:

a. Three triaxial - for intermediate range monitors
b. Two triaxial - for source range monitor
c. 48 coaxial - for local power range monitor.

All penetrations are sized for a 12-inch diameter nozzle and are hermetically sealed, with provisions for continuous leak detection at design pressure. The penetrations are factory-assembled, prewired and tested, and do not require field welding for installation due to the flange mount design. Radiation shielding is integral, thus minimizing radiation shine; and eliminating overhanging moments which would occur if shielding were mounted externally.

6.2.1.2.1.6 Traversing In-Core Probe Penetration

A total of five traversing in-core probe (TIP) guide tubes pass through the primary containment. Penetrations of these guide tubes, through the primary containment, are sealed by means of brazing which meets the requirements of the ASME Boiler and Pressure Code, Section III. These seals also meet the intent of Section III of the Code even though the Code has no provisions for qualifying the procedures or performance.

6.2.1.2.1.7 Personnel and Equipment Access Lock

One personnel access lock is provided for access to the drywell (Figure 6.2-6). The lock has two gasketed doors in series, and the doors are designed and constructed to withstand the drywell design pressure. The doors are mechanically interlocked to ensure that at least one door is locked. The locking mechanisms are designed so that a tight seal will be maintained when the doors are subjected to either internal or external pressure. The seals are capable of being tested for leakage.

Two equipment access hatches and a rod drive removal hatch are in the spherical portion of the drywell. These hatches have double testable seals and are bolted in place (Figure 6.2-7).

6.2.1.2.1.8 Access to the Pressure Suppression Chamber

Access from the reactor building to the pressure suppression chamber is provided at two locations. These are two four-foot diameter manhole entrances with double-gasketed bolted covers connected to the chamber by four-foot diameter steel pipes. These access ports are bolted closed when primary containment is secured.

6.2.1.2.1.9 Access for Refueling Operations

The head, or top portion of the drywell vessel is removed during refueling operations. This head is held in place by studs and is sealed with a double seal. It is closed when primary containment is required and is opened only when the primary coolant temperature is below 212°F and the pressure suppression system is not required to be operational.

A double seal on the head flange is provided to permit checking leaktightness after the drywell head has been replaced.

6.2.1.2.1.10 Venting and Vacuum Relief System

The primary containment is designed for an external pressure of two psi. It can be vented through the SGTS to limit pressure fluctuations caused by temperature changes during various operating modes. This is accomplished through ventilation purge connections which are normally closed while the reactor is at a temperature greater than 212°F.

The containment atmospheric control system provides makeup nitrogen. Makeup is automatically supplied to the suppression chamber and/or drywell as the pressure drops below a slightly positive pressure design point. Makeup gas is supplied from the nitrogen storage tank via the inerting system. Vacuum breakers are between the drywell and suppression chamber.

Automatic vacuum relief devices are used to prevent the external primary containment pressure from exceeding the design value. The drywell vacuum relief valves draw gas from the pressure suppression chamber, and the pressure suppression chamber vacuum relief device draws air from the reactor building.

A vacuum breaker in series with an air-operated normally closed butterfly valve is used in each of two lines to the atmosphere. One valve (a pilot-operated butterfly valve) is actuated by a differential pressure signal. The second valve is a self-actuating vacuum breaker, opening at a maximum ΔP of 0.5 psi. The valves are sized to provide sufficient mass flowrate to equalize the pressure between the suppression chamber and the reactor building in case of an inadvertent operation of the suppression chamber spray. The flowrate calculation assumed that the vacuum breaker valves failed to open until the differential

pressure reached 1.0 psid. The two separate lines are redundant in that either can provide adequate venting.

The vacuum breakers connecting the suppression chamber and the drywell are sized on the basis of the pressure suppression system test program conducted for Bodega Bay at Moss Landing (Reference 1). The vacuum breaker flow area is proportional to the flow area of the vents connecting the drywell and suppression pool. Their chief purpose is to prevent excessive water level variation in the portion of the vent discharge line which is submerged in suppression pool water. The tests relating to vacuum breaker sizing were conducted by simulating a small system rupture, which tended to cause vent water level variation as a preliminary step in the large rupture test sequence. The vacuum breaker capacity selected on this test basis is more than adequate (typically by a factor of four) to limit the suppression chamber-drywell pressure differential during post-accident drywell cooling operations to within containment system design values.

The Fermi 2 vacuum breakers are described in Table 6.2-4a. The number of suppression chamber to drywell vacuum breaker valves was chosen so that 25 percent (three of 12) could fail to open and adequate venting would still be provided.

The vacuum breaker valves are provided with a magnetic latch that holds the valve disk against the seat so that vibration does not cause the valve to chatter. The close limit switches, located near the bottom of the valve body, are actuated directly by the pallet. This design allows a precise adjustment of the limit switch set point to a very slight opening of the pallet. The transfer point of the switch from the close to open position is measured electrically using an ohm meter or other continuity device. With the switch properly adjusted, the maximum distance that the valve may be unseated and still indicate the closed position is 0.03 inch. After limit switch adjustment, the opening gap of the pallet at the switch is verified to be less than 0.03 inches. Inspection of vacuum breaker instrumentation during reactor refueling will include verification of the opening gap for switch actuation. The bypass opening for the suppression chamber to drywell vacuum breaker corresponding to a 0.03 inch disc opening is 0.009 ft^2, well within the maximum allowable leakage area of 0.25 ft^2 discussed in Subsection 6.2.1.3.9.

The torus-to-reactor building open and closed valve disk positions are indicated by lights on the main control room panel H11-P817. The drywell-to-torus vacuum breakers are provided with open and closed position indicators on panel H11-P817, and a second set of closed position indicators on panel H11-P808. The position indicator lights are controlled by sealed switches designed to comply with immersion requirement MIL-E5272. The drywell-to-torus closed indicating circuits are powered by Class 1E power supplies and are wired to meet the requirmeents of IEEE 279-1971. 12

There is no annunciation of the valve position. The position switches and circuits do not control or affect the operation of the vacuum breakers. Any single failure of the indicating circuits or switches will not prevent proper action of the vacuum breakers.

The drywell-to-torus and the torus-to-reactor building vacuum breakers are equipped with pneumatic actuators operated by pushbuttons from the main control room. The purpose of these actuators is to enable verification of the operability of the vacuum breakers by observing the response of limit switches. The operability of the vacuum breakers will be verified on a periodic basis and at the discretion of the operators.

The actuators are sized such that they have insufficient power to open the vacuum breakers if a back-flow differential pressure exists. The vacuum breakers and the testing actuators are designed to Category I criteria.

A post-design drywell cooldown transient analysis has been performed to verify that the Fermi 2 drywell vacuum relief system has an adequate margin. To develop a set of initial conditions for this analysis, the major parameters that govern the depressurization must be considered. Mass and energy balances and thermodynamic principles determine that the following conditions have a predominant effect on the depressurization transient:

a. Minimum mass of air or air/vapor mixture in the drywell

b. The maximum mass flow rate and minimum temperature of the spray water

c. The minimum energy of mass in the drywell.

The analysis consisted of computing drywell temperature/pressure transients resulting from activation of the containment spray systems following various operating conditions and the design basis LOCA. The drywell and torus pressure-temperature response was calculated on a computer program utilizing basic thermodynamic, flow, and gas law equations using 0.1 second intervals for each iteration.

It was found that the worst transient would occur from operation of both containment spray systems following a small steam leak within the drywell. A small steam leak in the drywell is assumed to pressurize the drywell to 16.7 psia, just below the isolation signal pressure set point. The operator is assumed to manually vent the drywell to keep the pressure nearly normal until procedural controls prohibit further venting.

These events result in the largest number of condensible moles of vapor in the drywell. The torus pressure is 14.7 psia, the

pressure just before the 4.5-foot vent pipe water leg breaks. It is postulated that the operator actuates both RHR system drywell spray loops simultaneously, and that the torus water is 50°F. One of the two torus-to-reactor building vacuum breakers is assumed to fail as well as two of the twelve drywell-to-suppression chamber vacuum breakers. The resulting pressure of -1.88 psid occurs 19 seconds after spray initiation (see Figure 6.2-7a).

For the post-LOCA transient, the initial conditions are developed assuming the LOCA blowdown forces all noncondensible gases into the torus, and the drywell is filled with steam. This assumption results in a torus pressure of 31.8 psia. With four feet of water in the vent pipes, the most conservative drywell pressure possible is 33.5 psia. Saturated steam conditions in the drywell are assumed giving the lowest possible internal energy.

With the torus and drywell at these conditions, it is again postulated that both RHR drywell spray loops are simultaneously activated, that the torus water is 50°F, and that one of the torus-to-reactor building and two of the drywell-to-suppression chamber vacuum breakers fail to open. The resulting maximum drywell differential pressure is -1.03 psid, occurring 38 seconds after spray initiation (see Figure 6.2-7b).

Table 6.2-4b lists in detail the initial and operating conditions for each transient. The adequacy of the drywell vacuum relief system was demonstrated in both transients, since the maximum negative pressure differential did not exceed -2.63 psid, which was the maximum allowable pressure determined in the drywell stress report for the weakest portion of the drywell cylinder.

As stated previously, the initial conditions were chosen to achieve the most severe transient. For example, the torus water temperature is conservatively chosen to be 50°F during the entire transient. Realistically, the temperature of the stagnant water in the RHR piping would be the (higher) temperature of the reactor building and would increase during the LOCA blowdown. In addition, two of the twelve drywell-to-suppression chamber vacuum breakers, as well as one of the two suppression chamber-to-reactor building vacuum breakers, are assumed to fail to open.

Many conservative assumptions are made in the calculational model. For example, the calculation assumes a 100 percent spray efficiency. Realistically, at full spray conditions, the residence time of the water droplets in the drywell is insufficient to reach their full heat sink capacity while in contact with the gas (typical cooling tower efficiencies are 60 to 70 percent). The calculation assumes heat transfer between gas and water, but neglects heat transfer from structures and equipment. The drywell would also thermally contract and reduce the free volume by approximately 500 cubic feet. The calculation assumes simultaneous operation of both drywell spray loops. This is almost impossible to accomplish, for it would require

simultaneous operation of four keylocked switches opening two valves on each loop. It is therefore concluded there is adequate margin in the drywell vacuum system design.

6.2.1.2.2 Secondary Containment System

The reactor building completely encloses the reactor and its pressure suppression primary containment.

This building provides secondary containment when the primary containment is closed and in service, and provides primary containment when the primary containment is open, as it is during refueling. The reactor building houses the refueling and reactor servicing equipment; new and spent fuel storage facilities, and reactor auxiliary and service equipment, including the reactor core isolation cooling (RCIC) system; reactor cleanup demineralizer system, standby liquid control system (SLCS), CRD system equipment, emergency core cooling system (ECCS), and electrical equipment components.

The reactor building includes the "tunnel" containing the outboard main steam isolation valves (MSIV), the main steam lines up to the turbine building, the feedwater lines, and the outboard feedwater line isolation valves. The tunnel is equipped with hinged doors which, upon pressure buildup due to a break in one of these lines, will relieve the steam pressure to the first and second floors of the turbine building. The net volume of the secondary containment is 2.8 x 10^6 cubic feet.

The reactor building is a Category I structure designed and constructed in accordance with all applicable local and state building code requirements.

Substructures and exterior walls of the building up to the refueling floor consist of poured-in-place, reinforced concrete. The building structure above the refueling floor is a steel frame covered with insulated metal siding, and is sealed against leakage. The building is designed for an external pressure of 0.295 psig and for low in-leakage and out-leakage (depending on wind conditions) during reactor operation.

6.2.1.2.2.1 Reactor Building Penetrations

Access openings for personnel and equipment are equipped with weather strip-type seals for airtightness. Penetrations for piping and ducts are designed for leakage characteristics consistent with containment requirements for the entire building. Electrical cables and instrument leads pass through ducts sealed into the building wall.

6.2.1.2.2.2 Reactor Building Ventilation Systems

The reactor building has two ventilation systems: the normal ventilation system and the SGTS. During normal power operation,

shutdown, or refueling, the normal ventilation system provides outside filtered air to all levels and building equipment rooms. This system provides one reactor building free volume change of air per hour. Air flows from the filtered supply to uncontaminated areas, to potentially contaminated areas and then to the release v t (a short stack) on the reactor building roof.

The fans for the normal ventilation system are automatically shut down in the event a high radiation level in the building exhaust ducts is detected by the radiation monitoring system (RMS), or if there is high pressure in the drywell, low RPV water level, or high static pressure in the building.

Shutting down the fans closes the dual ventilation duct isolation valves. The fans are controlled from the main control room.

During emergencies when the normal ventilation system is down, the reactor building is ventilated through the SGTS. The SGTS filters and exhausts the atmosphere of the reactor building to the roof vent when containment isolation is required.

6.2.1.2.2.3 Bypass Leakage Paths

One purpose of the secondary containment (reactor building) is to collect and filter leakage from the primary containment prior to release to the environment and thereby reduce offsite doses after a loss-of-coolant accident. This purpose is accomplished by

a. Minimizing reactor building leakage

b. Maintaining the reactor building at a negative pressure

c. Passing all exhaust from the reactor building through the SGTS after a LOCA.

A study has been made to evaluate the secondary containment system and determine all potential paths that could result in a fraction of the primary containment leakage going directly to the environment (i.e., without passing through the SGTS). The study encompasses three areas:

a. Lines that are connected to the primary containment and pass through the secondary containment

b. Electrical penetrations

c. Reactor building leakage.

Primary Containment Lines

Lines that are connected to the primary containment and pass through the secondary containment are potential paths for leakage of radioactivity directly from the primary containment to the

environment, bypassing the SGTS. The detailed review of primary containment piping for potential leakage paths is given in Appendix 6A and Table 6.2-2.

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Leakage through the primary containment exhaust lines is collected by the SGTS and is not discharged through the exhaust fans. The
12 primary containment inerting lines are tied to the SGTS at all times except during the inerting purge. The isolation valve leakage is collected by the SGTS. See Figure 9.3-12.

Category I design requirements are met (1) on the main steam piping from the reactor, up to and including the turbine stop valves, bypass valves, and other turbine-associated valves (e.g., live steam to reheater, gland seal, and flange warming valves); and (2) on all branch piping, up to and including the first valve that is either normally closed or capable of automatic closure during all modes of normal NSSS operation. Leakage through this system is discussed in detail in Subsection 10.3.3.

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The drywell drain lines are also isolated in the event of a LOCA. Leakage through the isolation valves is limited to 10 cubic centimeters liquid per hour per inch of valve diameter. Consequently, the maximum total bypass leakage from these lines is 60 cm^3/h. This leakage is to holding tanks in the radwaste building.

Electrical Penetrations

Electrical cables exit from the primary containment via penetrations sealed at both internal and external ends; the external end

is within the secondary containment. The cables leaving these penetrations run in cable trays. Thus there are no electrical wiring conduits or ducts that go directly from the primary containment to the environment bypassing the secondary containment.

Reactor Building Leakage

The reactor building under both normal and emergency conditions is maintained at a negative pressure so that leakage is inward. However, due to the kinetics of gas at high velocities, the pressure on the leeward side of the building will be negative at high wind speeds. Consequently, above a threshold wind speed, air could be drawn from the reactor building, bypassing the SGTS.

An exfiltration/infiltration analysis has been made on the building to determine inward and outward leakage rates as a function of wind speed. The analysis was based on the following.

a. The SGTS maintains the building at 1/4 in. H_2O negative pressure.

b. Leakage to the environment occurs only through the metal siding and only when the pressure differential across the siding is outward.

c. The rate of leakage is 0.015 $ft^3/min/ft^2$ at 1/4 in. H_2O and varies as the square root of pressure differential. The leakage rate is the same for positive and negative differentials.

d. The wind force acts on two sides of the building; the other two sides are at a negative pressure.

e. The positive and negative pressures due to wind are based on the equation

$$P = 0.002558S\ (GV)^2$$

where

P = wind pressure (lb/ft^2)

S = shape factor = 0.9 windward side
= 0.5 leeward side

G = gust factor = 1.1

V = wind velocity (mi/h).

The study shows the threshold wind velocity for any leakage outward from the building is 30 mi/h. The study also shows that the net leakage (inward) through the siding is not a strong function of wind velocity; consequently, the operating parameters of the SGTS are independent of wind velocity.

Since there is siding only above the refueling floor, this leakage path is not directly from the primary containment to the environment, but rather from the secondary containment--the reactor building. The estimate of the fraction of primary leakage bypassing the SGTS will be conservative if this fraction is assumed to be equal to the fraction of building leakage to total discharge from the reactor building. This statement can be expressed by the following equation:

$$B = \frac{S}{S + G}$$

where

B = fraction of primary leakage bypassing SGTS

S = outward leakage rate of siding (function of wind speed) (scfm)

G = discharge rate of SGTS (scfm).

The results of this study are summarized in the following table:

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Wind Velocity (mi/h)	Fraction of Time per Year[a]	Reactor Building Leakage: Total Outward from Siding (scfm)	Reactor Building Leakage: Fraction of Primary Leakage Bypassing SGTS
0		0	0
	0.65		
10		0	0
	0.34		
20		0	0
	0.01		
30		52	0.017
	0.001		
40		246	0.076
	--		
50		370	0.110

[a] Winds of 15-minute duration as measured from the 10-meter level on the 60-meter tower during the 12-month period from June 1, 1974, to May 31, 1975.

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6.2.1.3 Design Evaluation

6.2.1.3.1 Introduction

In the design of the primary containment vessel, certain extreme conditions were hypothesized; the design then proceeded so that maximum stress levels under these conditions did not exceed the maximum allowable values specified in the appropriate code.

Two key parameters of stress are vessel temperature and pressure. The containment vessel for Fermi 2 was designed under ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels (1968), including Summer 1969 Addenda. This code specifies that the internal pressure used for design conditions shall not be less then 90 percent of the maximum containment internal pressure, and that the design temperature shall not be less than the maximum containment temperature at the coincident maximum containment pressure.

In the following sections, various extreme conditions are hypothesized and an analysis is made of the resulting temperature and pressure transients inside the containment vessel.

The maximum drywell pressure occurs during the reactor blowdown phase of a LOCA. It is dependent upon the rate at which primary system energy and fluid enter the drywell. The largest pipe in the primary coolant system is the 28-inch diameter main recirculation line. The instantaneous guillotine rupture of this pipe is the DBA for the containment design pressure. The same pressure is conservatively used for suppression chamber design.

The most severe drywell temperature condition would occur as a result of a small primary system rupture above the reactor water level that results in the blowdown of reactor steam to the drywell. Because of the nature of the blowdown process, this would produce high temperature steam in the drywell.

In order to demonstrate that breaks smaller than the rupture of the largest primary system pipe can be accommodated safely without any of the containment design parameters being exceeded, the blowdown phase of an intermediate size break is described.

All of the analyses have assumed that the primary system and containment system are operating at the maximum normal conditions. References are provided that describe relevant experimental verification of the analytical models used to evaluate the containment response to a LOCA.

6.2.1.3.2 Recirculation Line Break - Short-Term Response

Descriptions of the models used for the short-term response calculations are given in Reference (2).

The instantaneous guillotine rupture of a main recirculation line results in the maximum flowrate of primary system fluid and energy into the drywell. This in turn results in the maximum containment differential pressure. Figure 6.2-8 is a diagram showing the location of a recirculation line break.

Immediately following the rupture, the flow out both sides of the break will be limited to the maximum allowed by critical flow considerations. Figure 6.2-8 shows a schematic view of the flow paths to the break. In the side adjacent to the suction nozzle, the flow will correspond to critical flow in the 3.667 ft^2 pipe cross-section. In the side adjacent to the injection nozzle, the flow will correspond to critical flow at the ten jet pump nozzles associated with the broken loop, providing an effective break area of 0.538 ft^2. In addition, there is a four-inch cleanup line crosstie that will add 0.08 ft^2 to the critical flow area, yielding a total of 4.825 ft^2.

This accident is analyzed using the following assumptions:

a. The conditions in the containment and primary system at the time of the accident are the maximum normal conditions described in Subsection 15.1.13

b. A complete loss of normal power occurs simultaneously with the pipe break. This additional condition results in the longest delay time for the ECCS to become operational

c. The recirculation line is considered to be severed instantly. This results in the most rapid coolant loss and depressurization, with coolant being discharged from both ends of the break

d. The reactor is assumed to shut down at the time of accident initiation because of void formation in the core region

e. The sensible heat released in cooling the fuel to 545°F and the core decay heat are included in the RPV depressurization calculation. The rate of energy release is calculated using a conservatively high heat transfer coefficient throughout the depressurization. Because of this assumed high energy release rate, the RPV is maintained at nearly rated pressure for approximately 15 seconds. The high RPV pressure increases the calculated blowdown flowrates; this is conservative for containment analysis purposes. With the RPV fluid temperature remaining near 545°F, however, the release of sensible energy stored below 545°F is negligible during the first 15 seconds. The later release of of this sensible energy does not affect the peak drywell pressure. The small effect of this energy on the end-of-transient suppression pool temperature is included in the calculations

f. The MSIV's are assumed to start closing at 0.5 second after the accident. They are assumed to be fully closed in the shortest possible time of three seconds following closure initiation. Actually, if the closure of the MSIV's is expected to be the result of low water level, these valves may not receive a signal to close for more than four seconds, and the closing time could be as long as ten seconds. By assuming rapid closure of these valves, the RPV is maintained at a high pressure, which maximizes the discharge of high energy steam and water into the primary containment

g. The feedwater flow is assumed to stop instantaneously at time zero. This conservatism is used because the relatively cold feedwater flow, if considered to continue, tends to depressurize the RPV, thereby reducing the discharge of steam and water into the primary containment. In addition, the rapid closure of the MSIV's cuts off motive power to the steam-driven feedwater pumps

h. The vessel depressurization flowrates are calculated using Moody's critical flow model (Reference 3) assuming "liquid only" outflow, because this assumption maximizes the energy release. "Liquid only" outflow means that all vapor formed in the RPV by bulk flashing rises to the surface rather than being entrained in the existing flow. Some of the vapor would be entrained and would significantly reduce the RPV discharge flowrates. Moody's critical flow model,

which assumes annular, isentropic flow, thermodynamic phase equilibrium, and maximized slip ratio, accurately predicts vessel outflows through small diameter orifices. These are conservative assumptions. Actual rates through larger flow areas have been measured to be less than the model indicates

i. The pressure response of the primary containment is calculated assuming the following:

   1. Thermodynamic equilibrium exists in the drywell and suppression chamber. Because complete mixing is nearly achieved, the error introduced by assuming complete mixing is negligible and in the conservative direction

   2. The constituents of the fluid flowing through the drywell-to-suppression chamber vents are based on a homogeneous mixture of the fluid in the drywell. The consequences of this assumption are complete liquid carryover into the drywell vents and a substantial increase in the calculated irreversible pressure losses in the vents

   3. The flow in the drywell-to-suppression chamber vents is compressible except for the liquid phase

   4. No heat loss from the gases inside the primary containment is assumed. This adds extra conservatism to the analysis; that is, the analysis will tend to predict higher containment pressures than would actually result.

Figures 6.2-9 and 6.2-10 show the blowdown flowrates from the primary system to the containment. Table 6.2-5 shows the primary system energy distribution at time of break.

The calculated primary containment pressure and temperature responses to this LOCA are shown in Figures 6.2-11 and 6.2-12. The calculated peak drywell pressure is 56.5 psig, which is 8.9 percent below the maximum allowable pressure of 62 psig for this design. After the discharge of primary coolant from the RPV into the drywell, the temperature of the suppression chamber water approaches 135°F and the suppression chamber pressure stabilizes at approximately 25 psig. The drywell pressure stabilizes at a slightly higher pressure, the difference being equal to the downcomer submergence. During the RPV depressurization phase, most of the noncondensable gases in the drywell initially are forced into the suppression chamber. However, the noncondensables will redistribute between the drywell and suppression chamber via the vacuum breaker system as the drywell pressure is decreased by steam condensation.

The LPCI and/or core spray system removes decay heat and stored heat from the core, thereby controlling core heatup and limiting metal-water reaction to less than 0.1 percent. At approximately 100 seconds, the RPV is flooded to the height of the jet pump nozzles and the excess flow discharges through the recirculation line break into the drywell. This flow of water transports the core decay heat out of the RPV, through the broken recirculation line, in the form of hot water that flows into the suppression chamber via the drywell to suppression chamber vent pipes. Steam flow is negligible. This flow, in addition to heat losses to the drywell walls, offers considerable cooling to the drywell atmosphere and causes a depressurization of the containment as the steam in the drywell is condensed.

The LPCI/RHR pumps that are used to flood the core are also used as the containment spray pumps. Prior to activation of the containment cooling mode (arbitrarily assumed at 600 seconds after the accident) all of the LPCI pump flow will be used only to flood the core. After 600 seconds, the RHR pump flow can be diverted from the RPV to the containment spray. This is a manual operation. Actually, the containment spray need not be activated at all to keep the containment pressure below the containment peak allowable pressure. In either case (LPCI/RHR pumps flow to the RPV or to the containment spray), the RHR heat exchangers must be activated within some reasonable time after the accident, e.g., one hour, to maintain the peak temperature of the suppression chamber below its allowable limit.

The Fermi 2 PSAR filed May 1969 documented the anticipated post-LOCA containment drywell pressure to be 47 psig using GE models then in effect. The maximum allowable internal pressure was set at 62 psig, and in accordance with Section III of the ASME Boiler and Pressure Vessel Code, the drywell design pressure was set at 90 percent of maximum allowable or 56 psig.

The post-LOCA containment drywell pressure is now calculated to be 56.5 psig or 5.5 psig below the Code allowable pressure. Sufficient pressure margin exists even though the calculated peak pressure has increased 9.5 psig from the PSAR to the FSAR stage of the project. The entire change noted is due to use of a more conservative model as described in GE report NEDO-10320. The principal difference in the two models used is a 300 percent increase in the fluid density used for vent flow. The PSAR model was shown to be conservative relative to test data, and the new vent flow model is even more conservative. No change in containment design pressure is necessary even though the above increase in calculated peak pressure has occurred.

6.2.1.3.3 Recirculation Line Break - Long-Term Response

To assess the primary containment long-term response after the accident, an analysis is made of the effects of various containment spray and containment cooling combinations. For all cases, one of the core spray loops is assumed to be in operation. The long-term pressure and temperature response of the primary containment was analyzed for the following RHR containment cooling mode conditions:

a. Case a - Operation of both RHR cooling loops - four RHR pumps, four service water pumps, and two RHR heat exchangers - with containment spray

b. Case b - Operation of one RHR cooling loop with two RHR pumps, two service water pumps, and one RHR heat exchanger - with containment spray

c. Case c - Operation of one RHR cooling loop with one RHR pump, two service water pumps, and one RHR heat exchanger - with containment spray

d. Case d - Operation of one RHR cooling loop with one RHR pump, two service water pumps, and one RHR heat exchanger - no containment spray.

The initial pressure response of the containment (the first 30 seconds after break) is the same for each case. During the long-term containment response (after depressurization of the RPV is complete) the suppression pool is assumed to be the only heat sink in the containment system. The effects of decay energy and stored energy on the suppression pool temperature are considered.

6.2.1.3.3.1 Case a

This case assumes that both RHR loops are operating in the containment cooling mode. This includes two RHR heat exchangers, four RHR main system pumps, and four service water pumps. The RHR pumps draw suction from the suppression pool and pump water through the heat exchangers and into the drywell as containment spray. This forms a closed cooling loop with the suppression pool. This suppression pool cooling condition is arbitrarily assumed to start at 600 seconds after the accident. Prior to this time the pumps are used to flood the core (LPCI mode).

The long-term containment pressure response to this set of conditions is shown as curve (a) in Figure 6.2-13. The corresponding drywell and suppression pool temperature responses are shown as curve (a) in Figures 6.2-14 and 6.2-15. After the initial blowdown and subsequent depressurization due to core spray and LPCI core flooding, energy addition due to core decay heat results in a gradual pressure and temperature rise in the containment. When the energy removal rate of the RHR system exceeds the energy addition rate from the decay heat, the

containment pressure and temperature gradually decrease to their pre-accident values. Table 6.2-6 summarizes the cooling equipment operation, the peak containment pressure following the initial blowdown peak, and the peak suppression pool temperature.

6.2.1.3.3.2 Case b

This case assumes that only one RHR loop is operating in the containment cooling mode. This includes one RHR heat exchanger, two RHR main system pumps, and two service water pumps. As in the previous case, the RHR containment cooling mode is assumed to be activated at 600 seconds after the accident. The long-term containment pressure response to this set of conditions is shown as curve (b) in Figure 6.2-13. The corresponding drywell and suppression pool temperature responses are shown as curve (b) in Figures 6.2-14 and 6.2-15. A summary of this case is given in Table 6.2-6.

6.2.1.3.3.3 Case c

This case assumes that one RHR loop is operating in the containment cooling mode at partial pumping capacity. This includes one RHR heat exchanger, one RHR main system pump, and two service water pumps. This reduction in RHR flow results in a decrease in the heat removal capacity of the heat exchanger, which in turn results in slightly higher containment temperature and pressure. It is assumed that this cooling condition is established at 600 seconds after the accident. The containment response to this set of conditions is shown as curve (c) in Figure 6.2-13. The corresponding drywell and suppression pool temperature responses are shown as curve (c) in Figure 6.2-14 and 6.2-15. A summary of this case is given in Table 6.2-6.

6.2.1.3.3.4 Case d

This case is exactly the same as the preceding one except that the drywell spray is not operated. During this mode of operation, RHR main system pumps draw suction from the suppression pool and discharge flow through the RHR heat exchangers where it is cooled and then injected back into the suppression pool. Core cooling is provided by the core spray system.

The containment pressure response to this set of conditions is shown as curve (d) in Figure 6.2-13. The corresponding drywell and suppression pool temperature responses are shown as curve (d) in Figures 6.2-14 and 6.2-15. A summary of this case is given in Table 6.2-6.

When comparing the "spray" case with the "no spray" case, it is seen that the suppression pool temperature response is the same. This is because the same amount of energy is removed from the pool whether the exit flow from the RHR heat exchanger is returned to the pool or injected into the drywell as spray.

However, the peak containment pressure is higher for the "no spray" case. This, however, is of no consequence because the pressure is still less than the containment maximum allowable pressure of 62 psig.

6.2.1.3.4 Intermediate Breaks

The failure of a recirculation line results in the most severe pressure loading on the drywell structure. However, as part of the containment performance evaluation, the consequences of intermediate breaks are also analyzed. This classification covers those breaks for which operation of the ECCS will occur during the blowdown and which result in reactor depressurization. These breaks can involve either reactor steam or liquid blowdown. This section describes the consequences to the containments of a 0.1 ft^2 break below the RPV water level. This break area was chosen as being representative of the intermediate break area range. Figures 6.2-16 and 6.2-17 show the drywell and pressure and temperature suppression chamber response; the response of the ECCS is discussed in Subsection 6.3.3.

Following the 0.1 ft^2 break, the drywell pressure increases at 0.5 psi per second. This drywell pressure transient is sufficiently slow so that the dynamic effect of the water in the vents is negligible and the vents will clear when the drywell-to-suppression chamber differential pressure is equal to the vent submergence pressure. For this containment design, the distance between the pool surface and the bottom of the vents is 4.0 feet. Thus, the water level in the vent will reach this point when the drywell-to-suppression chamber pressure differential reaches 1.7 psi, i.e., approximately three seconds after the 0.1 ft^2 break occurs. At this time, air, steam, and water will start to flow from the drywell to the suppression pool; the steam will be condensed and the air will enter the suppression chamber free space. After three seconds there will be a constant pressure differential of 1.7 psi between the drywell and suppression chamber. The continual purging of drywell air to the suppression chamber will result in a gradual pressurization of the latter. By approximately 300 seconds, all the drywell air will have been swept over to the suppression chamber and the pressure increase terminated. After this time, the drywell and wetwell pressures will remain relatively constant and all the steam being released to the drywell will be condensing in the pool. Some continuing containment pressurization will occur because of the continued pool heatup. The ECCS will be initiated by the 0.1 ft^2 break via high drywell pressure and will provide emergency cooling of the core. The operation of these systems is such that the reactor will be depressurized in approximately 600 seconds.

This will terminate the blowdown phases of the transient. The drywell will be at approximately 25 psig and the suppression chamber at approximately 23 psig.

In addition, the suppression pool temperature will be the same as from the recirculating line break because essentially the same amount of primary system energy would be released during the blowdown. After reactor depressurization, the flow through the break will condense the drywell steam and will eventually cause the drywell and suppression chamber pressures to equalize in the same manner as following a recirculation line rupture.

The subsequent long-term suppression pool and containment heatup transient that follows is essentially the same as for the recirculation break without containment spray.

From this description, it can be concluded that the consequences of an intermediate break are less severe than a recirculation line rupture over short time periods and essentially the same over a long time period.

6.2.1.3.5 Small Breaks

This subsection discusses the containment transient associated with small primary system blowdowns. The sizes of primary system blowdowns in this category are those blowdowns that will not result in reactor depressurization due either to loss of reactor fluid or automatic operation of the ECCS equipment. The underlying assumption is that, following the manifestation of a break of this size, the reactor operators will initiate an orderly shutdown and depressurization of the plant.

The thermodynamic process associated with the blowdown of primary system fluid is one of constant enthalpy. If the primary system break is below the water level, the blowdown flow will consist of reactor water. Upon depressurizing from reactor pressure to the drywell pressure, approximately one third of this water will flash to steam and two thirds will remain as liquid. Both phases will be at saturated conditions corresponding to the drywell pressure. Thus, if the drywell is at atmospheric pressure, the steam and liquid associated with a liquid blowdown would be at 212°F. Similarly, if the containment is assumed to be at its maximum allowable pressure, the reactor liquid would blow down to approximately 309°F steam and water.

If the primary system rupture is located so that the blowdown flow consists of reactor steam, the resultant steam temperature in the containment is significantly higher than the temperature associated with liquid blowdown. This is because a constant enthalpy decompression of high pressure, saturated steam will result in a superheat condition. For example, decompression of 1000 psia steam to atmospheric pressure will result in 298°F superheated steam (86°F of superheat).

The conclusion is that a small reactor steam leak will impose the most severe temperature conditions on the drywell structures and the safety equipment in the drywell. The superheat temperature

for large steam-only blowdowns would be the same as for small breaks, but the duration of the high temperature condition would be less. This is because the larger breaks will depressurize the reactor more rapidly than the orderly reactor shutdown that is assumed to terminate the small break.

For drywell design evaluation, the following sequence of events was assumed to occur. With the reactor and containment operating at the maximum normal conditions defined in Table 6.2-1, a small break occurs that allows blowdown of reactor steam to the drywell.

The resulting pressure increase in the drywell will lead to a high drywell pressure signal that will scram the reactor and activate the containment isolation system. The drywell pressure will continue to increase at a rate dependent upon the size of the assumed steam leak. This pressure increase will depress the water level in the vents until the level reaches the bottom of the vents. At this time, air and steam will start to enter the suppression pool. The steam will be condensed and the air will pass to the suppression chamber free space. The latter will result in a gradual pressurization of the containment at a rate dependent upon the air carryover rate. Eventually, the entrainment of the drywell air in the steam flow through the vents will result in all the drywell air being carried over to the suppression chamber. At this time, pressurization of the containment will cease and the system will reach an equilibrium condition with the drywell pressure at 25 psig and the suppression chamber at approximately 23 psig. The drywell will be full of superheated steam. Continued blowdown of reactor steam will be condensed in the pool.

The reactor operators will be alerted to the incident by the high drywell pressure signal and the reactor scram. For the purposes of evaluating the duration of the superheat condition in the drywell, it is assumed that their response is to shut the reactor down in an orderly manner using the RHR condensing heat exchangers, or main condenser, and limiting the reactor cooldown rate to 100°F per hour. This will result in the reactor primary system being depressurized within six hours. At this time, the blowdown flow to the drywell will cease and the superheat condition will be terminated. If the plant operators elect to cool down and depressurize the reactor primary system more rapidly than at 100°F per hour, then the drywell superheat condition will be shorter.

The temperature resulting from the blowdown is determined by finding the combination of primary system pressure and containment pressure that produces the maximum superheat temperature. These are 450 psia, 35 psig, and 340°F, respectively. This temperature is assumed to exist for the initial three hours of the blowdown.

6.2.1.3.6 Analytical Models - Post-Blowdown

The analytical models, assumptions, and methods used by GE to evaluate the containment response during the reactor blowdown phase of a LOCA are described in References (4) and (5).

Once the RPV blowdown phase of the LOCA is over, a fairly simple model of the drywell and suppression chamber may be used.

The key assumptions employed in the post-blowdown model are as follows:

a. Drywell and suppression chamber atmosphere are both saturated (100 percent relative humidity)

b. The drywell atmosphere temperature is equal to the temperature of the liquid flowing in from the RPV or to the spray temperature if the latter is activated

c. Suppression chamber atmosphere temperature is equal to the suppression pool temperature or to the spray temperature if the latter is activated

d. No credit is taken for heat losses from the primary containment.

Because the ECCS flow path during the long-term, post-blowdown containment transient is a closed loop, the suppression pool mass will be constant. Schematically, the loop is shown in Figure 6.2-18, together with the identification of some of the terms.

There is no storage other than in the suppression pool (the RPV is reflooded during the blowdown phase of the accident); thus,

$$\dot{m}_{D_o} = \dot{m}_{s_o} = \dot{m}_{eccs} \qquad (6.2.1)$$

The rate of change or energy in the suppression pool, E_p, is given by

$$\frac{d}{dt}(E_p) = \frac{d}{dt}(M_{w_s} \cdot h_s) \qquad (6.2.2)$$

$$= h_s \cdot \frac{d}{dt}(M_{w_s}) + M_{w_s}\frac{d}{dt}(h_s).$$

Since

$$\frac{d}{dt}(M_{w_s}) = 0$$

(because there is no storage: and if it is assumed that for water at the conditions that will exist in the containment)

$$\frac{d}{dt}(h_s) = C_p \cdot \frac{d}{dt}(T_s), \qquad (6.2.3)$$

where

C_p = specific heat of pool water, $\frac{Btu}{lb/°F}$

T_s = pool temperature, °F

then the pool energy balance yields

$$M_{w_s} C_p \cdot \frac{d}{dt}(T_s) = \dot{m}_{D_o} h_D - \dot{m}_{s_o} h_s \qquad (6.2.4)$$

Assuming C_p = 1.0 Btu/lb-°F for water, this equation can be rearranged to yield

$$\frac{d}{dt}(T_s) = \frac{\dot{m}_{D_o} h_D - \dot{m}_{s_o} h_s}{M_{w_s}} \qquad (6.2.5)$$

An energy balance on the RHR heat exchanger yields

$$h_c = h_s - \frac{\dot{q}_{H_x}}{\dot{m}_{s_o}} \qquad (6.2.6)$$

where,

h_c = enthalpy of ECCS flow entering the reactor, Btu/lb

Similarly, an energy balance on the RPV will yield

$$h_d = h_c + \frac{\dot{q}_D + \dot{q}_e}{\dot{m}_{eccs}} \qquad (6.2.7)$$

Pump heat rate is handled as a constant in the computer input for the decay heat term.

Combining Equations 6.2.1, 6.2.2, 6.2.3, and 6.2.4 gives

$$\frac{d}{dt}(T_s) = \frac{\dot{q}_D + \dot{q}_e - \dot{q}_{H_x}}{M_{w_s}} \quad (6.2.8)$$

This differential equation is integrated by finite difference techniques to yield the suppression pool temperature transient. The drywell and wetwell atmospheric temperatures can then be calculated.

For the case in which no containment spray is operating, the wetwell temperature, T_w, at any time will be equal to the current temperature of the pool, T_s, and the drywell temperature, T_d, will be equal to the temperature of the fluid leaving RPV:

$$T_D = T_s + \frac{\dot{q}_D + \dot{q}_e - \dot{q}_{H_x}}{\dot{m}_{eccs}} \quad (6.2.9)$$

and

$$T_w = T_s$$

For the case in which the containment spray is assumed to be operating, both the drywell and suppression chamber atmosphere will be at the spray temperature, T_{sp},

where

$$T_{sp} = T_s - \frac{\dot{q}_{H_x}}{\dot{m}_{eccs}} \quad (6.2.10)$$

and

$$T_D = T_w = Tsp$$

With the suppression chamber and drywell atmosphere temperatures known and with assumption (a) (drywell and suppression chamber saturated), it is now possible to solve for the total pressures:

$$P_D = P_{a_D} + P_{V_D} \quad (6.2.11)$$

$$P_s = P_{a_s} + P_{V_s} \tag{6.2.12}$$

where

P_D = drywell total pressure, lb/in.^{2}

P_{a_D} = partial pressure of air in drywell, lb/in.^{2}

P_{V_D} = partial pressure of water vapor in drywell, lb/in.^{2}

P_s = suppression chamber total pressure, lb/in.^{2}

P_{a_s} = partial pressure of air in the suppression chamber, lb/in.^{2}

P_{V_s} = partial pressure of water vapor in the suppression chamber, lb/in.^{2}

and, from the gas laws,

$$P_{a_D} = \frac{M_{a_D} RT_D}{V_D\ 144} \tag{6.2.13}$$

$$P_{a_s} = \frac{M_{a_s} RT_W}{V_s\ 144} \tag{6.2.14}$$

where

M_{a_D} = mass of air in the drywell, lb

M_{a_s} = mass of air in the suppression chamber, lb

R = gas constant for air, ft-lb/lb

V_D = drywell free space volume, feet

V_S = suppression chamber free volume, ft^3

With known values of T_D and T_W, Equations 6.2.11, 6.2.12, 6.2.13, and 6.2.14 can now be solved if M_{a_D} and M_{a_s} are known. The following procedure is used to calculate the values.

The transient analysis is based on a finite time step integration of the suppression pool temperature. When this integration has been performed and the value of T_s at the end of a time step has been calculated, the following check is made.

Using values of M_{a_D} and M_{a_s} from the end of the previous time step and the updated values of T_D and T_s, check to see if $P_s \geq P_D$ using Equations 6.2.11, 6.2.12, 6.2.13, and 6.2.14.

If $P_s \geq P_D$, then the two values are made equal. This is done because in the actual containment, P_s cannot be greater than P_D. The vacuum breakers between the drywell and suppression chamber are provided for this purpose.

Hence, with

$$P_D = P_s$$

and knowing that

$$M_{a_D} + M_{a_s} = \text{the known total initial mass of air in the suppression chamber and drywell prior to the accident} \qquad (6.2.15)$$

Equations 6.2.11, 6.2.12, 6.2.13, and 6.2.14 can be solved for M_{a_s}, M_{a_D}, and P_s/P_D. It should be noted that assuming the total mass of air to remain constant conservatively ignores any containment leakage that might occur during the transient.

If, as a result of the end-of-time-step pressure check,

$$P_s \leq P_D \leq P_s + \frac{H}{V_w}, \qquad (6.2.16)$$

where

H = submergence of vent, feet

V_w = specific volume of fluid in vent, ft^3/lb

then the pressure in the drywell is higher than the pressure in the suppression chamber but not sufficiently so to depress the water to the bottom of the vents and thus permit air to flow from the drywell to the suppression chamber. Under these circumstances, no air transfer is assumed to have occurred during the time step, and Equations 6.2.11, 6.2.12, 6.2.13, and 6.2.14 are solved using the updated temperatures and the M_{a_s} and M_{a_D} values from the previous time step.

If the end-of-time-step pressure check shows

$$P_D \geq P_s + \frac{H}{V_w} \tag{6.2.17}$$

then

$$P_D = P_s + \frac{H}{V_w} \tag{6.2.18}$$

This assumes that the drywell pressure can never exceed the suppression chamber pressure by more than the hydrostatic head associated with the submergency of the vents. To maintain this condition, some transfer of drywell air to the suppression chamber will be required. The amount of transfer is calculated by using Equation 6.2.15 and combining Equations 6.2.11, 6.2.12, 6.2.13, 6.2.14, and 6.2.15 to give

$$P_{V_D} + \frac{M_{a_D} R T_D}{144 V_D} = P_{V_s} + \frac{M_{a_s} R T_W}{144 V_s} + \frac{H}{V_w}$$

and solving for the unknown values of air masses. The total pressures can then be evaluated.

6.2.1.3.7 <u>Accident Chronology</u>

The drywell and containment responses to the DBA have been analyzed based on two cases: all systems of the ECCS in operation, and minimum systems of the ECCS available after losing offsite power. Results of the analysis are plotted and shown in Figures 6.2-11 through 6.2-15.

Corresponding to those figures mentioned, the accident chronology for the design basis LOCA is tabulated in Table 6.2-7.

6.2.1.3.8 Transient Energy Release Rates

In order to establish the energy distribution as a function of time (short-term, long-term), the following energy rates are required

a. Blowdown energy rates
b. Decay heat rate, fuel relaxation energy
c. Sensible heat rate
d. Pump heat rate value
e. Heat removal rate from suppression pool.

In general, a very conservative analytical approach is taken in that all possible sources of energy are accounted for, and the suppression pool is assumed to be the only available heat sink. No credit is taken for either the heat that will be stored in the primary containment structures, or the heat that will be transmitted through the containment and dissipated to the environment.

Section IV of Table 6.2-1 shows a tabulation of the blowdown flowrates that are assumed to occur as a result of the instantaneous rupture of a main steam line with all subsystems of the ECCS in operation. (Note that the first 30 seconds is independent of ECCS.)

The following is a tabulation of the core decay heat values that have been used in the evaluation of the containment response to a LOCA.

Time (Seconds)	Normalized Core Heat*	Time (Seconds)	Normalized Core Heat*
0	1.0	30.0	.0471
.9	.9987	10^2	.0381
2.1	.7662	10^3	.0223
5.0	.5005	10^4	.0119
6.93	.3850	10^5	.0067
9.03	.2955	10^6	.0027
15.93	.1491		

*Includes fuel relaxation energy

These data are based on GE decay heat curves. The GE decay heat curves are conservative and were explained in detail in a letter from GE (S. Levy) to the AEC (Dr. Peter Morris), dated April 16, 1971. On some recent dockets the AEC has requested the applicant to calculate on the basis of five percent zirconium-water reaction. Such a calculation adds less than 1°F to the suppression pool temperature previously calculated for one percent zirconium-water reaction.

Following a LOCA, the sensible energy stored in the reactor primary system metal will be transferred to the recirculating ECCS water and will thus contribute to the suppression pool and containment heatup. Figure 6.2-19 shows the temperature response of the RPV. Heat addition or removal credits are not included due to negligible effects of core/containment cooling equipment or containment wall or piping heat losses. It should be pointed out that such consideration would not contribute more than about 2°F or 3°F to or from the system. These secondary effect considerations certainly do not negate or compete with the conservatisms assumed in the other heat loads.

The pump heat rate value that has been used in the evaluation of the containment response to a LOCA for the maximum number of pumps in use is 6222 Btu/s.

Figure 6.2-20 shows the rate at which the RHR system heat exchanger will remove heat from the suppression pool following a LOCA. The heat removal rate is shown for two cases. The first assumes that all the ECCS equipment is available following the LOCA, including both RHR heat exchangers and the necessary service water pumps. The second case is for the degraded cooling condition that would limit the heat removal capacity to one heat exchanger. For both cases, it was conservatively assumed that at the time of the accident the residual heat removal service water (RHRSW) temperature was 90°F.

6.2.1.3.9 Steam Bypass

The Fermi 2 containment has been examined to determine what leakage between the drywell and suppression chamber can be tolerated as a function of primary system break area; i.e., what leakage will result in a peak pressure equal to the maximum allowable pressure for the system. For this calculation, the following assumptions were made

a. Flow through the postulated leakage path is pure steam. For a given leakage path, postulating that the leakage flow consisted of a mixture of liquid and vapor would increase the total leakage mass flowrate but would decrease the steam flowrate. Since it is the steam entering the suppression chamber free space that is resulting in the containment pressurization, this is a conservative assumption

b. There is no condensation of the leakage flow on either the suppression pool surface or the torus and vent system structures. Since any condensation results in less steam being in the suppression chamber free space, this is a conservative assumption. In practice, there would be condensation, especially for the larger primary system breaks when there will be vigorous agitation at the pool surface during blowdown.

Leakage capacity is expressed in terms of A, the area of the leakage flow path, and K, the geometric loss coefficient. These terms are interrelated such that the allowable leakage capacity for a system is expressed in units of $A/\sqrt{K}$.

The calculation shows that the limiting leakage capacity occurs for a primary system break area of 0.4 ft^2. For this break area, the allowable leakage capacity is 0.147. Typically, the geometric loss factor, K, would be three or greater; thus, the maximum allowable leakage area would be about 0.25 ft^2. This corresponds to a seven-inch line.

Primary system breaks greater than about 0.4 ft^2 will result in rapid system depressurization, and, for the given primary allowable leakage area, would result in the containment pressure being less than the maximum allowable pressure at the end of the reactor blowdown period.

Primary system breaks less than about 0.4 ft^2 will not result in rapid primary system depressurization and some operator action is required to terminate the pressure rise in the containment. The operators have several options available to them. If the source of the leakage is undefined, they would probably depressurize the primary system via either the main condenser or relief valves, or they could activate the suppression chamber or drywell sprays.

6.2.1.3.10 Small Break Temperature Consideration

The Fermi 2 containment vessel was designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels (1968), including the 1969 Summer Addenda. The primary containment design parameters, as shown in Part I of Table 6.2-1, were chosen on the basis of conditions discussed in the Fermi 2 PSAR. The design basis conditions have since changed, as discussed in Subsection 6.2.1.3.2. However, as stated there, no change in design pressure was necessary.

Figure 6.2-12 shows an initial transient temperature peak in the drywell atmosphere of about 304°F. The temperature reaches the maximum value ten to 15 seconds after the LOCA begins and persists above 281°F for 40 to 45 seconds.

A small steam leak inside the primary containment, followed by an orderly shutdown and RPV depressurization, presents a different drywell atmosphere temperature transient. This situation, which is discussed in Subsection 6.2.1.3.5, was also discussed in the Fermi 2 PSAR Amendment 15 in response to question 2.8.3.e. The drywell temperature is calculated to be 340°F for three hours, and 320°F for six hours. During this period the calculated maximum drywell pressure is 35 psig, and during the following 24-hour period the temperature is 250°F with a pressure maximum of 25 psig. The containment vendor has analyzed the containment capability and found it to be adequate for these conditions.

6.2.1.3.11 Line Breaks in Sacrificial Shield Annulus

6.2.1.3.11.1 Description of System Configuration

The sacrificial shield is a cylindrical shell with a 25'-7" ID and a 29'-1" OD, 50 feet high and 1'-9" thick. It has steel liners on its exterior and interior surfaces, and is meridionally stiffened by 12 vertical steel columns. The steel liner plates are welded to the columns, and the annular space between these plates is filled with concrete. The wall is rigidly attached to the reactor support at the bottom and attached to the drywell and RPV at the top by means of stiff leg supports and snubbers respectively. The RPV sits inside the sacrificial shield with annular clearance of approximately 18 inches. Of this 18 inches approximately three inches is occupied by insulation and three inches by a ventilation space between the shield and the insulation. This leaves a 12-inch annular space between the insulation and the RPV. A detailed description of the sacrificial shield is
1 given in Subsections 3.8.3.1.1 and 3.8.3.3.1.

Openings are provided in the shield for the passage of lines from the RPV to the drywell. Those openings which lie within an area nine feet above and 16 feet below the centerline of the core are required to be shielded, and are equipped with shielding doors; these doors are locked closed and will not open during a pipe break within the annulus. The openings above and below this band have no shielding requirements; these openings are covered with a light weight rupture diaphragm designed to help relieve the annulus pressure should a break occur.

The nozzles of the RPV are connected to the main piping using a short transition piece called a safe-end. The postulated break is the weld at either end of the safe-end. There are 26 penetrations in the wall, of which 17 occur where shield doors are required. Of these 17 lines the major ones are the two 28-inch diameter recirculation outlet lines and the ten 12-inch recirculation inlet lines. The safe-end welds for these nozzles lie within the thickness of the shield wall or in the annular space. These two sets of lines were considered the critical cases, because the rest of the lines either are smaller, or may vent

directly to the drywell because of the absence of any shield doors. One more case was considered: the feedwater line safe-end break. Because this line is located at the top of the sacrificial shield, forces generated during a postulated line break have a large moment which can lead to high stresses. The analysis requires modeling the system to predict what forces and pressures are generated following a postulated failure of safe-ends from these three lines and then using these in a structural design assessment.

6.2.1.3.11.2 Summary of Study

A study was performed in 1973 using state-of-the-art methods. A detailed report of that study was filed with the AEC in response to Open Item No. 12 and Question 12.4, Amendment 11 of the PSAR. (Refer to Reference 11 in Section 3.8). During the review of the FSAR, the NRC questioned whether certain aspects of the model used to predict the pressure distribution were adequately conservative and requested that the calculation be repeated using models currently available.

The recalculation was broken down into three tasks: calculation of mass energy release, calculation of annulus pressure distribution history, and the structural design assessment.

MASS ENERGY RELEASE

This task was performed using a method developed by the General Electric Co. The method assumes that the initial fluid velocity is zero. After the break, a finite time is required to accelerate the fluid to steady-state velocities; this is called the inventory period. The flow rate during this period is computed by two methods: one method includes the effect of inventory and subcooling on flow in the pipe, the other method accounts for the finite break opening time. The smaller of the two flow rates at any time is used. Both methods produce maximum flow rates based on different limiting areas. The transfer from one curve to the other represents a change in the point where the flow is choked. Following the inventory period the flow is assumed to be choked at the limiting cross-sectional flow area. Mass flux is calculated using the Moody steady slip flow model with subcooling. Results of this calculation are in Reference 9. 12

ANNULUS PRESSURIZATION

The computation of pressure distribution in the annulus following these breaks was based on the use of the computer code COMPARE. The model for Fermi 2 used 42 nodes in the annulus and four nodes in the drywell. The analysis considered movement of insulation at penetrations and the resulting venting of fluid to the drywell. The code was modified to account for variable junction area as a function of time. A Moody multiplier of 0.6 was used for all junctions except that from the break to the annulus, where 1.0 was used. All junctions had an inertia term, and subcritical flow was calculated on the basis of a solution to the momentum equation

with constant density. Reference 8 is the report of this work. Three copies of the reference were included with the submittal of Amendment 12.

STRUCTURAL DESIGN ASSESSMENT

The structure was analyzed using the Sargent & Lundy thin shell of revolution computer code, DYMAX. The Fermi 2 model for this study consisted of 76 nodes. Reference 10 is the report of this work. Three copies of this reference were included also with the submittal of Amendment 12.

The loads included in the study were

12 a. annulus pressurization

b. jet impingement

c. pipe whip reaction

d. dead load

e. thermal effect due to accident

f. seismic effect due to OBE and SSE.

The structural components assessed were

a. sacrificial shield

b. reactor pedestal

c. stabilizer truss

d. reactor anchor bolts

e. sacrificial shield anchor bolts.