APPENDIX 19B EX-VESSEL SEVERE ACCIDENT PHENOMENA

One of the key AP600 severe accident design features is the capability to retain the core debris within the reactor vessel for a large number of severe accident sequences by flooding the reactor cavity and submerging the outer surface of the reactor vessel. The heat removal capability of the water on the external surface of the reactor vessel prevents the reactor vessel wall from reaching temperatures where failure of the reactor vessel could occur. This has been termed in-vessel retention and is described in detail is other sections of this chapter. The primary benefit of in-vessel retention of the core is that ex-vessel severe accident phenomena associated with relocation of core debris to the containment, which can be a dominant containment failure mechanism, are physically prevented. Thus, retention of the core within the reactor vessel results in a significant reduction in the potential for significant fission product releases to the environment for core damage accidents.

The probability of various levels of fission product releases (release categories) has been determined in the AP600 Level 2 PRA, using a containment event tree which describes the various severe accident phenomena that can impact the fission product release quantities and probability of release. In the quantification of the AP600 Level 2 PRA it was conservatively assumed that the containment would fail at the time of reactor vessel failure for all core damage sequences in which the core debris could not be retained within the reactor vessel. The two principle ways identified in the Level 2 PRA of retaining the core within the reactor vessel are reflooding the core with water before the core begins to relocate within the reactor vessel and submerging the outer surface of the reactor vessel to the reactor coolant loop nozzles. Using this conservative approach, the regulatory and industry severe accident performance targets for the AP600 design criteria were met. Therefore, it was considered unnecessary to investigate the consequences of reactor vessel failure on a realistic basis, including quantification of uncertainties.

The purpose of this section is to provide the results of a limited number of deterministic investigations of the consequences of ex-vessel severe accident phenomena for the AP600 design. The results of these deterministic investigations show that the challenges to the integrity of the containment posed by ex-vessel severe accident phenomena are generally within the structural capability of the containment. From these investigations, the conclusion is the capability to prevent large fission product releases to the environment does not depend on the ability to retain the core within the reactor vessel for core damage accident sequences.

The AP600 design includes features to enhance the likelihood of retaining the core within the reactor vessel for severe accident sequences. These features include:

- Depressurization of the reactor coolant system in the event of an accident by either automatic or manual actuation of the automatic depressurization system

- A containment layout wherein the water relieved from the reactor coolant system (either from the ADS discharge or a break in the RCS) would accumulate in the reactor cavity region

- The capability to manually initiate flooding of the reactor cavity by gravity draining the in-containment refueling water storage tank into the reactor cavity

- The absence of in-core penetrations in the reactor vessel bottom head eliminates a possible reactor vessel failure mode

- The reactor cavity layout provides for rapid flooding of the reactor vessel to the reactor coolant loop nozzle elevation

- The unique reactor vessel insulation design assures the ingress of water to the region between the insulation and the reactor vessel and the egress of steam from that same region to promote cooling of the external surface of the reactor vessel.

Some of the AP600 design features to reduce the probability of a core damage accident and to enhance the likelihood of in-vessel retention of core debris in the event of a core damage accident are counter to the design philosophy that would be used to mitigate the consequences of ex-vessel severe accident phenomena. In particular, two of the design features are mutually exclusive between preventing ex-vessel phenomena and mitigating the consequences of ex-vessel phenomena. On the balance, the AP600 severe accident risk profile is substantially reduced by the features that prevent ex-vessel severe accident phenomena. Two of the more noteworthy features are:

- The large mass and low power density of the AP600 core provides for a much slower accident progression which enhances the capability to prevent a core damage accident (i.e., a reduced core damage frequency). The larger mass of core materials may result in more severe consequences from some of the potential ex-vessel phenomena such as core debris coolability, direct containment heating and core concrete interactions.

- The small reactor cavity floor area reduces the amount of water required to completely submerge the reactor vessel. The small cavity floor area also provides for a more rapid flooding of the cavity if manual initiation of IRWST draining to the reactor cavity is required to submerge the reactor vessel. The small reactor cavity floor area may result in more severe consequences from some of the severe accident ex-vessel phenomena such as core debris coolability and core concrete interactions.

The limited deterministic investigations of ex-vessel severe accident phenomena described in this section includes: ex-vessel steam explosions, direct containment heating and core concrete interactions. These ex-vessel phenomena are strongly dependent on the assumptions made concerning the mode of reactor vessel failure for the AP600 design. Therefore, the reactor vessel failure mode is described first, followed by a description of the ex-vessel phenomena investigations.

19B.1 Reactor Vessel Failure

The AP600 reactor vessel has a main cylindrical section and a hemispherical bottom head. The bottom head is made of carbon steel with an inner cladding of stainless steel to prevent contact between reactor coolant and carbon steel during normal plant operations. The bottom head of the reactor vessel does not contain any discontinuities or penetrations that could impact the mode of reactor vessel failure as the molten core material relocates to the bottom head.

Based on investigations of the possible failure modes for the AP600 reactor vessel, as documented in reference 19B-1, the most likely failure mode is creep failure of the vessel wall due to heating of the vessel wall by the core debris that has relocated to the reactor vessel bottom head. Since creep failure is a strongly temperature-dependent phenomena, the location of the failure is predicted to be at the upper surface of the core debris pool that has relocated to the reactor vessel bottom head. For most severe accident sequences, this location is near the junction of the hemispherical bottom head and the cylindrical portion of the vessel.

As described in reference 19B-2, the presence of water on the external surface of the reactor vessel, as in the case of a flooded reactor cavity, does not alter the conclusion that the highest heat fluxes to the reactor vessel walls will be at a point near the top of the in-vessel molten core pool. This would correspond to the region of the reactor vessel most susceptible to creep failure. However, as described in reference 19B-2, the failure of the reactor vessel for the case in which the reactor coolant system is depressurized and the reactor cavity is filled with water to the reactor coolant loop elevation is physically unreasonable. Even considering the uncertainties related to in-vessel retention of core debris for this case, the potential for vessel failure is physically unreasonable.

For the case in which the outside of the reactor vessel is initially submerged but a sufficient in-flow of water to the reactor cavity cannot be maintained, the reactor vessel wall location experiencing the highest heat fluxes would uncover and lose its external cooling before other locations on the reactor vessel lower head. Thus, creep failure of the vessel would be expected to occur at the same location as the case with no water in the reactor cavity.

Two reactor vessel creep failure cases, as described below, are carried through the deterministic analyses of ex-vessel steam explosions and core concrete interactions. However, in the case of direct containment heating, a failure of the reactor vessel at an elevation at or above the surface of the core debris in the reactor vessel would result in only a small fraction of the total core debris being transported from the reactor vessel. With only a small fraction of the core debris transported to the containment, direct containment heating does not represent a challenge to the integrity of the containment. Thus, the deterministic analysis of direct containment heating considers an undefined reactor vessel failure near the bottom of the vessel bottom head such that all of the core debris that has relocated to the bottom head at the time of vessel failure will be expelled from the reactor vessel into the reactor cavity before the discharge of reactor coolant system gases into the cavity.

For the consideration of ex-vessel steam explosions and core-concrete interactions, it is assumed that the reactor vessel is initially submerged in water but that gravity draining of water from the IRWST does not occur. In this case, the heat removed from the reactor vessel by the water in the reactor cavity results in a boil-down of the water in the cavity. The steam produced by the boiling is condensed by the containment heat sinks and returned to the

IRWST and the refueling canal where it is held-up and cannot return to the cavity. As the water in the reactor cavity boils down, the outside of the reactor vessel at the elevation at the top of the in-vessel core pool will dry-out and begin to heat up. As the vessel wall heats up, it undergoes thinning due to dissolution and melting until failure occurs. The manner in which the reactor vessel fails is treated in two separate cases described below.

In the first case, the formation of a localized opening occurs due to asymmetric heating around the circumference followed by the vessel tearing around nearly all of its circumference. This would result in the bottom part of the reactor vessel and the bottom head hinging such that the lower head swings downward and comes to rest on the cavity floor. This behavior is illustrated in Figure 19B-1. A hinging type of failure would result in an immediate pouring of core debris onto the cavity floor with metal flowing ahead of oxide. The relationship between the height of the reactor vessel above the floor is such that all but a minor part of the oxide melt would be free to flow immediately out of the head.

In the second case, the head and bottom part of the vessel do not hinge downward. In this case, the formation of a localized opening permits molten core debris to drain into the cavity lowering the in-vessel core debris depth and thereby decreasing the thermal load on the vessel wall formerly adjacent to the melt. This type of failure is illustrated in Figure 19B-2. In this case, the continued boildown of water level is followed by the release of the core debris located above the water level after a delay interval during which heatup, thinning, and localized failure of the wall will occur. Over time, the elevation of the failure location moves downward over the vessel wall and lower head. This type of failure gives rise to a very slow release rate with the core debris first relocating downward through the water before collecting and spreading on the cavity floor.

19B.2 Direct Containment Heating

Direct Containment Heating (DCH) is defined as the rapid energy addition to the containment atmosphere as a result of several physical and chemical processes that can occur if the core debris is forcibly ejected from the reactor vessel. The prerequisites for direct containment heating are vessel failure occurs at a location where a substantial portion of the core debris that has relocated to the lower head is ejected into the reactor cavity before the RCS gases are discharged from the RCS and that the RCS is at a high pressure (sometimes called high pressure melt ejection or HPME). Under these conditions, it is postulated that the molten core debris on the reactor cavity floor will be swept out of the reactor cavity with the gases that are discharged from the reactor cavity. The airborne, fragmented core debris then rapidly transfers its sensible heat to the containment atmosphere in one of the containment compartments, as dictated by the gas flow from the reactor cavity. In addition to transfer of sensible heat, the unoxidized metal in the core debris can undergo an exothermic oxidation reaction in the presence of the oxygen in the containment compartment. This heat is also added to the containment atmosphere in that compartment. Finally, if the flammable gases in that containment compartment (including the added flammable gases from the oxidation reactions in that compartment) are ignited, the heat of combustion will be added to the containment atmosphere gases in that compartment. Experimental evidence (reference 19B-3) shows that containment compartmentalization and the flow paths from the reactor cavity to each compartment have a strong effect on the containment conditions that can result from a high pressure melt ejection. A screening model for predicting the potential impact of direct containment heating on the containment integrity was developed from the experimental considerations (reference 19B-4).

The Pilch 2-Cell model presented in reference 19B-4 was used to determine the potential impact of the direct containment heating on the integrity of the containment for the AP600 design.

The application of the Pilch 2-Cell model to the AP600 design leads to the conclusion that direct containment heating would not challenge the integrity of the containment. The results of the bounding analysis show that the final containment pressure will be well below the point where containment failures are predicted to occur.

19B.3 Ex-Vessel Steam Explosions

19B.3.1 Ex-Vessel Steam Explosion Loads

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19B.3.2 Structural Response to Steam Explosions

An evaluation of the structural integrity of the reactor vessel cavity concrete structure, and the response behavior of the reactor pressure vessel subjected to ex-vessel steam explosion loadings was performed. The ex-vessel steam explosion scenarios are of short time duration impulsive loadings and therefore, the dynamic response characteristics of the structures and the reactor vessel are considered.

19B.3.2.1 Assessment of the Reactor Cavity Concrete and Steel Structures

The interior concrete structures in the vicinity of the reactor cavity were analyzed for the dynamic impulsive loading associated with a hinged reactor vessel failure mode. The dynamic impulse loadings associated with the localized mode of reactor vessel failure are enveloped by the hinged failure loadings.

The analysis considered the reactor cavity floor and the wall to be equivalent one-degree of freedom dynamic systems. It was determined that both the floor and wall structures will not retain their structural integrity with loading associated with hinged vessel failure ex-vessel steam explosion loading. Failure of the wall will not impair the overall structural integrity of the interior concrete structure. To determine the effect of the loss of the floor structure on safety function (leakage of radioactivity into the atmosphere) an evaluation of the structural integrity of the containment vessel was made.

The results of the assessment showed that the reactor vessel cavity structural concrete may fail locally inside and outside the containment vessel. However, the containment vessel will be less than approximately 20 percent of its ultimate strain capacity, and therefore, the containment vessel will not leak any radioactivity into the atmosphere, and there is no danger due to this postulated steam blast loading. The containment structure can withstand the peak postulated loading from a hinged reactor vessel failure.

19B.3.2.2 Reactor Pressure Vessel Response

The vertical uplift of the reactor pressure vessel (RPV) as a result of the steam explosion was assessed, as well as the effects of the RPV motion on the steam generator and attached piping.

The resulting calculated displacement of the RPV due to the impulse loading is dependent on the system stiffness and mass, with the maximum displacement occurring when the RPV support stiffness approaches zero. This is the case where there is an instantaneous separation of the RPV from the attached piping and no resistance to RPV motion is present. Results show that the maximum lift of the RPV is not sufficient to exceed the height of the walls associated with the biological shield and refueling water canal. Therefore, the RPV motion is contained and will not affect the integrity of the containment structure and associated equipment.

The reactor coolant loop piping deforms plastically with significant RPV uplift due to the steam explosion. Plastic hinges would form in the piping at the biological containment wall penetrations, and at the RPV equipment nozzles, isolating the movement of the RPV from other reactor coolant loop piping and primary components.

The energy released by the worst case ex-vessel steam explosion is insufficient to propel the reactor pressure vessel outside of the walls associated with the biological shield and refueling water canal. The RPV is therefore contained, and will not impact the containment vessel. The effect on other systems and components attached to the loop piping and components is minimized due to the plastic deformation of the loop piping which would isolate the effects of the RPV response due to the ex-vessel steam explosion. The containment vessel will not be compromised as a result of the ex-vessel postulated steam explosion events.

A similar assessment of the potential downward movement of the reactor vessel following an ex-vessel steam explosion in reactor cavity was performed. The downward movement might result from damage to the cavity walls which are part of the reactor vessel support. The assessment concluded that plastic deformation of the loop piping would isolate the effects of the RPV response. In addition, the potential impact of dropping the reactor vessel onto the cavity floor produces loads that are small compared to the steam explosion loads.

19B.4 Core Concrete Interactions

If the reactor vessel fails when the RCS is at a low pressure, the molten core debris will pour from the reactor vessel onto the reactor cavity floor. If a steam explosion does not occur, the pour will spread over the cavity floor and begin to transfer heat to the concrete floor of the reactor cavity. Due to the predicted mode of reactor vessel failure and the shape of the AP600 reactor cavity, analyses of the possible spreading of the core debris over the cavity floor were conducted. The results were then used as input for analysis of core concrete interactions.

An investigation of the spreading of core debris that pours into the reactor cavity was conducted for reactor vessel failure that occurs at low RCS pressure. The investigation considered the vessel failure mode and location, as well as the recognition that the oxide and metal components of the in-vessel core debris are predicted to be separated. Since the oxide and metal components of the core debris have very different physical characteristics (e.g., viscosity, heat capacity, etc.), the separated in-vessel layers influence the spreading of the core debris in the reactor cavity. A melt spreading analysis was conducted for both the hinged and the localized reactor vessel failure modes.

For the hinged vessel failure case, the analysis results show that the core debris is spread relatively uniformly over the reactor cavity floor. However, the distribution of the metal and oxide components of the core debris are not uniformly distributed over the reactor cavity floor. In the region directly under the reactor vessel, the core debris consists primarily of the oxide component. At the opposite end of the reactor cavity, the core debris consists mainly of the metal component of the core debris released from the reactor vessel. The core debris is still almost totally molten at the end of the spreading analysis.

A different behavior is predicted for the localized reactor vessel failure case. The analysis predicts that the core debris will accumulate at the reactor vessel end of the reactor cavity. The distribution of the metal and oxide components of the core debris are not uniformly distributed over the reactor cavity floor. In the region directly under the reactor vessel, the core debris consists primarily of the oxide component. At the opposite end of the reactor cavity, the core debris consists mainly of the metal component of the core debris released from the reactor vessel. The core debris is almost totally frozen at the end of the spreading analysis.

The core concrete interactions for the AP600 design were analyzed for two concrete types: basaltic concrete and common limestone-sand concrete. The common limestone-sand concrete has a significantly higher noncondensible gas generation rate, compared to basaltic concrete and should therefore present a more severe containment pressurization transient. On the other hand, the basaltic concrete has a significantly higher core penetration rate, due to the physical properties of the concrete, and should therefore present a more severe basemat penetration failure mode, compared to common limestone-sand concrete.

Based on the analyses, it can be concluded that:

- The goal of protecting the containment fission product boundary during the first 24 hours of a core melt accident is met.

- It is not necessary to specify a concrete type for the containment basemat since credible containment basemat failures that could lead to fission product releases to the atmosphere are likely to occur at times well beyond 24 hours.

- The reactor cavity sump is adequately protected such that it is not a weakness in containment basemat integrity during postulated accidents that lead to core concrete interactions.

The reactor cavity design incorporates features that extend the time to basemat melt-through in the event of reactor pressure vessel failure. The cavity design includes:

- A minimum floor area of 48 m^2 available for spreading of the molten core debris.
- A minimum thickness of concrete above the embedded containment liner of 0.85 m.
- There is no piping buried in the concrete beneath the reactor cavity; sump drain lines are not enclosed in either of the reactor cavity floor or reactor cavity sump concrete. Thus, there is no direct pathway from the reactor cavity to outside the containment in the event of core-concrete interactions.
- The openings between the reactor cavity and cavity sump are small diameter openings in which core debris in the cavity will solidify. Thus, there is no direct pathway for core debris to enter the sump, except in the case where it might spill over the sump curbing.

19B.4.1 Containment Pressurization Due to Core Concrete Interactions

The containment pressurization due to steam and noncondensible generation during the episodes of core concrete interactions described above was assessed to determine the effect of core concrete interactions on the containment integrity.

The indicator of a challenge to containment integrity for the containment pressurization due to the noncondensible gases produced from core concrete interactions is the Service Level "C" pressure which is well below the 50% containment failure probability value.

From the assessment, it is concluded that the containment conditions at 24 hours after reactor vessel failure are not sensitive to the assumed mode of reactor vessel failure.

The results also show that, in all cases except the localized reactor vessel failure with a common limestone-sand concrete basemat, the containment does not pressurize to Service Level C containment challenge indicator value prior to the time that the core debris completely penetrates the containment basemat. Thus, for these cases there is no potential challenge to containment integrity due to overpressurization. In the case of a localized reactor vessel failure and an assumed basemat concrete composition of common limestone-sand the analysis results show that the indicator for containment overpressurization challenge (Service Level "C") falls within the same time frame as the complete basemat penetration indicator for basemat integrity challenge. Since the complete basemat penetration indicator of a basemat integrity challenge is the optimistic bounding indicator and the containment Service Level C pressure is the pessimistic bounding indicator for containment overpressure challenge, the union of the two sets represents a very small conditional probability. Based on the subjective uncertainties represented by these bounding indicators, it is likely that even in this case, containment basemat failure would occur, allowing the containment to depressurize, thereby precluding a containment overpressure failure.

Based on the core-concrete interaction assessments, it can be concluded that it is not necessary to specify a concrete type for the containment basemat since containment overpressure failure due to non-condensable gas generation from core concrete interactions is not likely for any credible severe accident scenarios.

19B.5 Conclusions

The results of the limited deterministic analyses of ex-vessel severe accident phenomena presented in this section show that early containment failure is not a certainty if the reactor vessel fails. Based on the deterministic analyses, direct containment heating that might ensue from a high pressure melt ejection would not challenge the integrity of the containment. Ex-vessel steam explosions, assessed on a very conservative basis would not produce impulse loads that would challenge the integrity of the containment due to localized failures of the reactor cavity floor and walls. In addition, these analyses indicate that the ex-vessel steam explosion loads are not strong enough to displace the reactor vessel from its location inside the biological shield. Thus, there is no challenge to any containment penetrations connected to the reactor vessel or to the reactor coolant loops. In the case of a vessel failure at a low RCS pressure, the core concrete interactions analyses indicate that the containment integrity would not be challenged in the first 24 hours of the event and thus no significant releases of fission products are predicted in that time frame.

Thus, it is concluded that prevention of large fission product releases to the environment is not dependent on the integrity of the reactor vessel. If reactor vessel failure occurs, there may be challenges to the containment integrity, but these challenges are highly uncertain and the most likely challenge (containment failure by core penetration of the cavity basemat) would not occur in the first 24 hours of the accident.

19B.6 References

19B-1 "AP600 Phenomenological Evaluation Summaries," WCAP-13388 (Proprietary) Rev. 0, June 1992 and WCAP-13389 (Nonproprietary), Rev. 1, 1994.

19B-2 Theofanous, T.G., et al., "In-Vessel Coolability and Retention of a Core Melt," DOE/ID-10460, July 1995.

19B-3 "The Influence of Selected Containment Structures on Debris Dispersal and Transport Following High Pressure Melt Ejection From the Reactor Vessel," NUREG/CR-4914, September 1988.

19B-4 Pilch, M.M., "Adiabatic Equilibrium Models For Direct Containment Heating," SAND-91-2407C, December 1992.

19B-5 Deleted.

19B-6 Sienicki, J.J., et al., "Analysis of Melt Spreading in an AP600-Like Cavity," DOE/ID-10523, February 1996.

19B-7 Deleted.

TABLES B-1 THROUGH B-5 NOT INCLUDED IN THE DCD.

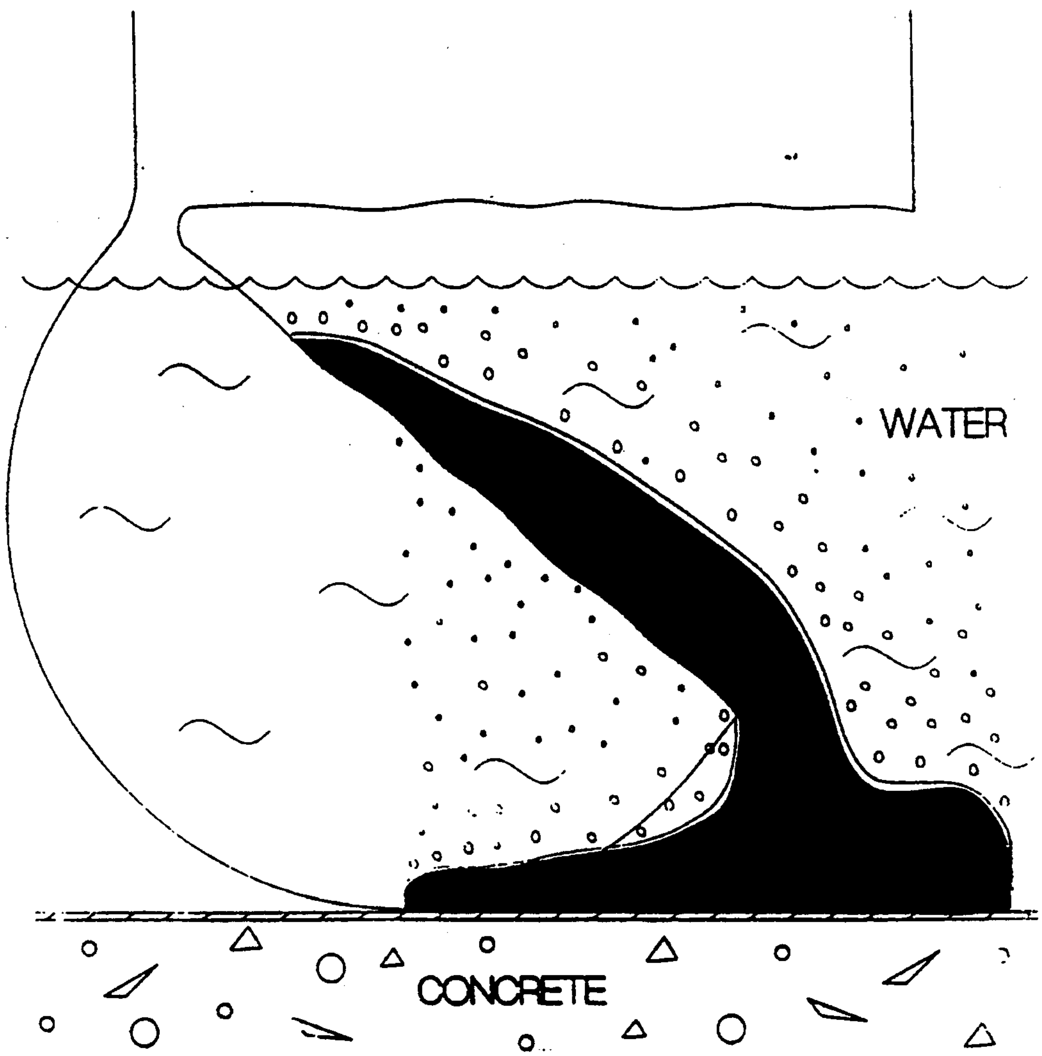

Figure 19B-1

Illustration of Hinging Type of Failure Resulting in Rapid Melt Release (from Reference 19B-6)

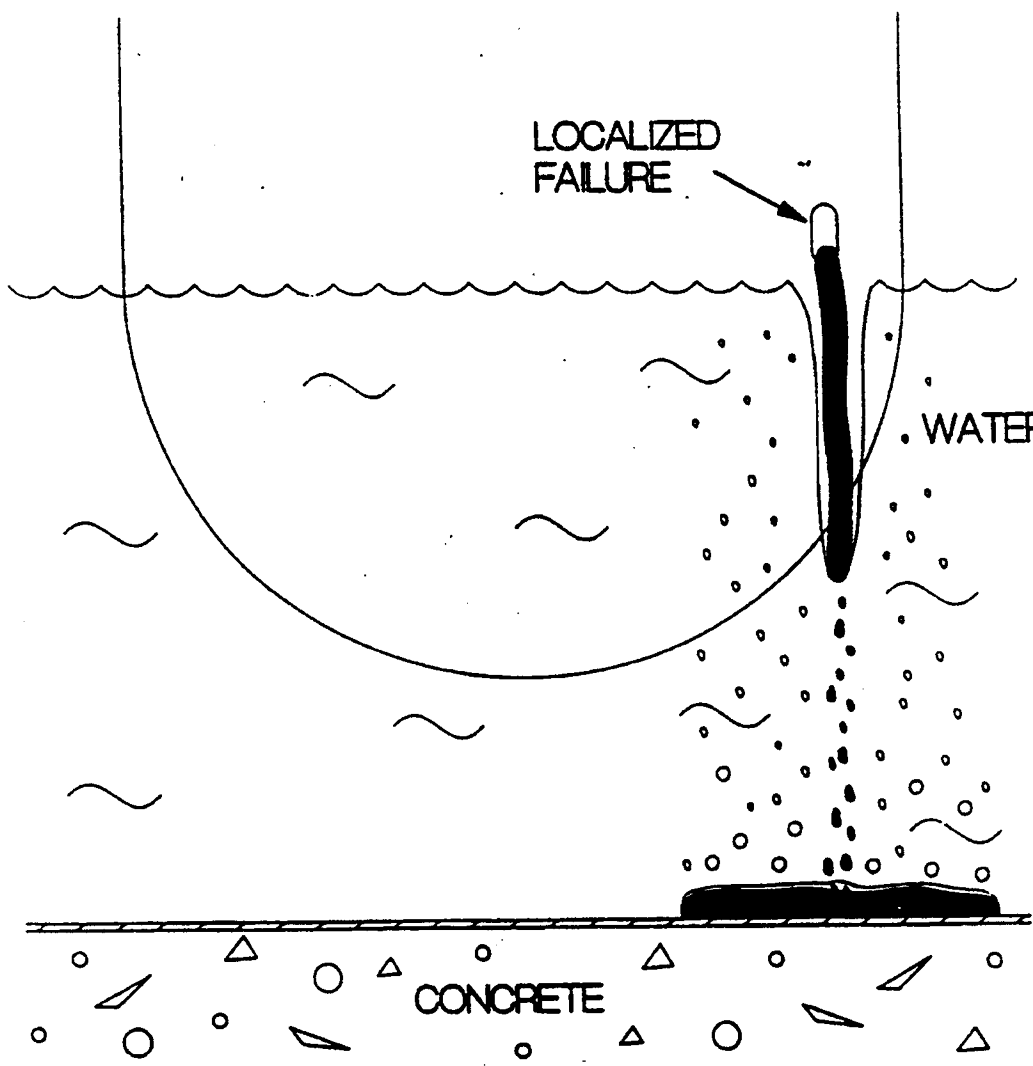

Figure 19B-2
Illustration of Localized Type of Failure Resulting in Slow Melt Release (from Reference 19B-6)

FIGURES B-3 THROUGH B-7d NOT BINCLUDED IN THE DCD.