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Analysis of Hypothetical Severe Core Damage Accidents for the Zion Pressurized Water Reactor

F. E. Haskin, J. L. Darby, W. B. Murfin

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Abstract

This report describes analyses of the response of a Pressurized Water Reactor at the Zion Plant to hypothetical core meltdown sequences. The analyses consider the progression of core meltdown, containment response, and consequences to the public for many specific accident sequences within the categories of Loss of Coolant Accidents (LOCAs), transient-initiated accidents, and containment-bypass accidents. The report does not not deal with the probability of the accidents occurring. Strategies for accident management and mitigation of consequences are suggested. Uncertainties in the calculated plant responses are described.

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Summary

As part of the Severe Accident Sequence Analysis (SASA) program of the Nuclear Regulatory Commission (NRC), we have analyzed hypothetical severe-core-damage accidents for the Zion plant. The Zion plant consists of two, virtually identical units each having a Pressurized Water Reactor (PWR) located in a large dry containment building. We have dealt with core meltdown, containment response, and consequences to the public. In this study, we have not addressed the probability that a core meltdown accident occurs.

We have analyzed many specific accident sequences within the following categories: Loss of Coolant Accidents (LOCAs), transients, and containment bypass accidents. Operator actions to mitigate the consequences of severe accidents have been identified and examined.

A severe accident sequence can be divided into three time regions: before core damage, during core damage and meltdown, and after core meltdown. We have summarized ways by which core damage can be prevented from the SASA work by EG&G Idaho, Inc. and Los Alamos National Laboratory. If an accident leads to core melting, we cannot demonstrate that melting can be arrested, because phenomenological uncertainties associated with the melting process are large. We do not believe that the overall strategy of coping with severe accidents should focus on the ability to cool a partially damaged core. Preserving containment integrity should be stressed from the onset of an accident. Substantial amounts of radionuclides are released from the core once core degradation begins, and these radionuclides must be retained within containment even if complete core melting is prevented. Also, the radiological consequences associated with a complete core meltdown are small if containment integrity is maintained.

We have examined the potential for massive above-grade failure of containment at Zion for accidents which progress through core meltdown. The Zion containment is large and strong. Our best estimate is that overpressurization will not occur early in any accident sequence, and the steam spike due to vessel breach will not fail containment.* We used 1.03 MPa (149 psia) as the best-estimate failure pressure for Zion containment. [This value is based on analyses by other investigators. We did not examine the possibility that containment penetrations might fail below this level, because the necessary data was not available to us.] Based upon a thermodynamic analysis, we estimate that overpressurization would not occur before six hours into any accident sequence. Overpressurization would require the failure of both containment sprays and fan coolers which would otherwise be available to provide containment

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*The term overpressurization excludes hydrogen burning by convention.

heat removal. Actuation of one fan cooler within six hours should prevent overpressurization.

If containment heat removal is continuously available, hydrogen burning initiated at $\leq$ 10 mole percent should not fail containment. If no containment heat removal is available, combustion of hydrogen will not occur because the containment atmosphere will be steam inert (noncombustible due to the large quantity of steam present). If containment heat removal is initially unavailable, but is recovered late in an accident after the core has melted, a rapid condensation of steam could produce containment-threatening hydrogen burns.

The degree to which containment is isolated has a major effect on radiological consequences. Severe accidents in which containment is bypassed produce the highest consequences of any of the many accidents which we have analyzed. If containment is not bypassed and if containment does not fail by overpressurization or hydrogen burning, the rate at which radionuclides leak from containment dominates radiological consequences. If containment can be isolated to maintain the design leakage limit, consequences are small.

We recommend the following operator actions during severe accidents:

- Check for containment bypass and isolate if possible. Manually close containment isolation valves as necessary.

- Begin immediate containment heat removal if possible.

- If containment fan coolers are operating, limit the use of containment sprays before clad failure in order to delay the necessity for recirculation of water from the containment sump.

- Operate containment sprays continuously, if possible, after clad failure. Attempt to maintain or re-establish the capability for spray injection following switchover to spray recirculation from the containment sump.

- Attempt to ensure that all available water from the refueling water storage tank is injected into containment. Avoid injecting only small amounts of water which could decrease the time to containment failure due to overpressure.

- If hydrogen burning has been suppressed by steam inerting long enough to permit combustible gases to accumulate to levels which could fail containment given ignition, carefully control containment pressure (preferably with sprays) to remain below failure limits but above the steam-inert limit.† Containment conditions would remain potentially

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†Maintain steam-inert conditions to prevent hydrogen burning.

hazardous, and eventual inerting by addition of ignition suppressants or by removal of hydrogen would be required; precautionary evacuation would be appropriate pending permanent solution of the hydrogen problem.

If existing plant instrumentation survives pressures, temperatures, and radiation levels associated with severe accidents, the operator has enough information to initiate the actions which we recommend.* However, an instrument that provides real-time analysis of hydrogen/oxygen/steam content within the containment atmosphere would facilitate operator mitigative actions during severe accidents. No such instrument is presently available.

The conclusions and recommendations of this study apply only to the Zion plant and should not be extrapolated to other nuclear power plants.

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*We have not examined equipment survivability in this study.

1. INTRODUCTION

The Reactor Safety Study (RSS) investigated the risk from core meltdown accidents and found that, although the likelihood of a core meltdown accident was very low, the consequences to the public could be significant [WASH-1400, 1975]. The RSS found that the risk from meltdown accidents dominated the risk from all LWR accidents. The RSS assumed that a meltdown accident, once started, continued inexorably to its conclusion. Penalties were assessed for human failures to properly initiate essential safety functions but there was no treatment of human intervention to exacerbate, mitigate, or halt the accident. The event at Three-Mile Island showed that this RSS assumption might be either conservative or overly optimistic [TMI, 1980], [Kemeny, 1979], [NSAC, 1979]. The accident was initiated by a normal transient and exacerbated by hardware failures and human errors. However, later human actions halted core damage and established means for long term cooling with the result that public radiological consequences were insignificant [TMI, 1980].

Investigations and follow-ups pointed out the need for a better understanding of the man-machine interface in potentially severe accidents, and especially for an understanding of the strategies that might be employed to halt or mitigate what would otherwise be a meltdown accident with high public consequences [NUREG-0585, 1979].

In response to this need, the Nuclear Regulatory Commission instituted the Severe Accident Sequence Analysis (SASA) program. The objective of SASA is to improve the understanding of accident phenomenology and of the human-machine interaction over a broadened spectrum of accident sequences.

Sandia National Laboratories (SNL), Idaho National Engineering Laboratory (INEL), and Los Alamos National Laboratory, are cooperating in an investigation of Pressurized Water Reactor (PWR) plants in the SASA program. INEL and Los Alamos are analyzing the initiation of accident sequences, before any core damage takes place. SNL has been asked to concentrate on the "back-end," that is, the period from the time the core begins to be uncovered to either eventual recovery or meltdown and (if applicable) containment failure.

For investigation of the man-machine interface, and of the operator responses, it is necessary to evaluate the information presented to the operator. The conservative analyses presented for licensing applications often do not accurately represent the plant responses. Ideally, the calculated results should be neither conservative nor optimistic. Unfortunately, there is considerable uncertainty in some of the data and in the models used to calculate the responses. This uncertainty prohibits a truly realistic calculation.

It is sometimes possible to bound the plant responses by using both optimistic and pessimistic assumptions and models; it is then known that the true plant responses lie somewhere between the bounds.

This procedure was followed to some extent in this study. However, attempting to bound the responses greatly increases the number of calculations required, and thus increases the cost of the study. Also, the upper and lower bounds are often so far apart that few meaningful conclusions can be drawn.

In our investigation of accident progression we have generally used best-estimate, state-of-the-art techniques. We have interpreted "best estimate" to mean that our assumptions should reflect, as a goal, a 50% confidence level. Because of uncertainty, it is impractical to quantify confidence levels, and we have been unable to determine whether that goal has been met. We have found that many of the state-of-the-art techniques are inherently conservative. Wherever conservative techniques or assumptions had to be used for lack of more realistic techniques, we have pointed out the conservatism. In other cases there is so much uncertainty that it was not possible to choose a best estimate with complete objectivity. We have attempted to choose assumptions and techniques that reflect our personal best estimates; we make no claim that the choices we have made are unique.

The first plant chosen for study was the Zion plant. Zion is a two-unit, Westinghouse-PWR plant operated by the Commonwealth Edison Company. The plant was the subject of an NRC study to determine whether special containment venting arrangements were warranted [Murfin, 1980]. A probabilistic risk analysis of Zion [Zion PRA, 1982] is currently being reviewed by the NRC.

Detailed plant data were not available to us. Sufficient information could generally be garnered from such sources as the Final Safety Analysis Report, Piping and Instrumentation Diagrams, and the Plant Systems Descriptions. However, we believe that a more complete analysis would require more plant details than were available for this study. We found information on plant instrumentation to be especially difficult to obtain.

This report summarizes our analyses of a number of severe accident sequences at Zion. It is a follow-on and an expansion of an earlier report which examined one particular sequence (TMLB' - a transient-initiated sequence involving total loss of feedwater and AC electric power) in detail [Haskin et al., 1981].

2. EVENT TREES

An event tree identifies the possible outcomes resulting from an initiating event. A path from the initiating event through subsequent events to a final outcome is called a sequence. In this section we present three event trees which identify (1) paths leading to possible core damage, (2) paths leading to different degrees of core damage, and (3) paths leading to different radiological consequence levels associated with possible containment failure modes. A complete accident sequence is a path through all three event trees.

The three event trees are general in nature. They apply for all of the accident initiators considered in this report. Many events depicted on the trees represent successes or failures of rather general functions which could result from a variety of more specific events. For example, a failure to "provide adequate cooling via steam generators" could result from a loss of coolant accident (LOCA) by precluding natural circulation in the primary system or it could result from a loss of (main) feedwater transient followed by a failure of auxiliary feedwater.

The general event trees discussed in this section serve to introduce the more specific, potentially significant sequences selected for analysis in Section 4. Insofar as possible, such sequences are identified herein using the nomenclature for PWR events from the Reactor Safety Study [WASH-1400, 1975]. This nomenclature is reproduced in Table 2-1. Parenthetical references to the single-letter event designators in Table 2-1 are made throughout the report.

Another type of functional event tree is the "operator action" event tree. Each event in this type of tree is a function which might be achieved by operator action(s). For instance, the function "restore core injection" could be accomplished if any of several pieces of equipment could be activated. Operator action event trees are particularly applicable to mitigating actions [Fletcher et al., 1980].

2.1 Early-Sequence Event Tree

Figure 2-1 is a simplified early-sequence event tree which identifies the functional failures which can lead to core damage. The first function on the event tree is reactor scram. Since accidents involving failure to scram (Event K, Table 2-1) are not dominant contributors to risk for PWRs [WASH-1400, 1975], [Carlson et al., 1981], [Kolb et al., 1981], they are not analyzed in this report. However, as indicated in Outcome 4 on Figure 2-1, it should be recognized that failure to scram does not always lead to core damage. For example, given a large LOCA, injection of borated water is sufficient to maintain subcriticality and control rod insertion is not required. Also it should be noted that the physical phenomena governing core meltdown and associated threats to containment integrity are the same for accidents involving failure to scram as for those accidents explicitly analyzed in this report.

Table 2-1

PWR Event Nomenclature*

A	- Intermediate to large LOCA.
B'	- Failure of electric power to ESF.
B	- Failure to recover either onsite or offsite electric power within 1.5 hours following an initiating transient which is a loss of offsite AC power.
C	- Failure of the containment spray injection system.
D	- Failure of the emergency core cooling injection system.
F	- Failure of the containment spray recirculation system.
G	- Failure of the containment heat removal systems.
H	- Failure of the emergency core cooling recirculation system.
K	- Failure of the reactor protection system.
L	- Failure of the secondary system steam relief valves and the auxiliary feedwater system.
M	- Failure of the secondary system steam relief valves and the power conversion system.
Q	- Failure of the primary system safety relief valves to reclose after opening.
R	- Massive rupture of the reactor vessel.
S_1	- A small LOCA with an equivalent diameter of about 2 to 6 inches.
S_2	- A small LOCA with an equivalent diameter of about 1/2 to 2 inches.
T	- Transient event.
V	- Low pressure injection system (LPIS) check valve failure.
α	- Containment rupture due to a reactor vessel steam explosion.
β	- Containment failure resulting from inadequate isolation of containment openings and penetrations.
γ	- Containment failure due to hydrogen burning.
δ	- Containment failure due to overpressure.
ϵ	- Containment vessel melt-through.

*Reproduced from [WASH-1400, 1975].

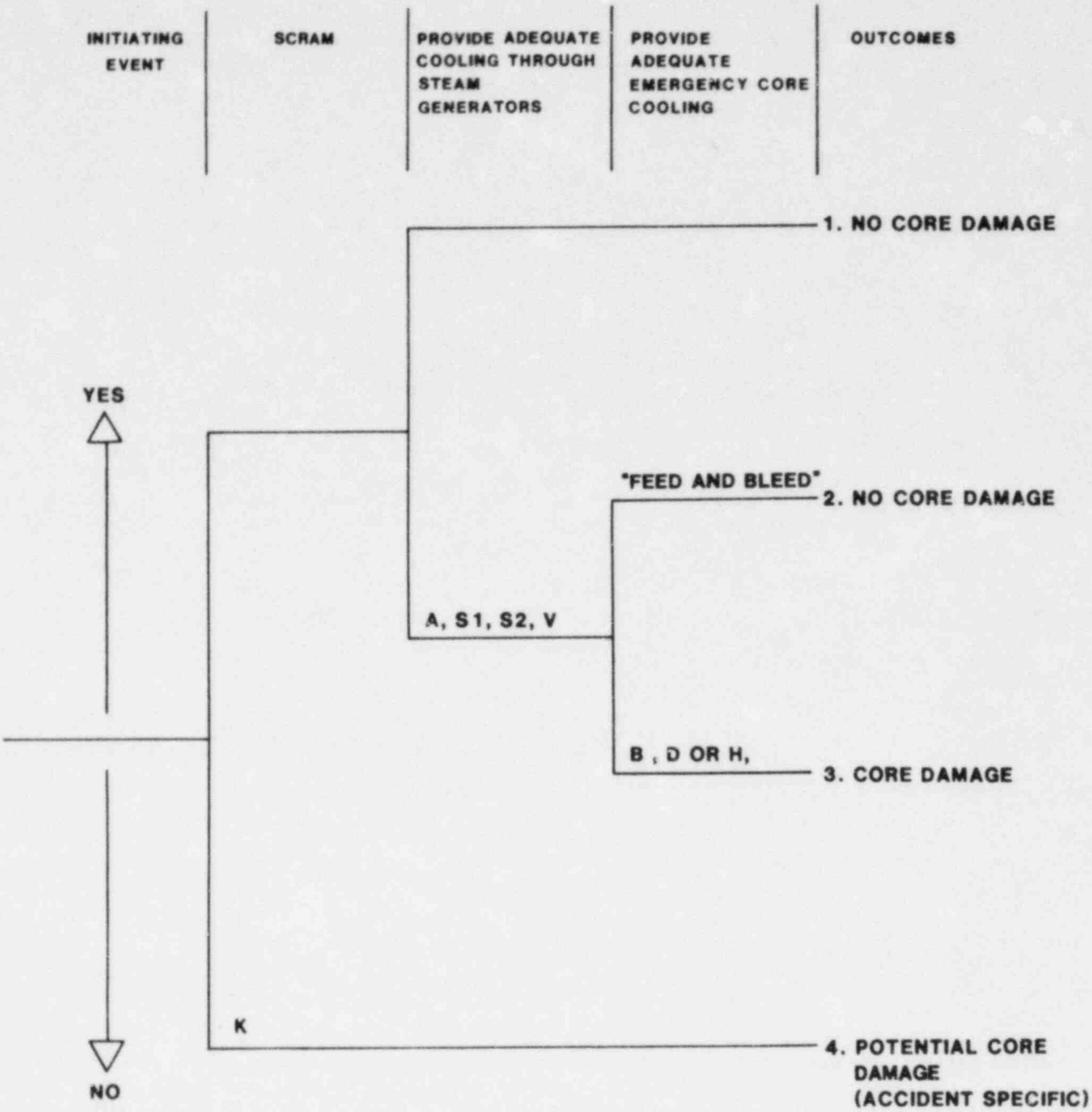

Figure 2-1. Early-Sequence Event Tree

Outcomes 1, 2, and 3 in Figure 2-1 result from accident sequences in which reactor scram occurs. Core damage is avoided in sequences leading to Outcomes 1 and 2 but occurs in sequences leading to Outcome 3.

Outcome 1 is reached when adequate cooling of the primary system is provided via the steam generators. This requires that the primary system pressure boundary be substantially intact. Primary coolant can then flow through the steam generators by natural circulation or be forced through the steam generators if the reactor coolant pumps are available.* Steam from the secondary side of the steam generators would either be relieved to the outside atmosphere or, as long as condenser vacuum and cooling can be maintained, dumped to the condenser. Feedwater to the steam generator secondaries would usually be pumped by the auxiliary feedwater system from the condensate storage tank or from backup sources, if required. The main feedwater pumps (used intermittently) provide backup pumping capability. The plant could be maintained in hot shutdown as long as feedwater could be supplied to the steam generator secondaries or until primary system makeup was required to compensate for primary system leakage (e.g., through pump seals) and/or thermal contraction upon cooldown. The residual heat removal system would be required to cool the plant to cold shutdown after initial cooling via the steam generators.

For some accident initiators, adequate cooling via the steam generators is not possible. In particular, for LOCAs (initiating Events A, S1, S2, and V in Table 2-1) failure of forced and natural circulation would occur. The time required for such failure obviously decreases with increasing break size. Also, for transient-initiated accidents involving total loss of feedwater (TML-initiated sequences per Table 2-1), cooling via the steam generators would become ineffective when all the secondary side liquid boiled away.

If scram occurs, but cooling via the steam generators is not possible, the core can still be cooled if adequate emergency core cooling (ECC) is available. This is true both for LOCAs and TML initiated accidents at Zion. In the latter case, the plant can be maintained at hot shutdown through ECC "feed" and "bleed" via the primary system safety valves or power operated relief valves (PORVs) [DeMuth et al., 1981]. If adequate ECC is provided, Outcome 2 is reached and there is no core damage. Inadequate ECC leads to core damage, as indicated by Outcome 3.

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*At later stages of accidents when primary water inventory is low, steam produced in-vessel could condense on cold steam generator tubes. Based on MARCH code analyses, heat removed in this manner would be inadequate to prevent core melting.

Initially ECC injection of water from the refueling water storage tank (RWST) is attempted. If ECC injection fails (Event D, Table 2-1) or the RWST inventory is depleted, switchover to ECC recirculation of water from the containment sump is required. Failure of the ECC recirculation system is Event H in Table 2-1. The loss of power to ESFs (Events B or B' Table 2-1) also implies failure of ECC (D and H), since AC power is required for system operation. A power outage of limited duration may or may not lead to core damage depending on the accident initiator and the time at which power is reestablished. Event D, ECC injection system failure, does not necessarily imply Event H, ECC recirculation system failure; however, this is a likely result because the two systems differ principally in their source of water (RWST versus containment sump) -- they both utilize the same pumps and much of the same piping. Even if the cause of the ECC injection failure (D) were blocked flow from the RWST and the ECC pumps were still operable, recirculation failure (H) might still result. Such blockage would also result in containment spray injection failure (C), and if the blockage occurred early, water in the containment sump would be from primary system blowdown alone and could be insufficient to permit sustained ECC recirculation. A failure of ECC recirculation could also result from a failure of containment heat removal systems (fan coolers and RHR heat exchanger -- Event G, Table 2-1). The sump water could then heat up and eventually cause recirculation failure due to pump or pump-motor overheating.

2.2 Core-Damage Event Tree

Inadequate core cooling, Outcome 3 of Figure 2-1, would eventually lead to a sustained core uncovering, core degradation through meltdown, breach of the reactor vessel, discharge of molten material into the reactor cavity, and core-concrete interactions. This progression of events can be terminated only when three conditions are satisfied:

(a) water must be continuously available to the core, core debris, or melt in quantities sufficient to quench the material and remove decay heat and heat associated with metal-water reactions,

(b) the core, core debris, or melt configuration must be coolable,

(c) means must be available for cooling the water or condensing the steam produced.

Water could be delivered in-vessel by the ECCS. Water could be delivered ex-vessel by the containment spray system or by the ECCS with the water entering the vessel and running out of the breach into the reactor cavity. Possible heat sinks include the steam generators, the RHR heat exchangers, and the containment fan coolers. Figure 2-2 is an event tree which shows the outcomes obtained by meeting all three termination conditions at various stages of core damage.

If adequate ECC is re-established after core uncovering, but early enough to prevent any melting, the core geometry would still be coolable and releases would be limited to activity in the fuel-clad gap (Outcome 1). If adequate ECC is re-established later, but in time to prevent extensive meltdown (Outcome 2), the resulting configuration would be a damaged but coolable core, perhaps with some coolable debris in the lower head. This condition has been suggested as likely for TMI-2 [TMI, 1980]. Damage could range from minor clad failure to severe core disruption. Halting accident progression at this point would require that some part of the core remain above the lower grid plate in nearly its original configuration. A large fraction of the fission products would be released from the fuel-clad gap. The possibility also exists that some fraction of the volatile fission products would be released from the fuel.

The phenomenology of fragmention and debris bed formation is not well understood; however, a coolable in-vessel debris configuration seems unlikely following complete meltdown [Haskin et al., 1981]. Coolability of ex-vessel debris (Outcomes 3 and 6 in Figure 2-2) can neither be assured nor excluded; however, at least two means for cooling water or condensing steam could be available. If the ECC recirculation system were available, heat could be removed via the RHR heat exchangers. Long-term cooling of water covering debris in the reactor cavity using the ECC recirculation system would require complete filling of the reactor cavity at Zion. Water from the reactor cavity could then overflow into the containment sump, be recirculated through the RHR heat exchangers, and returned to containment via the containment sprays or via the ECCS with water running out of the vessel breach and into the reactor cavity. The containment fan coolers could also be available to condense water vapor from and to cool the containment atmosphere. Fine debris fragments could be swept out of the cavity by the flow of water vapor and could settle into the containment sump. Use of containment air coolers alone could be less effective for cooling but would eliminate the circulation of highly radioactive coolant outside containment. The fission product release would be similar to that for in-vessel cooling (Outcome 2).

If some, but not all, of the necessary termination conditions can be met the accident progression can be delayed. For example, if only a limited amount of water can be supplied to a coolable ex-vessel debris configuration, the accident progression may be delayed until the water supply is exhausted (Outcomes 4 and 7, Figure 2-2).

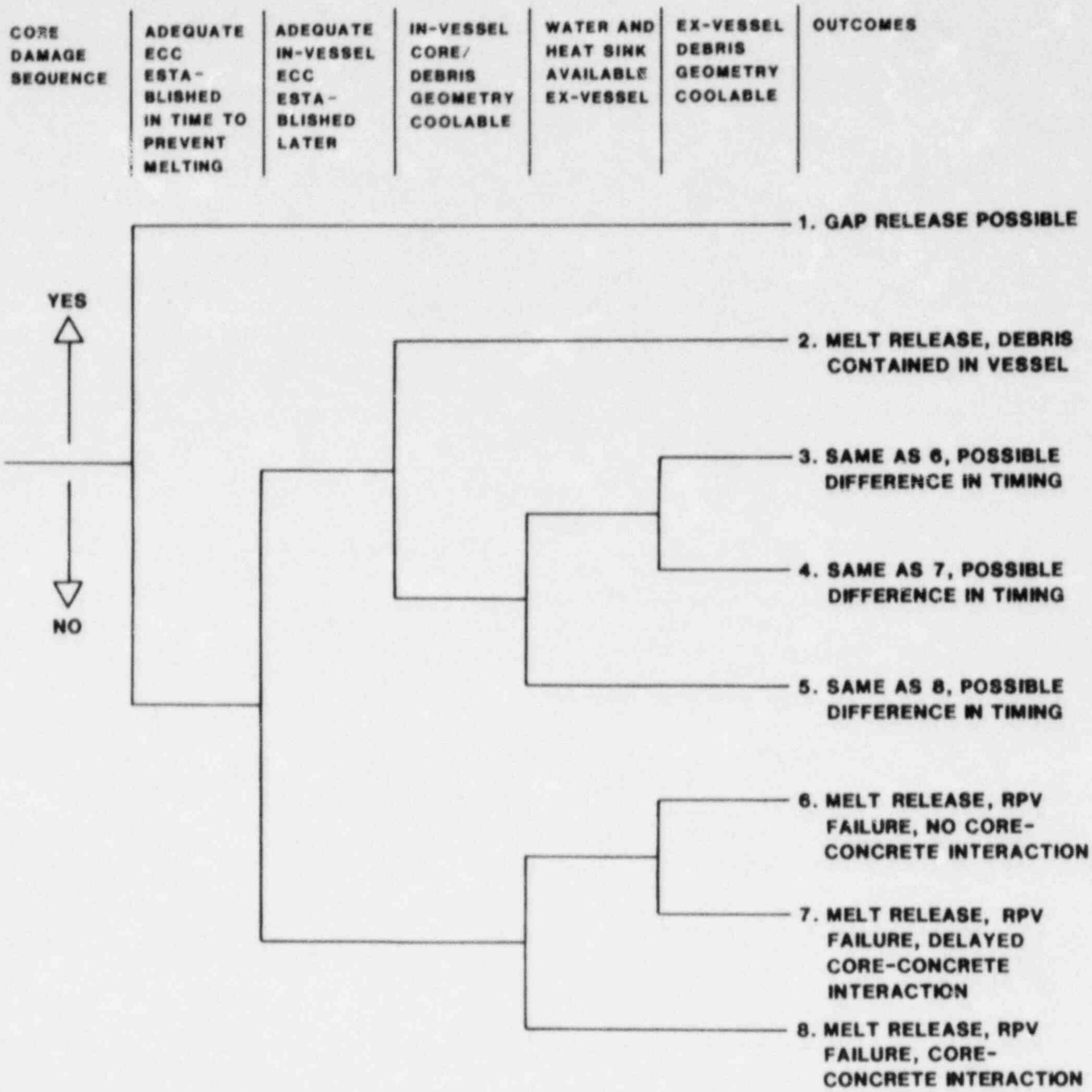

Figure 2-2. Core-Damage Event Tree

2.3 Radiological-Consequence Event Tree

The Reactor Safety Study [WASH-1400, 1975] considered containment failure caused by missiles propelled by a steam explosion (failure mode α), failure of isolation (β), hydrogen burning (γ), overpressure (δ), and basemat melt-through (ϵ). Accident sequences in which containment was bypassed (V) were also considered. The Reactor Safety Study assumed that melt-through was inevitable in all core melt accidents.

More recent studies [Swenson and Corradini, 1981] have determined that containment failure due to steam explosions is unlikely, and this failure mode will not be considered here. Another study [Murfin, 1980] has called into question the inevitability of basemat meltthrough. The Reactor Safety Study showed that radiological consequences of basemat melt-through were much lower than those from other containment failure modes. Because of lower consequences associated with basemat melt-through and the unresolved question of its inevitability, basemat melt-through is not considered in this work.

Figure 2-3 is an event tree which illustrates the consequences of containment failure with and without core melting. Note that the events listed in Figure 2-3 are all undesirable in the sense that the occurrence of any of these events would increase radiological consequences. The outcomes of the containment event tree identify order-of-magnitude differences in mean radiological consequences. Quantitative estimates of radiological consequences for specific accidents are given in Section 5 and Appendix D. Events which do not, within current uncertainties, lead to order-of-magnitude differences in radiological consequences are not included in Figure 2-3. For example, for PWRs, there are no major differences in radiological consequences based solely on whether or not core melting is terminated in-vessel (see Section 5). Similarly, removal of fission products from the containment atmosphere by containment sprays or coolers would not lead to more than an order-of-magnitude reduction in radiological consequences (see Section 5 and Appendix D).*

Sequences in which containment is bypassed, such as the V-sequence interfacing systems LOCA, have the highest potential radiological consequences (Outcome 1). For accidents in which primary system blowdown does not bypass containment, the radiological consequences will depend on the extent of containment isolation or, assuming isolation is adequate, on whether above-ground containment failure occurs due to overpressure or hydrogen burning. There is approximately a three order-of-magnitude range in radiological consequences corresponding to a range of containment leak rates from design leakage to leakage from a large opening (Outcome 2).

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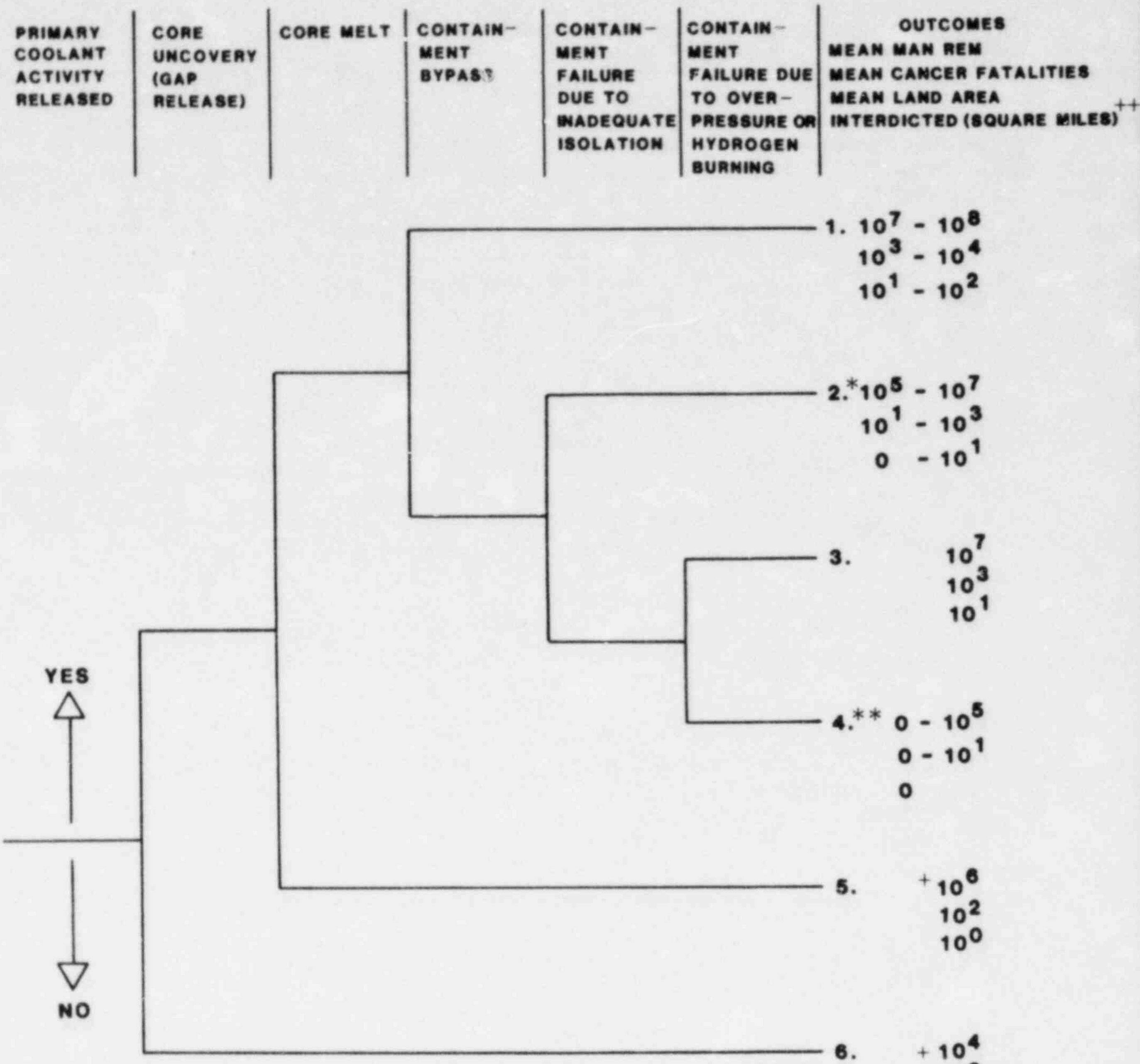
*This statement assumes that sprays and coolers do not operate sufficiently long to affect containment integrity.



*LOWER BOUND FOR DESIGN LEAKAGE, UPPER BOUND FOR LARGE HOLE.

**LOWER BOUND FOR NO LEAKAGE, UPPER BOUND FOR DESIGN LEAKAGE (0.1 VOL % PER DAY).

+MAXIMUM VALUES.

++LAND EXCLUDED FROM OCCUPATION FOR 10 YEARS OR MORE FOLLOWING RELEASE OF RADIATION.

Figure 2-3. Event Tree for Radiological Consequences

If containment is not bypassed and containment isolation is successful, the radiological consequences will be relatively low unless the containment fails due to overpressure or hydrogen burning. At Zion, energy can be removed from the containment atmosphere by the fan coolers or by the containment sprays. With containment sprays, heat removal in the recirculation mode via the RHR heat exchangers must eventually be provided. If either fan coolers or sprays are available, containment failure due to overpressure can be prevented. If these containment heat removal systems are inoperable, containment failure due to overpressure would not occur for many hours after the initiating event (see Sections 4.1, 4.2, and Appendix C).

The failure pressure of the Zion containment has been estimated to be between 0.93 MPa (134.7 psia) [Meyer, 1980] and 1.16 MPa (169 psia) [Stevenson, 1980]. As discussed in Section 8, the amount of hydrogen required for the Zion containment to fail due to hydrogen burning exceeds the amount corresponding to 100 percent oxidation of metal in the core fuel assemblies. However, considering the potential generation of hydrogen and carbon monoxide (which also burns exothermically) from core-concrete interactions, containment failure due to burning is still possible. As discussed in Section 8, for ignition to occur before H_2 and CO have accumulated to levels which could result in containment failure requires that containment ESFs be available to lower the pre-burn containment pressure (thereby decreasing the mole fraction of steam and increasing the mole fraction of hydrogen).

The effects of hydrogen burns which do not fail containment on instruments and equipment within containment have not been determined in this report. Equipment survivability during severe accidents is being investigated in separate, ongoing research programs.

If ignition is delayed, allowing the buildup of H_2 and CO to levels which could result in containment failure upon burning, two actions may still be available which could prevent such failure. First, the use of containment sprays could lower the pressure rise associated with the burn to the point where containment would not fail. Second, hydrogen burning could be precluded as long as containment steam concentrations could be held high enough to ensure steam inerting ($\geq$ 56 mole percent steam). Possible operator actions to reduce the potential for containment failure due to hydrogen burning are discussed in Section 8.

3. COMPUTER MODEL DESCRIPTION

3.1 Thermohydraulic Models

The MARCH computer code [Wooten and Avci, 1980] was used to simulate the responses of primary system and containment both during the initial stage of the severe accident and during subsequent core uncovering, melting and slumping, attack and failure of the pressure vessel, and interaction of the molten debris with water and concrete in the reactor cavity. A February 1981 version, MARCH 1.1, updated to include the contribution of heavy elements to decay heat [ANS 5.1-1979] was used for this report.

RELAP4 predictions of primary and secondary system responses through time of initial core uncovering were obtained from EG&G Idaho, Inc. [Fletcher et al., 1980], [Dearien, 1980a]; [Dearien, 1980b], [Fletcher, 1981] for comparison with MARCH calculations.

Figures 3.1-1 and 3.1-2 show the MARCH and RELAP4 representations of the primary and secondary systems for Zion. MARCH approximates the primary system as one large volume with liquid at the bottom and any vapor at the top. RELAP4 divides the systems into several volumes, each of which may contain a mixture of liquid and vapor and each having a temperature and pressure of its own. Thus MARCH does not provide the same degree of detail regarding the responses of the parameters of primary and secondary systems as does RELAP4. The impact of such modeling differences on predicted parameter responses is discussed and shown graphically in Sections 4.1 and 4.2.

Appendix F includes base case MARCH input decks for the Zion sequences analyzed in the report. Most input parameters describing the plant, such as dimensions, areas, volumes, initial conditions, pump flows, etc., were obtained from the Zion Station System Descriptions [Zion-SD], the Zion Station FSAR [Zion-FSAR], the RELAP4 model [Fletcher et al., 1980], [Fletcher, 1981], and the radiological technical specifications [Zion-RTS].

Section 6 summarizes uncertainties, associated with MARCH, which are relevant to the accidents analysed in this report.

3.2 Radiological-Consequence Models

The analysis of radiological consequences is divided into two areas: behavior of radioactive material in containment, and behavior of radioactive material and consequences to the public after release from containment.

Behavior of radionuclides within containment is modeled by the CORRAL computer code [CORRAL II Users Manual, 1977]. CORRAL is described in Appendix VII of the Reactor Safety Study (RSS)

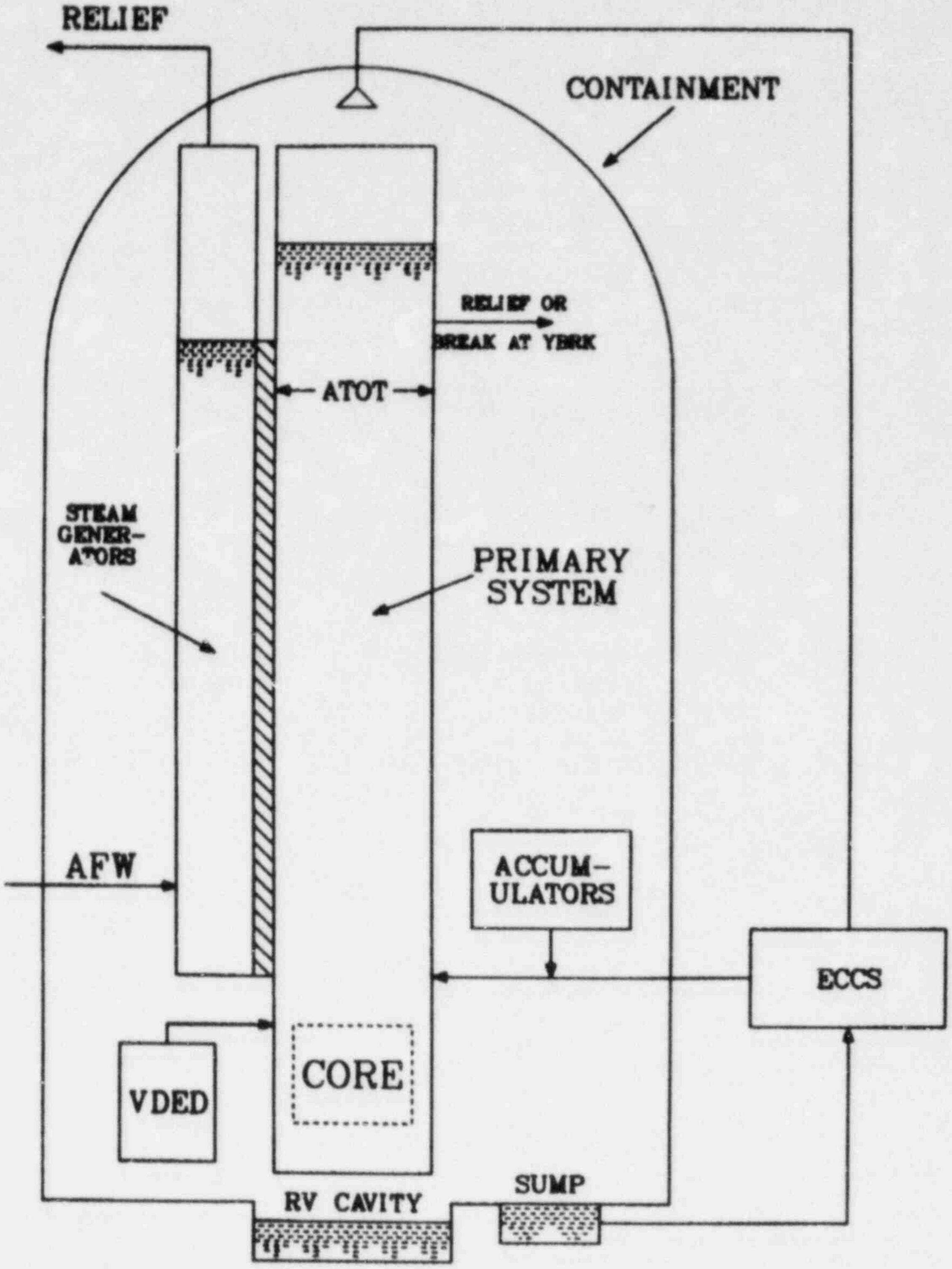


Figure 3.1-1. MARCH Conceptual Nodalization for Zion 1

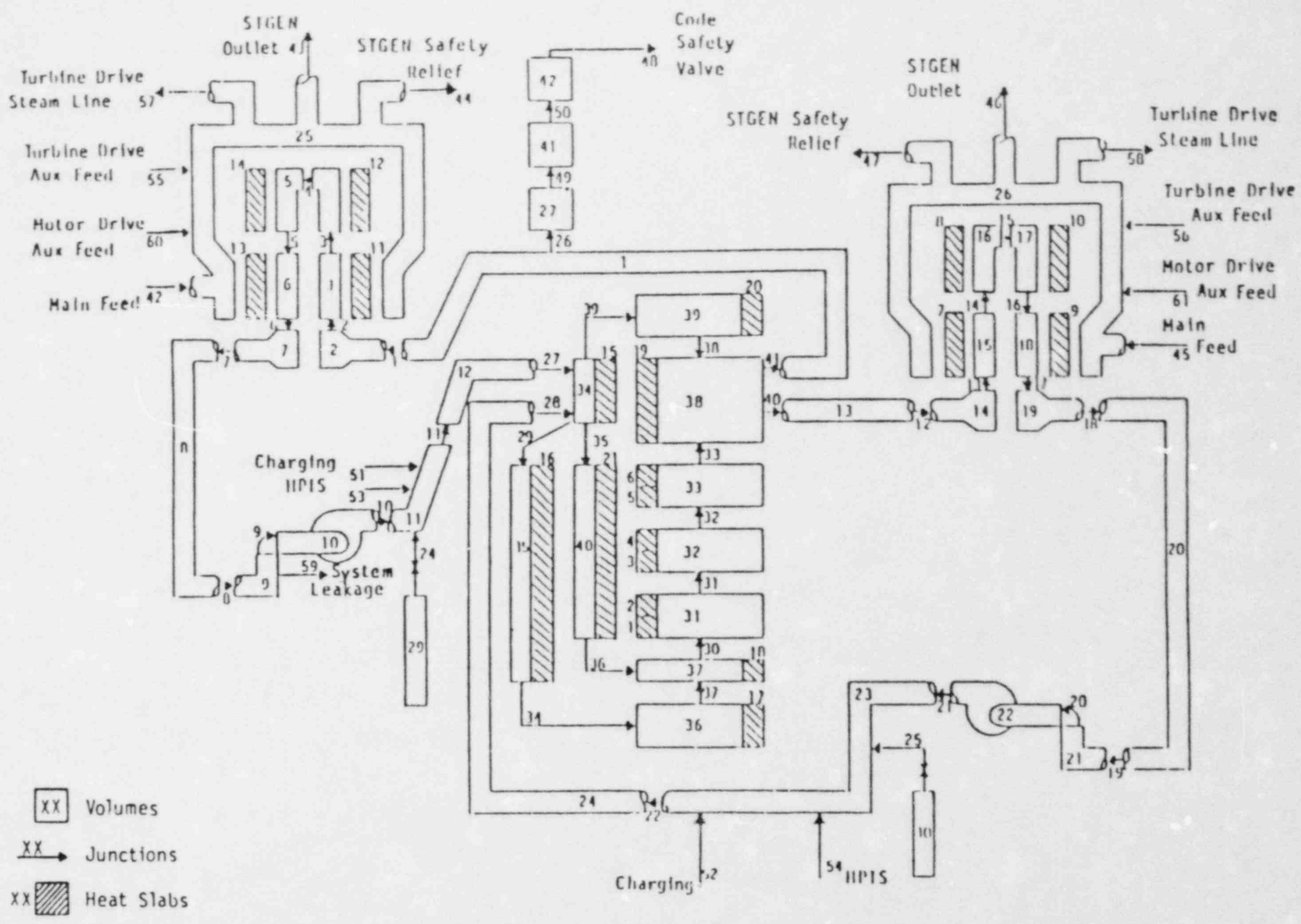


Figure 3.1-2. RELAP4 Nodalization for Zion 1

[WASH-1400, 1975]. The CORRAL models have been discussed elsewhere [NUREG-0205, 1977]. Other codes for calculating the transport and removal of fission products in containment are NAUA [Bunz, 1979], MATADOR [Baybutt, 1981], and CONTAIN [Senglaub et al., 1981]. These codes are more mechanistic than CORRAL. However, CORRAL has been widely used in the nuclear energy community, whereas NAUA, MATADOR, and CONTAIN are not yet fully developed. The version of CORRAL used in this study [Burian and Cybulskis, 1977] has been somewhat generalized from the version described in the RSS [WASH-1400, 1975]. Uncertainties in the CORRAL code, including the current controversy over radionuclide chemistry, are discussed in Section 6.

Figure 3.2-1 shows the containment model used for CORRAL. The solid lines indicate flow paths for all sequences except interfacing system LOCAs; dashed lines indicate flow through the auxiliary building for interfacing system LOCAs. The compartment atmospheres were considered well mixed within containment. In the absence of forced circulation by fan coolers, intercompartmental flow rates were assumed to provide 10 volume changes per hour for the smaller compartment connected by a given path [WASH-1400, 1975]. Double arrows on a flow path indicate that flow rates between the connected compartments are equal. Flow rates to the auxiliary building and to the environment depend on the size of the opening and the magnitude of the driving pressure.

Containment parameters -- pressure, temperature, water vapor mole fractions -- for the accident sequences of interest were taken from MARCH code results.

Fission product release fractions, iodine partition coefficients, and aerosol particle diameters were taken from Appendix VII of the RSS [WASH-1400, 1975].

Dispersion and transport of radioactive material after release from containment are calculated with the CRAC2 code [CRAC2]. CRAC2 divides the area surrounding the reactor into 16 angular segments and 34 concentric radial intervals. Site-specific demographic and meteorological data were used.

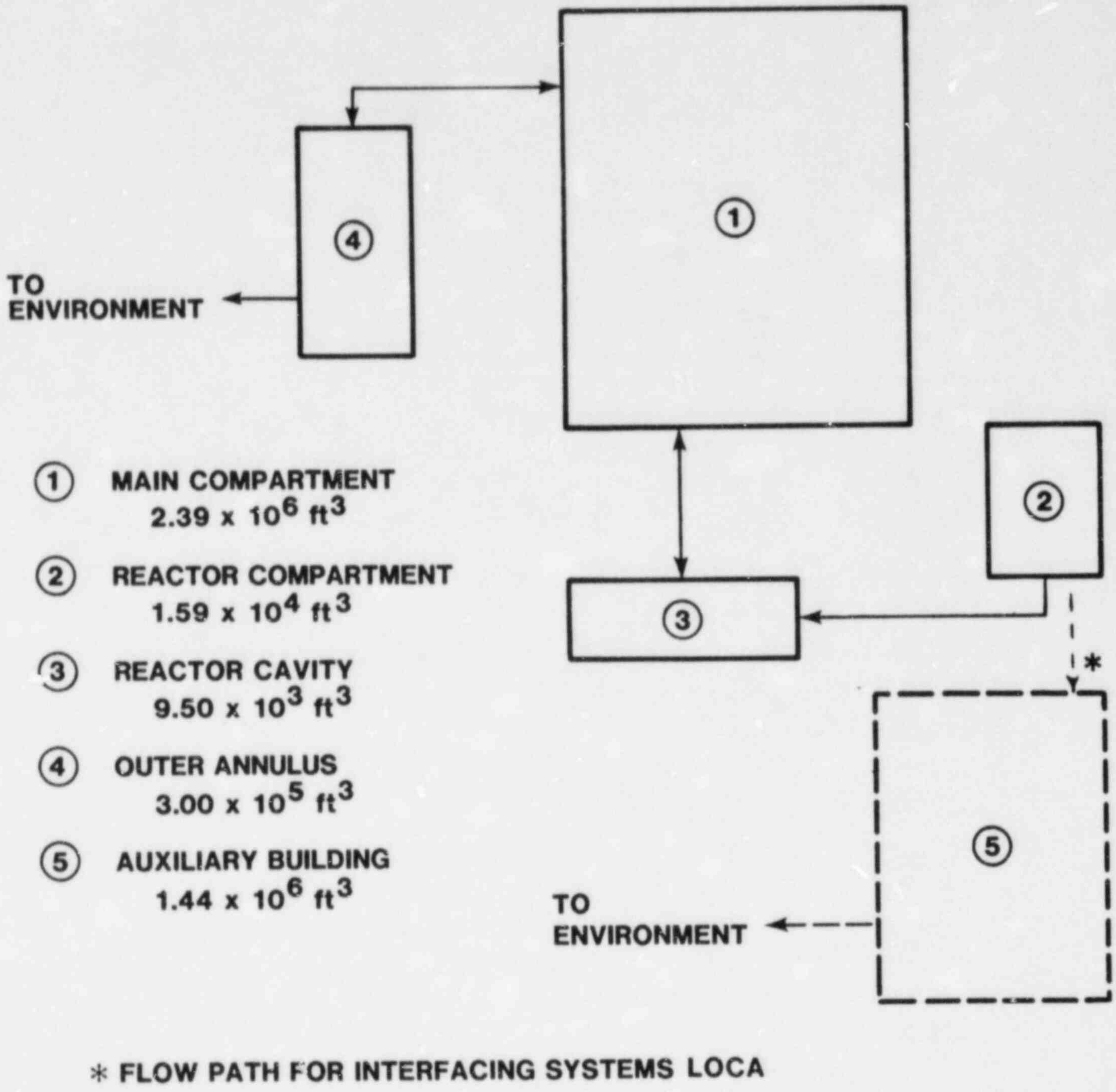


Figure 3.2-1. Containment Model

4. PLANT PARAMETER BEHAVIOR DURING POTENTIALLY SEVERE ACCIDENTS

As discussed in Section 2.3, accident sequences with the highest radiological consequences are those which involve both core degradation and above-ground failure of containment. Such accident sequences contribute to PWR release categories 1 through 5 as defined in the Reactor Safety Study [WASH-1400, 1975]. For the PWR analyzed in the Reactor Safety Study, the accident sequences which are dominant contributors to these release categories include ACD, AD, S1CD, S1H, S2C, S2D, S2DG, S2H, TMLB', and V.

Zion plant parameter responses during a number of potentially severe accident sequences are discussed in this section. All of the sequences from the above list are covered. Two sequences, TMLB' and S2H, are described in detail in Sections 4.1.2 and 4.2.1, respectively. RELAP4 results describing the initial stages of these two sequences were available, and a detailed treatment was deemed appropriate in order to establish the degree of approximation involved with using the MARCH computer code exclusively for other sequences.

The parameter responses presented in this section and the corresponding radiological consequences presented in Section 5 are based on best-estimate assumptions and input parameters. Examples of such best estimates include: realistic containment failure pressure as discussed in Appendix B, no containment failure due to in-vessel steam explosions [Swenson and Corradini, 1981], hydrogen deflagrations initiated at 10 mole percent hydrogen with no hydrogen detonations, and radiological releases adjusted for removal of fission products by the fan coolers as discussed in Section 6.3.2. A summary of MARCH input parameters is provided in Appendix F. Uncertainties regarding input data, modeling, and physical phenomena are discussed in Section 6.

The sequences discussed in this section involve few and in most cases no operator actions. Possible mitigating operator actions and their effects on plant parameter responses and radiological consequences are discussed in Sections 8 and 9, respectively.

4.1 Transient-Initiated Accidents

This section discusses the plant response to two accidents initiated by transients, TML and TMLB'. Transients followed by failure to scram are not treated in this report because they are not dominant contributors to risk for PWRs [WASH-1400, 1975] [Carlson et al., 1981], [Kolb et al., 1981].

In transient-initiated accidents, there is no break in the primary system pressure boundary, and core decay heat can be removed via the steam generators or by using the ECCS in a "feed and bleed" mode of operation. Both of the accidents discussed below, TML and TMLB', involve loss of heat removal capability via the steam generators. The TML accident is only briefly discussed because the ECCS are still available in this accident so that core uncovering can be prevented. The TMLB' accident is treated in considerable detail. Loss of both ECCS and steam generator heat removal capability are involved in the TMLB' accident. Termination of transient-initiated accidents by manually actuating or restoring ECCS or steam generator heat removal capability is discussed in Sections 8.1 and 8.2.

4.1.1 TML

The TML accident is initiated by a transient (T) and involves the total loss of main (M) and auxiliary (L) feedwater to the steam generators. Plant parameter responses during TML acccidents at Zion have been investigated by Los Alamos National Laboratory [DeMuth, 1981] with and without operator actions and compounding equipment failures.

Although there is no feedwater to the steam generator secondaries in the TML accident, core decay heat is removed via the steam generators until all of the initial secondary inventory is evaporated and discharged out the atmospheric dump valves (ADVs). Assuming no operator action is taken, this early primary system response to the TML accident would be very similar to that for the TMLB' accident discussed in Section 4.1.2., and steam generator dryout is predicted roughly 100 minutes into the TML accident at Zion.

Following steam generator dryout, the primary system pressure and temperature would rapidly increase, and steam would begin to be discharged from the primary system to containment through the power operated relief valves (PORVs). ECCS would be actuated on high containment pressure (0.129 MPa [4 psig]). The ECCS would feed relatively cold liquid into the primary system where it would mix with existing in-vessel coolant, remove heat from the core, and eventually be bled from the primary system via the PORVs. The high containment pressure and resulting ECCS actuation in the Zion TML accident would occur in time to prevent core damage [DeMuth, 1981].

4.1.2. TMLB'

The TMLB' accident sequence is initiated by a transient (T) and involves the loss of both main (M) and auxiliary (L) feedwater and failure of AC electric power (B') to engineered safety features (ESFs). A plausible TMLB' scenario would begin with a loss of off-site power. The three diesel generators which supply one of the units would then have to fail to start or load resulting in a station blackout -- a total loss of both onsite and offsite AC power -- to one of the two units. Station blackout would disable all ESFs including emergency core cooling, containment sprays and fan coolers, and motor-driven auxiliary feedwater.

Two of the three auxiliary feedwater pumps at each Zion unit are AC motor driven. In the event of a station blackout the third steam-turbine-driven auxiliary feedwater pump should function, permitting removal of decay heat from the primary system via the steam generators. The TMLB' sequence results, however, if turbine-driven auxiliary feedwater also fails.* Decay heat would then be sufficient to raise the primary system pressures to the (code) safety valve setpoint. Primary coolant would then be discharged to the containment in a process eventually leading to core damage.

Figure 4.1.2-1 shows results of two base case Zion TMLB' analyses, one performed by EG&G using RELAP4 [Dearian, 1980a] and the other performed for this report using MARCH. Figure 4.1.2-1 is a composite figure which includes the response of various parameters as functions of time into the accident. Secondary system plots are located in the first column, primary system plots are located in the middle two columns, and containment plots are located in the fourth column. Temperature plots are placed across the fourth (bottom) row, pressure plots are placed across the third row, level/inventory plots are placed on the second row, and miscellaneous plots are placed on the first (top) row. Subsequent parenthetical references to individual plots will indicate row and column; for example, the parenthetical reference (Figure 4.1.2-1 r2c1) corresponds to the reactor vessel liquid inventory plot in Row 2, Column 1 of Figure 4.1.2-1.

Decay heat (Figure 4.1.2-1 r1c1) is initially removed by relieving steam from the steam generators via the main steam safety valves. Both codes predict a rapid increase in secondary pressure (Figure 4.1.2-1 r3c1). The constant secondary relieving pressure specified for the MARCH calculation (8.2 MPa [1190 psia]) closely approximates the secondary relieving pressure for the RELAP4 calculation (Figure 4.1.2-1 r3c1).

Since auxiliary feedwater is not available, the steam generators dry out (Figure 4.1.2-1 r2c1). RELAP4 predicts steam generator dryout at ~ 54 minutes. This compares to the MARCH-predicted dryout at ~100 minutes. The time to steam generator dryout is sensitive to the decay heat level in the TMLB' accident as discussed in Section 6.1.

Following steam generator dryout, residual steam in the steam generators is superheated (Figure 4.1.2-1 r4c1, RELAP4 plot) until it closely approaches the primary coolant temperature of 627 K (670 F). MARCH does not model the superheating of residual secondary steam after steam generator dryout.

*The failure of turbine-driven auxiliary feedwater postulated here is a failure upon demand as opposed to a failure due to loss of DC controls several hours later when the station batteries fail.

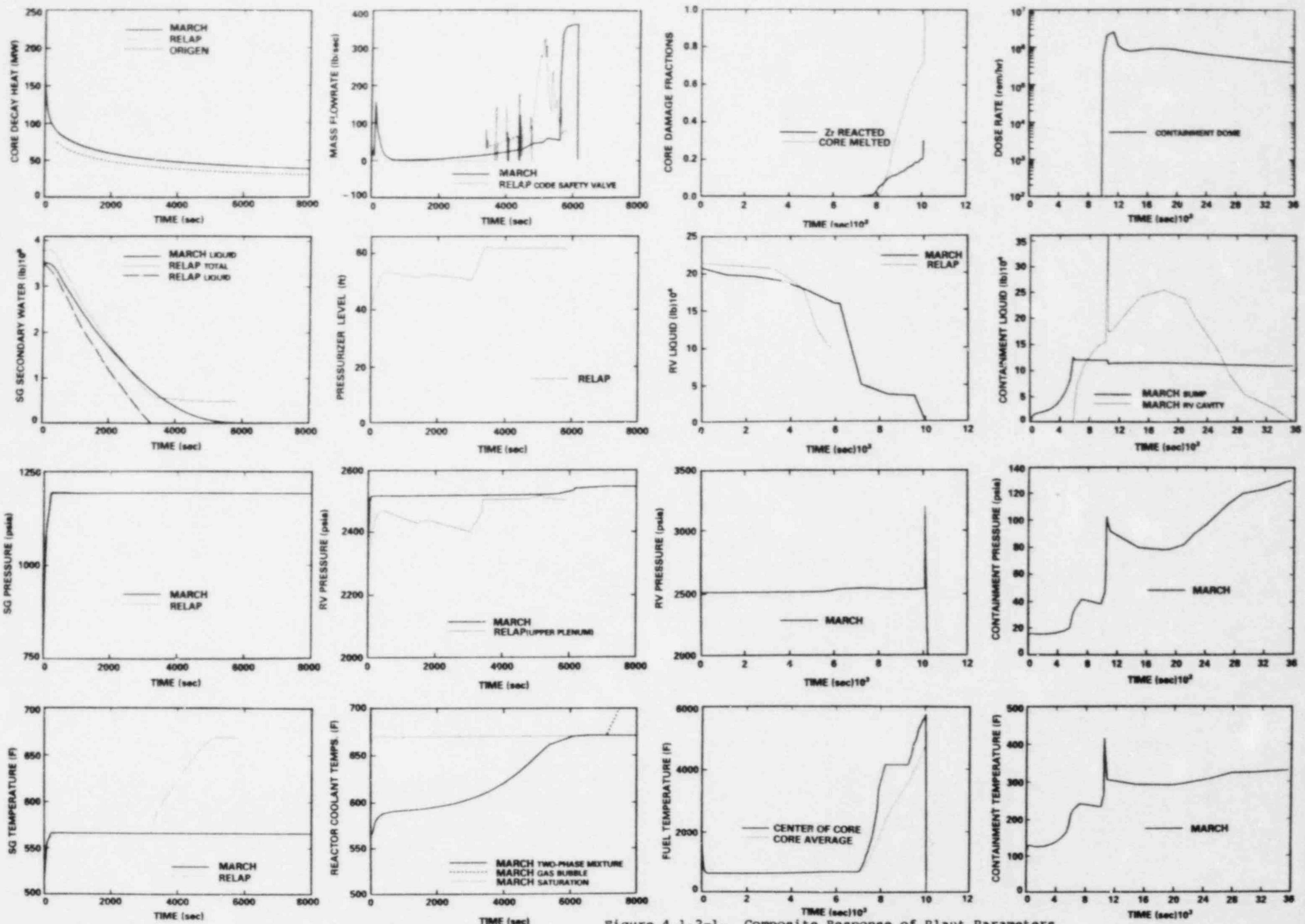


Figure 4.1.2-1. Composite Response of Plant Parameters for Zion TMLB'

Following steam generator dryout, the primary system pressure and temperature as predicted by RELAP4 increase rapidly (Figure 4.1.2-1 r3&4c2), the pressurizer goes solid (Figure 4.1.2-1 r2c2) due to thermal expansion, and the primary pressure increase initiates primary system relief through the safety valves (Figure 4.1.2-1 r1c2). The MARCH and RELAP4 calculations were based on primary system relief at a code safety valve set point of 17.2 MPa (2500 psia). No credit was taken for the power operated relief valves (PORVs). The PORVs are air operated, and control air availability one hour or more into a station blackout cannot be assured. Furthermore, control air would be blocked by containment isolation.

Since MARCH has a single volume model for the primary system, it does not provide an indication of pressurizer level. Also, the pressure and temperature of the primary system as modeled in MARCH only coarsely approximate the behavior predicted by RELAP4.

RELAP4 predicts the formation of a steam bubble in the upper plenum after steam generator dryout. In RELAP4, bubble formation begins at approximately 80 minutes. MARCH does not model in-vessel bubble formation; however, MARCH predicts that the liquid in its single volume model reaches its saturation temperature at approximately 120 minutes (Figure 4.1.2-1 r4c2).

As the bubble in the upper plenum enlarges into the hot-leg region, a full pressurizer can no longer be maintained. Based on a TRAC-P1A calculation for a TMLD sequence [Burns, 1980], the drop in pressurizer level should begin when the total liquid inventory in the primary system is approximately 8.3×10^4 kg (1.83×10^5 lb). When the pressurizer level falls, blowdown from the primary system to containment changes from a two-phase mixture to steam. This blowdown is approximated in MARCH by setting the discharge elevation in the primary system, YBRK (Figure 3.1-1), at the value corresponding to a primary system inventory of 8.3×10^4 kg.

RELAP4 does not treat those stages of the accident involving significant core damage (i.e., beyond initial core uncovering). Consequently, the RELAP4 calculations were terminated at 5800 s. MARCH, on the other hand, treats the entire accident sequence including core melt, vessel breach, and core-concrete interactions. Results of the MARCH calculations during the later stages of the Zion TMLB' base case are included in Figure 4.1.2-1.

Figure 4.1.2-1 r1c3 shows the progression of the $Zr-H_2O$ reaction and core melting. An observable $Zr-H_2O$ reaction precedes initial core melting by approximately 400 s. The MARCH input (Appendix F, FDROP) was specified so core slump into the lower plenum would occur when the fuel temperature at the core center (Figure 4.1.2-1 r4c3) approached 3590 K (6000 F).* At this point, MARCH predicts

*Values for the UO_2 boiling temperature range from 3200 K (5300 F) [Hesson et al., 1971] to 3653 K (6116 F) [Gabelnick and Chasnov, 1972].

that over 70 percent of the core is melted and 20 percent of the Zr has reacted (Figure 4.1.2-1 r1c3). Uncertainties in the MARCH models for core melting and $Zr-H_2O$ reactions are discussed in Section 6.

The extent of the $Zr-H_2O$ reaction after the core slumps into the residual water in the lower head was specified by MARCH input (Appendix F, FDCR) to be approximately 40 percent (Figure 4.1.2-1 r1c3). This kept the average fuel temperature prior to vessel breach below 3590 K. The remaining Zr oxidized in the reactor cavity following failure of the bottom head.

MARCH predicts very rapid rates of Zr-oxidation and evaporation of water when molten material slumps into either the lower head or the reactor cavity. This causes a pressure spike in the primary when the core slumps into the lower plenum (Figure 4.1.2-1 r3c3) and a pressure spike in containment when the bottom head fails dumping molten material into the reactor cavity (Figure 4.1.2-1 r3c4). The validity of MARCH assumptions affecting evaporation rates when molten material slumps into water are discussed in Section 6.3.

The liquid inventories in the containment sump and the reactor cavity are depicted in Figure 4.1.2-1 r2c4. The containment sump was assumed to overflow into the reactor cavity when the water on the containment floor exceeded a volume of approximately 56.6 m^3 (2000 ft^3). As discussed in Section 6.1, overflow into the reactor cavity is conservative with respect to containment pressure as predicted by MARCH since more liquid is evaporated by direct contact with molten core material. Overflow is consistent with the basemat configuration of the Zion containment. The spike in the plot of reactor cavity liquid inventory corresponds to dumping of the accumulators when bottom head failure causes the primary system to depressurize. Some of the water in the reactor cavity is then rapidly evaporated (MARCH subroutine HOTDROP) and the remainder is evaporated more gradually during the prolonged core-concrete interaction (MARCH subroutine INTER).

The pressure and temperature in containment during the Zion TMLB' base case are depicted in Figure 4.1.2-1 r3&4c4. These curves are substantially similar in shape and show the effects of many of the events discussed above.

The initial jumps in containment pressure (from 14.7 psia to 16 psia) and containment temperature (from 110 F to 129 F) result from an early, short blowdown spike associated with an early, rapid rise of primary pressure to the relief valve set point which is predicted by MARCH (Figure 4.1.2-1 r3c2). Such early blowdown is not correct and is not predicted by RELAP4.

Containment pressure and temperature remain substantially constant until steam-generator dryout (100 minutes per MARCH). Containment pressure and temperature rise during the ensuing primary system relief. There is a distinct change in the slope of the containment pressure and temperature curves at 100 minutes when the blowdown changes from a two-phase mixture to steam. When the core is substantially uncovered, the boiloff rate decreases and containment pressure and temperature actually decline slightly until failure of the bottom head. The spikes in containment pressure and temperature at the time of bottom head failure are caused in part by the release of the remaining primary coolant and H_2 inventory and to a larger degree by the rapid evaporation rate (predicted by MARCH subroutine HOTDROP) when melt slumps into the reactor-cavity liquid.

Heat transfer to the passive heat sinks inside containment reduces the containment pressure and temperature thus forming the tails of the pressure and temperature spikes. The subsequent pressure and temperature buildups are caused by the continued evaporation of water from the reactor cavity and the generation of noncondensible gases in the core-concrete interactions. At ~ 10 hours the base case containment pressure is 0.932 MPa (135 psia), slightly exceeding the lower bound for containment failure [0.929 MPa (134.7 psia), see Section 8.3]. The best-estimate failure pressure of 149 psia would not be exceeded for several days, if at all.

4.2 Small LOCAs

This section presents predicted plant parameter responses to accidents initiated by small breaks in the primary system pressure boundary inside containment (S1 and S2 initiating events per Table 2.1). These responses demonstrate that the small-break initiated sequences which were dominant contributors to release categories 1 through 5 in the Reactor Safety Study [WASH-1400, 1975] would not result in gross containment failure due to overpressure or hydrogen burning at Zion because the Zion fan coolers are designed to operate following such breaks. (At the PWR studied in the Reactor Safety Study, the containment fan coolers are turned off on receipt of an ESF actuation signal.) One of the dominant small-break sequences from the Reactor Safety Study, S2C, would not even lead to core damage at Zion. The results presented in this section, which stress containment pressure response, support the following conclusions for Zion:

1. Failure of containment sprays alone following a small break would not lead to core melt. Sufficient water would be available in the containment sump at the end of ECC injection to permit switchover to recirculation.

2. Failure of containment sprays following a small break would not lead to containment failure if the containment fan coolers operate. One fan cooler could prevent containment failure due to overpressure (see Appendix C). Four fan coolers could prevent containment failure due to hydrogen burning provided ignition occurs at hydrogen concentrations less than or equal to 10 mole percent as assumed throughout this report.

3. Successful ECC injection can be prolonged and switchover to ECC recirculation delayed by preserving RWST inventory

   a. by limiting containment spray operation (three spray trains would deplete the RWST in about 45 minutes)

   b. by replenishing RWST inventory from alternative supplies.

4. Failure of both containment spray injection and ECC injection could result in insufficient water in the containment sump to permit switchover to recirculation. Injecting water into containment by alternative means could permit ECC recirculation to be established in time to prevent core degradation.

5. Complete failure of containment heat removal systems (fan coolers and spray recirculation heat exchangers) could lead to core melt followed by containment failure due to overpressure but only after approximately 11 hours.

4.2.1 S2HF and S1HF

The S2HF sequence is a small LOCA followed by successful scram, emergency core cooling (ECC) injection from the refueling water storage tank (RWST), and auxiliary feedwater initiation. ECC recirculation is postulated to fail when switchover to recirculation from the containment sump is necessitated by RWST depletion.

The containment air coolers at Zion are used during normal power operation, and are designed to continue operating* during a LOCA (as sensed by high containment pressure, > 4 psig). Independent failures of containment air coolers are not postulated in the S2HF sequence. Containment sprays would initially be available in an S2HF sequence at Zion but they would not be actuated automatically because blowdown from the primary system would be insufficient to cause high-high containment pressure (23 psig) with the air coolers operating. Recirculation failure is assumed to disable the containment sprays in addition to ECC at Zion because recirculation flow for both functions is taken from the containment sump via the low pressure injection (LPI) pumps. Therefore, containment sprays are not a factor in the S2HF results presented herein.

Figure 4.2.1-1 compares three accident calculations for an S2HF sequence initiated by a 3.81-cm (1.5-inch) diameter, cold-leg break at Zion. RELAP4 calculation results [Dearian, 1980b] are shown as

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*The fan coolers operate in a cooling mode (high speed) in normal operation and in a steam condensing mode (low speed) on ESF actuation.

solid lines. MARCH calculations are shown as chain-dot lines. Dashed lines are used to show the extension of RELAP4 calculations using MARCH. This latter RELAP4-MARCH calculation was performed by forcing the initiation stage of the calculation using primary system blowdown data from the RELAP4 output as input to MARCH subroutine INITIAL. The MARCH and RELAP4-MARCH results shown in Figure 4.2.1-1 extend through ~ 73 percent core melting.

Figure 4.2.1-1 includes plots of RWST, accumulator, and containment inventories across the top row. Liquid inventories in the steam generators and the reactor vessel are shown on the second row along with the total energy released to containment with the primary system blowdown. The bottom row shows the pressure responses in the secondary system, the primary system, and the containment.

The rapid decrease in primary system pressure from 2250 psia (Figure 4.2.1-1 r3c2) initiates several automatic actions. In the RELAP4 calculation, scram occurs at 90 seconds, a safety actuation signal is generated at 93 seconds, emergency core cooling is inititiated at 98 seconds, steam generator secondaries are fully isolated at 94 seconds, main feedwater is terminated at 100 seconds and motor-driven auxiliary feedwater is initiated at 150 seconds.

RELAP4 results indicate that the secondary pressure initially rises (Figure 4.2.1-1 r3c1) as a result of secondary isolation. At 1050 seconds, RELAP4 predicts primary and secondary fluid temperatures cross and the secondaries begin acting as a heat source until these temperatures reverse again at 16000 seconds. RELAP4 predicts an initial decrease in the steam generator mixture level with little change in inventory as the bubbly froth of normal operation collapses. The secondary inventory increases slightly as a result of auxiliary feedwater flow because liquid replaces froth until ~ 750 seconds when normal level is attained. Auxiliary feedwater is controlled in the RELAP4 calculation to be "full on" or "full off" depending on whether normal secondary level is established. RELAP4 predicts a gradual increase (until ~ 16000 seconds) in the secondary inventory required to maintain normal level due to thermal contraction as the secondary cools. In MARCH, the initial and maximum steam generator inventories are input-specified constants WTRSG and FULSG, respectively. Since MARCH uses a maximum secondary inventory instead of a maximum secondary level, MARCH does not predict an increase in secondary inventory as the secondary cools.

ECC is first initiated via the two centrifugal charging pumps (CCPs). The CCPs have a shutoff head which exceeds normal primary system operating pressure. If the primary system pressure falls below 1520 psia, injection via the high pressure injection (HPI) pumps is initiated. The two CCPs can provide up to 844 gpm, and the two HPI pumps can provide up to 952 gpm; however, the actual flow will depend on primary system pressure as depicted in Figure 4.2.1-2 [based on data from Fletcher, 1981].

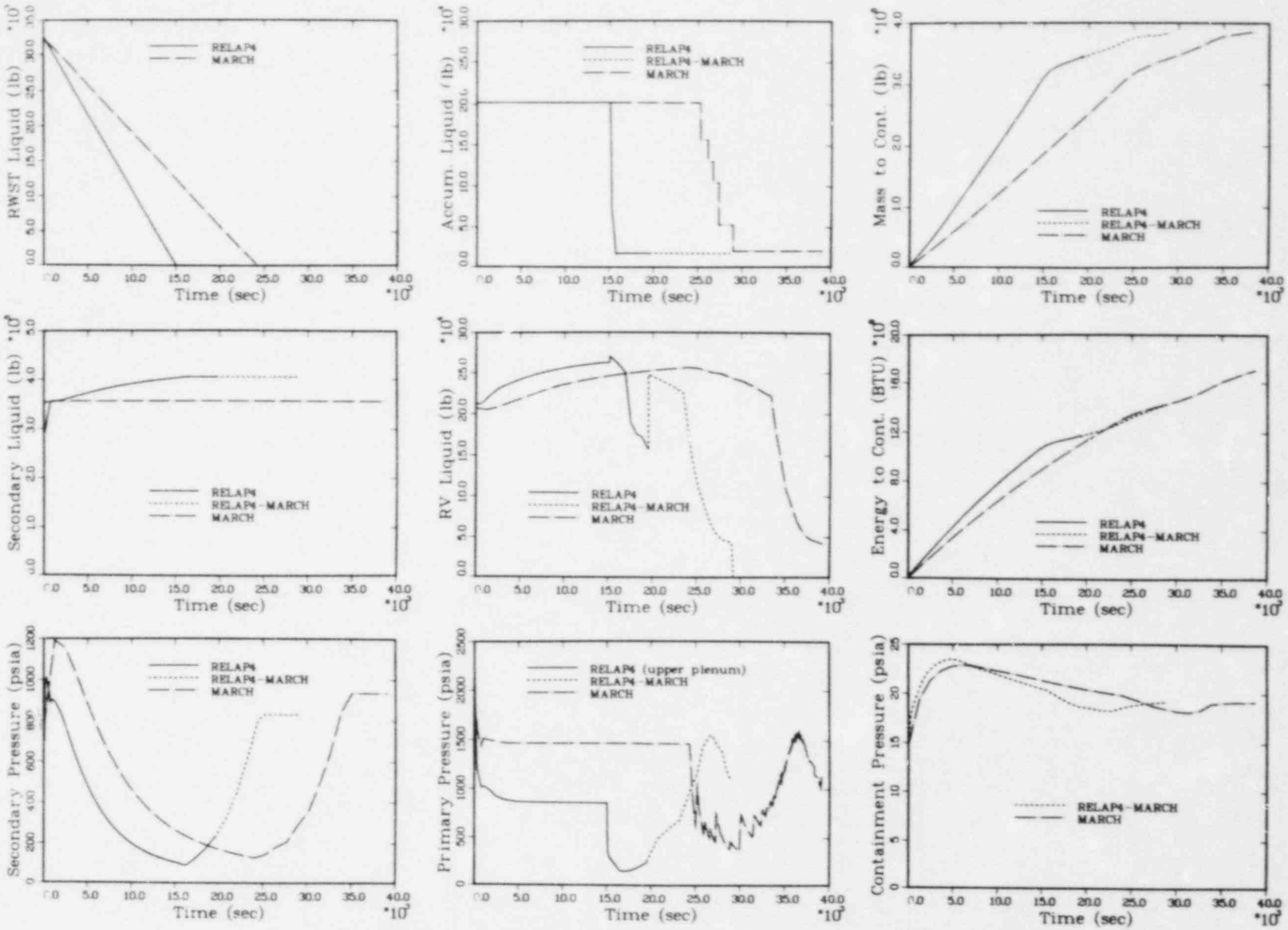

Figure 4.2.1-1. Composite Response of Plant Parameters for Zion S2HF (1.5-Inch Diameter Break).

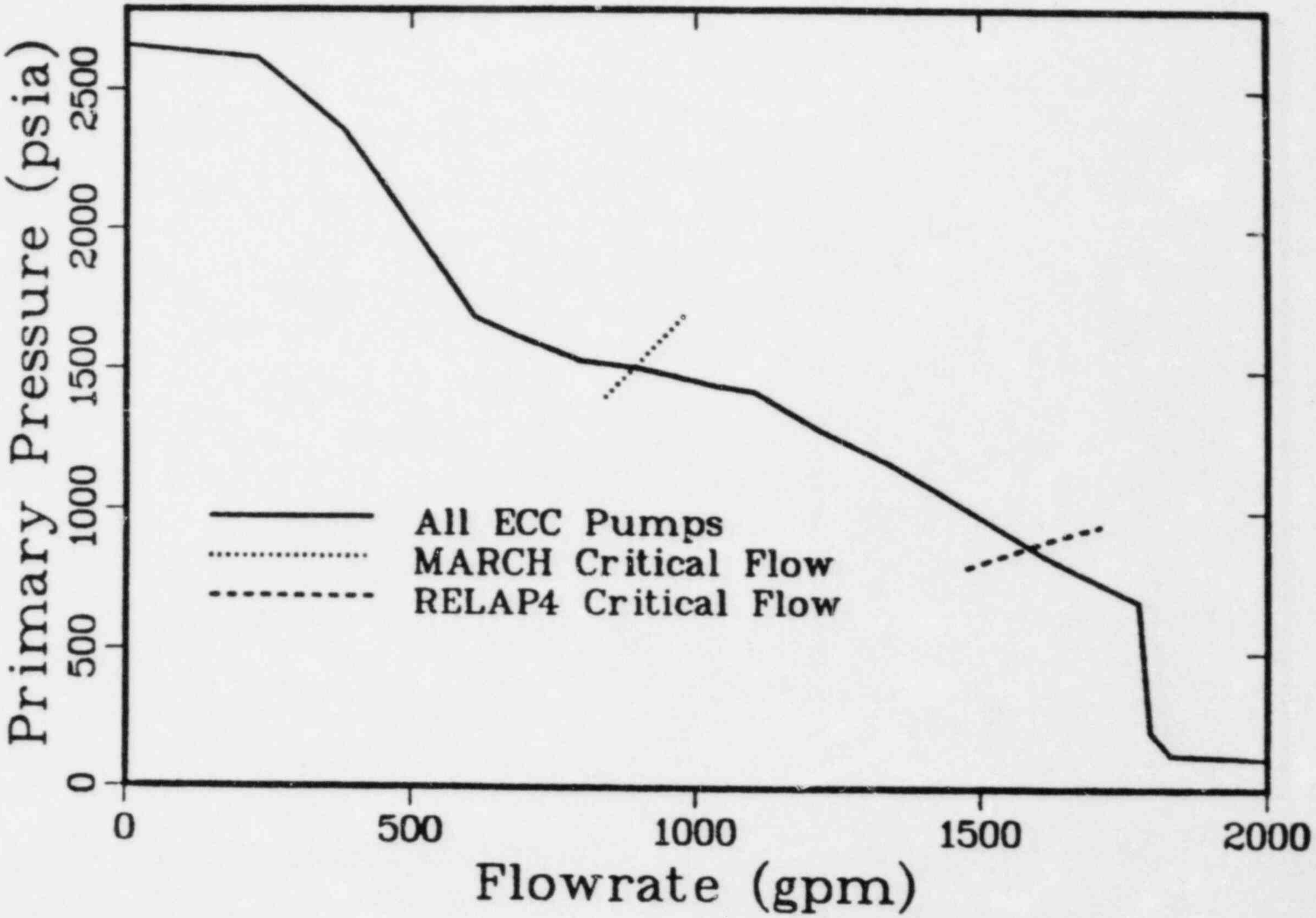

Figure 4.2.1-2. Pump Flow Curve Used for S2H Calculations Showing Differences Between MARCH and RELAP4 Critical Liquid Flows Through 1.5-Inch Diameter Break

The RWST and containment inventory plots (Figure 4.2.1-1 r1c1&3) show that RELAP4 predicts substantially higher blowdown from the break than does MARCH. RELAP4 also predicts a substantially lower primary system pressure (Figure 4.2.1-1 r3c2). The MARCH blowdown flow rate for liquid from the primary system is based on a very simple critical-flow approximation [Wooton and Avci, 1980; Eq. III.A 56]. Figure 4.2.1-2 shows both the MARCH critical-flow curve and the RELAP4 critical-flow curve. The intersection of the RELAP4 critical-flow curve with the ECC pump curve is far removed from the intersection of the MARCH critical-flow curve with the ECC pump curve. The simple critical-flow model used in MARCH for transients and small breaks disagrees with the more sophisticated RELAP4 models. Verification of RELAP4 critical-flow models is discussed elsewhere [Aerojet Nuclear, 1976]. In the S2HF case shown, this difference results in a nonconservative (with respect to RELAP4) delay in the MARCH predicted RWST depletion time of 2.5 hours.

Following depletion of the RWST (at 15000 s for RELAP4 or 24000 s for MARCH), the S2HF sequence postulates failure of all recirculation from the containment sump. As a result, flows from the CCPs and the HPI pumps are terminated. Low-pressure injection is not available.

Upon termination of charging and HPI flows, the primary pressure (Figure 4.2.1-1 r3c2) falls sharply causing accumulator injection to begin (Figure 4.2.1-1 r1c2). When accumulator injection is completed the primary system begins to repressurize (Figure 4.2.1-1 r3c2).

Some RELAP4 parameter predictions are beyond the capability of the simple MARCH models. The following parameter responses could be important to operating personnel and, indirectly, to authorities contemplating emergency actions designed to protect the public. Following accumulator depletion, RELAP4 predicts the hot leg temperature increases above the secondary temperature at 16000 s causing the steam generators to again act as heat sinks. Heat addition to the secondaries after 16000 s causes the secondary level to swell slightly due to thermal expansion of the secondary fluid. Loop flows dramatically increase after 16000 s as natural circulation is re-established. The pressurizer level increases to the top following accumulator flow initiation, then drops approximately 30 feet, and again recovers as a steam bubble is formed in the upper head and plenum. The bubble causes an increase in hot leg quality at 17400 s.

The RELAP4 calculation was terminated at 19500 s with the pressurizer full and the upper plenum level 4.5 ft above the top of the active core and decreasing. The discontinuity in the RELAP4-MARCH in-vessel liquid inventory at this time (Figure 4.2.1-1 r2c2) occurs because MARCH does not model a bubble in the vessel or flow from the pressurizer and steam generators when the bubble enlarges sufficiently to uncover the vessel nozzles.

The RELAP4-MARCH calculation gives a higher initial containment pressure rise than the MARCH calculation because of the more rapid blowdown predicted by RELAP4. It is apparent from Figure 4.2.1-1 that the timing of the key events is strongly dependent on the predicted blowdown rate. For slightly larger breaks, the potential for actuating containment sprays on high-high (> 23 psig) containment pressure and thereby accelerating RWST depletion also requires accurate blowdown data. Assuming that the models in RELAP4 are fairly realistic, the comparisons made in Figure 4.2.1-1 and 4.2.1-2 demonstrate that MARCH can be nonconservative with respect to primary system blowdown and event times for small LOCAs.

Figure 4.2.1-3 is a continuation of the RELAP4-MARCH calculation from Figure 4.2.1-1. Figure 4.2.1-3 shows the MARCH-predicted containment response for times both before and after vessel breach. Note that because the fan coolers operate throughout the S2HF sequence, containment pressures are sufficiently low that containment failure due to hydrogen burning is precluded.* For the reasons stated above, MARCH as a stand-alone code predicts significantly later event times, but such later event times would not change the conclusion that above-ground containment failure is unlikely for the S2HF sequence.

Figure 4.2.1-4 shows the MARCH-predicted accident event times for S1HF and S2HF sequences as a function of break diameter. Note that there is a discontinuity in the rate of decrease in the event times between 5- and 6-inch breaks. For the 6-inch break, the blowdown is sufficiently rapid that high-high containment pressure (23 psig) is exceeded. Containment sprays are actuated on high-high containment pressure, and the effects of such actuation on event times are apparent in Figure 4.2.1-4. Shutting off the containment sprays following a LOCA could delay RWST depletion by as much as 45 minutes based on three-spray train flowrate of 7800 gpm. The discussion of the S1C and S2C sequences in Section 4.2.2 demonstrates that containment sprays are not required to protect the Zion containment from failure due to overpressure when the fan coolers are operating. Although the MARCH-predicted event times shown for the smaller break sizes in Figure 4.2.1-4 may be nonconservative due to the MARCH critical-flow model discussed above, they do illustrate the acceleration associated with increasing break size.

4.2.2 S1C and S2C

The S1C and S2C sequences are small LOCAs followed by failure of containment spray injection. At the PWR investigated in the Reactor Safety Study [WASH-1400, 1975], the containment fan coolers are switched completely off upon receipt of an ESF actuation signal,

*Based on complete combustion following ignition at 10 mole percent hydrogen. See also Section 8.3.

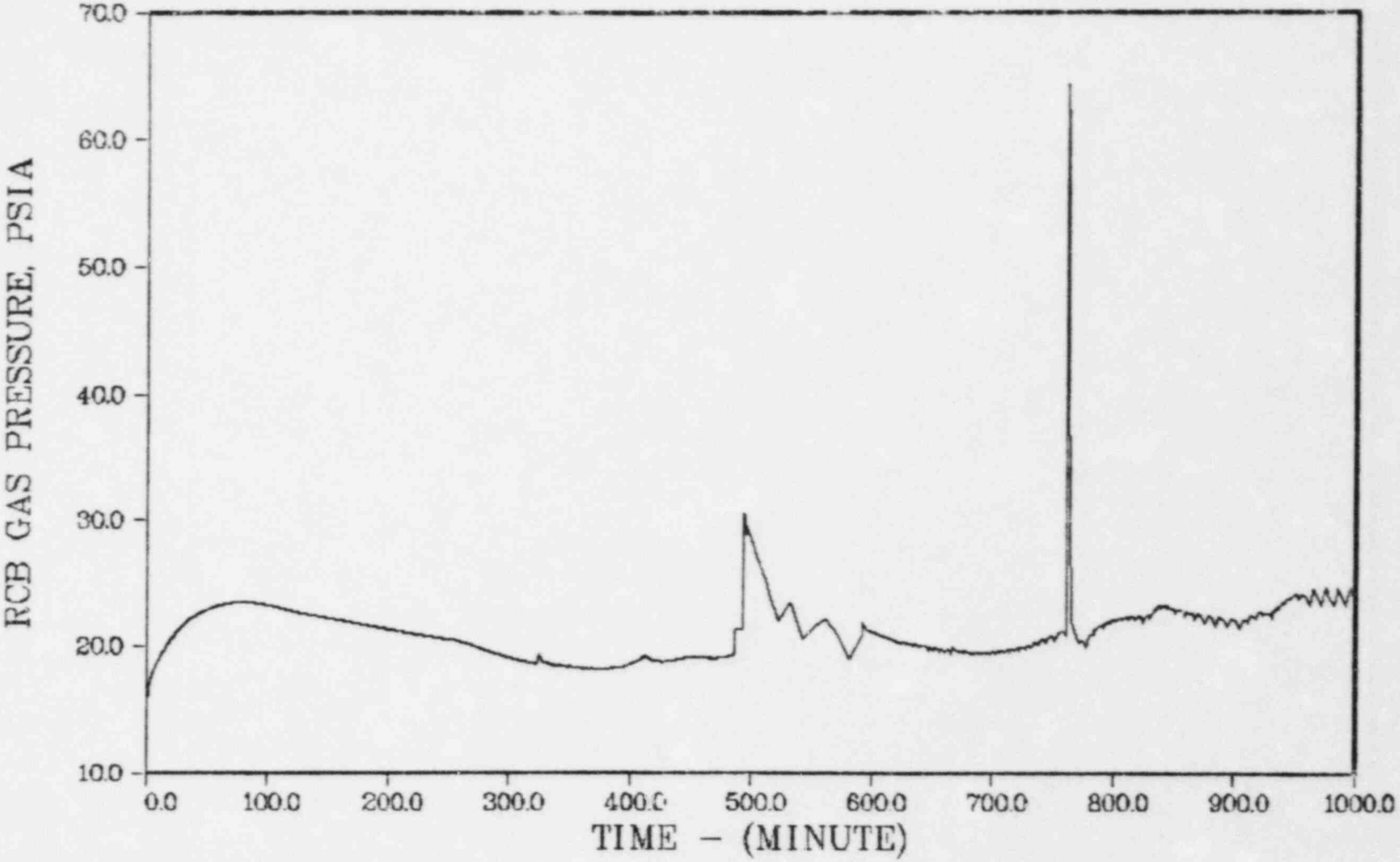

Figure 4.2.1-3. Zion S2HF (1.5-Inch Diameter Break)
Containment Pressure Versus Time,
RELAP4-MARCH Calculation

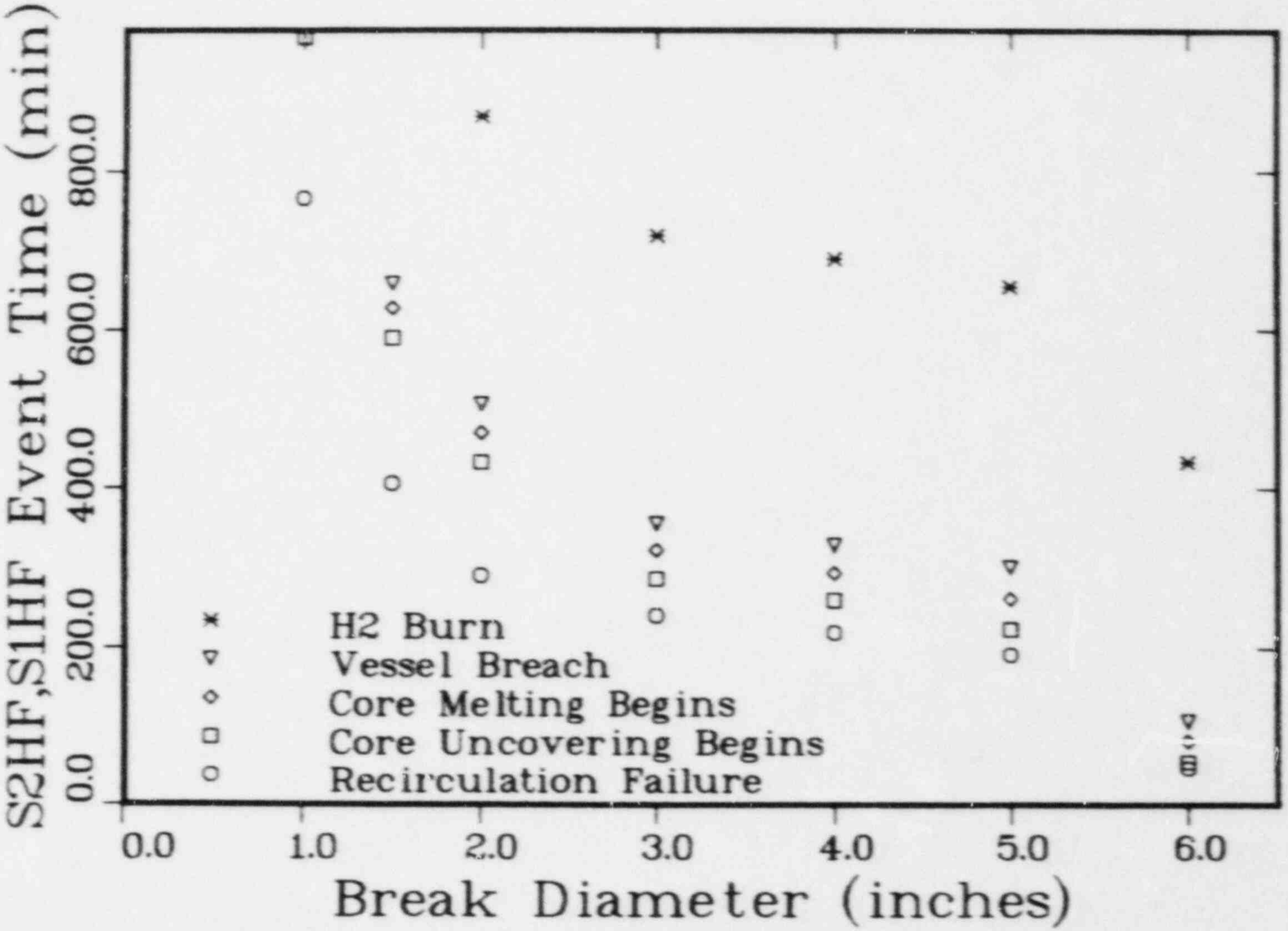

Figure 4.2.1-4. Zion S2HF and S1HF Accident Event Times Versus Break Size, MARCH 1.1 Calculations

leaving only the containment sprays to protect the containment against failure due to overpressure. Failure of containment spray injection (Event C) following an S2 break can then lead to spray recirculation failure due to inadequate water in the containment sump. Containment failure follows, resulting in flashing of water from the containment sump and ECC recirculation failure.

At Zion, in contrast, failure of containment spray injection (Event C) alone would not lead to containment failure or core damage. The containment fan coolers at Zion automatically switch from normal to accident mode (reduced speed) of operation upon receipt of an ESF actuation signal. There is no reasonable likelihood of losing ECC recirculation with the fan coolers operating due to failure of containment sprays. Condensate running from the fan coolers to the containment sump would be highly subcooled. In fact, with four fan coolers operating as designed for Zion, high-high containment pressure (0.26 MPa [23 psig]) would not be reached for S2 sequences or for S1 sequences initiated by breaks up to 5 inches in diameter.* In the absence of high-high containment pressure the containment sprays would not be actuated automatically.

Figure 4.2.2-1 shows the S1C containment pressure response for a larger (15.24-cm [6-inch] diameter) break. High-high containment pressure is reached but the containment sprays fail to operate; however, considering the 0.93 MPa (134.7 psia) threshold for containment failure at Zion (see Appendix B), Figure 4.2.2-1 clearly shows that four fan coolers are adequate to prevent containment failure due to overpressure. In fact, calculations in Appendix C demonstrate that containment failure due to overpressure can be prevented at Zion with only one fan cooler.

4.2.3 S1G, S2G, S1CG, and S2CG

The S1G and S2G sequences are small LOCAs followed by failures of containment heat removal systems (G). At Zion, heat can be removed from the containment building using either the containment fan coolers or the RHR heat exchangers in the ECC recirculation system. Failure of both (G) could occur, for example, due to a total loss of service water or emergency component cooling water. Without containment heat removal, water in the containment sump would heat up and ECC recirculation failure could result (for example due to recirculation pump or pump-motor overheating). Figure 4.2.3-1 shows the MARCH-predicted containment pressure response for two S1G sequences initiated by 15.24-cm (6-inch) diameter breaks. In one sequence, no recirculation failure was postulated and meltdown prior to containment failure was prevented by recirculating hot water from the containment sump to the reactor vessel. In the second case, ECC recirculation failure was assumed to occur approximately 10 minutes

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*There is a coupling between break size and the minimum number of fan coolers required to avoid high-high containment pressure. A single fan cooler is adequate for very small breaks.

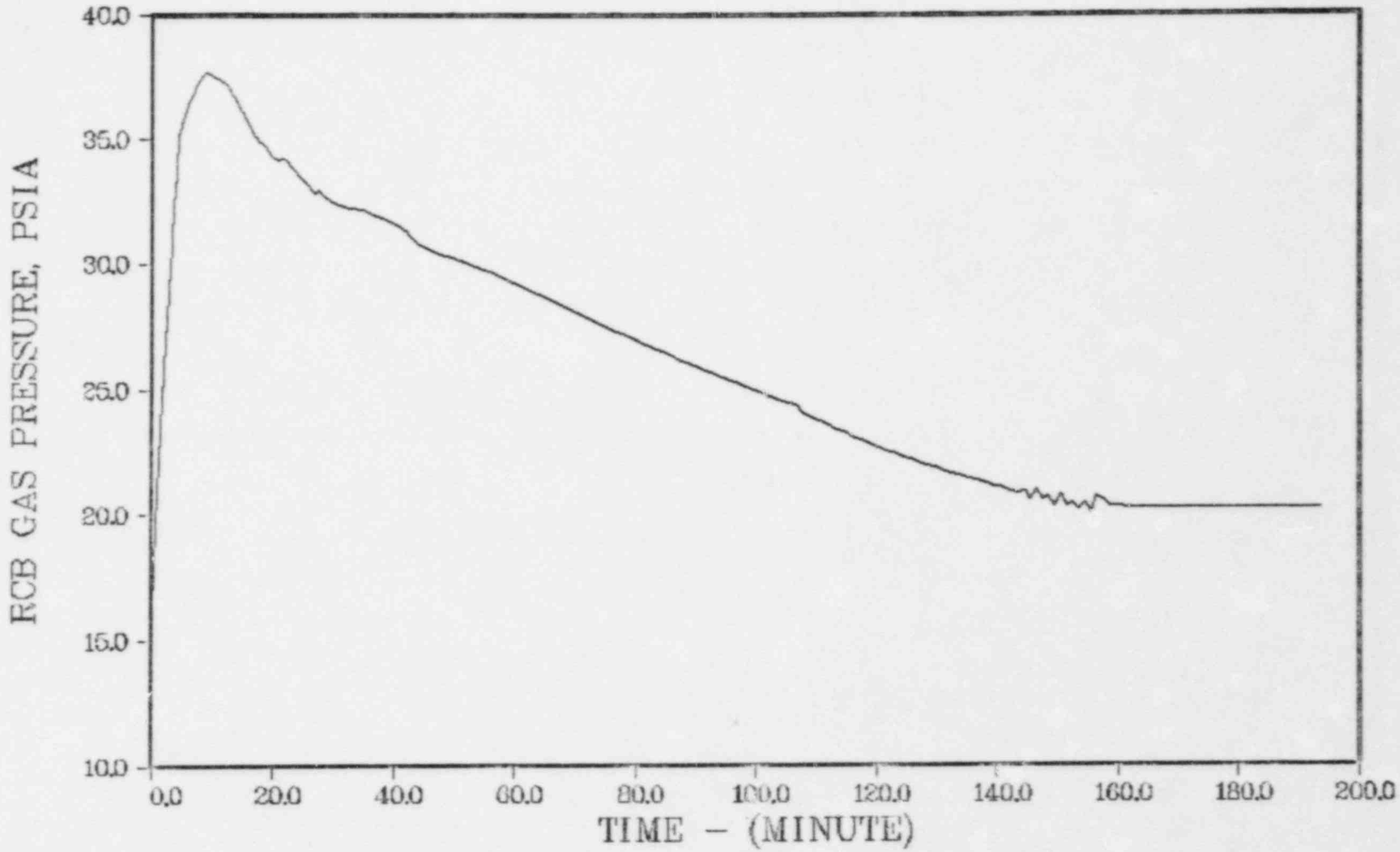

Figure 4.2.2-1. Zion S1C (6 Inch Diameter Break) Containment Pressure Response, March 1.1 Calculations

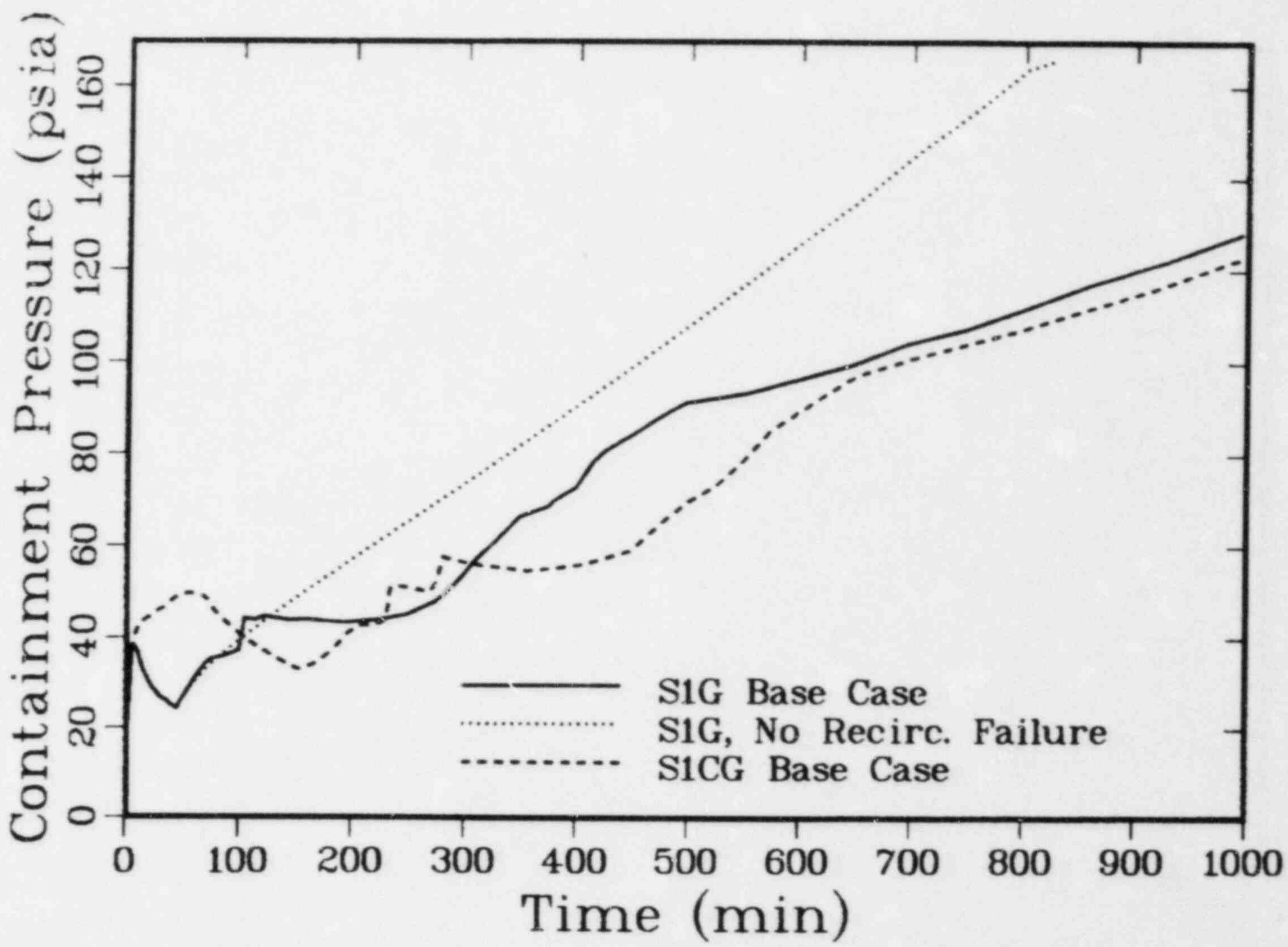

Figure 4.2.3-1. Zion S1G and S1CG (6-Inch Diameter Break) Containment Pressure Responses, MARCH 1.1 Calculations

after switchover to ECC recirculation (i.e., 10 minutes after the depletion of water in the RWST caused a cessation of ECC injection). At this time the predicted temperature of the water in the containment sump reached ~367 K (~200F). Figure 4.2.3-1 shows that unless containment heat removal is restored, containment failure due to overpressure will occur. The lower bound for containment failure is 0.93 MPa (134.7 psia), see Appendix B. Containment failure is predicted earlier (at approximately 660 minutes for 134.7 psia) for the case in which recirculation failure is not postulated. In this case primary coolant is continuously heated by the core, discharged through the break to containment, condensed, and recirculated from the sump. Flashing of the remaining sump water upon containment failure could lead to core meltdown for this case. The other case, in which recirculation is postulated to fail after 10 minutes, is more likely. In this case, melting is predicted to begin at about 85 minutes, bottom head failure is predicted at about 120 minutes, but containment failure is not predicted before 1100 minutes.

The S1CG and S2CG sequences are small LOCAs followed by failures of containment spray injection (C) and containment heat removal systems (G). As in the S1G and S2G sequences, recirculation failure in the S1CG and S2CG sequences would eventually occur due to the heatup of water in the containment sump. Figure 4.2.3-1 includes the predicted containment pressure response for a S1CG sequence initiated by a 15.24 cm (6 inch) diameter break with recirculation again postulated to fail after 10 minutes. Figure 4.2.3-1 shows that the S1G sequence proceeds more rapidly than the corresponding S1CG sequence. The containment sprays deliver 0.49 m^3/s (7800 gpm) and consequently accelerate RWST depletion by approximately 45 minutes compared to the S1CG sequence. Figure 4.2.3-1 illustrates the advantage associated with minimizing the use of containment sprays to preserve RWST inventory in accident sequences in which ECC injection is operable.

4.2.4 S1D, S2D, S1CD, S2CD

The S1D, S2D, S1CD, and S2CD sequences are small LOCAs followed by failure of ECC injection (D) or containment spray injection and ECC injection (CD). For all S2 breaks and for sufficiently small S1 breaks at Zion, D and CD sequences are equivalent since for such small breaks the containment fan coolers keep the containment pressure below the 0.26 MPa (23 psig) setpoint for automatic actuation of containment sprays.

The D and CD sequence results presented in this section assume, as does the Reactor Safety Study [WASH-1400, 1975], that injection failures (C and D) lead to corresponding recirculation failures (F and H). This is a likely result for several reasons. The respective injection and recirculation systems share the same pumps and much of the same piping. The systems differ primarily in their sources of water (RWST for injection and containment sump for recirculation). Also, emergency operating procedures specify manual switchover to recirculation only upon reaching a low water level in the RWST.

With no ECC injection (D), if containment sprays also fail or are not actuated, low RWST level would not be reached. The only water added to the containment under such circumstances would be the result of primary system blowdown to containment. This could, depending on the time and the break size, be insufficient for establishing or maintaining ECC recirculation.

In spite of the above logic, there could be injection failures which leave ECC recirculation operable, in particular, failures associated with the RWST. In such cases, if sufficient water levels exist or can be established in the containment sump, switchover to recirculation in time to prevent core damage could be feasible (see Section 8.1.2).

Figure 4.2.4-1 shows the MARCH-predicted containment pressure response for S1D and S1CD sequences initiated by a 15.24 cm (6 inch) diameter break. Hydrogen burns are predicted in these sequences as in the S2HF and S1HF sequences discussed in Section 4.2.1. Again, however, the peak pressures associated with such hydrogen burns (specified to be from 10 to 0 mole percent hydrogen) are well below the threshold for failure of the Zion containment (see Appendix B). Note that, although the time to vessel breach is essentially the same for both the S1D and S1CD sequences, the sprays in the S1D sequence substantially lower the containment pressure and delay the onset of core-concrete interactions by providing additional water for quenching debris in the reactor cavity. For the input parameters selected for these baseline runs, hydrogen from core-concrete interactions is necessary for 10 mole percent H_2 to be reached in containment. Consequently, spray operation in the S1D sequence delays the H_2 burn with respect to that for the S1CD sequence.

Figure 4.2.4-2 shows the MARCH-predicted event times as a function of break size for the S2D and S1D sequences. For the reasons discussed in Section 4.2.1, these MARCH-predicted event times may be nonconservative especially for the smaller diameter breaks; however, they illustrate the correct trends. For large enough S1 breaks, high-high containment pressure is reached and automatic actuation of containment spray injection is attempted. Failure or success of containment spray injection (S1CD versus S1D) would not affect the timing depicted in Figure 4.2.4-2, except for a possible delay in hydrogen burning as discussed above. As discussed in Section 8.3.4, the containment sprays could, if available, be used to decrease the pressure rises associated with hydrogen burns; however, as noted above, even without containment sprays containment failure due to hydrogen burning is not predicted.

4.2.5 S1DG, S2DG, S1CDG, and S2CDG

The S1DG and S2DG sequences are small LOCAs followed by failures of ECC injection (D) and containment heat removal systems (G), including both the containment fan coolers and the containment spray recirculation heat exchanger. Containment spray injection

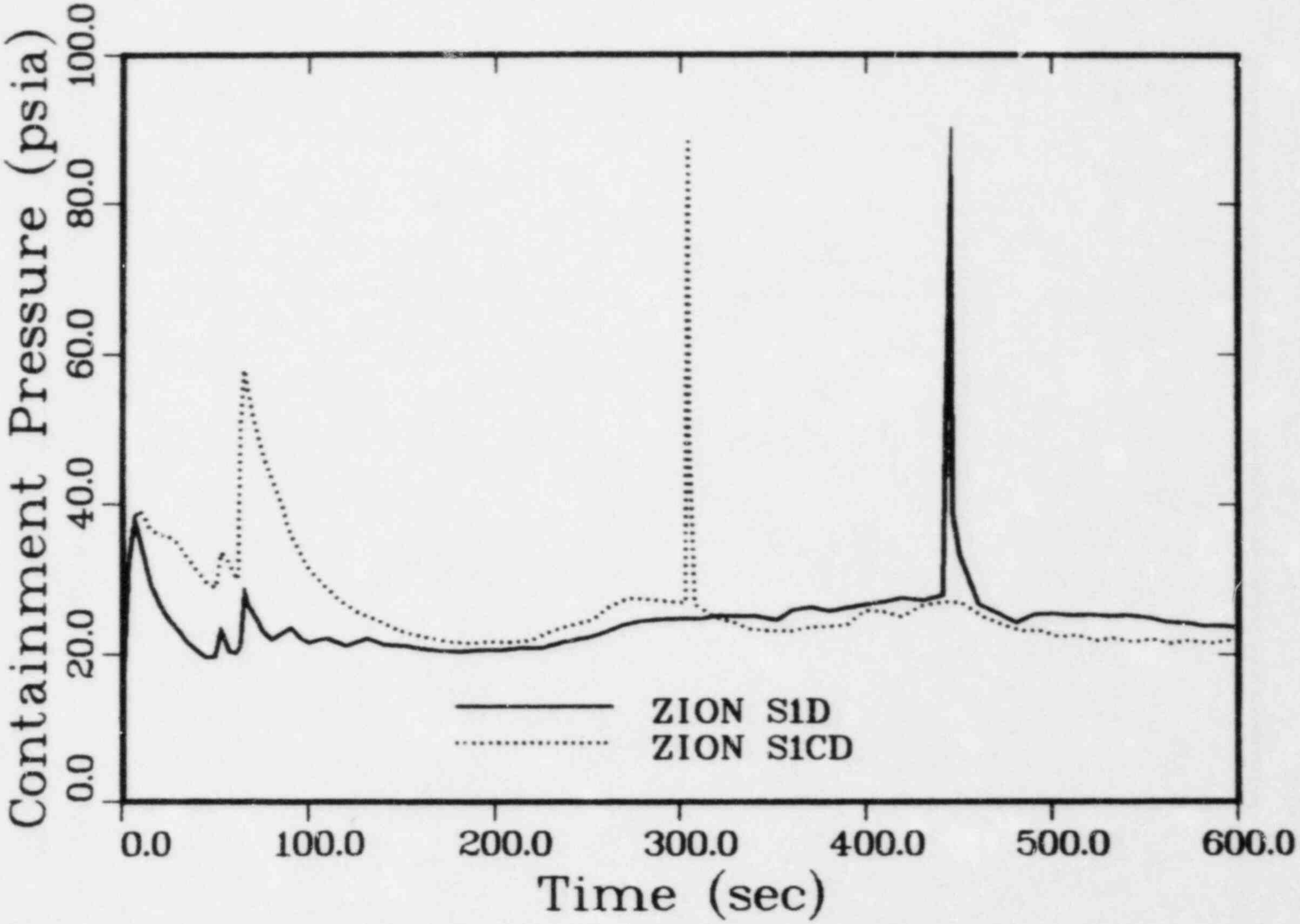

Figure 4.2.4-1. Zion S1D and S1CD (6 Inch Diameter Break) Containment Pressure Versus Time, MARCH 1.1 Calculations

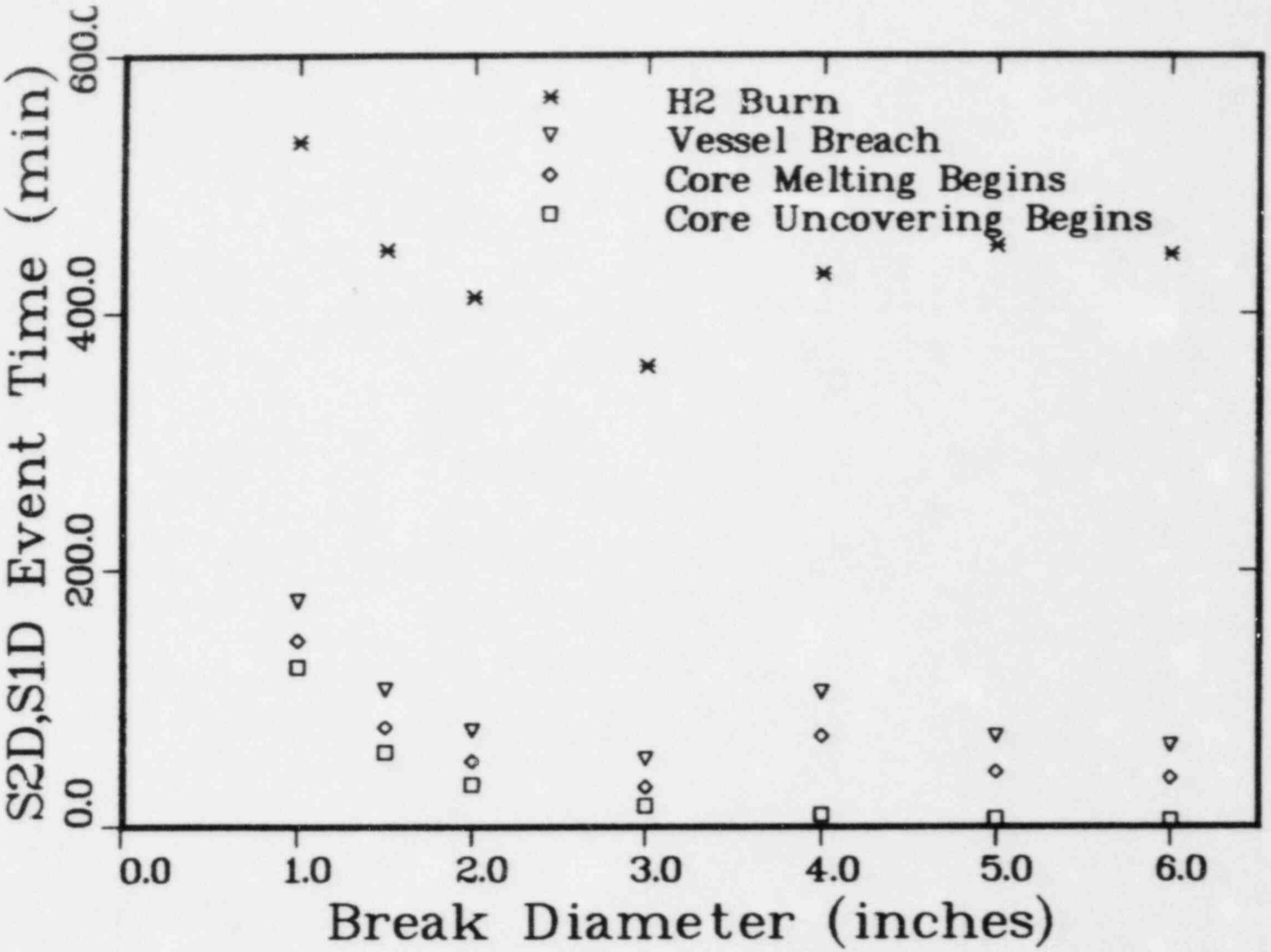

Figure 4.2.4-2. Zion S1D and S2D Accident Event Times Versus Break Diameter, MARCH 1.1 Calculations

is assumed to function in the S1DG and S2DG sequences but spray recirculation is assumed to fail after 10 minutes (as in Section 4.2.3). In the S1CDG and S2CDG sequences containment spray injection is postulated to fail. In this section, as in Section 4.2.4, injection failures (C and D) are assumed to lead to corresponding recirculation failures (i.e., Event C implies Event F and Event D implies Event H).

Hydrogen burns are suppressed in DG sequences because failure of the containment fan coolers leads to high steam concentrations (> 56 mole percent) in the containment atmosphere; however, consideration of the S1DG, S2DG, S1CDG, and S2CDG sequences provides some insight regarding the time required for containment failure due to steam overpressure. Figure 4.2.5-1 shows the MARCH-predicted containment pressure responses for S1DG and S1CDG sequences initiated by a 15.24-cm (6-inch) diameter break. This is the largest diameter which is categorized as a small break [WASH-1400, 1975]. It was selected to reduce the time to containment failure due to overpressure. Figure 4.2.5-1 shows that MARCH-predicted containment failure due to overpressure does not occur for small-break DG sequences before approximately 16 hours. The lower pressures in the S1DG sequence are attributable to energy being imparted to spray water which has collected in the reactor cavity. The relatively cool RWST water, injected into containment via the spray, overflows into the reactor cavity and collects sensible heat, thereby delaying the buildup in containment pressure. The same effect is observed for the large LOCA in Section 4.3 and is discussed further in Appendix C.

4.3 Large LOCAs

Three large LOCA (Event A initiated) sequences are discussed in this section: AHF, AD, and AHCG. Results for these sequences parallel the results for corresponding small-break-initiated sequences discussed in Section 4.2. Events for the large LOCA's occur sooner, of course, due to the more rapid blowdown from the primary system.

The AHF sequence is a large LOCA followed by successful ESF operations, including ECC and containment spray injection, from the RWST. Failures of ECC and containment sprays are postulated to occur when switchover to recirculation from the containment sump is necessitated by RWST depletion. Figure 4.3-1 is the containment pressure response for a Zion AHF sequence initiated by a double-ended cold-leg break. The RWST is depleted by the combined ECC and containment spray flows approximately 33 minutes after the pipe break. Until this point in time, the plant response is essentially as described in the safety analysis report [Zion FSAR, Sect. 14.3.2]. The failure of ECC upon switchover to recirculation leads to core degradation through meltdown. Bottom head failure, as predicted by MARCH 1.1, for the Zion AHF sequence occurs approximately 126 minutes after the double-ended cold leg break. This compares with a predicted bottom head failure at 127 minutes for

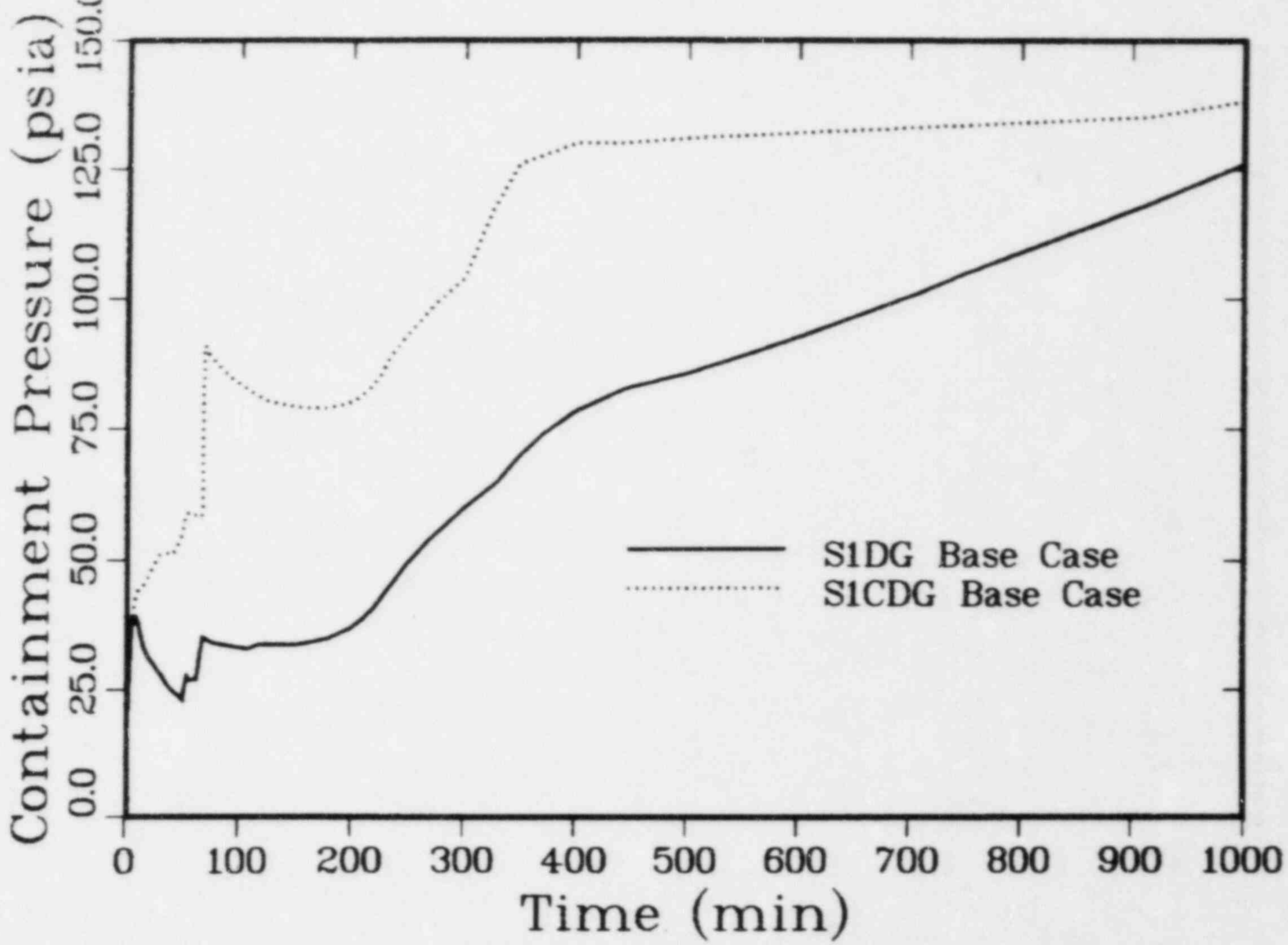

Figure 4.2.5-1. Zion S1DG and S1CDG (6-Inch Diameter Break) Containment Pressure Versus Time, MARCH 1.1 Calculations

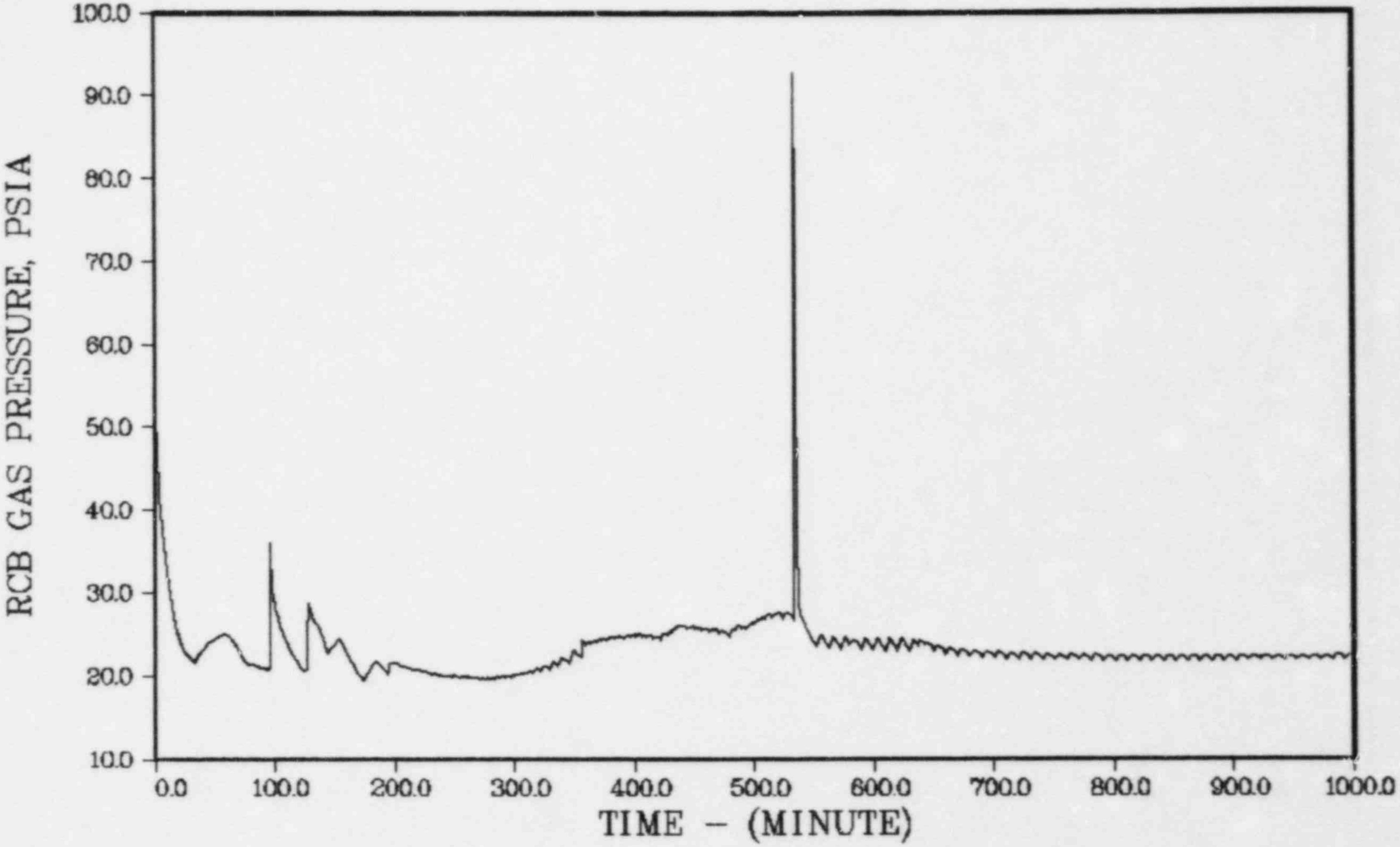

Figure 4.3-1. Zion AHF (Double-Ended Cold-Leg Break) Containment Pressure Versus Time, MARCH 1.1 Calculation

an S1HF sequence initiated by a 15.24 cm (6 inch) diameter break (see Figure 4.2.1-4 for small LOCA HF sequence event times).

Four containment fan coolers were assumed to operate as designed in the Zion AHF sequence. As a result, the containment pressure remains below the design pressure (0.429 MPa [47.5 psig]) until hydrogen burning occurs at approximately 530 minutes. With the fan coolers operating, steam concentrations in the containment atmosphere are not high enough to suppress hydrogen burns. Hydrogen is assumed to burn from 10 to 0 mole percent hydrogen. However, with four fan coolers operating, the containment pressure preceding the hydrogen burns is low enough that the lower bound pressure for containment failure (0.93 MPa [134.7 psia]) is not reached as a result of hydrogen burning (see also Section 8.3).

The AD sequence is a large LOCA followed by a failure of ECC injection. All ECC injection pumps -- low, intermediate, and high pressure -- are assumed to fail and, since the low pressure pumps are required for recirculation from the containment sump, both ECC and containment spray recirculation also fail. Figure 4.3-2 shows the containment pressure response for a Zion AD sequence initiated by a double-ended cold-leg break. Because ECC failure occurs immediately in the AD sequence, subsequent events occur more rapidly than in the AHF sequence. Bottom-head failure in the AD sequence is predicted by MARCH ~35 minutes after the pipe break. This compares with ~64 minutes for a 15.24-cm (6-inch) diameter S1D sequence (Figure 4.2.4-2). The conclusions regarding containment failure are the same for the AD sequence as for the AHF sequence; that is, with fan coolers operating as designed, containment failure due to overpressure or hydrogen burning would not occur.

The AHCG sequence provides some insight into the time required for containment failure due to overpressure at Zion. In the AHCG sequence, as in the AHF sequence, a large LOCA is followed by ECC recirculation failure. However, in the AHCG sequence, containment heat removal via the containment sprays or fan coolers is not available (CG). Figure 4.3-3 shows the containment pressure response for a Zion AHCG sequence initiated by a double-ended cold-leg break. Without containment sprays, or fan coolers, containment failure due to overpressure would occur eventually. However, the base case AHCG sequence does not yield a minimum time to containment failure due to overpressure since all of the relatively cool RWST inventory has been added to the containment. As discussed in Appendix C, the minimum time to containment failure due to overpressure occurs when just enough RWST water is added to permit saturation of the containment atmosphere at the containment failure pressure.

Figure 4.3-3 also shows the containment pressure response for an AHCG sequence in which this minimum amount (150,000 lb) of RWST

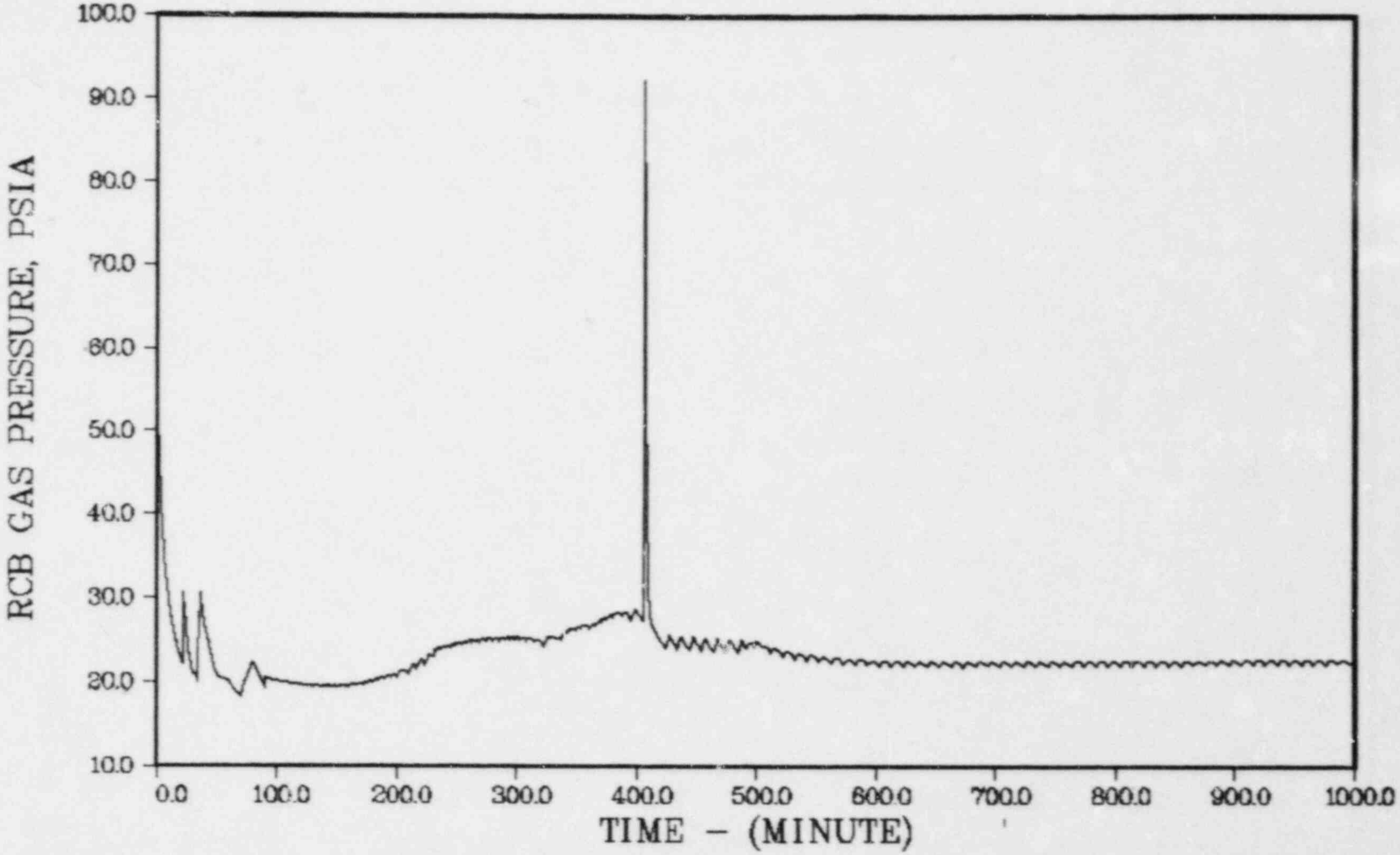

Figure 4.3-2. Zion AD (Double-Ended Cold-Leg Break) Containment Pressure Versus Time, MARCH 1.1 Calculation

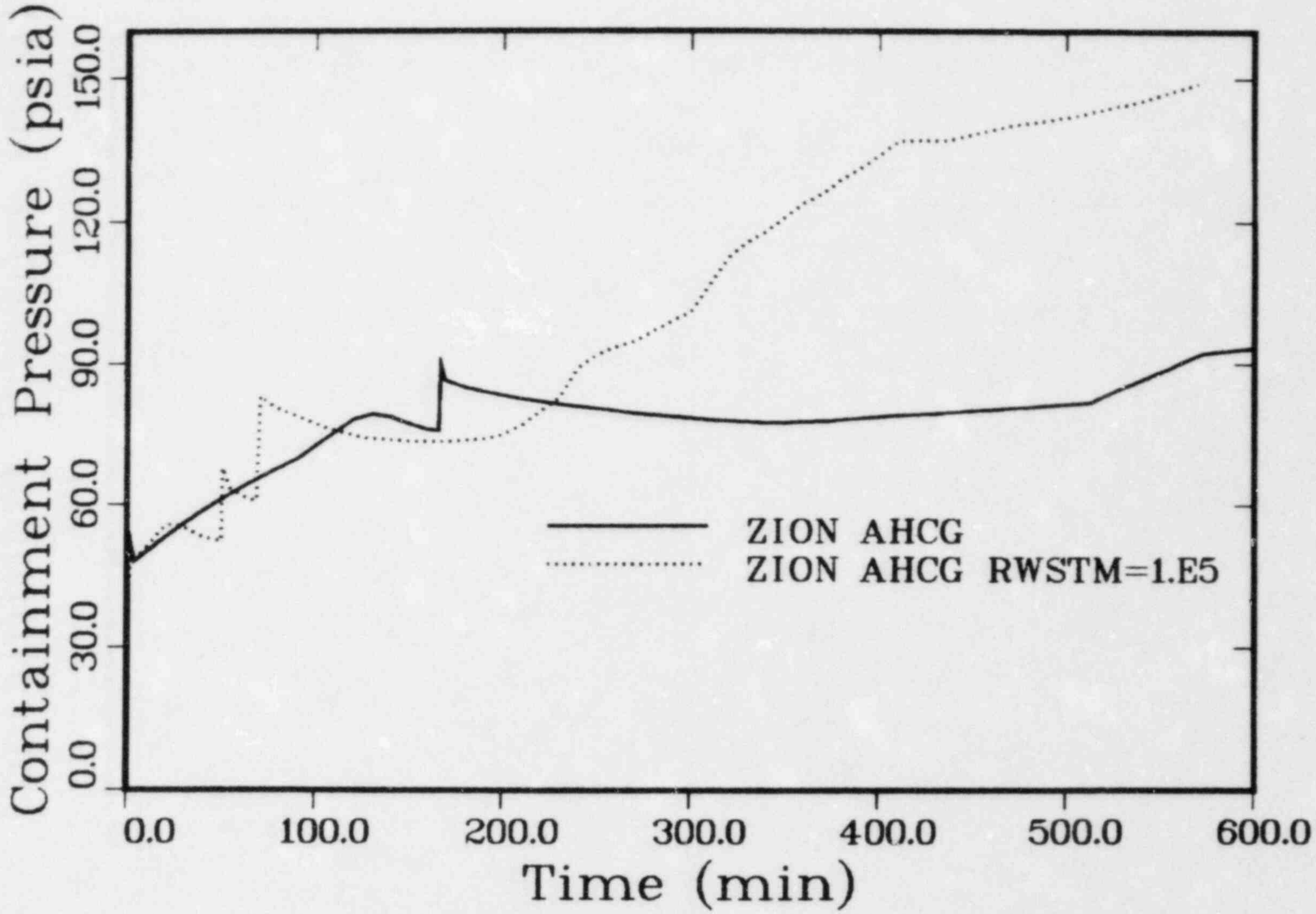

Figure 4.3-3. Zion AHCG (Double-Ended Cold-Leg Break) Containment Pressure Versus Time, MARCH 1.1 Calculation

water is added before ECC failure. For this case the containment pressure reaches 1.03 MPa (149 psia) at ~570 minutes.

4.4 V-Sequence LOCA

The V-sequence interfacing system LOCA is caused by failure of the valves (in series) which isolate the low pressure injection system in the auxiliary building from the high pressure reactor coolant system inside containment. At Zion, the most likely V-sequence LOCA would be initiated by failures of the two motor-operated valves (RH8701 and RH8702) in the RHR suction line [Zion PRA, 1981]. Figure 4.4-1 is a piping schematic for the Zion RHR suction line. This line is used during plant shutdown when the RHR system is in operation. The postulated valve failures would result in flow from the reactor coolant system into the low pressure injection system (LPIS). Should the backflow through the failed valves exceed the capacity of the LPIS relief valves,* a break could occur in the LPIS pressure boundary due to overpressure or dynamic loading beyond the design basis. Such a break would cause primary coolant to be discharged directly to the auxiliary building.

The ultimate consequences of V-sequence valve failures would depend on the extent of the resulting ECCS failures. The Reactor Safety Study [WASH-1400, 1975] assumed that such valve failures would "almost surely" lead to total LPIS failure. The Reactor Safety Study postulated a 15.24-cm (6-inch) diameter break in the LPIS pressure boundary as a result of the valve failures. With a total LPIS failure, even if ECC injection from the RWST were to function, switchover to ECC recirculation would not be possible because the LPIS pumps are required to take suction from the containment sump.

Depending on the nature of the valve failure and the details of the ECCS design and layout, it is conceivable that flow through the valves could be accommodated by the LPIS relief valves. If not, a break in small interconnecting process or instrument lines could well occur before a larger (e.g. 6-inch diameter as per RSS) process line break. It is also conceivable that only one LPIS train, the one in which the break occurred, would fail. To test such hypotheses would require detailed design information and thermohydraulic and fracture analyses. Also, the possible effects of steam flooding on the operability of ECCS pump motors and valve operators in the auxiliary building would have to be assessed.

Such detailed analyses are beyond the scope of this study. We assumed that V-sequence valve failures would result in a 3.81-cm (1.5-inch) diameter break and that the response of the primary and

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*Flow from the LPIS relief valve is returned to containment and hence is not released to the environment.

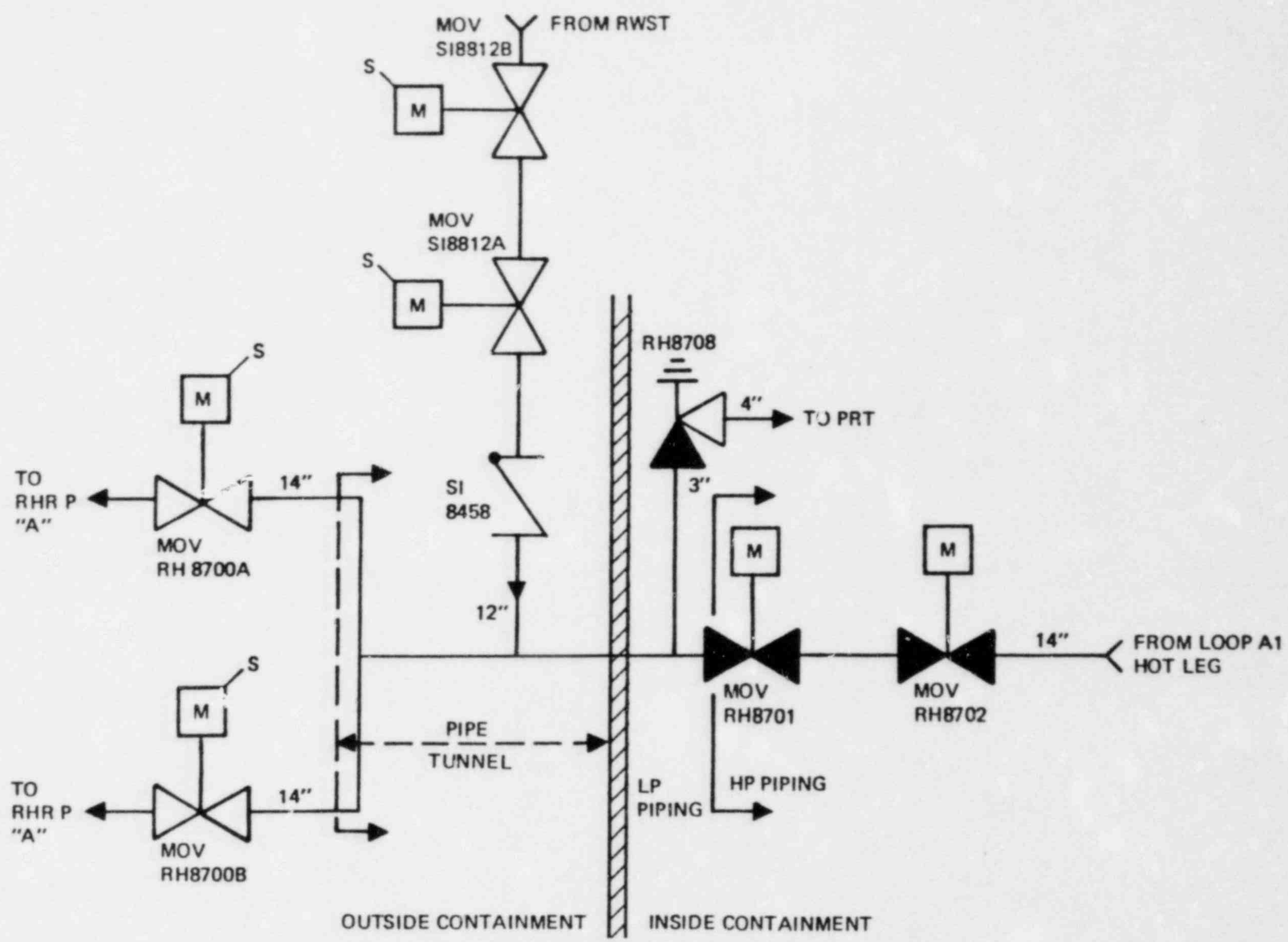

Figure 4.4-1. Schematic Diagram of Portion of RHR Suction Line Susceptible to V-Sequence Valve Failures

secondary coolant systems would be essentially equivalent to that for the corresponding S2HF accident discussed in Section 4.2.1. Such an accident would eventually lead to core meltdown and vessel breach. Since the containment would be bypassed in the V-sequence LOCA, containment pressure, temperature, and radiation levels would be unaffected until the time of vessel breach. The signature of a V-sequence LOCA would thus be distinct, showing no initial change in containment parameters but sharp increases in auxiliary building pressure, temperature, and radiation levels. Operator action to isolate the break was not assumed in this section but is discussed in Section 8.1.4. Obviously, for a larger break size (e.g., 6-inch diameter as per the RSS) less time would be available for the operator to isolate the break.

An unisolated break in the auxiliary building would provide a pathway for release of radionuclides from the containment both before and after vessel breach. There would, of course, be plateout of less volatile species in the auxiliary building. Radiological consequences for the unisolated V-sequence LOCA are discussed in Section 5.

5. RADIOLOGICAL CONSEQUENCES FOR BASE CASE SEVERE ACCIDENTS

This section summarizes our estimates of the radiological consequences for a number of accident sequences for the Zion plant. No mitigative measures are considered in the results of this section; mitigation is covered in Section 8 and 9. Best-estimate values were used in all calculations; best estimate is discussed in the introduction and in Section 6.

5.1 Method of Calculation

Transport, removal, and leakage of radioactive material in the containment atmosphere have been calculated using the CORRAL code from Appendix VII of WASH-1400. A generalized version of the code CORRAL-2 was used [Burian and Cybulskis, 1977].

Release fractions of radionuclides from containment calculated by CORRAL were used to compute two types of radiological consequences for the Zion site: mean latent cancer facilities, and land area interdicted.

Consequences were computed by the CRAC2 code [CRAC2] for two classes of release, "high" and "low", using demographic and meterological data for the Zion site. The "high" release was based on the calculated core fractions released in a TMLB' sequence with an arbitrarily early, catastrophic containment failure postulated six hours after accident initiation. The "low" release was based on the calculated core fractions released in the first 15 hours of a TMLB' sequence with design leakage. Shortlived radionuclides have decayed to insignificant concentrations at these times and do not affect the consequences. All consequences are normalized to the core fraction release so that the details of the sequences used for the "high" and "low" releases have little effect on subsequent calculations of consequences. Consequences for specific accidents were computed by taking CORRAL release fractions and interpolating between the CRAC2 "benchmark" data. A short computer code, CONSEQ, was written to perform this interpolation (see Appendix D).

Tables 5.1-1 and 5.1-2 summarize the CRAC2 "high" and "low" results. Consequences were calculated separately for each of seven nuclide groups (Kr-Xe, I-Br, Cs-Rb, Te, Ba-Sr, Ru, and La); also, consequences were calculated for a simultaneous release of all seven groups. Note that the mean man rem (population dose) and mean latent cancer fatalities, which are proportional to population dose, are additive: a release of all seven groups is essentially equivalent to the sum of the releases of each separate group. (Acute fatalities are not additive in this manner since below a threshold individual dose, such fatalities do not occur.)

As the tables indicate, CRAC2 predicts the mean man rems per core fraction released and mean latent cancer fatalities per core fraction released are relatively independent of the fraction

Table 5.1-1

CRAC2 Consequences For "High" Release

Nuclide Group	Fraction of Core Inventory Released	Mean Man † Rem	Mean Latent Cancer † Fatalities	Mean Land Area Interdicted (square (miles)	Mean Man Rem Per Core Fraction Released	Mean Latent Cancer Fatalities Per Core Fraction Released	Mean Land Area Interdicted Per Fraction Released (square miles)
Kr-Xe*	0.82	9.6×10^4	6	0	1.2×10^5	8	0
I-Br*	0.38	2.0×10^6	157	0	5.3×10^6	413	0
Cs-Rb*	0.29	2.1×10^7	988	36	7.3×10^7	3410	124
Te*	0.35	5.5×10^5	130	0	1.6×10^6	371	0
Ba-Sr*	0.030	1.1×10^6	54	0	3.5×10^7	1790	0
Ru*	0.026	1.6×10^6	306	0	6.1×10^7	11800	0
La*	0.0045	2.9×10^6	216	0	6.4×10^8	48000	0
All**	Sum of Above	2.4×10^7	1560	38	mean early fatalities = 179 maximum early fatalities = 22900		

*Consequences calculated separately for each group.
**Consequences calculated for simultaneous release of all groups.
†Maximum values are about 10 times mean values.

Table 5.1-2

CRAC2 Consequences For "Low" Release

Nuclide Group	Fraction of Core Inventory Released	Mean Man Rem †	Mean Latent Cancer Fatalities †	Mean Land Area Interdicted (square miles)	Mean Man Rem Per Core Fraction Released	Mean Latent Cancer Fatalities Per Core Fraction Released	Mean Land Area Interdicted Per Fraction Released (square miles)
Kr-Xe*	1.1×10^{-3}	1.7×10^{2}	1.1×10^{-2}	0	1.6×10^{5}	10	0
I-Br*	1.6×10^{-5}	1.7×10^{2}	1.2×10^{-2}	0	1.0×10^{7}	747	0
Cs-Rb*	1.3×10^{-5}	1.0×10^{4}	4.7×10^{-2}	0	7.5×10^{8}	35600	0
Te*	4.2×10^{-5}	6.8×10^{1}	1.1×10^{-2}	0	1.6×10^{6}	252	0
Ba-Sr*	8.2×10^{-7}	4.7×10^{1}	2.1×10^{-3}	0	5.8×10^{7}	2630	0
Ru*	2.6×10^{-6}	3.4×10^{2}	3.9×10^{-2}	0	1.4×10^{8}	15200	0
La*	5.0×10^{-7}	5.4×10^{2}	2.5×10^{-2}	0	1.1×10^{9}	50300	0
All**	Sum of Above	1.1×10^{4}	5.7×10^{-1}	0	mean early fatalities = 0 maximum early fatalities = 0		

*Consequences calculated separately for each group.
**Consequences calculated for simultaneous release of all groups.
†Maximum values are about 10 times mean values.

released, for all nuclide groups except for Cs-Rb group. Above a certain Cs-Rb release fraction (about 10^{-3} for Zion), interdiction of land is assumed by CRAC2 in order to reduce cancer deaths from chronic exposure; cancer deaths per core fraction are reduced by CRAC2 at the expense of interdicting the land. The area of land interdicted is a "consequence" that should not be neglected. Table 5.1-3 indicates the tradeoff specified by CRAC2 between cancer deaths and land area interdicted for a range of Cs-Rb release fractions.

CONSEQ estimates mean man rem, mean latent cancer fatalities, and mean land area interdicted as indicated in Figure 5.1-1. Appendix D describes the interpolation scheme used by CONSEQ.

5.2 Containment Failure

Radioactive material can be released to the public by any mechanism that causes bypass or failure of the containment barrier. We have analyzed releases via normal containment leakage, failure of containment isolation, massive rupture caused by static overpressure, massive rupture caused by hydrogen burning, and bypass of containment.

The penetrations and liner-plate welds in the Zion containment are pressurized. Containment leakage should be essentially zero as long as the penetrations and liner welds remain pressurized and the failure limits on containment are not reached. A conservative assumption is that pressurization is lost during the accident of concern. The design leakage is 0.1 percent of containment volume for 24 hours at 0.425 MPa (61.7 psia). Figure 5.2-1 shows fractional containment leakage which was used in our calculations as a function of containment internal pressure. Flow of mixtures of saturated steam and air through an equivalent orifice was calculated using standard fluid-flow formulas [Marks, 1978]. The knee in the curve indicates where choked flow begins. The curve exceeds design leakage of 0.1 percent per day (42×10^{-6} per hour) at 0.425 MPa because of rounding up of the equivalent orifice diameter and the use of saturated atmospheric compositions differing from those for which the equivalent orifice was chosen. The conservatism in leak rate is believed unimportant in comparison with other major uncertainties.

The largest penetrations through the containment are isolated by air-operated valves. These would close automatically upon loss of power. However, in addition to the air-operated valves, there are also several motor-operated valves which require power for isolation. Isolation failure could occur either from failure of air-operated valves to close, or via paths to the atmosphere through open motor-operated valves. The effective leakage area is determined by the valve alignment at the time of the accident, and is accident specific.

Table 5.1-3

CRAC2 Cs-Rb Consequences

Fraction Cs-Rb Released	Mean Man Rem	Mean Latent Cancer Fatalities	Mean Area Land Interdicted (square miles)
0.5	2.7×10^{7}	1270	63
0.29	2.1×10^{7}	988	36
0.10	1.3×10^{7}	609	14
0.01	2.7×10^{6}	125	0.89
0.001	4.1×10^{5}	19.1	0.014
0.0001	5.2×10^{4}	2.4	0
0.000013	1.0×10^{4}	0.47	0

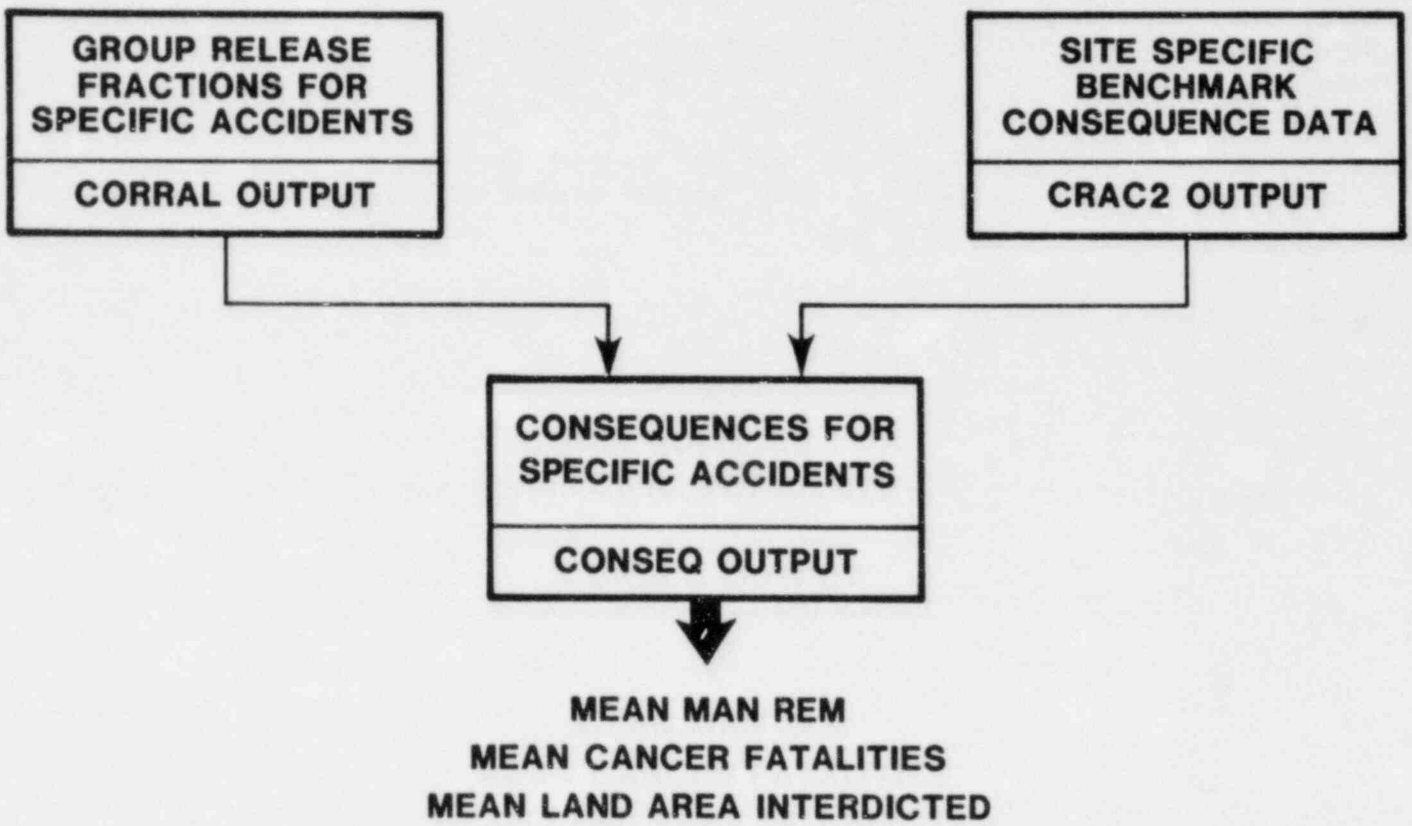

Figure 5.1-1. Method for Calculating Consequences

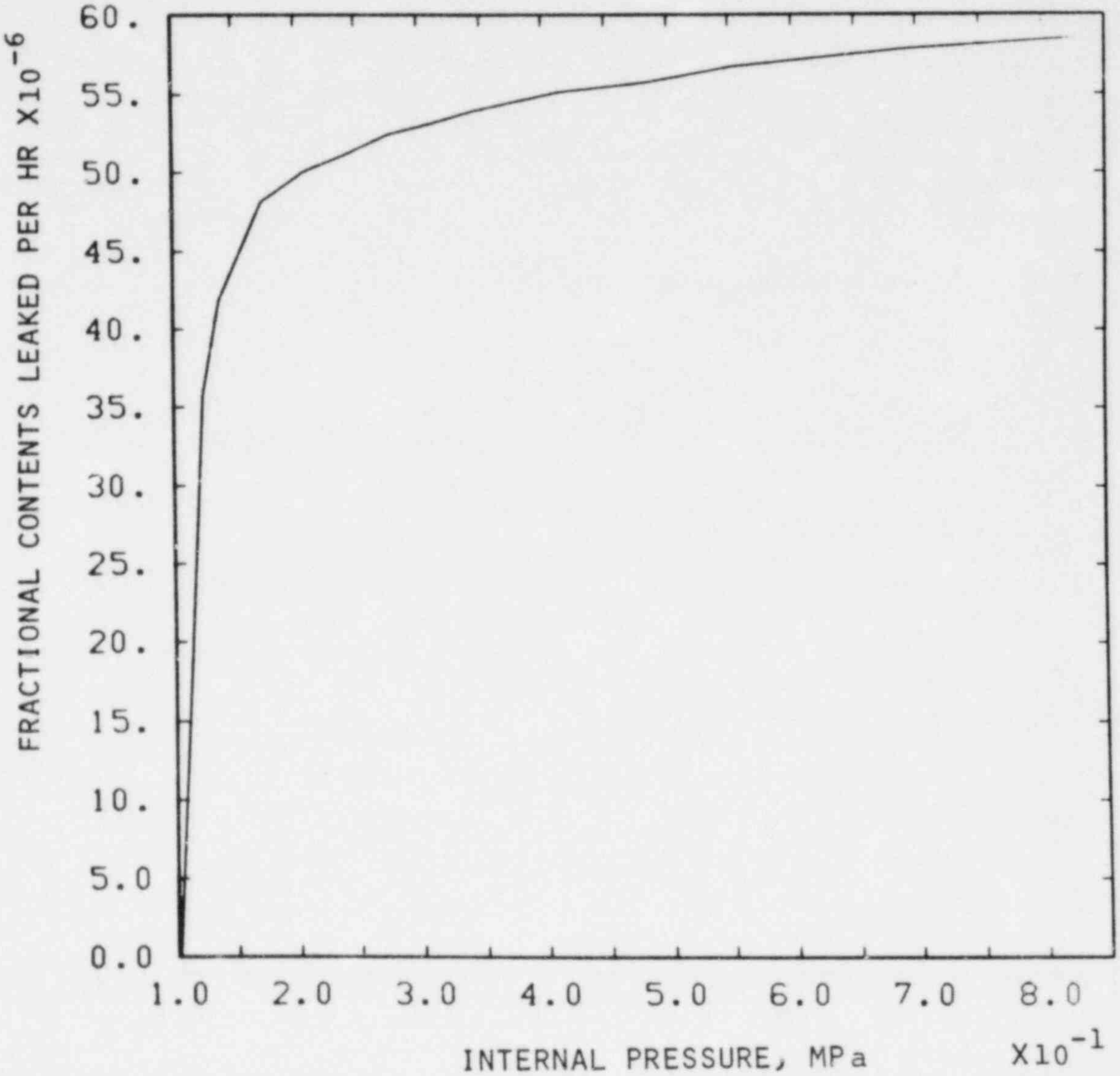

Figure 5.2-1. Containment Leakage as a Function of Internal Pressure

The Reactor Safety Study [WASH-1400, 1975] assumed a leakage rate of 1000 times the design rate when the containment fails to isolate. For the Zion plant, this would be equivalent to a hole slightly over 8 cm in diameter, assuming choked flow. This leak rate is plausible for isolation failure in some circumstances; however, a lower leak rate of 100 times the leakage depicted in Figure 5.2-1 has been used for this study. This leak rate produces consequences intermediate between the design leakage and gross containment rupture. We have not analyzed in detail possible leakage paths to determine the most probable leak rate; the possible leakage paths are accident specific, and such an analysis requires detailed plant data that were not available.

Containment failure from overpressure is discussed in detail in Appendix B. Because of the high strength and the large volume of the Zion containment, the probability of containment failure is lower for this plant than for the PWR studied in the RSS. If containment overpressure failure occurs, it would be expected rather late, on the order of 10 hours after the initiating event.

Pressures due to hydrogen burning are not expected to fail containment except for situations in which hydrogen builds up in a steam-rich atmosphere, containment is then cooled, and a severe hydrogen burn occurs (see Section 8.3). Under these circumstances, containment failure is likely with consequences essentially identical to those from overpressure failure of containment.

5.3 Results of Consequence Calculations

In comparing consequence calculations among different accident sequences, the inherent uncertainty of the models should be considered. We believe that order-of-magnitude differences among results are significant; smaller differences are not as significant, although they may indicate trends.

5.3.1 LOCAs in Containment

The S2-, S1-, and A-initiated LOCAs are all discussed in this section because of the similarity in consequence results for these types of accidents.

We have investigated the effects of break size, containment isolation, and containment cooling on radiological consequences.

For sequences in which containment is cooled and isolated, the results of Table 5.3-1 are applicable. The effect of break size on estimated consequence is minor and is due to such factors as timing of key events (end of emergency core cooling, core uncovering, vessel breach) and differences among deposition phenomena in containment (time during which sprays operate, when HEPA filters plug

Table 5.3-1

Radiological Consequences for LOCAs With Cooled, Isolated Containment

Accident Sequence	Break Diameter (inches)	State of Containment	Mean Man Rem	Mean Latent Cancer Fatalities	Mean Land Area Interdicted (square miles)
S2HF	1.5	Design Leakage	2×10^4	1	0
S1HF	4	Design Leakage	2×10^4	1	0
S1HF	6	Design Leakage	3×10^4	1	0
AHF	Cold Leg	Design Leakage	4×10^4	2	0
S2D	1.5	Design Leakage	2×10^4	1	0
S1D	6	Design Leakage	4×10^4	2	0
AD	Cold Leg	Design Leakage	2×10^4	1	0

up, and so on). However, extending the time prior to core degradation does increase the likelihood that appropriate mitigative actions would be taken. The impact of such mitigative actions on radiological consequences is discussed in Section 9.

The cooling of containment is of paramount importance in reducing consequences of core melt accidents initiated by pipe breaks within containment. At Zion, containment cooling can be accomplished by sprays or fan coolers. For sequences in which containment cannot be cooled, the consequences are much more severe than for those in which containment cooling can be maintained, as indicated in Table 5.3-2.

Due to the high failure pressure of the Zion containment, loss of containment cooling does not lead to overpressure failure until late -- around 10 hours -- into the accident. The effect of restoring containment cooling is examined in Section 9 where mitigative measures are considered.

If containment cooling is available, the degree of containment isolation can significantly affect consequences. Table 5.3-3. indicates the importance of containment isolation.

5.3.2 Interfacing-Systems LOCA

The particular sequence examined is a 1.5 inch break in a line which connects the RHR system to the primary system (see Section 4.4). This results in a direct discharge of coolant and radionuclides into the auxiliary building; containment is bypassed. Blowdown from the reactor vessel to the auxiliary building is modeled as an isentropic process from vessel pressure/temperature to atmospheric pressure. (The auxiliary building is assumed to fail at just above atmospheric pressure.) Natural deposition of radionuclides as they pass through the auxiliary building is the only process by which the concentrations of radionuclides are reduced due to the presence of the building. The flow rate from the auxiliary building was assumed to be that required to maintain it at atmospheric pressure.

The consequences of the interfacing systems LOCA are given in Table 5.3-4; this accident sequence gives the highest consequences of all the sequences examined for the Zion plant.

5.3.3 Transients

The TMLB' sequence was analyzed in detail for Zion [Haskin et al., 1981]. The estimated consequences of such a sequence are given in Table 5.3-4. These consequence estimates assume that AC power is not restored so that overpressurization eventually occurs due to lack of containment heat removal. The estimated consequences are higher than for LOCA sequences involving overpressure failure,

Table 5.3-2

Effect of Containment Cooling on Radiological Consequences for S1D LOCA (6-Inch Diameter Break)

Accident Sequence	State of Containment	Mean Man Rem	Mean Latent Cancer Fatalities	Mean Land Area Interdicted (square miles)
S1D-6"	Design Leakage	4×10^4	2	0
S1CG-6"	Overpressure after 18 Hours	2×10^7	840	16

Table 5.3-3

Effect of Containment Isolation on Radiological Consequences for S1D LOCA (6-Inch Diameter Break)

State of Containment*	Mean Man Rem	Mean Latent Cancer Fatalities	Mean Land Area Interdicted (square miles)
Pressurization of Penetrations and Welds	0	0	0
Design Leakage (0.1 volume % per day)	4×10^4	2	0
Isolation Failure (100 times Design Leak Rate)	8×10^5	37	0.2

*Section 5.2 discusses state of containment.

Table 5.3-4

Radiological Consequences for V-Sequence LOCA and TMLB'

Accident Sequence	State of Containment	Mean Man Rem	Mean Latent Cancer Fatalities	Mean Land Area Interdicted (square miles)
V	Bypassed	4×10^7	2070	65
TMLB'	Design Leakage	6×10^4	3	0
TMLB'	Overpressure Failure at 10 Hours*	2×10^7	1210	23

*See discussion in text and in [Haskin et al., 1981].

due to the complete unavailability of containment sprays and fan coolers in the TMLB' sequence. The length of time between accident initiation and containment failure, tens of hours, provides a substantial time during which AC power could be restored. Our best estimate is that overpressure failure is not likely for the TMLB' sequence at Zion because of the likelihood of early AC power restoration.

We have not evaluated in detail the recoverability of containment sprays or fan coolers when AC power restoration is delayed until after vessel breach.

5.3.4 Other Sequences

Radiological consequences for other Zion sequences were not analyzed. We believe that the previous sequences bound the consequences for all potential accidents involving core melt, except for steam explosion accidents that rupture containment, which are considered unlikely [Swenson and Corradini, 1981]. The state of containment is the primary factor affecting consequences.

An accident sequence involving core melting belongs to one of the four following categories: containment is bypassed, containment fails by overpressurization or by hydrogen burning, containment does not fail but does not isolate, or containment does not fail and does isolate. Figure 2-3 of Section 2, and Appendix E summarize consequences for Zion based on these categories.

6. UNCERTAINTIES

The analysis of accident progression requires models of relevant plant characteristics and physical phenomena. Inputs for these models must be specified based on existing knowledge. Uncertainties about the inputs, the models, and the phenomena lead to uncertainties in the predicted results. These uncertainties and their potential magnitudes must be recognized if meaningful conclusions are to be drawn.

Uncertainties in the inputs will be both probabilistic (random) and systematic. Uncertainties in initial conditions are often probabilistic. For example, the level in the refueling-water storage tank (RWST) varies over a controlled range during normal operation. A probability-distribution function could be developed, based on data from the plant, to characterize the probability of any particular level at the time of the initiating event. Other inputs which are probabilistic in nature include boundary conditions (e.g., meteorological conditions), equipment failures, and operator actions. Systematic uncertainties also exist for many inputs and for most models. For example, it may not be feasible to determine the exact mass and exposed surface area associated with all passive heat sinks in the reactor containment building. Even if all extensive and intensive properties of heat sinks were known with total accuracy, practical restrictions on computation times would necessitate modeling with a small number ($\leq$ 15 in MARCH) of composite slabs having discrete spatial nodes. Thus systematic uncertainties arise from incomplete knowledge of inputs and from modeling approximations. Modeling approximations may be due to computational restrictions or incomplete knowledge of the phenomena being modeled.

In principle, if all systematic uncertainties involved with determining and modeling plant characteristics and physical phenomena could be eliminated or bounded and distribution functions for the probabilistic inputs to these models could be developed, a bounded consequence-versus-probability relationship could be established using Monte Carlo sampling techniques [Hall, 1979]. In practice, however, systematic input and modeling uncertainties cannot always be eliminated or bounded and the generation and sampling of distribution functions for probabilistic inputs is prohibitively time consuming and expensive.

Figure 6-1 illustrates an effective (though less rigorous) process for evaluating and selectively reducing uncertainties. In this process a relative uncertainty in predicted result(s) is assigned to selected input or modeling uncertainties. This assignment may of necessity be qualitative; however, if input bounds or bounding models are available or can be developed, then a quantitative assignment can be made with reference to a base case. By this process, relative measures of uncertainty are provided and priorities for their reduction can be established. Because a large number of analyses are required for this process, fast-running analytic vehicles such as MARCH are required.

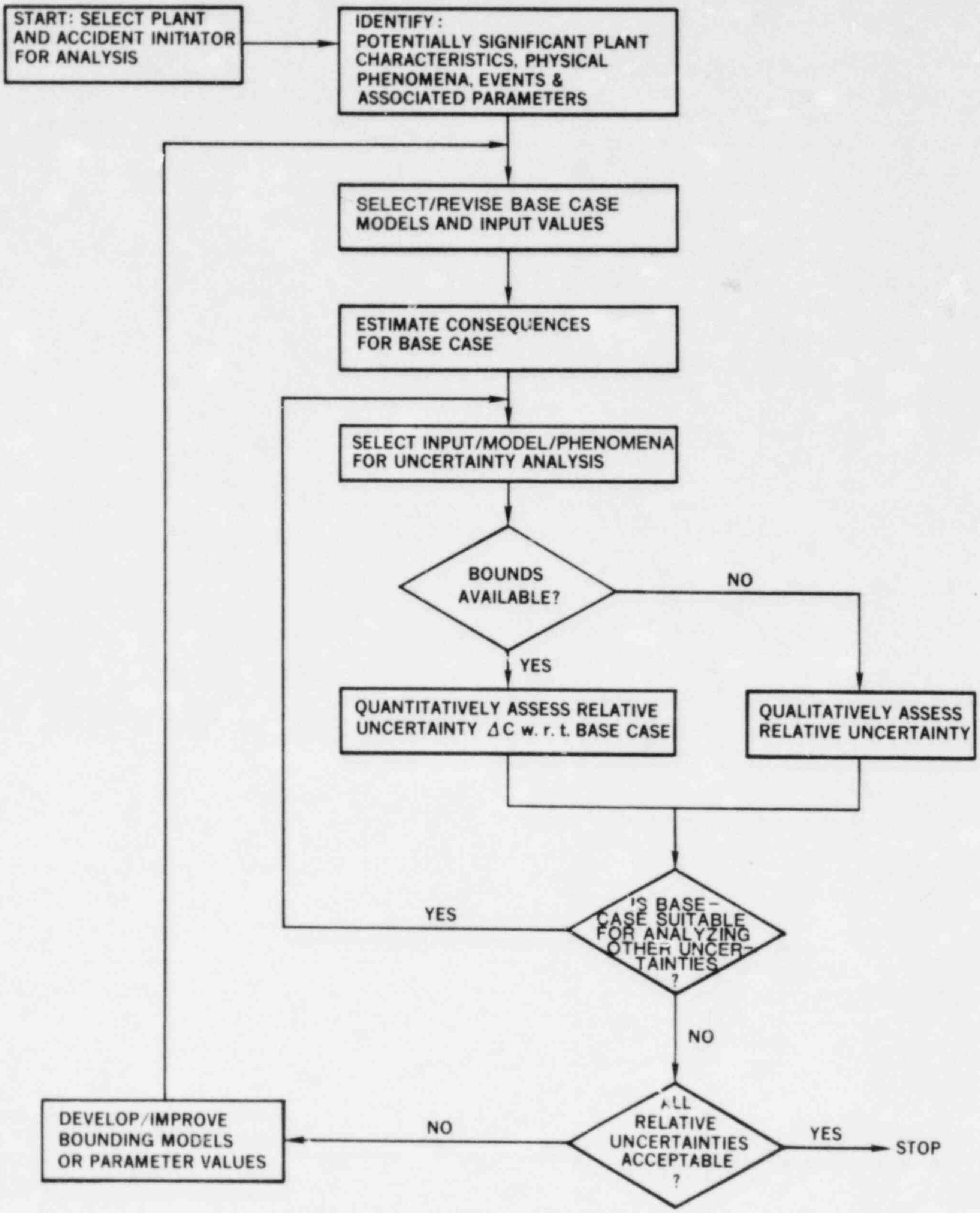

Figure 6-1. Process for Reducing Severe Accident Sequence Analysis Uncertainties

The discussions of uncertainties presented in the following subsections represent a limited application of the process described above. The emphasis is on identifying inputs and models which the authors felt could significantly affect calculated results. A more comprehensive application of the process is beyond the scope of this report. Many relevant uncertainties have been identified and discussed in more detail elsewhere [Rivard, 1980], [Haskin et al., 1981], [Rivard et al., 1981], [NUREG-0850, 1981].

The uncertainties discussed below fall into three categories: input uncertainties, modeling uncertainties, and phenomenological uncertainties. These categories reflect the authors' current judgment regarding appropriate methods for reducing or bounding the uncertainties. Input uncertainties are those which could be reduced or bounded by developing better input data for use with existing models. The reduction or bounding of modeling uncertainties would require further verification, improvement, or development of models based on existing phenomenological knowledge.

Phenomenological uncertainties result primarily from incomplete information about physical phenomena. Significant reductions in phenomenological uncertainties would require new experimental data; however, in some instances a phenomenological uncertainty might be bounded using limiting (bounding) models.

Note that assignment of an uncertainty to one of the three categories does not imply a unique reason for the uncertainty. An uncertainty in predicted result(s) is often caused by a combination of factors: imperfect knowledge of relevant input data, modeling approximations, and imperfect knowledge of the phenomena involved.

6.1 Input Uncertainties

Uncertainties regarding certain plant characteristics can lead to corresponding uncertainties in predicted results. Plant input uncertainties which have been investigated [Haskin et al., 1981] include the decay power level, the containment-liner-to-concrete heat transfer coefficient, the passive heat sinks within the containment, the threshold for overflow of water from the containment floor to the reactor cavity, and the composition of concrete in the containment basemat.

Decay Power

The decay power used for the MARCH calculations in this report was based on an average irradiation period of 2 years and the current American Nuclear Society standard [ANS 5.1, 1979]. Decay power for the end of cycle core was also estimated using the more detailed ORIGEN code [Bennett, 1979]. Both decay power estimates included contributions due to heavy element decay as shown in Figure 4.1.2-1. The decay power estimated with ORIGEN is roughly 9 percent lower than that estimated per ANS 5.1, 1979. For accidents such as TMLB' (no break in primary pressure boundary), in which decay power is the dominant driving force for reactor coolant discharge to containment, in-vessel event times vary roughly linearly with decay power. For example, using decay heat based on ORIGEN rather than ANS 5.1, 1979 delays the time for bottom-head failure by more than 20 minutes. More importantly, the integral decay heat controls the time to possible containment overpressure failure and a 9 percent reduction in decay power can easily extend the time to containment overpressure failure by an hour or more.

Liner-to-Concrete Heat Transfer

When containment ESFs (containment sprays and air coolers) are not available, the ability to transfer heat out of the containment is a strong function of the liner-to-concrete heat transfer coefficient, HIF. Values of HIF ranging from 0 to 568 $W/m^2/K$ (0 to 100 $BTU/hr/ft^2/°F$) have been used in safety analysis reports. Our best estimate of HIF for Zion is 114 $W/m^2/K$ (20 $BTU/hr/ft^2/°F$). This value is uncertain due to a lack of detailed design data, analysis of such data, and questions regarding the impact of concrete offgassing and liner thermal expansion on effective gap conductance at prolonged high containment temperatures [Haskin et al, 1981]. However, increasing HIF from 23 to 568 $W/m^2/K$ only decreases the TMLB' containment pressure by less than 7 percent (8 psig) during the first 10 hours of the TMLB' sequence.

Passive Heat Sinks

The passive heat sinks in containment can also influence the rate of pressure buildup if containment ESFs are not available. Steel heat sinks in particular can rapidly absorb energy. However, nominal ($\pm$10 percent) variations in the Zion heat sink masses and areas resulted in only minor variations in predicted containment pressures using the MARCH 1.1 condensing heat transfer correlation [Haskin et al., 1981].

Dry Versus Wet Reactor Cavity

The MARCH runs for this report (Appendix F) assumed that overflow into the reactor cavity would occur when 56.6 m^3 (2000 ft^3) of water had accumulated in the sump and on the containment floor. This assumption was based on statements by utility personnel about basemat layout and pathways from the containment floor to the lower reactor cavity. If overflow were delayed until 198 m^3 (7000 ft^3) of water had accumulated in containment, the reactor cavity would still be dry at the time of vessel breach for TMLB'. The only water in the cavity after breach of the vessel would be from the accumulators which would empty due to depressurization of the reactor vessel.

There are many phenomenological uncertainties concerning the interactions of molten debris with water in the cavity. These phenomenological uncertainties are addressed separately in Section 6.3. Our purpose here is to point out that some uncertainty exists about the threshold for overflow from the containment floor into the reactor cavity at Zion and this leads to an associated uncertainty in containment pressure as predicted by MARCH.

MARCH predicts significantly higher containment pressures with overflow of water from the containment floor to the reactor cavity. This is due to the evaporation of the water in contact with molten debris in the reactor cavity following a breach of the vessel. The assumption of overflow at approximately 56.6 m^3 (2000 ft^3) therefore seems appropriate (in the absence of updated information) for our Zion analyses with MARCH. (See also [NUREG-0850, 1981].)

Basemat Concrete Properties

When molten core material contacts concrete, water vapor and other gases, mostly carbon dioxide, are released. The amount of carbon dioxide released depends on the amount of calcium carbonate in the concrete mix. Basalt concrete contains only small amounts of calcium carbonate (~1 percent by weight), whereas limestone concrete can contain up to 80 percent calcium carbonate by weight. Assuming limestone concrete leads to predicted containment pressures 20 percent or more higher than those predicted based on basalt concrete [NUREG-0850, 1981]. A detailed chemical analysis of Zion basemat concrete was not available for this report; however, some limestone concrete is known to have been used at Zion [NUREG-0850]. The calculations in this report were therefore based on limestone concrete with a calcium carbonate content of 80 percent (Appendix F). It should also be noted that water vapor and carbon dioxide can react chemically with molten metals to produce flammable carbon monoxide and hydrogen. The amount of flammable gases produced in this manner can be comparable to the amounts of hydrogen generated by metal-water reactions in-vessel. Thus limestone concrete can result in higher combustible gas concentrations in containment. Such concentrations can be significant unless the maximum extent of combustion in containment is already oxygen limited.

Containment Penetrations

There are many piping and electrical penetrations of containment. Also, the equipment hatch and personnel airlock penetrate containment.

We did not have sufficient information available to analyze failure of penetrations during severe accidents. We assumed that the containment structure itself would fail before any penetrations would fail.

If the penetrations meet the ASME code [ASME III-1, 1980], [ASME III-2, 1980], their failure due to overpressurization is unlikely to occur before gross failure of containment. Degradation of electrical penetration insulation due to high temperatures is a potential failure mode that we did not examine.

Sandia has a containment safety margin program which is addressing the ultimate capacity of containment buildings [Blejwas, 1982]. A few tests will examine the effects of penetrations.

6.2 Modeling Uncertainties

Limitations associated with the MARCH computer code modeling have been discussed extensively in the Interim Technical Assessment of the MARCH Code [Rivard et al., 1981]. Some additional limitations have been identified elsewhere [NUREG-0850, 1981], [Greene, 1981]. In an effort to reduce the number and severity of those limitations, personnel at Battelle Columbus Laboratories are collecting improvements to MARCH 1.1 which have been developed at Sandia National Laboratories [Haskin, et al, 1982], Brookhaven National Laboratories [NUREG-0850, 1981], Oak Ridge National Laboratories [Greene, 1981], the Tennessee Valley Authority, and elsewhere, and consolidating these together with their own improvements into a new version, MARCH 2. Since the MARCH modeling limitations are well-documented elsewhere we will not discuss them further here; however, Table 6.2-1 summarizes the modeling changes tentatively planned for MARCH 2 which have the potential for altering the MARCH predictions presented in this report. It is our current belief that these modeling changes will not significantly alter the findings in this report; however, the proposed changes may be more significant for smaller containments and containments with lower failure pressures than Zion. MARCH 2 is currently scheduled to be publicly released by the end of 1982.

6.3 Phenomenological Uncertainties

The accuracy of thermal hydraulic modeling of primary and containment system responses during core meltdown accidents is constrained by our physical understanding. Modeling can only be improved to a level consistent with our current understanding of the underlying phenomenology. This section points out certain areas in which underlying phenomenological uncertainties may limit our ability to develop predictive models. In most cases these are areas of ongoing experimental research.

6.3.1 Radionuclide Chemistry

Radionuclide source terms for reactor accidents have recently been studied by several groups [Nucl. Tech., 1981], [NUREG-0772, 1981]. Based on these references, we have investigated what effects the uncertainties in radionuclide chemistry have on estimated consequences for Zion.

NUREG-0772 examined radionuclide transport to the environment by analyzing state-of-the-art information on the following mechanisms:

Table 6.2-1

Planned Improvements for MARCH2

Program Modification	Limitation Identifier*	Organization Providing Model**
Coding and Structure		
1. MARCH Overlay - Because of size of code it is difficult to run on some machines. This will be particularly true with planned modifications.	---	Battelle
2. MACE Subdivision - Because of the size of this routine, it cannot be compiled on IBM machines.	G3	TVA
3. Routine Interface Flexibility - the rigid sequential behavior assumed in transferring from BOIL to HEAD to HOTDROP will be relaxed, permitting consideration of some phenomena in parallel. (Under evaluation)	G14, TR21, TR22	Battelle
Internal Aids to User		
1. Rewrite in FORTRAN5 - In order to aid installation on different machines, some variations from standard practice will be changed.	---	Battelle
2. ANSI Conversion Factors - For consistency, standard conversion factors will be used in data statements.		SNL
3. Reformatted Output - Convenient output formats will be included as options.	G5, G6	SNL/TVA Battelle
4. Energy and Mass Balance - Contributions to the energy and mass balances will be calculated at various stages of the analysis and provided as optional printout.		Battelle

Program Modification	Limitation Identifier*	Organization Providing Model**
Timestep Management		
1. HOTDROP Timestep Control - This control limits the timestep based on the rates of containment temperature and pressure increase. The need for change will be evaluated.		BNL
Core Modeling		
1. Decay Heat Correlation - Current ANS standard including actinide contribution will be incorporated.	CNT9, CNT10	SNL/ORNL/Battelle
2. Axial Conduction in Fuel Rods.	TR30	SNL
3. Heat Transfer from Partially Covered Nodes - This model smooths the transition from completely uncovered to completely covered. Since axial noding is typically fairly coarse (e.g., 6 inches), this effect could be important during core uncovery.	TR8	Battelle
4. Rod-to-Gas Heat Transfer - Improvements will be made in the convective heat transfer coefficient to gas and in the rod to steam radiation heat transfer coefficient.	TR10, TR11	SNL
5. Parallel Plane Radiation Heat Transfer - This model accounts for radiation heat transport between radial zones and from the periphery of the core to the core barrel.	TR4, TR5, TR19, TR20 SD1	SNL
6. Control Parameter on Zircaloy Oxidation - The FDCR parameter will be redefined to better represent the cladding oxidation in the whole core.	CNT18	Battelle
7. Gas Phase Diffusion Limited Clad Oxidation - A term will be included in the metal-water reaction equations to account for the limited availability of steam to the cladding surface as it diffuses through hydrogen.	CNT27	Battelle

Program Modification	Limitation Identifier*	Organization Providing Code **
In-Vessel Modeling		
1. ECC Pump Curves - The number of ECC pumps that can be characterized with pump performance curves will be increased.		Battelle
2. Debris Bed Model - A debris bed dryout heat flux correlation is used in this model to determine the coolability of a debris bed in the lower head of the vessel.	TR25	BNL
3. Instrumentation Tube or Control Rod Drive Housing Failure - A simple model will be developed to evaluate the timing of local failure of vessel penetrations. The effect of localized failure will be simulated. (Under evaluation)		
4. Boiling Model - An improvement is made in the in-vessel coolant boiling model which reduces the occurrence of steam flow oscillations.	CNT1, CNT26	SNL
5. Fuel Slumping and Grid Plate Failure - In this model the thermal attack and failure of grid plates is analyzed.		BNL
6. PROP Routine for Radiation Heat Transfer - Upgraded physical properties for steam and hydrogen are included in the PROP routine.		SNL
7. Source Pressure for ECCS - The source pressure for the injection phase of ECCS will be made appropriate to the borated water storage tank.	SS4	Battelle

Program Modification	Limitation Identifier*	Organization Providing Model**
In-Containment Modeling		
1. BWR Flowpaths - Flowpaths from the vessel will be modified to permit simultaneous flow from a break to the drywell and through relief valves to a suppression pool.		Battelle
2. Pressure Differential - An option will be incorporated for a break outside containment with atmospheric pressure at the break outlet.		Battelle
3. Cavity Boiloff - Changes to the boiloff routine with control value IHOT=100 will be evaluated.		BNL
4. Heat Transfer to Cavity Structures - This model provides a sink for the transfer of heat by radiation from the molten core to the structures in the reactor cavity.	TR32	BNL
5. Debris Heat Transfer - An alternative correlation for heat transfer from hot debris to water in the reactor cavity is used in this model. A dryout heat flux correlation is used to determine debris bed coolability.	TR12	BNL
6. Hydrogen Combustion - The BURN subroutine is modified to include the burning of CO as well as hydrogen. Changes in the logic of burn propagation will be included if available in time.	CNT12, CNT15, CNT16, CNT17, CNT22	BNL/SNL
7. Steam Properties - ASME steam table values are incorporated in a revised PROPS routine.	G4	TVA
8. Critical Flow Model - The Moody model is encoded in this routine as an optional two-phase break flow model.	TR44	TVA
9. Blowdown Treatment - A more flexible treatment of blowdown is made available in the INITIAL subroutine.		SNL

Program Modification	Limitation Identifier*	Organization Providing Model**
10. Critical Flow of Liquids - The critical flow model in MARCH is inaccurate when the flow through the break is subcooled liquid. The RETRAN model will be incorporated into MARCH2.	TR44	Battelle
11. Fission Product Transport - A routine has been [illegible]loped which provides an approximate means [illegible] tracking fission products as they are released from the fuel and transported through the reactor coolant system and containment.		Battelle
12. Containment Inerting - Flexibility will be included in defining the initial concentration of gases in the containment to represent inerted designs.	SS6	ORNL/SNL

*Limitation identifiers are taken from [Rivard et al., 1981].

**Battelle = Battelle Columbus Laboratories
BNL = Brookhaven National Laboratory
SNL = Sandia National Laboratory
TVA = Tennessee Valley Authority

- release of radionuclides from the fuel during various stages of the accident,
- transport of radionuclides through the primary system to containment,
- transport of radionuclides from containment to the environment.

The document focused on iodine, but discussed other nuclides to some extent.

For accidents involving core melting, NUREG-0772 concluded that releases of radionuclides from the <u>fuel</u> due to core melting may be significantly higher for Te, Sb, Ba, and Sr than releases considered in the Reactor Safety Study. Table 6.3.1-1, taken from NUREG-0772, illustrates this conclusion.

NUREG-0772 examined iodine transport through the primary system in detail. If iodine is released from the fuel in molecular form, I_2, there is essentially no retention if the pathway to containment is dry (no water). If iodine is released as CsI, Table 6.3.1-2, taken from NUREG-0772, indicates that for a dry pathway to containment, as much as 60 percent of the CsI can be retained in the primary system. For a wet pathway to containment, almost all of the CsI would be retained in the water (CsI has an ionic bond and is soluble in water); a substantial amount of I_2 could be retained, the exact amount depending on the pH and length of time that the I_2 is in contact with water.

For severe accidents, the retention of iodine within containment is not appreciably different whether the form is I_2 vapor or CsI particles, according to NUREG-0772.

To estimate the effect of greatly reduced release of the I and Cs group radionuclides to the environment, consequences were calculated assuming no release of these radionuclides. Also, consequences were calculated using the NUREG-0772 fuel melt release fractions, but still with no Cs or I group nuclides released to the environment. Tables 6.3.1-3 and 6.3.1-4 summarizes these calculations. If no I or Cs is released, mean latent cancer fatalities are reduced by a factor of about 2.5, and land interdiction is decreased from 55 square miles to 0 (due to no Cs group release). These are meaningful reductions, but the consequences are not negligible even if no I or no Cs is released.

Until the radionuclide chemistry is better understood, we will continue to model retention of I and Cs in the primary system as negligible. Our rationale is threefold:

- The chemistry is complicated and is accident specific.

Table 6.3.1-1

Fuel Release Fractions

[NUREG-0772, 1981]		[WASH-1400, 1975]	
Fission Product Group	20 Min Release Fraction for AB Sequence	Fission Product Group	Melt Release Fraction
I	1.0	I, Br	0.9
Cs	1.0	Cs, Rb	0.8
Te, Ag, Sb	1.0	Te, Sb, Se	0.15
Ba	0.5		
Sr	0.3	Ba, Sr	0.1
Zr	0.03		
Ru	0.02	Noble metals	0.03
Structure	0.005		
Clad	0.002	Rare earths	0.003
Fuel	0.003		

Table 6.3.1-2

Summary of Predictions of Iodine Distribution Among the Four States at the End of the Accidents Considered (For a dry pathway to containment*)

			CsI Released to Containment		CsI Retained in Primary	
Case	time (s)	I_2 Released to Containment[a] (%)	Vapor[b] (%)	Suspended Particles[c] (%)	Deposited Vapor[d] (%)	Deposited Particles[e] (%)
TMLB'-1[f]	1320	>99	0.4	92.6	6.2	0.7
TMLB'-2	1320	>99	0.4	92.8	6.3	0.5
TMLB'-3	1320	>99	4.3	86.1	9.3	0.3
TMLB'-4	1320	>99	22.5	40.2	37.1	0.1
AD-1/2	600	>99	1.3	80.2	18.0	0.3
AD-1	900	>99	10.8	70.7	16.6	1.6
AD-2	900	>99	11.5	53.7	33.9	0.6
AD-3	900	>99	12.4	26.2	61.1	0.1
AD-4	600	>99	11.3	22.8	51.9	13.8
AD*	800	>99	86.1	13.6	0	0.1

(a) Percent of I_2 mass released from fuel which escapes to containment.
(b) Percent of CsI mass released from fuel remaining in vapor state.
(c) Percent of same deposited on surfaces of suspended particles.
(d) Percent of same deposited on primary system surfaces from vapor state.
(e) Percent of same deposited on system surfaces via particle deposition mechanisms.
(f) Key to sequence designations: 1-base case; 2-large particle source; 3-weak particle source; 4-altered thermal hydraulic conditions; 1/2-delayed ECI; *-hot leg break.

* From NUREG-0772

- All the radionuclides contribute to consequences and a reduction in I and Cs releases alone does not reduce consequences to negligible values.

- The consequence models themselves have an order of magnitude uncertainty in their predictions.

6.3.2 Atmospheric Cleanup by Fan Coolers

We have examined the effects of the fan cooler system in removing fission products from the containment atmosphere.*

During accident conditions, air is drawn through HEPA filters and roughing filters, then cooled by finned cooling coils.† The HEPA filters are designed to remove 99.97 percent of particulate matter from the air stream. However, these filters are not designed for the very heavy aerosol loadings expected in a meltdown accident. Most of the aerosols are produced at the time of and after vessel failure. Our best estimate is that the HEPA filters become overloaded at this time.

After the filters are loaded, our best estimate is that bypass dampers open so that cooling is not interrupted. Some particulates would still be removed by the wet surfaces of the cooling coils. Our best estimate is a 10 percent removal fraction for particles after vessel failure. An optimistic assumption is that the wet tubes are quite efficient and remove 90 percent of the particulates; a pessimistic assumption is that no particles are filtered after vessel failure. An even more pessimistic assumption would be that air flow is completely stopped when the HEPA filters overload. However, based on conversations with Zion plant personnel, a spring loaded damper in the inlet duct work should open due to high pressure drop across the filters. This would enable air flow into the cooling units to bypass the HEPA filters.

When hydrogen burning or substantial steam spikes occur, our best estimate is that the filters are destroyed by differential overpressure across their surfaces.

The condensate on the cooling coils should serve as a trap for iodine, since iodine can be removed by water to an appreciable extent even in its molecular state [NUREG-0772, 1981]. We have taken the minimum iodine removal fraction by cooling coils to be 10 percent. As an optimistic estimate, we assume that 90 percent of the entering iodine is removed. We have not found any documentation of experiments that directly address the removal of iodine

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*The effects of sprays are explicitly considered in the CORRAL calculations.

†No charcoal filters are present [Zion-FSAR].

by cooling coils and a clear-cut best estimate of the removal fraction is not possible. We have taken the best estimate to be 50 percent removal fraction.

Table 6.3.2-1 summarizes assumptions used for cleanup by the fan cooler system. Table 6.3.2-2 shows the results of applying these assumptions to sequence S1HF, a 4-inch diameter LOCA in which the fan coolers operate and the containment leaks at its design rate. The effect of fan coolers (or sprays) in cleaning the atmosphere is of secondary importance compared to their effect on maintaining containment integrity.

6.3.3 Other Phenomenological Uncertainties

The phenomena of core melting and fuel relocation have been characterized as the greatest uncertainties in the MARCH modeling of meltdown accident progression. Analyses of these uncertainties as well as those associated with in-vessel melt-water interactions, termination of severe core damage, pressure vessel breach and materials discharge, melt/water interactions in the reactor vessel cavity, hydrogen generation, transport, and burning, and core-concrete interactions are available elsewhere. [Rivard, 1980], [Haskin et al., 1981], [Rivard and Haskin, 1981], [Rivard et al., 1981], [NUREG-0850, 1981]. This section summarizes many of the findings in these references.

The uncertainty in fuel relocation and deformed core geometry could have a major impact on heat-up and Zirconium-water reaction rates, and could also influence subsequent events and their timing. The liquefaction temperature would be expected to vary over a wide range [Hagen and Malauschek, 1979], [Politis, 1975], depending on the local composition. However, MARCH only allows the use of a single "melt" temperature. The choice of the melt temperature has a significant effect on the progression of melting. Hydrogen generation rates predicted by MARCH might be at times either too high or too low; the total quantity of hydrogen predicted by MARCH before core slump is probably too low. Extensive experimental and theoretical investigations would be required to refine the MARCH models for hydrogen generation.

The MARCH models for introduction of slumped fuel into the water in the lower vessel are not realistic. If the melt is coherent, water boiloff would be limited to the film boiling rate, and steam generation and Zirconium oxidation would be quite slow. If the melt fragments, the rates of oxidation and steaming could be initially high, but would later be limited by the countercurrent flow rate into and out of the particle bed. MARCH allows all the water in the lower head to boil away and residual Zirconium to react in a single timestep. Thus, whatever the form of the melt/water interface, MARCH predictions of steam generation and hydrogen production rates during this phase are probably too high.

Table 6.3.1-3

Consequence Calculations for an Accident With No Containment ESF's in Which Containment Fails at 3.4 Hours*

Fuel Release Fractions: gap-melt-vaporization*	Fractions Released to Environment (CORRAL Calculations)	Mean Man Rem	Mean Latent Cancer Fatalities	Mean Land Area Interdicted, (square miles)
Kr-Xe: .03, .87, .10	1.	4×10^7	2370	55
I-Br: .017, .883, .10	.62			
Cs-Rb: .05, .76, .19	.40			
Te: .0001, .150, .850	.46			
Ba-Sr: 10^{-6}, .10, .01	.04			
Ru: 0, .03, .05	.03			
La: 0, .003, .01	.006			

*Reactor Safety Study [WASH-1400, 1975].

Table 6.3.1-4

Consequence Calculations for an Accident With No Containment ESF's in Which Containment Fails at 3.4 Hours: All I and All Cs Group Nuclides Retained in Primary

Fuel Release Fractions: gap-melt-vaporization Based on [WASH-1400, 1975]	Fractions Released to Environment (CORRAL Calculations)	Mean Man Rem	Mean Latent Cancer Fatalities	Mean Land Area Interdicted, (square miles)
Kr-Xe .03, .87, .10	1.	8×10^6	869	0
I-Br .017, .883, .10	0.			
Cs-Rb .05, .76, .19	0.			
Te .0001, .15, .85	.46			
Ba-Sr 10^{-6}, .10, .01	.04			
Ru 0, .03, .05	.03			
La 0, .003, .01	.006			
Fuel Release Fractions Through Core Melting: Based on [NUREG-0772, 1981]				
Kr-Xe: 1.	1.0	1×10^7	933	0
I-Br: 1.	0.			
Cs-Rb: 1.	0.			
Te: 1.	.55			
Ba-Sr: 0.4	.12			
Ru: .02	.02			
La: .003	.006			

Table 6.3.2-1

Cleanup by Fan Coolers

Assumption	Iodine Removal Fraction	Particle Removal Fraction Before Vessel Failure	Particle Removal Fraction After Vessel Failure	Particle Removal Fraction After Hydrogen Burn or Substantial Steam Spike
Optimistic	0.9	.9997	0.9	.9
Best-Estimate	0.5	.9997	0.1	0.
Pessimistic	0.1	.9997	0.	0.

Table 6.3.2-2

Effect of Fan Cooler Cleanup on Radiological Consequences in S1HF Sequence

Assumption	Mean Man Rem	Mean Latent Cancer Fatalities	Mean Land Area Interdicted (square miles)
Optimistic	8×10^2	0	0
Best-Estimate	2×10^4	1	0
Pessimistic	7×10^4	3	0

The MARCH models do not adequately represent the core geometry following significant melting and fuel relocation, and do not allow for steam explosions or formation of debris beds. Therefore, MARCH cannot predict whether reintroduction of water will terminate an accident after a significant fraction of fuel has melted.

The MARCH model for bottom-head failure and exit of material is neither the only conceivable model nor a particularly reasonable model. The steam generation rate predicted by MARCH after molten material falls into the cavity may be conservative for the time of initial contact; however, if a particle bed forms the later generation rate would be limited to the dryout flux [SAND80-1304]. Also MARCH does not consider steam explosions or other mechanisms which tend to cause fuel relocation; fuel dispersed by such mechanisms might be more readily cooled.

The predicted containment pressure spike at reactor vessel failure is probably too high and rises too rapidly. However, so little is definitely known about the detailed physics of vessel breach, materials discharge, and fuel/coolant interactions that it is not possible to quantify the degree of conservatism in MARCH.

Not all hydrogen sources are accounted for in MARCH, and it is not immediately apparent whether the MARCH calculations presented in this study are conservative or unconservative with respect to hydrogen generation. The MARCH assumption of a uniform, well-mixed containment atmosphere might be conservative in some cases and unconservative in others. MARCH can be used to establish the approximate time when hydrogen burning becomes possible and to bound the resulting containment pressures. However, significantly more sophisticated models would be required to assess the degree of conservatism in the hydrogen burn pressures.

Melt/concrete interaction phenomena are obviously important to the analysis of basemat penetration in meltdown accidents, but such phenomena are important to above-ground containment integrity also. Water vapor and noncondensible gases (CO_2, CO, and H_2) generated in the melt/concrete interactions contribute to containment pressurization. In addition, both CO and H_2 are combustible. The vertical penetration rate and corresponding gas generation rates predicted by the INTER subroutine in MARCH are known to be conservative. The more recent CORCON1 code [Muir et al., 1981] provides more realistic rates; however, neither code is applicable when the debris in contact with the concrete begins to solidify. A preliminary model has been developed to treat the solid penetration regime [Berman et al., 1981]; however, both regimes require more experimental data:

"Gas generation dominates the nature of melt/concrete interactions. Characterization of processes associated with or induced by gas generation should be the focus of future tests. If analytic models of melt interactions with concrete are to be constructed, understanding of gas generation and behavior will be essential." [Powers and Arellano, 1982]

Upward heat transfer from the core debris is an area of significant uncertainty. In a wet reactor cavity, upward heat transfer from debris to water could slow or conceivably arrest basemat penetration. In a dry cavity, radiation heat transfer from the top of the debris would heat the concrete walls of the reactor cavity, resulting in additional releases of water vapor and CO_2.

In open experiments, H_2 and CO have been observed to spontaneously ignite when they emerge from the debris. However, in an actual meltdown accident, such prompt ignition might be precluded either as a result of oxygen depletion in the reactor cavity or due to the presence of water above the debris. The kinetics of the emerging gases under such conditions are not clearly understood.

If aerosols released during melt/concrete interactions are predominately non-radioactive, they would act to reduce the concentration of radioactive aerosols in containment by agglomeration and settling. On the other hand, airborne radioactivity would be increased if melt/concrete aerosols were predominately radioactive. In either case, melt/concrete aerosols may be detrimental to containment heat removal ESF's. The mechanism and rate of aerosol generation during melt/concrete interactions require further experimental research.

7. INSTRUMENTATION TO MONITOR ACCIDENT PROGRESSION

Instrumentation for the Zion Plant consists of the following types: ex-vessel nuclear, engineered safety features actuation, in core, and process. Information from sensors provides input to both control systems and protective systems (Reactor Protection System and Engineered Safety Features Systems) [Zion-FSAR].

Ex-vessel nuclear instrumentation consists of three sets of neutron detectors to measure neutron flux at three different power ranges of the core: source, intermediate, and power. Engineered safety features actuation instrumentation measures temperatures, pressures, flow rates, and liquid levels in the coolant system, steam system, containment building, and auxiliary systems. In-core instrumentation consists of thermocouples to measure coolant temperatures at core-outlet positions, and miniature neutron flux detectors. These in-core sensors provide localized information on temperature and flux at various core positions; they do not provide input to any control or protective systems. Process instrumentation provides information necessary for normal plant operation.

The control systems are not of primary interest for severe accidents and they will not be discussed further in this report. The Reactor Protection System scrams the reactor when allowable operating ranges on neutron flux, coolant temperature, or primary pressure are exceeded; inputs are from ex-vessel nuclear instrumentation and from engineered safety features actuation instrumentation. Engineered Safety Features Systems provide emergency core cooling, containment cooling, and containment isolation upon receipt of the appropriate inputs from actuation sensors.

In Regulatory Guide 1.97, Revision 2 [RG 1.97, 1981], the U.S. Nuclear Regulatory Commission (NRC) defines a set of plant parameters and anticipated ranges to be monitored during light water reactor accidents. The Reg. Guide 1.97 list for PWRs is included in Appendix H. We have not established a need for any additional instrumentation in our analyses of hypothetical severe accidents at Zion. NRC has requested operating plants, such as Zion, to meet the provision of Reg. Guide 1.97, with minor modifications, by June of 1983.

Specific information on all existing Zion instruments was not available to us. We believe the scope of the existing Zion instrumentation should permit the operators to monitor accident progression and determine the plant state. However, some improvization required with the existing instrumentation would be eliminated if all Reg. Guide 1.97 monitoring guidelines were met.

No instrumentation is currently provided at Zion, or at other operating PWR's, for direct measurement of vessel water level or core damage. Two types of indirect measurements have been proposed: heating of core thermocouples during core heatup, and variations in source detector count rate during core uncovery [TMI, 1980].

The thermocouples measure coolant temperature at the top of the core; if the core uncovers, fuel heats up and the temperature measured by the thermocouples will eventually increase. Difficulties in correlating the temperature indicated by the thermocouples to the state of the core make the usefulness of this indirect measurement uncertain at present.

The source detectors are two BF_3 gas-filled proportional counters used to measure low energy neutrons. They are positioned outside the reactor vessel at an elevation of approximately one-quarter of the core height. When the reactor is shutdown, the source detectors detect nutrons produced from a subcritical assembly with a source. The neutron source consists of spontaneous fission from heavy even-even isotopes, (α, n) reactions from low atomic number isotopes, reactor startup sources, and the D(γ, n)H photoneutron reaction which is endoergic with a Q value of -2.2 MeV. The latter source is the most important one [TMI, 1980]. As the water level in the vessel drops, competing effects determine the count rate at the positions of the source detectors. The neutron source strength in the core decreases because less moderation decreases the neutron multiplication factor; however, the attenuation of neutrons in passing through the vessel to the source detectors also decreases as the water shield lowers. Studies have shown that a time plot of source counts can possibly be correlated with core water level [TMI, 1980]. At present, this indirect measurement technique has not been shown to provide unambiguous indication of water level.

At Zion, as at other PWRs, operators focus on flow rates, pressures, and temperatures of the primary and secondary systems in assessing the state of the plant. Such measurements are sufficient before core uncovering, if the operators correctly interpret the information presented to them. Once core uncovering and core damage begin, no detailed information on the progression of the accident is available until the vessel fails. (Containment radiation monitors can provide indication of gross melting.) The containment pressure sensors allow the state of containment to be monitored following vessel breach. Also, sampling of containment atmosphere can provide information on the possibility that H_2 burns may occur.

Table 7-1 describes radiation monitors in containment. Table 7-2 lists dose rates during severe accidents. Existing radiation monitors would be driven out of range by a meltdown accident; however, radiation levels inside containment can be inferred by measuring levels at the outside wall.

The Zion sampling system consists of four intakes inside containment. Filter capsules and sample bottles are located outside containment. Filter papers or air bottles are taken to an in-plant laboratory for analysis. The sample system could be used to detect high radiation or hydrogen levels in containment, although there would be delays in getting readings. The isolation valves for the

Table 7-1

Containment Radiation Monitors

Manufacturer	Type	Range
Westinghouse	G-M	.1mR - 1000 R/hr
NMC	Scint.	.1mR - 1000 mR/hr

Table 7-2

Potential Containment Dose Rates

Accident Type	Dose Rate in Containment	Dose Rate Outside Lower Containment Wall
Terminated before core damage	~20 R/hr	10^{-4} - 10^{-3} R/hr
Terminated before complete melt	~10^6 R/hr	1 - 10 R/hr
Meltdown	~10^7 R/hr	10 - 100 R/hr

air sample lines are manually operated. The containment-air samples would be radioactive; hence, precautions would be required when drawing and analyzing samples.

Information on the amounts of radionuclides released from containment during an accident can be gathered from radiation monitors in areas surrounding the plant. In addition to the area radiation monitors, hand carried or semi-permanently emplaced dosimeters or air samplers could be used in several locations outside containment.

The survivability of instrumentation within containment must be considered. Electrical equipment required to survive the accident has been qualified for a containment atmosphere of 60 psig and 275°F for 10^4 seconds [Zion-FSAR]. Predicted pressures and temperatures during severe accidents are higher, on the order of 100 psig and 300°F for long time intervals if containment heat removal systems have failed, and much higher during hydrogen burns. In our discussion of operator actions in Section 8, we assumed that essential instrumentation would survive. Bonzon [Bonzon, 1977] has tabulated test specifications for many Class IE electrical items. A radiation dose of 2×10^8 rads is common to all. The calculated dose (human equivalent) is approximately 2×10^7 rems in the upper containment. No instrumentation failures are expected as a result of radiation, assuming the Zion instruments meet the test specifications. It should be noted that a radiation detector at Three Mile Island Unit 2 may have failed due to spray actuation during the 1979 accident [Murphy, 1981]. We have not performed a detailed evaluation of the capability of instruments to survive calculated accident conditions. An experimental program is underway at Sandia to test instrument survivability.

All instrumentation will fail when the batteries are exhausted during accidents involving total loss of AC power. Each unit has two batteries, and a third battery is in place for control of the "swing" diesel. Each battery is rated at 996 ampere-hours [Zion-SD]. All five batteries can be interconnected, although this is not a normal procedure. A previous report [Haskin, et al., 1981] pointed out that total instrumentation failure at 4.5 hours is possible, but not inevitable, under conditions of total failure of AC power.

The correlation between parameters measured and the way in which operating personnel receive information and assess plant state based upon these measurements was not addressed in this study. Plant-specific information beyond what was available to us would be required to study this important area.

8. OPERATOR ACTIONS

This section discusses operator actions which might be taken during severe accident sequences. We have only examined operator actions which might logically be taken to terminate or delay accident progression or to reduce potential radiological consequences; although, in some cases, seemingly logical actions can have exacerbating effects. As illustrated in Figure 2.3, the most effective actions for reducing radiological consequences are those which prevent core degradation. Radiological releases are then limited by the activity in the primary coolant. Operator actions to prevent or delay core degradation are discussed in Section 8.1. Section 8.2 discusses operator attempts to terminate core degradation in-vessel. If termination can be achieved before significant melting has taken place, radiological consequences would be limited since gap releases are roughly two orders-of-magnitude less than melt releases. Section 8.3 discusses operator actions to prevent or delay above-ground containment failure. Preserving above-ground containment can reduce potential radiological consequences by roughly three orders-of-magnitude. The containment sprays and fan coolers can be used not only to protect against above-ground containment failure, but also to remove radionuclides from the containment atmosphere; however, removal alone would not lead to order-of-magnitude reductions in radiological consequences. Section 8.5 discusses actions by plant personnel to notify public authorities so that public evacuation and/or sheltering can, if necessary, be initiated. The quantitative impact of operator actions on radiological consequences are discussed in Section 9.

8.1 Actions to Prevent or Delay Core Degradation

So-called "front-end" severe accident sequence analyses deal with actions directed at preventing or delaying core degradation. This contrasts with "back-end" analyses of actions directed at terminating core degradation or mitigating the radiological consequences associated with core-damage accidents.

Front-end analyses for the Zion station have been performed primarily by EG&G [Fletcher, 1980], [Fletcher, 1981] and Los Alamos National Laboratory [Burns, 1980], [DeMuth, 1981]. Some of their results are summarized in this section for the sake of completeness and because there are some interdependencies between front-end and back-end operator actions. For more detail regarding the front-end analyses the reader should consult the cited references.

A front-end operator action event tree for accidents initiated by loss of offsite power (LOP) has been developed by EG&G [Fletcher, 1980]. This operator-action event tree illustrates several front-end actions which could be required to establish and maintain adequate core cooling for LOP and other accidents. Table 8.1-1 is a list of front-end operator actions including those from EG&G's operator-action event tree for LOP-initiated accidents. Note that most of

Table 8.1-1

Possible Front-end Operator Actions*

Manually scram reactor

Manually start diesel generators

Manually load diesel generators

Manually start motor-driven auxiliary feedwater

Manually start turbine-driven auxiliary feedwater

Throttle auxiliary feedwater properly

Open steam-generator atmospheric dump valves (ADVs)

Line up service water supply to auxiliary feedwater

Isolate interfacing-systems break

Manually initiate ECC injection

Minimize use of containment sprays to preserve RWST inventory

Replenish RWST inventory

Switchover from ECC injection to ECC recirculation

Properly throttle ECC flow

Open power-operated relief valves (PORVs)

Isolate PORVs

---

*The automatic response of plant safety systems would obviate the need for many of these manual actions. The appropriate subset of the listed actions would depend on the specific equipment failures involved in the given accident sequence.

these operator actions are aimed (directly or indirectly) at establishing and maintaining adequate core heat removal using the steam generators, the ECCS, or both.

8.1.1 Transient-Initiated Sequences

Restoration of Auxiliary Feedwater

Heat removal via steam generators requires forced or natural circulation of reactor coolant through the primary side, feedwater to the secondary side, and secondary steam discharge through the main steam safety valves or the atmospheric dump valves (ADVs). For transient-initiated sequences such as TML or TMLB there is no break in the reactor coolant system pressure boundary and natural circulation through the steam generators can be established. The ADVs or the main steam safety valves would permit steam to be discharged from the secondary side; however, the postulated loss of all feedwater to the steam generators (Events M and L) eventually leads to secondary side dryout. If auxiliary feedwater can be recovered in time, core uncovering can be prevented even if ECC injection is unavailable (as in the TMLB' sequence). Turbine-driven auxiliary feedwater would have to be established within approximately 1.2 hours or motor-driven auxiliary feedwater would have to be established within approximately 1.3 hours in order to prevent core uncovering in the Zion TMLB' sequence [Fletcher, 1980].

Feed and Bleed

For transient-initiated accidents in which heat removal via the steam generators cannot be established, the emergency core cooling system must be used to replenish primary coolant lost from the primary system after steam generator dryout. The ECCS feeds relatively cold water into the primary system where it mixes with existing coolant. Steam and/or liquid coolant are then "bled" out the PORVs or safety relief valves. For example, in the TML sequence, heat removal via the steam generators is not available, but ECC injectior via the charging pumps, with relief out the PORVs (feed and bleed), could prevent core uncovering. ECC injection would not be actuated automatically until high containment pressure [0.129 MPa (4 psig)] was reached as a result of discharge through the PORVs following steam generator dryout. Such ECC actuation, although delayed, would occur in time to prevent core degradation at Zion [DeMuth, 1981]. The same conclusion was reached for a TML sequence in which only 70 percent of rated ECC flow was available and for another sequence (TMLQ) in which a PORV was assumed to fail open [DeMuth, 1981].

8.1.2 Small LOCAs

In LOCAs, discharge of primary coolant through the break prevents sustained heat removal via the steam generators. For small breaks (S1 and S2), if the steam generator secondaries are isolated as is usually the case, depressurization reduces the primary coolant temperature below the secondary coolant temperature and the steam generators actually act as heat sources. The time required for this temperature crossover decreases with increasing break size. For breaks with equivalent diameters approaching 6 cm (2.4 inches), the primary pressure drops below the accumulator set-point [Fletcher, 1981] thereby increasing the temperature drop between the secondary and primary coolants. Eventually, if the core uncovers, the primary system will again heat up and the steam generators will again act as heat sinks (see Section 4.2.1 and Figure 4.2.1-1). Of course, for large enough breaks, primary coolant is rapidly lost, and any interim behavior of the steam generators as heat sources is relatively inconsequential.

Feed and Bleed

Since heat removal via the steam generators cannot be maintained during LOCAs, core cooling must be achieved using the ECCS. ECCS "feed," with "bleed" out the break, is the intended mode of post-break heat removal. The ECC systems include the accumulators and three sets of pumps which are designed to operate over the entire range of possible post-break primary-system pressures. The charging pumps have the capability to inject water at pressures up to or above the safety-relief-valve set point. The high pressure injection (HPI) pumps have the capability to inject water when primary pressures are less than or approximately equal to the low pressure scram set-point [11.0 MPa (1600 psia)]. The accumulators contain a fixed amount of water which is injected at a rate depending on the primary system pressure when that pressure falls below approximately 4.14 MPa (600 psia). The low pressure injection pumps are capable of injecting water into the primary system only if the primary system pressure is below approximately 1.3 MPa (189 psia) [Fletcher, 1981].

If the automatic response of the ECCS occurs as intended, then core degradation during LOCAs would be prevented by feed and bleed operation at pressures which would depend on the break size. If the ECCS fails then operator action to manually actuate or restore ECC could still prevent or terminate core degradation. To prevent core degradation, ECC would have to be actuated or restored shortly after the onset of a sustained core uncovering. In-vessel termination of core degradation is discussed in Section 8.2.

Delayed Switchover to ECC Recirculation

The charging pumps, HPI pumps, and LPI pumps are designed to provide continuous injection. Initially, suction for ECC injection is taken from the refueling water storage tank (RWST). Upon depletion of water in the RWST, suction for ECC recirculation must be switched to the containment sump. In ECC recirculation, the LPI pumps draw water from the containment sump, and the HPI and charging pumps, if required, are aligned in series with the LPI pumps. In the case of a small LOCA, the LPI pumps would not be required until switchover to recirculation.

Assuming ECC injection operates as intended during a LOCA, it is to the operator's advantage to delay switchover to recirculation and thereby delay the possibility of ECC failure due to recirculation equipment malfunction (e.g., LPI pumps, sump isolation valves, etc.) or operator errors. Also, the circulation of radioactive primary coolant through the auxiliary building would be delayed, perhaps giving plant personnel additional time to checkout recirculation equipment.

Two operator actions are available which could delay the necessity for switchover to recirculation. First, as demonstrated in Section 4, if containment fan coolers are operating, the containment pressure can be adequately controlled without using the containment sprays. Operation of the three containment spray trains alone would deplete the RWST in approximately 45 minutes. Consequently ECC injection from the RWST can be significantly prolonged by minimizing the use of containment spray injection. Second, it may be possible for the operator to replenish the RWST inventory with processed water from the liquid radwaste system. The two evaporator-monitor tanks at Zion could hold up to 16,000 gallons and the two lake-discharge tanks could hold up to 60,000 gallons [Zion SD, 1981]. This compares to a nominal initial RWST inventory of about 350,000 gallons.

Early Switchover to ECC Recirculation

If ECC injection fails following a small break, the time until core uncovery begins would depend strongly on the break size, for example a few minutes for a 15.24-cm (6-inch) break but over two hours for a 2.54-cm (1-inch) break (see Figure 4.2.4-2). Should the fault in the ECCS be associated with the RWST supply only, so that ECC recirculation would still be operable, manual switchover to ECC recirculation could be initiated in time to prevent core uncovering for S2 and some smaller S1 breaks at Zion. At Zion, this switchover would currently have to be performed manually; however, instrumentation and control systems to sense the lack of ECC injection flow and automatically initiate switchover to recirculation are used at other plants.

For either manual or automatic switchover to ECC recirculation, sufficient water would have to be available in the containment sump to permit ECC recirculation to be established and maintained. Adequate sump inventory is potentially a problem because ECC injection failure may be caused by a fault in the RWST supply, and that same fault might also cause containment spray injection failure. The only water in the containment sump would then be that due to primary system blowdown. Results from the S2D calculations, discussed in Section 4.2.4, indicate that before core uncovering begins water would already be overflowing from the sump and containment floor into the reactor cavity. The threshold for such overflow at Zion was estimated to be approximately 57 m^3 (2000 ft^3 or 15000 gal, see Section 6.1). The ECC makeup flow required for a 3.81-cm

(1.5-inch) diameter break, for example, based on RELAP4 [Dearian, 1980a], is approximately 0.1 m^3/s (1600 gpm). Thus, the sump inventory due to primary system blowdown alone would be adequate to permit ECC recirculation to be established and maintained for S2- and smaller S1-initiated sequences.

Bleed and Feed

For breaks smaller than about 6-cm (2.4-inches) equivalent diameter, the primary system does not depressurize to the accumulator setpoint so the charging pumps and/or HPI pumps are required for feed and bleed operation. If both of these ECC subsystems fail, there would be no way to replenish primary coolant loss through the break by feed and bleed because the primary system pressure would remain well above the pressure at which injection via the LPIS pumps becomes feasible. Possible operator actions to depressurize (bleed) the primary system so that the LPIS pumps could be used have been investigated [Fletcher, 1981]. Manual opening of the four ADV's ten minutes after the break was found to result in sufficient depressurization; however, possible water-hammer effects during subsequent LPI feed were not investigated.

8.1.3 Large LOCAs

In large LOCAs, depressurization to the LPI range of operation occurs very rapidly. Very little time would be available for operator action should ECC injection fail. Consequently, ECC injection failure following large breaks has been assumed in this report to lead to core meltdown.

If ECC injection functions as intended, the measures outlined in the preceding section for preserving or replenishing RWST inventory would still be appropriate. The operator would, of course, have less time to take such actions due to the higher ECC injection flowrates associated with large LOCAs.

8.1.4 V-Sequence LOCA

The most important operator action in the case of a V-sequence LOCA is to isolate the break. If the break is not isolated promptly, the motor for the isolation-valve operator may overheat since it is physically located in the vicinity of the postulated break. The ECC pump motors could also fail due to steam flooding unless prompt action is taken. The symptoms of a V-sequence LOCA would be primary-system depressurization accompanied by an increase in temperatures, pressures, and radiation levels in the auxiliary building (rather than in containment).

8.2 Actions to Terminate Core Degradation

Significant uncertainties exist regarding the potential effectiveness of attempts to quench a damaged core [Rivard and Haskin, 1981]. The modeling of in-vessel quenching in MARCH is not mechanistic [Haskin et al., 1981], [Rivard et al., 1981]. Phenomenological and modeling uncertainties affecting predictions of in-vessel quenching are summarized in Section 6. In spite of these uncertainties and the limitations associated with MARCH, some significant observations can be illustrated with MARCH calculations.

Figure 8.2-1 shows the MARCH-predicted containment pressure for the Zion TMLB' accident with reactor coolant makeup restored via the centrifugal charging pumps. Two restoration times are depicted in Figure 8.2-1. Restoration at 145 minutes just precedes initial core melting. Restoration at 160 minutes occurs when approximately one third of the core has melted. For comparison, the containment pressure response for the Zion TMLB' base case (no ECC restoration) is also included in Figure 8.2-1.

When ECC is restored at 145 minutes, just before the onset of core melting, MARCH predicts over 50 percent of the core will melt by 160 minutes but continued coolant makeup will quench all melting by 280 minutes. The containment pressure rise associated with steam generated in this quenching process is clearly evident in Figure 8.2-1. Containment ESF's (sprays and/or fan coolers) were not assumed to be restored with ECC in Figure 8.2-1.

For ECC restoration at 160 minutes, when one-third of the core has melted, the core melting is actually accelerated with respect to the base case calculation (no ECC restoration). This is shown by the relative locations of the containment pressure spikes at vessel breach in Figure 8.2-1. Accelerated melting was also observed in the 145 minute ECC actuation case (52.4 percent of core melted at 160 minutes versus 32.4 percent for the base case). Accelerated melting is predicted because the additional water acts as a source which drives the metal-water reaction. While the uncertainties summarized in Section 6 do not inspire quantitative confidence in the MARCH in-vessel quenching calculations, the possibility of accelerated oxidation and core melting is consistent with phenomenological considerations [Rivard and Haskin, 1981].

Figure 8.2-2 shows the containment pressure responses for the same ECC restoration times but with containment sprays and fan coolers actuated simultaneously with the centrifugal charging pumps. To avoid high containment pressures due to steam generated in the quenching process, a prudent action would be to ensure that containment ESFs are operable before initiating ECC if the core has been uncovered for more than a few minutes. Restoring ECC without restoring containment ESFs could conceivably lead to containment failure due to overpressure for some accidents. The fact that containment failure due to overpressure is not indicated in Figure 8.2-1 for the Zion TMLB' with only ECC restored at 145 minutes is

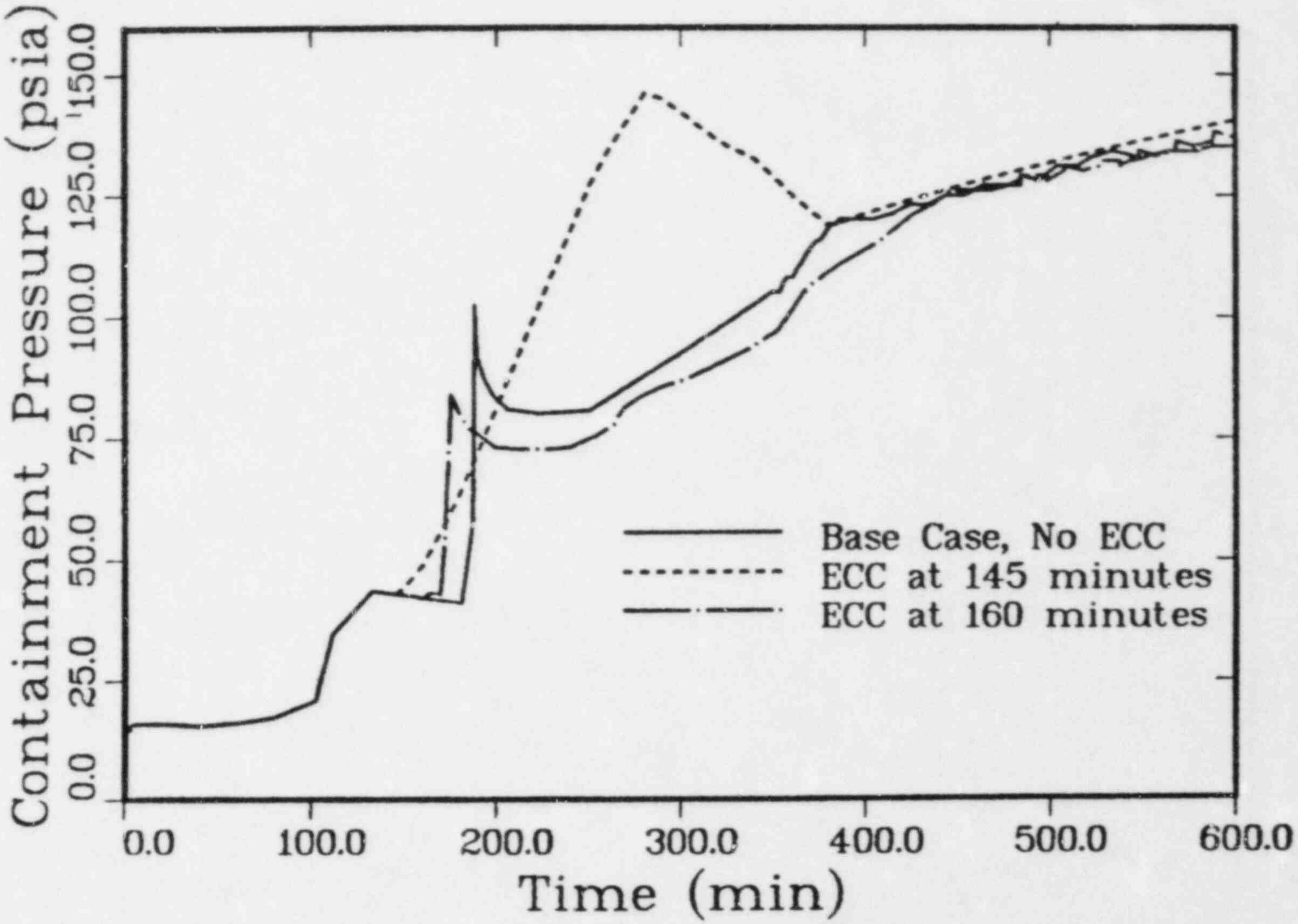

Figure 8.2-1. Zion TMLB' Containment Pressure Versus Time With Restoration of ECC but not Containment ESFs, MARCH 1.1 Calculations

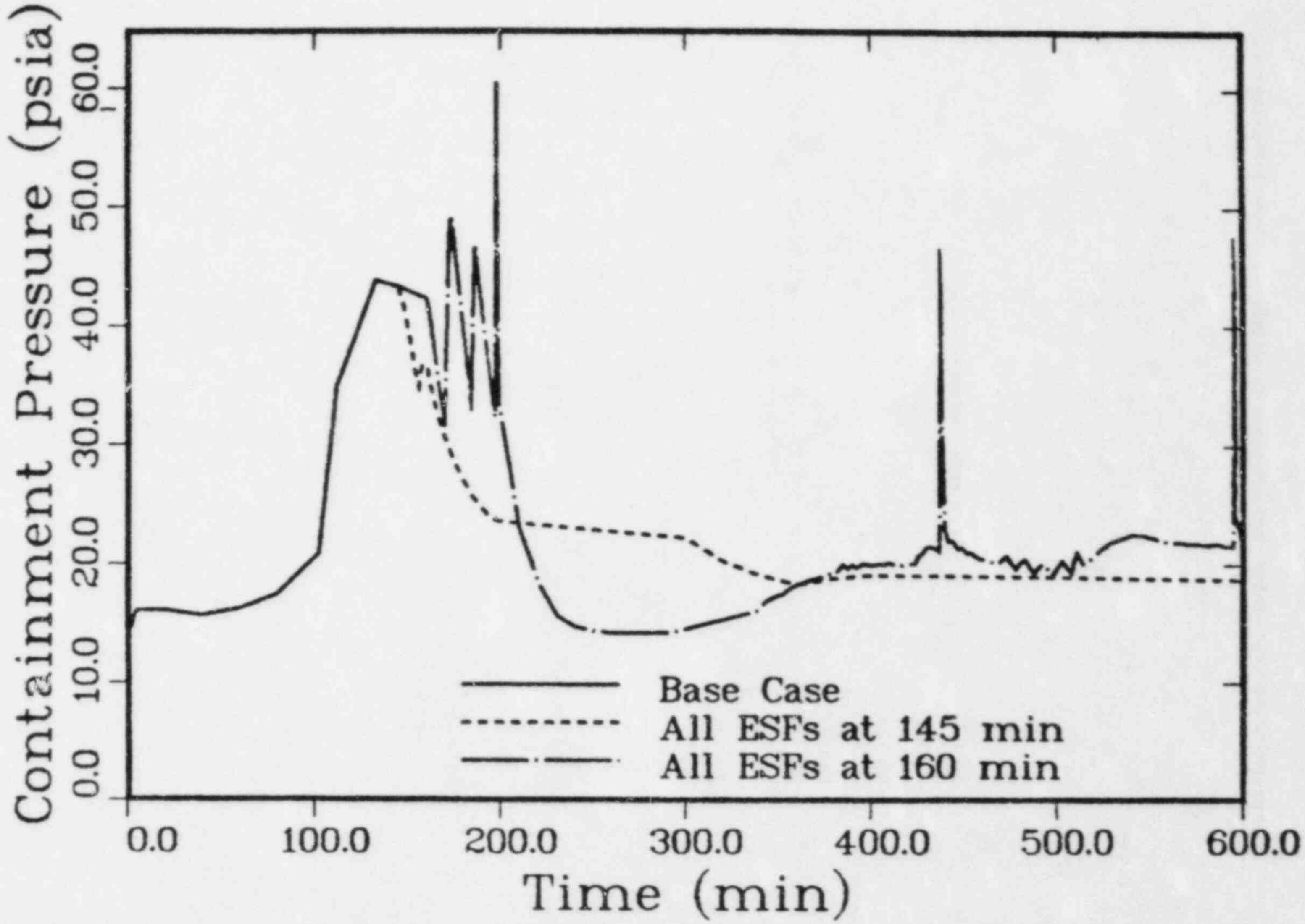

Figure 8.2-2. Zion TMLB' Containment Pressure Versus Time With Restoration of ECC and Containment ESFs, MARCH 1.1 Calculations

consistent with the finding in Appendix C that the earliest time to containment failure due to overpressure at Zion is ~400 minutes.

When containment ESFs and centrifugal charging pumps are both actuated at 160 minutes, several interesting spikes occur in the MARCH-predicted containment pressure response (Figure 8.2-2). A lower containment pressure at the time of core slumping with containment ESFs (Figure 8.2-2 versus 8.2-1 for actuation at 160 minutes) apparently results in a lower MARCH-predicted primary pressure. The MARCH-predicted primary pressure with simultaneous actuation of containment ESFs remains below the shutoff head for the centrifugal charging pumps and the pumps continue to operate after slumping of the core, delaying vessel dryout by ~10 minutes compared with Figure 8.2-1. The first broad pressure spike in Figure 8.2-2 corresponds to vessel dryout. The second broad pressure spike corresponds to breach of the vessel. The subsequent sharp spikes at ~197 minutes, ~437 minutes, and ~595 minutes correspond to MARCH-predicted hydrogen burns. Hydrogen burns are predicted because the condensing of steam from the containment atmosphere by the containment ESF's raises the concentration of H_2 (from metal-H_2O reactions and core-concrete interactions) to the assumed flammability limit (10 percent by volume). However, even with H_2 burns, the containment pressures with containment ESFs (i.e., Figure 8.2-2) are well below the corresponding containment pressures without containment ESFs (Figure 8.2-1). Based on the MARCH calculations in this section and in Section 8.3, the actuation of containment ESFs prior to breach of the vessel seems desirable. However, Section 8.3 illustrates that delaying the actuation of containment ESFs until after breach of the vessel may result in containment failure due to hydrogen burning.

8.3 Actions to Prevent or Delay Above-Ground Containment Failure

Above-ground containment failure could occur due to containment bypass (V-sequence), containment isolation failure (β), steam overpressure (δ), hydrogen (and possibly carbon monoxide) burning (γ), or missiles generated as a result of steam explosions (α). As noted in Section 2.3, neither hydrogen detonations (as opposed to deflagrations) nor internally generated missiles capable of penetrating containment are considered likely at Zion. Operator actions to isolate a V-sequence, interfacing-systems LOCA are discussed above in Section 8.1.4. In this section, we discuss operator actions to assure containment isolation and operator actions to prevent containment failure due to steam overpressure or hydrogen burning.

The design pressure of the Zion containment is 0.43 MPa (47.5 psig) [Zion-FSAR]. However, the design pressure is not a good indicator of the pressure at which the containment would actually fail. Sargent and Lundy [Meyer, 1980] performed an approximate analysis of the Zion containment design and concluded that the most likely failure mode would be hoop-tendon yielding. They calculated an ultimate pressure capacity of 0.93 MPa (135 psia) ignoring the contribution of the liner, or 1.03 MPa (149 psia) including the liner. Los Alamos National Laboratory [Stevenson et al., 1980] calculated a failure pressure of 1.16 MPa (169 psia), although extensive yielding and cracking were predicted at lower pressures. Neither of these analyses explicitly considered failure of penetrations, although the Sargent and Lundy analysis referred to the possibility of failure at the equipment hatch. Based on these analyses, the ratio of the calculated failure pressure to design pressure ranges from 2.17 to 2.72 for Zion.

Figure 8.3-1 shows information characterizing the likelihood of containment failure due to hydrogen burning at Zion. In Figure 8.3-1, the containment pressure existing before a hydrogen burn is scaled along the vertical axis and the amount of hydrogen in containment is scaled along the horizontal axis. The amount of hydrogen in containment is expressed as a multiple of the amount which could be obtained if all metal in the fuel assemblies [26943 kg (59400 lbs)] were reacted with water. On this scale unity corresponds to 100-percent "core" oxidation. Actually, the fuel assemblies contain only 20206 kg (44547 lbs) of zircalloy [Zion-FSAR, Table 3.2.3-1]; the balance of the non-UO_2 mass is predominately steel.

In constructing Figure 8.3-1 it was assumed that the steam in the containment atmosphere was saturated and that the containment was completely isolated. It was also assumed that hydrogen in excess of that from 100-percent core oxidation comes from core-concrete interactions, and that 2.0 moles of CO and 2.5 moles of CO_2 are released with each mole of H_2 from the core-concrete interactions. With these assumptions the partial pressure of steam is equal to its saturation pressure at the temperature of the containment atmosphere, and the partial pressures of noncondensable gases (O_2, N_2, H_2, CO, and CO_2) can be approximated for a given temperature using the ideal gas law. In this manner, the locus of pressures corresponding to fixed H_2 or steam concentrations can be plotted as a function of H_2 in containment.

The line labeled 4 percent H_2 corresponds roughly to the upward flame propagation limit. To the left of this line hydrogen burns are not likely. The line labeled 9 percent H_2 corresponds roughly to the downward flame propagation limit. The line labeled 56 percent steam $+CO_2$ is used to approximate the region (above) where hydrogen burning would be suppressed.

For concentrations below 4 percent H_2, hydrogen combustion can start near ignition sources, but the flame will not propagate far. For mixtures between 4 percent H_2 and 9 percent H_2, in quiescent atmospheres, flames propagate upward, not in a continuous front,

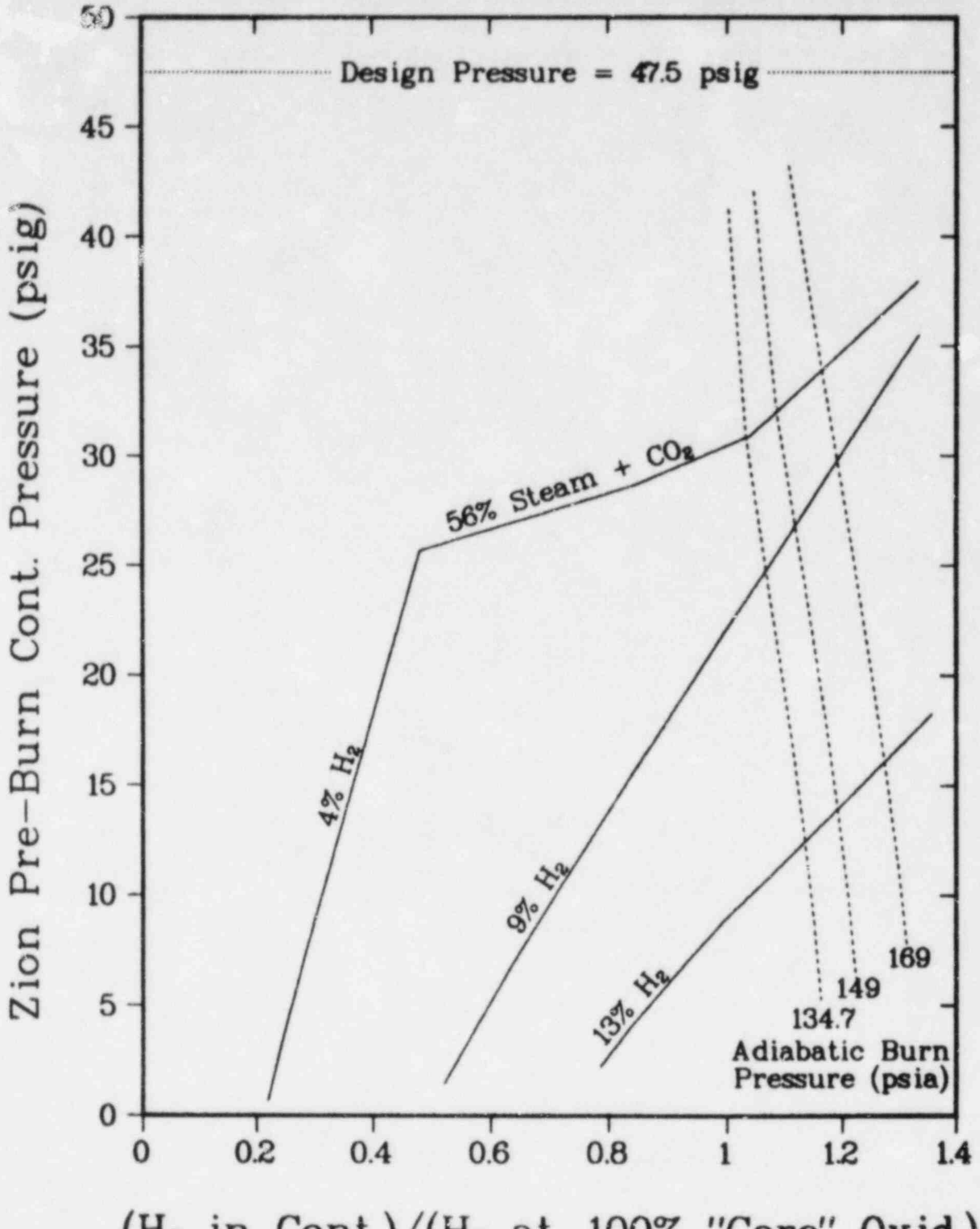

Figure 8.3-1. Containment Failure Estimates and Approximate Hydrogen Deflagration and Detonation Limits for the Zion Containment

but in "balls of fire." Such combustion is known to be incomplete, with low fractions of the hydrogen burned for mixtures below 8 percent hydrogen. If the atmosphere is in motion, the degree of completeness of hydrogen burning can be much higher. Even without fans or sprays, because of the large size of containment and the gas motions induced by flame-generated turbulence, lean burns will probably consume a higher fraction of the hydrogen than found in quiescent atmosphere tests. Complete burns of hydrogen (and CO when present) would be expected to the right of the 9 percent H_2 line; however, until the first dashed curve is reached, the adiabatic pressure rise associated with such burns would not be sufficient to result in a containment pressure of 134.7 psia. Similarly, complete burns initiated at points to the left of the second dashed curve would not result in containment pressures in excess of the best-estimate threshold for containment failure (149 psig). The line labeled 13 percent H_2 is used to approximate the region (below) where propagation of the H_2 detonation becomes possible.

Figure 8.3-1 will be referred to in the following subsections which discuss actions to prevent or delay above ground containment failure.

8.3.1 Manual Assurance of Containment Isolation

Manual assurance of containment isolation could be required in some accidents. For example, in the TMLB' sequence, AC power which is required for some isolation valves is not available. One indication of a containment isolation failure would be high auxiliary building radiation levels due to leakage; however, such an indication might be delayed and would not necessarily pinpoint the location of the isolation failure. The most probable candidates for isolation failure and subsequent leakage would be isolation valves on HVAC penetrations where direct leakage from the containment atmosphere is possible.

8.3.2 Actuation of Fan Coolers

Four of the five fan coolers at each Zion unit operate during normal power operation. When an ESF signal is generated on high (>4 psig) containment pressure, the fan speed is reduced, but the fan coolers continue to operate. With four fan coolers operating as intended, containment failure due to overpressure would not occur. In fact, Appendix C demonstrates that if just one fan cooler is actuated, even after a considerable delay during which the containment pressure approaches the threshold for overpressure failure, containment overpressure failure can be prevented at Zion.

Furthermore, if four fan coolers operate as intended, containment failure due to hydrogen burning is not likely. Pre-burn containment pressures would then be under 25 psig and steam concentrations would not be sufficient to suppress hydrogen burns;

however, a hydrogen burn would most probably be ignited between 8 and 10 mole percent hydrogen and the resulting peak containment pressure would not exceed the lower threshold for containment failure. That is, ignition would occur to the left of the first dashed curve in Figure 8.3-1.

Suspended aerosols released to the containment atmosphere during the core meltdown or subsequent core-concrete interactions could easily plug filters in the fan cooler inlet lines [NUREG - 0772, 1981]; however, based on conversations with Zion plant personnel, a spring-loaded damper in the inlet ductwork should open due to high pressure drop across the filters so that the cooling capability of the fan coolers would be preserved. We did not have access to fan cooler, filter, damper, or ductwork design details or qualification data so that we cannot preclude the possibility of common-mode failure of the fan coolers as a result of filter blockage or other effects of the severe, post-meltdown containment environment. Should the fan coolers fail, the operators would have to rely on the containment sprays for long-term cooling (see Section 8.3.3).

If both containment sprays and fan coolers fail, as in the TMLB' sequence, the resulting containment pressures would be high enough that hydrogen burning would be steam suppressed thereby allowing hydrogen to accumulate.* Figure 8.3.2-1 shows the containment pressure response for a TMLB' accident in which fan coolers are restored after vessel breach, six hours after the initiating event. This point would occur on Figure 8.3-1 at a relative hydrogen inventory of 1.05. Removal of steam from the containment atmosphere by the fan coolers lowers the containment pressure (along a downward sloping path on Figure 8.3-1) until hydrogen burning is no longer steam suppressed. Should a uniform, complete hydrogen burn then occur, the resulting pressure rise would result in containment failure as shown on Figure 8.3.2-1. Under such conditions, if only the fan coolers are available, the operator should keep the containment pressure high enough to ensure steam suppression. The containment atmosphere is still potentially hazardous, even when inerted by steam. Natural condensation processes are beyond the operator's control and could eventually reduce the steam fraction to a combustible level. Steam inerting should be considered a temporary measure that allows more time for carrying out other actions such as Halon inerting or reducing the burden of hydrogen in containment. Steam inerting also allows more time for evacuation of the surrounding population.

The Lawrence Livermore National Laboratories recently tested combustion properties of hydrogen using glowplug devices in a 2-foot diameter cylindrical vessel with rounded ends. In two tests combustion was initially steam inerted and the steam fraction was

*The hydrogen recombiners would not function with hydrogen concentrations in excess of 4 mole percent without modifications. If they could be actuated under such circumstances, they would not have sufficient capacity to arrest the buildup of hydrogen from core-concrete interactions (Appendix G).

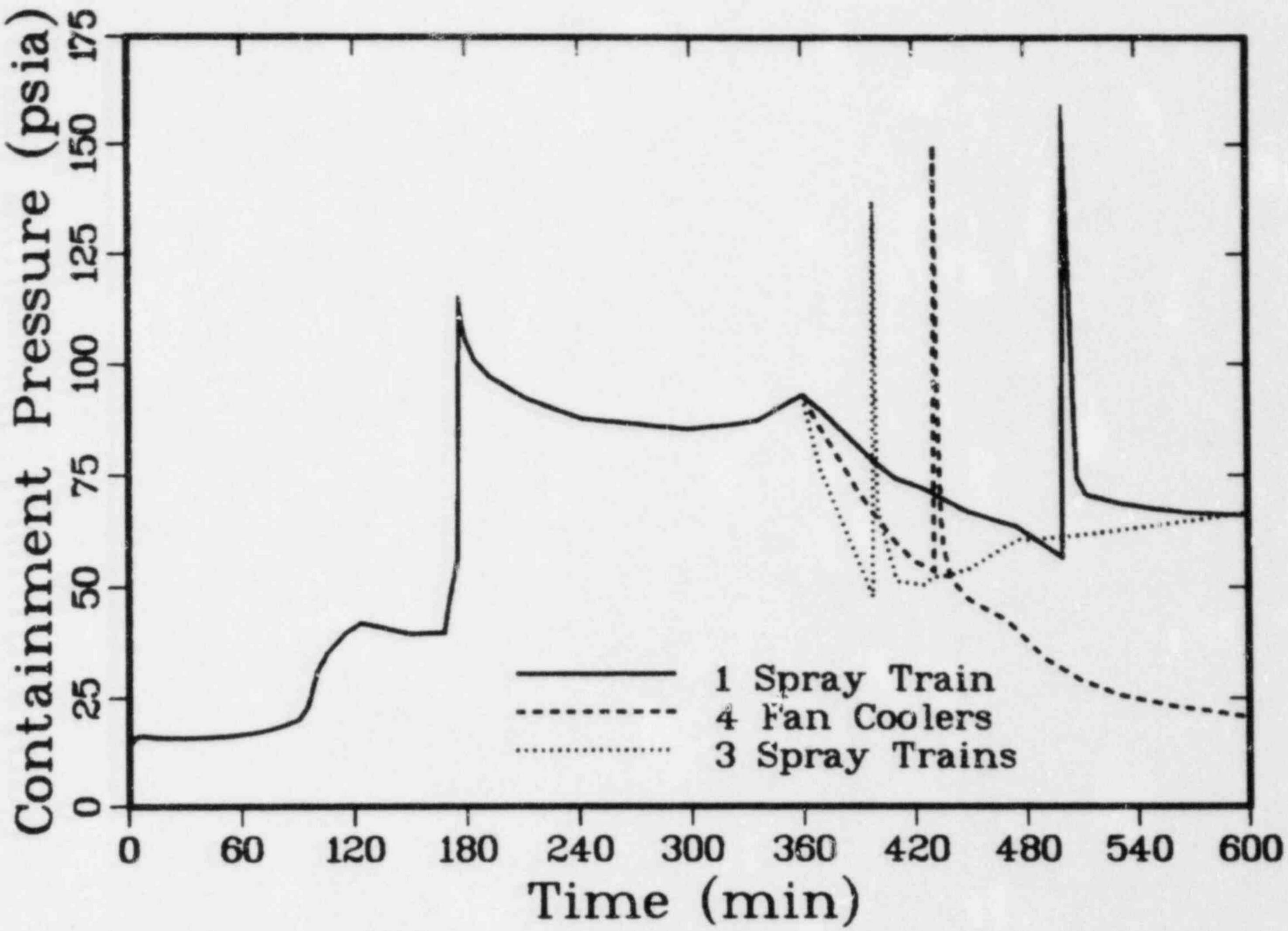

Figure 8.3.2-1. Zion TMLB' Containment Pressure Versus Time With Containment Cooling Restored After Vessel Breach, MARCH 1.1 Calculation

gradually reduced by condensation. The steam fraction was reduced to levels where combustion should no longer have been steam inerted; however, no substantial pressure increase due to hydrogen burning was recorded. These observations have not yet been satisfactorily explained [NUREG-0850, 1981]. If a reliable physical basis can be established which would preclude hydrogen burning in containment under similar circumstances, the above recommendation to maintain steam inerting could be withdrawn.

8.3.3 Actuation of Containment Sprays

The containment sprays, like the containment fan coolers, are easily capable of preventing containment failure due to steam overpressure. However, as explained in Section 8.1.2, for accidents in which ECC injection is operable, it is desirable to preserve RWST inventory by minimizing use of containment spray injection. Also, operation of containment sprays in the recirculation mode, in particular after vessel breach, requires highly radioactive liquid from the containment sump be circulated through the auxiliary building.

Containment sprays are efficient scavengers of aerosols from the containment atmosphere. Prior to fuel-cladding failure, the atmospheric fission product burden is relatively low. However, a marked increase in atmospheric fission products is expected at the time of cladding failure. Operation of sprays should be initiated at that time, if possible, to remove radioactive material from the atmosphere. Spray operation should be continued as long as possible to remove the additional fission products released during fuel melt and fuel-concrete interactions.

Figure 8.3.2-1 shows the MARCH-predicted containment pressure for TMLB' sequences in which containment sprays are restored after vessel breach, six hours into the accident. Numerical results for the runs depicted in Figure 8.3.2-1 are included in Table 8.3.3-1. The magnitude of the peak pressure predicted by MARCH 1.1 for these calculations is almost entirely correlated with the amount of H_2 present at the time of the burn. The more rapid the pressure decrease before the burn, the less time available for H_2 buildup from the core-concrete interactions, and the lower the MARCH 1.1 predicted peak pressure. When ignitable amounts of hydrogen have accumulated in the containment atmosphere, the containment sprays may offer an advantage over the fan coolers. The containment sprays may suppress the pressure rises associated with hydrogen burns as some of the energy released is consumed in evaporating spray droplets. Spray pressure rise suppressions exceeding 50 percent have been reported [Carson et al., 1973]. However, such pressure suppressions are not predicted by MARCH 1.1. To obtain realistic estimates of peak pressure suppression due to containment sprays, we used the Sandia National Laboratories HECTR (hydrogen event: containment transient response) computer code [Cummings et al., 1982]. Figure 8.3.3-1 illustrates that the peak pressure due to hydrogen burning, with containment sprays operating, decreases rapidly with increasing burn time. For a 0.1 minute burn time, such

Table 8.3.3-1

MARCH 1.1 H_2 Burn Results for TMLB' Sequences With Restoration of Containment Cooling After Vessel Breach.*

Components Restored	Restoration Time (min)	H_2 Burn Time (min)	H_2 Burned* (lb)	Peak Pressure† (psia)
Three (3) Spray Trains	360	398.17	3163	138
Four (4) Fan Coolers	360	430.91	3439	150
One (1) Spray Train	360	501.17	3797	164
Four (4) Fan Coolers	428	500.85	3801	162
Three (3) Spray Trains	458	500.87	3808	161

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*Complete H_2 burns initiated at < 56% H_2O (g), > 9% H_2, > 5% O_2.

†In situations where containment sprays are operating, the MARCH 1.1 results for Zion do not indicate any reduction in the peak pressure associated with a hydrogen burn. Apparently this is because the MARCH 1.1 logic precludes spray droplet evaporation when the spray droplet fall time exceeds the code's internal time step [Wooten and Avci, 1980].

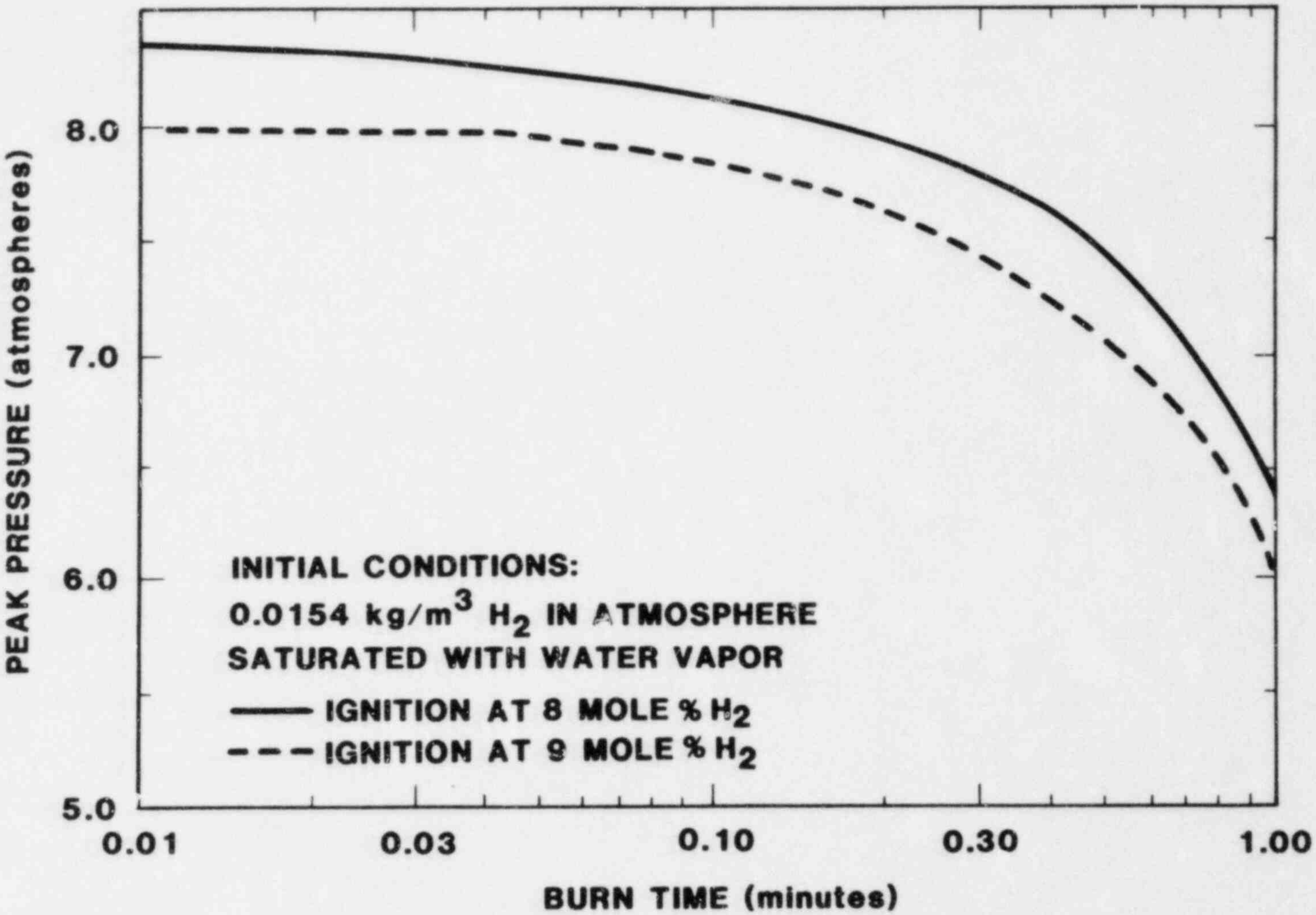

Figure 8.3.3-1. Effect of Hydrogen Combustion Burn Time on the Single Compartment Peak Pressure Predicted by HECTR for Zion With Three Containment Spray Trains Operating.

as occurred at Three Mile Island Unit-2 [Hertzberg, 1981], the predicted peak pressure with three spray trains operating exceeds 97 percent of the corresponding adiabatic, isochoric burn pressure for Zion. As the burn time approaches 1.0 minute, more time is available for spray droplet evaporation during the burn, and the peak pressure is substantially reduced. However, such long burn times imply a lack of turbulent mixing which may not be achievable in a large, dry containment.

8.3.4 Venting Containment Before Core Uncovering

Air in the Zion containment free volume contains roughly twice as much oxygen (40,000 lb) as would be needed to react with the hydrogen (2600 lb) from 100 percent core oxidation; that is, the amount of hydrogen in the Zion containment could accumulate to approximately 2 on the horizontal scale of Figure 8.3-1 before combustion would be oxygen-limited. In an attempt to reduce the potential for containment failure and associated radiological consequences, one might vent the containment, thereby reducing its oxygen content, prior to the buildup of extremely high radiation levels in the containment atmosphere, that is, before core uncovering.

Figure 8.3.4-1 shows the containment pressure for a TMLB' sequence in which the containment is vented and re-isolated prior to core uncovering. Venting was initiated immediately after steam generator dryout and terminated just before the onset of core uncovering. Such venting reduced the predicted oxygen content in containment by a factor of 4.2 for a 2-foot diameter vent opening. For a 10 inch diameter opening, an oxygen-reduction factor of only ~1.3 was predicted.

It should be noted that each Zion unit has a 3 foot diameter purge-supply-air penetration and a 2 foot diameter purge-exhaust-air penetration. The purge-exhaust-air containment-isolation signal could be manually overridden at Zion; however, the existing purge system has not been designed for high pressure, high flow venting to the unit vent. Design considerations associated with backfitting a system for anticipatory venting at Zion are discussed in a previous report [Murfin, 1980]. The purging of noncondensables from containment would require that vacuum breaking capability be included in the system to preclude containment failure to to subsequent spray actuation or long-term condensation.

The TMLB' accident depicted in Figure 8.3.4-1 is assumed, as in the TMLB' base case, to proceed through vessel breach and core-concrete interactions. Due to the lower pressure before vessel breach in Figure 8.3.4-1, the pressure spike associated with discharge of material into the reactor cavity is considerably lower than for the TMLB' base case. Also, upon restoration of the fan coolers (assumed to occur in Figure 8.3.4-1) or containment sprays after vessel breach, the containment pressure can be reduced without risking a hydrogen burn capable of failing containment. Containment venting before core uncovering could, of course, only be used in

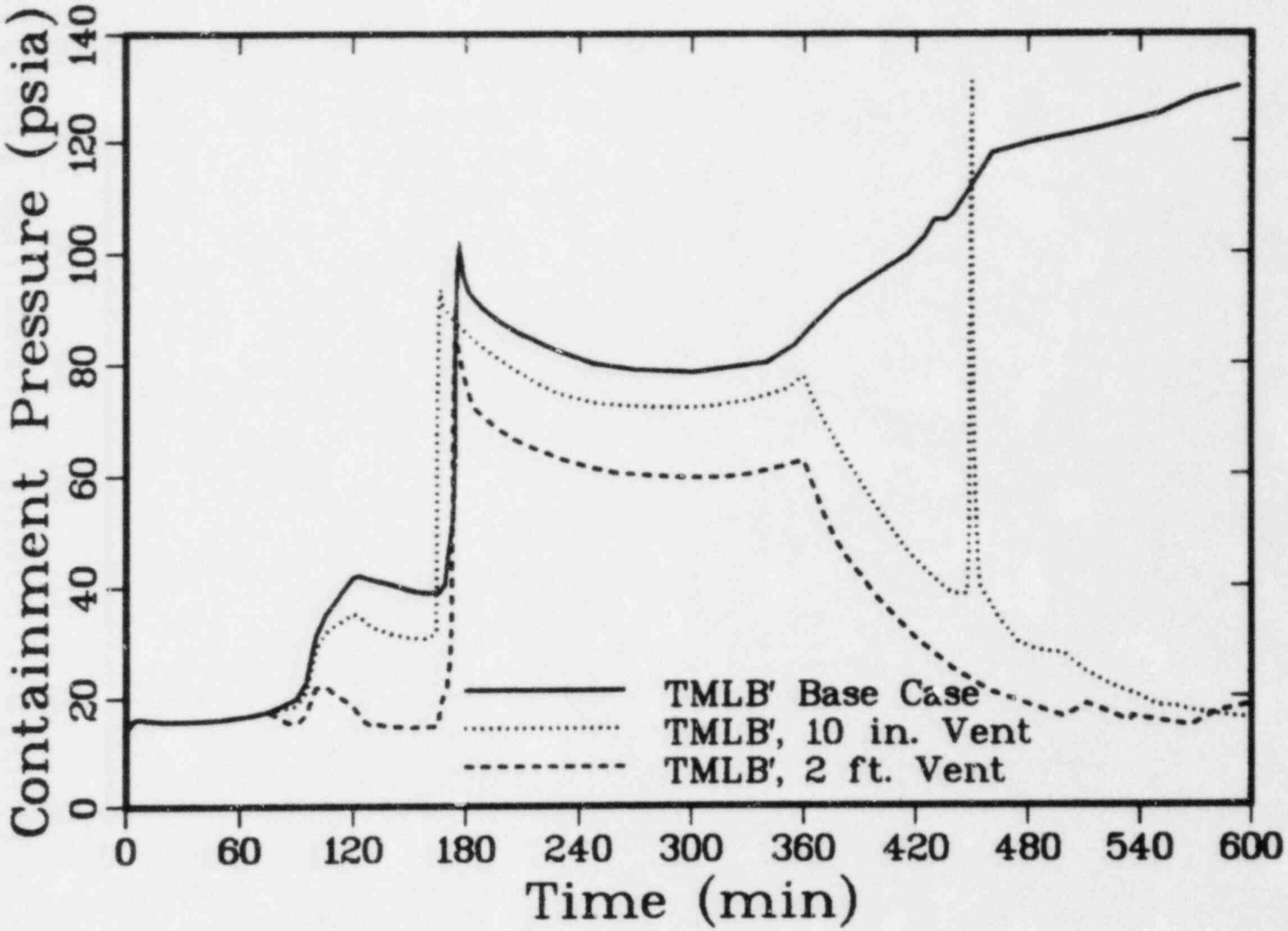

Figure 8.3.4-1. Zion TMLB' Containment Pressure Versus Time With Containment Venting Before Core Uncovering, MARCH 1.1 Calculation

those accident sequences in which core uncovering was sufficiently delayed to allow time for diagnosis and action on the part of the operator. The operator would have to conclude that core uncovering and meltdown were imminent before initiating such action. He would be releasing some primary coolant activity in an effort to substantially reduce the likelihood of a much larger release later. The radiological consequences associated with venting before core uncovering are discussed in Section 9.

9. RADIOLOGICAL CONSEQUENCES WITH MITIGATING ACTIONS

For accidents which proceed beyond core uncovering, two types of mitigative strategies have been investigated: cooling a slightly distorted core in-vessel, and preserving containment once the vessel is penetrated. As indicated in Section 6.3, there are many uncertainties associated with cooling a highly distorted core geometry. There is a time period during a severe accident, lasting from the beginning of significant core damage until vessel failure, in which no credit for cooling the core has been taken. The time at which the core geometry becomes significantly distorted, so as to preclude cooling, is accident specific; also, phenomenological uncertainties preclude an exact prediction of this time. We have assumed that restoration of core cooling just before initial core melting can cool the core in-vessel. Because there is no direct measurement of the beginning of core melting, operators must rely on indirect measurements for indications of core damage. For example, significant increases in containment radiation level would occur due to cladding failures and later, due to core melting.

We feel that in-vessel cooling merits consideration for all accident sequences in which there is enough time between accident initiation and the start of core melting for restoration of cooling to be possible. This mitigative option applies to all the sequences analyzed except large LOCAs in which the time between the pipe break and the start of core melting is only about 6 minutes.

The effects of in-vessel cooling on radiological consequences have been examined for two accident sequences. For the TMLB' accident, if power is restored between the time of core uncovering (~127 min) and the beginning of core melting (~146 min), in-vessel cooling may be possible. For the interfacing systems LOCA, if the break is isolated before the beginning of core melting (114 minutes), in-vessel cooling may be possible. However, such late isolation could be precluded due to the effects of steam flooding on the isolation valve's motor operator. This steam flooding could also result in failure of the (ECC) emergency core cooling function because the ECC equipment is located in the same region of the auxiliary building. High radiation levels in the auxiliary building (~10^3 rem/hr) due to expulsion of primary coolant, coincident with low containment pressure and radiation levels, should alert operators to isolate the break early in the accident before the motor operator is damaged and before the core uncovers (at~101 minutes).

Mitigation of radiological consequences after vessel breach requires cooling of containment and containment isolation. Pressure sensors provide information on containment pressure, and sampling of the containment atmosphere provides delayed information on combustibility.

Table 9-1

Effects of Mitigative Actions on Radiological Consequences

Accident Sequence	Mitigative Action	Mean Man rem	Mean Latent Cancer Fatalities	Mean Land Area Interdicted, Square Miles
TMLB'	None-Containment Fails due to Overpressure	2×10^7	1210	23
TMLB'	Containment Cooling Restored	6×10^4	3	0
TMLB'	Core Cooling and Containment Cooling at 145 min*	6×10^3	0	0
V	None	4×10^7	2070	65
V	Break Isolated and Core Cooling at 114 min*	2×10^7	822	18
V	Break Isolated and Core Cooling Before Core Uncovery -- Only Primary Coolant Released	1×10^4	0	0
SICG	None-Containment Fails After ~18 Hours	2×10^7	840	16
SICG	Containment Cooling Restored	9×10^4	4	0

*Cooling is assumed to be restored just before core melting begins; however, in-vessel termination is predicted by the MARCH code only after substantial (~50 percent) melting has occurred.

As discussed in Sections 4 and 8, if containment cooling is not available early in an accident but is recovered later, cooldown must be carefully controlled to avoid H_2 burns which could threaten containment integrity.

Table 9-1 indicates reductions in radiological consequences for successful mitigative actions. HF and D sequences, not listed in the table, already provide mitigation through containment cooling.

10. IMPLICATIONS OF RESULTS

10.1 Instrumentation

Prior to core uncovering, existing plant instrumentation is adequate to follow accident sequence progression and assess the plant state. Between the time of core uncovering and vessel breach, no instrumentation for directly assessing core damage is provided; this is true of all operating reactors (see Section 7). Containment radiation monitors may provide enough information for operators to decide when significant core degradation begins.

The state of containment can be monitored continuously with existing instrumentation augmented by sampling and chemical analysis of containment atmospheric contents. Mitigative strategies which rely on cooling containment can be applied using existing instrumentation at Zion; however, gathering and analyzing containment air samples is time consuming. Containment high-range radiation monitors having ranges up to 10^7 or 10^8 rads/hr are desirable. Hydrogen monitors would also be useful, although currently available instruments may not be completely adequate [Neidel et al., 1981].

Although instrumentation is available for all the required information, gathering the data would sometimes be awkward or difficult.

Because we did not have sufficiently detailed information, we were unable to evaluate the survivability of existing Zion instrumentation during severe accidents. However, survivability is achievable, with commercially available instrumentation, for monitoring of many important parameters such as: containment pressure, containment temperature(s), containment radiation level(s), and containment water level(s). Experimental research is underway at Sandia National Laboratories to investigate equipment survivability during severe accident conditions including those imposed by hydrogen burns.

10.2 Operator Preparedness

We have reviewed the symptom-oriented procedures being developed by reactor vendors. They do not address situations in which substantial core melting has occurred. We recommend that procedures and training be instituted to enable operators to reduce radiological consequences during such accidents. The philosophy of coping with severe accidents differs somewhat from that of coping with design-basis accidents. For design-basis accidents, the primary goals of operator actions are:

1) keep the core covered with water
2) stabilize primary pressure
3) provide controlled cooldown.

Containment heat removal and containment isolation are of secondary concern and are considered only as they impact the above three goals. This approach is valid for accidents in which substantial core melting cannot occur. The amount of fission products in the containment atmosphere is then relatively small and the philosophy of focusing on core cooling is consistent with that of reducing radiological consequences.

In contrast, during severe accidents, a primary objective of operator actions should be to preserve containment integrity. Substantial core melting can release great quantities of fission products to the containment atmosphere. Cooling the core after this point has been reached will not in itself greatly reduce the radionuclide source term within containment. We cannot demonstrate that core cooling is possible following the initial onset of core melting; furthermore, without containment cooling, core cooling contributes to raising containment pressure.

A further objective of operator action should be to delay containment failure as long as possible. Delay allows more opportunity for intervention and also allows reduction of the contained burden of radioactivity via decay and natural removal processes. For example, we recommend careful control of containment cooling in the TMLB' sequence if power is restored after core melting and buildup of hydrogen to levels which could fail containment if ignited. The purpose of control of containment cooling is to maintain a steam concentration high enough to inert the atmosphere against hydrogen combustion. Concomitant with cooling control, intervention is required to provide more permanent inerting or to dispose of the hydrogen.

Operator actions for reducing the radiological consequences of severe accidents should have the following goals:

1) isolate containment
2) preserve above-grade containment integrity by careful use of containment heat-removal systems
3) if containment heat removal is absent, attempt core cooling only if water is available in amounts comparable to the refueling water storage tank inventory.*

It is beyond the intent of this report to develop detailed procedures for coping with severe accidents. It is important that such procedures be integrated with existing design-basis accident procedures to ensure that conflicting recommendations are not presented to plant operators.

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*The effect on containment of adding water to a degraded core depends on the amount of water added. See Appendix C.

10.3 Systems Design

Options for reducing radiological consequences must focus on reducing the likelihood of core uncovering and on preserving containment integrity. This study focuses on mitigating the effects of meltdown, and hence the latter objective is examined.

Zion has a strong containment with adequate cooling through the use of sprays or fan coolers. No additional cooling capability is deemed necessary. An instrument that can measure containment atmospheric contents would be useful for avoiding H_2 burns during certain cooldown conditions. If practical, a containment isolation system that is independent of AC power would be useful in mitigating consequence; however, procedures for manual isolation may suffice.

Zion has one diesel-driven and two motor-driven containment spray pumps; however, the diesel-driven spray pump cannot automatically provide water to the spray headers in the event of a loss of AC power. The plant operators would, at minimum, have to manually open a motor-operated valve in the discharge line from the diesel-driven spray pump. Furthermore, AC powered cooling water to the diesel engine cooler would not be available; so that, if continuous operation of the diesel-driven pump were attempted, the diesel would eventually fail due to overheating. It is conceivable that the operators could achieve intermittent or limited continuous operation of the diesel-driven spray train during severe accidents involving loss of AC power; however, the current design does not provide the diversity one might intuitively associate with such a three-train system.

A fairly low-cost option for accident management through both the "front-end" and "back-end" regions would be an accident-process computer. This has been strongly suggested by other studies [Kemeny, 1979]. An accident-process computer would sense plant variables, realize that something was amiss, diagnose the cause of the accident, and suggest optimum strategies. The course of the accident could be followed and important events and changes of strategy would be signalled. For example, when the computer sensed total AC power loss and failure of auxiliary feedwater, it would tell the operator to manually operate turbine-driven feedwater. If manual operation failed, the computer would alert the operator that sequence TMLB' had started, and tell the emergency director to inform public authorities of a general emergency. The accident-process computer would also tell the operator what not to do. Some readings might have to be manually entered, for example, containment-air analyses, but the number of manual entries should be kept to a minimum.

10.4 Emergency Response and Accident Management

Appendix A summarizes guidelines for emergency action levels [NUREG-0654, 1980]. For accidents in which core melt is imminent, a general emergency exists. There is much debate over the value of evacuation versus sheltering.

For most accident sequences at Zion, containment failure is estimated to not occur for many hours into an accident; thus, evacuation in a general emergency may be effective. If at any time during the evacuation containment integrity is unexpectedly compromised, we recommend that switching from evacuation to sheltering be considered. For example, if an evacuation is in progress prior to the time of vessel breach, and a steam explosion occurs which fails containment, the evacuating populace could be instructed to shelter in their cars or surrounding buildings. The option of switching from evacuation to sheltering must be considered because, even though our best estimate is that containment will not fail early in an accident, phenomenological uncertainties during the core-melt-to-vessel-breach phase of the accident mean that there is a nonzero probability that early containment failure may occur.

For accidents in which containment is bypassed, sheltering downwind of the site should be considered instead of immediate evacuation.

The plant provides information to local authorities and to the NRC. The decisions as to what specific steps to take to evacuate or shelter people depend on the logistics of communication and transportation within the community; these decisions are made by the appropriate authorities, not the plant management.

A logic diagram which indicates how plant readings support decisions on evacuation is given in Figure 10-1. If containment integrity is maintained but isolation is not achieved, the decision to shelter rather than evacuate depends upon specific meteorological conditions and on logistics of evacuation. We are unable to provide guidance for this situation, other than to recommend training operators to manually isolate containment.

The emergency action levels [NUREG-0654, 1980] are consistent with the results of this study. We have found nothing to justify revising the guidelines. Some of these Emergency Action Levels are tied to the EPA Protective Action Guidelines [EPA-520, 1975], that is, to individual doses at the plant boundary. As was noted in Section 7, doses near the plant have an inherent uncertainty of an order-of-magnitude or more due to meteorological conditions. It is therefore impossible to know whether the Protective Action Guides will be exceeded without knowing the weather, the time of release, and the amount to be released. However, if the release is no worse than design leakage, there is essentially no chance of exceeding the Protective Action Guidelines at the plant-exclusion radius (400 meters). On the other hand, if the containment fails by overpressure, there is a virtual certainty of exceeding the Protective Action Guidelines to distances of several miles. Between these two extremes, an estimate of the amount likely to be released and good knowledge of current weather conditions are needed.

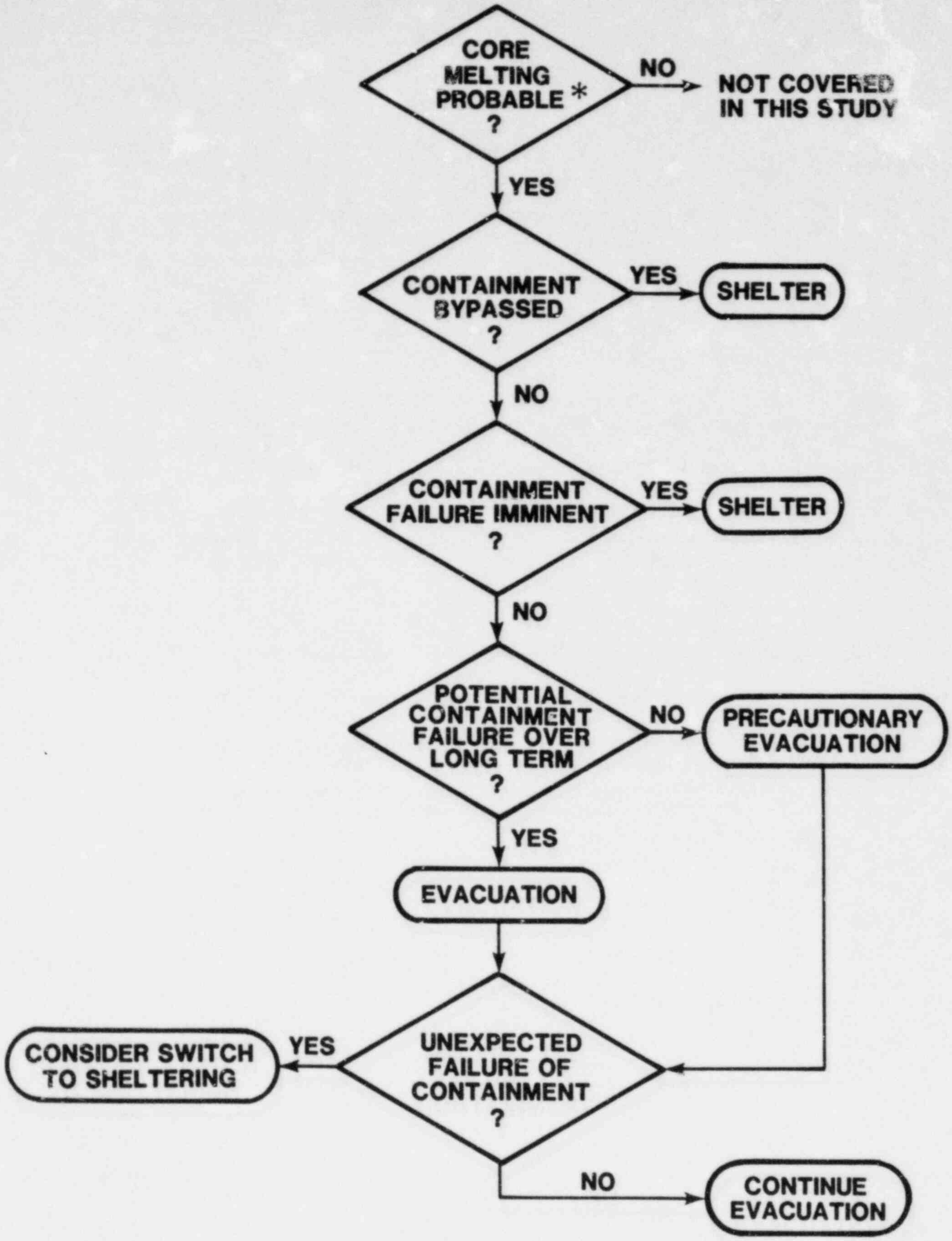

Figure 10-1. Logic for Deciding Emergency Actions

10.5 Future Research

This section summarizes areas in which additional research is necessary to reduce uncertainties associated with this study.

The strength of containment is of primary importance in severe accidents. Our conclusions for Zion are based on a relatively high failure pressure; however, we did not examine failure of containment penetrations (airlocks, piping penetrations, and electrical penetrations). The integrity of such penetrations during accidents in which pressures and temperatures are above design limits should be analyzed.

The ability to isolate containment is accident specific. Detailed plant information (beyond that available to us) should be collected and a more rigorous analysis of the degree to which containment can be isolated during severe accidents should be performed.

The likelihood of hydrogen burning during rapid cooldown from steam inert conditions and the associated peak presure cannot be established. Experimental data, in particular, for large volumes with sprays operating would be helpful.

Phenomena which lead to long-term overpressure cannot be accurately quantified at present. In particular, core-concrete interactions and debris-coolant interactions in the reactor cavity affect the time at which containment could fail. We recommend further analysis to better understand these phenomena.

Fission-product transport from the core, through the primary system and containment, and into the environment is a phenomenological area that is receiving a great deal of attention at present. As more definitive information on the mechanisms which affect transport becomes available, we recommend that our estimates of radiological consequences be revised accordingly.

11. REFERENCES

[ANSI N18.1-1971]
"Selection and Training of Nuclear Power Plant Personnel," ANSI N18.1-1971, American Nuclear Society, 1971.

[ANSI 5.1-1979]
American Nuclear Society Standards Committee Working Group ANS-5.1, "American National Standard for Decay Heat Power in Light Water Reactors," ANSI/ANS-5.1-1979, August 1979.

[ASME III-1, 1980]
ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Division 1 - Subsection NE, Class MC Components, The American Society of Mechanical Engineers, July 1980 (earlier editions may be applicable).

[ASME III-2, 1980]
ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Division 2 - Code for Concrete Reactor Vessels and Containments, The American Society of Mechanical Engineers, July 1980 (earlier editions may be applicable).

[Baybutt et al., 1981]
P. Baybutt, S. Raghuram, and M. Kuhlman, MATADOR: A Computer Code for the Analysis of Radionuclide Behavior During Degraded Core Accidents in Light Water Reactors, Battelle Draft Report, November 1981.

[Bennett, 1979]
D. E. Bennett, SANDIA-ORIGEN User's Manual, NUREG/CR-0937, SAND79-0299, Sandia Laboratories, October 1979.

[Berman et al., 1981]
M. Berman et al., LWR Safety Research Program Quarterly Report, January-March 1981, NUREG/CR-2163, 1 of 4, July 1981.

[Blejwas et al., 1982]
T. E. Blejwas et al., "Background Study and Preliminary Plans for Safety Margins of Containments," Sandia National Laboratories, NUREG/CR-2549, SAND82-0324, March 1982.

[Bonzon, 1977]
L. L. Bonzon, Radiation Signature Following the Hypothesized LOCA, NUREG76-6521, SAND76-0740, Sandia National Laboratories, September 1977.

[Bunz, 1979]
H. Bunz, Der Naua-Mod3: Ein Computerprogramm zur Beschreibung des Aerosolverhaltens in Kondensierender Atmosphaere, PNS-Bericht Nr. 419/79, Kernforschungszentrum Karlsruhe, October 1979.

[Burian and Cybulskis, 1977]
R. J. Burian and P. Cybulskis, CORRAL II User's Manual, Battelle Columbus Laboratories, January 1977.

[Burns, 1980]
Robert D. Burns III, letter to F. E. Haskin, Q-7-80-148, Los Alamos Scientific Laboratory, July 28, 1980.

[Carlson and Hickman, 1978]
D. D. Carlson, J. W. Hickman, "A Value Impact Assessment of Alternate Containment Concepts," NUREG/CR-C165, SAND77-1344, June 1978.

[Carlson, et al., 1981]
D. D. Carlson, et al., Reactor Safety Study Methodology Applications Program: Sequoyah #1 PWR Power Plant, NUREG/CR-1659, 1 of 4, SAND80-1897 (1 of 4), February 1981.

[Carlson et al., 1973]
L. W. Carlson, R. M. Knight, and J. O. Henrie, "Flame and Detonation Initiative and Propagation in Various Hydrogen-Air Mixtures, With and Without Water Sprays," AI-73-29, May 11, 1973.

[CRAC2]
"Calculations of Reactor Accident Consequences Version 2, CRAC2 Computer Code User's Guide," NUREG/CR-2326, to be published.

[Cummings et al., 1982]
John C. Cummings, Allen L. Camp, Martin P. Sherman, Michael J. Wester, David Tomasko, Rupert K. Byers, and Blaine W. Burnham, "Review of the Grand Gulf Hydrogen Igniter System," NUREG/CR-2530, Revised March 5, 1982.

[Darby, 1981]
J. L. Darby, "A Review of the Applicability of Core Retention Concepts to Light Water Reactor Containments," NUREG/CR-2155, SAND81-0416, September 1981.

[Dearian, 1980a]
J. A. Dearian, "Severe Accident Sequence Analysis for the Westinghouse Zion I PWR, JAD-190-80, letter to W. B. Murfin, EG&G Idaho, Inc., Idaho Falls, ID, July 29, 1980.

[Dearian, 1980b]
J. A. Dearian, "Preliminary Results of S_2 Initiated Sequences," CDF-12-80, letter to J. Von Herrmann, EG&G Idaho, Inc., Idaho Falls, ID, November 21, 1980.

[DeMuth, et al., 1981]
N. S. DeMuth, D. Dobranich, R. J. Henninger, Loss of Feedwater Transients for the Zion-1 Pressurized Water Reactor, LA-SASA-TN-81-S, July 1981.

[EPA-520/1-75-001, 1975]
Manual of Protective Action Guides and Protective Actions for Nuclear Incidents, EPA-520/1-75-001, Environmental Protection Agency, September 1975.

[Fletcher et al., 1980]
C. D. Fletcher, S. J. Bruske, and H. M. Delaney, "Loss of Offsite Power Scenarios for the Westinghouse Zion I Pressurized Water Reactor," EGG-CAAP-5156, Idaho National Engineering Laboratory, May 1980.

[Fletcher, 1981]
C. D. Fletcher, "Accident Mitigation Following A Small Break with Coincident Failure of Charging and High Pressure Injection for the Westinghouse Zion I Pressurized Water Reactor, EGG-CAAD-5428, April 1981.

[Greene, 1981]
S. R. Green, Oak Ridge National Laboratory, letter to R. T. Curtis, U. S. Nuclear Regulatory Commission, "March BWR Applications," January 13, 1982.

[Hall, 1979]
R. E. Hall, "A Risk Assessment of a Pressurized Water Reactor for Class 3-8 Accidents," NUREG/CR-0603, Brookhaven National Laboratory, October 1979.

[Haskin, et al., 1981]
F. E. Haskin, et al., "Analysis of a Hypothetical Core Meltdown Accident Initiated by Loss of Offsite Power for the Zion Pressurized Water Reactor," NUREG/CR-1988, SAND81-0504, Sandia National Laboratories, 1981.

[Hertzberg, 1981]
M. Hertzberg, "Flammability Limits and Pressure Development in H_2: Air Mixtures," Proceedings of the Workshop on the Impact of Hydrogen on Water Reactor Safety, Vol. III, NUREG/CR-2017, SAND81-0661, September 1981.

[Kemeny, 1979]
J. G. Kemeny (Chairman), Report of the President's Commission on the Accident at Three Mile Island, U. S. Government Printing Office, October 1979.

[Kolb, et al., 1981]
G. J. Kolb, et al., "Reactor Safety Study Methodology Applications Program: Oconee #3 PWR Power Plant," NUREG/CR-1659, 2 of 4, SAND80-1897 (2 of 4), May 1981.

[Marks, 1978]
Marks Standard Handbook for Mechanical Engineers, T. Baumeister, ed., 8th Edition, McGraw-Hill, 1978.

[Meyer, 1980]
J. F. Meyer (Compiler), "Viewgraphs for June 17, 1980, Technology Exchange Meeting #5 'Containment Structural Response'," U. S. Nuclear Regulatory Commission, June 1980.

[Muir, et al., 1981]
J. F. Muir, et al., "CORCON-MOD1: An Improved Model for Molten-Core/Concrete Interactions," SAND80-2415, NUREG/CR-2415, 1981.

[Murfin, 1980]
W. B. Murfin (Compiler), "Report of the Zion/Indian Point Study, Vol. I, NUREG/CR-1410," SAND80-0617/1, Sandia National Laboratories, April 1980.

[Murphy, et al., 1981]
M. B. Murphy, et al., "Examination Results of the Three Mile Island Radiation Detector HP-R-211," SAND81-0725, Sandia National Laboratories, September 1981.

[Neidel, et al., 1981]
E. C. Neidel, J. G. Castel, Jr., and J. E. Grover, "A Review of H_2 Detection in Light Water Reactor Containments," SAND81-0326, NUREG/CR-2080, December 1981.

[NSAC-1, 1979]
Nuclear Safety Analysis Center, Analysis of the Three Mile Island - Unit 2 Accident, NSAC-1, Electric Power Research Institute, July 1979.

[Nuclear News, 1981]
Nuclear News, p. 36, May 1981.

[Nuclear Technology, 1981]
Nuclear Technology, Vol. 53, May 1981.

[NUREG-0205, 1977]
Core Meltdown Experimental Review, NUREG-0205, SAND74-0382, Sandia Laboratories, March 1977.

[NUREG-0585, 1979]
Lessons Learned Task Force, TMI-2 Lessons Learned Task Force Final Report, NUREG-0585, U. S. Nuclear Regulatory Commission, October 1979.

[NUREG-0654, 1980]
Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants, NUREG-0654, FEMA-REP-1, U. S. Nuclear Regulatory Commission and Federal Emergency Management Agency, November 1980.

[NUREG-0772, 1981]
Technical Bases for Estimating Fission Product Behavior During LWR Accidents, NUREG-0772, Draft Report, 1981.

[NUREG-0850, 1981]
U. S. Nuclear Regulatory Commission, Preliminary Assessment of Core Melt Accidents at the Zion and Indian Point Nuclear Power Plants and Strategies for Mitigation their Effects: Analysis of Containment Building Failure Modes, Preliminary Report, NUREG-0850, Vol. I, November 1981.

[Powers and Arellano, 1982]
D. A. Powers, F. E. Arellano, "Large-Scale, Transient Tests of the Interaction of Molten Steel with Concrete," NUREG/CR-2282, January 1982.

[RG 1.97, 1981]
U. S. Nuclear Regulatory Commission, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Conditions During and Following an Accident," Regulatory Guide 1.97, Revision 2, December 1980.

[Rivard, 1980]
J. B. Rivard, "Review of In-Vessel Meltdown Models," NUREG/CR-1493, Sandia National Laboratories, July 1980.

[Rivard et al., 1981]
J. B. Rivard, et al., Interim Technical Assessment of the MARCH Code, SAND81-1672, NUREG/CR-2285, 1981.

[Senglaub, et al., 1981]
M. E. Senglaub, J. P. Odem, M. J. Clauser, J. E. Kelley, and P. S. Pickard, "CONTAIN, A Computer Code for the Analysis of Containment Response to Reactor Accident - Version 1A," NUREG/CR-2224, SAND81-1495, June 1981 Draft.

[Sherman, et al., 1980]
M. P. Sherman, et al., The Behavior of Hydrogen During Accidents in Light Water Reactors, NUREG/CR-1561, SAND80-1495, Sandia National Laboratories, August 1980.

[Stevenson, 1980]
M. G. Stevenson (Compiler), Report of the Zion/Indian Point Study, Vol. II, NUREG/CR-1141, LA-8306-MS, Los Alamos National Laboratory, April 1980.

[Swenson and Corradini, 1981]
D. V. Swenson and M. L. Corradini, "Monte Carlo Analysis of LWR Steam Explosions," NUREG/CR-2307, SAND81-1092, October 1981.

[TMI, 1980]
N.R.C. Special Inquiry Group, Three Mile Island: A Report to the Commissioners and to the Public, U. S. Nuclear Regulatory Commission, January 1980.

[Wall et al., 1977]
I. B. Wall, et al., Overview of Reactor Safety Study Consequence Model, NUREG-0340, U. S. Nuclear Regulatory Commission, June 1977.

[WASH-1400, 1975]
Reactor Safety Study, WASH-1400 (NUREG-75/014), U. S. Nuclear Regulatory Commission, 1975.

[Wooton and Avci, 1980]
R. O. Wooton and H. I. Avci, "MARCH (Meltdown Accident Response Characteristics) Code Description and User's Manual," NUREG/CR-1711, BMI-2064 (October 1980).

[Zion-FSAR]
Zion Station Final Safety Analysis Report, Commonwealth Edison Company, 1973.

[Zion-SD]
Zion Station System Descriptions, Commonwealth Edison Company, Received 1980.

[Zion PRA, 1982]
Zion Probabilistic Safety Study, Pickard, Lowe, and Garrick, Inc., review draft, January 1982.

[Zion-RTS]
Radiological Technical Specifications for Zion Units 1 and 2, Commonwealth Edison Company, Docket Numbers 50-295 and 50-304, 1973.

APPENDIX A

EMERGENCY-ACTION GUIDELINES

Guidance for emergency action is provided in NUREG-0654 [NUREG-0654, 1980]. Four classes of emergency-action levels are established: Notification of Unusual Event, Alert, Site Area Emergency, and General Emergency. Initiation of a specific emergency class is based on plant-specific instrument readings. For accident situations in which the core is uncovered, a general emergency is called for. A general emergency is defined as follows:

> "Events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area." [NUREG-0654, 1980]

The question of whether to evacuate or shelter is discussed in general terms in the guidance document:

> "For core melt sequences where significant releases from containment are not yet taking place and large amounts of fission products are not yet in the containment atmosphere, consider 2 mile precautionary evacuation. Consider 5 mile downwind evacuation (45° to 90° sector) if large amounts of fission products (greater than gap activity) are in the containment atmosphere. Recommend sheltering in other parts of the plume exposure Emergency Planning Zone under this circumstance.
>
> "For core melt sequences where significant releases from containment are not yet taking place and containment failure leading to a direct atmospheric release is likely in the sequence but not imminent and large amounts of fission products in addition to noble gases are in the containment atmosphere, consider precautionary evacuation to 5 miles and 10 mile downwind evacuation (45° to 90° sector).
>
> "For core melt sequences where large amounts of fission products other than noble gases are in the containment atmosphere and containment failure is judged imminent, recommend shelter for those areas where evacuation cannot be completed before transport of activity to that location." [NUREG-0654, 1980]

APPENDIX B

CONTAINMENT FAILURE ANALYSES

Detailed containment structural drawings were not available for this study. As a result, we were not able to conduct a thorough analysis of containment structural capability. Analyses had previously been carried out by other organizations. These analyses were examined by R. L. Woodfin of the Systems Safety Technology Division at SNL; it was his opinion that the methods of analysis are reasonable.

The design pressure of the Zion 1 containment is 0.43 MPa (62 psia) [Zion FSAR, 1973]. However, the design pressure is not a good indicator of the pressure at which the containment would actually fail. Sargent and Lundy [Meyer, 1980] performed an approximate analysis of the Zion 1 containment, and concluded that the most likely failure mode would be hoop-tendon yielding. They calculated an ultimate pressure capacity of 0.93 MPa (135 psia) ignoring the contribution of the liner, or 1.03 MPa (149 psia) including the liner. Los Alamos National Laboratory [Stevenson, 1980] calculated a failure pressure of 1.16 MPa (169 psia), although extensive yielding and cracking were predicted at lower pressures. Neither analysis explicitly considered failure of penetrations, although the Sargent and Lundy analysis does make reference to the possibility of failure at the equipment hatch. The ratio of calculated failure pressure to design pressure ranges from 2.17 to 2.72.

We assumed the Sargent and Lundy estimate including the liner contribution to be a "best estimate,"; that is, at this pressure the containment has a 50% probability of failure. We further assumed that at the lower Sargent and Lundy estimate there would be a low probability of failure, say 10%, and, at the Los Alamos estimate, a low probability of survival. We assumed a normal distribution of failures as shown in Figure B-1.

Containment loadings of concern in this study are quasi-static; the load on containment is long in duration compared with the structural response time. This is the case for both steam spikes and hydrogen burns [NUREG/CR-1561, 1980]. Impulsive loads, as from hydrogen detonation, were not addressed. We have assumed complete mixing of containment atmospheric contents; computed mole fractions of hydrogen, steam, and air led to deflagration before detonation conditions were ever achieved.

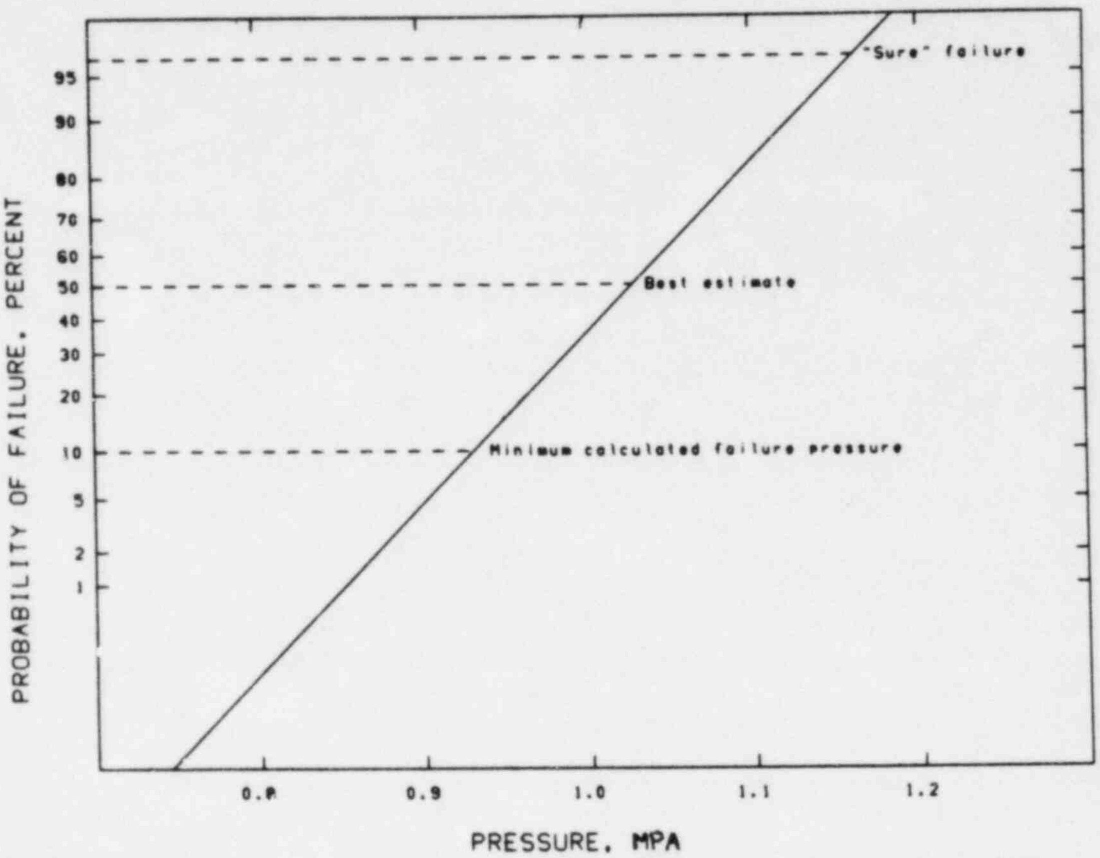

Figure B-1. Assumed containment failure probability distribution.

APPENDIX C

MINIMUM TIME OF CONTAINMENT FAILURE: FAN COOLER EFFECTIVENESS

The earliest time at which containment can fail due to overpressure is estimated. If one fan cooler is turned on just before failure pressure is reached, our best estimate is that overpressure of containment will not occur. (Hydrogen burning is not considered in this analysis.)

Physical insight into the parameters that affect the timing of containment failure can be gathered by applying the first law of thermodynamics to the containment and its internals. The net heat added to the system, Q, equals the net change in internal energy of the system, U.* Q and U can be expressed as:

$$\begin{aligned} Q &= \mathrm{QTOT} = \mathrm{QCO} - \mathrm{QSG} + \mathrm{QDH} - \mathrm{QCON} \\ U &= \mathrm{DELU} = \mathrm{CHAU} + \mathrm{CHSU} + \mathrm{CHCS} + \mathrm{FUWS} - \mathrm{UPC} - \mathrm{MWAEN}^{**} \end{aligned} \qquad (1)$$

where: QCO is the heat added from oxidation of zirconium.

QSG is the heat removed by steam generator secondaries.

QDH is decay heat.

QCON is the heat removed by concrete.

CHAU is the change in internal energy of the containment atmosphere.

CHSU is the change in internal energy of steel structures in containment.

CHCS is the change in internal energy of core and structural materials in the core and vessel.

FUWS is the final internal energy of the water in the sump and cavity.

UPC is the initial internal energy of the primary coolant, including pressurizer and accumulator inventories.

MWAEN is the internal energy of the water added to the system by the ECCS.

---

*Internal energies are referenced to the triple point of water.

**These variables correspond to names in the attached listing of computer program CHGU. American engineering units are used.

Containment atmosphere is assumed to always be saturated following primary system blowdown. The time to failure can be estimated using equation (1) by computing final internal energies corresponding to containment failure pressure. This equation assumes that no heat is removed by containment heat removal systems (sprays and fans) and that no heat is transferred directly from corium to concrete. Obviously, time to failure is minimized without containment heat removal. If heat goes directly into concrete, the time to failure is increased because the rate of pressure rise due to generating noncondensible gases is small compared to that due to generating steam from water in the cavity, given no containment heat removal. (For Zion, the long-term rate of pressure rise due to all decay heat going into concrete should be less than ~2 psi per hour.*)

A short computer program, CHGU, was written to calculate time of containment failure based on equation (1). A listing of the program CHGU is given in Table C-1.

Equation (1) implies two interesting points: the greater the final internal energy of corium (the greater the value of CHCS), the greater the time until failure; and the greater the mass of water left in containment at failure (the greater the value of FUWS), the greater the time until failure.**

At a failure pressure of 150 psia (about 128 psi partial steam pressure), Zion will contain 7.76×10^5 lb of water vapor (saturated state).*** Zion contains 7.44×10^5 lb of water in the primary system, pressurizer, and accumulators. 1.18×10^5 lb of water end up in the sump. 1.74×10^4 lb of water are reacted in the oxidation of the 45,000 lb of zirconium. The 6.09×10^5 lb of water available to be evaporated cannot supply the 7.76×10^5 lb needed for failure at saturation. Without the addition of ECCS water, Zion containment cannot fail at its earliest time. (With no water added, the cavity dries out and core/concrete interactions cannot be avoided. However, the rate of pressure rise from such interactions is believed to be small, as previously discussed.)

The minimum failure time is achieved when just enough water is added by the ECCS to enable the final saturated state to be reached with no water left in the reactor cavity. About 1.7×10^5 lb of water need to be added. If more than this minimum amount

---

*This estimate is based on conversations with R. K. Cole, a principal investigator of core-concrete interactions [Muir et al., 1981]. This point is also discussed in work by Brookhaven [NUREG-0850, 1981]

**It takes energy to heat up water in the cavity and sump.

***Based on CHGU calculation of steam and gas pressures, assuming 100% of the zirconium oxidizes. Zion containment volume is 2.72×10^6 ft^3.

Table C-1

FORTRAN Listing of Program CHGU

```
      PROGRAM CHGU(INPUT,OUTPUT,TAPE10,TAPE2=INPUT,TAPE3=OUTPUT)
C     COMPUTE CHG INT ENERGY IN CONT.$EARLIEST TIME CONT OVERPRESSURE
C PIN,INITIAL PRESS(PSIA)-TIN,INITIAL TEMP(F)-PFI,FINAL PRESS(PSIA)-VOL,
C VOLUME(FT**3)-VSUMP,VOL SUMP(FT**3)-VCAV,VOL CAVITY(FT**3)
C RMZ,ZIRCONIUM LBS-QSG,STEAM GEN BTU-PO,POWER MW(T)
C RMS,STEEL MASS LBS-CSFU,CORE/STRUC FINAL INT EN-CSIU,CORE/STRUC
C     *INITIAL INTERNAL ENERGY
C BTU-T,FAILURE TIME,SEC-UPC,INIT INT
C EN PRIM COOL,BTU-MWA,LBS WATER ADDED-MWP,LBS WATER IN PRIMARY$ACCUM
C AREA,UNLINED CONCRETE SURFACE AREAS(FT**2)-KC,CON.THER. COND,(BTU/HR/
C /FT/F)-A,CON. THER. DIFFUSIVITY(FT**2/HR)
C AREAL,LINED CONCRETE SURFACE AREAS-KCL,LINED CON THER CON-AL,
C LINED CON THER DIFF
C TAVG IS AVG TEMP IN CONTAINMENT DURING ACCIDENT
C DT,TEMP DROP AIR GAP(LINER TO CONCRETE),F
      DIMENSION PSAT(24),VSAT(24),USAT(24),TSAT(24),ID(5),CHCS(15),
     *T(15),ATIT(5),AABS(5),AORD(5),AREA(2),KC(2),A(2),PFI(11),
     *AREAL(2),KCL(2),AL(2),LAB1(7),LAB2(7)
      REAL MWA,MWP,MWS,KC,KCL
      DATA PSAT/45.,50.,55.,60.,65.,70.,75.,
     *80.,85.,90.,95.,100.,105.,110.,115.,
     *120.,125.,130.,135.,140.,145.,150.,160.,170./
      DATA VSAT/9.4,8.5,7.8,7.2,6.6,6.2,5.8,
     *5.5,5.2,4.9,4.6,4.4,4.2,4.0,3.9,3.7,3.6,3.4,3.3,3.2,
     *3.1,3.0,2.8,2.7/
      DATA USAT/1094.,1095.,1097.,1098.,1099.,1100.,1101.,
     *1102.,1103.,1104.,1104.,1105.,1106.,1106.,
     *1107.,1108.,1108.,1109.,1109.,1110.,1110.,1110.,
     *1111.,1112./
      DATA TSAT/274.,281.,287.,293.,298.,303.,308.,
     *312.,316.,320.,324.,328.,331.,335.,338.,341.,344.,347.,
     *350.,353.,356.,358.,363.,368./
      DATA ATIT,AABS,AORD,LAB1,LAB2/29(10H
      CALL ENTFLM(10)
      CALL CTXTC2(ATIT(1),50,0)
      CALL CTXTC2(AABS(1),50,0)
      CALL CTXTC2(AORD(1),50,0)
      XMAX=200000.
      XMIN=10.
      YMAX=200.
      YMIN=60.
      AABS(1)=10HTIME, SEC
      AORD(1)=10HPRESSURE,
      AORD(2)=10HPSIA
      ATIT(1)=10H        ZI
      ATIT(2)=10HON 1, MWA=
      READ(2,*)VSUMP,VCAV,DT
      WRITE(3,44)
      WRITE(3,45)VSUMP,VCAV,DT
      READ(2,*)(AREA(I),I=1,2)
```

Table C-1 (Cont'd)

```
      READ(2,*)(KC(I),I=1,2)
      READ(2,*)(A(I),I=1,2)
      READ(2,*)(AREAL(I),I=1,2)
      READ(2,*)(KCL(I),I=1,2)
      READ(2,*)(AL(I),I=1,2)
      WRITE(3,47)
      WRITE(3,48)AREA(1),KC(1),A(1)

      WRITE(3,71)PO,RMS,CSIU,UPC,MWA,MWP
C TAVG IS AVG TEMP(F)WHICH CONCRETE STRUCTURES SEE
      TAVG=TFIN-30.
      WRITE(3,150)
      WRITE(3,175)TAVG,CHCS(K)
      WRITE(3,100)
      WRITE(3,200)PGAS,PST,TFIN,CHAU,MWS
C QCO IS CLAD OXID BTU
C USE 140 KCAL/MOLE ZIRC
      QCO=RMZ*2.71E3
C TS IS TIME IN SEC
      TS=1.
      DO 3 J=1,350
      TS=TS+60.*J
C QDH IS DECAY HEAT BTU
      QDH=150./.71*PO*TS**(.71)
C QCON IS HEAT REMOVED BY CONCRETE UP TO TIME TS(=CHG INT EN CON.)
C UNITS,BTU(TS IN SEC) SEE CARSLAW$JAEGER SEC.2.4 2ND ED.
C QCOU IS UNLINED CONCRETE,QCOL IS LINED CONCRETE,QCON IS ALL CONCRETE
      QCOU=(KC(1)*(TAVG- TIN)*AREA(1)/(SQRT(3.14*A(1)))+KC(2)*
     $(TAVG- TIN)*AREA(2)/(SQRT(3.14*A(2))))*(3.33*10.**(-2.)*SQRT(TS))
      QCOL=(KCL(1)*(TAVG-DT- TIN)*AREAL(1)/(SQRT(3.14*AL(1)))
     $+KCL(2)*(TAVG-DT- TIN)*AREAL(2)/(SQRT(3.14*AL(2))))
     $*(3.33*10.**(-2.)*SQRT(TS))
      QCON=QCOU+QCOL
C FIT MARCH CONCRETE HEAT REMOVAL
      QCON=QCON*.9
CQTOT IS NET HEAT ADDED
C QSG,POSITIVE IS HEAT REMOVED BY STEAM GENERATORS
      QTOT=QCO-QSG+QDH-QCON
C DELU IS NET CHANGE IN INTERNAL ENERGY
C WATER ADDED,MWA,AT 100 F
      DELU=CHAU+CHSU+CHCS(K)+FUWS-UPC-MWA*1.0*(100.-32.)
      DIFF=QTOT-DELU
      T(L)=TS
      IF(DIFF.GE.0.) GO TO 40
 3    CONTINUE
 40   WRITE(3,300)
      WRITE(3,400)QCO,QDH,QCON,T(L)
      WRITE(3,500)
      WRITE(3,600)DELU,QTOT
 7    CONTINUE
      CALL PLOTPR(1,0,T,PFI,ITAL,1,1,0,1H*,ATIT,AABS,AORD,IF,1,
     *XMIN,XMAX,1,YMIN,YMAX)
```

```
      IF=0
 6    CONTINUE
      ENCODE (50,54,LAB2) CHCS(1),CHCS(4),CHCS(7),CHCS(10),CHCS(13)
      CALL PLABPR(LAB1,LAB2)
      IF=2
 8    CONTINUE
      GO TO 1
 44   FORMAT(1H0,5HVSUMP,5X,4HVCAV,5X,15HGAP TEMP DROP F)
 45   FORMAT(1H0,E8.3,2X,E8.3,1X,E8.3)
 46   FORMAT(E10.3)
 47   FORMAT(1H0,16HUNLINED CONCRETE,2X,5HAREAS,5X,4HKC"S,5X,3HA"S)
 48   FORMAT(1H0,18X,E8.3,2X,E8.3,1X,E8.3)
 49   FORMAT(1H0,16HLINED   CONCRETE,2X,5HAREAS,5X,4HKC"S,5X,3HA"S)

      WRITE(3,48) AREA(2),KC(2),A(2)
      WRITE(3,49)
      WRITE(3,48) AREAL(1),KCL(1),AL(1)
      WRITE(3,48) AREAL(2),KCL(2),AL(2)
 1    READ(2,50)ID
      IF(EOF(2).NE.0) GO TO 750
      WRITE(3,60)ID
      READ(2,*)PIN,TIN,VOL,RMZ,QSG,PO,RMS,CSIU,UPC,MWP
      MWA=.5E5
      DO 8 M=1,33,8
      MWA=MWA+M*1.E5
      ENCODE(10,46,ATIT(3)) MWA
C ASSUME STEAM PART. PRESS OF 1.3 PSI
      PIA=PIN-1.3
      MAIR=29.*PIA*VOL/10.73/(TIN+460.)
C IF IS FOR PLOT
      IF=2
      DO 6 K=1,15,3
      PRESS=69.
      ITAL=0
      DO 7 L=1,11
      PFI(L)=PRESS+L*10.
      DO 2 I=1,24
C ASSUME SATURATED FINAL STATE
      TFIN=TSAT(I)
      VFIN=VSAT(I)
      UST=USAT(I)
C ACCOUNT FOR ZR+2H2O/ZRO2+2H2 EXPLICITELY IN ATMOS.
C IMPLICITELY IN FUEL
      PST=10.73*(TFIN+460.)/VFIN/18.
      PAIR=MAIR*10.73*(TFIN+460.)/VOL/29.
      PH2=RMZ/93.*2.*10.73*(TFIN+460.)/VOL
      PGAS=PAIR+PH2
      DELP=PST+PGAS-PFI(L)
      IF(DELP)2,20,20
 2    CONTINUE
```

Table C-1 (Cont'd)

```
C COMPUTE MASS WATER LEFT IN CAVITY,MWS,SUMP WATER NOT IN CAVITY
C SUBTRACT MASS H2O CONSUMED IN ZIRC OXID
 20    MWS=MWA+MWP-VOL/VFIN-VSUMP/.017-RMZ/93.*2.*18.
       IF(MWS)4,5,5
 4     WRITE(3,700)
       GO TO 7
C COMPUTE FINAL INT EN WATER IN CONTCAVITY AND SUMP,FUWS.
 5     FUWS=(MWS+VSUMP/.017)*1.0*(TFIN-40.-32.)
       ITAL=ITAL+1
C CHAU IS CHG IN ATMOS INT EN BTU
       CHAU=(VOL/VFIN*UST)-(18.*1.3*VOL/10.73/(TIN+460.)*1046.)+(MAIR*
      $.17*(TFIN-TIN))+RMZ/93.*2.*2.*1.43*(TFIN-32.)
C CHSU IS CHANGE IN STEEL INT EN BTU
       CHSU=RMS*.12*(TFIN-TIN)
C CHCS IS CHANGE IN CORE/STRUC INTERNAL ENERGY
       CSFU=-1.E7
       CSFU=CSFU+1.E7*K
       CHCS(K)=CSFU-CSIU
       WRITE(3,65)
       WRITE(3,70)PIN,TIN,PFI(L),VOL,RMZ,QSG
       WRITE(3,75)

 50    FORMAT(5A8)
 54    FORMAT(5E10.3)
 60    FORMAT(1H0,5A8)
 65    FORMAT(1H0,/,4H PIN,6X,3HTIN,6X,3HPFI,6X,3HVOL,6X,3HRMZ,6X,3HQSG)
 70    FORMAT(1X,E8.3,1X,E8.3,1X,E9.3,1X,E8.3,1X,E8.3,1X,E8.3)
 71    FORMAT(1X,E8.3,1X,E8.3,1X,E9.3,1X,E8.3,1X,E8.3,1X,E8.3)
 75    FORMAT(1H0,3HPO,6X,3HRMS,6X,4HCSIU,6X,3HUPC,6X,3HMWA,6X,3HMWP)
 100   FORMAT(1H0,7HGAS PSI,5X,9HSTEAM PSI,5X,6HTEMP F,5X,10HCHG AT BTU,
      *5X,10HLB H20 CAV)
 150   FORMAT(1H0,4HTAVG,5X,4HCHCS)
 175   FORMAT(1H0,F8.3,1X,F9.3)
 200   FORMAT(1X,E8.3,4X,E8.3,6X,E8.3,2X,E8.3,8X,E8.3)
 300   FORMAT(1H0,11HCLAD OX BTU,2X,10HDEC HT BTU,2X,12HBTUS IN CONC,2X,
      *8HTIME SEC)
 400   FORMAT(1X,E8.3,5X,E8.3,4X,E8.3,6X,E8.3)
 500   FORMAT(1H0,17HNET CH INT EN BTU,2X,10HTOT HT BTU)
 600   FORMAT(1X,E8.3,11X,E8.3)
 700   FORMAT(1H0,15HNOT ENOUGH WATER)
 750   CALL EXTFLM(0)
 800   STOP
       END
```

of water is added, the time to failure is increased due to the internal energy remaining within water in the reactor cavity.

Figures C-1 through C-3 are CHGU results for Zion.* Each figure gives time to reach a given pressure for a fixed amount of water added (MWA). The five curves on each figure (left to right) correspond to the five values (left to right) of CHCS.

CHCS cannot be accurately estimated due to phenomenological uncertainties associated with corium fragmentation, quenching, and debris coolability. A conservative assumption is that the corium is quenched to the saturation temperature of the containment atmosphere. A more realistic assumption, that is still probably conservative, is that CHCS = 0.

The minimum time to failure for Zion (149 psia) is about 2.3×10^4 s (383 minutes). However, this may not be a realistic estimate since it assumes that only about 7×10^4 kg (1.5×10^5 lb) of water are added by the ECCS. The RWST contains 1.4×10^6 kg. If water is available to be added, it is unlikely that only a small amount of it would be added. If all the RWST water is added, containment would not fail until about 1400 minutes.

To examine the ability of one fan cooler to preserve containment integrity, we will consider the earliest possible time of containment failure, 2.3×10^4 s. If containment reaches 128 psi steam, at 2.3×10^4 s, the rate of heat addition due to decay heat is approximately $(4.9 \times 10^5)(2.3 \times 10^4)^{-.29} = 2.7 \times 10^4$ Btu/sec. Figure C-4 shows the heat removal rate for one fan cooler at Zion. Data beyond 80 psig is not available. At 80 psig, one cooler removes 2.8×10^4 Btu/sec. The cooler removes heat due to condensation of steam; the increase in heat removal rate with steam pressure is due to the increase in steam density with pressure. Based on the MARCH model of fan coolers, one cooler should remove about 3.5×10^4 Btu/sec at 128 psi steam pressure. One cooler can remove more heat than is being added to containment due to decay heat, even if the best-estimate failure limit on containment pressure is reached at the earliest possible time.

If one fan cooler is turned on at any time prior to 383 minutes into an accident, the Zion containment would not fail due to overpressure.

*QSG = 0 in these figures.

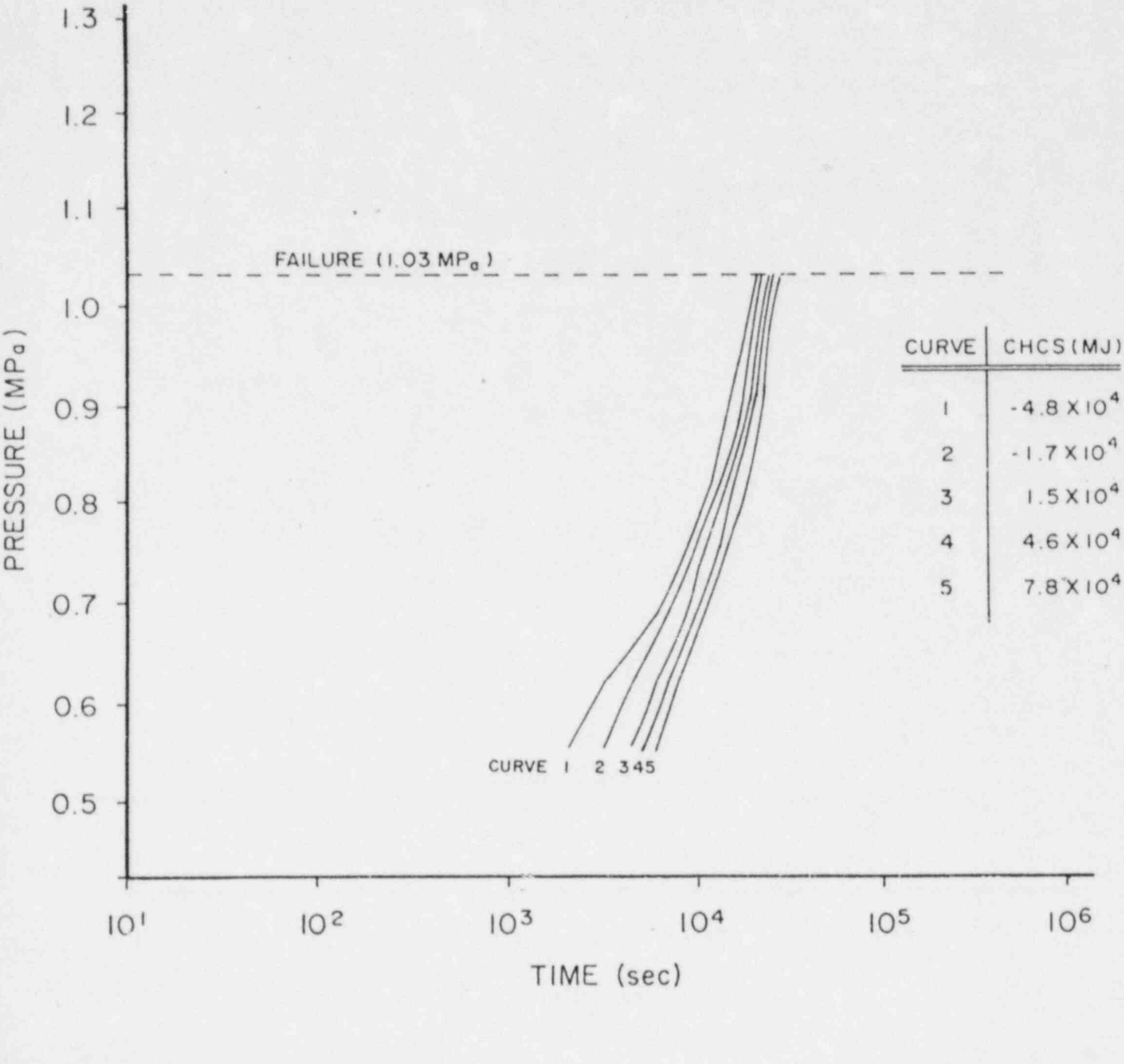

Figure C-1. Maximum Containment Pressurization When 7 x 10^4 lb Water is Added.

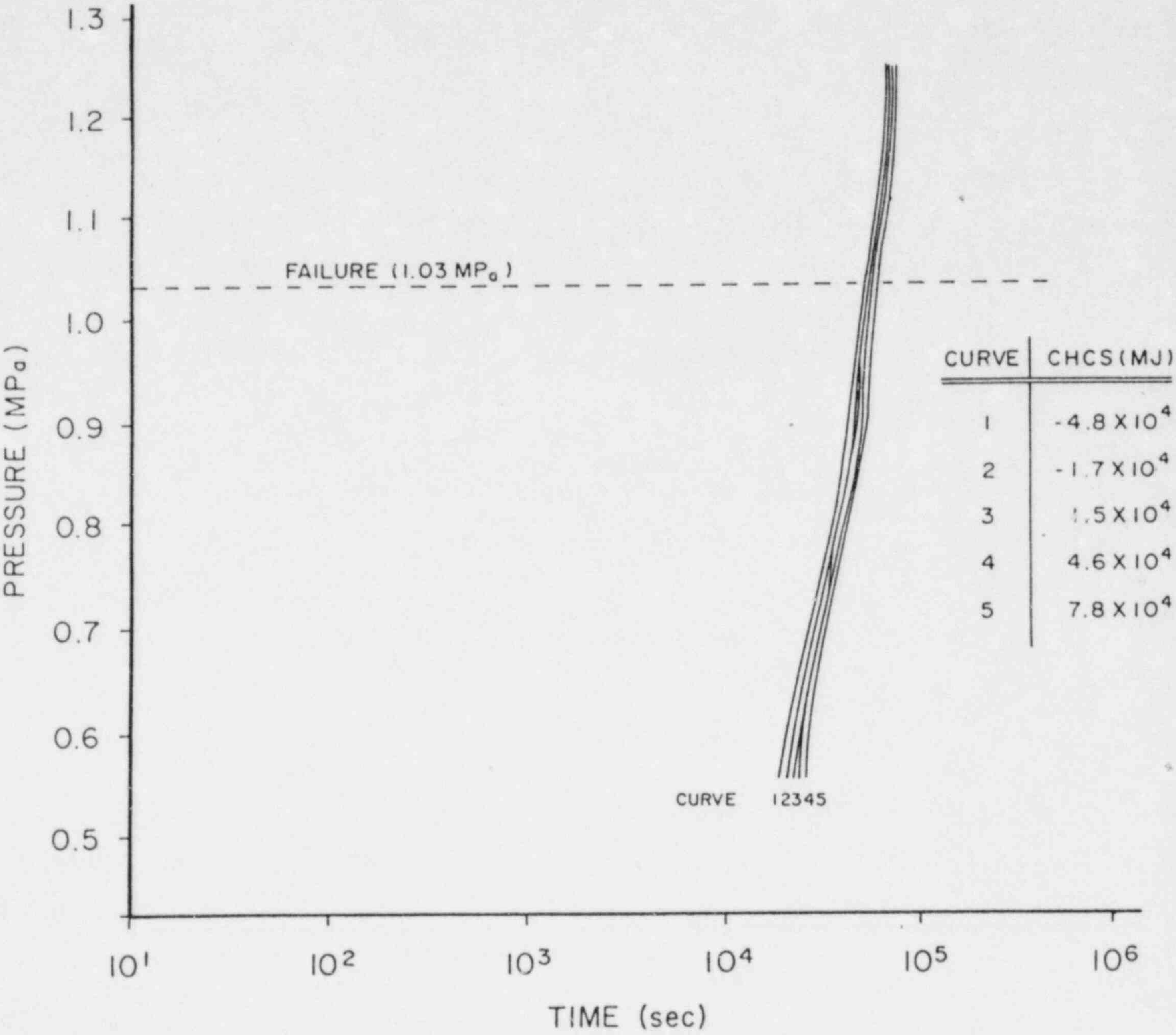

Figure C-2. Maximum Containment Pressurization When 1.2×10^6 lb of Water is Added.

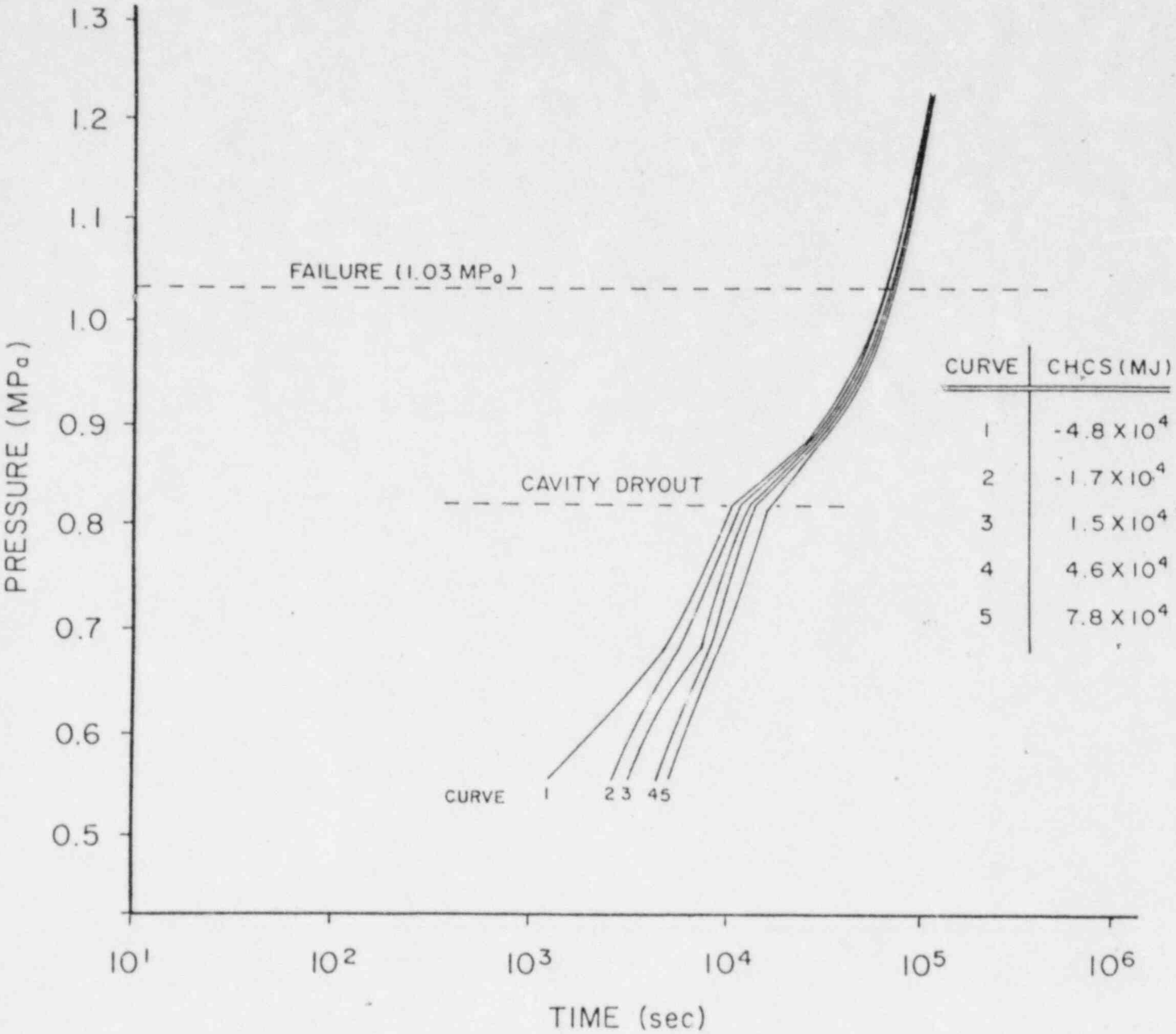

Figure C-3. Maximum Containment Pressurization When no Water is Added.

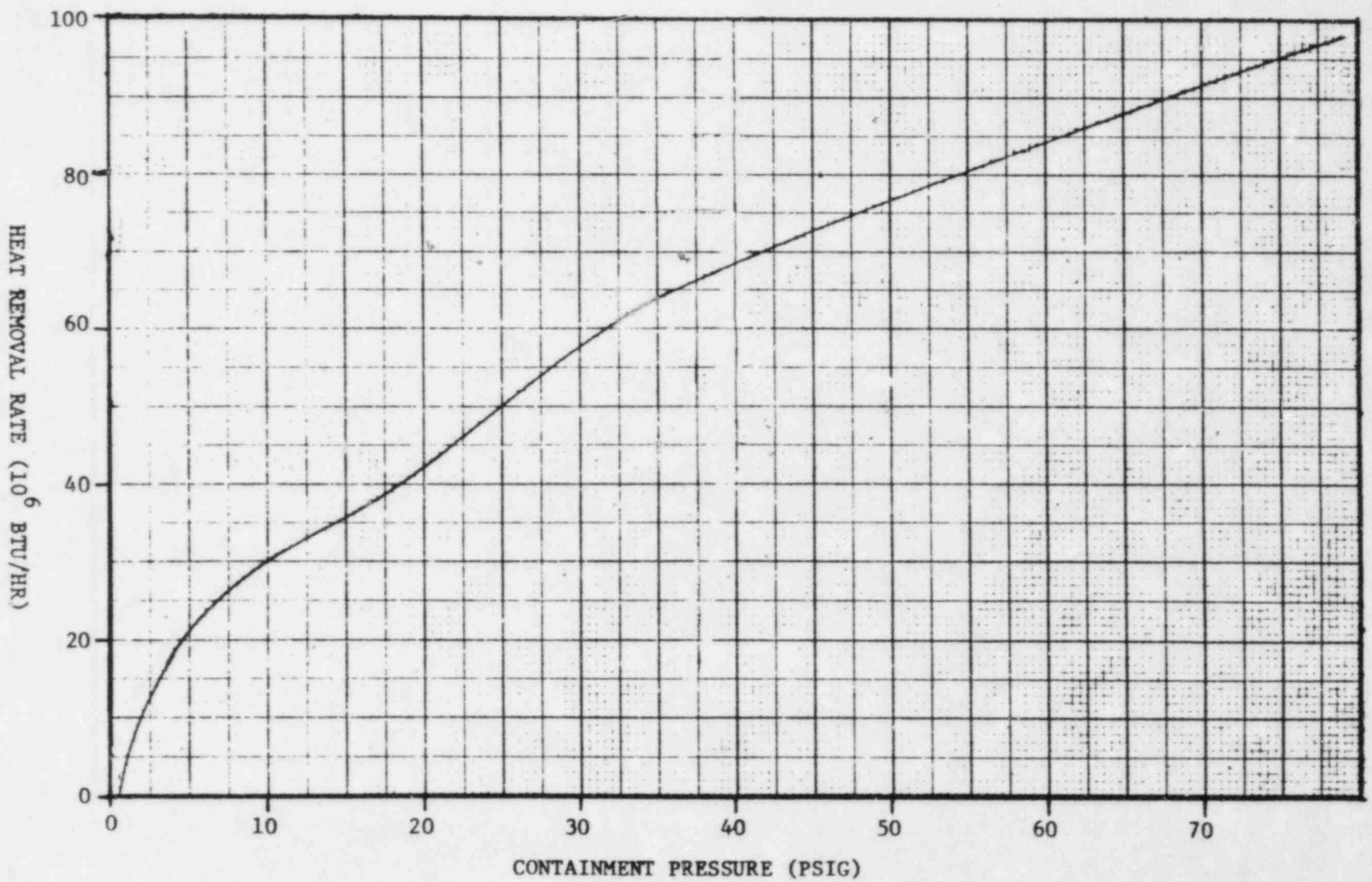

Figure C-4. Zion Fan Cooler Heat Removal Rate

APPENDIX D

THE CONSEQ COMPUTER CODE

The CONSEQ computer code calculates mean man rems, mean latent cancer fatalities, and mean land area interdicted for a specific accident at a specific site. Inputs are CORRAL release fractions for the specific accident and CRAC2 benchmark calculations. To use CONSEQ, one must construct the following families of graphs:

- mean man rem per core fraction released in a given nuclide group versus core fraction released in that group (based upon CRAC2 benchmark calculations)

- mean latent cancer fatalities per core fraction released in a group versus fraction released in that group (based upon CRAC2 benchmark calculations

- land area interdicted per fraction Cs-Rb released versus fraction Cs-Rb released (based upon CRAC2 benchmark calculations).

All of these curves can be fit reasonably well by a power law, or equivalently, logarithms of quantities can be fit by straight lines.

The CONSEQ computations are based on the linear equation $Y = S * X + B$. For mean latent cancer fatalities: Y is the (natural) logarithm of the latent cancers per core fraction released in a given group, X is the logarithm of the core fraction released in the group, and S and B are appropriate constants for that group. For mean man rem: Y is the logarithm of the man rem per core fraction released in a given group, X is the logarithm of the core fraction released in a given group, and S and B are appropriate constants for that group. For land area interdicted: Y is the land area in square miles per core fraction released in the Cs-Rb group, X is the logarithm of the Cs-Rb fraction released, and S and B are appropriate constants.

Given S's and B's from the graphs of the CRAC2 benchmark calculations, CONSEQ reads X's for a particular accident (CORRAL data), computes Y's, and computes and prints consequences for that specific accident. Table D-1 summarizes the curve fits for Zion. Note that the logarithms of the man rem (or latent cancers) per fraction released in a group are strongly dependent on fraction released only for the Cs-Rb group.* This is due to the trade-off between chronic exposure and land area interdicted for the Cs-Rb group.

*S's (slopes) are large in absolute magnitude only for Cs-Rb.

Table D-1

Appropriate CONSEQ Curve Fit Constants for Zion

Nuclide Group	S's for Latent Cancers	B's for Latent Cancers	S's for Man Rem	B's for Man Rem	S for Land Inter.	B for Land Inter.
Kr-Xe	-3.8×10^{-2}	2	-4.3×10^{-2}	11.7	--	--
I-Br	-5.4×10^{-2}	6	-7.3×10^{-2}	15.4	--	--
Cs-Rb	-0.265	7.8	-0.250	17.9	15.7	15.3
Te	$+4.0 \times 10^{-2}$	5.9	0.	14.3	--	--
Ba-Sr	-3.6×10^{-2}	7.4	-5.0×10^{-2}	17.2	--	--
Ru	-3.1×10^{-2}	9.2	-8.5×10^{-2}	17.6	--	--
La	0.	10.8	-5.7×10^{-2}	70.	--	--

The CONSEQ program is listed in Table D-2.

The CONSEQ output for the accidents analyzed in this study is given in Table D-3.

Table D-2

FORTRAN Listing of Program CONSEQ

```
      PROGRAM CONSEQ(INPUT,OUTPUT,TAPE2=INPUT,TAPE3=OUTPUT)
C S"S $B"S ARE CRAC CONSTANTS IN Y=S*X+B
C X IS LN OF CORE FRACTIONS RELEASED,FOR LATENT CANCERS(LC)$ MAN
C REM(MR) Y IS LN OF LC OR MR PER CORE FRAC RELEASED
C FOR AREA LAND INTERDICTED(ALI)Y IS AREA(MI**2) PER FRAC CS RELEASED
C SLC"S ARE S"S FOR LATENT CANCER DEATHS,BLC"S ARE B"S FOR LAT CAN"S
C SMR"S ARE S"S FOR MEAN MAN REM,BMR"S ARE B"S FOR MAN REM
C SALI AND BALI ARE S AND B FOR AREA LAND INTERDICTED
C LAND INTER BASED ON CS GROUP ONLY
C RF"S ARECORRAL RELEASE FRAC"S FOR KR,I,CS,TE,BA,RU,LA
      DIMENSION RF(7),ID(07),SLC(7),BLC(7),SMR(7),BMR(7),ARF(7)
      DATA SLC/-3.8E-2,-5.4E-2,-.265,+4.0E-2,-3.6E-2,-3.1E-2,0./
      DATA BLC/2.,6.,7.8,5.9,7.4,9.2,10.8/
      DATA SMR/-4.3E-2,-7.3E-2,-.250,0.,-5.E-2,-8.5E-2,-5.7E-2/
      DATA BMR/11.7,15.4,17.9,14.3,17.2,17.6,20./
      SALI=15.7
      BALI=153.
      DO 10 I=1,100
      READ(2,500)ID
      READ(2,*)RF
      RMR=0.
      RLC=0.
      DO 5 K=1,7
      IF(RF(K))5,5,2
 2    ARF(K)=ALOG(RF(K))
      RMR=RMR+EXP(SMR(K)*ARF(K)+BMR(K)+ARF(K))
      RLC=RLC+EXP(SLC(K)*ARF(K)+BLC(K)+ARF(K))
      GO TO 5
 5    CONTINUE
      ALI=(SALI*ARF(3)+BALI)*RF(3)
      IF(ALI)6,6,7
 6    ALI=0.
 7    WRITE(3,600)ID
      WRITE(3,620)RF
      WRITE(3,650)
      WRITE(3,700)RMR,RLC,ALI
 10   CONTINUE
 30   STOP
 500  FORMAT(7A8)
 600  FORMAT(1H0,10A8)
 620  FORMAT(1H0,7(E8.3,1X))
 650  FORMAT(1X,7HMAN REM,5X,14HLATENT CANCERS,5X,10HINTER LAND)
 700  FORMAT(1X,E8.3,4X,E8.3,4X,E8.3)
      END
```

Table D-3

CONSEQ Results for Zion

```
V BYPASS

.710E+00 .540E+00 .460E+00 .800E-01 .600E-01 .200E-01 .200E-02
MAN REM      LATENT CANCERS      INTER LAND
.408E+08     .207E+04     .648E+02

V PRIM

.350E-03 .840E-04 .100E-04 .990E-08 .0        .0        .0
MAN REM      LATENT CANCERS      INTER LAND
.114E+05     .576E+00     .0

V 1/2 MELT

.310E+00 .160E+00 .150E+00 .200E-01 .200E-01 .600E-02 .600E-03
MAN REM      LATENT CANCERS      INTER LAND
.169E+08     .822E+03     .185E+02

TMLB RED PAR ISOL FAIL

.400E-01 .100E-01 .300E-01 .300E-01 .300E-02 .200E-02 .400E-03
MAN REM      LATENT CANCERS      INTER LAND
.498E+07     .250E+03     .294E+01

TMLB 6 HR FAIL

.970E+00 .430E+00 .340E+00 .380E+00 .400E-01 .300E-01 .500E-02
MAN REM      LATENT CANCERS      INTER LAND
.359E+08     .208E+04     .463E+02

TMLB ISOL FAIL

.440E-01 .140E-01 .140E-01 .140E-01 .150E-02 .100E-02 .200E-03
MAN REM      LATENT CANCERS      INTER LAND
.284E+07     .143E+03     .120E+01

TMLB ECC 145 MIN NO CONT ESF

.170E-03 .640E-04 .430E-04 .740E-05 .500E-05 .150E-05 .150E-06
MAN REM      LATENT CANCERS      INTER LAND
.329E+05     .160E+01     .0

TMLB SPRAY 100 PSIG

.630E+00 .200E-02 .200E-01 .300E-01 .200E-02 .200E-02 .400E-03
MAN REM      LATENT CANCERS      INTER LAND
.383E+07     .201E+03     .183E+01
```

Table D-3 (Cont'd.)

```
TML3 FEED BLEED

.230E-03 .450E-07 .270E-08 .270E-08 .0         .270E-08 .270E-08
MAN REM      LATENT CANCERS      INTER LAND
.674E+02     .379E-02     .0

TML3 ALL ESF 145 MIN

.410E-04 .400E-05 .430E-05 .810E-06 .540E-06 .160E-06 .160E-07
MAN REM      LATENT CANCERS      INTER LAND
.575E+04     .286E+00     .0

TML3 RED I2 ISOL FAIL

.400E-01 .120E-04 .110E-01 .110E-01 .110E-02 .840E-03 .140E-03
MAN REM      LATENT CANCERS      INTER LAND
.227E+07     .112E+03     .904E+00

S1HF 4 IN PESSIM

.520E-03 .200E-04 .110E-03 .130E-03 .120E-04 .990E-05 .170E-05
MAN REM      LATENT CANCERS      INTER LAND
.679E+05     .331E+01     .109E-02

S1HF 4 IN OPTIMIS

.520E-03 .700E-06 .280E-06 .490E-06 .290E-07 .340E-07 .660E-08
MAN REM      LATENT CANCERS      INTER LAND
.838E+03     .441E-01     .0

S1HF 6 IN

.360E-03 .600E-05 .320E-04 .440E-04 .360E-05 .320E-05 .560E-06
MAN REM      LATENT CANCERS      INTER LAND
.267E+05     .132E+01     .0

S2HF 1.5 IN

.380E-03 .580E-05 .180E-04 .290E-04 .190E-05 .200E-05 .360E-06
MAN REM      LATENT CANCERS      INTER LAND
.174E+05     .862E+00     .0

S2D 1.5 IN

.480E-03 .650E-05 .190E-04 .230E-04 .210E-05 .180E-05 .300E-06
MAN REM      LATENT CANCERS      INTER LAND
.180E+05     .389E+00     .0
```

Table D-3 (Cont'd.)

```
2 S1CG NO CONT FAIL

3 .650E-03 .130E-03 .150E-03 .160E-03 .160E-04 .120E-04 .200E-05
  MAN REM      LATENT CANCERS      INTER LAND
4 .864E+05     .421E+01     .221E-02

5 S1D 6 IN

6 .520E-03 .610E-05 .510E-04 .770E-04 .550E-05 .550E-05 .960E-06
  MAN REM      LATENT CANCERS      INTER LAND
7 .381E+05     .188E+01     .0

8 S1D 6 IN ISOL FAIL

9 .370E-01 .540E-03 .290E-02 .570E-03 .380E-03 .110E-03 .110E-04
  MAN REM      LATENT CANCERS      INTER LAND
10 .790E+06     .369E+02     .178E+00

11 S1HF BEST EST

12 .510E-03 .590E-05 .250E-04 .460E-04 .250E-05 .310E-05 .570E-06
  MAN REM      LATENT CANCERS      INTER LAND
13 .224E+05     .111E+01     .0

14 S1CG CONT FAIL

15 .830E+00 .200E-01 .130E+00 .140E+00 .100E-01 .100E-01 .200E-02
  MAN REM      LATENT CANCERS      INTER LAND
16 .157E+08     .840E+03     .157E+02

17 AD

18 .490E-03 .190E-05 .280E-04 .810E-04 .220E-05 .500E-05 .970E-06
  MAN REM      LATENT CANCERS      INTER LAND
19 .249E+05     .125E+01     .0

20 AHF

21 .370E-03 .640E-05 .620E-04 .790E-04 .690E-05 .590E-05 .100E-05
  MAN REM      LATENT CANCERS      INTER LAND
22 .439E+05     .215E+01     .553E-04

23 TMLB DIESEL 118 MIN 30 MIN

24 .180E-03 .160E-04 .280E-04 .280E-04 .270E-04 .280E-04 .280E-04
  MAN REM      LATENT CANCERS      INTER LAND
25 .522E+05     .294E+01     .0

26 TMLB DESIGN LEAK

27 .480E-03 .140E-03 .960E-04 .120E-03 .150E-04 .100E-04 .160E-05
  MAN REM      LATENT CANCERS      INTER LAND
28 .628E+05     .310E+01     .745E-03
```

Table D-3 (Cont'd.)

```
TMLB RUPTURE 10 HRS

.990E+00 .160E+00 .180E+00 .220E+00 .200E-01 .170E-01 .280E-02
MAN REM      LATENT CANCERS      INTER LAND
.215E+08     .121E+04      .227E+02

TMLB" RUPTURE 10 SPRAY FAN

.990E+00 .300E-02 .700E-01 .200E+00 .600E-02 .100E-01 .200E-02
MAN REM      LATENT CANCERS      INTER LAND
.108E+08     .647E+03      .779E+01

TMLB RUPTURE 10 NO I CS

.990E+00 .0       .0       .220E+00 .200E-01 .170E-01 .280E-02
MAN REM      LATENT CANCERS      INTER LAND
.415E+07     .449E+03      .0

TMLB RUPTURE 10 GAP ONLY

.300E-01 .200E-02 .900E-02 .200E-04 .170E-06 .0       .0
MAN REM      LATENT CANCERS      INTER LAND
.176E+07     .779E+02      .711E+00

TMLB RUPUTURE 10 NO VAP

.900E+00 .140E+00 .140E+00 .260E-01 .180E-01 .500E-02 .500E-03
MAN REM      LATENT CANCERS      INTER LAND
.159E+08     .770E+03      .171E+02

TMLB RUPTURE 3.4 HRS

.100E+01 .620E+00 .400E+00 .460E+00 .400E-01 .300E-01 .600E-02
MAN REM      LATENT CANCERS      INTER LAND
.409E+08     .237E+04      .554E+02

TMLB RUPUTURE 3.4 NO I CS

.100E+01 .0       .0       .460E+00 .400E-01 .300E-01 .600E-02
MAN REM      LATENT CANCERS      INTER LAND
.793E+07     .869E+03      .0

TMLB RUPTURE 3.4 GAP ONLY

.300E-01 .100E-01 .200E-01 .320E-04 .320E-05 .0       .0
MAN REM      LATENT CANCERS      INTER LAND
.323E+07     .143E+03      .183E+01

TMLB RUPTURE 3.4 NO VAP

.900E+00 .550E+00 .300E+00 .600E-01 .400E-01 .100E-01 .100E-02
MAN REM      LATENT CANCERS      INTER LAND
.298E+08     .150E+04      .402E+02
```

Table D-3 (Cont'd)

```
TML3 RUP 3.4 0772 FRAC"S

.100E+01 .0        .0        .460E+00 .120E+00 .200E-01 .600E-01
MAN REM      LATENT CANCERS      INTER LAND
.402E+08     .355E+04     .0

TML3 RUP 3.4 0772 MOD.

.100E+01 .0        .0        .550E+00 .120E+00 .200E-01 .600E-02
MAN REM      LATENT CANCERS      INTER LAND
.101E+08     .933E+03     .0

V BYPASS HIGH TE

.710E+00 .540E+00 .460E+00 .460E+00 .600E-01 .200E-01 .200E-02
MAN REM      LATENT CANCERS      INTER LAND
.414E+08     .220E+04     .648E+02

HIGH TEST

.820E+00 .330E+00 .290E+00 .350E+00 .300E-01 .260E-01 .450E-02
MAN REM      LATENT CANCERS      INTER LAND
.317E+08     .184E+04     .387E+02

LOW TEST

.106E-02 .158E-04 .133E-04 .421E-04 .815E-06 .255E-05 .493E-06
MAN REM      LATENT CANCERS      INTER LAND
.144E+05     .732E+00     .0

ULTIMATE

.100E+01 .100E+01 .800E+00 .600E+00 .100E+00 .800E+00 .100E-01
MAN REM      LATENT CANCERS      INTER LAND
.102E+09     .113E+05     .120E+03

ZION RISK 2 RELEAS

.900E+00 .700E+00 .500E+00 .300E+00 .600E-01 .200E-01 .400E-02
MAN REM      LATENT CANCERS      INTER LAND
.453E+08     .239E+04     .711E+02

ZION RISK Z/1A RELEAS

.900E+00 .700E+00 .100E+00 .350E+00 .500E-01 .210E+00 .300E-02
MAN REM      LATENT CANCERS      INTER LAND
.290E+08     .329E+04     .117E+02

END-OF-FILE ENCOUNTERED, FILENAME -    INPUT
ERROR NUMBER    65   DETECTED BY INPF=    AT ADDRESS 000175
CALLED FROM CONSEQ  AT  LINE 20
```

APPENDIX E

HIERARCHY OF FINAL PLANT STATES

The core of a pressurized water reactor (PWR) could be uncovered by loss of primary coolant initiated by a pipe break or transient event. If the primary coolant is rapidly replenished, the core can be promptly covered again, and little or no core damage would be expected. In this case, only the radioactivity in the primary coolant would be released to the containment.

If the core remains uncovered for several minutes, the cladding could be damaged and radioactive gases could be released from the gap between the fuel pellets and the cladding; the gases could be released to the containment through pipe breaks or open relief valves. If the core is promptly recovered, the accident progression could be halted and melting could be averted.

If the core melts, some of the less refractory radioactive material could be released. If water is added at an adequate rate after melting, and a coolable configuration is formed in the pressure vessel, it is conceivable that pressure vessel failure could be prevented and no further release to containment would occur.

If the pressure vessel does fail, the core and associated structural materials would drop into the reactor cavity. If water is now added at an adequate rate, and a coolable debris bed is formed, interaction with concrete could be prevented or controlled. If the core material does interact with the concrete, additional releases of radioactive material to the containment would be expected.

If the containment Engineered Safety Features (ESFs) are operative, some of the radioactive material will be removed from the containment atmosphere, and buildup of internal pressure will be mitigated. There will, therefore, be less material in the atmosphere for release to the public and a lower driving force for release.

If the containment remains intact--that is, not grossly ruptured--only relatively small pathways exist for release to the public. If the containment isolates as designed, leakage would be very small.

In certain accidents such as interfacing-system LOCAs, the containment could be bypassed and a large fraction of the core inventory could be released to the environment.

Table E-1 shows final plant states arranged in order of outcome. More desirable outcomes are at the top of the table. The "Xs" across a given line indicate the events which have occurred. Blanks indicate that either the events have not occurred or are irrelevant. The solid lines enclose sets of plant states having

Table E-1

Hierarchy of Final Plant States for Zion

Core Damage (Gap Release)	Core Melt	Containment By-Passed	Pressure Vessel Failure	Concrete Attack	Containment ESF Failure	Gross Containment Failure	Isolation Failure[a]	Mean Values of: Man Rem, Cancer Fatalities, Area Land Interdicted (Square Miles)
					X			Negligible Consequence
							X	
					X		X	10^3,0,0[b]
X								
X					X			10^4,1,0[b]
		X					X	
X							X	10^5,10^1,0[c]
X	X							
X	X		X					
X	X		X	X				
X					X		X	
X		X						10^6,10^2,1
X	X						X	
X	X		X				X	
X	X		X	X			X	
X	X		X	X		X		10^7,10^3,10^1[d]
X	X		X	X	X	X		
X	X		X					10^7,10^3,10^2

(a) 10 Vol. %/day leak rate with isolation failure.
0.1 Vol. %/day leak rate with no isolation failure.
(b) Assuming primary coolant activity is released.
(c) Maximum consequences given that containment is not bypassed, does not fail in a gross manner above grade, and does not fail to isolate.
(d) Delayed operation of containment ESFs (see text).

nearly the same outcomes. Within each band, the plant states can be considered "equivalent" in terms of consequences.

Note that if containment is not bypassed, does not fail in a gross manner, and does not fail to isolate, mean cancer deaths are less than 10 and land interdicted is zero. This points out the importance of preserving containment in reducing consequences.

These results were calculated using MARCH, CORRAL, CRAC2, and CONSEQ. All outcomes are given to the nearest order-of-magnitude. Finer shadings of outcome are not believed to be meaningful in view of the many uncertainties involved in the calculations.

If the containment is bypassed, all subsequent branches are irrelevant, because the additional material that might be leaked from containment is very small in comparison with that which enters the environment through the bypass. Note that the bypass could potentially be isolated at any time: before core damage or before melt. The consequences would be different after melt depending on how long afterward the bypass is isolated. The outcome shown for core melt is for a bypass that is never isolated.

Pressure vessel failure and concrete attack appear not to make any difference in the results. This is due to rounding the consequences to the nearest order-of-magnitude. If the melt is promptly cooled either in- or ex-vessel, there is no vaporization release. However, the vaporization release component makes less than an order of magnitude change in consequences. Of course, clean up would be easier after the accident if the pressure vessel has not failed, so that it is preferable not to have pressure vessel failure.

The containment ESFs serve two functions: to control the pressure within containment, and to remove radionuclides from the containment atmosphere. Without containment ESFs, containment will probably eventually fail in a gross manner given core meltdown. The effect on consequences due to the ESFs removing radionuclides is minor compared to the effect due to preserving containment.

There is the possibility of gross containment failure if containment ESF's are not available until after concrete attack has begun. Hydrogen can build up to appreciable levels and not burn due to the high steam content of the atmosphere. With restoration of containment ESFs, steam is condensed and hydrogen burning may take place and fail containment.

If containment is not bypassed and does not fail in a gross manner, the degree to which containment can be isolated affects consequences. The leak rate in the event of isolation failure depends on the particular path to the atmosphere that happens to be open. It is assumed that the leak rate for isolation failure is 100 times the design rate. (See Section 5.2.)

APPENDIX F

MARCH INPUT

Table F-1 lists all of the MARCH input parameters for an S2HF accident initiated by a 1.5-inch diameter cold leg break at Zion (see Section 4.2.1), Table F-2 indicates the changes to this S2HF run required to generate the other MARCH runs discussed in Sections 4 and 8 of this report. Table F-3 explains our seven-character MARCH-input-file designators.

The primary sources of information used in constructing the MARCH input for Zion were the Zion FSAR [Zion-FSAR], the Zion system descriptions [Zion-SD], and the INEL RELAP4 input deck for Zion [Dearien, 1980b]. Because some of our calculations were used to compare MARCH with existing RELAP4 results for Zion and because the RELAP4 deck was constructed based on actual plant data, we often selected RELAP4 input values rather than corresponding values from the Zion-FSAR or system descriptions. In general, differences between these primary sources of data are not large.

Table F-1

MARCH Input Parameters for S2HF
(1.5-Inch Diameter Break)

```
   820129ZSHF4A    ZSHF4A --ZION S2HF BASE CASE
 $CHANGE
  ACBRK=-1.,   CPSTP=180.,  FDRP=-1.,    HIMX=-1.,    HIOX=-1.,    ID=820129,
  IFISH=-1,    IFPM=10,     IFPV=10,     IGASX=10,    IHOTX=10,    IPLOT=0,
  IS=10,       JS=0,        LST7=1,      MEL=10,      NCRST=1,     NCT7=1,
  PFAIL=-1.,   PRST=200.,   TFX=-1.,     TMX=-1.,     TRST=1000.,  WALLX=-1.,
  $END
                ZSHF4A --ZION 1.5' S2HF BASE CASE
 $NLMAR
  DTINIT=.02,  H2HI=.10,    H2LO=0.,
  IBRK=1,      IBURN=1,     ICBRK=1,     ICE=0,       ICKV=0,
  IECC=-2,     IFPSM=2,     IFPSV=2,     IPDEF=0,     IPDTL=7,     IPLOT=0,
  ISPRA=-1,    ITRAN=1,     IU=0,        IXPL=0,      NINTER=20,   NPAIR=0,
  TAP=1.0512E6,TBURN=.1,    TIME=0.,                  VOLC=2.715E6,
  $END
 $NLINTL
  EW(1)=20*0.,
  T(1)=20*0.,
  W(1)=20*0.,
  $END
STEEL     STEEL+    CONCRETE  CONCRETE+
CYLINDER  DOME      FLOOR     RV CAVITY CRANE WALLOP.DECK   SHIELD WLSCANAL
MISC.STEEL
 $NLSLAB
  DEN(1)=511.,511.,150.,146.,0.,
  DTDX(1)=15*0.,
  HC(1)=.11,.11,.186,.24,0.,
  HIF(1)=2*20.,13*0.,
  IPRINT=0,
  IVL(1)=9*1,6*0,
  IVR(1)=9*1,6*0,
```

Table F-1 (Cont'd.)

```
 MAT1(1)=1,1,4,1,3,3,3,2,1,6*0,
 MAT2(1)=3,3,4,1,3,3,3,2,1,6*0,
 NMAT=4,
 NMO1(1)=4,4,8,4,8,8,7,4,4,6*0,
 NMO2(1)=8,8,13*0,
 NOD(1)=1,4,5,12,
 NSLAB=9,
 SAREA(1)=7.75E4,1.95E4,1.55E4,2.E3,3.1E4,5.E3,7.E3,1.6E4,5.E4,6*0.,
 TC(1)=26.,9.4,.8,1.6,0.,
 TEMP(1)=200*110.,
 X(1)=.0,     .005,  .01,   .02083,
      .04083,.07083,.12083,.22083,.52083,1.02083,2.02083,3.52083,
      .0,     .005,  .01,   .02083,
      .04083,.07083,.12083,.22083,.52083,1.02083,2.02083,2.6875,
      .0,     .01,   .02,   .05,   .1,     .2,     .5,    1.,
      .0,     .005,  .01,   .02083,
      .0,     .01,   .02,   .05,   .1,     .2,     .5,    1.,
      .0,     .01,   .02,   .05,   .1,     .2,     .5,    1.,
      .0,     .01,   .02,   .05,   .1,     .25,    .75,
      .0,     .005,  .01,   .02083,
      .0,     .005,  .01,   .02083,
 $END
$NLECC
 ACMO=201900.,CSPRC=0.,    DTSUB=-100., ECCRC=0.,    NP=0,
 P(1)=6*0.,
 PACMO=614.7, PHH=2649.,   PLH=200.,  PSIS=1602., PUHIO=0., RWSTM=3.2251E6,
 STP(1)=6*1.E6,
              STPHH=1.E6,  STPLH=1.E6,  STPSIS=1.E6,
 TACM=125.,
 TM(1)=6*1.E6,
              TMHH=0.,     TMLH=0.,     TMSIS=0.,               TRUST=100.,
                                                     TUHI=100.,
                                                     UHIO=0.,
 WEC(1)=6*0.,
              WHH1=-949.,  WLH1=-2332., WSIS1=-1045.,
 WTCAV=100.,
 $END
```

Table F-1 (Cont'd.)

```
$NLECX
 EQR=5.6E7,   ETP1R=137.5, ETS1R=107.1, EWPR=3.9E6,  EWSR=4.96E6,
 $END
$NLCSX
 SQR=0.,      STP1R=0.,    STS1R=0.,    SWPR=0.,     SWSR=0.,
 $END
$NLCOOL
 CQR=3.24E8,  CTPR=271.,   CTSR=80.,    CWPR=2.12E5, CWSR=6.48E4,
 JCOOL=1,     NCOOL=0,     PCOOL=0.,    POFF=0.,     QRCOOL=0.,TCOOL=0.,
 $END
$NLMACE
 AREA(1)=1.55E4,9*0.,
 AVBRK=0.,
 CVBRK=0.,
 C1(1)=169.,37.7,8*0.,
 C2(1)=.583,7800.,8*0.,
 C3(1)=7.,100.,8*0.,
 C4(1)=0.,500.,8*0.,
 DCF=100.,    DCFICE=100., DTO=.05,     DTS=5.E3,    DTPNT=20.,
 FALL=.7,     FSPRA=0.,    HMAX=280.,
 HUM(1)=10*1.,
 IBETA=0,     ICECUB=0,    IDRY=1,      IVENT=0,     IWET=2,
 KT(1,1)=100*0,
 N=2,
 NC(1)=1,1,8*0,
 NCAV=1,      NCUB=1,      NRPV1=1,     NRPV2=1,
 NS(1)=2,2,8*0,
 NSMP=1,      NSMP2=1,
 NT(1)=-7,1,8*0,
 PO=14.7,     PUNT=0.,     STPECC=1.E6, STPSPR=1.E6,
 TEMPO(1)=110.,9*100.,
 TICE=20.,    TPOOL=0.,    TSTM=105.,   TVNT1=0.,    TVNT2=0.,
 TWTR=190.,   TWTR2=130.,
 VC(1)=2.715E6,9*0.,
 VCAV=4.5E4,  VDRY=0.,     VFLR=2.E3,   VTORUS=0.,
 WICE=0.,     WPOOL=0.,    WVMAKS=5.6E5,WVMAX=0.,
 $END
```

Table F-1 (Cont'd.)

```
$NLBOIL
 AB(1)=16*0.,
 ABRK=.01227, ACOR=53.386,
 AH(1)=100.,188.,158000.,150.,150.,365.5,0.,
 ANSK=0.,
 AR(1)=50.,.78,36.,-.323,-3.653,-4.323,0.,
 ATOT=98.457,
 CM(1)=1200.,912.,504000.,1050.,3000.,9350.,0.,
 CLAD=.001896,CSRV=1E2.,    D=.03517,    DC=10.,
 DD(1)=.3,1.,.065,.29,.83,.4609,0.,
 DF=.03041,   DH=.04453,   DPART=.0164, DTK=1000.,    DTPN=1.,      DTPNTB=20.,
 DUO2=.03049,
 F(1)=0.49,0.63,0.80,0.94,1.07,1.18,1.27,1.35,1.41,1.43,1.45,1.44,
      1.43,1.39,1.32,1.24,1.14,1.02,0.89,0.74,0.59,0.43,0.26,0.09,
 FCOL=.728,
 FDCR=.4,     FDROP=.728,  FM=0.,        FR=0.,        FULSG=352507.,FZMCR=1.,
 FZOCR=1.,    FZOS1=0.,    F12=.445,     H=12.,        HO=104.93,    HW=300.,
 ICON=-1,     IFP=2,       IGRID1=1,     IGRID2=0,     IHC=0,        IHEAD=-1,
 IHR=1,       IMWA=3,      IMZ=100,      ISAT=0,       ISG=3,        ISRV=1,
 ISTM=0,      ISTR=3,      KRPS=0,       MELMOD=1,     MWORNL=0,
 NDTM=100000, NDZ=24 ,     NDZDRP=2,     NNT=43425,    NR=39372,
 PF(1)=1.09,1.11,1.1,1.1,1.12,1.11,1.09,1.1,1.01,.75,
 PSET=2349.7, PSG=1190.,   PVSL=2257.57,
 QPUMP1=0.,   QPUMP2=0.,   QZERO=1.10487E10,
 RHOCU=53.053,R1=1,        R2=10,
 TAFW=100.,   TALF1=1.E10, TALF2=1.E10,
 TB(1)=16*1.E6,
 TCAV=1251.0, TDK=0.,      TFAIL=1832., TFEOO=563.,  TFUS=5320.,  TGOO=563.,
 TMAFW=1.2133,TMELT=4130.,
 TMLEG(1)=3*1.E6,
 TMSG1=1.E6,  TMSG2=1.E6,  TMUP1=1.E6,  TMUP2=1.E6,  TMYBK=1.E6,  TPM=1.,
 TPN=1.E6,    TPUMP1=1.E6, TPUMP2=1.E6, TRPS=0.,
 TSB(1)=.25,3*1.,
 TSCT(1)=0.,3*1.E10,
 TT(1)=563.,563.,512.,563.,563.,563.,0.,
 VF(1)=.047,.062,.083,.062,.062,.062,.083,.124,.166,.249,
 VOLP=12481.6,VOLS=422.3,  WAFW=8197.2, WATBH=47210.,WCST=1.E8,  WDED=30489.,
 WFE2=8000.,  WMUP1=0.,    WMUP2=0.,    WTRSG=352507.,XOO=3.28E-6,
 YB=0.,       YBRK=16.,    YBRK2=1.E3,  YLEG=16.,    YLEG2=1.E3,  YT=0.,
 $END
```

Table F-1 (Cont'd.)

```
$NLRAD
 ECROS=.7,      ELONG=.9,      ESTRU=.6,      EWAT=.95,      IAXC=0,        ICONV=0,
 IRAD=0,        PITCH=.04692,WBAR=4770.,
$END
$NLHEAD
 COND=5.,       DBH=14.83,     E1=.8,         E2=.5,        FOPEN=0.,
 THICK=.4609,  TMLT=4130.,    TVSL=500.,
 WFEC=14853., WGRID=37000.,WHEAD=85000.,WUO2=216600.,WZRC=44547.,
 $END
$NLHOT
 CON=5.,        DP=.19685,     FLRMC=0.,      IHOT=0,        MWR=0,         TPOOLH=125.,
 WTR=0.,
 $END
$NLINTR
 CAYC=.015,     CPC=1.45,      DENSC=2.35,    DPRIN=1200., DT=.5,
 EPSI(1)=.5,.5,8*0.,
 FC1=.46,       FC2=.14,       FC3=.36,       FC4=.04,       FIOPEN=.5,
 HIM=.01,       HIO=.01,       IGAS=1,        IWRC=1,        NEPS=2,
 R=6000.,       RBR=.135,      R0=313.,       TAUL=.5,       TAUS=5.,
 TEPS(1)=0.,3.6E7,8*0.,
 TIC=293.,      TF=3.601E4,    TPRIN=1200., WALL=300.,     ZF=1000.,
 $END
```

Table F-2

MARCH Input Parameters for Various Zion Accidents

Figure	Input File	Description/ Changes from Reference File	Reference File
4.1.2-1	ZTMLBA*	Zion TMLB' Base Case CHANGE: TRST=600., NLMAR: IBRK=0, IECC=0, ISPRA=0, NLECC: TMHH=TMLH=TMSIS=1.E6, NLCOOL: TCOOL=1.E6, NLMACE: N=1, NLBOIL: ABRK=0., PSET=2499.7, TMAFW=1. E6, YBRK=37.65,	ZSHF4A
4.2.1-1 4.2.1-2 4.2.1-3	ZSHF4A**	Zion 1.5" S2HF Base Case	ZSHF4A
	ZSHF4A1	Zion 1.5" S2HF, RELAP4-MARCH NLMAR: NPAIR=189, DTINIT=.1, IBLDF=1, NLECC: ACMO=17780., RWSTM=0., NLBOIL: HO=70.236, PVSL=248., TCAV=404., TFEOO=380., TGOO=396., TT(1)=380., 393., 3*380., WATBH=56490., WDED=67841., WTRSG=4.042E5,	ZSHF4A
4.2.1-4	ZSHF4A1	Zion 1.0" S2HF Base Case CHANGE: TRST=2000., NLBOIL: ABRK=.01091,	ZSHF4A
	ZSHF4A	Zion 1.5" S2HF Base Case	ZSHF4A
	ZSHF4A2	Zion 2.0" S2HF Base Case NLBOIL: ABRK=.01282,	ZSHF4A6
	ZSHF4A3	Zion 3.0" S1HF Base Case NLBOIL: ABRK=.04909,	ZSHF4A6
	ZSHF4A4	Zion 4.0" S1HF Base Case NLBOIL: ABRK=.08727,	ZSHF4A6
	ZSHF4A5	Zion 5.0" S1HF Base Case NLBOIL: ABRK=.13635,	ZSHF4A6
	ZSHF4A6*	Zion 6.0" S1HF Base Case CHANGE: TRST=1000., NLBOIL: ABRK=.19635,	ZSHF4A
	ZSHF4A6	Zion 6.0" S1C Base Case CHANGE: TRST=200., NLMAR: IECC=+2, ISPRA=0, NLMACE: N=1,	ZSHF4A6

4.2.3.1	ZSGGGA6	Zion 6.0" S1G Base Case NLMAR: IECC=2, ISPRA=+1, NLECX: EQR=0., NLCOOL: TCOOL=1.E6, NLMACE: STPSPR=54.44, STPECC-54.44,	ZSHF4A6
	ZS63GA6	Zion 6.0" S1G, No Recirc. Failure NLMAR: IECC=+2, ISPRA=1, NLECX: EQR=0., NLCOOL: TCOOL=1.E6,	ZSHF4A6
	ZSGCGA6	Zion 6.0" S1CG Base Case NLMAR: IECC=+2, ISPRA=0 NLECX: EQR=0., NLCOOL: TCOOL=1.E6, NLMACE: N=1,	ZSHF4A6
4.2.4-1	ZSDF4A6*	Zion 6.0" S1D Base Case NLMAR: IECC=+2, NLECC: TMHH=TMLH=TMSIS=1.E6,	ZDHF4A6
	ZSDC4A6	Zion 6.0" S1D Base Case NLMAR: ISPRA=0, NLMACE: N=1,	ZSDF4A6
4.2.4-2	ZSDF4A1	Zion 1.0" S2D Base Case NLBOIL: ABRK=.01091,	ZSDF4A6
	ZSDF4A	Zion 1.5" S2D Base Case NLBOIL: ABRK=.01227,	ZSDF4A6
	ZSDF4A2	Zion 2.0" S2D Base Case NLBOIL: ABRK=.02182,	ZSDF4A6
	ZSDF4A3	Zion 3.0" S1D Base Case NLBOIL: ABRK=.04909,	ZSDF4A6
	ZSDF4A4	Zion 4.0" S1D Base Case NLBOIL: ABRK=.08727,	ZSDF4A6
	ZSDF4A5	Zion 5.0" S1D Base Case NLBOIL: ABRK=.13635,	ZSDF4A6
	ZSDF4A6*	Zion 6.0" S1D Base Case	ZSDF4A6
4.2.5-1	ZSDFGA6	Zion 6.0" S1DG Bise Case NLECX: EQR=0.0, NLCOOL: TCOOL=1.E6, NLMACE: STPSPR=65.2,	ZSDF4A6

	ZSDCGA6	Zion 6.0" SlCDG Base Case NLMAR: ISPRA=0, NLCOOL: TCOOL=1.E6, NLMACE: N=1,	ZSDF4A6
4.3-1	ZAHF4A*	Zion AHF Base Case NLMAR: ITRAN=0, NPAIR=2, NLINTL: EW(1)=2*594.72, T(1)=0.,0.36, W(1)=2*1.6386E6, NLECC: ACMO=1.035E5, NLMACE: WVMAKS=1.68E5, NLBOIL: DTPNTB=5., HO=-1.507, HW=300., PVSL=70., TCAV=430., TGOO=303., TFEOO=422., TT(1)=2*422., TT(4)=3*422., VOLS=11580, WATBH=60000., WDED=100.,	ZSHF4A6
4.3.-2	ZADF4A	Zion AD Base Case NLMAR: IECC=+2, NLECC: TMHH=TMLH=TMSIS=1.E6,	ZAHF4A
4.3-3	ZAHCGA	Zion AHCG Base Case NLMAR: ISPRA=0, NLCOOL: TCOOL=1.E6, NLMACE: N=1,	ZAHF4A
	ZAHCGA	Zion AHCG, Minimum Injection NLMAR: ISPRA=0, NLECC: RWSTM=1.5E5, NLCOOL: TCOOL=1.E6, NLMACE: N=1,	ZAHF4A
8.2-1	ZTMLBAB	Zion TMLB', ECC on at 145 min. CHANGE: MEL=-1, NLMAR: IECC=+2, NLECC: TMHH=TMSIS=145,	ZTMLBA
	ZTMLBAC	Zion TMLB', ECC on at 160 min. CHANGE: MEL=-1, NLMAR: IECC=+2, NLECC: TMHH=TMSIS=160.,	ZTMLBA
8.2-2	ZTMLBAD	Zion TMLB', All ESFs at 145 min. CHANGE: MEL=-1, NLMAR: IECC=+2, ISPRA=+1, NLECC: TMHH=TMSIS=145., NLCOOL: TCOOL=145., NLMACE: C1(2)=145., N=2, NS(2)=1,	ZTMLBA

	ZTMLBAE	Zion TMLB', All ESFs on at 160 min. CHANGE: MEL=-1, NLMAR: IECC=+2, ISPRA=+1, NLECC: TMHH=TMSIS=160., NLCOOL: TCOOL=160., NLMACE: C1(2)=160., N=2, NS(2)=1,	ZTMLBA
8.3.2-1	ZTMLBAF	Zion TMLB', 4 Fan Coolers at 360 min. CHANGE: TRST=600., NLMAR: H2HI=0.09, IBRK=0, IECC=0, ISPRA=0, NLECC: TMHH=TMLH=TMSIS=1.0E+06, NLCOOL: TCOOL=360.0, NLMACE: N=1, NLBOIL: ABRK=0., CLAD=2.566E=03, PSET=2499.7, TMAFW=1.0E+06, YBRK=37.65, NLHEAD: WFEC=0.0, WZRC=5.940E+04, NLHOT: IHOT=0,	ZSHF4A
8.3.2-1	ZTMLBAG	Zion TMLB', 1 Spray Train at 360 min. NLMAR: H2HI=0.09, ISPRA=+1, NLCSX: SQR=5.6E+07, SWPR=3.9E+06, SWSR=4.96E+06, STP1R=137.5, STS1R=107.1, NLMACE: NS(2)=1, C1(2)=360.0, C2(2)=2.6E+03, N=2, UPDATE Change: *DELETE MARCH.5079 IF (FHYD.GT.H2HI.AND.FOXY.GT.0.050) IBURN(I)=1	ZTMLBA
8.3.2-1	ZTMLBAH	Zion TMLB', 3 Spray Trains at 360 min. NLMAR: H2HI=0.09, ISPRA=+1, NLCSX: SQR=5.6E+07, SWPR=3.9E+06, SWSR=4.96E+06, STP1R=137.5. STS1R=107.1, NLMACE: NS(2)=1, C1(2)=360.0, N=2,	ZTMLBA
8.3.3-1	ZTMLBAI	Zion TMLB', Fan Coolers at 428 min. NLMAR: H2HI=0.09, ISPRA=0, NLCOOL: TCOOL=428.0, NLMACE: N=1, UPDATE Change: *DELETE MARCH.5079 IF (FHYD.GT.H2HI.AND.FOXY.GT.0.050) IBURN(I)=1	ZTMLBA

8.3.3-1	ZTMLBAJ	Zion TMLB', 3 Spray Trains at 458 min. NLMAR: H2HI=0.09, ISPRA=+1, NLCSX: SQR=5.6E+07, SWPR=3.9E+06, SWSR=4.96E+06, STP1R=137.5, STS1R=107.1, NLMACE: NS(2)=1, C1(2)=458.0, N=2, UPDATE Change: *DELETE MARCH.5079 IF (FHYD.GT.H2HI.AND.FOXY.GT.0.050) IBURN(I)=1	ZTMLBA
8.3.4-1	ZTMLBAK	Zion TMLB', Vent before Core Melting NLCOOL: TCOOL=360.0, NLMACE: NT(1)=7, C1(1)=17.68, C2(1)=-.785, C3(1)=0.545, C4(1)=1.00, NS(2)=1, NT(2)=7, C1(2)=131.95, C2(2)=0.785, C3(2)=0.00, C4(2)=0.00,	ZTMBLA
8.3.4-1	ZTMLBAL	Zion TMLB', Vent before Core Melting, 2 Ft. Dia. Vent, Fan Coolers at 360 minn. NLCOOL: TCOOL=360.0, NLMACE: NT(1)=7, C1(1)=17.68, C2(1)=0.785, C3(1)=3.142, C4(1)=1.00, NS(2)=1, NT(2)=7, C1(2)=131.70, C2(2)=0.785, C3(2)=0.00, C4(2)=0.00	ZTMBLA

*Used as a reference file elsewhere in table.
**This is the reference file listed in Table F-1.

Table F-3

Seven-Character MARCH-Input-File Designation

First Character - Plant

I ______ - Indian Point
Q ______ - Sequoyah
S ______ - Surry
X ______ - Example
Z ______ - Zion

Second Character - Accident Initiator

_A______ - Large LOCA, double-ended cold leg break unless otherwise indicated by seventh character.
_S______ - Small LOCA, 1.5" diameter cold leg break unless otherwise indicated by seventh character.
_T______ - Transient Event.
_V______ - Interfacing ECSS LOCA.

Third, Fourth, and Fifth Character - Events

For LOCAs, Third Character - ECCS
__D_____ - Only accumulators function.
__G_____ - ECC recirculation failure due to high sump liquid temperature.
__H_____ - ECC pump trains fail on swithover to recirculation.
__6_____ - All ECC pump trains operate on demand.

For LOCAs, Fourth Character - Containment Sprays

___C____ - All spray trains inoperativve.
___F____ - All spray trains function on demand in injection mode, inoperative in recirculation mode.
___G____ - All spray trains function on demand in injection mode, fail in recirculation mode due to high sump liquid temperature.
___3____ - All spray trains operate on demand.

For LOCAs, Fifth Character - Containment Fan Coolers

____G___ - All fan coolers are inoperable.
____4___ - Four fan coolers operate continuously.

Sixth Character - MARCH Version and Plant Deck

Z____A__ - MARCH 1.1, 811224 Zion Plant Deck

Seventh Character - Variations

For Small LOCAs, Seventh Character - Break Size

_S_____1 - 1.0" diameter.

_S_____2 - 2.0" diameter.

_S_____3 - 3.0" diameter.

_S_____4 - 4.0" diameter.

_S_____5 - 5.0" diameter.

_S_____6 - 6.0" diameter.

APPENDIX G

HYDROGEN RECOMBINER EFFECTIVENESS

Zion employs a hydrogen recombiner system to keep hydrogen concentration within containment below the lower flammability limit during the design-basis loss-of-coolant accident. During an accident, one recombiner unit can be moved into place in the auxiliary building. Containment gas is drawn into the recombiner at 50 cfm; radiant heat from electric heaters enables hydrogen and oxygen to combine into water vapor. The product gases are returned to containment. Tests indicate that the system can safely recombine all hydrogen for concentrations up to 5% by mole in the inlet gas [Zion-FSAR].

The recombiner is not sized to keep hydrogen below flammable concentrations during degraded-core accidents in which gross cladding oxidation occurs.

In certain accident sequences, TMLB' for example, containment heat removal is not available. Hydrogen burning does not occur during such accidents due to the high steam content of the containment atmosphere. If heat removal capability is restored late in the accident and if plant operators attempt to condense steam to reduce the pressure inside containment, the hydrogen that has accumulated throughout the course of the accident may burn and lead to containment failure. To avoid hydrogen burning, the operators might attempt to remove hydrogen using the recombiner, before condensing steam. The recombiners would probably not function in such an environment [Sherman, et al., 1980]. Even if they were modified to function, for this mitigative strategy to work, the recombiner would have to remove hydrogen faster than it was being generated by core/concrete interactions.

For the TMLB' accident, at 600 minutes the containment atmosphere consists of 4.8 mole% hydrogen, 2.9 mole% oxygen, and 77 mole% steam. Assuming that a recombiner can function properly with such a high-steam, low-oxygen content and that it can remove all hydrogen that passes through it, the removal rate would be 3.8×10^{-2} lb mole H_2/min. The INTER subprogram of MARCH estimates the hydrogen generation rate from concrete to be 8.1×10^{-1} lb mole H_2/min at 600 minutes into the accident. The recombiner cannot remove hydrogen quickly enough to be useful until very late (on the order of days) into the accident when the hydrogen generation rate from concrete is substantially reduced.* Steam pressure

*Both the INTER and CORCON1 codes predict hydrogen generation rates from core/concrete interactions considerably in excess of the recombiner capacity. However, neither of these codes is applicable for solid-debris penetration of concrete. During such penetration (i.e., after approximately one day) the recombiners could possibly be used in a mitigative strategy, if they can function in a high steam environment.

within containment must be controlled to be kept below failure limits but above limits at which hydrogen could burn, until the recombiner can be used to effectively remove hydrogen.

Containment loading of concern in this study are quasi-static; the load on containment is long in duration compared with the structural response time. This is the case for both steam spikes and hydrogen burns [NUREG/CR-1561, 1980]. Impulsive loads, as from hydrogen detonation, were not addressed. We have assumed complete mixing of containment atmospheric contents; computed mole fractions of hydrogen, steam, and air led to deflagration before detonation conditions were ever achieved.

APPENDIX H

INSTRUMENTATION FOR MONITORING SEVERE ACCIDENTS

The NRC has divided accident monitoring parameters into categories based on ANSI/ANS-4.5-1980, Criteria for Accident Monitoring Functions in Light Water Cooled Reactors [ANSI/ANS-4.5, 1980]. This guide was prepared by Working Group 4.5 or Subcommittee ANS-4 with two objectives:

1. To address that instrumentation that permits plant operators to monitor the expected parameter changes during an accident.

2. To address extended range instrumentation deemed appropriate for the possibility of encountering previously unforeseen events.

The standard defines three types of plant variables for the purpose of selecting appropriate instrumentation for accident monitoring. The three types of variables defined are:

Type A - those variables that provide "primary information" needed to permit control room operating personnel to take specific manually controlled actions for which no automatic control is provided, and that are required for safety systems to accomplish their safety functions for design basis events.

Type B - those variables that provide information to indicate whether plant safety functions are being accomplished.

Type C - those variables that provide information to indicate the potential for being breached or the actual breach of the barriers to fission product release.

"Primary information" is defined as information which is essential for the direct accomplishment of specified safety functions.

In addition to the accident monitoring parameter classifications defined in ANSI/ANS-4.5-1980, the NRC has recommended installation of the capability to monitor the operation of individual safety systems and continuously assess radioactive releases to the environment. Two additional plant parameter classifications are defined in Regulatory Guide 1.97, Revision 2 [RG1.97, 1981]:

Type D - those variables that provide information to indicate the operation of individual safety systems and other systems important to safety.

Type E - those variables to be monitored as required for use in determining the magnitude of releases of radioactive materials and continuously assessing such releases.

The minimum sets of Type B, C, D, and E variables to be monitored for PWRs as indicated in Regulatory Guide 1.97 are presented in Table H.1. The necessary instrument ranges and qualification categories are also included in the tables. Type A variables are plant specific, and are not included in Regulatory Guide 1.97. Categories B-E are not mutually exclusive. Instrumentation to monitor variables in more than one category must meet the more stringent qualification requirements.

There is disagreement as to whether the set of variables defined in Regulatory Guide 1.97 is too inclusive or does not include enough variables to ensure that adequate information will be available in accident situations. One problem with the new requirements is that the technology does not exist for monitoring some of the parameters as indicated in the new guidelines. Efforts are underway aimed at developing new instrumentation for monitoring plant parameters over wider ranges with a high degree of reliability.

All plants which will go into operation after June of 1983 are to meet the provisions of Revision 2 of Regulatory Guide 1.97. Plants currently in operation should meet the provisions of the guide, except for minor modifications, by June of 1983.

Table H-1

PWR Variables to be Monitored per Regulatory Guide 1.97, Revision 2

PWR VARIABLES

TYPE A Variables: those variables to be monitored that provide the primary information required to permit the control room operator to take specific manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for design basis accident events. Primary information is information that is essential for the direct accomplishment of the specified safety functions; it does not include those variables that are associated with contingency actions that may also be identified in written procedures.

A variable included as Type A does not preclude it from being included as Type B, C, D, or E or vice versa.

Variable	Range	Category (see Regulatory Position 1.3)	Purpose
Plant specific	Plant specific	1	Information required for operator action

TYPE B Variables: those variables that provide information to indicate whether plant safety functions are being accomplished. Plant safety functions are (1) reactivity control, (2) core cooling, (3) maintaining reactor coolant system integrity, and (4) maintaining containment integrity (including radioactive effluent control). Variables are listed with designated ranges and category for design and qualification requirements. Key variables are indicated by design and qualification Category 1.

Variable	Range	Category	Purpose
Reactivity Control			
Neutron Flux	10^{-6}% to 100% full power	1	Function detection; accomplishment of mitigation
Control Rod Position	Full in or not full in	3	Verification
RCS Soluble Boron Concentration	0 to 6000 ppm	3	Verification
RCS Cold Leg Water Temperature[1]	50°F to 400°F	3	Verification
Core Cooling			
RCS Hot Leg Water Temperature	50°F to 750°F	1	Function detection; accomplishment of mitigation; verification; long-term surveillance
RCS Cold Leg Water Temperature[1]	50°F to 750°F	1	Function detection; accomplishment of mitigation; verification; long-term surveillance
RCS Pressure[1]	0 to 3000 psig (4000 psig for CE plants)	1[2]	Function detection; accomplishment of mitigation; verification; long-term surveillance

[1] Where a variable is listed for more than one purpose, the instrumentation requirements may be integrated and only one measurement provided.

[2] The maximum value may be revised upward to satisfy ATWS requirements.

Table H-1 (continued)

Variable	Range	Category (see Regulatory Position 1.3)	Purpose
TYPE B (Continued)			
Core Cooling (Continued)			
Core Exit Temperature[1]	200°F to 2300°F (for operating plants - 200°F to 1650°F)	3[3]	Verification
Coolant Level in Reactor	Bottom of core to top of vessel	1 (Direct-indicating or recording device not needed)	Verification; accomplishment of mitigation
Degrees of Subcooling	200°F subcooling to 35°F superheat	2 (With confirmatory operator procedures)	Verification and analysis of plant conditions
Maintaining Reactor Coolant System Integrity			
RCS Pressure[1]	0 to 3000 psig (4000 psig for CE plants)	1[2]	Function detection; accomplishment of mitigation
Containment Sump Water Level[1]	Narrow range (sump), Wide range (bottom of containment to 600,000-gallon level equivalent)	2 1	Function detection; accomplishment of mitigation; verification
Containment Pressure[1]	0 to design pressure[4] (psig)	1	Function detection; accomplishment of mitigation; verification
Maintaining Containment Integrity			
Containment Isolation Valve Position (excluding check valves)	Closed–not closed	1	Accomplishment of isolation
Containment Pressure[1]	10 psia to design pressure[4]	1	Function detection; accomplishment of mitigation; verification

[3] A minimum of four measurements per quadrant is required for operation. Sufficient number should be ins lled to account for attrition. (Replacement instrumentation should meet the 2300°F range provision.)

[4] Design pressure is that value corresponding to ASME code values that are obtained at or below code-all able values for material design stress.

Table H-1 (continued)

TYPE C Variables: those variables that provide information to indicate the potential for being breached or the actual breach of the barriers to fission product releases. The barriers are (1) fuel cladding, (2) primary coolant pressure boundary, and (3) containment.

Variable	Range	Category (see Regulatory Position 1.3)	Purpose
Fuel Cladding			
Core Exit Temperature[1]	200°F to 2300°F (for operating plants - 200°F to 1650°F)	1[3]	Detection of potential for breach; accomplishment of mitigation; long-term surveillance
Radioactivity Concentration or Radiation Level in Circulating Primary Coolant	1/2 Tech Spec limit to 100 times Tech Spec limit, R/hr	1	Detection of breach
Analysis of Primary Coolant (Gamma Spectrum)	10 μCi/gm to 10 Ci/gm or TID-14844 source term in coolant volume	3[5]	Detail analysis; accomplishment of mitigation; verification; long-term surveillance
Reactor Coolant Pressure Boundary			
RCS Pressure[1]	0 to 3000 psig (4000 psig for CE plants)	1[2]	Detection of potential for or actual breach; accomplishment of mitigation; long-term surveillance
Containment Pressure[1]	10 psia to design pressure[4] psig (5 psia for subatmospheric containments)	1	Detection of breach; accomplishment of mitigation; verification; long-term surveillance
Containment Sump Water Level[1]	Narrow range (sump), Wide range (bottom of containment to 600,000-gal level equivalent)	2 1	Detection of breach; accomplishment of mitigation; verification; long-term surveillance
Containment Area Radiation[1]	1 R/hr to 10^4 R/hr	3[6,7]	Detection of breach; verification
Effluent Radioactivity - Noble Gas Effluent from Condenser Air Removal System Exhaust[1]	10^{-6} μCi/cc to 10^{-2} μCi/cc	3[8]	Detection of breach; verification

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[5] Sampling or monitoring of radioactive liquids and gases should be performed in a manner that ensures procurement of representative samples. For gases, the criteria of ANSI N13.1 should be applied. For liquids, provisions should be made for sampling from well-mixed turbulent zones, and sampling lines should be designed to minimize plateout or deposition. For safe and convenient sampling, the provisions should include:

a. Shielding to maintain radiation doses ALARA,
b. Sample containers with container-sampling port connector compatibility,
c. Capability of sampling under primary system pressure and negative pressures,
d. Handling and transport capability, and
e. Prearrangement for analysis and interpretation.

[6] Minimum of two monitors at widely separated locations.

[7] Detectors should respond to gamma radiation photons within any energy range from 60 keV to 3 MeV with an energy response accuracy of ±20 percent at any specific photon energy from 0.1 MeV to 3 MeV. Overall system accuracy should be within a factor of 2 over the entire range.

[8] Monitors should be capable of detecting and measuring radioactive gaseous effluent concentrations with compositions ranging from fresh equilibrium noble gas fission product mixtures to 10-day-old mixtures, with overall system accuracies within a factor of 2. Effluent concentrations may be expressed in terms of Xe-133 equivalents or in terms of any noble gas nuclide(s). It is not expected that a single monitoring device will have sufficient range to encompass the entire range provided in this regulatory guide and that multiple components or systems will be needed. Existing equipment may be used to monitor any portion of the stated range within the equipment design rating.

Table H-1 (continued)

Variable	Range	Category (see Regulatory Position 1.3)	Purpose
TYPE C (Continued)			
Containment			
RCS Pressure[1]	0 to 3000 psig (4000 psig for CE plants)	1[2]	Detection of potential for breach: accomplishment of mitigation
Containment Hydrogen Concentration	0 to 10% (capable of operating from 10 psia to maximum design pressure[4]) 0 to 30% for ice-condenser-type containment	1	Detection of potential for breach; accomplishment of mitigation. long-term surveillance
Containment Pressure[1]	10 psia pressure to 3 times design pressure[4] for concrete; 4 times design pressure for steel (5 psia for subatmospheric containments)	1	Detection of potential for or actual breach; accomplishment of mitigation
Containment Effluent Radioactivity - Noble Gases from Identified Release Points[1]	10^{-6} µCi/cc to 10^{-2} µCi/cc	2[8,9]	Detection of breach; accomplishment of mitigation; verification
Radiation Exposure Rate (inside buildings or areas, e.g., auxiliary building, reactor shield building annulus, fuel handling building, which are in direct contact with primary containment where penetrations and hatches are located)[1]	10^{-1} R/hr to 10^{4} R/hr	2[7]	Indication of breach
Effluent Radioactivity[1] - Noble Gases (from buildings as indicated above)	10^{-6} µCi/cc to 10^{3} µCi/cc	2[8]	Indication of breach

TYPE D Variables: those variables that provide information to indicate the operation of individual safety systems and other systems important to safety. These variables are to help the operator make appropriate decisions in using the individual systems important to safety in mitigating the consequences of an accident.

Variable	Range	Category	Purpose
Residual Heat Removal (RHR) or Decay Heat Removal System			
RHR System Flow	0 to 110% design flow[10]	2	To monitor operation
RHR Heat Exchanger Outlet Temperature	32°F to 350°F	2	To monitor operation and for analysis

[9] Provisions should be made to monitor all identified pathways for release of gaseous radioactive materials to the environs in conformance with General Design Criterion 64. Monitoring of individual effluent streams is only required where such streams are released directly into the environment. If two or more streams are combined prior to release from a common discharge point, monitoring of the combined stream is considered to meet the intent of this regulatory guide provided such monitoring has a range adequate to measure worst-case releases.

[10] Design flow is the maximum flow anticipated in normal operation.

Table H-1 (continued)

Variable	Range	Category (see Regulatory Position 1.3)	Purpose
TYPE D (Continued)			
Safety Injection Systems			
Accumulator Tank Level and Pressure	10% to 90% volume 0 to 750 psig	2	To monitor operation
Accumulator Isolation Valve Position	Closed or Open	2	Operation status
Boric Acid Charging Flow	0 to 110% design flow[10]	2	To monitor operation
Flow in HPI System	0 to 110% design flow[10]	2	To monitor operation
Flow in LPI System	0 to 110% design flow[10]	2	To monitor operation
Refueling Water Storage Tank Level	Top to bottom	2	To monitor operation
Primary Coolant System			
Reactor Coolant Pump Status	Motor current	3	To monitor operation
Primary System Safety Relief Valve Positions (including PORV and code valves) or Flow Through or Pressure in Relief Valve Lines	Closed-not closed	2	Operation status; to monitor for loss of coolant
Pressurizer Level	Bottom to top	1	To ensure proper operation of pressurizer
Pressurizer Heater Status	Electric current	2	To determine operating status
Quench Tank Level	Top to bottom	3	To monitor operation
Quench Tank Temperature	50°F to 750°F	3	To monitor operation
Quench Tank Pressure	0 to design pressure[4]	3	To monitor operation
Secondary System (Steam Generator)			
Steam Generator Level	From tube sheet to separators	1	To monitor operation
Steam Generator Pressure	From atmospheric pressure to 20% above the lowest safety valve setting	2	To monitor operation
Safety/Relief Valve Positions or Main Steam Flow	Closed-not closed	2	To monitor operation
Main Feedwater Flow	0 to 110% design flow[10]	3	To monitor operation

Table H-1 (continued)

Variable	Range	Category (see Regulatory Position 1.3)	Purpose
TYPE D (Continued)			
Auxiliary Feedwater or Emergency Feedwater System			
Auxiliary or Emergency Feedwater Flow	0 to 110% design flow[10]	2 (1 for B&W plants)	To monitor operation
Condensate Storage Tank Water Level	Plant specific	1	To ensure water supply for auxiliary feedwater (Can be Category 3 if not primary source of AFW. Then whatever is primary source of AFW should be listed and should be Category 1.)
Containment Cooling Systems			
Containment Spray Flow	0 to 110% design flow[10]	2	To monitor operation
Heat Removal by the Containment Fan Heat Removal System	Plant specific	2	To monitor operation
Containment Atmosphere Temperature	40°F to 400°F	2	To indicate accomplishment of cooling
Containment Sump Water Temperature	50°F to 250°F	2	To monitor operation
Chemical and Volume Control System			
Makeup Flow - In	0 to 110% design flow[10]	2	To monitor operation
Letdown Flow - Out	0 to 110% design flow[10]	2	To monitor operation
Volume Control Tank Level	Top to bottom	2	To monitor operation
Cooling Water System			
Component Cooling Water Temperature to ESF System	32°F to 200°F	2	To monitor operation
Component Cooling Water Flow to ESF System	0 to 110% design flow[10]	2	To monitor operation
Radwaste Systems			
High-Level Radioactive Liquid Tank Level	Top to bottom	3	To indicate storage volume
Radioactive Gas Holdup Tank Pressure	0 to 150% design pressure[4]	3	To indicate storage capacity

Table H-1 (continued)

Variable	Range	Category (see Regulatory Position 1.3)	Purpose
TYPE D (Continued)			
Ventilation Systems			
Emergency Ventilation Damper Position	Open-closed status	2	To indicate damper status
Power Supplies			
Status of Standby Power and Other Energy Sources Important to Safety (hydraulic, pneumatic)	Voltages, currents, pressures	2[11]	To indicate system status

TYPE E Variables: those variables to be monitored as required for use in determining the magnitude of the release of radioactive materials and continually assessing such releases.

Variable	Range	Category (see Regulatory Position 1.3)	Purpose
Containment Radiation			
Containment Area Radiation - High Range[1]	1 R/hr to 10^7 R/hr	1[6,7]	Detection of significant releases; release assessment; long-term surveillance; emergency plan actuation
Area Radiation			
Radiation Exposure Rate[1] (inside buildings or areas where access is required to service equipment important to safety)	10^{-1} R/hr to 10^4 R/hr	2[7]	Detection of significant releases, release assessment; long-term surveillance
Airborne Radioactive Materials Released from Plant			
Noble Gases and Vent Flow Rate			
• Containment or Purge Effluent[1]	10^{-6} µCi/cc to 10^5 µCi/cc 0 to 110% vent design flow[10] (Not needed if effluent discharges through common plant vent)	2[8]	Detection of significant releases, release assessment
• Reactor Shield Building Annulus[1] (if in design)	10^{-6} µCi/cc to 10^4 µCi/cc 0 to 110% vent design flow[10] (Not needed if effluent discharges through common plant vent)	2[8]	Detection of significant releases, release assessment
• Auxiliary Building[1] (including any building containing primary system gases, e.g., waste gas decay tank)	10^{-6} µCi/cc to 10^3 µCi/cc 0 to 110% vent design flow[10] (Not needed if effluent discharges through common plant vent)	2[8]	Detection of significant releases, release assessment, long-term surveillance

[11] Status indication of all Standby Power a.c. buses, d.c. buses, inverter output buses, and pneumatic supplies.

Table H-1 (continued)

Variable	Range	Category (see Regulatory Position 1.3)	Purpose
Type E (Continued)			
Airborne Radioactive Materials Released from Plant (Continued)			
Noble Gases and Vent Flow Rate (Continued)			
• Condenser Air Removal System Exhaust[1]	10^{-6} μCi/cc to 10^{5} μCi/cc 0 to 110% vent design flow[10] (Not needed if effluent discharges through common plant vent)	2[8]	Detection of significant releases; release assessment
• Common Plant Vent or Multipurpose Vent Discharging Any of Above Releases (if containment purge is included)	10^{-6} μCi/cc to 10^{3} μCi/cc 0 to 110% vent design flow[10] 10^{-6} μCi/cc to 10^{4} μCi/cc	2[8]	Detection of significant releases; release assessment; long-term surveillance
• Vent From Steam Generator Safety Relief Valves or Atmospheric Dump Valves	10^{-1} μCi/cc to 10^{3} μCi/cc (Duration of releases in seconds and mass of steam per unit time)	2[12]	Detection of significant releases; release assessment
• All Other Identified Release Points	10^{-6} μCi/cc to 10^{2} μCi/cc 0 to 110% vent design flow[10] (Not needed if effluent discharges through other monitored plant vents)	2[8]	Detection of significant releases; release assessment; long-term surveillance
Particulates and Halogens			
• All Identified Plant Release Points (except steam generator safety relief valves or atmospheric steam dump valves and condenser air removal system exhaust). Sampling with Onsite Analysis Capability	10^{-3} μCi/cc to 10^{2} μCi/cc 0 to 110% vent design flow[10]	3[13]	Detection of significant releases; release assessment; long-term surveillance

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[12]Effluent monitors for PWR steam safety valve discharges and atmospheric steam dump valve discharges should be capable of approximately linear response to gamma radiation photons with energies from approximately 0.5 MeV to 3 MeV. Overall system accuracy should be within a factor of 2. Calibration sources should fall within the range of approximately 0.5 MeV to 1.5 MeV (e.g., Cs-137, Mn-54, Na-22, and Co-60). Effluent concentrations should be expressed in terms of any gamma-emitting noble gas nuclide within the specified energy range. Calculational methods should be provided for estimating concurrent releases of low-energy noble gases that cannot be detected or measured by the methods or techniques employed for monitoring.

[13]To provide information regarding release of radioactive halogens and particulates. Continuous collection of representative samples followed by onsite laboratory measurements of samples for radiohalogens and particulates. The design envelope for shielding, handling, and analytical purposes should assume 30 minutes of integrated sampling time at sampler design flow, an average concentration of 10^{2} μCi/cc of radioiodines in gaseous or vapor form, an average concentration of 10^{1} μCi/cc of particulate radioiodines and particulates other than radioiodines, and an average gamma photon energy of 0.5 MeV per disintegration.

Table H-1 (continued)

Variable	Range	Category (see Regulatory Position 1.3)	Purpose
TYPE E (Continued)			
Environs Radiation and Radioactivity			
Radiation Exposure Meters (continuous indication at fixed locations)	Range, location, and qualification criteria to be developed to satisfy NUREG-0654, Section II.H.5b and 6b requirements for emergency radiological monitors		Verify significant releases and local magnitudes
Airborne Radiohalogens and Particulates (portable sampling with onsite analysis capability)	10^{-9} μCi/cc to 10^{-3} μCi/cc	3[14]	Release assessment; analysis
Plant and Environs Radiation (portable instrumentation)	10^{-3} R/hr to 10^{4} R/hr, photons 10^{-3} rads/hr to 10^{4} rads/hr, beta radiations and low-energy photons	3[15] 3[15]	Release assessment, analysis
Plant and Environs Radioactivity (portable instrumentation)	Multichannel gamma-ray spectrometer	3	Release assessment; analysis
Meteorology[16]			
Wind Direction	0 to 360° (±5° accuracy with a deflection of 15°). Starting speed 0.45 mps (1.0 mph). Damping ratio between 0.4 and 0.6, distance constant ≤ 2 meters	3	Release assessment
Wind Speed	0 to 30 mps (67 mph) ±0.22 mps (0.5 mph) accuracy for wind speeds less than 11 mps (25 mph) with a starting threshold of less than 0.45 mps (1.0 mph)	3	Release assessment
Estimation of Atmospheric Stability	Based on vertical temperature difference from primary system, -5°C to 10°C (-9°F to 18°F) and ±0.15°C accuracy per 50-meter intervals (±0.3°F accuracy per 164-foot intervals) or analogous range for alternative stability estimates	3	Release assessment

[14]For estimating release rates of radioactive materials released during an accident.

[15]To monitor radiation and airborne radioactivity concentrations in many areas throughout the facility and the site environs where it is impractical to install stationary monitors capable of covering both normal and accident levels.

[16]Guidance on meteorological measurements is being developed in a Proposed Revision 1 to Regulatory Guide 1.23, "Meteorological Programs in Support of Nuclear Power Plants."

Table H-1 (continued)

Variable	Range	Category (see Regulatory Position 1.3)	Purpose
TYPE E (Continued)			
Accident Sampling[17] Capability (Analysis Capability On Site)			
Primary Coolant and Sump	Grab Sample	3[5,18]	Release assessment; verification, analysis
• Gross Activity	10 μCi/ml to 10 Ci/ml		
• Gamma Spectrum	(Isotopic Analysis)		
• Boron Content	0 to 6000 ppm		
• Chloride Content	0 to 20 ppm		
• Dissolved Hydrogen or Total Gas[19]	0 to 2000 cc(STP)/kg		
• Dissolved Oxygen[19]	0 to 20 ppm		
• pH	1 to 13		
Containment Air	Grab Sample	3[5]	Release assessment; verification, analysis
• Hydrogen Content	0 to 10% 0 to 30% for ice condensers		
• Oxygen Content	0 to 30%		
• Gamma Spectrum	(Isotopic analysis)		

[17] The time for taking and analyzing samples should be 3 hours or less from the time the decision is made to sample, except for chloride which should be within 24 hours.

[18] An installed capability should be provided for obtaining containment sump, ECCS pump room sumps, and other similar auxiliary building sump liquid samples.

[19] Applies only to primary coolant, not to sump.

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