ABB
ASEA BROWN BOVERI

August 28, 1992
LD-92-091

Docket No. 52-002

Attn: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C 20555

Subject: System 30+™ Severe Accident Features

Reference: NRC Letter, Severe Accident Design Features, April 9, 1992

Dear Sirs:

This letter transmits our responses to the Requests for Additional Information (RAIs) in the reference. Attachment 1 provides a response to each RAI and Attachment 2 provides the report which is referenced in the responses.

If you have any questions, please call me or Mr. Stan Ritterbusch at (203) 285-5206.

Very truly yours,

COMBUSTION ENGINEERING, INC.

G. D. Hess for

C. B. Brinkman
Acting Director
Nuclear Systems Licensing

CBB/ser
cc: J. Trotter (EPRI)
T. Wambach (NRC)

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ATTACHMENT 1

NOTE: RESPONSES TO RAIs 410.156 THROUGH 410.160 WERE PROVIDED VIA LETTER LD-92-064, DATED 5/8/92.

Q. 410.140 Enclosure 2 provides an outline on Severe Accident closure issues which expands on the guidance provided in SECY-90-016. The staff will use this outline in the review of Advanced Light Water Reactor (ALWR) Severe Accident Issues closure. Since this document represents the staff's opinion as to what issues should be addressed for closure of the severe accident issues, show where each of the line items are discussed in the CESSAR-DC. If not currently available, provide a schedule for when the information will be provided to the staff.

Response:

Detailed discussion of the severe accident prevention and mitigation features of the System 80+ design is presented in References 1, 2, 3, and 4. The information contained in References 3 and 4 will be incorporated into CESSAR-DC Appendix B at the completion of the System 80+ updated PRA. The following table summarizes the status of the line items identified in Enclosure 2 of NRC letter dated 4/9/92.

STATUS OF ENCLOSURE 2 LINE ITEMS

I. BACKGROUND AND OVERVIEW

DEFENSE IN DEPTH PHILOSOPHY	- References 2 and 3 discuss how the defense in depth strategy is maintained in th evaluation of severe accidents.
BALANCE BETWEEN PREVENTION AND MITIGATION	- Reference 2, Section 6, contains descriptions of severe accident prevention and mitigation features. Reference 3, Section 3 contains descriptions of severe accident mitigative features and discusses how these features minimize containment system challenges during a severe accident.
CONTAINMENT PERFORMANCE GOALS	- The probabilistic approach for evaluating containment performance is used in Appendix B of CESSAR-DC (Reference 1) and the PRA report (Reference 2). Deterministic analyses of containment performance are contained in Reference 3, Section 5.

Response to 410.140 (Cont'd)

2. SEVERE ACCIDENT PHENOMENOLOGY - Reference 3, Section 4, contains information on severe accident phenomenology and its significance to the System 80+ design.

3. DISTINCTION BETWEEN SEVERE ACCIDENTS AND DBAs - Reference 1, Sections 6 and 15, contains the analyses of design basis accidents (DBAs) which consider single active failures and do not result in any significant core damage. Reference 3, Section 5, describes best estimate analyses of System 80+ containment performance for representative severe accident scenarios for which multiple component failures are postulated and core melt occurs.

4. CONTAINMENT PHILOSOPHY RELATIVE TO EARLY FAILURES

ACCIDENT SEQUENCES/CONTAINMENT CAPABILITY - Appendix B of CESSAR-DC (Reference 1) and the PRA Report (Reference 2) contain information pertinent to accident sequences and containment capability. Reference 3, Section 3, discusses System 80+ containment capability based on various failure considerations.

PRA CONSIDERATIONS - Appendix B of CESSAR-DC (Reference 1), the PRA Report (Reference 2), and Reference 4 contain this information. Reference 3, Section 4, discusses severe accident phenomenology and its application to the System 80+ PRA.

EXPERIENCE AND RESEARCH INSIGHTS - Reference 3, Section 4, contains this information.

FEATURES TO PREVENT AND MITIGATE SEVERE ACCIDENTS - Reference 4 discusses the prevention and mitigation features and Reference 3, Section 3 describes how the mitigative features minimize containment system challenges during a severe accident.

5. ACCIDENT MANAGEMENT - The basis for severe accident management is contained in References 2 and 3. Detailed accident management guidance will be developed based on NUMARC/NRC guidelines and the information contained in References 2 and 3. This will address the following major operator actions for mitigating severe accident consequences:

Response to 410.140 (Cont'd)

1) Actuation of hydrogen ignitors,
2) Flooding of cavity prior to reactor vessel failure, and
3) Safety Depressurization System Operation prior to core melt.

References:

1. System 80+ Standard Design, CESSAR Design Certification, 1990.

2. DCTR-RS-02 Rev. 0, "Probabilistic Risk Assessment for the System 80+ Standard Design", Combustion Engineering, Inc., January 1991.

3. ALWR-FS-DCTR-33, Rev.0, "System 80+ Severe Accident Phenomenology and Containment Performance", Combustion Engineering, Inc., August 1992.

4. "Application of Probabilistic Risk Assessment for the System 80+ Standard Design", Combustion Engineering, Inc., May 1992.

Q. 410.141 Your response to RAI 440.20 lists, in part, the hydrogen mitigation system igniters and cabling, as well as valves for the reactor cavity flooding system, as equipment that is relied upon to mitigate consequences of severe accidents. SECY-90-016 requires that there be high confidence that this equipment will survive severe accident conditions for the period that is needed to perform its intended function. However, SECY-90-016 has concluded that it is not necessary for redundant trains to be qualified to meet this goal.

With this general background, there are several areas where information is missing in your response to RAI 440.20. Therefore, please provide the following:

a. Provide the results of the calculations used to establish the environmental conditions for severe accident mitigative equipment. These conditions should include pressure, temperature, and radiation, as a function of time. In addition, provide the basis for concluding that the above conditions are bounding for the range of severe accidents.

b. In addition to the environmental conditions, provide any further criteria that will be imposed on the mitigative equipment. Indicate if these added criteria are to justify that there is reasonable assurance that this equipment will perform its function. Provide and justify the seismic design of this equipment.

c. Describe the electric power supplies for post accident mitigative equipment, including train and bus configurations supplying class 1E and alternate power sources. Describe the provisions for switching between the power sources, if required in the course of a severe accident.

Response to 410.141 a:

The equipment used in severe accident mitigation include:

(1) hydrogen mitigation system igniters and cabling,
(2) reactor cavity flooding system (CFS) valves, and
(3) safety depressurization system (SDS) valves.

The capability of igniters to function in harsh environment has been demonstrated via a number of NRC and EPRI sponsored test programs. For System 80+ application the igniters and associated cabling are expected to be available to perform their intended function if they survive the environment corresponding to the most limiting containment environment during a design basis LOCA or Main Steam Line Break. Since hydrogen combustion is not a significant threat to System 80+, the primary intent of the igniters is to minimize potential containment combustion loadings. The design basis accident (DBA) qualification range is sufficiently restrictive to encompass most severe accident scenarios. Because the low likelihood of exceeding DBA limits a more restrictive qualification criteria is considered unnecessary.

Response to 410.141 a (Cont'd)

The CFS valves are intended for operation prior to a reactor vessel breach. Therefore, these valves are not required to be qualified to extreme temperature, pressure, and radiation conditions representative of the later portions of a severe accident scenario. Thus, acceptable operation of the CFS valves is obtained during a severe accident scenario by qualifying them to design basis accident containment environmental conditions.

The SDS valves are expected to be employed for severe accident mitigation prior to, or immediately following, core uncovery. Therefore, no additional qualification testing (other than that is required for design basis accident containment environment) is considered necessary.

Response to 410.141 b:

Cavity Flooding System (CFS) and Safety Depressurization System (SDS) piping and components are designated in accordance with ASME Section III and ANSI/ANS 51.1. The CFS piping and components are Code Class 2 and Safety Class 2. The SDS piping and components that are part of the RCS pressure boundary are Code Class 1 and Safety Class 1. The remaining portions of the SDS are Code Class 2 and Safety Class 2. ASME Code and ANSI Safety Class designations for these piping and components are specified in CESSAR-DC Section 3.2 and Tables 6.7-2 and 6.8.2-1. As described in CESSAR-DC Section 3.2.1, all components in Safety Classes 1, 2, and 3 are Seismic Category I. Use of the specified classifications is intended to provide reasonable assurance that the CFS and SDS equipment will appropriately perform their functions.

Response to 410.141 c:

The major severe accident mitigative equipment that requires electric power supplies consist of (1) the hydrogen igniters, (2) the cavity flooding system, and (3) the safety depressurization system.

The hydrogen igniters are powered from the Class 1E 120V AC Vital Instrumentation and Control (I&C) Power system as described in CESSAR-DC Section 8.3.2.1.2.1 and Table 8.3.2-3. This system normally receives power from offsite power sources, with the Diesel Generators, Alternate AC Source (combustion turbine generator), or the emergency batteries supplying power if offsite power is unavailable. As described in CESSAR-DC Section 6.2.5.2.2, each igniter location consists of two igniters, one powered from each electrical division.

The cavity flooding system valves are powered from the Class 1E DC Vital Power System. Each of the four holdup volume flooding valves are powered from separate Class 1E channels and each of the two cavity flooding valves are powered from separate Class 1E divisions as seen from CESSAR-DC Table 8.3.2-4. The power to the Class 1E buses is normally supplied by either of two offsite power sources. Upon loss of both offsite power sources, the Class 1E Diesel Generators and the Class 1E batteries supply power to the buses. The diverse Alternate AC source combustion turbine generator can power these buses if power from all other sources is lost.

Response to 410.141 c (Cont'd)

The source of power for the rapid depressurization valves of the safety depressurization system is the Class 1E DC Vital Power System. The power to the Class 1E buses is normally supplied by either of two offsite power sources. Upon loss of both offsite power sources, the class 1E diesel generators and the Class 1E batteries supply power to these buses. The diverse Alternate AC Source (combustion turbine generator) can power these buses if power from all other sources is lost.

Q. 410.142 Describe any systems or methods such as on-line monitoring that will be utilized to ensure that the containment leakage rate is maintained below the value assumed.

Response:

In accordance with 10 CFR 50 , Appendix J, the required Type A containment integrated leak rate test (CILRT) is performed at approximately equal intervals during each 10-year service period. The CILRT methodology is described in CESSAR-DC Section 6.2.6.1. This test is the regulatory and industry established method for ensuring that the containment leakage is maintained below the value assumed.

In addition to Type A containment integrated leak rate testing, 10 CFR 50, Appendix J, Type B and C individual component tests are performed at least once every two years. The resultant leakage rates of these individual component tests are summed to obtain an equivalent overall containment leakage rate, which must be less than $0.6 \times L_a$ (i.e., 0.6 times the assumed Design Basis Accident leakage rate). Type B and C leak rate testing methodologies are described in CESSAR-DC Sections 6.2.6.2 and 6.2.6.3.

The EPRI Utility Requirements Document (Volume II, Chapter 5 Engineered Safety Systems) Section 6.3.2.5 specifies that "Means shall be provided to enable the operator to perform a periodic check for gross leakage of containment atmosphere during normal operation". System details for such means (required hardware, software, interfaces with Appendix J required CILRT, test pressure parameters, procedures, etc.) are not known at this point of Design Certification. These will be specified at a later stage of detailed design.

Q. 410.143 RAIs 722.13 and 730.7 (b) requested a description of the location of the hydrogen igniters. In addition to this information, provide the separation distance between igniters and a general discussion of where the igniters will be located. For example how were the various areas considered in the placement of igniters; under overhangs, in all compartments, on the ceiling, and at the source of possible hydrogen? If there is a particular separation distance between igniters, please provide the associated analytical input parameters that were used in conjunction with this value.

Response:

The updated CESSAR-DC Section 6.2.5 submitted via letter LD-92-050, dated April 15, 1992 contains the hydrogen igniter locations which can be used to determine the approximate separation distance between them. These igniters are designed and placed within the containment to burn hydrogen at low concentrations where hydrogen could accumulate (e.g., closed compartments and dead ended regions in containment).

There is no specific separation distance between igniters within the containment. However, each igniter location includes two igniters (one powered from each electrical division) to account for failure of a division's power sources.

Q. 410.144 How many igniter assemblies will be allowed in an igniter circuit, and how many are allowed to be inoperable before the Hydrogen Mitigation System is declared inoperable? Also, would inoperable igniter assemblies be allowed to be adjacent to one another and if complete loss of igniters in a compartment will be allowed? Provide the justification for this type of multiple failure criteria.

Response:

As indicated in DESSAR-DC Section 6.2.5.2.2, one (1) to (10) igniter assemblies will be contained in an igniter circuit.

Each igniter location contains two igniter assemblies, each one powered from a separate electrical division. Therefore, upon loss of an entire electrical division, the Hydrogen Mitigation System is fully operable. Thus, loss of a complete igniter circuit or an entire division will not result in the complete loss of igniters in a compartment.

Q. 410.145 Please identify all CE 80+ design features which prevent core melt or provide a recovery capability.

a. Describe how in the design process these features were selected.

b. provide some quantification of each features risk benefit worth.

c. Identify which of these features came from existing designs, and which were new or possess new capabilities.

d. Describe the process used to decide which severe accident enhancements should be incorporated into the CE 80+ and which to exclude (if any).

Response:

The System 80+ design features which prevent core melt or provide a recovery capability are discussed in Reference 1. In addition, severe accident mitigation features are described in Section 3 of Reference 2 and discussed in the response to RAI 410.164 (attached).

Response to 410.145 a:

Reference 1 discusses how in the design process the above design features were selected.

Response to 410.145 b:

Reference 1 summarizes the each features' risk benefit worth. Reference 3 contains the complete design alternatives analysis which has been submitted to NRC as a separate report.

Response to 410.145 c:

All design enhancements are evolutions of existing design features. As discussed in Reference 1 many of the major design enhancements for preventing core melt or providing a recovery capability are based on insights gained from the System 80 "baseline" PRA as well as previous risk assessments for both CE and non-CE plants. These enhancements were incorporated into the ALWR Utility Requirements Document (Reference 4).

Response to 410.145 d:

The process used to decide which severe accident enhancements should be incorporated into the System 80+ design is described in References 1 and 2.

References for Response to 410.145:

1. Letter LD-92-064 (Attach. 13), "Application of Probabilistic Risk Assessment for the System 80+ Standard Design", Combustion Engineering, Inc., May 1992.

2. ALWR-FS-DCTR-33, Rev. 0, "System 80+ Severe Accident Phenomenology and Containment Performance", Combustion Engineering, Inc., August 1992.

3. "Design Alternatives for the System 80+ Nuclear Power Plant", Rev. 0, Combustion Engineering, Inc., April 1992.

4. NP-6780-L, "Advanced Light Water Reactor (ALWR) Utility Requirements Document", Rev. 3, Electric Power Research Institute, 1992.

Q. 410.146 In addition to the reactor cavity drawings requested via RAI 722.1, provide the following:

a. Provide the following design details:
   . location and size of any ledge-like surfaces,
   . location and configuration of all penetrations,
   . location and configuration of all openings to the drywell compartment,
   . size, elevation, and configuration of floor vents; and

b. Identify and provide the results of any experimental tests that support the design of the cavity. Show to what degree the results demonstrate the design objective of the cavity to retain corium debris.

Response to 410.146 a:

The locations and sizes of ledge-like surfaces in the reactor cavity area are as follows:

	Item	Location	Size
1.	Excore Detectors (4 total)	Elevation 102+0-3/4	33.96 ft^2 (Total area)
2.	Reactor Vessel Support Corbels (2 total)	Elevation 84+3-1/4	501.58 ft^2 (Total area)
3.	Shield Plugs (6 total)	Elevation 106+0-3/4	127.64 ft^2 (Total area neglecting cutouts for nozzles)
4.	Lid of Core Debris Chamber	Elevation 75+6	213.26 ft^2

The location, configuration, and sizes of all penetrations, openings, and vents in the reactor cavity are as follows:

	Item	Location	Size
1.	Reactor Vessel Hot Leg Piping (2 total)	Elevation 107+3-3/4	42 in diameter pipe size
2.	Reactor Vessel Cold Leg Piping (4 total)	Elevation 107+3-3/4	30 in. diameter pipe size
3.	Direct Vessel Injection Piping (4 total)	Elevation 114+2-1/2	10 in diameter pipe size

Response to 410.146 a (Cont'd)

	Item	Location	Size
4.	In-Core Instumentation Lines (61 total)	Elevation 120+0	1 in diameter pipe size
5.	Reactor Cavity Cooling Ducts (2 total)	Elevation 97+10 & 102+10	36 in diameter duct size
6.	Door into Maintenance, Access, Ventilation, and Equipment Chase (MAVEC)	Elevation 91+9	3 ft x 7 ft
7.	Pressure Relief Dampers (2 total)	Elevation 92+9 (bottom)	6 ft x 8 ft
8.	Pool Purification and Cleaning System Piping (6 total)	Elevation 109+6 Elevation 105+6 Elevation 106+6 Elevation 107+6 Elevation 108+6 Elevation 109+6	8 in diameter 1 in diameter 1 in diameter 1 in diameter 1 in diameter 1 in diameter (pipe sizes)
9.	Excore Detectors	Information not yet available	Information not yet available
10.	Reactor vent Piping (2 total)	Not yet routed	3/4 in diameter pipe size

Response to 410.146 b:

The System 80+ design employs a cavity design concept that is consistent with the ALWR Utlity Requirements Document (Reference 1). Specifically, the reactor cavity is designed such that the corium debris should be retained in the reactor cavity, preventing any credible direct containment heating (DCH) threat to the containment and thereby minimizing the potential for an early containment failure.

The System 80+ cavity design is based on the cumulative experience from several DCH tests performed by Fauske and Associates (FAI), Sandia National Loaboratories, and other national laboratories. These tests established correlations for debris entrainment/impingement, quantified the importance of intervening structures on the magnitude of DCH induced pressure loads and identified the ability of recessed regions to retain the core debris. The Sandia debris impingement model (Reference 2) established from a review of

Response to 410.146 b (Cont'd)

HIPS experiments were specifically used to bound the level of corium retention within the System 80+ cavity. Additional details on the use of experimental data in the design of the System 80+ cavity can be found in Appendix D of Reference 3 and Section 4.1 of Reference 4.

References:

1. NP-6780-L, "Advanced Light Water Reactor (ALWR) Utility Requirements Document, Rev. 3, Electric Power Research Institute, 1992.

2. NUREG/CR-5039, "Reactor Safety Research Semi-Annual Report", J. V. Walker, July-December, 1987.

3. DOE/ID-10271, "Prevention of Early Containment Failure due to High Pressure Melt Ejection and Direct Containment Heating for the Advanced Light Water Reactor", J. Carter, et. al., March 1990.

4. ALWR-FS-DCTR-33, Rev. 0, "System 80+ Severe Accident Phenomenology and Containment Performance", Combustion Engineering, Inc., August 1992.

Q. 410.147 Describe the methodology used to determine ex-vessel corium debris coolability.

a. Discuss the basis for the methodology used.

b. Include initial conditions, assumptions, results, and conclusions.

c. Quantify and describe the basis for the mass composition and temperature assumed of the debris in the lower head at the time of lower head failure.

d. Please provide the analysis used to determine the amount of debris ejected from the reactor vessel.

e. Please provide the depth of erosion into both the basemat and the reactor vessel pedestals for at least the first 24 hours or until the debris was quenched, whichever came first.

f. What is the maximum penetration that can be tolerated into the pedestals, such that their structural integrity is maintained.

g. Please provide the basis (i.e., calculations, assumptions, and test data) for the penetration rate used in the analysis.

h. What total thickness was assumed for the basemat?

i. Please provide the supporting containment pressure temperature response profile.

j. Please provide a plot of the integrated and instantaneous production rate of non-condensible gases as a function of time.

k. How does the core debris cooling rate affect containment integrity, and what is the maximum time that the containment can withstand with no core debris cooling before integrity is breached?

Response to 410.147 a:

The methodology used for the assessment of ex-vessel coolability in System 80+ varies based on the context of the analysis. Analyses demonstrating deterministic predictions of corium coolability and corium-concrete attack are based on the DECOMP core-concrete interaction module of MAAP (Reference 1). In past PRA applications, the ALWR support work provided in Reference 2 was used to support the conclusion that meeting the URD floor area/thermal power guidance of 0.02 m^2/MWt was sufficient to provide reasonable assurance of a coolable debris

Response to 410.147 a (Cont'd)

bed provided a continuous source of water was available to the cavity. In the PRA, DECOMP is used parametrically along with concrete attack rates estimated from NUREG-1150 CORCON basemat analyses (See Reference 3) and experimental observations to establish debris coolability assessments and their probabilistic consequences.

Response to 410.147 b:

Analyses supporting future PRA activities will investigate the consequences of reduced coolability on concrete erosion using DECOMP. This study will investigate the impact of reducing the corium debris bed heat flux from about 1 MWt/m^2 down to 0.4 MWt/m^2.

Since the System 80+ cavity is designed to be debris retentive, core concrete attack analyses assume the full core inventory of UO_2 and Zircaloy in addition to 100,000 lbm of stainless steel are available to interact with the concrete.

Initial conditions, assumptions, and results of these analyses will be included in the final PRA submittal scheduled for early 1993.

Response to 410.147 c:

As discussed in item B, the corium concrete attack is based on total RCS inventory of UO_2 and Zr. Thus, the precise inventory at the time of vessel failure is not critical to the issue of corium concrete attack since, immediately after vessel failure, all the corium is assumed to reside in the cavity.

Response to 410.147 d:

All the corium is conservatively assumed to be ejected from the reactor vessel and is available for corium concrete attack.

Response to 410.147 e:

Based on information contained in Reference 3, it is conservatively estimated that a partially cooled corium debris bed will erode concrete at a constant rate of 0.035 cm/min in the axial direction. Reviews of Beta test data suggests that radial concrete erosion rates will be only 20 to 50 % of that value. Thus, after 24 hours the radial erosion profile in a partially cooled melt will progress to between 0.33 and 0.83 feet. This level of erosion is not sufficient to cause collapse of the reactor cavity walls (see response to 410.147 f).

Response to 410.147 f:

A very conservative estimate of the maximum radial erosion (penetration) that can be tolerated into the cavity walls, and still maintain their structural integrity is 1.25 ft.

Response to 410.147 g:

See discussion in Response to 410.147 e.

Response to 410.147 h:

The total thickness of the System 80+ basemat is 22 feet.

Response to 410.147 i:

Containment temperatire and pressure responses for a station blackout scenario for the System 80+ design are presented in Figures 5.3.1.4 and 5.3.1.5 of Reference 4. These results are obtained from a MAAP assessment of degraded core coolability with successful cooling of the corium by an overlying liquid pool. Consistent with the above discussions, it is assumed that 100 % of the corium resides in a water filled reactor cavity. Additional details of this scenario can be found in Section 5.3 of Reference 4.

Response to 410.147 j:

In the scenario discussed in response to 410.147 i, the corium is rapidly cooled without release of any non-condensable gases. For the analysis discussed in response to 410.147 e, concrete erosion at a rate of 0.035 cm/min will outgas carbon dioxide from the concrete at a rate of approximately 28.5 lbm/min for limestone/common sand concrete and 2.03 lbm/min for basaltic concrete.

MAAP analyses performed to parametrically study degraded debris heat removal indicate that the maximum concrete erosion that can be expected in a "wet" cavity is 0.75 foot (See response to Q. 410.164). Erosion of limestone/common sand concrete to a 0.75 foot depth would release approximately 19,000 lbm of carbon dioxide to the containment atmosphere. Release of carbon dioxide due to erosion of a basaltic basemat is small (approximately 1300 lbm).

Response to 410.147 k:

Corium cooling in the absence of containment heat removal will result in eventual containment failure. Results of the analysis presented in response to 410.147 i indicate that for a station blackout scenario with battery power available for 4 hours, containment pressures will reach ASME Level "C" values by 60 hours and containment ultimate strength levels by 80 hours. Longer time estimates would result if the full 8 hour battery availability was considered in the analysis. Additional information concerning this scenario may be found in Section 5.3.1 of Reference 4.

During dry cavity sequences, containment integrity can be compromised via the following:

1. basemat melt-through into the containment subsoil
2. basemat melt-through into the containment subsphere

Response to 410.147 k (Cont'd)

3. corium induced failure of cavity walls
4. temperature induced failure of the penetration sealant
5. overpressure failure of containment

The above failure mechanisms are discussed in Reference 4, Section 4.2.

REFERENCES:

1. "MAAP 3.0B Users Manual", Volume 2, Part 2, Fauske and Associates (FAI), 1990.

2. DCTR-RS-02 Rev. 0, "Probabilistic Risk Assessment for the System 80+ Standard Design", ABB Combustion Engineering, Inc., January, 1991.

3. NUREG/CR-5567,"PWR Dry Containment Issue Characterization", J. W. Yang, July, 1990.

4. ALWR-FS-DCTR-33, Rev. 0, "System 80+ Severe Accident Phenomenology and Containment Performance, Combustion Engineering, Inc., August 1992.

Q. 410.148 Appendix B, Section 9.2.2.7 entitled, "Top Event 5: Late Containment Failure," includes a scenario in which containment spray is unavailable for 24 hours. MAAP code analyses have apparently shown that for these sequences where the cavity is initially flooded with no containment heat removal capability, it takes longer than 48 hours to overpressurize the CE 80+ containment. Please provide the supporting analysis or basis for this 48 hours, including all initial conditions and assumptions used.

Response:

MAAP code analyses have shown that for sequences where the cavity is initially flooded with no containment heat removal capability, it takes longer than 48 hours from the initiation of the transient to overpressurize the System 80+ containment. The dominant sequence for this event is a loss of offsite power transient where the RCPs have tripped and both the diesel generators and the gas turbine generator are unavailable. One turbine-driven auxiliary feedwater pump is available to deliver flow to the steam generators for an eight hour time frame. At the end of eight hours, the pump becomes unavailable and the operators actuate the cavity flooding system. At approximately 15.7 hours into the transient, the vessel fails to a flooded cavity. Following vessel failure, the MAAP run continues for an additional 48 hours (63.7 total hours into the transient) at which time a single train of containment sprays is made available to reduce the containment building pressure. At 63.7 hours into the accident the containment pressure is still below ultimate failure pressure and the sprays successfully reduce the pressure to acceptable safe levels. In this transient, leakage of fission products from containment is limited to releases resulting from normal containment leakage of 0.5% by volume per day.

Q. 410.149 Please provide the consequences of a high pressure core melt ejection accident (via the MAAP code) assuming no ingression of coolant into the mass of debris--i.e, assuming the corium is not in a coolable geometry. Please include the following:

a. a discussion of the phenomenon of ingression into molten core debris as it is cooled,

b. the basis for the assumption used in the CESSAR-DC with regard to ingression into the debris,

c. the expected location of maximum heat generation in the debris bed (Is it at the base and center of the debris bed?),

d. a description of the corium-concrete interaction and its consequences at this location,

e. the earliest projected containment failure under this scenario, and

f. a profile of the thermal effects on the cavity floor for up to 48 hours following vessel failure under this scenario.

Response:

A high pressure core melt accident in the System 80+ design results in the retention of more than 90% of the corium debris within the cavity. An evaluation of the maximum corium concrete interaction was performed using MAAP 3.0B for the condition where the cavity was dry. This plant damage state was estimated to have a frequency of 2.1×10^{-8} per year. Under these circumstances the earliest projected containment failure due to erosion of the basemat is associated with a potential ingression of the melt into the SI pump room of the containment subsphere. This containment failure can potentially occur in beyond the 100 hour time frame. Erosion of the basemat into the containment subsoil is estimated to occur in the 250 to 300 hour time frame.

The potential for these large erosion rates to be obtained in the presence of an overlying water pool is not considered credible. Experiments consistently indicate that in instances where water was present during corium-concrete attack, the concrete erosion rate was reduced and energy was removed from the melt.

CESSAR-DC does not explicitly model water ingression into the core debris. Based on recent experiments, it is believed that water ingression into the debris bed will occur, and the corium debris bed will be quenched, at least in the long term. This position is supported by results of MACE Test 1B which demonstrated the ability of an overlying water pool to significantly penetrate a corium debris bed.

Q. 410.150 In Section 9.2.2 entitled, "Quantification of the Containment Event Tree Top Events," it is repeatedly stated that if the vessel fails at high pressure, the corium will be widely distributed through containment.

a. Please provide the basis for the assumption that corium debris would be widely distributed (please include test references, if applicable).

b. Could the corium be blown into one location in one mass?

c. Which assumption--the concentrated heat generation of one softened mass or the wide dispersal of fine fragments--would be more conservative?

Response to 410.150 a:

The small amount of corium (< 10 % of the total core inventory, see Reference 1, Section 4) that is potentially capable of exiting the reactor cavity will be distributed in the refueling pool and in the lower compartment above the IRWST pool access walkway. Design changes to the System 80+ cavity indicate that corium leaving via the IRWST walkway will either be deposited within the ventilation room or outside the room in the vicinity of the ventilation duct louvers and HVAC room door. All corium will be retained within the crane wall.

Response to 410.150 b:

While corium that is ejected from the cavity is not likely to be blown in one location, it may be accumulated in small localized quantities in several areas. The impact of this projected dispersion of corium on containment performance is considered negligible.

Response to 410.150 c:

In either case the amount of corium ejected outside cavity is expected to be small. Since the predominant containment threat is due to non-condensed steam overpressurization, any corium ejected outside the cavity where it would not vaporize water will have a net beneficial effect on containment integrity.

Reference:

1. ALWR-FS-DCTR-33, Rev.0, "System 80+ Severe Accident Phenomenology and Containment Performance", Combustion Engineering, Inc., August 1992.

Q. 410.151 A series of ACE/MACE tests are underway at Argonne National Laboratory to demonstrate core debris coolability. Several of these tests have been completed. Discuss the applicability of these tests to the CE 80+ design. Include a discussion on the applicability of the test parameters, assumptions, and results.

Response:

The issue of core debris coolability in the context of the ACE/MACE tests is discussed in Section 4.2.2 of Reference 1. It indicates that while existing test data do not support the immediate coolability of a corium debris, evidence is sufficient to support the long term coolability of a corium debris bed for times greater than twelve (12) hours into a core melt scenario. In addition, test data indicate that debris cooling via overlying water pools will significantly reduce the concrete erosion rate from that determined for dry cavity erosion sequences.

Reference:

1. ALWR-FS-DCTR-33, Rev. 0, "System 80+ Severe Accident Phenomenology and Containment Performance, Combustion Engineering, Inc., August 1992.

Q. 410.152 Section 6.3.15 entitled, "Cavity Flooding System," says the flooder valves in the system will undergo a surveillance every refueling outage.

a. Please explain the recommended surveillance for these valves.

b. Is it recommended that these flooder valves be tested (stroked) periodically?

c. Are these valves expected to have a reliability value higher than normal isolation valves? If so, what is the value?

Response to 410.152 a and b:

The Cavity Flooding System (CFS) flooder valves are designated as Code Class 2 in accordance with ASME Boiler and Pressure Vessel Code Section III and Safety Class 2 in accordance with ANSI/ANS 51.1 (see CESSAR-DC Section 3.2 and Table 6.8.2-1). The surveillance requirements for valve testing, including stroke testing, are specified in accordance with the ASME Boiler and Pressure Vessel Code Section XI (see CESSAR-DC Sections 3.9.2 and 3.9.6.2).

Response to 410.152 c:

The reliability of the CFS flooder valves is expected to be as good as the reliability of other Safety Class 2 valves.

Q. 410.153 Is there a consistent thickness throughout the containment of at least 3 feet of concrete to protect the steel containment liner?

Response:

A concrete thickness of 3 feet is maintained in the floor region of the reactor cavity. The thickness of concrete between the floor of the debris chamber (which is offset from the reactor cavity) and the containment steel shell is 2'-0".

It should be noted that penetration of the 2'-0" thickness of the reactor cavity floor transition region does not constitute of breach of the containment. In addition to eroding this 2'-0" thickness of concrete, the molten core debris would also have to attack and penetrate the 1-3/4" containment steel plate and approximately 20 feet of basemat concrete before making contact with the environment. For a release via the SI pump room about 10 feet of radial erosion of concrete would be required in addition to the penetration of the 2'-0" thickness of the reactor cavity floor transition region and the 1-3/4" containment steel shell.

Q. 410.154 Please explain how the containment system can accommodate the following challenges resulting from the thermal decomposition of concrete by molten corium:

a. the degradation of containment cooling and of cleanup capability due to aerosol formation,

b. slow overpressurization resulting from the evolution of noncondensible gases,

c. functional degradation of structural concrete by erosion, including basemat penetration, and

d. combustion of carbon monoxide.

Response:

The challenges identified in this question will be insignificant as long as the cavity flood system operates and the cavity is flooded. For this condition concrete erosion will be slow and limited thus ancillary processes associated with core concrete interaction such as, aerosol formation, noncondensable gas formation, basemat penetration, and carbon monoxide production will not pose a threat to containment integrity (see Reference 1, Section 4.2).

Response to 410.154 a:

Generation of significant quantities of aerosols suggests that cavity flooding was not accomplished. The most likely cause of this situation would be a station blackout scenario. This initiating event would also disable the containment spray system. Thus, the adverse effects of aerosolization becomes a moot point.

In the rare instance that sprays are operational and the Cavity Flood System (CFS) has not been actuated to control the corium concrete attack, large aerosol concentrations may develop in containment. The initial availability of sprays is expected to control the aerosol threat by promoting settling of the aerosol. A continuous supply of water to the containment spray nozzles should prevent any potential aerosol clogging.

If the containment sprays are not on, and the cavity does not flood, significant levels of corium concrete interaction could potentially degrade the inoperable containment spray, reducing the system's potential for recovery. However, since the CFS can be actuated via station batteries, the inability to actuate either the CFS or the containment sprays is remote.

Response to 410.154 b:

The dry cavity basemat scenario following a station blackout event is analyzed in Section 4.2.2 of Reference 1. Based on this study it appears that generation of noncondensable gases during the basemat attack will not be sufficient to overpressurize the containment.

Response to 410.154 c:

The anticipated outcome of an unmitigated corium concrete attack of the basemat is the penetration of the corium into the containment subsoil. This breach is expected to be followed by a vitrification of the corium in the foundation soil and ultimate arrest of the corium progression. Radiological consequences of this scenario are considered to be insignificant.

Alternate basemat failure modes include the potential for corium to radially spread within the basemat and penetrate into the auxiliary building via the SI pump room. In the initial System 80+ PRA (Reference 2), this scenario was conservatively modeled as an unfiltered above ground radiation release. Furthermore, this scenario was also selected to represent the "dry cavity" basemat release class.

Response to 410.154 d:

While the presence of carbon monoxide complicates the combustion process, there is no fundamental difference in igniting a mixture of hydrogen/air /steam with that of hydrogen/carbon monoxide/air and steam. The containment response to the ignition of these sources depends on several factors including the existence of prior burns, the potential for auto-ignition of carbon monoxide/ hydrogen as they traverse the melt, and the use of igniters. In the PRA, the presence of carbon monoxide is conservatively treated as a concentration of hydrogen.

REFERENCES:

1. ALWR-FS-DCTR-33, Rev.0, "System 80+ Severe Accident Phenomenology and Containment Performance", Combustion Engineering, Inc., August 1992.

2. DCTR-RS-02, Rev. 0, "Probabilistic Risk Assessment for the System 80+ Standard Design", Combustion Engineering, Inc., January 1991.

Q. 410.155 Describe how the above challenges could affect equipment required for containment cooling and atmospheric cleanup, if they could result in leakage that exceeds the rate specified in General Design Criteria 16, and whether they could result in release through the basemat following the onset of the corium-concrete interaction.

Response:

The challenges identified in Q. 410.154 will be insignificant as long as the cavity flood system operates and the cavity is flooded. For this condition, concrete erosion will be slow and limited. Therefore, ancillary processes such as, aerosol formation, noncondensable gas formation, basemat penetration, and carbon monoxide production will not pose a threat to containment integrity.

If the Cavity Flooding System is not operational, and a vigorous corium concrete attack occurs, a basemat melt-through scenario can be postulated. Unmitigated corium concrete attack would result in the penetration of the corium into the containment subsoil. This breach is expected to be followed by a vitrification of the corium in the foundation soil and ultimate arrest of the corium progression. Radiological consequences of this scenario are considered to be insignificant.

An alternate failure mode is the potential for for corium to radially spread within the basemat and penetrate into the auxilary building via the SI pump room. In the System 80+ PRA this failure mode was conservatively modeled as an unfiltered above ground radiation release.

Q. 410.161 In SECY paper 90-016, in the "Containment Performance" section, the staff position indicates that a containment design may utilize controlled elevated venting, diverse containment heat removal systems, or may rely on the restoration of normal heat removal systems if sufficient time is available for major recovery actions...for example, 48 hours. CE appears to take credit for the SECY paper "example" of 48 hours, even though this time period is not applicable to the CE 80+ design. For instance, in Section 4.8.2.1.8 of Appendix B, containment failure is projected in approximately 41 hours. Please clarify this inconsistency.

Response:

The loss of offsite power sequence described in Section 4.8.2.1.8 of the System 80+ PRA report (Reference 1), which results in the failure of the containment at about 41 hours, represents a very conservative scenario for which a total loss of main feedwater is postulated with no auxiliary feedwater delivery. The core heat removal is accomplished by successful operation of the feed (Safety Injection) and bleed (Safety Depressurization System operation). No containment spray or containment heat removal is assumed available with IRWST cooling unavailable. This sequence has a core damage frequency less than 1.0E-11 events/reactor-year. This is not considered a credible core damage sequence to be included in the severe accident sequences for which the SECY-90-016 recommended 48 hours is applicable.

SECY-90-016 also allows for the use of alternate containment heat removal systems after maintenance of containment integrity (defined as containment stresses not exceeding ASME Service Level "C" limits) for a minimum period (e.g., 24 hours). The System 80+ containment design will include a connection to the containment spray system which would enable it to be supplied from an external water supply system. Based on the SECY paper, utilization of this diverse containment heat removal system can take place immediately after the miniumum time period of 24 hours (as opposed to restoration of normal containment heat removal systems which would be termed a "major recovery action" for which "sufficient time" [e.g., 48 hours] would be required.). With the use of this alternate containment heat removal capability, the containment pressure will remain well below the containment failure point for time periods greater than 48 hours.

Reference:

1. DCTR-RS-02, Rev. 0, "Probabilistic Risk Assessment for the System 80+ Standard Design", Combustion Engineering, Inc., January 1991.

Q. 410.162 Did CE consider providing containment (filtered) vents for containment overpressure protection?

Response:

ABB-CE considered adding a containment vent for severe accident mitigation purposes, but did not incorporate such a feature for three reasons. First, the System 80+ design has a large dry containment (3.337E+6 cubic feet free volume) and early in the design process it was judged that the consequences of severe accidents could be mitigated without a vent. Second, it was judged that public acceptance of the design would be more likely without a containment vent and, third a feasibility study (letter LD-92-056, dated April 24, 1992) showed that there would not be a significant benefit from a vent (over 90% of the risk comes from the intact-containment release class).

Q. 410.163 In accordance with SECY-90-016, the design pressure used for severe accident analysis may be calculated one of two ways--either applying a conditional containment failure probability (CCFP) guideline of 0.1, or using a deterministic method (based on the ASME schedule) offering comparable protection. Therefore, please provide the following:

a. the pressure used for CE 80+ severe accident analysis,

b. the method used to arrive at that pressure (i.e., Service Level C),

c. the rationale for using the above method, and

d. a description of the use of uncertainties in the analysis.

Response to 410.163 a:

The System 80+ Probabilistic Risk Assessment documented in Appendix B of CESSAR-DC shows that the conditional probability of containment failure given a core melt condition was 0.099. This result was obtained using containment fragility (probabilistic failure) data generated for the PRA. Fragility data based on more detailed analyses has been recently generated. This data is presented in Figure 3.1-3 of Reference 1 and would be used in the revised System 80+ PRA.

In addition to the probabilistic approach, the containment performance was calculated on a deterministic basis using ASME Boiler & Pressure Vessel Code, Service Level "C" stress criterion for determining the containment pressure limit. The results of this analysis as reported in response to NRC RAI 220.55 shows that for a station blackout scenario, the containment pressure remains well below the ASME Service Level C stress criterion based pressure limit of 156 psia (at a temperature of 350 degree F) for more than 50 hours.

Response to 410.163 b:

The method used to determine the containment pressure limits is described in Section 3.1.2 of Reference 1.

Response to 410.163 c:

The rationale for the use of the above method is provided in Section 3.1.2 of Reference 1.

Response to 410.163 d:

Uncertainties have not been accounted for in the analysis of the containment pressure limits documented in Reference 1. As part of the revised System 80+ revised PRA effort, the uncertainties in material properties and modeling will be included in the analysis.

Response to 410.163 (Cont'd)

Reference:

1. ALWP-FS-DCTR-33, Rev. 0, "System 80+ Severe Accident Phenomenology and Containment Performance", Combustion Engineering, Inc., August 199[illegible]

Q. 410.164 Please provide the analyses that support those design features necessary to mitigate severe accidents. Include initial conditions, assumptions, results, and conclusions. Also identify which design objectives are supported solely by analysis (i.e., having little or no historical or experimental basis).

Response:

The design features included in the System 80+ design to mitigate severe accidents include:

1. Large Containment Volume

2. Rapid Depressurization Valves of the Safety Depressurization System

3. Hydrogen Igniters of the Hydrogen Mitigation System

4. The Reactor Cavity Flood System

5. Reactor Cavity Debris Retentive Design

The experimental and analytical bases of these mitigating features are discussed below.

1. Large Containment Volume

The desirability of a large, strong containment was evident from a review of much of the severe accident literature over the past decade. The large size of the System 80+ containment was desired to reduce global hydrogen concentrations. Large containment volumes are also desirable in mitigating DCH effects. These features were quantitatively evaluated under the DOE Advanced Reactor Severe Accident Program (ARSAP) for a typical evolutionary PWR of similar size to System 80+ in References 1 and 2. Additional discussion of this topic can be found in Reference 3.

2. Safety Depressurization System

Following an unrecovered station blackout (or other high pressure core meltdown scenario), the role of the safety depressurization system in severe accident is to depressurize the RCS from high system pressures (near 2500 psia) to 250 psia such that the low RCS pressures are achieved prior to a reactor vessel lower head failure. The intent of this feature is to ensure that, should a core melt scenario develop, a high pressure melt ejection situation can be averted and the threat of DCH induced containment failure could be eliminated.

The ability of the SDS to perform this function was established via MAAP parametric analyses for an unrecovered total loss of feedwater event. The analyses confirmed that in order to accomplish the above objective, the operator has more than 2 hours following steam generator dryout and consequent PSV "lift" to actuate the SDS system.

Response to 410.164 (Cont'd)

3. Hydrogen Igniters

The use of igniters to provide a means of controlled combustion of hydrogen is well established and is consistent with the requirements of the EPRI URD and draft NRC Safety Evaluation Report (SER) on the URD. The ability of igniters to function in harsh environments and perform their function has been demonstrated by numerous experimental programs. A discussion of hydrogen igniter testing is provided in Reference 3.

4. Reactor Cavity Flood System

The reactor cavity flood system provides both (1) cooling to the corium debris contained in the reactor cavity and (2) scrubbing of fission products. The capability of the cavity flood system to perform its intended function has been established using the System 80+ version of MAAP 3.0B. These analyses included a parametric evaluation of the debris bed heat removal for a baseline station blackout scenario. These analyses varied the FCHF pool boiling heat transfer parameter from a minimum value of 0.03 to a nominal value of 0.1. The selected nominal evaluation for this study is representative of the heat removal characteristics observed for the SWISS experiments. The lower value conservatively bounds the reduced heat removal observed in the WETCOR and MACE debris coolability experiments. The results of this study are presented in Table 1. These results indicate that even in the presence of considerably degraded heat removal the concrete erosion will be limited and will not pose a threat to containment integrity.

5. Reactor Cavity Debris Retentive Design

In order to mitigate the consequences of a High Pressure Melt Ejection from the reactor vessel, the System 80+ plant has been designed with a debris retentive cavity. The experimental/analytical bases for this design are discussed in Reference 1 (Appendix D) and Reference 3, Section 4.1 (See also the response to Q. 410.146b).

REFERENCES:

1. DOE/ID-10271, "Prevention of Early Containment Failure due to High Pressure Melt Ejection and Direct Containment Heating for Advanced Light Water Reactors", Carter, J.C. et. al., March, 1990.

2. DOE/ID-10290, "Tecnnical Support for the Hydrogen Control Requirement for the EPRI Advanced Light Water Reactor Requirements Document", January, 1990.

3. ALWR-FS-DCTR-33, Rev.0, "System 80+ Severe Accident Phenomenology and Containment Performance", Combustion Engineering,Inc., August 1992.

Response to 410.164 (Cont'd)

TABLE 1: EFFECT OF DEGRADED HEAT TRANSFER ON CORIUM POOL COOLABILITY

	FCHF=.10	FCHF=.05	FCHF=.03
MAX. EROSION DISTANCE (FT)	0.0	0.087	0.75
TIME CONCRETE ATTACK ENDS (HR)	NA	11.5	15

ATTACHMENT 2

ALWR-FS-DCTR-33 Rev. 0

SYSTEM 80+
SEVERE ACCIDENT PHENOMENOLOGY
AND
CONTAINMENT PERFORMANCE

August 1992

Prepared

by

ABB Combustion Engineering
Nuclear Power

Combustion Engineering, Inc.

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LIST OF ACRONYMS

ALWR	Advanced Light Water Reactor
AICC	Adiabatic Isochoric Complete Combustion
CFS	Cavity Flooding System
CHR	Containment Heat Removal
DCH	Direct Containment Heating
EPRI	Electric Power Research Institute
EVSE	Ex-Vessel Steam Explosion
FITS	Fully Instrumented Test Facility
HEDL	Hanford Engineering Development Laboratory
HMS	Hydrogen Mitigation System
HPME	High Pressure Melt Ejection
LWR	Light Water Reactor
MAAP	Modular Accident Analysis Program
MWR	Metal Water Reaction
NTS	Nevada Test Site
IRWST	In-Containment Refueling Water Storage Tank
IVSE	In-Vessel Steam Explosions
PDS	Plant Damage State
PRA	Probabilistic Risk Assessment
PSV	Primary Safety Valve
RCS	Reactor Coolant System
RSG	Rapid Steam Generation
RV	Reactor Vessel
SASM	Severe Accident and Sealing Methodology
SBO	Station Blackout
SDS	Safety Depressurization System
SIT	Safety Injection Tank
SGTR	Steam Generator Tube Rupture
VB	Vessel Breach
V/O	Volume Percent

SEVERE ACCIDENT PHENOMENOLOGY AND CONTAINMENT PERFORMANCE FOR THE SYSTEM 80+ PWR

1.0 INTRODUCTION

The PRA indicates that design improvements of System 80+ have decreased the likelihood of core damage by a factor of about 120 over System 80. As a result of this effort a more balanced design has been achieved, and the contribution to severe accident risk from loss of offsite power (LOOP) events (including station blackout) has been significantly reduced. The design enhancements are discussed in response to NRC Requests for Additional Information (RAIs) on CESSAR-DC and are contained within the Level 1 PRA section (CESSAR-DC Appendix B).

This document provides a description of the severe accident mitigation features of the System 80+ design and provides a technical basis for the severe accident phenomenology modeling assumptions employed in the PRA. In addition it discusses containment performance of the design for selected event sequences leading to severe accidents. It also contains the responses to several RAIs relating to containment performance and establishes a fundamental basis for the interpretation of System 80+ Level 2 PRA.

2.0 SCOPE

This document will describe major System 80+ severe accident mitigation features. However, it is not intended to restate the basic design information found in CESSAR-DC. Instead, this document will highlight the details of various reactor systems, discuss them from the viewpoint of severe accident mitigation and management and clarify the impact of these mitigation features on the System 80+ PRA.

Section 3 of the document describes the System 80+ severe accident mitigation design features. The System 80+ design features considered include (1) a large dry steel primary containment, (2) a reinforced concrete secondary containment with an annulus ventilation system, (3) a reactor cavity flooding system, (4) a hydrogen mitigation system to prevent in-containment hydrogen concentration from reaching detonation levels, (5) a safety depressurization system, (6) a large reactor cavity designed for retention and cooling of core debris, (7) missile protection structures, and (8) an integrated shutdown cooling and containment spray system.

Section 4 provides a concise discussion of severe accident phenomenological issues and a technical basis for their treatment within the System 80+ PRA. Experimental data or analyses used to support PRA conclusions/assumptions and Supporting Logic Model (SLM) structure are identified.

Section 5 provides representative analytical assessments of the severe accident containment performance of the System 80+ design. Assessments presented in these sections are obtained from analyses performed with an enhanced version of the MAAP 3.0B Rev 16 (Reference 2.1). Enhancements have been included to allow proper representation of unique System 80+ design features. Representative performance of the System 80+ containment to a typical spectrum of severe accidents is provided.

Section 6 contains the conclusions with regard to the System 80+ design's containment performance under severe accident conditions.

3.0 SYSTEM 80+ DESIGN FEATURES FOR SEVERE ACCIDENT MITIGATION

3.1 CONTAINMENT DESIGN

3.1.1 Description of the Steel Containment

The System 80+ containment vessel, including all its penetrations, is a low leakage spherical steel shell which is designed to withstand the postulated Loss-of-Coolant-Accident (LOCA) or a Main Steam Line Break (MSLB) while limiting the postulated release of radioactive material to within the requirements of 10 CFR 100. Additionally, the containment and shield building provide a barrier against the release of radioactive materials which may be present in the containment atmosphere following an accident.

The containment spherical shell is 200 feet in diameter and is constructed of steel plate with a wall thickness of one and three-quarter inches. The material of construction is SA357 Class 2 steel. The containment shell is supported by sandwiching its lower portion between the building foundation concrete and the internal structure base of a spherical depression in an intermediate floor of the shield building. The shield building (see Section 3.2) is a reinforced concrete cylindrical building with a hemispherical dome which totally encloses the containment. There is no structural connection either between the containment and the internal structure, or between the containment and the shield building.

The spherical containment provides 3.34 million cubic feet of net free volume with its internal structures arranged in a manner to (1) protect the steel shell from missile threats (see Section 3.7), (2) promote mixing throughout the containment atmosphere (See Figure 3.1-1), and (3) comfortably accommodate condensable and non-condensable gas releases from design basis and severe accidents. The internal structures, which are made of reinforced concrete. enclose the reactor vessel and other primary system components. The internal structures provide biological shielding for the containment interior and missile protection for the reactor vessel and containment shell.

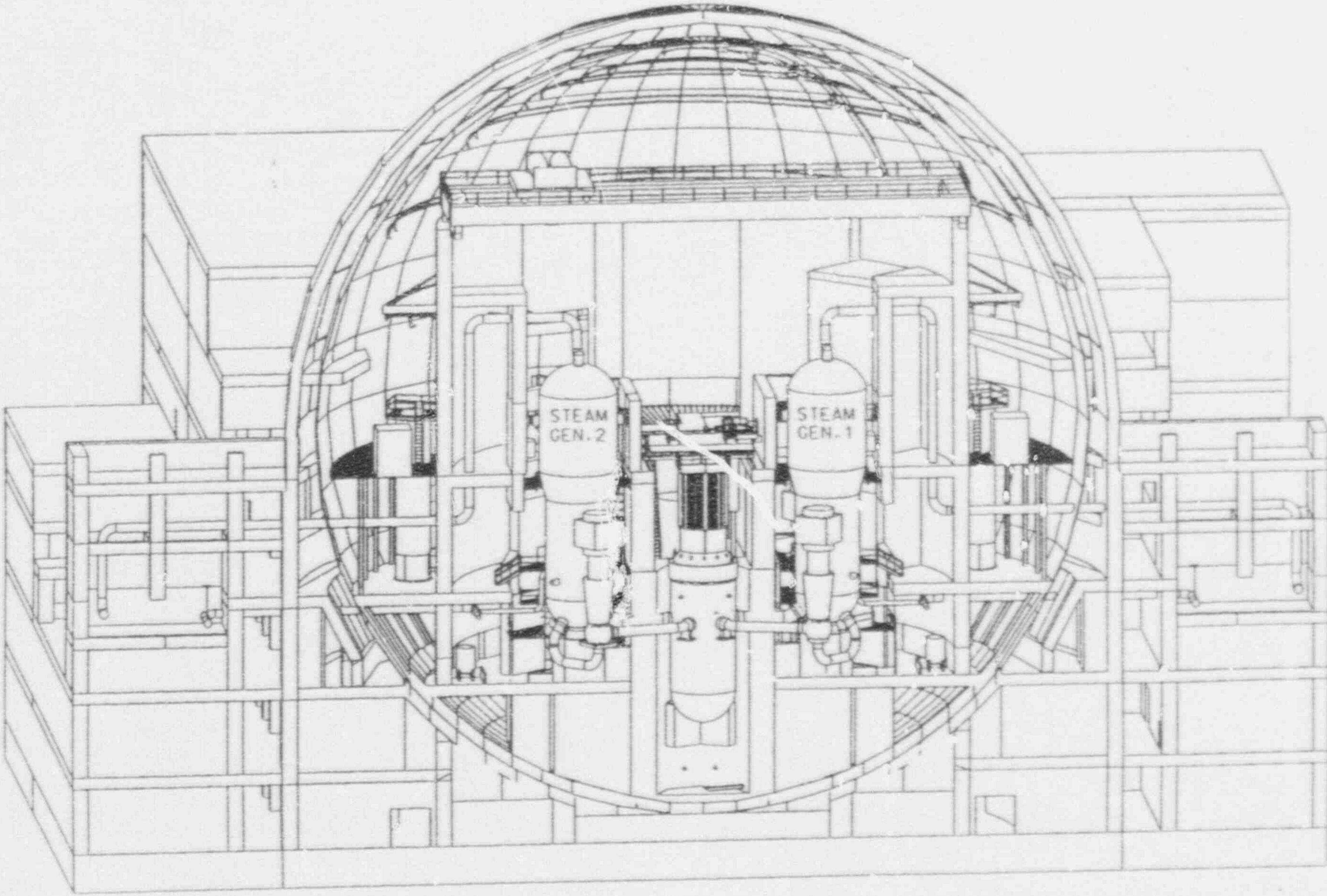
3.1.2 Containment Shell Pressure Limits

In severe accident scenarios the containment vessel is the last fission product barrier protecting the public from potentially large radiation releases. Therefore, it is of paramount importance to provide a strong robust containment design to meet severe accident internal pressurization challenges. To this end, several structural analyses have been performed to characterize the System 80+ containment strength. These analyses have investigated containment strengths based on design, ASME Service Level "C" and ultimate failure criteria. The results of these assessments are summarized below.

Figure 3.1-1

The System 80+™ Standard Design

Containment Internal Structure Arrangement



Design Basis Pressure Capacity

The containment vessel is analyzed to determine all membrane, bending and shear stresses resulting from the specified static and dynamic design loads. The vessel is idealized as a three dimensional thin shell using the finite element method of analysis. The stresses and deflections produced in the shell under the applied loads are calculated with the ANSYS computer program (Reference 3.1).

Seismic stresses and deflections are calculated using the response spectrum method. The frequencies of vibration and corresponding mode shapes are determined using the normal mode method. Modal responses are combined as described in Regulatory Guide 1.92 (Reference 3.2). The appropriate damping level for the applied response spectra is defined in Regulatory Guide 1.61 (Reference 3.3).

The critical buckling stresses in the containment vessel are determined by applying the appropriate safety factors and capacity reduction factors to the results of a three-dimensional analysis using an ANSYS finite element model similar to that constructed for the static and dynamic analyses. These methods are described in Article NE-3222 of the ASME Code and ASME Code Case N-284 (Reference 3.4). Code Case acceptability is in concurrence with Regulatory Guide 1.84 (Reference 3.5).

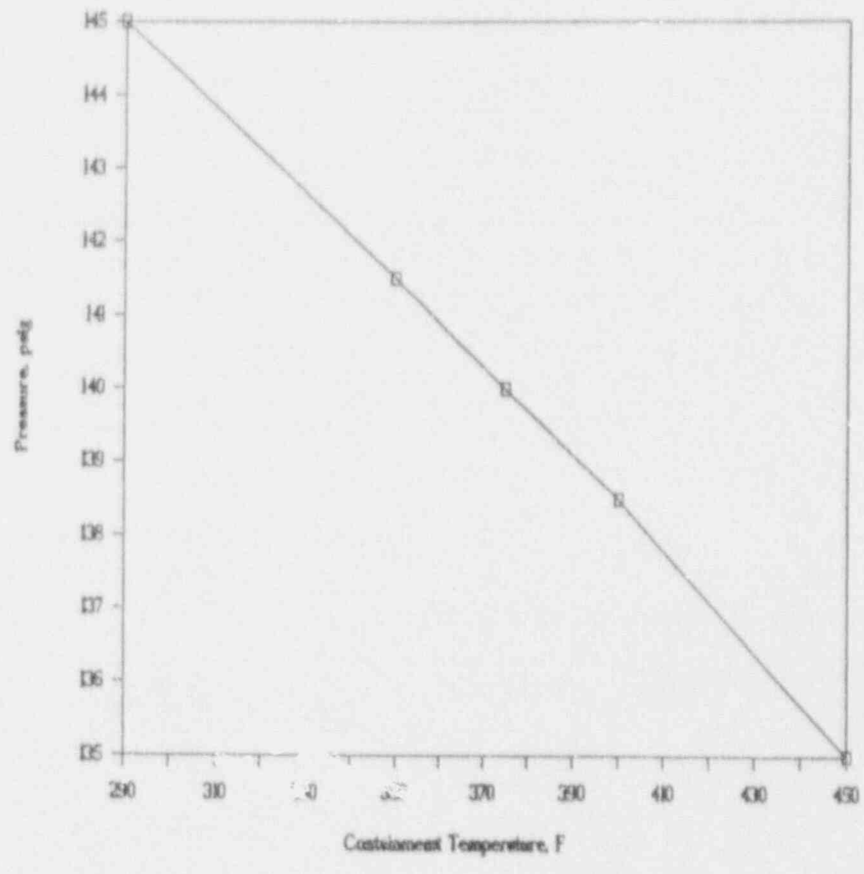
Based on these evaluations, the design basis pressure for the containment was calculated to be 49 psig. The analyses documented in CESSAR-DC Section 6.2.1.1 demonstrate that pressures resulting from large break LOCAs or main steam line breaks within the containment will not exceed this design pressure.

ASME Service Level "C" Stress Evaluation

An evaluation was performed to determine the containment pressure that may be reached without exceeding the ASME Boiler & Pressure Vessel Code, Service Level "C" allowable stress intensities. ASME Service Level "C" loading conditions allow material strains representative of incipient yield and provide a conservative estimate of the containment ultimate capacity.

The analyses considered a range of containment temperature values representative of severe accident degraded containment performance including the effects of dead weight. These evaluations were performed using an ANSYS model of the containment spherical shell. Results of this evaluation are presented in Figure 3.1-2 . These calculations indicate that pressure limits determined in accordance with ASME Service Level C criteria decrease from 145 psig at an average steel shell temperature of 290 degree F to 135 psig at a temperature of 450 degree F.

Figure 3.1-2: Pressure Limit Based on ASME Service Level "C" Criteria



System 80+ Ultimate Capacity Evaluation for PRA

A well defined set of criteria for evaluating the precise pressure at which the steel containment shell would actually fail is unclear. The expert elicitation for the Sequoyah Steel Containment evaluation (Reference 3.6) suggested the following guidance for relating the shell failure probability with the calculated shell membrane strain.

(1) For peak strains typical of incipient yield conditions, the probability of containment failure is less than 0.05.

(2) At shell membrane strains around 0.03 in/in, containment failure is most likely (probability of 0.5).

(3) At shell membrane strains greater than 0.05 to 0.1 in/in containment failure is virtually certain (probability of 0.9 to 1.0).

It was also noted that strains in the 3% equivalent membrane strain range would probably have a high probability of tearing at a containment penetration. In assessing the ultimate pressure for the System 80+ containment the PRA uses a conservatively low estimate of ultimate capacity based on assigning a 50% failure criteria to membrane strains only marginally greater than minimum yield (0.5% equivalent membrane strain) and absolute failure at 1% equivalent membrane strain. The methodology for establishing the containment pressure strain relationship is described below.

The System 80+ spherical steel containment vessel has been analyzed as an axisymmetric thin shell using the ANSYS finite element computer code. The analysis was a nonlinear elastic-plastic evaluation based on small strain, large displacement theory consistent with the ASME collapse load criteria found in Appendix II, Article 1430 of Section III of the ASME Code. The material properties were represented by a bilinear stress-strain curve which was assumed to be essentially elastic-perfectly plastic in nature. The overall stiffness matrix of the finite element model was updated with a stress stiffness matrix at the start of each iteration using the stress state computed in the previous iteration.

Stresses and strains are computed using the Von Mises failure criterion. ANSYS calculates an equivalent strain based on the Von Mises theory and determines the corresponding equivalent stress from the defined stress-strain curve. The equivalent stress is compared to the yield stress to determine when yielding has commenced. The strains calculated in the model when yield is reached are approximately 0.002 in/in and the strain at the maximum pressure of 193 psig is approximately 0.003 in/in. The exact value varies depending upon element location and whether the midsurface or inner/outer surface is examined.

It should be noted that material strains associated with the ASME ultimate capacity and Service Level C limits are well below the 0.02 in/in actual tensile failure point of SA537 Class 2 material used in the containment shell construction. Thus, the ultimate capacity referred to in the Standard Review Plan and Safety Analysis Reports refer to the ASME code defined maximum pressure capacity which is slightly lower than the pressure reached at tensile failure of the material. The ultimate pressure of 193 psig calculated using the methodology described above is consistent with the ASME code defined ultimate capacity and is selected to represent the 0.05 probable failure point at a steel shell temperature of 110 degree F (nominal operating conditions). The maximum radial deflections at a pressure of 193 psig are approximately 4 inches which correspond to the above defined failure strain of 0.003 in/in. As the mean steel shell temperature is increased to 290 degree F [typical of Design Basis Accident (DBA)] assumptions the ultimate pressure reduces to 169 psig. Extrapolation of these results to higher temperatures suggest that incipient yield will be reached at 161 psig and 147 psig at average shell temperatures of 350 and 450 degree F, respectively.

The containment ultimate pressure capacity based on 0.5% and 1% membrane strains were also determined using an extrapolation of the pressure-strain analyses performed with ANSYS for the System 80+ containment. The following results were obtained for a range of containment shell temperatures.

Shell Temperature (degree F)	Membrane Strain (%)	Ultimate Pressure (psig)
290	0.5	185
350	0.5	180
450	0.5	172
290	1.0	220
350	1.0	214
450	1.0	204

The temperature range selected for the analysis was based on anticipated plant transient performance during design basis accidents, as well as "wet" and "dry" cavity severe accident scenarios.

Containment Fragility Curve

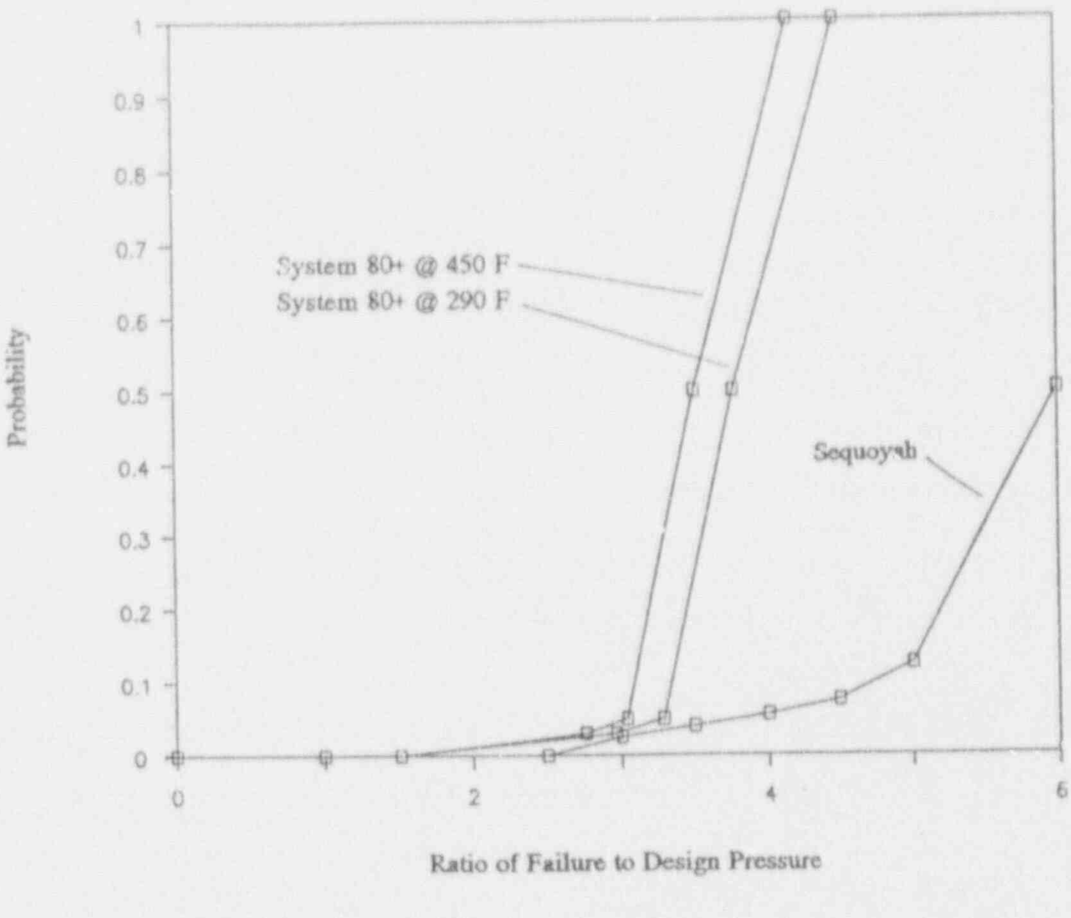
The stress-strain relationships for the steel containment presented in Section 3.1.2 were converted into probabilistic failure curves. To ensure a conservative treatment of containment strength, the following relationships were maintained between calculated average membrane strain and the probability of containment failure:

Pressure Level	Internal Pressure (psig)	Membrane Strain (% in/in)	Failure Probability
Design	49	0.0003	0.00
1.5 x Design	73.5	0.0007	0.00
ASME Level "C"	145-135	<0.002	0.03
Ultimate Capacity ASME	161-147	0.003	0.05
0.5% Strain	185-172	0.005	0.50
1.0% Strain	220-204	0.01	1.00

This method was used to translate data obtained from containment stress analyses to fragility (probabilistic failure) curves at temperatures typical of early and late containment failure. It was assumed that early failure stress curves allow greater strength because of the lower shell temperatures expected prior to containment failure. In these instances, containment failure is due to a rapid pressurization process to which the shell cannot thermally respond. The DBA peak temperature (290°F) was selected as the conservative temperature for evaluation of the early containment failure. Late containment failure includes a gradual overpressurization process that takes from hours to days; therefore, failure is expected to occur with a "hot" wall. The late containment failure fragility curve was conservatively established assuming the 450°F peak containment environmental temperature for the dry cavity overpressurization scenario was applicable.

Fragility curves generated using the pressure-failure probability points of the above table are shown in Figure 3.1-3. These fragility curves are presented in a normalized form and compared with the Reference 3.6 mean normalized fragility curve for the Sequoyah Steel Containment in Figure 3.1-3. As previously discussed and illustrated by this figure, the System 80+ fragility for use in the PRA has been intentionally skewed to lower failure pressures by limiting the maximum yield strain to 1%.

Figure 3.1-3: Comparison of Fragility Curves for System 80+



3.1.3 Containment Penetrations

The System 80+ containment pressure boundary is made up of the containment shell and several mechanical and electrical containment penetrations. These penetrations include a twenty-two foot diameter equipment hatch, two ten foot diameter personnel locks, containment piping penetration assemblies to provide for the passage of process, service, sampling and instrumentation pipe lines into the containment, electrical penetrations for power, control and instrumentation and a fuel transfer tube. Since the capacity of the pressure boundary can be estimated as the failure mode with the lowest predicted pressure capacity at elevated temperatures, the penetrations will be designed such that they will withstand containment pressures beyond the calculated ASME ultimate pressure capacity at containment temperatures typical of severe accident conditions. Details of the containment penetrations are presented in Section 3.8.2.1.3 of CESSAR-DC.

3.1.4 Containment Penetration Seals

Seals around penetrations are designed to seat under internal containment pressurization. This feature ensures minimal containment leakage at higher pressures.

Temperature degradation of seals around penetrations have been studied by Sandia Laboratories within the Containment Integrity Programs (Reference 3.7) for many common penetration seals (e.g., silicon, neoprene). These studies concluded that the temperature properties of sealant can vary considerably depending on the particular product. Several types of seals, notably silicon rubber and fluoroelastomer, were determined to have good high temperature stability. The material selection for penetration seals for the System 80+ design has not been specified at this time. When this selection is made, the temperature stability of the seal materials will play a major role in the selection process.

3.2 SECONDARY CONTAINMENT DESIGN

3.2.1 Purpose of System

The secondary containment consists of the containment shield building and the annulus between the steel containment vessel and the shield building. The containment shield building, which houses the containment vessel and safety-related equipment, is designed to provide biological shielding and external missile protection for the containment vessel and safety-related equipment. It is a reinforced concrete structure consisting of a right cylinder and hemispherical dome. The shield building shares a common foundation base with the nuclear system annex as shown in Figure 3.2-1. In addition, the annulus ventilation system provides a mechanism for substantially reducing and/or eliminating unfiltered fission product releases following design basis and severe accidents.

3.2.2 Description of the System

As described in Section 3.8.4.1.1 of CESSAR-DC, the containment shield building has an inner radius of 105 feet, with the cylinder wall thickness of 4 feet up to the nuclear annex roof elevation and 3 feet above as well as a dome thickness of 3 feet and a maximum height of 215 feet. An annular space between the containment vessel and the shield building above the 91 3/4 feet elevation is provided for structural separation and access to penetrations for testing and inspection.

An Annulus Ventilation System (AVS) serves the space between the primary containment and the containment shield building. The AVS does not perform any normal ventilation function. It is primarily designed to minimize and/or prevent radioactivity release following an accident. The system is designed as an engineered safety feature and is capable of functioning during startup, power operation, hot standby, and hot shutdown. A description of the AVS is provided in Section 6.2.3 of CESSAR-DC.

Post-accident operation of the AVS produces and maintains a negative pressure zone in the annulus and passes the annulus air through HEPA filters. This mitigates the consequences of airborne products of radiation that might otherwise become an environmental hazard during and following accident sequences including those leading to a severe accident.

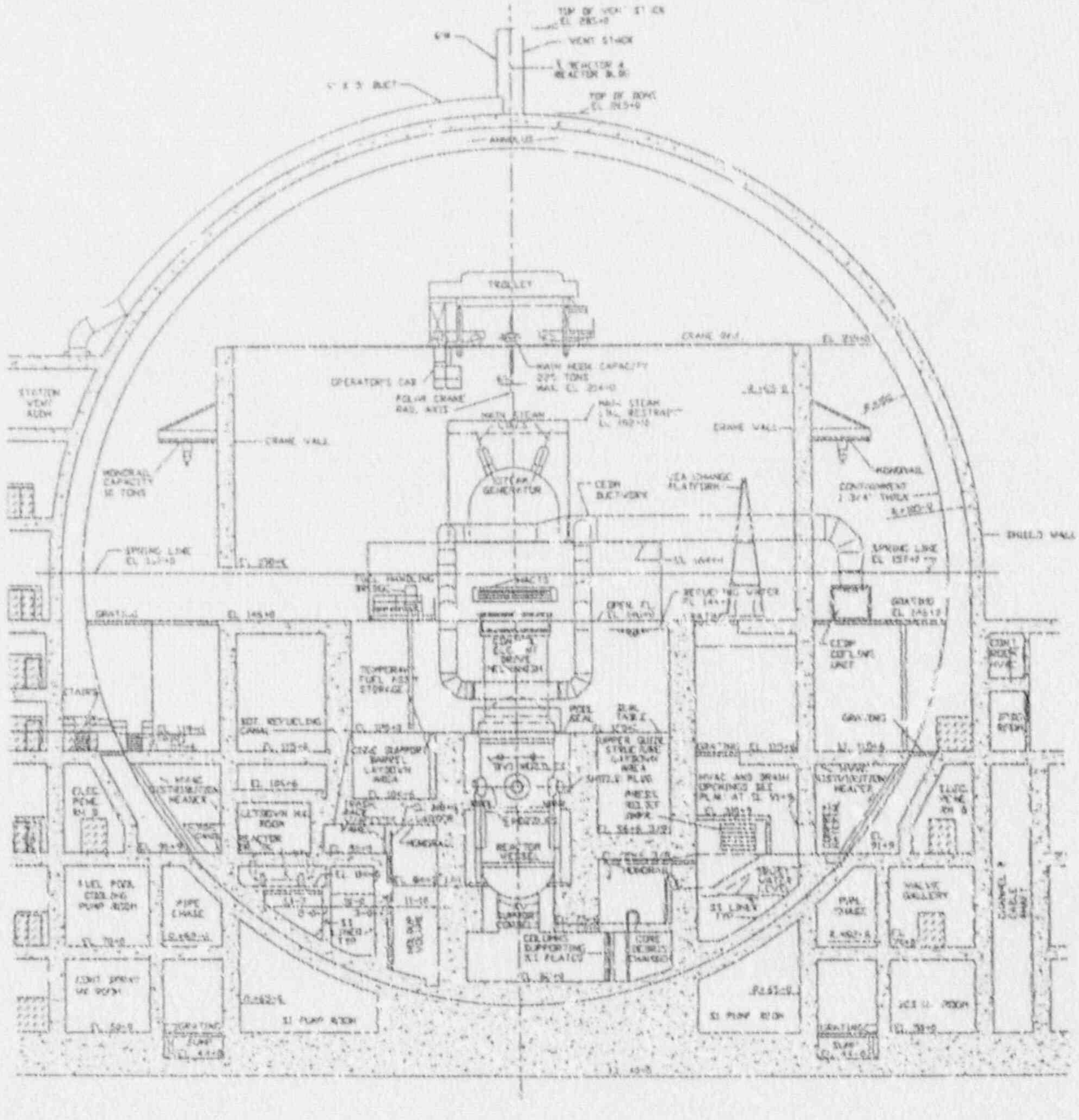
The AVS will filter annulus air at a minimum rate of up to 16,000 cfm. This discharge is sufficient to create a negative pressure of about -0.5 in water gauge with respect to the outside atmosphere following a LOCA. The AVS is a two-train system which is activated by the Containment Spray Actuation Signal (CSAS) and is designed to function during a seismic event. The system has no containment penetrations and is single failure proof. Filters are designed to accommodate design basis fission product loadings.

Each train of the AVS is powered by Class 1E Emergency Diesel Power. The AVS can minimize radiation releases to the environment following a severe accident scenario for which the containment vessel remains intact and emergency power is available.

3.2.3 Impact on PRA

Although the secondary containment with the AVS is included in the System 80+ design, no credit is taken for it in the System 80+ PRA presented in Appendix B of CESSAR-DC. In the updated PRA effort currently underway, the impact of this system will be considered.

Figure 3.2-1: Elevation View of System 80+ Containment shell and Shield Building



3.3 CAVITY FLOODING SYSTEM (CFS)

3.3.1 Purpose of the CFS

The function of the CFS is to provide a means of flooding the reactor cavity in the event of a severe accident for the purpose of cooling the core debris in the reactor cavity and scrubbing fission product releases. The cavity flood system is manually actuated and is designed (in conjunction with the containment spray system) to provide an inexhaustible continuous supply of water to quench the core debris.

3.3.2 System Description

A detailed description of the CFS can be found in Section 6.8 of CESSAR-DC. The CFS is designed as a manually actuated severe accident mitigation system. Procedures and guidance for actuating the CFS will be included in the System 80+ Severe Accident Management Guidelines. The CFS is designed to make use of available containment water sources along with active and passive System 80+ design features.

The components of the CFS include the In-Containment Refueling Water Storage Tank (IRWST), the reactor cavity Holdup Volume Tank (HVT), the reactor cavity, connecting piping, valves and associated power supplies. This system is used in conjunction with the containment spray system (see Section 3.8) to form a closed or recirculating water cooling system by providing a continuous cooling water supply to the corium debris. The quenching of the corium produces steam which is condensed by the containment spray flow. A schematic of the CFS is shown in Figure 3.3-1. The CFS takes water from the IRWST and directs it to the reactor cavity. The water flows first into the HVT by way of the four 12 inch diameter Holdup Volume Tank (HVT) spillways and then into the reactor cavity by way of two 10 inch diameter reactor cavity spillways.

The CFS valves are powered from the Class 1E 125 VDC Vital Instrumentation and Control Power System as described in CESSAR-DC Section 8.3.2.1.2.1 and Table 8.3-2-4. Each of the four holdup volume flooding valves are powered from separate Class 1E channels and each of the two cavity flooding valves are powered from separate Class IE divisions (See Table 8.3.2-4 of CESSAR-DC). The Class 1E busses are normally supplied from offsite power sources. Upon loss of offsite power, power to the busses can be supplied by the Class 1E diesel generators or the Class 1E batteries. In addition, the diverse Alternate AC source (combustion turbine generator) can power these busses upon loss of all other AC power.

Once actuated, movement of the water from the IRWST source to the cavity occurs passively due to the natural hydraulic driving heads of the system. Actuation of the CFS results in the opening of the HVT spillway valves allowing water from the IRWST to flood the HVT. The motive force for this flooding is gravity and the static head of water between the IRWST water level

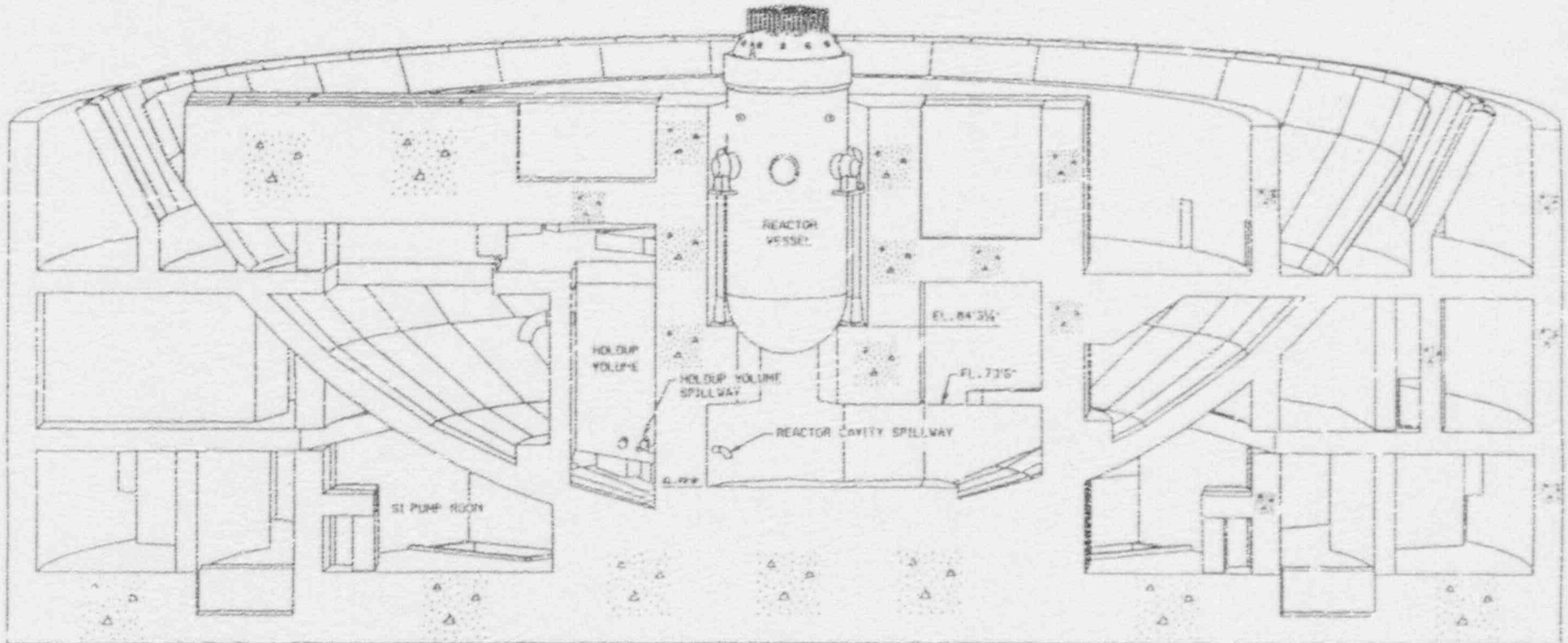


FIGURE 3.2-1: IRWST SPILLWAY AND CAVITY FLOODING SYSTEM

and the HVT water level. Flooding of the HVT progresses until the water level in the HVT reaches the reactor cavity spillway, at which time reactor cavity flooding commences. Flooding ceases when water levels in the IRWST, HVT and reactor cavity equalize.

To ensure the most rapid water delivery to the reactor cavity, the HVT spillways are located as low as possible (approximate elevation: 67.0 ft) to provide the greatest head. The HVT spillways and the reactor cavity spillways are equipped with remote manual motor operated valves. The reactor cavity spillways are located low enough to ensure sufficient flooding of the reactor cavity when the IRWST water level is at its minimum.

Flooding of the reactor cavity is an EPRI URD evolutionary plant design requirement and serves several purposes in the overall strategy to mitigate the consequences of a severe accident. These include:

(1) minimize or eliminate corium-concrete attack.

(2) minimize or eliminate the generation of combustible gases (hydrogen and carbon monoxide).

(3) reduce fission products released due to core-concrete interaction.

(4) scrub fission products released from the trapped core debris.

These features are discussed in detail in Section 4.3.2.

3.3.3 Role of the CFS in Accident Management

The CFS is a manually actuated system so that guidance will be developed to ensure that the system will be properly employed. The accident management guidance will require that the CFS be actuated once a core melt condition (i.e. prolonged and irreversible core uncovery) has been confirmed. Initiating CFS at this time will allow sufficient water inventory in the reactor cavity prior to RV failure. Ensuring a water filled reactor cavity will reduce the initial concrete ablation following RV breach and lead to a corium quench and provide scrubbing of fission products released in the cavity. Additional discussion of core concrete interaction phenomenon is provided in Section 4.

3.4 HYDROGEN MITIGATION SYSTEM

3.4.1 Purpose of the HMS

During a severe accident, large quantities of hydrogen can be generated during the core degradation and melting process. While it is unlikely the hydrogen generated will be sufficient to fail the containment, a Hydrogen Mitigation System (HMS) has been incorporated into the System 80+ design to provide added assurance that hydrogen concentrations will be maintained at non-detonable levels even during the most limiting severe accident. To this end, the Hydrogen Mitigatio System (HMS) is designed to accommodate the hydrogen production from 100% fuel clad metal-water reaction and maintain the average containment hydrogen concentration limit below 10% in accordance with 10 CFR 50.34(f) for a degraded core accident.

3.4.2 System Description

The HMS is a control room actuated system designed to allow controlled burning of hydrogen at low concentrations in order to preclude hydrogen concentration build-up to detonable levels. The system is designed to prevent the average hydrogen concentration in containment from reaching 10% by volume during a degraded core accident by burning hydrogen throughout the containment as the local concentration reach levels of between 4 to 6 %. Frequent small hydrogen burns will not threaten containment while simultaneously removing the possibility of a significant combustion threat later in the accident.

Experimental studies performed by Acurex Corporation (Reference 3.8) and Sandia National Laboratories (Reference 3.9) suggest that hydrogen burning from the igniters will not jeopardize the survivability of critical plant equipment.

Each igniter is an AC powered glow plug powered directly from a step down transformer. Each igniter assembly consists of a 1/8" thick steel enclosure (8" H x 6" W x 8" D) which contains the transformer and all electrical connections and partially encloses the igniter. The enclosure meets National Electrical Manufacturers Association (NEMA) Type 4 specifications for water-tight integrity under various environmental conditions, including exposure to water jets. The sealed enclosure incorporates a heat shield to minimize the temperature rise inside the igniter assembly, and a spray shield to reduce water impingement on the glow plug from above. The igniter assembly is designed to meet Seismic Category I requirements.

The igniters are powered from the Class 1E 120 VAC Vital Instrumentation and Control Power System as described in CESSAR-DC Section 8.3.2.1.2.1 and Table 8.3.2-3. The system normally receives power from offsite sources. In the event of a loss of offsite power, the igniters will be powered from the emergency diesel generators. Group A igniters will be powered from the Division I diesel generator and Group B igniters from the Division II diesel generator. On loss of offsite power and failure of the emergency diesel generators to start or run (Station Blackout), the igniters can be powered from the alternate AC source combustion turbine generator or the station emergency batteries via DC-to-AC inverters.

The HMS consists of 42 igniters which are divided into redundant groups, Group A and Group B. Each group has independent and separate control, power, and igniter locations to ensure adequate coverage within the containment. Although the detailed individual circuit design for the igniter power supply is not currently available, as described in CESSAR-DC Section 6.2.5.2.2, one to ten igniters are expected to be included within an electrical circuit. Each igniter location has two igniters, one powered from each division of the Class 1E Vital Instrumentation and Control Power System. Upon loss of an entire division, the system is still operable. The loss of a complete circuit or an entire division will not result in the complete loss of igniters in a compartment.

3.4.3 Igniter Placement

The hydrogen igniters are placed so as to achieve controlled hydrogen burning. Although the containment is designed to promote mixing, the igniters will be positioned in areas where hydrogen is produced most rapidly. Local areas of potentially high hydrogen concentrations will have two igniters, one from Group A and one from Group B. Considerations for igniter positioning are as follows:

A. Local positioning at the In-Containment Refueling Water Storage Tank (IRWST) safety valve outlets and the IRWST spillway outlet in the holdup volume, due to hydrogen production from in-vessel oxidation.

B. Local positioning in the cavity area, due to dry cavity scenario where zirconium and steel are freely oxidized using steam from concrete as a water source.

C. Vulnerability to damage from a pipe rupture during a LOCA.

The placement of the igniters within the containment considered the potential for damage to the igniters due to blowdown during a LOCA. The locations were selected to minimize the potential for damage to the igniters and to provide redundant coverage within a specific location. The updated CESSAR-DC, Section 6.2.5 submitted via letter LD-92-050 shows the approximate location of the igniters within containment and the separation between the igniters. These igniters are designed to burn hydrogen at low concentrations and located where hydrogen is produced or could accumulate (i.e., closed compartments or dead-ended regions) in the containment.

3.4.4 Role of the HMS in Accident Management

The HMS igniters are intended to be used in controlling the concentration of hydrogen within the containment once the operator confirms that an extended core uncovery is in progress. The operator will use the HMS based on specific accident management guidance which will rely on RCS and containment instrumentation, such as in-vessel level monitoring instrumentation, core exit thermocouples, ex-vessel level monitoring instrumentation, and containment and RCS pressure indications and a direct measure of containment hydrogen. Once activated the igniters will produce small burns and/or diffusion flames that serve to reduce the containment hydrogen concentration and thereby prevent the potential for destructive hydrogen detonations within the containment. The specific severe accident management guidance will be developed based on ongoing NUMARC and NRC Accident Management Guidelines effort.

3.5 SAFETY DEPRESSURIZATION SYSTEM

3.5.1 Purpose of the SDS

The Safety Depressurization System (SDS) is a multi-purpose dedicated safety system specifically designed to serve important roles in severe accident prevention and mitigation. Section 6.7 of CESSAR-DC provides details of the SDS. In the context of severe accident prevention, the SDS performs the following functions:

A. Venting of the Reactor Coolant System (RCS)

The Reactor Coolant Gas Vent (RCGV) function of the SDS provides a safety-grade means of venting non-condensable gases from the pressurizer and the reactor vessel upper head to the Reactor Drain Tank (RDT) during post-accident conditions. In addition, the RCGV provides:

1. Safety-grade means to depressurize the RCS in the event that pressurizer Main Spray and Auxiliary Spray systems are unavailable.

2. Means of venting the pressurizer and reactor vessel upper head during pre-refueling and post-refueling operations.

B. Rapid Depressurization (bleed process) of the RCS.

The Rapid Depressurization (RD) function, or bleed function, provides a manual safety-grade means of quickly depressurizing the RCS when normal and emergency feedwater (EFW) are unavailable to remove core decay heat through the steam generators. This function is achieved via remote manual operator control. Whenever any event e.g., a total loss of feedwater (TLOFW) results in a high RCS pressure with a loss of RCS liquid inventory, the SDS rapid depressurization or bleed valves may be opened by the operator, resulting in a controlled rapid depressurization of the RCS. As the RCS pressure decreases, the Safety Injection (SI) pumps start, initiating feed flow to the RCS and restoring the RCS liquid inventory. The RD function allows for both short or long-term decay heat removal.

The rapid depressurization feature of the SDS also serves an important role in severe accident mitigation. In the event a high pressure meltdown scenario develops and the feed portion of feed and bleed cannot be established due to unavailability of the SI pumps, the SDS can be used to depressurize the RCS to ensure that a High Pressure Melt Ejection (HPME) event does not occur thereby minimizing the potential for direct containment heating.

3.5.2 System Description

Design Requirements for Rapid Depressurization

The Rapid Depressurization (RD) design requirements are summarized in Section 6.7 of CESSAR-DC. Of particular interest to severe accident mitigation is the capability to depressurize the RCS from 2500 to 250 psia prior to reactor vessel melt-through. This is accomplished by designing the SDS rapid depressurization valves for initial bleed flow of approximately 412,000 lbm/hr of steam.

Power to SDS Valves

The power supply for each rapid depressurization valve is from a DC bus. The power is provided such that in case of a loss of offsite power, both EDGs, the combustion turbine, and the loss of one battery bank, a RD bleed path can be established. Each DC load group is provided with a separate and independent 125 volt battery charger. The battery chargers are powered from Division I and II of the Class 1E Auxiliary Power Systems. Each battery is sized to supply the continuous emergency loads of its own load group for a period of 4 hours.

3.5.3 System Performance During Severe Accidents

Rapid Depressurization (RD) Evaluation

The RD valve size was selected to meet both the feed and bleed and DCH severe core damage depressurization goals. The following RD valve sizing criteria were established for the worst case Total Loss of Feedwater (TLOFW) event to ensure that feed and bleed capability:

Criterion 1. The primary system shall maintain a two-phase mixture level two feet above the top of the core when a single feed valve and bleed valve are opened simultaneously with the primary safety valves and two SI pumps operational.

Criterion 2. The primary system shall maintain a two-phase mixture level of two feet above the core when two equally sized feed and bleed valves are opened after the primary safety valves lift with four SI pumps operational.

The severe core damage depressurization goal is to ensure that the RD can depressurize the RCS from 2500 to 250 psia prior to a reactor vessel melt-through.

The ability of the RD system to accomplish these goals were validated via CEFLASH analyses. MAAP analyses confirmed that the RD system can be used to depressurize the RCS prior to vessel failure provided the RD is actuated within two hours following pressurizer safety valve lift during either an extended total loss of feedwater or station blackout scenario.

3.5.4 Role of the SDS in Accident Management

The RD function is performed by opening the rapid depressurization valves located on the top of the pressurizer. In situations where rapid depressurization is used to establish once-through-core-cooling (OTCC), the Safety Injection pumps can be activated to provide a continuous source of RCS inventory.

The RD function is normally not used and is primarily intended for mitigating the consequences of a beyond design basis event such as a total loss of normal and emergency feedwater, or as an emergency means of RCS depressurization and pressure relief when the main spray, auxiliary spray, and the reactor coolant gas vent system (RCGVS) functions are not available. In a severe accident environment, the RD may also be used to depressurize the RCS prior to a projected RV lower head breach. This action will add residual SIT water inventory to the RCS and drop the RCS pressure to below the anticipated corium dispersal threshold value.

The OTCC RD function is accomplished by means of a manually operated, safety-grade system, utilizing certain components and equipment from the following safety systems:

A. The In-containment Refueling Water Storage Tank (IRWST) which provides a quench volume and heat sink; (see CESSAR-DC Section 6.8).

B. Four Safety Injection pumps and associated direct vessel injection lines which provide the feed function; (see CESSAR-DC Section 6.3).

C. Two Shutdown Cooling System heat exchangers; (see CESSAR-DC Section 5.4.7).

D. Two Shutdown Cooling System pumps (see CESSAR-DC Section 5.4.7).

Opening the rapid depressurization or bleed valves results in a rapid depressurization of the RCS which allows the SI pumps to be automatically started to refill the RCS and provide cooling of the core.

Core decay heat removal, using the RD function, is accomplished by a once-through cooling process in which water is injected directly into the reactor vessel downcomer via the Safety Injection System. Once in the reactor vessel, the cooling fluid passes through the vessel downcomer to the lower plenum, up through the core (where decay heat is removed) and out to the hot leg, through the surge line to the pressurizer and out through the dedicated rapid depressurization bleed valves to the piping sparger in the IRWST where quenching and cooling of the bleed flow is accomplished. The volume within the IRWST allows a feed and bleed operation to be maintained for about thirty minutes before external cooling of the IRWST should be initiated. IRWST cooling is provided by the safety grade Shutdown Cooling System heat exchangers which in turn are cooled by the component cooling water system. In addition, the Containment Spray System heat exchangers may be used to cool the IRWST.

3.6 REACTOR CAVITY DESIGN

3.6.1 Reactor Cavity Design Philosophy

The System 80+ reactor cavity is configured to promote retention of and heat removal from the postulated core debris during a severe accident, thus, serving several roles in accident mitigation. Corium retention in the core debris chamber virtually eliminates the potential for significant DCH induced containment loadings. The large cavity floor area allows for spreading of the core debris enhancing its coolability within the reactor cavity region.

3.6.2 Description of the Reactor Cavity

Figure 3.6-1 shows a schematic of the System 80+ reactor cavity design. The important features of the System 80+ cavity include:

1. a large cavity volume
2. a weakly vented vertical instrument shaft
3. a convoluted gas vent
4. a large recessed corium debris chamber
5. a large cavity floor area

The significance of these features to severe accident plant performance are discussed in the following paragraphs.

3.6.2.1 Large Cavity Volume

The System 80+ cavity includes 32,000 cu. ft. of free volume. This large volume benefits the plant design when cavity pressurization issues are considered. Large (and well vented) volumes, such as those in System 80+, are not prone to significant pressurization resulting from vessel breach or during the corium quench processes. Post-accident cavity pressurization analyses performed for System 80+ indicate peak cavity pressure loadings to be less than 100 psid (see Section 4.1.4.2). This loading is within the cavity design basis.

3.6.2.2 Weakly Vented Vertical Shaft

The instrument shaft design serves an important purpose in the severe accident mitigation for System 80+. First, by orienting the instrument shaft vertically and providing limited gas venting in this path, the possibility of corium carryover is minimized. Analyses provided in Reference 3.10, Appendix D suggests that only 10% of entrained corium could be expected to initially be carried upward into the vertical shaft even if the shaft were vented to accommodate significant gas flows. The remainder of the corium not entering the vertical shaft will be captured in a large debris retention chamber located at the base of the instrument shaft (see Figure 3.6-1).

3.6.2.3 Convoluted Gas Vent Escape Pathway

In the design of the System 80+ containment a significant effort has been made to ensure that actual venting to the upper containment either by the vertical shaft (See Figure 3.6-2) or around the RV flange is negligible. Thus, the actual steam exits via a convoluted pathway above the top of the core debris chamber and through louvered vents under the refueling pool (See Figure 3.6-2). As a consequence the dominant hot gas and corium carryover pathway will be to the lower portion of the containment where the containment shell is fully protected by the crane wall (See Section 3.7).

3.6.2.4 Core Debris Chamber

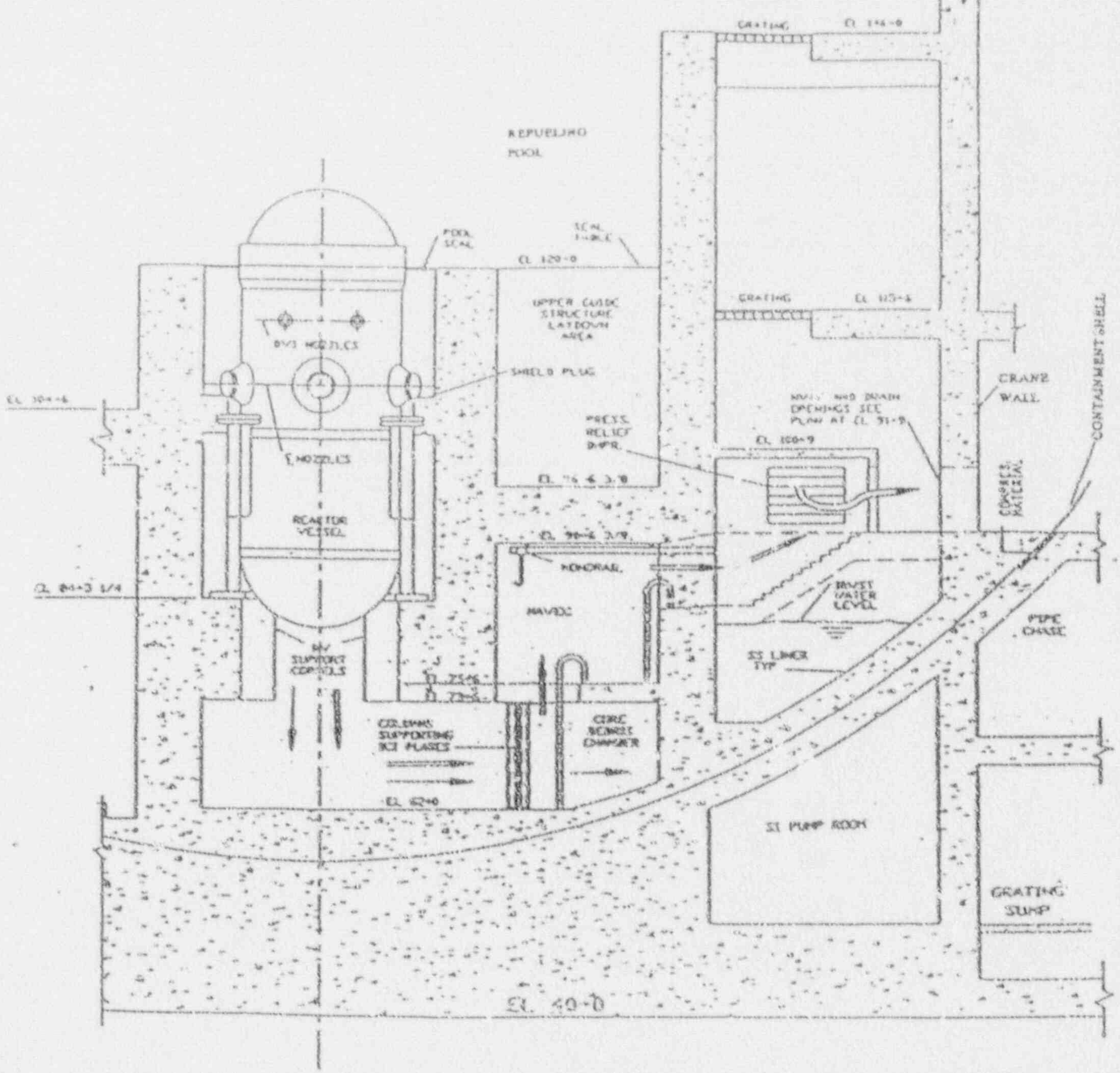
The cavity region of the System 80+ design is designed to minimize debris entrainment and subsequent debris dispersal into the upper compartment of the containment. As discussed above, by maximizing debris retention and minimizing debris dispersal to the upper compartment, containment threats resulting from High Pressure Melt Ejection (HPME) processes (in particular, DCH) can be virtually eliminated.

System 80+ is equipped with an offset core debris chamber designed to de-entrain and trap the debris ejected during a reactor vessel breach. The reactor cavity debris chamber and exit shaft have been designed such that following a failure of the reactor vessel high inertia corium debris would de-entrain and collect in the debris chamber while the lower inertia steam/hydrogen/air mixture would negotiate a right angle turn and exit the reactor cavity via a convoluted vent path. The chamber has been sized according to ARSAP guidance (Reference 3.10) to hold twice the post-severe accident maximum corium volume. Once deposited in the debris chamber, the debris would be difficult to re-entrain since the retention zone would exhibit a low velocity recirculation flow pattern. Any corium negotiating the 90 degree turn would be de-entrained by the reactor cavity concrete ceilings and seal table structure.

3.6.2.5 Floor Area

The reactor cavity is sized and configured to spread out the ejected core debris over the floor surface area during a postulated severe accident so as to meet the proposed EPRI URD (Reference 3.11) criteria of 0.02 m^2/MWt of flat surface area below the vessel. As a consequence of this large floor area the approximate depth of corium covering the cavity floor would be relatively shallow.

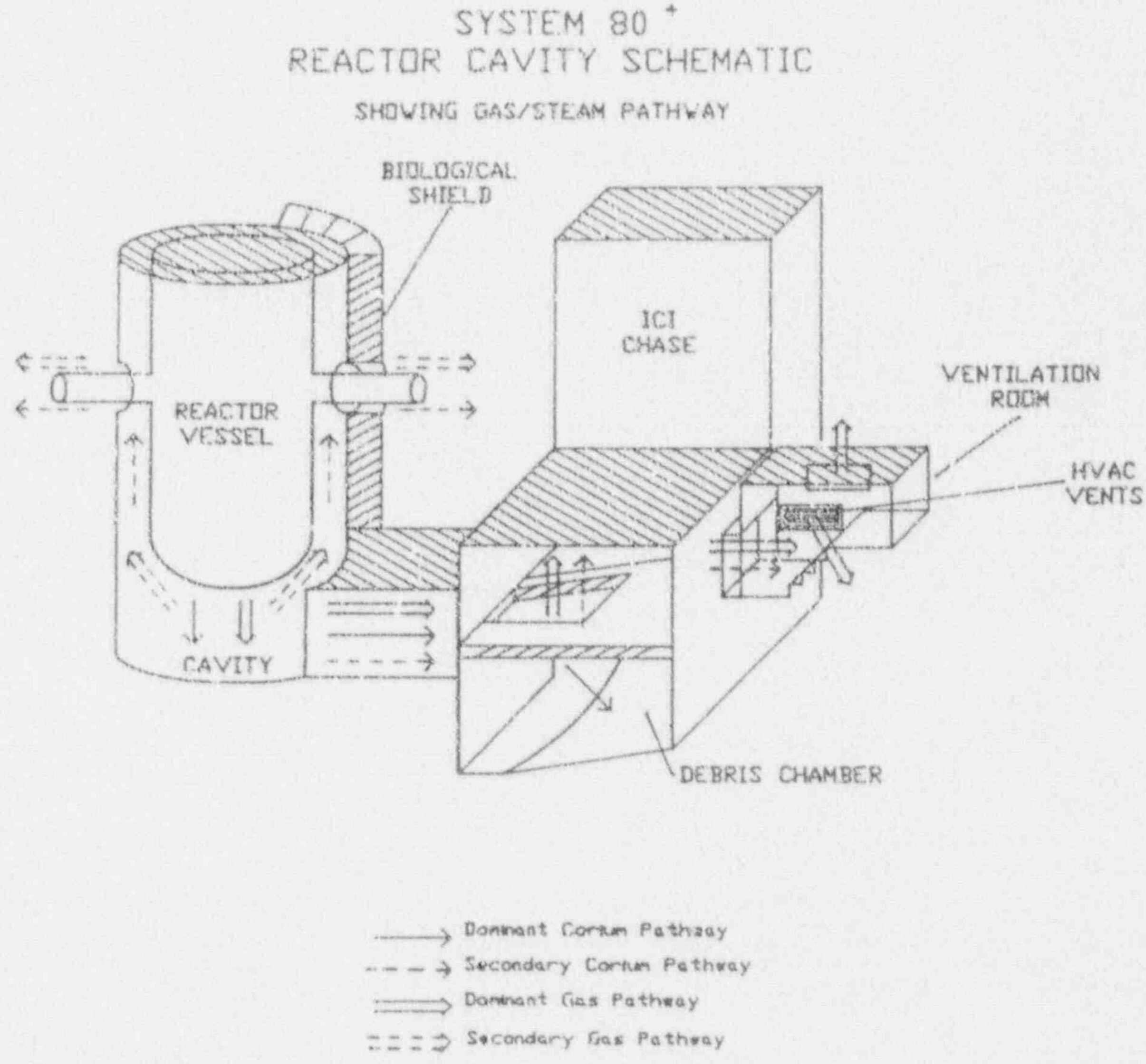
Figure 3.6-1: System 80+ Reactor Cavity with Core Debris Chamber



DOMINANT CORIUM FLOWPATH

DOMINANT GAS FLOWPATH

Figure 3.6-2: System 80+ Reactor Cavity Schematic



Experimental data appears to support the coolability of corium debris beds at least in the long term (Reference 3.12). For System 80+ this depth represents a near upper bound on the corium depth expected. Even if these deep debris beds are not fully cooled, partial cooling will quench the upper layers of corium and retard any concrete attack. Additional details on the core concrete interaction process and the System 80+ anticipated performance are presented in Sections 4.2.1 and 4.2.2.

The containment liner is adequately embedded in concrete in the reactor cavity area to preclude direct contact of the core debris with the containment. Note that the exit path for steam is separate from the incore instrumentation chase and that once out of the reactor cavity, the containment is protected from the hot gases exiting the cavity by the crane wall.

3.6.3 Response to Severe Accidents

The current models, supported by available test data, predict that the System 80+ cavity design mitigates the DCH effect by limiting the amount of debris leaving the cavity as finely fragmented particles.

The estimate of the fraction of the debris able to negotiate the turn into the vertical cavity shaft was established using a Sandia correlation for debris impingement determined from high pressure melt ejection tests (See Reference 3.10, Appendix D). Application of this model to the System 80+ cavity geometry results in a prediction that 90% of the corium debris would be de-entrained into the debris chamber and that 10% of the debris could potentially negotiate the turn into the reactor cavity shaft. Consequently, the probability of dispersing corium into the upper compartment was conservatively assumed to be 0.1. This estimate neglects the significant debris de-entrainment capability of the cavity ceiling and internal shaft structures and walls.

3.7 MISSILE PROTECTION

3.7.1 Purpose

The missile protection design features for Seismic Category I structures, systems and components of System 80+ design are described in Section 3.5 of CESSAR-DC. This section discusses those features of the System 80+ reactor cavity design which are specifically designed to protect the containment from the hot corium debris and induced missiles during a severe accident.

During severe accidents, missile protection of the containment shell is primarily accomplished by the use of protective shields and barriers either near the source of a potential missile or in front of the containment shell (such as the crane wall).

During a severe accident containment threatening missiles can be generated by a variety of sources including:

(1) the core debris which may be ejected from a breach in the reactor vessel.

(2) RV top head/CEAs resulting from in-vessel steam explosions (Alpha mode Failures)

(3) Containment internal structures following a rapid flame acceleration or hydrogen detonation

The means by which System 80+ protects the containment shell from these loadings are summarized in the following sections.

3.7.2 Protection From Hot Core Debris

A multi-faceted approach is used to ensure containment integrity during a severe accident. The first emphasis in the protection of containment integrity is prevention. That is both prevention of a severe accident and should a severe accident occur, prevention of a direct challenge to the containment shell by "in cavity" retention. The missile shields provide still another defense barrier containment shell defence barrier.

Section 3.5 describes the cavity geometry. The cavity is arranged such that any corium debris leaving the cavity will exit via the door or louvers in the HVAC room above the IRWST pool (see Figure 3.6-2) or via the nozzle cutouts. Corium debris released in these areas will likely interact with either the crane wall or refueling pool wall and ultimately deposit in these areas. In the highly unlikely event that the corium debris is projected upward out of the cavity annulus, the RV missile shield serves to protect the containment shell from a direct corium attack of the containment shell.

Within the reactor cavity the containment shell is protected from the corium debris by a concrete basemat layer varying from 3 to about 5 feet thick. Assuming flooded cavity conditions, it is conservatively estimated that this level of concrete flooring will protect the containment shell from corium debris contact for upwards of 30 hrs at its thinnest point.

3.7.3 Protection From Missiles Generated via Top Head Failures

Details of the System 80 + missile protection features are presented in Section 3.5 of CESSAR-DC. Missiles generated via failure of the top head and top head components are considered under the general severe accident category of "in-vessel" or "alpha" mode failure. In these scenarios the upper head or control rod missile will be intercepted by a 3 ft thick concrete shield located directly above the RV upper head. Thus, consequential damage of the containment steel shell due to failure of a pressurized RV is highly unlikely. Additional discussion of "alpha" mode failures is presented in Section 4.1.1.

3.7.4 Protection Following Induced Missiles Caused by Rapid Deflagrations/ Hydrogen Detonations and Ex-Vessel Steam Explosions

This section is concerned with protection of the containment shell from missiles produced either via a steam explosion or hydrogen burn in containment. The phenomenology of the Ex-Vessel Steam Explosion (EVSE) is described in Section 4.1.2. The EVSE may occur when corium debris contacts a water pool. Such processes are expected in the reactor cavity following a RV breach. While EVSEs are considered plausible, their consequences on containment integrity are insignificant. This arises from the fact that EVSEs are not expected to capable of damaging the reactor cavity structures required for support of the RV/RCS. All "in-cavity" structures that may be damaged by such explosions will be confined non-load bearing structures and thus will not compromise containment integrity.

Rapid deflagrations and detonations have the potential of generating missiles of sufficient strength to potentially damage the steel shell. To prevent such potential challenges, care has been taken to locate potential missiles within the crane wall and to protect the steel shell with a protective barrier of at least three feet of concrete wherever practical. This barrier level is maintained in the containment with the exception of a small region above the top of the refueling pool and in restricted basemat areas beneath the corium debris chamber and the HVT sump. These regions are not considered to be prone to missiles.

3.7.5 Application of Missile Generation Within the PRA

While the expectation is that containment failure due to missile impact on the containment structure highly unlikely, this issue is explicitly and implicitly considered within the System 80+ PRA. Details of the containment failure phenomenology are presented in Section 4. The containment failure modes that involve missile induced containment failure are summarized below:

1. Alpha mode failure considers the potential for the RV upper head or upper head component to failure containment via missile generation. The potential o this event has been established based on results of the Steam Explosion Working Group (SERG) to be about .001 conditional upon a low pressure core melt failure sequence (See Section 4.1)

2. An assessment of EVSE is presented in Section 4.1.2. Results of this assessment suggest that while EVSEs are credible, they will not pose a credible threat to containment integrity. The probability of an induced EVSE containment failure was therefore assessed as zero.

3. Detonations are not expected within the System 80+ containment. However, in the highly unlikely event that a detonation does occur in containment, it is conservatively assumed that a containment failure (due to missile generation, etc.) will result.

While missile damage to the containment is not believed to be credible, the updated System 80+ PRA conservatively assumes that in the highly unlikely event a detonation occurs in containment, containment failure occurs.

3.8 CONTAINMENT SPRAY SYSTEM

3.8.1 Purpose of the CSS

The Containment Spray System (CSS) is a safety grade system designed to reduce containment pressure and temperature from a main steam line break, loss-of-coolant-accident or a severe accident and to remove iodine from the containment atmosphere. Iodine removal is required so that in the event of containment leakage, activity at the site boundary due to radioactive iodine will be reduced.

The Containment Spray System is designed to provide adequate cooling of the containment atmosphere to limit post-design basis accident building temperatures and pressures to less than the containment design values (49 psig and 290 degree F, See Section 6.2.1 of CESSAR-DC). Additionally, it reduces the release of radioactive material from the containment in the event of a primary or secondary break (the limiting events are a Loss of Coolant Accident (LOCA) and a Main Steam Line Break (MSLB) in two ways:

1. Reduction of containment pressure to nearly atmospheric pressure thereby reducing the potential leakage rate from containment; and

2. The boric acid solution minimizes the fission product iodine in the building atmosphere by the removal of iodine through the absorption of iodine by the spray droplets.

3.8.2 System Description

The CSS uses the In-Containment Refueling Water Storage Tank (IRWST) and has two independent trains (two containment spray pumps, two containment spray heat exchangers, two independent spray headers, and associated piping, valves, and instrumentation). The pumps and remotely operated valves may be operated from the control room. Table 6.5-1 of CESSAR-DC provides a summary of containment spray system design parameters.

The CSS provides sprays of borated water to the containment atmosphere from the upper regions of the containment. The spray flow is provided by the containment spray pumps which take suction from the IRWST. The containment spray pumps starts upon the receipt of a Safety Injection Actuation Signal (SIAS). The pumps discharge through the containment spray heat exchangers and the spray header isolation valves to their respective spray nozzle headers, then into the containment atmosphere. Spray flow to the containment spray headers is not provided until a Containment Spray Actuation Signal (CSAS) automatically opens the containment spray header isolation valves. The spray headers are located in the upper part of the containment building to allow the falling spray droplets time to approach thermal equilibrium with the steam-air atmosphere. Condensation of the steam by the falling spray results in a reduction in containment pressure and temperature.

The CS pumps and CS heat exchangers can be manually aligned to provide cooling of the IRWST during post-accident feed and bleed operations when the steam generators are not available to cool the RCS.

The CS pumps are designed to be functionally interchangeable with the Shutdown Cooling System (SCS) pumps. Though not required for normal operation or accident mitigation, interchangeability of the pumps allows backup of the CS pumps and increases the reliability of the containment spray function.

A two out of four containment pressure high-high signal from the Engineered Safety Features Actuation System generates a Containment Spray Actuation Signal (CSAS) which initiates containment spray operation. Upon receipt of a CSAS, the Containment Spray Header isolation valve opens and the containment spray pump starts in each of the two redundant divisions. The pumps take suction from the In-Containment Refueling Water Storage Tank (IRWST) and discharge through the containment spray heat exchangers and the spray header isolation valves and to their respective spray nozzle headers, then into the containment atmosphere.

The CSS pumps, valves, and instrumentation are capable of being powered from the plant turbine generator (onsite power source), plant startup power source (offsite power), and the emergency generators (emergency power). Power connections are through a minimum of two independent buses so that in the event of a LOCA in conjunction with a single failure in the electrical supply, the flow from at least one containment spray train is available for containment heat and fission product removal. An independent electrical bus, as described above, supplies one containment spray pump and associated valves and instrumentation.

To further increase the reliability of the containment spray function, the containment spray headers are designed to accept spray flow from an external source of water supply via a "tee" connection to the spray line.

In case of unavailability of normal containment spray flow, the external source can supply water to the headers, allowing for containment cooling and depressurization.

3.8.3 Role of CSS in Accident Management

The containment spray pumps are automatically started by an SIAS. Containment spray flow to the containment does not occur until a CSAS opens the containment spray header isolation valves. The specific sequence of pump and valve actuation depends on which power source is available. If offsite power is not available, the safeguards loads are divided between the two plant emergency diesel generators and are sequentially started after the diesel generators are running.

Once the spray pumps are started and the valves are opened, the spray water flows into the containment spray headers. These headers contain spray nozzles which break the flow into small droplets, thus enhancing the water's cooling effect on the containment atmosphere. As these droplets fall to the containment floor they absorb heat until they approach thermal equilibrium with the containment. When the water reaches the containment floor it drains to the holdup volume and subsequently back to the IRWST.

The CSS uses a nozzle that provides a drop size distribution which has been established by testing and found suitable for the fission product removal function. The CSS provides a nozzle pressure differential of 40 psid which fixes the drop size distribution. The mean drop size produced at this differential pressure is 530 microns.

The CSS is currently designed to provide coverage for 90% of the containment net free volume. The remaining 10% of the containment net free volume is assumed to be unsprayed. The transfer rate from the unsprayed region to the sprayed region is two volumes of unsprayed region per hour.

Following the initiation of a severe accident, the functions of the CSS include, maintaining a low containment pressure and scrubbing fission products from the containment atmosphere. MAAP analyses demonstrate that operation of one CSS train is sufficient to provide a reasonable assurance of containment integrity.

3.9 REFERENCES

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3.2 Regulatory Guide 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis".

3.3 Regulatory Guide 1.61, "Damping Valves for Seismic Design of Nuclear Plants".

3.4 ASME Code Case N-284, "Metal Containment Shell Buckling Design Methods, Section III, Division 1, Class MC".

3.5 Regulatory Guide 1.84, "Design and Fabrication Code Case Acceptability ASME Section III Division 16".

3.6 NUREG/CR-4551, (SAND86-1309), Vol. 2, Rev. 1, Part 3, "Evaluation of Severe Accident Risks; Quantification of Major Input Parameters: Experts' Determination of Structural Response Issues", Sandia National Laboratories, March 1992.

3.7 "Performance of Containment Penetrations Under Severe Accident Loadings", M. B. Parks and D. B. Clauss, Sandia National Laboratories.

3.8 EPRI NP-2953, "Hydrogen Combustion and Control Studies in Intermediate Scale", R. Torok, et. al., June 1983.

3.9 NP-3878, "large Scale Hydrogen Combustion Experiments", EPRI, October 1988.

3.10 DOE/ID-10271, "Prevention of Early Containment Failure due to High Pressure Melt Ejection and Direct Containment Heating for Advanced Light Water Reactors", J. C. carter, et. al., March 1990.

3.11 NP-6780-L, Advanced Light Water Reactor (ALWR) Utility Requirements Document, Rev. 3, EPRI-1992.

3.12 "Initial Report of MACE Test M1B", Memo, S. Additon (TENERA) to S. Sorrell (DOE-Idaho), April 14, 1992.

4.0 SEVERE ACCIDENT PHENOMENOLOGY

This section provides an overview of the severe accident methodology issues and discusses the relationship of the various postulated containment failure modes to the System 80 + PRA and the EPRI ALWR Utility Requirements Document (URD).

4.1 MECHANISMS FOR EARLY CONTAINMENT FAILURE

For purposes of the System 80+ PRA phenomenology discussion, early containment failure is defined as containment failure prior to or within 1 hour after the core debris penetrates the reactor vessel.

The above definition of early containment failure is a relative one, driven by severe accident phenomenological processes. (For the source term and risk assessments, early containment failure is driven by the severity of the potential radiological release and population evacuation concerns. In these instances early containment failure implies a containment failure within 12 hours of the severe accident initiation.) Early failures of containment are important since these events will result in reduced warning times for initiating off-site protective measures and reduced time available for the decay and deposition of radioactive materials within the containment. The mechanisms for producing early containment failure cover a range of phenomenological processes. Potential early containment failure modes include containment overpressurization due to direct containment heating, hydrogen combustion and steam generation and containment structural failure due to missile generation, cavity overpressure, and corium debris impact on the containment steel shell.

In designing System 80+, several containment/cavity enhancements (see Section 3.0) were made to existing PWR designs to minimize the risk of early containment failure. This was typically accomplished by developing the System 80+ design in accordance with EPRI/Utility Requirements Document (URD) for the Evolutionary PWR (Reference 4.1)

The following sections provide a summary overview of the early containment failure severe accident challenges, the associated phenomenological issues and the impact of these challenges on System 80+.

4.1.1 DIRECT CONTAINMENT HEATING

During certain severe reactor accidents, such as those initiated by station blackout or a small-break LOCA, degradation of the reactor core can take place while the reactor coolant system remains pressurized. If unmitigated, core materials will melt and relocate to the lower regions of the reactor pressure vessel and ultimately melt through the RPV lower head. Once the RPV is breached, fragmented core debris will be ejected from the RPV and transported directly to the containment atmosphere. During the ejection process, metallic constituents of the ejected material, principally zirconium and steel, exothermically react with oxygen and steam to generate chemical energy and hydrogen. Concomitant with the High Pressure Melt Ejection (HPME) process, there is the potential for hydrogen combustion and vaporization of available water. The sensible heat loss to the containment atmosphere and its associated features are typically referred to as "Direct Containment Heating". By directly transferring large quantities of sensible energy from the corium and corium-steam reactions into the containment atmosphere, the containment may pressurize to the level where failure is possible. Since the containment threat is typically associated with vessel breach (VB) high containment radiation releases will accompany any containment failure. This issue is of considerable concern to the present design of PWRs. Consequently, mitigation features have been factored into EPRI Utility Requirements Document (Reference 4.1) and the NRC advanced reactor licensing basis (Reference 4.2). A detailed discussion of DCH and HPME aspects of vessel breach associated with the evolutionary ALWR design is presented in Reference 4.3. The application of this information to System 80+ is discussed below.

4.1.1.1 Description of Phenomena

The containment loads following an HPME event can be loosely combined under the heading of "Direct Containment Heating" (DCH). DCH loads arise from the addition of mass and energy into the containment via several sources:

1. Blowdown of reactor coolant system steam and hydrogen inventory into the containment.

2. Combustion of hydrogen released prior to and during the High Pressure Melt Ejection (HPME)

3. Interactions between molten core debris and water on the containment floor.

4. Transfer of sensible heat from the debris to the containment atmosphere.

As can be seen from the above, the DCH containment threat is synergistic in nature, typically involving several of the above processes. Factors affecting the above processes are discussed below.

4.1.1.2 Parameters Affecting DCH

The magnitude and the occurrence of DCH is influenced by both phenomenological factors and the plant geometrical layout.

4.1.1.2.1 Phenomenological Aspects of DCH

4.1.1.2.1.1 Reactor Pressure at Time of Melt-Through

The ability of gases flowing from the reactor vessel breach to fragment, entrain, and "sweep out" a significant amount of core debris from the reactor cavity into the lower compartment containment volume is dependent upon gas velocity which is functionally dependent upon the reactor pressure at the time of failure.

In general, the dispersal function follows an "s" shaped entrainment curve with three distinct regions. At low RCS pressures, the entrainment curve is characterized by low pressure entrainment cutoff. At RV failure pressures below this level, debris entrainment will not occur. At pressures greater than this threshold, the entrainment function monotonically increases with the RV failure pressure. Above RV failure pressures of 600 to 800 psia, the entrainment from the reactor cavity is relatively constant. Analyses and experiments have been performed on debris sweepout to establish details of the debris entrainment function and the RV failure low pressure threshold, in particular. This sweepout threshold pressure is a function of the failure area at vessel breach and the cavity configuration. For typical cavity configurations, a threshold pressure below 250 psi (see for example, References 4.3 and 4.34) will preclude entrainment of core debris into the upper compartment. In a similar assessment BNL estimated that the threshold vessel breach pressure can be as high as 350 psia (Reference 4.8). Thus, DCH can be significantly mitigated by taking steps to depressurize the RCS, or via induced RCS piping or SG tube creep failures that may occur during the core melt progression (See Reference 4.5).

4.1.1.2.1.2 Debris Mass Released at Reactor Failure

Figure 4.1-1 presents bounding calculations illustrating the influence of the ejected debris mass at VB on the post HPME containment pressure (Reference 4.3) for an evolutionary ALWR. These analyses indicate that to challenge a System 80+ type containment by DCH, a HPME event must involve a mass of finely fragmented debris equivalent to at least 50% of the total core inventory injected into a dry cavity. If the amount of debris that is available for release to the reactor cavity is not a large fraction of the core mass or if significant quantity of water initially resides in the reactor cavity, a containment challenging DCH event could not occur.

It should be noted that the DCH calculations presented in Figure 4.1-1 conservatively assume (1) 100% efficient energy transfer from the debris to the containment atmosphere, (2) complete burning of all hydrogen produced to the point of RV breach and during HPME and, (3) an initial molten debris temperature of 2533 ° K. More realistic assumptions regarding heat exchange efficiency and hydrogen combustion would result in lower predicted cavity pressurizations.

4.1.1.2.1.3 Debris Fragmentation

Very finely fragmented debris allows very efficient means of energy exchange with the environment. If the debris particle is large, both the heat transfer to the surrounding gases and the rate of chemical reactions will be relatively slow and insufficient to cause a DCH threat. HPME experiments suggest corium debris will fragment into 0.1 to 10 mm particles. These particle sizes will allow efficient heat transfer and particulate oxidation (References 4.9 and 4.34).

4.1.1.2.1.4 Chemical Reaction Kinetics

Chemical reactions can occur with unoxidized metals in the discharged debris. If steam is available and is well mixed with finely fragmented corium, oxidation processes will occur producing a large amount of hydrogen. This hydrogen could in turn burn, further contributing to the DCH pressure buildup. If oxygen remains in the cavity and the debris is mixed with it, the metals will oxidize, producing a large amount of energy without producing hydrogen.

4.1.1.2.1.5 Containment Transport and Mixing

Effective mixing in the containment atmosphere is essential to sustaining a DCH event that is containment threatening. Mixing is typically good in open containments and poor in containments with significant subcompartmentalization. However, even partial subcompartmentalization such as that afforded by the crane walls and operating decks found in typical large dry PWR containments will be effective in reducing the fraction of the containment atmosphere that can efficiently mix with the corium debris (see also Section 4.1.1.2.2) and therefore can be a "major mitigator of DCH" (See Reference 4.34).

4.1.1.2.1.6 Water Availability

The impact of the availability of water within the ALWR reactor cavity at the time of RV breach has been investigated in Reference 4.3. This study concludes that DCH loads would be mitigated provided the reactor cavity is "wet". (For this analysis it implies a "wet cavity" contains 60,000 gallons of water.) Figure 4.1-1 indicates that even with 100% dispersal of the "in vessel" corium into containment, containment pressures will be less than 0.8 MPa (116 psia) if the cavity is "wet".

An additional issue associated with HPME of corium into a water filled cavity involves the potential for "ex-vessel" steam explosions. Since an "ex-vessel" steam explosion would essentially quench the corium debris, this issue will be treated separately from DCH. A discussion of "ex-vessel" steam explosions is presented in Section 4.1.2.2.

4.1.1.2.2 Physical/Geometrical Features Influencing DCH

The presence of structures within or just outside the reactor cavity can help de-entrain debris previously entrained by the HPME process. This feature has been demonstrated by several corium stimulant dispersal experiments and DCH tests performed in the SNL Surtsey and HIPS Test Facilities. The factors which govern the effectiveness of the geometrical cavity de-entrainment features include abrupt area changes, small flow turning radius, low velocity recirculation regions and intervening structures normal to the debris flight path.

In System 80+ the de-entrainment process results in the retention of large amounts of corium debris in the reactor cavity and a significantly smaller amount of debris in the lower containment region below the containment operating deck. Restricting the corium within these regions will significantly reduce the ability of the corium to efficiently mix and release its energy directly to the containment atmosphere. The presence of intervening structures in reducing corium-air mixing has been observed in large scale DCH experiments to significantly reduce DCH loadings even when complete debris dispersal from the reactor cavity is expected (See References 4.10 thru 4.12).

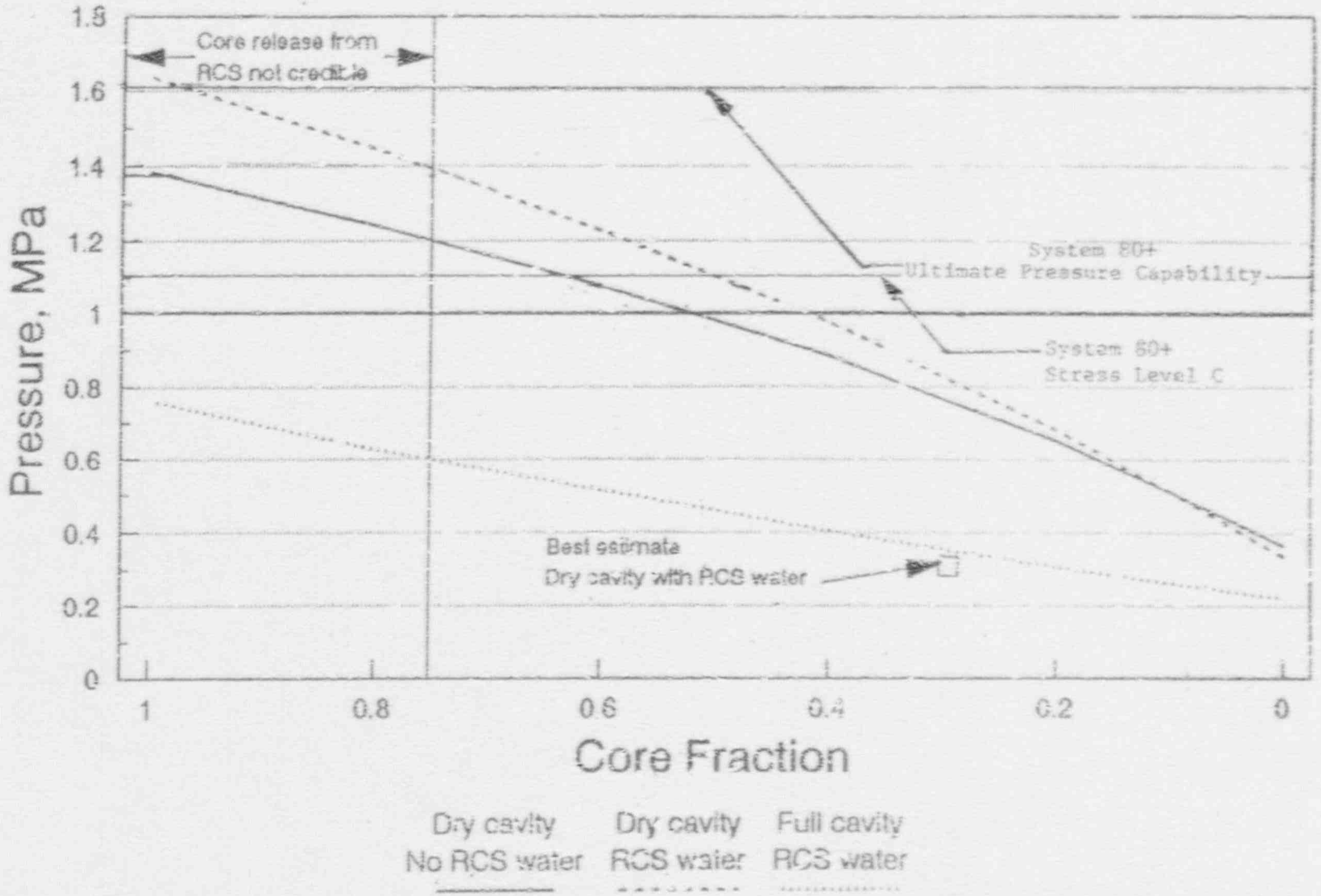


FIGURE 4.1-1: **BOUNDING ESTIMATES OF ALWR CONTAINMENT RESPONSES TO DCH**

4.1.1.3 Induced RCS Depressurization

As noted in Section 4.1.1.2.1.1 depressurization of the RCS prior to reactor vessel lower head failure can substantially reduce the DCH challenge to the containment. References 4.3 and 4.4 have indicated that the core uncovery and heatup process can potentially induce natural circulation flows within the RCS that can heat up the primary system hot leg, pressurizer surge line and/ or the steam generator tubes to the point where a credible creep failure of one or more of these components will occur prior to failure of the RV lower head. This position is supported by INEL structural integrity tests which demonstrate that at an average wall temperature of 1000 ° K, a hot leg is likely to fail in a few minutes time due to creep rupture. This issue was studied as part of the expert elicitation to NUREG-1150 "In-Vessel Issues" (Reference 4.6) for several representative PWR NSSSs.

Based on the expert elicitation on these issues, the following conclusions were reached regarding hot leg/surge line failure:

1. For conditions where (1) the RCS pressure is near a PORV/SRV setpoint, (2) steam generators are dry, and (3) no significant prolonged forced flows exist in the RCS during core melting, the likelihood of an induced hot leg/ surge line failure has a median frequency greater than 0.95.

2. Intermediate pressure core melt transients such as those associated with an unmitigated small LOCA with a total Loss of feedwater were not likely candidates for an induced hot leg/ surge line failure. The potential for hot leg/ surge line failure was estimated in Reference 4.6 to be only 0.13.

3. For transients where steam generators serve as an effective heat sink, the potential for induced hot leg failure was estimated to be zero.

4. Once the RCS undergoes an induced pipe failure, the potential for the RCS pressure to fall below 200 psia prior to RV lower head failure was 0.65 (See Reference 4.3).

While considered significantly less likely than induced hot leg failure, induced SGTRs were considered possible. Typically, the NUREG-1150 expert elicitation panel considered that for a station blackout, induced SGTRs would occur prior to hot leg failure only 1.5 % of the time. Induced steam generator tube ruptures are of potential concern because of the possibility of a highly energetic and radioactive containment bypass condition. It should also be noted that induced SGTR is possible for only those transients where the steam generator is allowed to dry out. Analyses presented in Reference 4.7 indicate that about 9 ft of water in the steam generator secondary side would be able to desuperheat the hot gases exiting the core to levels where creep failure of the steam generator tubes is no longer credible.

4.1.1.4 Summary of Experimental Evidence

4.1.1.4.1 DCH and HPME

Experiments contributing to the current understanding of DCH have been performed at Sandia National Laboratories (SNL), Argonne National Laboratory (ANL), Brookhaven National Laboratory (BNL), and Fauske and Associates Inc.(FAI) in the United States and at Winfrith in the U.K. A summary of these experimental efforts is presented in References 4.3 and 4.4. This information is summarized in Tables 4.1-1a and 4.1-1b. These tests investigated debris dispersal and DCH in test facilities with scale variations from smaller than 1:100 up to 1:10. It should also be noted that the vast majority of these experiments studied the highly dispersive Zion reactor cavity design. However, several different reactor cavity designs were experimentally investigated, particularly with unheated debris and at the smaller scales.

While these results could not be considered prototypic of System 80+ (see Reference 4.3) both due to the corium stimulant and the differing cavity design, these tests do provide considerable insight into the mechanical and thermodynamic processes associated with HPME and DCH process and in particular the role of geometry in limiting debris dispersal and heat transfer.

Key test observations were as follows:

1. Debris dispersal from the reactor cavity and expulsion into the upper containment can be significantly limited by judicious placement of obstructions within and just above the cavity exit.

2. RCS debris dispersal pressure thresholds were clearly observed. For modest RV failure sizes (typical of an instrument tube failure) threshold pressures for debris dispersal can exceed 600 psia for non-dispersive geometries similar to the Watts Bar cavity. For large lower head failures threshold pressures for debris dispersal were between 150 to 350 psia, depending on reactor cavity design.

3. Cavity offset areas, or low velocity regions within the reactor cavity could efficiently collect and retain corium debris (See Reference 4.3).

4. Heated debris dispersal tests indicate that the energy exchange efficiency between the corium debris and containment atmosphere was on the order of 30% at large scale conditions in a dry cavity. DCH experiments involving debris dispersal into wet cavities indicate that the energy exchange between the corium debris and the containment atmosphere is well below 10%.

5. The degree of subcompartmentalization within the containment will have a direct influence on DCH induced pressures. Results of the Limited Flight Path (LFP) (Reference 4.34) and Integrated Effects Tests (IETs) (Reference 4.13) performed at the SNL Surtsey facility indicate that even at levels of subcompartmentalization typical of large dry PWRs, DCH

induced pressure load contribution will be limited (typically less than several atmospheres).

6. In the presence of steam, unoxidized melt material released during a HPME can react to produce metal oxides and significant quantities of hydrogen.

These observations strongly suggest that given the importance of intervening structures and mixing processes on DCH, DCH induced containment failure for a System 80+ containment is highly unlikely. Quantitative estimates of the System 80+ de-entrainment features are discussed in Section 4.1.1.5.2.

TABLE 4.1-1a

SUMMARY OF LOW TEMPERATURE DEBRIS DISPERSAL STIMULANT EXPERIMENTS

LABORATORY	PLANT SIMULATION	SCALE	DESCRIPTION
ANL	ZION		These tests were performed in support of the IDCOR Reference design effort. The low-temperature experiment investigated the effects of containment geometry outside the reactor cavity on debris dispersal of corium into the Zion containment. In these tests the configuration of the Zion reactor cavity and instrument tunnel were mocked up along with the seal table and the biological shield in the lower containment compartment inside of the crane wall. Wood's metal was chosen to simulate the corium. High speed movies of the cavity "sweepout" process showed that a large fraction of the debris is initially transported as a large wave moving along the instrument tunnel surface towards the containment.
SANDIA	ZION	1:10	Debris dispersal tests were conducted with and without internal structures. Water was used to simulate the corium debris and both air and helium were used to simulate the reactor vessel blowdown. These experiments provided experimental data on the entrainment threshold and entrainment fraction and cavity flow distribution. Without internal structures the Zion cavity was found to be highly dispersive. The presence of structures in the cavity was noted to interfere with debris entrainment. Flow mapping within the reactor cavity revealed that a high velocity gas boundary layer is formed along the bottom of the reactor cavity. The dominant sweepout mechanism for these experiment was due to a film entrainment.
BNL	ZION SURRY WATTS BAR	1:42	Tests were performed to support NRC DCH scaling methodology assessment program. Three "representative" reactor cavities were investigated: Zion (IDCOR Type A Cavity), Surry (IDCOR Type D Cavity) and Watts Bar (IDCOR Type C Cavity). The corium was simulated by both water and Wood's metal. The driving fluid for the tests consisted of Nitrogen and Helium. For each cavity type a characteristic entrainment function was developed and basic information regarding the entrainment process was observed. Zion experiments largely confirmed findings of larger scale melt dispersion and DCH experiments. That is that the Zion cavity is highly dispersive.

TABLE 4.1-1a (Cont'd)

SUMMARY OF LOW TEMPERATURE DEBRIS DISPERSAL STIMULANT EXPERIMENTS

LABORATORY	PLANT SIMULATION	SCALE	DESCRIPTION
WINFRITH	SIZEWELL	1:25	Stimulant dispersal tests were performed using a 1:25 scaled mock-up of the Sizewell reactor cavity. Debris dispersal simulations were conducted with water (as the "corium" stimulant) and air or helium as the blowdown fluid. Results of the scaled Sizewell tests were comparable to those obtained by Sandia for the 1:10 Zion cavity blowdown simulations.
WINFRITH	SIZEWELL	1:25	Winfrith performed a series of steady flow gas-corium dispersal experiments. Both air and helium were used as the driving fluid and five similar debris fluids were employed. The experiments were initiated with debris fully spread on the cavity floor prior to initiating the entraining gas flow. Experiments were typically conducted under quasi-steady flow conditions for about 10 seconds. Results of these experiments suggested that the entrainment criterion be dependent upon the Euler Number (Eu)
FAI	ZION/ MILLSTONE 3	1:50 to 1:100	Pressurized Wood's metal injection into a series of experimental models simulating ZION and Millstone 3 cavities including cavity obstructions. Tests indicated that lower cavity and lower compartment design can be used to minimize the dispersal of debris to the upper compartment during HPME events.
WINFRITH	PARAMETRIC	1:132 to 1:21	Winfrith conducted a series of equivalent steady flow circular geometry entrainment/dispersal experiments at scales ranging from 1:132 to 1:21. The purpose of these experiments was to help identify the appropriate scaling parameters and dimensionless variables needed to define entrainment and debris dispersal processes.

TABLE 4.1-1b

SUMMARY OF DCH/HPME EXPERIMENTS

LABORATORY	TEST SERIES	PLANT SIMULATION	SCALE	DESCRIPTION/RESULTS
SNL	JETA-B	---	---	Iron-thermite tests to investigate high pressure discharge of hydrogen type saturated molten debris. Tests considered non-prototypical of severe accident reactor performance.
SNL	HIPS HIGH PRESSURE STEAMING TESTS	ZION	SPIT SERIES (1:20) HIPS SERIES (1:10)	SPITS consisted of high pressure scoping experiments. Tests were conducted with and without simulated reactor cavities. HIPS test series simulated HPME into ZION-like cavities with and without internal structures. Test indicated that ZION-like cavity is ineffective in holding up corium debris.
SNL	SURTSEY: DCH TEST	ZION	1:10	Tests injected 20 to 80 kg masses of iron thermite in a reactor cavity/containment structure. Larger scale melt ejection tests indicated energy exchange efficiencies between the debris and the atmosphere to be about 30 %.
ANL	CWTI CORIUM WATER THERMAL INTERACTION	ZION	1:30	Tests indicated that core debris could be effectively removed by structures in the lower compartment.
SNL	SURTSEY: LFP (LIMITED FLIGHT PATH)	SURRY	1:10	Tests studied the effects of compartmentalization in the containment, by placing a concrete slab in the path of dispersing debris. The presence of a concrete slab was effective in de-entraining corium debris, and reducing DCH loading.
SNL	SURTSEY: IET (INTEGRAL EFFECTS TEST)	ZION	1:10	Integrated effects test investigated the effect of simulated subcompartment structures on DCH. Lower compartment structures were noted to significantly de-entrain debris and substantially reduce containment loadings due to DCH.

4.1.1.5 Significance of DCH to System 80+

As discussed in Section 3.6, the System 80+ has been specifically designed to mitigate containment threats from HPME and DCH processes. System 80+ mitigation features principally arise from the availability of a safety grade Safety Depressurization System (SDS) to reduce RCS pressure to below the debris entrainment threshold and a debris retentive cavity configuration with sharp turns, overhangs, flowpath offsets and a "cavity trap", and a convoluted steam discharge path to de-entrain and retain core debris while allowing adequate steam relief.

In addition to the above features, the System 80+ cavity is designed to be floodable by the operator prior to RV lower head failure. As indicated in the NRC Draft Final Safety Evaluation Report on the EPRI Utility Requirements Document for the Evolutionary LWR, the ability to quench corium debris leaving the reactor vessel will ensure that the containment threat resulting from the HPME is minimal (Reference 4.52).

4.1.1.5.1 Analyses to Support DCH mitigation via the SDS

Analyses performed to support the design of the SDS have been performed at ABB and by FAI in support of the ALWR URD (Reference 4.3). FAI analyses demonstrated that if the operator takes action to depressurize the RCS before the time the core exit temperature reaches 1200 ° F the RCS can be successfully depressurized to below 250 psig prior to RV breach. Similar analyses were performed by ABB using a System 80+ version of the MAAP 3.0B code. These analyses confirmed that following a Total Loss of Feedwater (TLOFW) event the opening of the safety depressurization valves for more than two hours after the PSVs have actuated can successfully depressurize the RCS from 2500 psia to 250 psig (see Section 3.5.3). Failure of the RV at the 250 psig pressure level will not result in a credible DCH threat to containment.

4.1.1.5.2 Quantitative Estimate of Debris Entrainment

The de-entrainment potential of the corium debris has been approximately estimated via use of semi-analytical models developed from the HIPS experiments (See References 4.13). Results of these experiments were used to correlate the fraction of variously sized debris unable to negotiate a right angle turn when entrained within a low density gas flow. Reference 4.3 applied this model to estimate an upper bound for the upper compartment entrainment resulting from a high (2500 psia) pressure failure of the RV lower head. Even if all particles that negotiate the turn over the lid of the cavity debris chamber will be carried into the lower compartment, it can be concluded that less than 10 % of the corium dispersed to the cavity will reach the upper compartment. The remainder of the corium will be retained in the "cavity trap". In actuality, the small amount of debris that was not directly deposited in the debris trap will collect beneath various cavity overhangs or will be de-entrained in the cavity HVAC ventilation room prior to ever reaching the lower compartment. Any debris that enters the containment operating area will likely impinge upon the lower portion of the crane wall and deposit within the lower compartment (see Section 3.6). Significant transport of debris into the upper containment region is not expected.

At the level of entrainment anticipated for System 80+, DCH and ancillary processes pose a vanishingly small threat to containment. In fact, conservative estimates of HPME induced pressure loadings would yield peak containment pressures of between 40 and 90 psia (See Figure 4.1-1).

4.1.1.6 Application to the PRA

Based on the aforementioned phenomenological analyses and experimental information, DCH processes as it reflects on the System 80+ containment failure mechanism will be as follows:

(1) **High pressure transients with successful operation of SDS or low pressure core melt scenarios.**

For high pressure transients where the SDS is actuated in a timely manner or where RV failure follows low pressure RCS transients, the potential of a DCH induced containment failure will be taken as 0.0. That is, DCH induced containment failures from low RV failures under low RCS pressures is not deemed credible.

(Early containment threats following low pressure RV failure are considered primarily the result of a hydrogen combustion or a steam explosion; see Section 4.1.3 and 4.1.2.2, respectively.)

(2) **High-Intermediate pressure transient depressurization and the SDS is not actuated.**

Operator actuation of the SDS prior to RV failure will be included within the System 80+ Accident Management Guidelines (AMGs) and appropriate operator training on the use of this system is planned. Consequently, there is a high probability that the SDS will be actuated in a timely manner so that a DCH threat may be averted. However, for high pressure severe accident scenarios where the SDS is either unavailable or not activated the probability of an induced RCS depressurization caused by hot leg failure is assumed to be 0.50. This value is conservatively selected below the RCS hot leg failure probability of 0.95 established in Reference 4.15 . The lower value reflects the fact that a detailed analysis of this failure mechanism has not been performed on a plant specific basis for System 80+. This position is generally consistent with the ARSAP approach of Reference 4.3.

The probability of an induced RCS failure occurring as a result of a SG tube rupture in a dry steam generator (allowing potential fission product containment bypass) is estimated to be 0.015. This estimate is based on a review of NUREG-1150 expert elicitations for RV failure phenomena in Zion and Surry.

For intermediate pressure transients, the potential for an induced RCS hot leg failure will be conservatively neglected (See Section 4.1.1.3).

(3) **DCH overpressure potential following high / intermediate pressure failure**

Given a high pressure RV lower head failure occurs into a "dry" reactor cavity, the containment failure probability is conservatively based on a bounding discharge of 50% of core inventory as finely fragmented debris into the upper compartment. This results in a potential containment pressure of about 160 psia immediately following RV breach. A pressure spike of this magnitude will exceed ASME Service Level C limits and be conservatively assigned a 0.03 probability of causing a containment failure (See Figure 3.1-3).

Note that in the phenomenological assessment, it was concluded that less than 10% of the ejected debris could reach the upper compartment to be available for the DCH process. DCH processes with this level of corium loading would produce containment pressures only marginally greater than containment design strength and therefore would not pose a containment threat.

If an HPME occurs into a fully flooded reactor cavity, containment pressurization will be well below the containment failure threshold. A 0.001 probability is assigned to this event for purposes of performing sensitivity studies. The potential consequences of corium quenching on cavity pressure loadings, potential steam explosion behaviors and HPME induced missiles are discussed in later sections.

4.1.2 RAPID STEAM GENERATION

4.1.2.1 In-Vessel Steam Explosions (IVSEs)

4.1.2.1.1 Description of Phenomena

The concept of a fuel induced "steam explosion" within the reactor pressure vessel refers to a phenomenon in which molten fuel rapidly fragments and transfers its energy to the coolant resulting in steam generation, the development of shock waves and the acceleration of large RV internal masses with possible mechanical damage and failure of the RV. As a consequence of such explosions, there was a concern that missiles would be generated that might contact and locally penetrate the containment and allow for early radiation release to the environment. This containment failure mode was initially considered in the Reactor Safety Study (WASH-1400) as the alpha-mode failure. Recent assessments of steam explosion phenomena have suggested that IVSE "do not provide a credible threat to the integrity of either the primary system or containment"(Reference 4.5).

4.1.2.1.2 Parameters Affecting In Vessel Steam Explosions (IVSE)

For a steam explosion to produce a threat to the containment, the interaction process must have the following:

1. sufficient corium mass
2. favorable geometrical configuration
3. high energy conversion
4. triggering mechanism
5. production of a sufficiently energetic missile

These issues were investigated by the Steam Explosion Review Group (SERG) as they apply to steam explosions within the RV (Reference 4.14). Most members of the review group believed IVSE could occur but that the probability of producing a containment threatening IVSE was on the order of 10^{-3} (See Table 4.1-2). This conclusion was reached despite the expression of differing opinions on the basic steam explosion phenomenology (Reference 4.5). This failure potential assessment is an order of magnitude lower than that used for WASH-1400. In the NRC staff opinion expressed in Reference (4.15) the estimated probability of steam explosion has an upper limit of 0.01 and a mean of a considerably lower value.

Separately, a critical review of steam explosion phenomena and the consequences on containment loading was discussed in Reference 4.16. In that report, the components of an IVSE leading to containment failure were decomposed and reviewed in detail. This evaluation lead to the conclusion that the conditional probability of a steam explosion causing containment failure given a molten corium condition within the RV was on the order of 10^{-5}. Similar event decomposition methods were employed by Theofanous, et. al. in estimating the potential for low pressure steam explosion induced conditional containment failure (Reference 4.17). Such a failure was estimated to have a probability on the order of 10^{-4} and therefore expected to be physically unreasonable to occur.

A detailed investigation of the "in vessel" steam explosion issue was also considered within Phase B of the German Risk Study (Reference 4.18). This study assumed a steam explosion with 10,000 Kg of corium participating with a relatively high thermal to mechanical energy conversion efficiency of 10%. Even using these conservative assumptions, the resulting loadings would be insufficient to fail the RV. As a result of this study the GRS "ruled out" the possibility of destroying both the RV and the containment as a "risk relevant accident pathway".

In practice, the expert judgement developed for Reference 4.14 was incorporated into the NUREG-1150 Reference plant Surry PRA with a mean probability of this event (VB-ALPHA) to result in containment failure to be 0.0085 when the RCS is at a low pressure and an order of magnitude lower when the RCS meltdown occurs at high a pressure (Reference 4.19).

4.1.2.1.3 Summary of Experimental Evidence

Large scale steam explosion experiments were conducted by Sandia National Laboratory in the Open Geometry and FITS experimental test series. The purpose of these tests were to experimentally study the magnitude and time characteristics of pressure pulses and identify the initial conditions necessary to trigger and propagate explosive interactions between water and various molten materials.

4.1.2.1.3.1. Sandia Open Geometry Experiments

These experiments consisted of large scale (5 to 20 kg) fuel - coolant interaction tests. The test program was scoping in nature with the primary objective of assessing the efficiency of the fuel-coolant interaction thermal to mechanical energy conversion process. Approximately sixty tests were conducted in a minimally instrumented open vessel. The fuel stimulant used was a thermitically generated iron-alumina mixture and corium. Results of the iron-alumina-water interaction tests indicated energy conversion ratios at the lower end of the observed range (0.2 to 1.5%). No energetic explosions with corium were observed.

4.1.2.1.3.2 FITS Experiments

The purpose of the FITS experiments was to determine the triggering behavior, explosion threshold and to parametrically estimate the thermal to mechanical energy conversion ratio associated with steam explosions. The FITS facility was well instrumented and the interaction chamber was closed. The FITS program lasted several years and included over 100 experiments. Fuel masses varied between 2 and 20 Kg. The majority of the FITS experiments used thermitically generated iron-alumina, and the remaining tests used thermitically generated corium.

These tests indicated that energetic fuel - coolant interactions were possible for corium. Thermal energy conversion ratios were found to be in the 0 to 3% range. Parametric studies performed as part of FITS also provided the following:

For coherent melt deliveries (that is the melt is delivered in one mass) into water at ambient temperature and pressure 32 explosions were observed out of 37 tests (Reference 4.20). Thus, the probability of a spontaneous steam explosion under these conditions can be established at 0.86. Similar experiments conducted with saturated water and at ambient pressure indicated a steam explosion probability of 0.24 (4 observed explosions out of 17 tests).

The influence of pressure on the probability of a spontaneous steam explosion was established in FITS-C (Reference 4.21). No spontaneous steam explosions were observed for all five FCI tests conducted at ambient water temperature and a pressure of 5 bars (75 psia), thus, experimentally supporting the position that high pressure steam explosions are extremely unlikely events.

Additional experiments were conducted to ascertain the importance of melt delivery on the steam explosion process. While under certain circumstances, coherent melt deliveries were observed to produce steam explosions, a pre-dispersed delivery of fuel debris was not.

4.1.2.1.3.3 Ispra High Pressure Experiments

In Reference 4.14 , Dr. Mayinger referred to a steam explosion test program performed by EURATOM to establish the influence of system pressure on steam explosions. While details of this test series are unknown, Dr. Mayinger noted that a general conclusion drawn from these experiments was that initiation of steam explosions at pressures greater than about 300 psia require very strong detonative triggers.

4.1.2.1.4 Significance of IVSE to System 80+

Based on a review of available steam explosion data and analyses, it appears that sufficient information is available to conclude that the probability of containment failure resulting from a corium-coolant interaction (CCI) event is very low (on the order of 0.001 or less). Much of these assessments considered typical PWR geometries analogous to that of System 80+ and are, therefore, considered applicable to System 80+.

4.1.2.1.5 Applicability of IVSE to the PRA

The above information is believed to be generally applicable to the System 80+ PRA. Therefore, for the purpose of System 80+ risk assessment, the containment failure caused by an IVSE was taken to have a mean containment failure probability of 0.001.

Table 4.1-2: Summary of Subjective Containment Conditional Probability due to In-Vessel Steam Explosion

Investigator	Best Estimate	Upper Limit
Bankoff	$<10^{-4}$	
Bohl/Butler	$3x10^{-4}$	.10
Briggs	$<10^{-2}$	
Catton	$5x10^{-3}$	
Cho	WASH-1400 very conservative. Failure very unlikely	
Corradini	---------10^{-4} - 10^{-2}---------	
Cybulskis	10^{-4}	10^{-2}
Fauske	Vanishingly small (~0)	
Ginsberg	$4x10^{-3}$	$4x10^{-2}$
Mayinger	No endangerment of FRG/PWR containment	
Squarer	---------10^{-5} - 10^{-4}---------	
Theofanous	$<10^{-4}$	$<10^{-4}$
WASH-1400	10^{-2}	10^{-1}

* Table taken from Reference 4.5

4.1.2.2 Ex-Vessel Steam Explosions

4.1.2.2.1 Description of Phenomena

The discharge of molten debris into the reactor cavity region could potentially result in explosive interactions between the molten core debris and cavity water. Given water in the cavity, the initiation of an explosion, if any, would occur either when the debris initially contacts the water or when the debris penetrates the water and contacts the concrete surface at the bottom of the reactor cavity. The results of a potential s .am explosion would be to generate impulse loads on the reactor cavity walls and in-cavity structures. The resultant damage to important structures was considered negligible for the PWR cavity designs analyzed in NUREG-1150 (see for example Appendix C of Reference 4.19).

4.1.2.2.2 Parameters affecting EVSE

Parameters affecting ex-vessel steam explosions are essentially the same as those for IVSEs with the following exceptions. Ex-Vessel steam explosions occur exclusively at low pressure and typically the cavity geometry allows for steam relief. The details of the cavity/containment design will also influence the consequences of an EVSE event.

4.1.2.2.2.1 Containment Pressure

EVSEs will typically occur at low pressure. Results of FITS experiments at ambient temperature and pressure indicated that the probability of a spontaneous EVSE given a coherent melt subcooled water interaction was 0.86 (See Section 4.1.2.1). However, at a system pressure of only 5 bars (75 psia) spontaneous steam explosions could not be triggered by a discharge of corium into a subcooled water pool.

4.1.2.2.2.2 Cavity Water Temperature

The propensity for the development of a steam explosion was experimentally found to be dependent on the proximity of the water pool temperature to saturation. Results of FITs experiments indicate that when a melt interacts with a saturated water pool, the probability of an EVSE drops from 0.86 to about 0.25. Furthermore, steam explosions in saturated water typically occurred towards the top of the pool generating very low explosive forces within the pool.

4.1.2.2.2.3 Mass of Corium Involved in the Explosion

The short duration of the explosion process will limit the corium mass involved in the process. Estimates of corium involvement in ex-vessel steam explosions typically consider the mass of corium injected into the pool during the time interval in which corium initially enters the water pool and falls to the pool floor as the mass of corium involved in the steam explosion process. In System 80+, as with many PWRs, the corium ejection occurs primarily from the lower head instrument tube failure. Thus, the actual mass of corium expected to be involved in any one explosion is small (under 20 kg).

4.1.2.2.2.3 Vulnerability of Cavity/Containment Structure

The major concern with EVSE is the ability to cause damage to the containment either indirectly via failure of important RCS supporting structures or generation of containment threatening missiles or directly via dynamic loading of the containment. Proper location of support structures and cavity wall design can effectively eliminate the containment threatening potential of steam explosions. Previous assessments of PWRs with lower head instruments (Zion, Surry, Sequoyah) indicated that EVSEs posed a minimal threat to containment (See References 4.19, 4.21).

4.1.2.2.3 Summary of Experimental Evidence

A discussion of steam explosion experiments is provided in Subsection 4.1.2.1.3.

In addition to these experiments, steam explosions have also been observed in SANDIA HPME experiments performed as part of SPITS and HIPS test program, investigating pressurized melt ejection into the water pools. In several tests these explosions resulted in short duration (several millisecond) high amplitude pressure pulses disassembling the test apparatus. These tests were not conducted within the framework of the Severe Accident and Scaling Methodology (SASM) effort and therefore were not assessed to be prototypical of steam explosion loads within PWR cavities.

4.1.2.2.4 Significance of EVSE to System 80+

For System 80+, it is expected that the cavity flooding system will be operable and actuated prior to the reactor vessel failure. Therefore, water-corium interactions will occur following vessel breach and the possibility of "steam explosions" cannot be excluded. While steam explosions are of low consequence for reactors in general, several specific features of the System 80+ further support this conclusion. These features include:

1. Large cavity size

The total volume of the contiguous reactor cavity (including instrument chase and core debris chamber is about 32,000 ft^3. This is among the largest PWR reactor cavity volumes.

2. Relief Area

The reactor cavity has been equipped with several convoluted pathways for steam relief. These areas are ample (> 50 ft^2) to ensure appropriate cavity pressure relief following rapid pressurization.

3. High Cavity Strength

The System 80+ cavity design strength is expected to be about 225 psid. Cavity ultimate failure strengths will be considerably greater.

4. Lack of Vulnerable Critical Structures

The reactor cavity is isolated from the containment outer structure and contains no essential reactor supports in the region of the cavity expected to be flooded. Thus, while a steam explosion is credible the pressure induced shock damage on critical reactor supports or direct containment structure failure is not.

A review of the EVSE concern for the ALWR was provided in Reference 4.3 as part of the ARSAP study of ALWR HPME issues. The System 80+ cavity is designed to provide a flooded cavity in advance of RV lower head failure. Since the most likely RV failure mechanism for the ALWR will be via instrument tube failure, the extent of material release from the reactor vessel prior to explosion would be restricted to the mass of a corium jet between the RV lower head failed penetration and the concrete floor. The energetics of this type of an event were estimated in Reference 4.3 to produce localized cavity loads in the vicinity of 10 bar. Such loads are within the capability of the cavity walls. No structural elements are located in the lower portion of the reactor cavity and therefore an EVSE will not threaten RV support integrity. Since no portion of the containment wall is subjected to an in cavity EVSE, a direct containment failure due to a steam explosion is not considered credible.

4.1.2.2.5 Application of EVSE to the System 80+ PRA

While Ex-Vessel Steam Explosions can occur, EVSEs are not considered a credible threat to the System 80+ containment. However, for the purpose of completeness, an EVSE logic structure will be included in the PRA early containment failure supporting logic models with appropriate quantification of decomposed events including a containment failure probability of 0.001 conditional on the occurrence of an EVSE event.

4.1.2.3 Post-Vessel Breach Steam Spikes

4.1.2.3.1 Description of Phenomena

Steam spikes following RV breach result from the relatively rapid pressure increase within the containment produced by both (1) the discharge of high pressure water/steam from the reactor vessel and (2) generation of steam associated with the quenching of superheated core debris. These processes can provide a very rapid (occurring in several seconds or less) steam addition to the containment followed by a modest pressure spike. In general, pressure loadings resulting from this process will exceed containment design limits but will be well within ASME Service Level C limits (See Section 3.1). This containment challenge is discussed in more detail below.

4.1.2.3.2 Parameters Affecting Post-Vessel Breach Steam Spikes

In the context of the PRA, Post-Vessel Breach Steam Spikes will include both the steam released into the containment following vessel breach (VB) and the vaporization of liquid during the corium quenching process. The impact of these releases are considered along with the pressurization prior to VB to establish the rapid steam generation containment challenge. Consequently, the parameters affecting the magnitude of the post RV failure pressure spike are:

a. Containment pressure prior to VB

b. RCS conditions at VB

c. Mass and Superheat of corium ejected into the reactor cavity

d. Water availability in the reactor cavity.

4.1.2.3.2.1 Estimation of Containment Pressure at VB

The containment pressure at the time of VB is dependent on the RCS inventory discharge paths and the status of containment heat removal. In general, transients that discharge inventory through the IRWST or bypass containment will have containment pressures very near the initial containment state regardless of the availability of containment sprays. All transients that discharge into a cooled containment (sprays available) will have a containment pressure at VB 5 to 10 psi above the initial value. However, if the RCS discharges into the containment which has lost the heat removal function (sprays unavailable), containment pressures prior to vessel breach can be significant.

4.1.2.3.2.2 RCS Conditions at VB

The energy and mass associated with the RCS steam/water discharge following VB will establish the increment in containment loading due to direct mass and energy addition into the containment. This containment pressurization process is analogous to the containment pressurization following design basis pipe breaks.

4.1.2.3.2.3 Mass and Superheat of Corium Debris

Following VB, steam will be generated in the process of quenching the corium debris. In this process the stored energy from the corium is transferred to the water which in turn is vaporized. Experimental data on corium quenching indicates that the quenching process exhibits maximum heat fluxes of up to 30 Mw/m^2 for short time periods.

4.1.2.3.2.4 Availability of Water

The amount of corium that can be quenched is dependent on the availability of water. If insufficient water is available, quenching will not be complete and steam generation will be limited.

4.1.2.3.3 Significance to System 80+

The peak containment pressures resulting from rapid steam generation events following a System 80+ RV lower head breach are summarized in Table 4.1-3 for selected severe accident scenarios. These scenarios include a station blackout, a "V" sequence (interfacing systems LOCA) and a Large LOCA in containment. These events typically span the range of interest for estimating post VB rapid steam generation pressure spikes. For the first two scenarios, the initial containment pressure will be about 15 psia. The containment is subsequently pressurized by a combined high pressure steam release and steam generation due to a rapid quenching of the corium debris. For this study the corium mass quenched comprised 70% of the total core mass and was initially discharged into the containment at 2500°K, producing a 21 psi pressure spike. Appropriate energy balances performed on the containment indicate that the final containment pressure would be below 55 psia. Analogous analyses were performed for the Large Break LOCA. However, in this analysis containment sprays were not credited and the mass and energy of steam released from corium quenching at VB was negligible. As estimated previously, the corium quench steam release results in an incremental 21 psi pressure spike. Assuming the initial containment loading at VB to be at design limits (49 psig), the final containment pressure will be in the vicinity of 85 psia. While these loadings are above the design basis containment loadings, they are well below the pressure limit determined using the ASME Service Level C criterion.

4.1.2.3.4 Application to the PRA

Rapid steam generation events (or steam spikes) will not result in a significant challenge to the System 80+ containment. In fact, for scenarios where (1) primary discharge is through the IRWST , (2) the containment is bypassed or the (3) containment heat removal properly functions, the RSG pressures will be below or slightly above design pressure limits. These transients will be assumed to have no possibility of failing containment. For transients with direct steam discharge to the containment and without any containment heat removal the final containment pressure following the RSG event will be below 90 psia. This pressure level modestly exceeds the containment design limit and is well below Service Level C limits. The probability of containment failure under these circumstances has been established from an approximate containment fragility curve (See Figure 3.1-3) to also be negligible. For purposes of the PRA a failure probability of 0.001 is established for this later condition.

TABLE 4.1-3

STEAM INDUCED CONTAINMENT PRESSURE SPIKE FOR SYSTEM 80+ FOLLOWING VESSEL LOWER HEAD BREACH

	TYPICAL SCENARIO	CONTAINMENT PRESSURE FOLLOWING BREACH
1.	STATION BLACKOUT	< 55 PSIA
2.	"V" SEQUENCE LOCA	< 55 PSIA
3.	LARGE LOCA (W/O CONT. SPRAYS AVAILABLE)	85 PSIA

4.1.3 HYDROGEN COMBUSTION

The production of combustible gases (principally hydrogen) within the RCS and subsequent release to the containment following a severe accident has been noted to be a potential contributor to early containment failure for many PWR and BWR designs. Typically hydrogen combustion can influence containment failure by static (deflagration) or dynamic (detonation) overpressurization, missile generation, and equipment failure due to thermal or pressure effects.

4.1.3.1 Deflagrations

4.1.3.1.1 Description of Phenomena

A deflagration is a combustion process in which the combustion front moves at subsonic velocity with respect to the unburned gas. The pressure and temperature following a deflagration process are spatially uniform and can be conservatively bounded by the assumption of adiabatic, isochoric complete combustion (AICC). Factors that determine the type and level of combustion include the concentration of combustible gases (principally hydrogen), the concentration of the oxidant (oxygen in air) and inertents (nitrogen and steam) and the initial temperature and pressure conditions within the containment.

4.1.3.1.2 Parameters Affecting Hydrogen Combustion

4.1.3.1.2.1 Hydrogen Concentration

This section is concerned principally with the potential for a early hydrogen combustion induced failure of the containment. Other combustibles significant to severe accident progression, such as carbon monoxide are not considered in this section because they will not be available until a considerable time after VB. The concentration of hydrogen within the containment depends on (1) the amount of hydrogen produced in the RV during the early core melt, (2) how effectively the hydrogen is dispersed in the containment, (3) the threshold at which a hydrogen burn will occur, and (4) the occurrence of prior burns.

4.1.3.1.2.1.1 "In-Vessel" Hydrogen Production

During a severe accident in an LWR, significant quantities of hydrogen can be produced "in vessel" by oxidation reactions principally between the zircaloy constituents of the core and to a lesser extent the steel internal structural components and water. Assessments of the level of "in-vessel" hydrogen production were developed in support of the NUREG-1150 quantification (Reference 4.15). These assessments indicate that for PWRs, the maximum median expected level of "in-vessel" hydrogen generation is less than that due to oxidation of 70% of the zircaloy mass.

Recent assessments of zircaloy oxidation during the early stages of a severe accident progression has been established for NUREG-1150, Reference plants using RELAP/SCDAP and MELCOR/TRAC (Reference 4.15). Based on these investigations, Reference 4.4 concluded that for high pressure accident sequences, the degree of zircaloy oxidation will be in the vicinity of 50% of the available zircaloy mass. Accidents where core uncovery occurs at low pressure were observed to result in oxidation levels on the order of 60%. The higher oxidation levels occurring at lower pressures is a combined effect of the increased steaming potential associated with the passive discharge of the SITs, and larger lower head failure times due to reduced pressure loading on the lower head.

System 80+ is designed to accept and condense SDS and PSV discharge in the IRWST. Hydrogen discharge to the containment is released via several system vents and accumulates in the containment and the IRWST freeboard space. Consequently, any steam discharged to the IRWST will be condensed and steam inerting of hydrogen is not possible. A similar hydrogen ignition situation will exist for transients where the RCS fluid is discharged directly into the containment and sprays are functioning. In situations where the severe accident transient results in direct discharge of RCS fluid to the containment without containment heat removal, preburning of hydrogen will not occur in the containment due to the existence of a steam inerted condition.

4.1.3.1.2.1.2 Hydrogen Production During HPME

An HPME event may provide an efficient mechanism for unoxidized metals within the corium to mix with the cavity water or RCS steam and produce hydrogen. Hydrogen production during the HPME will be rapid and can be quite large.

4.1.3.1.2.1.3 Hydrogen Mixing within Containment

The transport and mixing of hydrogen inside containment are critical in determining the time and nature of hydrogen combustion. Rapid mixing could result in uniform distribution of hydrogen and burns that are global in nature. Slow mixing may lead to localized burning and locally detonable mixtures. The physical processes which govern the mixing in gaseous mixtures are forced convection, natural convection and diffusion. The mixing processes are affected by the rate and amount of hydrogen released into the containment and the operability of the containment heat removal systems, such as containment sprays.

Containment design is also important in establishing the potential for the development of localized high hydrogen concentrations. For typical large dry containments, the concentration variation of hydrogen throughout the containment is less than 3% (see Section 4.1.3.1.3). Should isolated containment regions exist, the localized hydrogen concentration could be quite high. Initial assessments of the hydrogen distribution within the evolutionary ALWR containment are presented in Reference 4.40 and in section 4.1.3.1.4.

4.1.3.1.2.1.4 Igniters

Operator activated igniters are included in the System 80+ design package so that in the unlikely event of a severe accident, the plant staff will have the option to burn off accumulated hydrogen in a controlled manner and at low hydrogen concentrations. Once igniters induce a hydrogen burn, that amount of hydrogen is no longer available to contribute to a large global burn, and hence the overall containment threat will be reduced. Igniter systems have been adopted by existing ice-condenser type PWR designs as a mechanism to mitigate containment threats due to hydrogen combustion (Reference 4.23).

4.1.3.1.2.2 Presence of Inertents

Inertents (such as nitrogen and steam) in the containment atmosphere, reduce the concentrations of the active combustion components and mitigate both the potential for and severity of a hydrogen burn. Of particular interest to hydrogen combustion is the availability of steam in the atmosphere. Experimental investigations on small scale facilities (see also Section 4.1.3.1.3) demonstrates that steam concentrations greater than about 56 v/o can effectively inert the containment and prevent combustion. The presence of an inert containment atmosphere early in an accident can be expected only for those severe accidents involving direct discharge of the RCS inventory to the containment without first passing through the IRWST at a time when containment sprays are unavailable. Under all other conditions, hydrogen combustion will be possible provided a sufficiently large concentration of hydrogen is available in the containment atmosphere.

4.1.3.1.2.3 Availability of Oxygen

The System 80+ PWR is designed to operate under standard atmospheric conditions. Thus, oxygen will be available for combustion.

4.1.3.1.3 Summary of Experimental Evidence

Considerable experimental work has been performed to understand the hydrogen mixing, and combustion processes. This survey provides only the most pertinent highlights of these efforts.

4.1.3.1.3.1 Hydrogen Mixing Experiments

An experimental study of hydrogen mixing and distribution has been performed at the Hanford Engineering Development Laboratory (HEDL) (Reference 4.5). A 20-m high, 7.6-m diameter vessel was used to simulate the lower compartment region of an ice-condenser containment under two different hydrogen-steam (or helium-steam) release conditions. Release locations were modeled to simulate hydrogen release from a postulated small pipe break or a pressurizer relief tank rupture disc. The results of the tests show that:

1. The compartment was well mixed during the source release period with maximum helium or hydrogen concentration differences of about 3 volume percent between points in the test compartment volume.

2. Gas entrainment caused by the high velocity jet was the dominant mixing process for the test compartment during the jet release period.

3. The test compartment was well mixed by natural convection after termination of the source gas for all cases.

4. The degree of mixing was not strongly dependent on source jet release orientation.

Additional limited scope hydrogen mixing tests were performed for a small scale mockup of a large dry containment at CEA in France using a helium-steam mixture (Reference 4.24). The results of these experiments were similar to that found by HEDL. In particular, natural convection was sufficient to mix the containment atmosphere to within 3 volume percent.

4.1.3.1.3.2 Hydrogen Combustion Experiments

There have been several experimental programs on hydrogen combustion performed in test volumes ranging in size from 0.017 to 2100 m^3. The large-scale simulation of accident environment was performed at DOE's Nevada Test Site (NTS) (Reference 4.5). The hydrogen combustion tests at NTS were conducted in a 16-m (52-ft) diameter spherical vessel whose internal volume was 2100 m^3 (74,000 ft^3).

The NTS vessel is about two orders of magnitude larger than that used in other, small-scale, experiments. One of the objectives of the NTS tests was to study hydrogen combustion behavior under simulated accident conditions in a reactor containment. Two types of tests were performed:

1) Premixed tests for simulating single burns which may occur in large open areas of a containment such as in a PWR dry containment.

2) Continuous injection tests for simulating continuous or intermittent hydrogen burning which may occur in containments with igniters.

The premixed tests were performed with hydrogen concentrations ranging from 5 to 13% and steam concentrations ranging from 5 to 40%. These conditions span the range of non-inerted hydrogen combustion conditions expected within the System 80+ containment. In the continuous injection tests, hydrogen flow rates were between 1 and 8 lbm/min and steam flow rates between 0 and 62 lbm/min. In some tests, fans and sprays were operated to simulate the plant emergency systems. The results applicable to hydrogen burn phenomena in reactor containments as summarized in Reference 4.5 are:

1) Primary combustion parameters, i.e., gas temperature, pressure, heat fluxes and burn fractions, increase with increasing hydrogen concentration.

2) Steam acts as a diluent and reduces gas temperature and pressure excursions.

3) Increasing the steam fraction in the continuous injection tests tends to inhibit combustion, resulting in a shorter burn time, smaller burn fraction and lower pressure rise.

4) Operation of fans and sprays enhances turbulence and promotes faster burn.

5) Spray operation results in lower peak pressure rise at the end of the combustion period. This is due to quenching of the gas and removal of steam by condensation.

NTS experiments did not include assessments of complete steam inerting. However, smaller scale experiments such as those discussed in Reference 4.25 clearly demonstrate steam inerting with steam environment concentrations in excess of 56%.

4.1.3.1.3.3 Igniters Effectiveness Experiments

The use of deliberate ignition strategies for controlling hydrogen in post-accident PWRs were investigated in the early 1980's. The emphasis of these tests were to (1) determine if lean mixtures of hydrogen can be reliably ignited, (2) establish what pressures are generated by deliberate ignition and (3) ascertain the effects on equipment and instrumentation caused by the temperature and pressure induced by the deliberate burn. An overview of the igniter test programs is provided below.

4.1.3.1.3.3.1 Small Scale Experiments

Several small scale test programs have been carried out to support the feasibility of deliberate ignition as a hydrogen control strategy. These tests included experiments performed by Livermore (LLNL), Sandia, Fenwall and the U.S. Bureau of Mines, AECL-Whiteshell, EPRI-ACUREX (See References 4.26, 4.21, and 4.33). The results of these experiments indicated that ignition can be initiated at about 4 % hydrogen concentration when the mixture is agitated, as by a fan cooler. Under quiescent conditions, hydrogen burns require hydrogen mole fractions closer to 8 %. In addition, these tests noted that hydrogen burns at low concentrations were inefficient. As the hydrogen concentration increased to about 9 % nearly complete combustion of available hydrogen was observed.

4.1.3.1.3.3.2 FITS Experiments (Reference 4.53)

Hydrogen combustion experiments were performed at the Sandia FITS facility. The purpose of these tests were to clearly define the combustion boundaries for a hydrogen-steam-air mixture in both quiescent and turbulent environments. These tests indicated that increasing the partial pressure of steam acts to reduce the pressure increase resulting from the burn. Specifically, the maximum pressure was observed to be between 40 and 90 % of the AICC calculated maximum pressure values for steam concentration in excess of 40 v/o. Furthermore, for hydrogen concentrations below 10 v/o the actual pressure was typically less than 50% of the AICC calculated value.

4.1.3.1.3.3.3 NTS Experiments

The effect of location on glow plug igniter performance was investigated under large scale conditions in the Nevada Test Site (NTS) tests. Igniters at four locations were examined: top, bottom, center, and test vessel equator. For the continuous injection tests, hydrogen and steam were released at locations about 6 to 8 ft. above the bottom center of the test vessel. The test results showed that:

1. Glow plugs could ignite mixtures down to 5.3 Vol% hydrogen and 4.2 Vol% steam during quiescent, bottom ignition in premixed combustion tests. Top ignition location was less effective than lower ignition locations. Flame quenching on the vessel dome could inhibit flame propagation.

2. During the continuous injection tests, the "best" igniter location appeared to be in non-stagnant regions above the hydrogen release point.

3. During the continuous injection tests, the turbulence promoted by fans and sprays caused hydrogen to be dispersed throughout the entire test vessel. Therefore, the time of ignition was delayed when the preactivated igniter was located in the upper portion of the vessel.

4.1.3.1.3.4 Equipment Survivability

The ability of critical equipment to survive a hydrogen burn was also investigated within the NTS program. In these experiments, selected equipment consisting of pressure and temperature measuring instruments, valves, switches, fan motors, containment penetrations, glow plug igniters and cables were subjected to pre-mixed hydrogen burns with test volumetric concentrations up to 13% hydrogen and 30% steam. All equipment was monitored for operability before, during, and after each burn test. The test results showed that most of the equipment operated successfully even for the most adverse burn condition exhibiting a peak temperature of 1155°C). No degraded operability was observed in 96.3% of the checks during the test and 99.6% of the post-test checks.

4.1.3.1.4 Significance of Early Hydrogen Burn to System 80+

Reference 4.1 contains a set of design requirements for ALWRs which are designed to limit the threat to containment integrity from a post-severe accident hydrogen combustion event. The ALWR hydrogen control guidance initially required ensuring that "... the hydrogen gas concentration in containment does not exceed 13 % under dry conditions for an amount of hydrogen equivalent to that generated by oxidation of 75% of the active fuel cladding surrounding the active fuel." In a later revision to this guideline the hydrogen gas concentration requirement was reduced to 10% and the equivalent oxidation level was increased to 100% of active fuel clad. In the current System 80+ design the containment is sufficiently large so that oxidation of 75 % of the active fuel clad will result in a maximum hydrogen concentration below 12 v/O. The corresponding oxidation of 100% active fuel clad equivalent will result in a maximum global average hydrogen concentration of below 15 volume percent in dry air.

The significance of early hydrogen burns on the evolutionary ALWR is discussed in Reference 4.40. These studies evaluated the ALWR combustion potential based on complete combustion of an initial concentration of hydrogen of 13 v/o in dry air. In these studies the steam was incrementally added to the containment atmosphere and the resultant mixture was assumed to undergo AICC. The post-burn containment pressure trajectory as a function of steam concentration is presented in Figure 4.1-2. In these analyses at steam concentrations greater than about 45 % the hydrogen/steam/air mixture was assessed to be below the flammability limit. These analyses predicted that the most probable post burn pressure would be about 93 psia, with the maximum possible pressure below 98 psia. Artificially extending the flammability limit to 55% steam concentration, the maximum burn pressure, assuming AICC, would be 104 psia. Typical peak burn pressures for various zircaloy oxidation levels are presented in Table 4.1-4. It should be noted that even for 100 % Metal Water Reaction (MWR), complete AICC hydrogen burns result in peak containment pressures of about 140 psia. This value is below ASME Service Level C limits of about 160 psia and are well below the containment ultimate failure pressure of 235 psia (See Section 3.1).

Thus, based on the above assessments, it can be concluded that provided a hydrogen burn of sufficient magnitude to damage containment early in a severe accident sequence is highly unlikely. The impact of nonhomogeneous hydrogen distributions were established for the ALWR in Reference 4.40. In this analysis a nodal representation of the ALWR containment was subjected to a forced hydrogen production representing an equivalent 75% clad oxidation during a Station Blackout (SBO). This analysis indicated that the vented IRWST hydrogen concentrations are only 2 v/o greater than the overall containment concentrations. Thus, any inhomogeneity in hydrogen gas concentration will not alter conclusions with regard to hydrogen.

4.1.3.1.5 Application to the PRA

In order to establish the probability of a hydrogen deflagration induced containment failure associated with various Plant Damage States (PDSs), the following assumptions are made in the PRA with regard to hydrogen availability and ignition conditions at or prior to vessel breach.

(1) Hydrogen Generation

Based on Section 4.1.3.1.2.1.1 it is assumed that hydrogen production within the RCS will be as follows:

- for all scenarios where core recovery cannot be accomplished (even temporarily) the maximum hydrogen concentration is bounded by 50% zirconium oxidation.

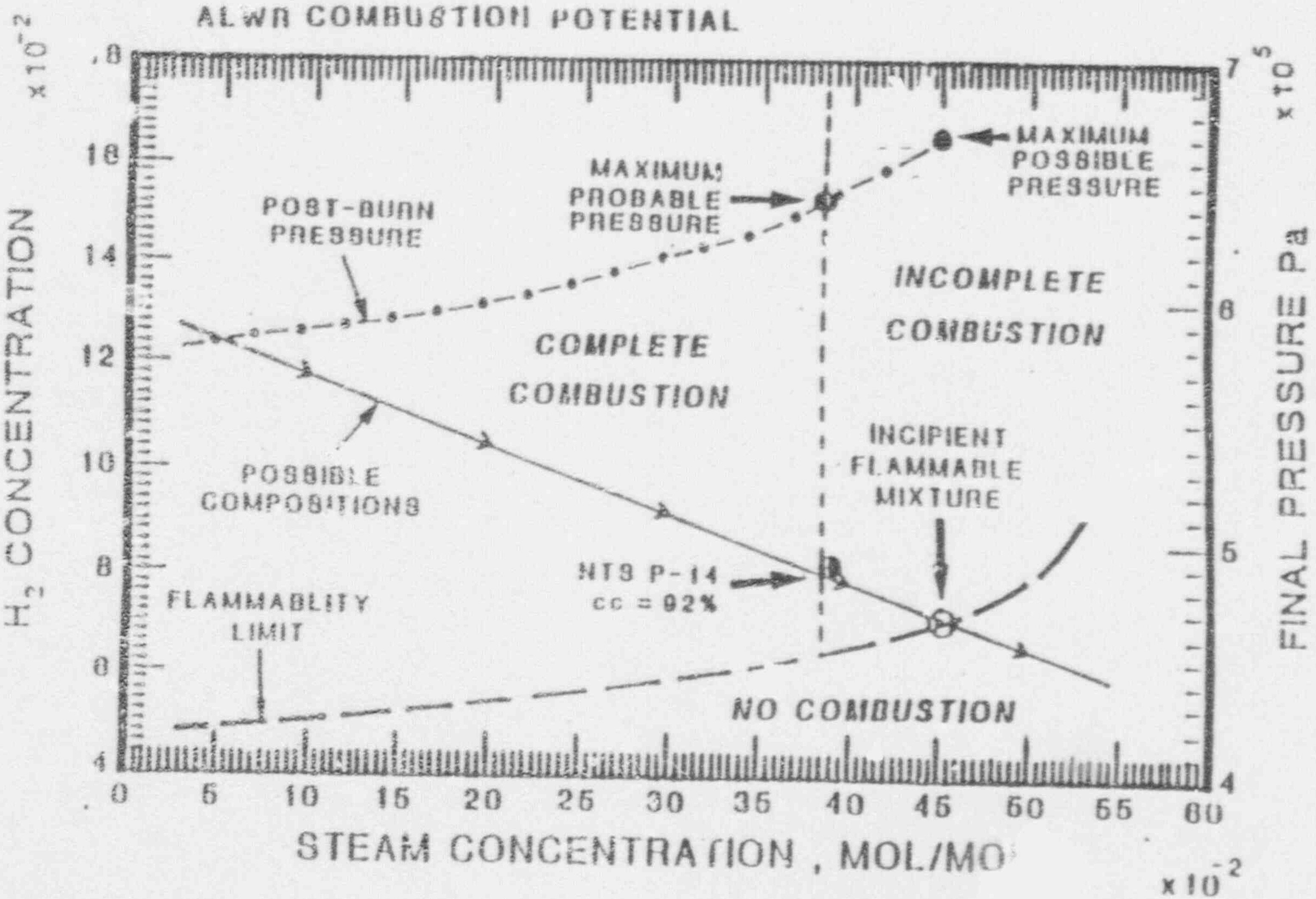


FIGURE 4.1-2: ALWR Combustion Potential.

- for all scenarios where core recovery is temporarily accomplished, a maximum of 75% MWR is assumed.

The selection of these values should conservatively bound expected hydrogen production rates prior to vessel breach.

This approach is generally consistent with the NRC position expressed in the Draft URD Final Safety Evaluation Report which states that, "a 75 percent-equivalent cladding reaction .. is a reasonable design basis for hydrogen generation for severe accidents in which the reactor pressure vessel remains intact".

It should also be noted that ISLOCAs and SGTRs that could potentially lead to core damage will not release significant quantities of hydrogen into containment and therefore do not have the potential to develop a containment threatening hydrogen burn upon vessel breach.

(2) Prior Burns/ Igniter Actuation

Hydrogen burns prior to vessel breach are only considered if igniter actuation has been credited. Igniters are assumed to burn off hydrogen as quickly as it is created. Thus, if hydrogen igniters are actuated and properly function, no containment threatening hydrogen burn can occur.

Since guidance and procedures for employing igniters is not yet available it will be assumed that for transients with sustained core uncovery the operator has a 0.75 probability of engaging igniters once a sustained core uncovery is confirmed. Igniter burns should produce pressure spikes less than that associated with a 50% core wide oxidation.

(3) Post Burn Pressures

Post burn containment pressures depend upon the following factors:

1. RCS steam discharge path prior to VB

2. IRWST subcooling

3. Availability of Containment Heat Removal Prior to Burn

For transients that discharge steam into the IRWST, or if the RCS inventory is discharged directly into the containment and the containment heat removal (CHR) is partially functioning (at least one containment spray pump and an associated heat exchanger), the hydrogen burn will be assumed to be initiated from a 30 psia base pressure.

Transients that result in significant steam discharge into the containment without availability of CHR will be assumed to have an inert hydrogen mixture and will not produce a containment threatening burn.

These PRA assumptions are summarized in Table 4.1-4. The resultant containment failure probabilities, based on applying the conservative burn pressure estimate to the System 80+ containment fragility curve (Figure 3.1-3) is presented in Table 4.1-5.

TABLE 4.1-4

SUMMARY OF PRA ASSUMPTIONS FOR SYSTEM 80+

HYDROGEN DEFLAGRATION INDUCED LOADING

PLANT DAMAGE STATE	FRACTION OF CORE ZIRCALLOY OXIDIZED	CONTAINMENT PRESSURE FOLLOWING PRIOR BURNS AND IGNITER OPERATION[1] (PSIA)	CONTAINMENT PRESSURE FOR TRANSIENTS WITHOUT EFFECTIVE CONTAINMENT PRESSURE CONTROL[2] (PSIA)	CONTAINMENT PRESSURE FOR TRANSIENTS WITH EFFECTIVE CONTAINMENT PRESSURE CONTROL[3] (PSIA)
CORE DAMAGE WITHOUT RECOVERY	0.50	< 50	84	64
CORE DAMAGE WITH RECOVERY	0.75	< 50	104	83

NOTES:

(1) Igniters assumed to operate early in the sequence and operation is continuous during the early hydrogen generation phase of the accident.

(2) AICC burn initiated from a 56 v/o steam atmosphere

(3) AICC burn initiated from 30 psia

TABLE 4.1-5

SUMMARY OF SYSTEM 80+ CONTAINMENT FAILURE PROBABILITY

DUE TO HYDROGEN DEFLAGRATION

	FAILURE PROBABILITY		
Fraction of Zr Reacted	With Prior Burns or Igniters	Without Containment Pressure Control	With Pressure Control
.50	0.0	0.0001	0.00
.75	0.0	0.01	0.0001

4.1.3.2 Hydrogen Detonation

4.1.3.2.1 Description of Phenomena

Detonations are combustion waves in which heating of the unburned gases is caused by compression from shock waves. The pressure loads developed during detonations are essentially dynamic loads (impulses) and can result in very short duration and localized pressure spikes many times greater than that of a deflagration initiated from similar conditions. As a result of these large loadings, detonations, if they should occur, can pose a threat to containment integrity and continued operation of mitigative equipment.

4.1.3.2.2 Parameters Affecting Hydrogen Detonation

Two classes of hydrogen detonations are typically distinguished: (a) detonation via direct initiation by high explosives and (b) Deflagration-to-Detonation Transition (DDT) resulting from large burn in a confined tube. Hydrogen detonations are influenced by (1) hydrogen concentration, (2) presence of inertents, (3) the ignition source and (4) system geometry (scale and configuration).

4.1.3.2.2.1 Hydrogen Concentration

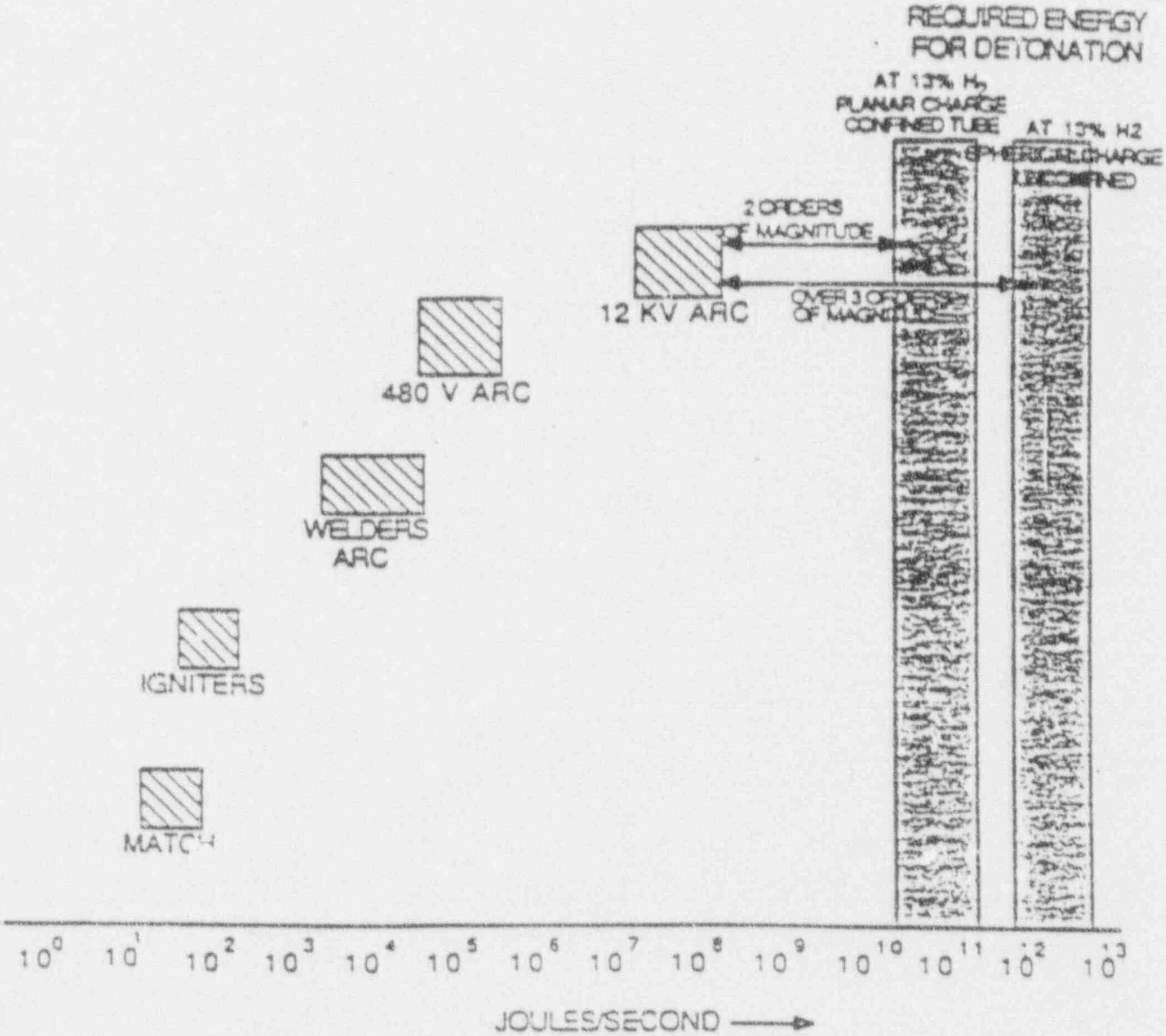
Experimental evidence has indicated that under favorable geometrical conditions a hydrogen detonation in dry air is possible at values of hydrogen concentration as low as 9.5 v/o. Detonations actually produced at these low hydrogen concentrations require a hot, dry mixture and the use of large explosive charges (See Reference 4.20). Based on this observation, the National Research Council reached the conclusion that mixtures of 9 to 11 v/o hydrogen might be detonable. In practice detonations at this low hydrogen concentration are not considered credible in a post severe accident LWR environment. Even at hydrogen concentrations of 13 v/o considerable energy would be required to initiate a detonation (See section 4.1.3.1.2.2).

Another mechanism for producing a detonation involves flame acceleration. Flame acceleration occurs due to turbulence induced by fans, structural roughness, obstacles, or changes in geometry. Flame acceleration is only important for mixtures that can be classified as highly flammable. Flame acceleration which results in sonic propagation of a detonation front is called a deflagration to detonation transition (DDT) and requires concentrations greater than 12 % in dry air. The lowest concentration for which DDT has been observed is 15% (See Section 4.1.3.1.3), and even then only with ideal geometric conditions.

4.1.3.2.2.2 Ignition Source

Direct initiation detonation of lean hydrogen mixtures (below 13 v/o) in an open containment would require a trigger of more than 10 MJ (See Figure 4.1-3). In contrast the energy required to initiate a deflagration is more than

Figure 4.1-3: Comparison of Ignition Source Energies with sources required for detonation



10 orders of magnitude lower than that for detonation. Therefore, wi[illegible] an appropriate energy source hydrogen detonations are not possible.

4.1.3.2.2.3 Steam Inerting of Containment

The presence of steam in the containment atmosphere can decrease the potential for, and severity of a hydrogen detonation. Experiments performed to date suggest that volumetric steam concentrations greater than about 30% will render even a stoichiometric mixture of hydrogen and oxygen in a non-detonatable state.

4.1.3.2.2.3 Geometry

Geometrical features can have an important influence on the potential for hydrogen detonation. In hydrogen mixtures which spontaneously undergo DDT, the ability of the system to detonate is dependent on the level of confinement and presence of obstacles. Open geometries are not typically favorable for the onset of detonation. Favorable geometries (such as confined areas containing obstructions) can even reduce requirements on ignition sources to induce detonation.

4.1.3.2.3 Summary of Experimental Evidence

Numerous studies have been performed to investigate the parameters affecting hydrogen detonation.

4.1.3.2.3.1 Small Scale Hydrogen Detonation Experiments

Small scale hydrogen detonation experiments include tests conducted by Atomics International (AI, see Reference 4.26), and tests performed by Sandia at the MINIFLAME facility (Reference 4.25). The AI tests investigated detonations in a 40 ft long, 1.25 ft diameter shock tube filled with various mixtures of hydrogen and air. A water spray was also included in the test facility in order to assess the impact of containment spray. Without sprays available detonation was observed at H2 levels of about 20 %. Use of a water spray delayed the onset of detonation and decreased the efficiency of the detonation process.

The MINIFLAME facility is a 1:12.6 scale model of the FLAME facility (see below). These tests were limited in scope and studied the detonation potential of hydrogen-air mixtures containing 20 and 30 % hydrogen mole fractions. Qualitatively, the MINIFLAME test results were similar to that of FLAME with the exception that at the smaller scale DDT was not observed during the 20 % hydrogen test series.

4.1.3.2.3.2 FLAME Experiments

The FLAME (Flame Acceleration Measurements and Experiments) Facility was designed and constructed for the USNRC to study hydrogen combustion problems associated with accelerated flames, transition to detonation and combustion in simulated containment geometries. The facility is a large rectangular channel

30.5 m in length. The experiments specifically investigated several effects that have been observed to be important to hydrogen detonation in small scale tests; hydrogen concentration, obstacles in the path of the combustion front and the degree of transverse venting. FLAME tests studied hydrogen concentrations between 12 and 30 v/o (near stoichiometric conditions).

The conclusions form the FLAME tests were as follows (See Reference 4.27):

1. The reactivity of the mixture as determined by the hydrogen concentration is the most important variable governing DDT. For very lean mixtures no significant flame acceleration and no transition to detonation was observed. The lowest hydrogen concentration found to result in a DDT occurred at 15% in a test with obstacles present and no transverse venting.

2. The presence of obstacles in the path of the flame front greatly increases flame speeds and overpressures. In fact, FLAME tests without obstacles did not result in DDT for hydrogen mixtures up to near stoichiometric conditions.

3. Large degrees of transverse venting reduce flame speeds and overpressures.

DDT results from the FLAME facility have been used by Sherman and Berman (Reference 4.28) to develop a quality risk ranking scheme for estimating the likelihood of a detonation in containment during a severe accident. This work was applied to the Bellefonte nuclear power plant with a large dry containment (Reference 4.54) and was used in the NUREG-1150 analysis of Sequoyah (ice condenser PWR). The application of this ranking scheme to the System 80+ is discussed in Section 4.1.3.1.4.1.

4.1.3.2.1.3 BMFT DDT Experiments (Reference 4.29)

This experimental effort involved a three test series with a total of 30 tests. The primary objective of these tests was to determine the influence of steam on DDT.

The hydrogen and steam concentrations in the experiments varied from 14 to 35 v/o and 0.4 v/o to 33 v/o, respectively. Of the thirty combustion experiments only 5 were observed to undergo DDT. The lowest hydrogen concentration DDT was observed was 18 v/o in a 1.2 v/o steam environment. As steam concentrations increased the hydrogen concentration at the onset of DDT also increased. No DDT was observed as the steam presence in the mixture increased to 30 v/o steam.

4.1.3.2.4 Significance of Hydrogen Detonation to System 80+

The potential for a direct hydrogen detonation, or a deflagration to detonation transition in the System 80+ is discussed in this section.

4.1.3.2.4.1 System 80+ Ranking of Deflagration to Detonation Transition Potential

The System 80+ containment consists of a large dry PWR with an In-Containment Refueling Water Storage Tank. The detonation potential for this containment configuration has been evaluated in a semi-quantitative fashion using the Sherman and Berman DDT detonation Ranking Scheme (see Reference 4.28). This procedure is based on the assumption that the likelihood of DDT can be expressed as a function of two variables; one based on the reactivity of the mixture, and a second based on the flame acceleration potential of the volume through which the flame propagates. The mixture reactivity or intrinsic flammability is based on the detonation cell width, which is related to the hydrogen concentration. The flame acceleration potential is based on the containment internal configuration.

The classification procedure for the System 80+ design is a three step process. In the first step intrinsic flammability is classified for various containment regions by classifying their maximum expected hydrogen concentrations according to the Sherman / Berman criteria (see Table 4.1-6a). In the second step the geometrical features of the various regions are compared against the Sherman/Berman geometric classifications (See Table 4.1-6b). In the last step of the process, the intrinsic flammability ranking and the geometric class rankings are combined to obtain a DDT likelihood ranking from Table 4.1-7. Application of this methodology to System 80+ suggests that for hydrogen concentrations typical of early severe accident containment failure scenarios, local hydrogen concentrations would be below 15 v/o even accounting for hydrogen stratification, and accumulation in the IRWST (see Reference 4.20). (Note that at 75% complete zirconium oxidation, the global hydrogen concentration in a dry atmosphere will be below 13 v/o and the expected local maximum hydrogen concentration would be below 15 v/o.) That would rank the hydrogen mixture as either class 4 (DDT possible but not observed) or class 5 mixtures (unlikely to undergo DDT). A similar ranking made for the containment geometric features indicated the containment to contain either class 4 or class 5 configurations. Such geometries are unfavorable to DDT. Mapping the flammability and geometric classifications on the Table 4.1-7 matrix indicates the System 80+ containment to be a class 5 containment. This classification implies that the potential for a DDT is highly unlikely to impossible. Using a probabilistic interpretation of impossible, the probability of a hydrogen deflagration undergoing a DDT was set at 0.001.

4.1.3.2.4.2 Direct Detonation of Hydrogen Within the System 80+ Containment

A second source of hydrogen detonation can arise from direct ignition of a flammable mixture. Direct ignition detonation typically requires an explosive charge within a highly flammable containment atmosphere. Reference 4.20 compared the energy required for a detonation with ignition sources typically available in PWR containments. This figure is reproduced as Figure 4.1-3. From this figure it can be clearly seen that containment ignition sources have energies which are more than three orders of magnitude lower than that

TABLE 4.1-6a

CLASSIFICATION OF MIXTURE DETONABILITY
(FROM REFERENCE 4.28)

MIXTURE CLASS	HYDROGEN MOLE FRACTION (V/O)	COMMENTS
1	24 TO 30	Highly detonable
2	21 TO 24	Less detonable than Class 1 mixtures
3	15 TO 21	Observed to undergo DDT in favorable geometries
4	13.5 TO 15	Detonations can propagate in mixture but DDT not observed
5	LESS THAN 13.5	Difficult to detonate

TABLE 4.1-6b

CLASSIFICATION OF GEOMETRIC FEATURES CONDUCIVE TO DDT

(FROM REFERENCE 4.28)

GEOMETRIC CLASS	DESCRIPTION
1	LARGE PARTIALLY CONFINED GEOMETRY WITH OBSTACLES IN THE PATH OF EXPANDING UNBURNED GASES. EXAMPLE: A LARGE TUBE WITH OBSTACLES AND IGNITION GOING FROM AN OPEN TO CLOSED END.
2	GEOMETRY IS SIMILAR TO CLASS 1 BUT TUBE MAY BE OPEN AT BOTH ENDS OR TRANSVERSE VENTING IS ALLOWED.
3	GEOMETRIES THAT YIELD MODERATE FLAME ACCELERATION. EXAMPLE: OPEN TUBES WITHOUT OBSTACLES.
4	LARGE VOLUMES WITH FEW OBSTACLE AND SIGNIFICANT VENTING TRANSVERSE TO FLAME PATH
5	UNCONFINED GEOMETRY

TABLE 4.1-7

DEPENDENCE OF SHERMAN / BERMAN RESULT CLASS

ON MIXTURE AND GEOMETRY CLASS

GEOMETRIC CLASS	MIXTURE CLASS				
	1	2	3	4	5
1	1	1	2	3	4
2	1	2	3	4	5
3	2	3	3	4	5
4	3	4	4	5	5
5	4	5	5	5	5

Result Class 1: DDT is highly likely
Result Class 2: DDT is likely
Result Class 3: DDT may occur
Result Class 4: DDT is possible, but unlikely
Result Class 5: DDT is highly unlikely to impossible

NOTE: Shaded area corresponds to the System 80 + design range

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necessary to detonate a 13 v/O dry hydrogen mixture in an unconfined geometry. On the other hand all ignition sources (even those of 10 orders of magnitude lower strength) are sufficient to cause a deflagration.

Based on the above work and supporting analyses presented in Reference 4.26, the possibility of detonation within the System 80+ containment is considered remote. Direct initiation of a hydrogen detonation would be improbable within the System 80+ containment while initiation of a deflagration during a severe accident is virtually certain. Similarly, an assessment of the intrinsic flammability and geometric features of the System 80+ containment indicates the potential for DDT is highly unlikely to impossible.

4.1.3.2.5 Application to the PRA

As discussed above the potential for hydrogen detonation within the System 80+ containment is remote. This is particularly so when considering early containment failure process since oxidation processes associated with core concrete attack are not considered (See Section 4.2.3). In developing the PRA the hydrogen combustion events were quantified as follows:

Conditional probability that a hydrogen burn would either be initiated as or become a detonation:

> For accident scenarios where the steam concentration is expected to exceed 30 v/o, detonations are not considered credible.
>
> For hydrogen concentrations below 10 v/o in dry air (<60 % zircaloy oxidation) detonations within the containment are considered impossible.
>
> For hydrogen concentrations above 10 v/o in dry air and steam concentrations below 30 v/o steam, the fraction of hydrogen burns that may become detonations is taken to be 0.001 (highly unlikely)

Furthermore, it will be assumed in the PRA that the occurrence of a detonation will fail containment. This is a very conservative position. While the ensuing pressure spike occurring following a detonation is very large compared to a deflagration pressure rise, the detonation spike is of very short duration (typically less than 10 ms) and consequently may not pose a threat to large structural components, and the containment structure.

4.1.4 OTHER EARLY CONTAINMENT FAILURE MECHANISMS

4.1.4.1 Direct Shell Attack via Corium Impingement

4.1.4.1.1 Description of the Phenomena

This failure mechanism considers the containment failure potential resulting from a high pressure RV discharge of highly energetic corium debris interacting with the stainless steel containment shell. Failure of the steel shell is assumed to be small and localized to the points of corium impingement.

4.1.4.1.2 Application to System 80+ Design

The System 80+ containment has been designed to provide adequate protection of the containment steel plate from debris and / or missile attack. In the lower portion of the containment below the 92'-9" elevation, the steel shell is imbedded in a minimum of 3 ft of concrete (See Section 3.3). Above the lower compartment floor, the crane wall provides a 5 ft thick (minimum) concrete barrier separating the potential escaping core debris from the lower portions of the containment shell. The upper containment shell is partially protected from corium and RV generated missiles by a substantial missile shield located above the Reactor Vessel top head. The remainder of the containment shell surface which is either not directly imbedded in concrete or separated via a substantial concrete shield is located in a small portion of the upper containment elevation where energetic missile contact is highly unlikely due to the large vertical distance the missile would have to travel.

4.1.4.1.3 Application to the PRA

The probability that corium debris could be ejected from the RV and reach the upper containment shell with sufficient energy to cause a localized containment failure was established via engineering judgement as follows:

1. For high pressure RV lower head failures, the conditional probability of containment failure due to direct corium impingement was assumed to be 0.001.

2. For low and intermediate RV lower head failures containment shell failure due to direct shell attack was not considered credible.

It should be noted that this failure mechanism does not include containment failure via combustion induced missile generation. This failure mechanism is included in the discussion of hydrogen detonation (See Section 4.1.3).

4.1.4.2 Cavity Overpressure Failure

4.1.4.2.1 Description of the Phenomena

Following a HPME, large quantities of steam and corium are discharged into the lower portion of the reactor cavity. This discharge can potentially challenge the integrity of the reactor cavity and thereby threaten containment integrity. Cavity overpressurization can potentially result in a structural failure of the reactor cavity and associated RV supports. Failure of the RV supports can produce excessive motions in the RCS and steam generators potentially failing a containment penetration or producing an unisolable breach in piping exiting the containment.

Potential sources of cavity overpressurization include the EVSE event and the energetic failure of the RV lower head. The EVSE induced failure of the cavity and or reactor internal supports is considered in Section 4.1.2. This section considers localized cavity pressurization induced by steam pressurization of the reactor cavity space immediately upon RV lower head failure.

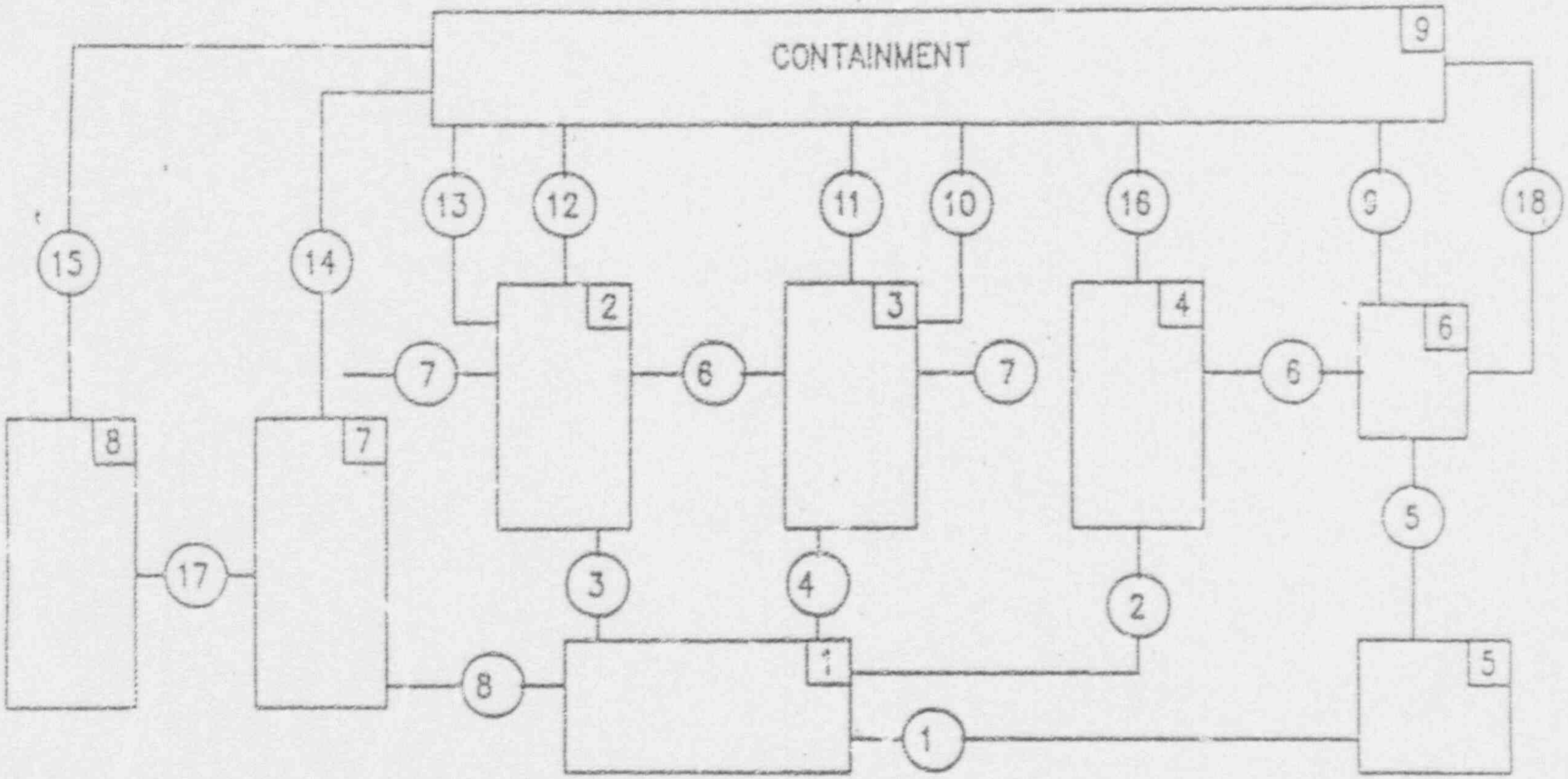
4.1.4.2.2 Significance to System 80+

System 80+ is expected to withstand cavity pressurization events following RV lower head failure. This capability of System 80+ arises from the (1) high reactor cavity wall strength and (2) large reactor cavity volume.

The post severe accident cavity pressurization performance of the System 80+ design was evaluated analytically for a simulated high pressure superheated steam blowdown from the RV lower head. Analyses were performed using the ABB-CE DDIF Mod 7 (Reference 4.30) cavity pressurization code. In this analysis a multi-compartment representation (see Figure 4.1-4) was assembled to simulate the detailed cavity pressurization process following the failure of the RV lower head. Pressurizations were established using RV lower head failure sizes equivalent to a lower head instrument tube failure and a larger creep failure of the RV lower head. Results of these indicate System 80+ cavity loadings to be below 100 psid. These loads are below the cavity wall design values of approximately 225 psid and consequently will not challenge cavity or containment integrity.

4.1.4.2.3 Application to the PRA

The cavity overpressurization induced containment failure is not considered a credible threat to containment integrity. However, for the purpose of completeness this failure mechanism has been included in the PRA supporting logic models with a probability of 0.0001.



LEGEND

1	LOWER CAVITY	6	WALKWAY TO LOWER COMPT.
2 3	1/2 RC ANNULUS	7	HOLDUP VOLUME
4	ICI CHASE	8	IRWST FREE BOARD
5	CORE DEBRIS CHAMBER	9	CONTAINMENT

NOTE: Circles represent flowpaths, Squares represent nodal volumes

FIGURE 4.1-4: NODAL MODEL OF SYSTEM 80+ REACTOR

4.1.4.3 Rocket Induced Containment Failure

4.1.4.3.1 Description of Phenomena

A rocket induced containment failure mechanism (Saturn V effect) was considered in the Oconee PRA (Reference 4.31) and later in the German Risk Study (Phase B) of the Biblis B PWR (Reference 4.18). In this scenario a full unzipping of the lower head at vessel breach induces tremendous momentary pressure and thrust forces on the RV. For RV failures at high pressure (greater than 1200 psia), sufficient forces can be developed to fail the RV supports and potentially lift the reactor vessel out of the cavity and impact the containment.

4.1.4.3.2 Significance to System 80+

A review of the System 80+ reactor cavity and lower head design suggests that this failure mechanism is not viable for the this plant design. This conclusion is based on the fact that (1) the RV lower head failure mode will be predominately governed by instrument tube failures and (2) the large System 80+ reactor cavity will not allow substantial pressurization.

4.1.4.3.3 Application to the System 80+ PRA

For completeness the rocket failure mode will be included in the supporting logic model for the System 80+ PRA with a corresponding probability of zero.

4.1.4.4 Synergistic Issues

Many early containment failure mechanisms are a result of several containment threatening processes occurring simultaneously. For the System 80+ PRA synergistic effects are typically considered under the umbrella of Direct Containment Heating (See Section 4.1.1).

Synergism between the hydrogen burn and steaming following VB can be established by estimating Hydrogen burn pressures at the uppermost de-inerted steam pressure (see Section 4.1.3).

Missile threats to the containment are considered under the phenomenological source of the missile. For example missiles generated by hydrogen burns / detonations, IVSE and EVSE are considered within their respective phenomenological section.

4.1.4.5 Loss of Containment Isolation Prior to Core Melt

4.1.4.5.1 Description of the Phenomena

During power operation, the containment is required to be closed. However, in rare instances procedures may be violated or common cause valve failures may result in loss of containment isolation during plant operation. Severe accidents initiated from this condition progress in the presence of a compromised final fission product barrier and may release considerable quantities of fission products early into the environment.

4.1.4.5.1 Application to the System 80+ Design

System 80+ is designed so that loss of containment isolation is highly unlikely. Preventive features in the System 80+ design to aid in maintaining containment isolation include:

1. use of double isolation valves for all containment penetrations

2. use of diverse means of powering isolation valves

3. selection of isolation valve failure position consistent with its safety related function.

Detailed information on the Containment Isolation System may be found in Section 6.2.4 of CESSAR-DC.

In the unlikely event of a loss of containment isolation preceding a severe accident, steam released from the RCS will pressurize the containment and drive out some of the initial containment air early in the transient. This feature leads to an interesting feature of this event. If for example an early steam discharge is immediately followed by spray actuation and successful containment heat removal, the pressure within containment may actually go subatmospheric (by less than 1 psig) for a period of many hours, thus, minimizing the environmental releases to that releases during the early steam / air discharge. As a result of this containment vacuum, the resulting radiation releases would be only marginally greater than releases from an intact containment. Hydrogen burns may be more likely during this scenario due to a low steam content in the containment and a smaller noncondensible gas content.

4.1.4.5.2 Application to the PRA

Containment isolation failures are considered as early breaches of containment. In these transients, the containment will lose noncondensibles and steam (and airborne radionuclides) from the initiation of the containment transient. As a result of the hole in the containment, the containment pressurization will be low.

Three characteristic plant responses can be expected for this transient. First, without containment sprays, radionuclide releases will enter the environment with little scrubbing. In a second instance, the containment spray function is available without the heat removal function. In this scenario, all noble gases will be completely released to the environment and iodine and cesium releases will be considerably scrubbed due to the scrubbing action of the containment sprays. In the final scenario, the full containment heat removal system (sprays and associated heat exchangers) functions. This last scenario is rather unique in its fission product release characteristic. Since noncondensibles are driven out of containment early in the severe accident (while radionuclide releases are relatively low), the operation of containment heat removal using containment sprays will ultimately condense the containment steam, and cool the containment atmosphere. Since the containment has a smaller and cooler non condensible gas component, a vacuum is created within the containment. This vacuum, while not indefinite, can retain fission products without any leakage into the environment for about 48 hours depending on the initial scenario (see also Section 5).

4.1.4.6 Containment Bypass

4.1.4.6.1 Description of the Phenomena

NUREG-1150 (Reference 4.32) identified containment bypass as an important contributor to the large, early releases of radionuclides for the Zion and Surry PWRs. The principal contributors to these severe accidents are steam generator tube ruptures (SGTRs) and interfacing-system LOCAs (ISLOCAs). Should these events progress into severe accidents, the radiation releases to the environment would be large and energetic and would pose a significant radiation exposure to the general public.

4.1.4.6.2 Significance to System 80+

4.1.4.6.2.1 ISLOCAs

The core damage frequency associated with System 80+ intersystem LOCA failures is estimated to be about $3x10^{-9}$ per reactor year. The low probability of these events is associated with the fact that the System 80+ design incorporated much of the EPRI URD guidance in designing to limit ISLOCAs. In particular, much of the connecting piping for the RCS has been designed to meet the ultimate rupture strength criteria required in SECY-90-016 and considerable piping that was previously routed outside of containment is now within the containment envelop. To further reduce the possibility of ISLOCA, interfaces between the RCS and connecting systems include, as appropriate, design features to leak test valves, indicate valve position and alarm high pressure in low pressure lines.

A detailed investigation of the sources of ISLOCAs has been reported to NRC in Reference 4.35. This report investigated the source of potential ISLOCAs from the System 80+

-Safety Injection System
-Shutdown Cooling System
-Chemical and Volume Control System
-Process Sampling System

4.1.4.6.2 Steam Generator Tube Rupture

A Steam Generator Tube Rupture (SGTR) can provide a significant pathway for radionuclide release from the RCS to the environment. In System 80+ considerable effort has been expended to both reduce the potential for and consequences of a steam generator tube rupture. Improvements to System 80+ to prevent and mitigate SGTRs include:

- Steam generator tubes made of thermally treated Inconel 690, which has favorable corrosion resistance properties including superior resistance to primary and secondary stress corrosion cracking,

- A deaerator in the condensate/feedwater system for the removal of oxygen,

- Condensate system with a full flow condensate polisher to remove dissolved and suspended impurities,

- Main condenser with provisions for early detection of tube leaks, and segmented design permitting the repair of leaks while operating at reduced power.

The SGTR precursor to severe accidents can be quite important to public risk. In fact, SGTR bypass scenarios presented for the initial System 80+ CESSAR-DC PRA (Reference 4.36) indicated that the doses due to SGTR initiated severe accident scenarios were between 2 to more than 10 times higher than that predicted for containment failure transients. While these transients can potentially be very serious threats to the public, most SGTRs (including those resulting in severe core damage) will be such that environmental radiation releases will be small. This is due to (1) secondary water that is available to the SG secondary side will produce a favorable environment (cool and low steaming rate) within the primary side of the steam generator tubes for fission product plateout, and (2) when the secondary side water level covers the broken tube elevation most iodine and cesium that leave the primary side will be "scrubbed out" in the secondary side water pool.

4.1.4.6.3 Application to the PRA

4.1.4.6.3.1 ISLOCAs

The dominant ISLOCA sequence involves the combined failure of check and isolation valves in the RHR line resulting in a catastrophic failure of this line outside of containment. This ISLOCA will typically deposit RCS inventory into a watertight area of the auxiliary building. Sufficient RCS inventory will be lost to the auxiliary building prior to substantial core damage so that the ISLOCA break location will be covered by several feet of water. This water will serve to scrub fission product releases from the ruptured safety injection pipe prior to entering the environment. The PRA conservatively assumes a decontamination factor of 10 associated with this water pool scrubbing.

4.1.4.6.3.2 SGTR

The PRA considers SGTRs due to both SGTR initiating events and thermally induced SGTRs. The consequences of SGTR events resulting from an initial SGTR will be established based on details of the plant damage state (PDS) prior to core damage. Induced SGTRs can only occur when the SG tubes are uncovered and water inventory in the steam generator is minimal. These events arise from the creep rupture assumptions presented in Section 4.1.1.3.

4.2 LATE CONTAINMENT FAILURE

Late containment failure refers to those severe accident scenarios where containment failure occurs more than 1 hour after VB and more than 24 hours after event initiation. The 24 hour definition of late containment failure is consistent with the deterministic containment integrity goal identified in the draft SER of the EPRI URD (Reference 4.52).

Four potential mechanisms for late containment failure are identified for System 80+. These are:

1. Gradual Containment Overpressurization
2. Basemat Melt-Through
3. Temperature Induced Penetration Seal Failure
4. Delayed Combustion

These failure mechanisms, and their role in the System 80+ PRA are discussed in the following sections.

4.2.1 GRADUAL OVERPRESSURIZATION

Without containment heat removal, the containment would fail by overpressurization due to the addition of steam and possibly noncondensable gases into the containment atmosphere. This pressurization process is typically gradual taking two or more days to reach the containment ultimate failure pressure. While the containment failure is energetic, the relatively long time to containment breach allows considerable time for recovery actions, as well as, providing time for fission product sources to decay and non-volatile fission product components to deposit themselves within the containment.

4.2.1.1 Steam Overpressurization

4.2.1.1.1 Description of the Phenomena

4.2.1.1.1.1 Containment Failure Before Vessel Breach

This category of containment failure arises when the containment heat removal function is irrecoverably lost and cooling of the RCS with a breach (either due to pipe rupture or open SDS Valve) is facilitated.

Scenario 1: Containment Failure with SDS Valve discharge into the IRWST

This scenario consists of an extended total loss of feedwater event (TLOFW) where Feed and Bleed core heat removal is successful and containment heat removal is unavailable due to failure of containment spray heat exchangers. This transient is described in detail in section 5.3.5. In this transient the SI pumps will inject IRWST inventory into the RCS and the RCS discharges steam generated in the cooling process into the IRWST. The outcome of the once-thru-core-cooling (OTCC) process is to maintain core temperatures at acceptable levels so long as makeup inventory is available. Once the IRWST reaches saturation, steam produced in the IRWST will be discharged into and pressurize the containment. Without restoration of containment heat removal function, the containment will fail and the IRWST liquid will flash, causing the SI pumps to cavitate. Without restoration of primary side inventory control, the core will slowly uncover and begin to melt.

Scenario 2: Containment Failure Following Primary Side Pipe Ruptures

In this scenario a large RCS primary side pipe rupture occurs which allows the discharge of superheated and saturated steam directly into the containment. Core heat removal, if available, will typically involve steaming of the corium debris and pressurization of the containment. In this scenario however, the RCS discharge is deposited into the containment directly. Thus, the IRWST is only heated via the collection of condensed steam. This scenario will result in a containment damaging condition in advance of reaching saturation conditions in the IRWST. Thus, containment depressurization is not expected to result in flashing of the IRWST water and therefore core heat removal will continue until the IRWST is depleted.

4.2.1.1.1.2 Containment Failure Following Vessel Breach

System 80+ employs a unique cavity design to trap corium debris in the reactor cavity (See Section 3.6) and a manually actuated cavity flooding system to rapidly arrest corium-concrete attack and cool the corium debris on the reactor cavity floor. The intent of these design features is to allow the reactor cavity to serve as a repository of most, if not all, the post-accident corium debris. As a consequence of this core debris cooling process, steam will be generated. If active core heat removal systems (containment sprays) are unavailable, the steam addition will pressurize the containment to the point of failure.

4.2.1.1.2 Parameters Affecting Steam Overpressurization

Steam overpressurization of the containment is influenced by the ability of the debris to produce steam and the ability of the containment active systems to condense steam. Analyses demonstrate that availability of one train of the containment spray system will be sufficient to control containment pressure well below the ultimate pressure threshold.

4.2.1.1.3 Significance to the System 80+

System 80+ has been designed with a very flexible and reliable containment spray system. MAAP analyses confirm that maintenance of the containment spray heat removal function will guarantee containment integrity. Should containment heat removal be unavailable, containment failure will occur in two to three days. This long time to failure, even in the absence of heat removal is a consequence of several System 80+ design features. These include (1) the spherical shell containment design which provides both a high resistive strength to internal pressurization and a large free volume for gas accumulation, (2) the presence of a large quantity of passive heat sinks (both steel and concrete) and (3) a CFS which is capable of circulating more than 500,000 gallons of initially subcooled water over the core debris.

4.2.1.1.4 Application to the PRA

For purposes of the baseline System 80+ PRA, the containment overpressure transients initiated by a loss of containment heat removal were assumed to be irrecoverable and ultimate containment failure was considered to occur between 155 and 200 psia (See Figure 3.1-3 based on 350 F steel shell temperature). Since all gradual overpressure transients require time frames of two days, or more prior to containment failure (and in some sequences prior to core melt) recovery of many failed systems or actuation of alternate cooling systems are highly likely. These recovery actions will be considered in performing the System 80+ Level 2 sensitivity studies.

4.2.1.2 Overpressure via Steaming in the Presence of Non-Condensables

4.2.1.2.1 Description of the Phenomena

Severe accidents leading to substantial core concrete interaction may also contribute to the containment overpressure process via the concrete destruction process. However, containment overpressure failure under these conditions is a result of combined pressurization of the containment atmosphere due to steam and non-condensible gases. For this process to be a threat to containment, the containment sprays must be unavailable and significant core concrete interaction must occur.

4.2.1.2.2 Parameters Affecting Overpressurization

The contribution of non-condensible gases to containment failure is a function of the degree of core concrete attack, the distribution of corium within the containment and the constituents of the basemat and structural concrete.

4.2.1.2.2.1 Core - Concrete Attack

The concrete destruction process can release potentially large quantities of non-condensable gases to the containment. These gases arise from the dehydration (release of H_2O) and decarboxylation (release of CO_2) processes associated with the heatup of concrete. In practice, two types of concretes are common to LWRs constructed in the United States. These are : limestone/common sand concrete and basaltic concrete. Properties of these concretes and Limestone concrete as obtained from Reference 4.46 are summarized in Tables 4.2-1 and 4.2-2. For purposes of gas generation, these concretes are distinguished primarily by the level of bound carbon dioxide within the concrete aggregate. Limestone/Common Sand concrete have carbon dioxide levels of more than 20 wt %, while basaltic concrete have only trace amounts of carbon dioxide (1.5 wt %).

From Table 4.2-3, it can be seen that non-condensable gases evolve from concrete at three temperature levels associated with the thermal decomposition process. At concrete temperatures greater than 212 ° F, the free water in the concrete is evaporated. If corium is available, this water will react with the metallic phase of the melt and be reduced to hydrogen. The total amount of hydrogen released from this process is equivalent to about 0.22% of the weight of concrete affected. Free water released due to thermal decomposition in areas not in contact with the corium melt corresponds to about 2 wt % of the concrete attacked. Bound water is typically released at higher concrete temperatures in the vicinity of 700°F. As with the free water, H_2O liberated from the concrete will be released as hydrogen (approximately (3 wt % of concrete). It is expected that release of water bound in the concrete will occur only in the vicinity of the corium melt. The last step of the gas evolution process involves the decarboxylation of concrete (that is the release of carbon dioxide). This release will occur in the vicinity of the corium concrete attack and will be dependent on the specific concrete being eroded.

4.2.1.2.2.2 Corium Distribution

The gas evolution due to corium-concrete attack is directly related to the amount of corium in contact and/or close proximity to the core debris. As discussed above, concrete not in close contact with the corium debris will not be heated to sufficiently high levels to complete the dehydration process or begin the decarboxylation process.

4.2.1.2.3 Significance to System 80+

An estimate of the level of non-condensible gas evolution from concrete can be established for System 80+ using the following bounding assumptions:

1. All cavity concrete releases both free and bound water. For wet cavity scenarios, basemat water releases are assumed to enter the containment as hydrogen. Sufficient unoxidized corium constituents (principally zirconium, iron and chromium) are assumed available to reduce water molecules into a metallic oxide and hydrogen. Hydrogen generated in this manner may be capable of entering the containment potentially increasing the containment hydrogen concentration to levels corresponding to 100 % (or more) core-wide zirconium water reaction (See Section 4.2.4).

   For dry cavity scenarios, hydrogen generated during the concrete decomposition process will likely undergo auto-ignition as the hydrogen gas leaves the corium bed. This implies the potential for a frequent lower concentration hydrogen burns.

2. All decomposed basemat concrete also releases carbon dioxide to the containment.

3. During a 48 hour interval, the maximum concrete erosion depth is 10 feet.

Based on these assumptions the maximum amount of non-condensables expected to be evolved during the concrete thermal decomposition will yield about 3650 lbm-moles of hydrogen and about 5240 lbm-moles of carbon dioxide. The resulting partial pressure contributions due to these non-condensables are about 0.1 psi and 13.5 psi for the hydrogen and carbon dioxide gases, respectively.

These results suggest that while non-condensible gas evolution will contribute to the containment overpressurization process, containment failure primarily due to non-condensable gas evolution is highly unlikely. In fact, significant non-condensable gas pressure contribution would require destruction of more than 5 million pounds of concrete.

4.2.1.2.4 Application to the PRA

The System 80+ PRA does not differentiate between containment overpressure failure due to primarily steam addition and that caused by a combination of steam and non-condensible sources. Containment overpressure failure caused primarily by non-condensible gas evolution is not considered credible.

4.2.2 BASEMAT MELT-THROUGH

4.2.2.1 Description of the Phenomena

Basemat melt-through refers to the process of concrete decomposition and destruction associated with a corium melt interacting with the reactor cavity basemat. The accident progression is slow (taking from several days onward to penetrate the reactor cavity basemat) and provided the corium melts through to the containment subsoil, the corium release to the environment is negligible. Once in contact with the subsoil, most of the corium will vitrify into a relatively impermeable substance. For some small number of System 80+ sequences the containment breach may include a pathway into the subsphere SI pump room. Under these circumstances, the basemat melt-through will respond as a filtered above ground release.

Basemat melt-through can also undermine and ultimately fail the reactor cavity walls, which in turn may cause a significant movement in the RCS and connecting piping. Should this situation develop, the RCS displacements may be sufficiently large to cause failures of containment penetrations. These failures will produce above ground radiation releases.

4.2.2.1.1 Overview of the Concrete Decomposition Process

LWR cavity basemats are typically constructed of concrete. The precise constituents of the concrete mix vary from reactor to reactor and typically reflects a concrete mixture that is indigenous to the plant site. Three general types of concretes have been used in reactor cavity basemat construction: limestone concrete, limestone-common sand concrete and basaltic concrete.

Concrete components consist of cement, sand and aggregate. The aggregate has the largest influence on which of the above concrete categories apply. It is common (and economical) to obtain concrete materials from sources in the same general area as the plant site. However, the selection and use of concrete in the cavity basemat construction can have a noticeable impact on the severe accident progression.

A comparison of the major properties of the various concrete types can be found in Table 4.2-1 and 4.2-2. These tables are based on information provided in Reference 4.46.

An overview of the concrete decomposition process is presented in Section 4.2.1.2.2.1.

TABLE 4.2-1. COMPARISON OF PROPERTIES AND CONSTITUENTS OF COMMON CONCRETES				
PROPERTY	UNITS	BASALTIC	LIMESTONE	LIMESTONE / COMMON SAND
AVG. SPECIFIC HEAT	J/KG/K	913	979	903
MELTING TEMPERATURE	K	1450	1750	1500
ENERGY ABSORBED IN ENDOTHERMIC CHEMICAL REACTIONS	J/KG	269 E+3	1735 E+3	1150 E+3
LATENT HEAT FOR CONCRETE MELTING	J/KG	555 E+3	760 E+3	560 E+3

TABLE 4.2-2 COMPARISON OF CONCRETE CONSTITUENTS

COMPONENT MASS FRACTION (WT%)	BASALTIC	LIMESTONE	LIMESTONE COMMON/ SAND
SiO2	.5484	.036	.358
CaO	.0882	.4540	.313
CO2	.015	.357	.2115
H2O-FREE	.0386	.0394	.027
H2O-BOUND	.0200	.0200	.020
OTHER	.2898	.0936	.0705

TABLE 4.2-3

GAS EVOLUTION DURING THE THERMAL DEGRADATION OF CONCRETE[1,2]

GAS RELEASE FROM CONCRETE	TEMPERATURE RANGE °F
1) FREE WATER	180 - 280
2) CHEMICALLY-BOUND WATER	660 - 950
3) CARBON DIOXIDE	1000 - 1800

NOTES:

1. Data obtained from Reference 4.20
2. Oxidation processes within the corium melt may result in the chemical reduction of water releases to hydrogen and the carbon dioxide releases to carbon monoxide.

4.2.2.2 Parameters Affecting Basemat Melt-Through

4.2.2.2.1 Concrete Properties

Several types of concrete have been used in the construction of nuclear power plants. The concretes vary as to their enthalpy of decomposition and bound gas content. As a result of these differences, the concrete type used in the cavity basemat construction can impact the overall accident performance including affecting the rate of basemat erosion, liberation of noncondensable and combustible gases and concrete water release.

Concrete decomposition is a thermally driven process. The energy required to decompose concrete results from the energy required to bring the concrete temperature to a point where many of the chemical bonds in the cement and the aggregate can be broken and the gaseous products be liberated. Since the composition of concrete vary, the temperature at which significant decomposition starts and the enthalpy of decomposition will also vary among concrete types. This results in different corium-concrete attack erosion profiles. In general, of the three common types of concretes used in reactor cavity basemat construction, limestone concrete has the largest enthalpy of decomposition and basaltic concretes have the lowest (see Table 4.2-1). Consequently, for similar core concrete attack situations basaltic concrete basemats are predicted to exhibit more pronounced erosion.

The general concrete composition is also important from the perspective of containment pressurization during severe accidents either via ideal gas pressurization or via the addition of large quantities of combustible gases into containment. Concretes with a large limestone content may be capable of producing significant quantities of carbon dioxide/carbon monoxide when subjected to core concrete attack. Both species of gas can contribute to the containment pressure as a non-condensible ideal gas. Carbon monoxide is combustible and may contribute to a late combustion pressure spike.

All concretes contain about 5% water by weight. Thus, dehydration of concrete can release potentially significant quantities of steam which may be added to the containment atmosphere as water vapor and/or hydrogen.

4.2.2.2.2 Corium Mass and Distribution Within the Reactor Cavity

The erosion of the basemat concrete is a thermally driven process. That is, heat transferred to the basemat and the subsequent heatup of the concrete is the driving mechanism for the various concrete decomposition and melting processes. Since the corium mass in the cavity also defines the cavity heat load the greater the corium mass the more energy available for concrete erosion.

4.2.2.3 Debris Bed Coolability

Debris coolability has been assessed for ARSAP in Reference 4.41 in support of the URD. Based on this assessment, it was concluded that the availability of a water source and a floor area of at least 0.02 square meter per Megawatt of reactor power will be sufficient to guarantee long term debris coolability for a typical ALWR. This assessment is based on (1) review of experimental data which suggests that the final state of the corium debris within the cavity would consist of a mixture of fragments and a relatively continuous, but porous and cracked, phase that would be distributed uniformly over the basemat and (2) an assessment of the heat removal mechanisms from the corium surface which guarantees that core debris in this configuration would be able to remove the expected 0.5 Mw/m^2 produced within the debris bed in the long term.

Experiments pertinent to debris coolability are summarized in Table 4.2-4. Additional details on these experiments as they relate to the ALWR are presented below and in Reference 4.41 .

4.2 2.2.3.1 Debris Configuration

It has been shown by several investigators that the morphology of quenched debris depends upon the relative amounts of liquid debris and water present. Breakup of debris jets can occur if the water depth is sufficient. Otherwise, channeling [illegible] accumulation of debris can occur. Small particulate debris breakup ([illegible] mm) is typically not conducive to debris cooling in that packed d[illegible] porosity exhibit a steam/water counter-current flow phenomenon [illegible] penetration difficult. Conversely, high porosity beds of mo[illegible] powers typical of that associated with decay heat at times greater than 3 hours after shutdown, should be easily coolable by an overlying water pool. Experimental evidence indicates that a mixture of both particulate and a continuous phase occur.

Experiments of particular note are simulated corium drop experiments performed by Benz (Reference 4.37) and the Corium Water Thermal Interaction (CWTI) tests performed by Spencer (Reference 4.38). In the Benz experiments molten steel or uranium dioxide changes were dropped into an interaction vessel containing excess water and the resultant debris fragmentation was measured. Based on these experiments the smallest average particle diameter was about 2mm with more than 60% of the debris being greater than 4 mm.

The CWTI tests covered a range of experimental conditions. Of particular interest were tests CWTI-7 and CWTI-8 which investigated the fragmentation of a zirconium-uranium oxide mixture dropped into a water filled interaction vessel. An examination of the debris indicated that debris was in the form of a solid but internally porous (about 50% porosity) slab.

Based on these tests, ARSAP (Reference 4.41) concluded that for prototypic debris and representative debris/water volumes, debris fragmentation would be limited and the majority of the debris will form a continuous porous slab.

4.2.2.2.3.2 Debris Bed Heat Transfer

Heat removal from the debris will be governed by the debris configuration. Experimental observations of cooling of particle beds indicate that for larger

TABLE 4.2-4

SUMMARY OF DEBRIS COOLABILITY INVESTIGATIONS*			
EXPERIMENT	SCALE	DEBRIS CONFIGURATION	COMMENTS
BENZ-ISPRA	10 cm 50 cm	continuous fragmented	observed fragmentation with excess water
ANL-CWTI	21 cm	continuous, some fragments	observed enhanced surface area, debris porous
SNL	22 cm	continuous, cracked	inferred water ingression on quench
SNL-SWISS	21 cm	continuous	observed stable crust, mettalic melt, high power
ANL	25 cm	continuous, cavern	observed stable crust attached to sidewall
ANL-ACE	50 cm	continuous	observed crust collapse onto debris surface
GRIMSVOTN MAGMA	90 m	deeply fissured	observed large lava field cooled and cracked to 12 m depth
TMI-2	6 m	continuous, some fragments	inferred lower plenum heat flux corresponding to CHF
MACE SCOPING	30 cm	continuous, some fragments	observed crust attached to sidewall with periodic breakup
WETCOR-1	32 cm	continuous	crust anchored to facility. corium simulant mixture of Al_2O_3 and CaO
MACE 1B	50 cm	continuous	stable crust with periodic breakup. long term cooling was approached, however complete debris quench was not achieved. Simulant was 950 lbm mixture of UO_2, ZrO_2 and Zr.

* Table extended from Reference 4.41

particle sizes (greater than about 3 mm) the heat removal rate from a particle bed are relatively independent of depth to 100 cm and can be bounded by the flat plate CHF limit (see Figure 4.2-1). This rate of heat removal was analytically found to be sufficient to guarantee corium coolability (See Reference 4.41).

The coolability of thick oxidic debris slabs have been demonstrated in large scale magma experiments conducted at Grimsvolth (Reference 4.42). In this test water was poured on unconfined magma and the magma was observed to solidify over time via water ingression to a depth of over 14 m.

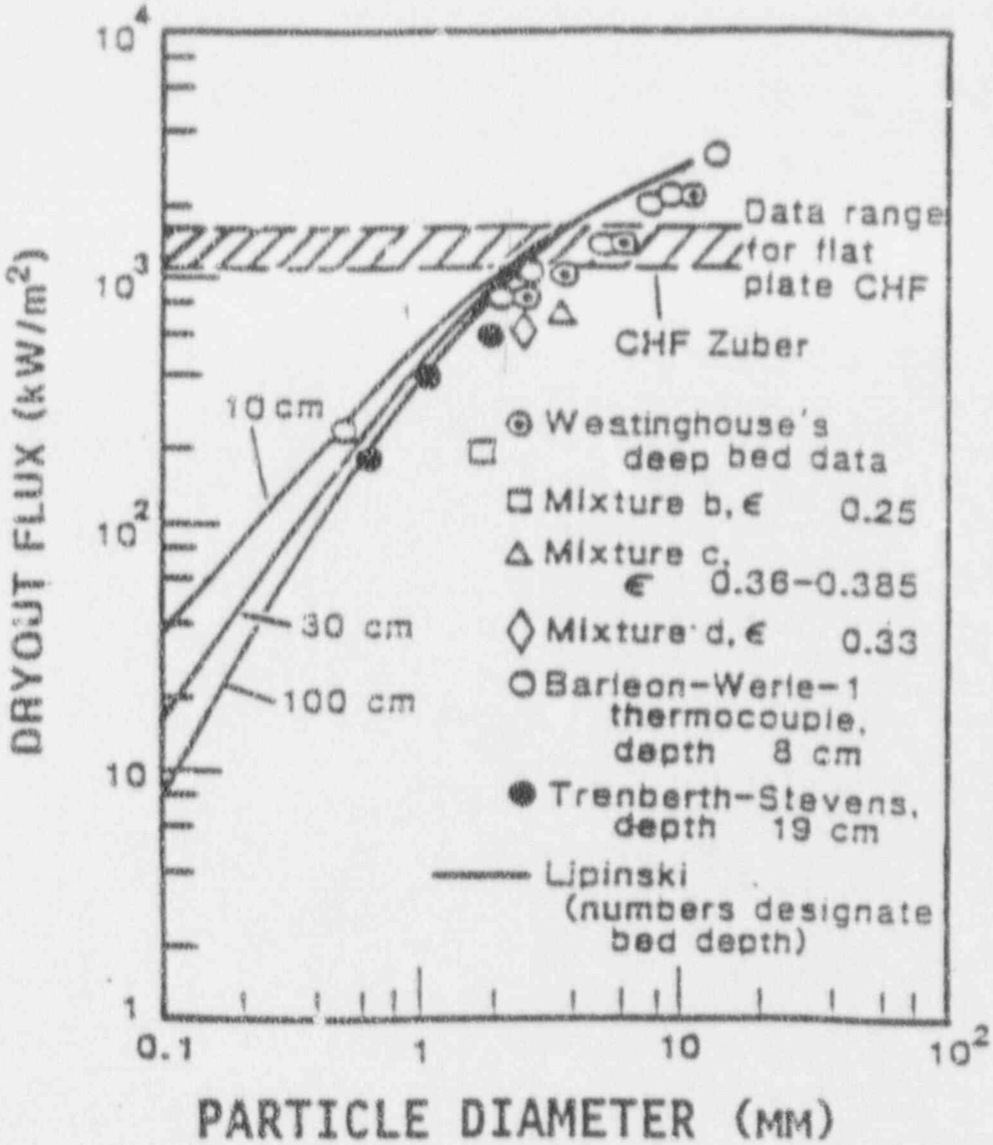
Corium coolability has also been studied in experiments with continuously heated simulated corium debris beds. In the SWISS (Sustained Water Interaction with Stairless Steel) program, a 45 kg charge of molten stainless steel was poured into a crucible with a limestone concrete basemat. The effects of instantaneous and delayed water cooling was studied. The SWISS tests indicated that the overlying water pool could remove heat from the debris at a sustained rate of 0.8 Mw/m^2. This was less than the 1.2 Mw/m^2 generated in the debris and consequently the downward erosion of concrete was not stopped. The lack of coolability in SWISS was largely attributed to the facility's small scale and the stable metallic crust which formed at the upper surface. Surface stability of stainless steel prevented cracking and any subsequent water ingression which limited the potential surface heat flux.

Sustained core-concrete interaction oxidic melt tests are being conducted as part of NRC WETCOR tests and EPRI MACE test series. To date, results of only one WETCOR test (WETCOR-1) and two MACE tests (including one scoping experiment and one test of the first test matrix point) have been performed. WETCOR-1 tests simulated the corium charge with a 70 lbm mixture of alumina and calcium oxide heated in a 12 inch crucible. Results of this test were reported in Reference 4.43. This test indicated an initial short period of high heat removal (1.5 Mw/m^2) followed by a longer period of reduced heat removal of about 0.4 Mw/m^2. The reduced heat removal was attributed to a stable crust that formed above the corium. Reference 4.7 suggests that at this small scale stable crusts may be expected and therefore the results are not prototypical of large scale reactor melts.

The MACE tests simulates the corium debris as a mixture of UO_2, ZrO_2 and Zr. To date, the MACE test results have provided mixed results regarding debris coolability. While the MACE scoping test only established a maximum stable heat flux of 0.6 Mw/m^2, it was noted that the details of the test facility may have contributed to providing an insulating debris surface. When the crust was penetrated during the test, heat fluxes were in excess of 2 Mw/m^2. The most Recent MACE test involving 960 lbm of simulated corium (Test 1B) indicated substantial debris quenching and a long duration vigorous heat removal of 2 Mw/m^2 was observed. Six hours into the test, concrete erosion was noted to be between 15 and 20 cm and the erosion rate had reduced to 1 cm/hr. At this time partial corium quenching was observed. A power reduction step was investigated to establish the debris erosion rates 24 hrs after shutdown. At this power level within the debris, the erosion process ceased and the debris conditions stabilized (Reference 4.44).

FIGURE 4.2-1

Comparison of Dryout Heat Flux Data



4.2.2.2.4 Melt Spreading

The ability of the corium melt to spread within a water pool was investigated by Greene (Reference 4.48). These experiments were used to establish a correlation between a nondimensional spreading thickness against a nondimensional spreading number. Applying typical ALWR geometric data into this correlation (see Reference 4.41), suggests that the spreading of the corium debris can be expected to be relatively complete for the ALWR.

Reference 4.41 also indicates that even if full spreading is not realized, the ability to remove heat via the sides of the debris bed would enhance debris coolability. Therefore, while full debris spreading is expected, it is not required in establishing corium coolability.

4.2.2.2.5 Debris Power

The debris power has been noted to have considerable impact on the debris coolability, and the concrete erosion rate and penetration profiles. These items are discussed below:

Debris Coolability

Based on debris bed heat and simulated corium - concrete attack experiments, the ability to quench corium debris is strongly dependent on the heat production rate within the corium pool (See item 4.2.2.2.3 above). Typically, an overlying pool of water at atmospheric conditions has been observed to sustain a heat flux on the upper corium surface of between 0.4 and 0.8 Mw/m^2. (This heat removal rate is less than the approximately 1 Mw/m^2 associated with the flat plate pool boiling CHF.) Thus, termination of core-concrete attack in the long term requires that the net heat production rate within the corium pool due to fission product and actinide decay and exothermic chemical reactions be below about .5 Mw/m^2.

Erosion Rate

The concrete destruction process is thermally driven. As a consequence, the greater the thermal power, the more rapid erosion is expected. For typical RCS powers the erosion rates expected for large dry PWRs are 0.08 to 0.14 cm/min for a dry cavity attack and 0.035 cm/min for a "wet" cavity attack of an uncoolable debris bed (See for example, Reference 4.5). MAAP analyses performed for representative System 80+ Station Blackout Scenarios indicate dry cavity concrete erosion rates will be on the order of .17 cm/min for the early dry cavity attack, to .06 cm/min for the longer stable erosion period.

Erosion Profile

The Beta experiments performed by KfK (References 4.50 and 4.51) have experimentally investigated Core-Concrete interaction for large simulated corium melts at various power levels. These tests allow introduction of simulated inductively heated sustained melts with a large concrete crucible. The crucible was designed so as to allow radial spreading of the corium within the basemat. The Beta tests investigated several parameters related to corium-concrete attack including the effects of debris power, and debris constituents on concrete erosion. Test results clearly indicate a strong dependence of the downward concrete heat flux (which for these tests is directly related to the corium power) and the basemat radial erosion profile. At high powers, Beta tests performed with various concretes suggest that a very effective downward heat transfer mechanism develops and very little radial spreading is observed. At low power levels, a more aggressive radial concrete attack is noted, however, the erosion rate is still considerably below that found for axial erosion. Furthermore, in both instances radial erosion ceases relatively early in the basemat melt-through process while the corium continues to burrow axially downward.

Based on a review of the low power corium concrete attack data for the Beta facility, it appears that the ratio of downward to radial erosion would be about 2:1 early in the transient and increase to well above that value as the core-concrete attack continues. That is, little spreading of the melt within the concrete was observed.

4.2.2.3 Significance to System 80+

The System 80+ reactor cavity has been designed with a large basemat area (consistent with the URD) and a Cavity Flood System (CFS) (See Section 3.6) to ensure the presence of water in the reactor cavity following severe accident scenarios.

The basemat penetration scenario for System 80+ is considered to be relatively benign because of the high likelihood of an overlying water pool, the large surface basemat area for corium spreading and the ample depth of the reactor cavity basemat (more than 20 feet). Furthermore, in the long term (> 12 hours after scram) energy production rates within the corium should enable the corium to be coolable even at the lower experimentally observed values of debris heat removal (See Table 4.2-5). MAAP analyses performed by parametrically varying the level of the pool boiling critical heat flux from a nominal value of about .8 Mw/m^2 (FCHF = .1) down to about .25 Mw/m^2 (FCHF=.03), show that erosion will increase as the debris heat removal limits are decreased. However, the erosion is of limited duration and even under conditions with significant degradation in debris heat removal capability, the erosion is ultimately arrested and the maximum penetration of the corium debris is limited to under 1 foot of concrete. The results of this study are summarized in Table 4.2-6.

It should be noted that, the subsphere design of System 80+ incorporates an offset "below reactor cavity" room, which under certain circumstances associated with a dry reactor cavity can be penetrated. Since the expected progression of the corium is primarily downward, the concrete erosion profile is not expected to extend into the subsphere (See Figure 4.2-2). In the remote possibility that such a corium penetration condition develops, any subsequent containment blowdown into this region will result in an above ground filtered radiation release from the containment.

TABLE 4.2-5

SURFACE HEAT FLUX FROM CORIUM DEBRIS REQUIRED TO PREVENT CONCRETE EROSION AT VARIOUS TIMES AFTER REACTOR SCRAM

TIME AFTER SCRAM	DECAY HEAT ONLY (Mw/m^2)	DECAY HEAT PLUS CHEMICAL REACTIONS* (Mw/m^2)
3 hrs	0.423	0.85
6 hrs	0.325	0.48
12 hrs	0.260	0.26
24 hrs	0.224	0.224

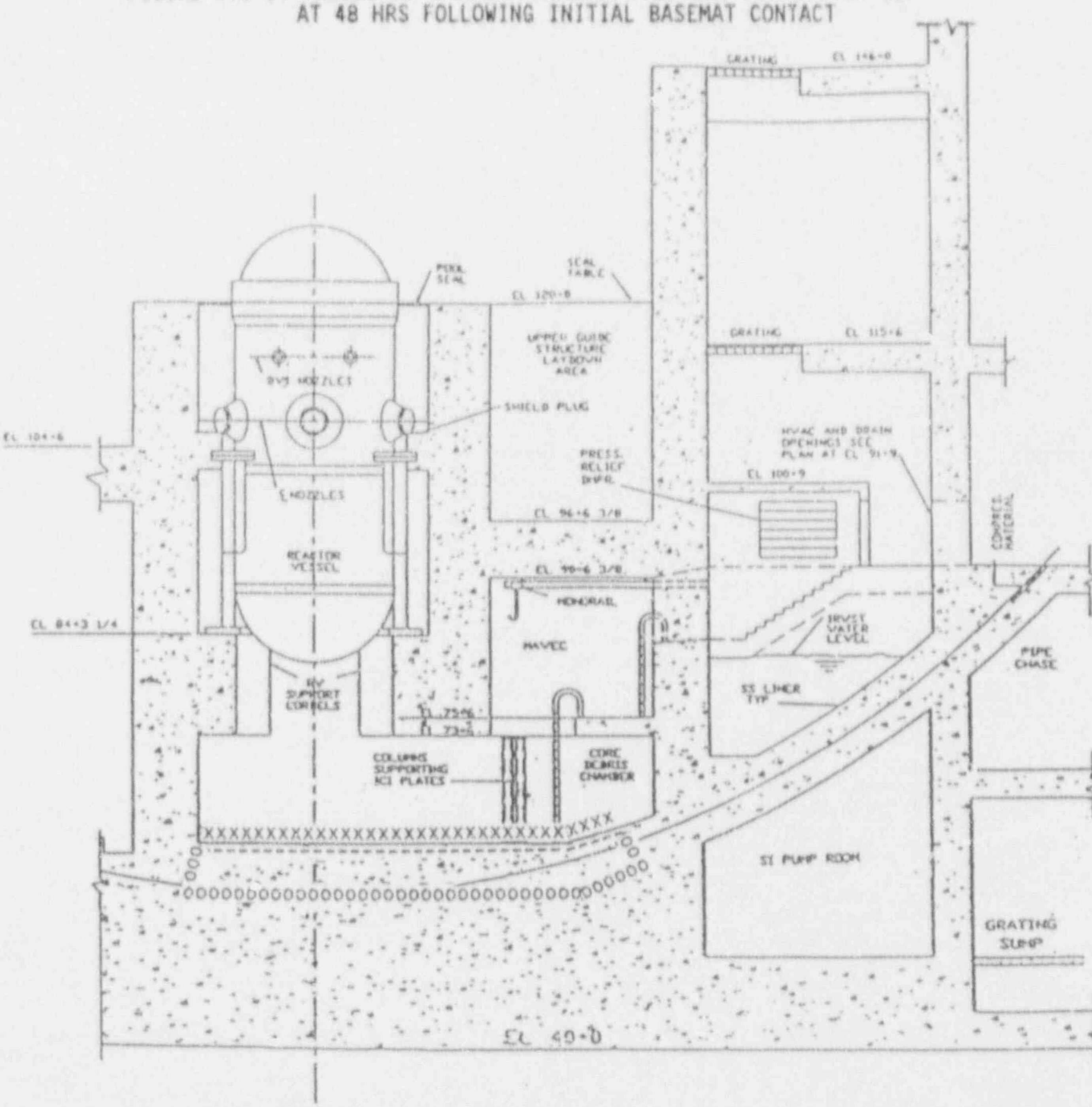
* Contribution due to chemical energy addition is approximate

FIGURE 4.2-6

EFFECT OF DEGRADED HEAT TRANSFER ON CORIUM DEBRIS COOLABILITY

	FCHF=.10	FCHF=.05	FCHF=.03
MAX. EROSION DISTANCE (FT)	0.0	0.087	0.75
TIME CONCRETE ATTACK ENDS (HR)	NA	11.5	15
CONTAINMENT PRESSURE AT 48 HOURS (PSIA)	118	139	145

FIGURE 4.2-2: PROJECTED BASEMAT EROSION PROFILES FOR SYSTEM 80+
AT 48 HRS FOLLOWING INITIAL BASEMAT CONTACT



xxxx Cavity flooded and corium cooled (No erosion)

---- Cavity flooded and corium uncooled (MAAP prediction with FCHR = 0.03)

oooo Dry cavity (MAAP prediction dry cavity)

4.2.2.4 Application to the PRA

In the existing System 80+ PRA (Reference 4.36), it was explicitly assumed that as a consequence of the cavity design, the availability and actuation of the CFS was sufficient to prevent a basemat melt-through scenario. While there is general agreement that water will retard the corium progression into the concrete basemat, there is not yet conclusive proof that a deep corium debris bed will be fully coolable by an overlying water pool. It is expected that the ongoing MACE (Melt/Debris and Coolability Experiment) program will shortly provide this information to confirm the existing PRA position. Until that time, future PRA assessments of System 80+ will allow for the potential for basemat failure in the presence of large quantities of water. It should be noted that while these sequences may progress to basemat melt-through, they will do so very slowly, extending the containment failure process to one week or more.

Dry cavity melt-through scenarios can occur if the CFS is disabled or not actuated. These sequences can result in:

1. basemat melt-through to the containment subsoil

2. corium penetration into the subsphere

3. corium erosion of cavity wall concrete, causing an induced containment failure

For purposes of the PRA radiological release calculations basemat melt-through scenarios will be assigned a benign fission product release classification. Failures into the subsphere or reactor cavity wall failures will be considered as potential atmospheric releases.

4.2.3 TEMPERATURE INDUCED FAILURE OF CONTAINMENT PENETRATION SEALANT

During dry cavity corium attack sequences, the containment atmosphere has the potential to undergo a gradual, but significant temperature transient. Analyses of typical System 80+ accident scenarios suggest that sustained temperatures in excess of 450 F can develop throughout the containment within 48 hours after accident initiation. At these temperature levels several common penetration sealants (e.g.. Nitril, Neoprene) will begin to degrade and potentially result in a localized containment failure. On the other hand, several other penetration sealants less prone to temperature failure are available on the market. By specifying the specific sealant at the time of actual equipment procurement, use of the best material available will be ensured.

4.2.3.1 Significance to System 80+

Specific elastomers for use in the System 80+ penetrations have not been finalized. To minimize the risk of thermal degradation of the penetration sealant, the best possible sealant available at the time of equipment procurement will be selected.

4.2.3.2 Application to the PRA

This failure mechanism is included for purposes of completeness and to allow it to be considered in PRA sensitivity studies to be conducted at a later time.

4.2.4 DELAYED COMBUSTION

4.2.4.1 Description of the Phenomena

Combustion events can occur late in a severe accident scenario due to the production of significant quantities of hydrogen and/or carbon monoxide. Delayed burns that significantly threaten containment can occur during transients where (1) previous hydrogen burns have not occurred and (2) a significant corium-concrete interaction has occurred which resulted in the oxidation of much of the unoxidized metal in the corium thereby producing hydrogen and carbon monoxide.

A delayed combustion event can occur anytime in a severe accident once a sufficient quantity of hydrogen / carbon monoxide is generated and the containment atmosphere is not inerted. The most common scenario where a delayed hydrogen burn can occur is when a hydrogen rich, steam inerted containment is sprayed with water. This process is typically operator initiated and can result in a hydrogen combustion event at pressures just below the steam inerting limit. Because of the large amount of steam and carbon dioxide initially available, the combustion event is far more likely to be a deflagration than a detonation. According to the NRC Final SER on the URD (Reference 4.52), the late hydrogen burn issue can be addressed by considering combustion of the hydrogen equivalent of 100% oxidation of the RCS zirconium. Results of analyses presented in this section indicate that pressures generated during this event will be below the containment Service Level C and are well below the ultimate containment failure pressure.

4.2.4.2 Significance to System 80+

As discussed in Section 4.1, System 80+ is equipped with igniters to burn off steam at low concentrations. These igniters have been demonstrated effective in steam environments and therefore, if actuated sufficiently early in the transient they should serve to fully eliminate any significant hydrogen induced containment threat.

In the event that a significant accumulation of combustible gases develop without the occurrence of early smaller burns, a single large burn corresponding to the ignition of the hydrogen equivalent of 100% zircaloy water reaction, initiated at a system pressure just below the inerting containment steam concentration , will produce an AICC burn pressure of below 140 psia. Burns in this range pose a small, but finite threat to containment integrity.

4.2.4.3 Application to the PRA

A delayed hydrogen burn capable of threatening containment are considered credible only if no previous burns have been assumed. The presence of early burns via deflagration or diffusion flames is assumed to consume sufficient hydrogen to make a large containment threatening burn impossible. In the PRA this implies that early operator actuation of the hydrogen igniters will prevent a late containment threatening burn.

Since quenching of the corium debris will produce hydrogen, as will the concrete attack process, the potential for a high level of hydrogen is assumed.

For situations where the igniters are actuated early during the uncovery sequence and operated continuously, the probability of a containment threatening hydrogen burn was considered to be 0.

For scenarios where igniters were not actuated and no prior burns occurred, the probability of a late containment burn was taken to be 1.0. The zircaloy oxidation level and pressure peak associated with the resultant burn was established as follows:

SUMMARY OF BURN CONDITIONS		
CONTAINMENT CONDITION	FRACTION OF ZR OXIDIZED	PEAK PRESSURE
1. Cavity dry with core-concrete attack	1.00	140
2. Cavity wet and corium quenched	0.75	104
3. Cavity Wet and corium unquenched	1.00	120

4.3 Fission Product Release, Transport, and Retention

The consequences of the severe accident scenario are dependent on the amount of fission products that ultimately are released from the fuel rods into the environment. This information, along with meteorological condition and site demographics will determine the man rem equivalent doses at various off-site locations possible during the various severe accident scenarios. It is a goal of the URD that the ALWR cumulative probability of releases greater than 25 rem one half mile from the site boundary be less than 10^{-6}/year.

4.3.1 Models for Fission Product Release and Depletion

The fission product release and distribution models for use in the System 80+ PRA are primarily based on MAAP 3.0B Rev 16. The models governing fission product release are contained in MAAP subroutines FPRATP and METOXA. MAAP also contains models that simulate all significant fission product transport and deposition due to both natural depletion and engineered safeguards systems. These models are contained in MAAP subroutines FPTRAN, FPTRNP.

4.3.1.1 Fission Product Release

Work performed on the development of the ALWR Evolutionary Plant source term (Reference 4.45) has come to several significant conclusions regarding fission product release from fuel rods. Based on investigations performed under this effort the fission product releases from melted and unmelted fuel can be estimated as follows:

ANTICIPATED FISSION PRODUCT RELEASES		
FISSION PRODUCT	MOLTEN FUEL RELEASE	UNMELTED FUEL RELEASE
NOBEL GASES	100%	25%
CESIUM AND IODINE	90%	25%
TELLURIUM	*	
SEMI-VOLATILES: BARIUM, STRONTIUM, RUTHENIUM, AND ANTIMONY	<1%	
REMAINDER	<.01%	

* When the local oxidation of zircaloy is equivalent to less than about 70% of the active clad, the release rate of Te is about 1/40 of that due to iodine and cesium, but equivalent to that of Cs and Iodine when the zircaloy oxidation exceeds 70% of active clad.

Fission product releases are modeled in MAAP subroutines FPRATP and METOXA. Fission product releases can be computed via either the steam oxidation Cubicciotti model or by NUREG-0772 models (Reference 4.46). Tellurium bonding to Zr can be approximated in MAAP through user input by setting ITEREL=0 and preventing "in-vessel" Te releases.

4.3.1.2 Fission Product Speciation in the Containment Environment

The ability for natural depletion and the engineered safeguards to remove fission products from the containment environment is dependent upon the airborne form of the fission products. This is particularly true for cesium (Cs) and iodine (I). Experimental evidence presented in Reference 4.45, suggest that airborne Cs will exist entirely as a water soluble particulate aerosol. Iodine will exist as a mixture of particulate iodine, primarily CsI,(97 %), elemental iodine (2.85 %), and organic iodine (0.15 %).

4.3.1.3 Fission Product Retention in the RCS

Fission products released from the fuel during core damage events will be affected by physical and chemical processes during the transport through the RCS which in turn affect the retention of aerosols in the RCS. Retention of fission product aerosols in the RCS can have an important effect on the ultimate source term.

Thermodynamic analysis and experimental evidence indicate that iodine, cesium and the less volatile radionuclides released from the fuel during core damage accidents in LWRs will behave primarily as aerosols. These aerosols will experience forces that deposit substantial fractions of the material on RCS surfaces or water reservoirs.

Experimental evidence of aerosol RCS retention processes is provided by the LACE and Marviken aerosol transport tests and the INEL severe core damage test series and LOFT test FP-2 (Reference 4.45). In general, the experiments consistently demonstrated high levels of fission product deposition within the RCS with significant levels of deposition noted in the first few meters close to the source.

Based on NUREG-1150 expert judgement elicitation, fission product retention was classified for various PWR sequences as shown in Table 4.3-1. The expert elicitation indicated that high pressure sequences were estimated to have nearly complete fission product retention while for low pressure sequences about 50 % of the released fission products were retained in the RCS.

4.3.1.4 Fission Product Removal In Containment

Following a severe accident, the containment environment can potentially contain significant quantities of noble gases , cesium and iodine and to a lesser extent tellurium. Trace amounts of other radionuclides may also be present. The distribution of fission products remaining in the atmosphere as the accident progresses is dependent upon the chemical form of the various fission products, the precise pathway the various species take to enter the

TABLE 4.3-1

NUREG-1150 EXPERT ELICITATION MEDIAN RETENTION FACTORS

CASE	CONDITIONS	IODINE	CESIUM	LOW VOLATILITY AEROSOLS
PWR1	PORV/PSV Setpoint Pressure	.91	.96	.97
PWR 2/3	High/ Intermediate Pressure	.59	.71	.76
PWR4	Low Pressure	.48	.60	.66

containment, the time after release and the presence of active engineered safeguards (containment sprays) to scrub fission products from the containment atmosphere.

Noble gases cannot be scrubbed from the containment atmosphere, and are relatively insoluble in water. Thus, once released to the containment, the only mechanism for changing the radiological makeup of these gases is via the decay chain. Cesium and iodine are considered to be primarily in the form of water soluble particulates. Thus, aerosol removal will be influenced by the natural depletion processes associated with sedimentation and diffusiophoresis and will be significantly affected by the operation of containment sprays and passage through water pools. Typical decontamination factors (DFs) associated with the use of sprays are between 30 and 50. Passage of scrubbable fission products through the IRWST pool or a flooded cavity will have a decontamination factor of about 100 if the pool is subcooled and between 10 and 20 if the pool is saturated.

4.3.1.4 Revaporization Processes

Because of the low vapor pressure of fission products such as CsI, CsOH,etc. they may condense to aerosol form and after being released from the high temperature fuel and "plate - out" in low temperature regions of the RCS and containment. For those conditions in which the deposited aerosol would be on a dry, uncooled surface, energy generated by the fission product decay may be sufficient to reheat the deposited fission products to revaporize. Revaporization is modeled in MAAP. However, MAAP predicted revaporization rates can be affected by user input and in particular the estimate of the "not-through insulation" heat losses. Large heat losses reflect a good ability to reject heat and decrease the potential for revolatilization. The System 80+ base MAAP model will establish "not through insulation" heat losses based on System 80 heat losses. In the final PRA sensitivity analyses on the effect of revaporization models on the radiation release will be performed.

4.3.2 Significance to System 80+

System 80+ is designed to minimize fission product release to the containment atmosphere by passing all discharges from the pressurizer safety valves and SDS valves through piping submerged deeply into a subcooled IRWST water pool. The effectiveness of fission product removal via overlying water pools can be significant. The Reactor Safety Study (WASH-1400) assumed the equivalent decontamination factor (DF) for a subcooled water pool to be 100 while saturated water pools were not credited for decontamination. More recent experimental evidence suggests that pool DFs can be larger for subcooled pools and that saturated pools can significantly contribute to decontamination.

The cavity flood system is also allowed to direct the IRWST liquid into the cavity to submerge, decontaminate and cool the corium debris.

The containment spray system is intended to both cool the containment atmosphere and scrub the atmosphere of fission products. This system has been designed to be highly reliable since it serves the combined heat removal function of the fan cooler/containment spray system of conventional PWRs. Sprays have been demonstrated to be highly effective in scrubbing particulate fission products from the containment atmosphere.

4.3.3 Application to the PRA

The PRA will estimate the environmental releases from the containment based on a group of bounding MAAP calculations representing general fission product release classes. MAAP models approximately incorporate all the above fission product release, retention and atmospheric depletion processes. This information will form the base System 80+ airborne concentrations which are to be applied to MACCS5.1 so that population dose estimates can be established. To demonstrate the uncertainty associated with this process several sensitivity/scoping studies will be performed using alternate approaches to fission product release and transport such as that developed by NRC consultants from the XSOR codes (Reference 4.47) and modified via Reference 4.45 and/or by performing limited parametric studies with MAAP model spray efficiency, aerosol agglomeration and late volatilization release processes.

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5.0 SYSTEM 80+ CONTAINMENT PERFORMANCE FOR SELECTED SEQUENCES

5.1 INTRODUCTION

This section provides a quantitative description of the System 80+ anticipated plant transient response and containment performance for representative severe accidents initiated from a station blackout sequence. The analyses have been performed with an enhanced version of MAAP 3.0B Rev 16 and are provided to demonstrate the unique features of the System 80+ design.

5.2 MODIFICATIONS TO MAAP 3.0B

Quantitative severe accident analyses are performed using a System 80+ version of MAAP 3.0B Rev 16.03 (Reference 5.1). Code modifications performed were required to simulate unique design features of System 80+. These models included:

1. The addition of In-Containment Refueling Water Storage Tank

2. A new Cavity Flooding System model

3. Changes to the Engineered Safety Features Systems to accommodate new ESF system line-ups

4. Detailed reactor cavity volume model

The IRWST model was written to replace the quench tank model in MAAP 3.0B. This was a logical model exchange since the IRWST receives water from the Pressurizer Safety Valves and Rapid Depressurization Valves, as would the quench tank in a conventional PWR. The model was developed to include all appropriate liquid and gas flow paths and considers hydrogen accumulation and combustion in the IRWST freeboard space, as well as, fission product scrubbing of the safety and SDS valve discharge in the IRWST water pool.

Cavity flooding was simulated employing a hydraulic model connecting the IRWST, Holdup Volume and Reactor Cavity. All flows into and out of these volumes were considered in the model formulation. Once actuated, the flooding of the reactor cavity is a passive process driven by the density heads developed in the IRWST and Holdup Volume.

The System 80+ engineered safeguards line-ups are similar to those used on contemporary C-E PWRs. The introduction of the IRWST into the evolutionary System 80+ design required modifications/additions to the containment suction and RHR heat removal models.

In order to accommodate the cavity flood model several code modifications were necessary to the reactor cavity models to both represent new flow paths and more rigorously consider reactor cavity volume distribution.

These models were typically verified by reviewing code changes and comparing predicted results to alternate hand and/or computer calculations.

5.3 SYSTEM 80+ SEVERE ACCIDENT TRANSIENT ANALYSES

The analyses selected for presentation in this section were chosen to provide the reader with a fundamental understanding of the System 80+ severe accident containment performance. This information is intended to supplement severe accident response descriptions presented in the body of the PRA.

5.3.1 Station Blackout Sequence with Battery Power Available and Cavity Flood System Actuated

The station blackout sequence consists of a total loss of all AC power. Station batteries are assumed available for only 4 hours. During this time the battery power is primarily directed towards maintaining auxiliary feedwater flow to the steam generators. Prior to battery depletion the operator floods the reactor cavity to ensure debris quenching and debris coolability following a potential failure of the RV lower head. The unavailability of containment heat removal results in an overpressure containment failure about two days after the initial Loss of Offsite Power (LOOP) condition.

5.3.1.1 Dynamic Response

In the SBO scenario the loss of power causes the control rods to drop into the core terminating the nuclear chain reaction. Since batteries are available to control the auxiliary feedwater pumps for several hours, core cooling can be temporarily maintained along with RCS pressure and inventory control. For this transient, auxiliary feedwater is assumed lost at 4 hours. (This time is conservative since battery management procedures will extend this time to at least 8 hours.)

5.3.1.1.1 RCS Plant Response

This scenario consists of an extended loss of AC power. The SBO results in unavailability of all engineered safeguards with the exception of auxiliary feedwater which is supplied via a steam turbine and electrically controlled via inverters. As a result of core heat removal via the steam generators the RCS pressure is maintained below 2200 psia. Approximately six hours into the event (almost two hours after all AFW is lost) the steam generators dry out and heat removal from the RCS is lost (See Figure 5.3.1-1). Loss of heat removal results in a repressurization of the RCS to the SRV setpoint pressure (See Figure 5.3.1-2). The cycling of the SRVs allows for an unreplenished loss of RCS inventory and incipient core uncovery at 8 hours (See Figure 5.3.1-3). Without any engineered safeguards operational the fuel rapidly heats up, melts, relocates to the lower plenum and fails the RV lower head. The RV failure mechanism is assumed to initially be failure of a single lower head penetration, opening an initial 0.052 ft radius hole in the RV lower head.

5.3.1.1.2 Containment Performance

The containment response during the SBO demonstrates the passive plant capabilities of System 80+. The upper containment temperature and pressures for this event are presented in Figures 5.3.1-4 and 5.3.1-5, respectively. Pressures and temperatures in other containment locations are similar. In the System 80+ design passive heat removal from the containment atmosphere is accomplished through heatup of the following heat sinks:

1. IRWST Inventory (222 million lbm)
2. Internal Structural Concrete (200 000 ft^2 surface area)
3. Internal Steel (gratings, polar crane, etc.)
4. Containment Shell and heat transfer to the Secondary Containment

For the SBO scenario, discharges from the primary system are ducted via the pressurizer pressure relief piping into the IRWST. During the early transient, steam discharged into the IRWST is ultimately condensed. Therefore, containment pressures remain near initial conditions until the IRWST reaches saturation conditions. At that time the IRWST water begins to boil, adding steam mass into the containment atmosphere. Without available containment heat removal systems, the steam addition is seen to directly result in a small containment pressure increase. At vessel breach, a rapid (but modest) containment pressurization is observed. This is due to the release of considerable quantities of steam and corium that are discharged into the reactor cavity.

It is expected that the operator will actuate the cavity flood system prior to vessel breach. The large floor area available for spreading the corium within the cavity results in a high confidence that the corium remaining within the cavity will be quenched. (For these analyses the corium heat removal rate at the upper corium surface is limited to about 70 % of the Zuber pool boiling heat flux.) Thus, the primary containment threat under these conditions becomes the gradual overpressurization of the containment due to vaporization of the water covering the corium. In the absence of containment heat removal, the steam will continue to gradually pressurize and heat up the containment atmosphere to the point of containment failure. Based on MAAP analyses of the containment pressure response the time required for the containment atmosphere to reach ASME Service Level C Limits will be about 60 hrs from the onset of the SBO (See Figure 5.3.1-4). (Longer times to reach the ASME Level C limits are expected for the more realistic 8 hour battery management scenario.) Ultimate stress levels (based on 1% strain in the containment shell) will not be reached for another 20 hours. Containment temperatures during this heatup process are illustrated in Figure 5.3.1-5. As can be seen containment temperatures are generally below temperature levels which would induce rapid degradation of the penetration sealant.

5.3.2 Station Blackout Sequence with Battery Power Available and Cavity Flood System Unavailable

The station blackout sequence consists of a total loss of all AC power. Station batteries are assumed available for only 4 hours. During this time the battery power is primarily directed towards maintaining auxiliary feedwater flow to the steam generators. In this scenario, the operator fails to actuate the CFS and RV failure occurs in the presence of a dry cavity. Initially, the RV lower head failure results in the deposition of a limited water mass into the reactor cavity. The sources of this inventory are the residual lower plenum liquid and the SIT inventory. The cavity remains wet and the debris is cool until the time of cavity dryout, about 16 hours (See Figure 5.3.2-1).

The dryout of the cavity results in an uncooled corium debris bed and significant core-concrete interaction. The unavailability of corium cooling in the reactor cavity results in an aggressive basemat erosion and an associated release of non-condensable gases. High temperatures in the containment may also attack containment penetration seals. As a result of the multiple attacks on containment integrity during these scenarios, the precise mechanism for containment failure is uncertain. Potential containment failures may be caused by either:

1. Basemat melt-through into the containment subsoil
2. Basemat melt-through into an SI pump room in the subsphere of the containment building
3. Basemat melt-through reactor cavity wall collapse
4. Temperature induced seal failure
5. Overpressure due to a combination of processes including corium concrete attack, concrete outgasing and containment atmosphere heatup.

These failure mechanisms are discussed below with reference to MAAP SBO analyses.

5.3.2.1 Basemat Melt-through Scenarios

As a result of the System 80 + design, the release of corium into a permanently dry cavity is considered remote. Under these circumstances an unmitigated corium concrete attack is expected to continue until either the basemat is penetrated and vitrifies in the basemat subsoil or the SI pump room in the auxiliary subsphere is penetrated. MAAP analyses provide an approximate timing of the basemat melt-through. Based on this failure mode it is estimated that basemat penetration of about 20 feet will require more than 200 hours for a limestone/common sand basemat and about 180 hours for a basemat constructed from basaltic concrete (See for example Figure 5.3.2-2). If the standard eight hour batteries were assumed in the analysis, basemat melt-through could be delayed an additional 50 to 100 hours.

The radial penetration of the corium is difficult to ascertain. Based on the Beta core concrete interaction experiments it appeared that initially the corium attack into the concrete would erode laterally at a rate of between 20 to 50% of the downward erosion rate. Conservatively assuming that these wall erosion rates are constant, corium entry into the SI pump room will be delayed beyond 100 hours following the initiation of corium concrete attack. It should be noted that experiments suggest lateral erosion rates rapidly become asymptotic, potentially eliminating the possibility of SI room penetration. The different consequences of the basemat erosion scenarios are significant in that they lead to different treatments in the PRA. A complete basemat penetration into the containment subsoil is considered to have negligible radiological consequences to the surrounding communities. Whereas, the corium penetration into the auxiliary building SI room is considered to be a partially filtered above ground radiological release.

Radial erosion of the concrete basemat may potentially cause the collapse of the Reactor Cavity walls, causing a significant displacement of the Reactor Vessel and associated RCS piping. The potential for these displacements causing containment penetration failures will be considered in the PRA.

5.3.2.2 Containment Overpressure Failure

For the SBO scenario where the IRWST is not actuated to flood the reactor cavity long term pressurization of the containment can come from a variety of sources, including:

1. Boiling of water in the IRWST prior to RV failure
2. Non-Condensible gases (CO,CO_2,H_2) generated via Core-Concrete interaction
3. Release of residual steam/water inventory in the RV at the time of lower head failure
4. Thermal Dehydration of free water from unlined concrete surfaces
5. Heatup of the containment atmosphere via radiatio nd convection to the containment atmosphere

MAAP analyses provide some guidance with respect to this heatup process. However, because of the high temperatures predicted to occur in the containment and large surface area of unlined concrete available for concrete outgassing, a potentially significant contributor to in containment steam release is not considered. However, based on a review of MAAP analyses containment integrity will not be compromised by non-condensable gas generation (See Figure 5.3.2-3).

5.3.2.3 High Temperature Failure of Penetration Seals

The high containment temperatures associated with dry cavity basemat attack sequences (see Figure 5.3.2-4) will challenge the performance of containment penetration seals. While specific penetration sealant materials have not been specified for System 80+, at temperatures above 450 F even high quality seals will begin to degrade with continuous exposure to a hostile environment. Typical seal lifetimes under these environmental conditions will be between 50 and 500 hours. Thus, for certain scenarios, it is possible for a high temperature seal failure to precede a complete basemat melt-through.

5.4 SUMMARY

Thermal-Hydraulic responses for various representative System 80+ station blackout sequences have been presented along with potential consequences and competing failure modes. These evaluations demonstrate that even for SBO sequences with a hypothetical reduced battery availability the containment failure times following station blackout scenarios are ample (greater than 48 hours from transient initiation). This time frame is consistent with the NRC goal of guaranteeing containment integrity for times greater than 24 hours.

5.5 REFERENCES

5.1 MAAP 3.0B Rev. 16.03, Fauske and Associates, Inc.

FIGURE 5.3.1-1

SYSTEM 80+ : STATION BLACKOUT WITH "WET" CAVITY

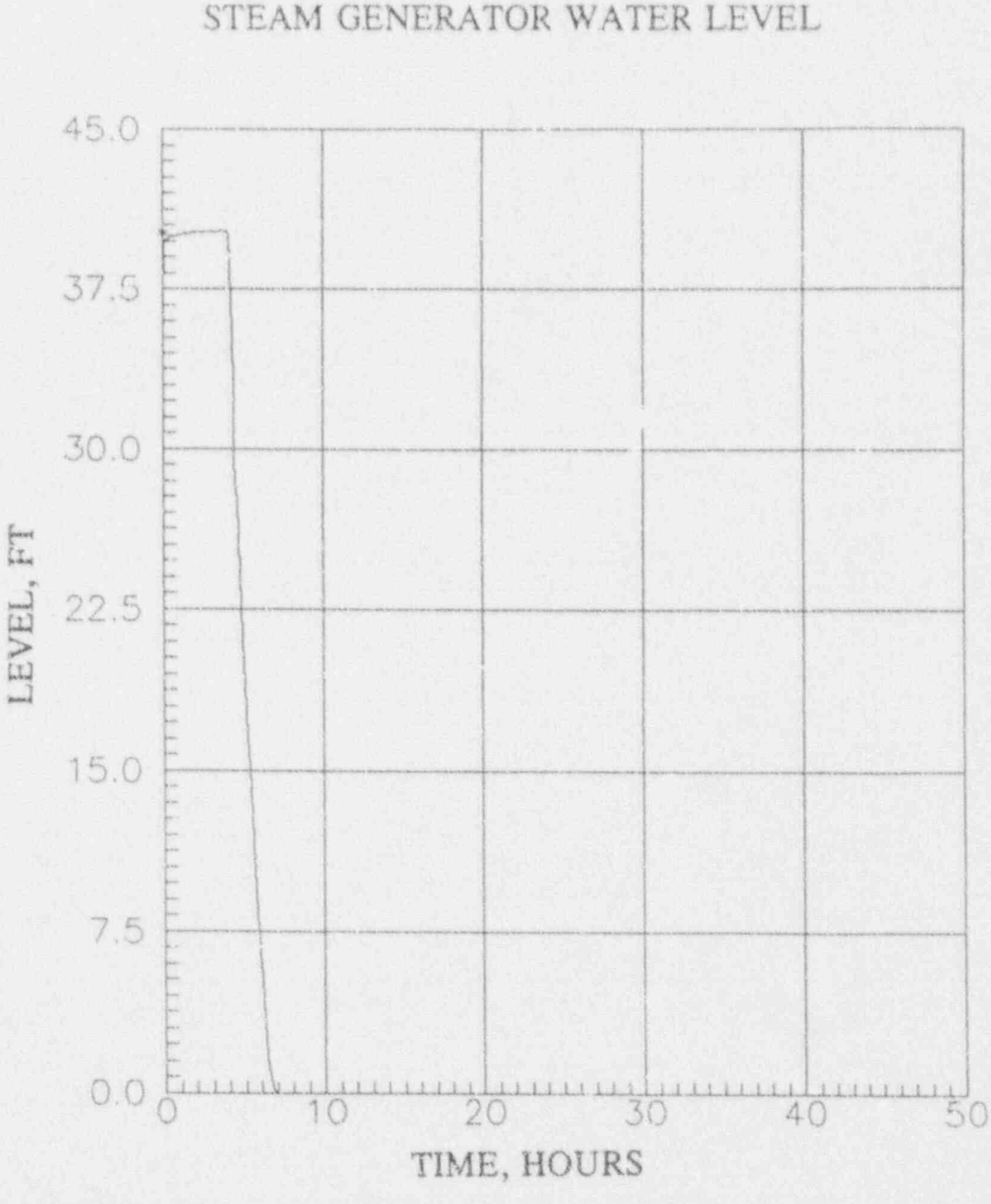


FIGURE 5.3.1-2

SYSTEM 80+ : STATION BLACKOUT WITH "WET" CAVITY

PRIMARY SYSTEM PRESSURE

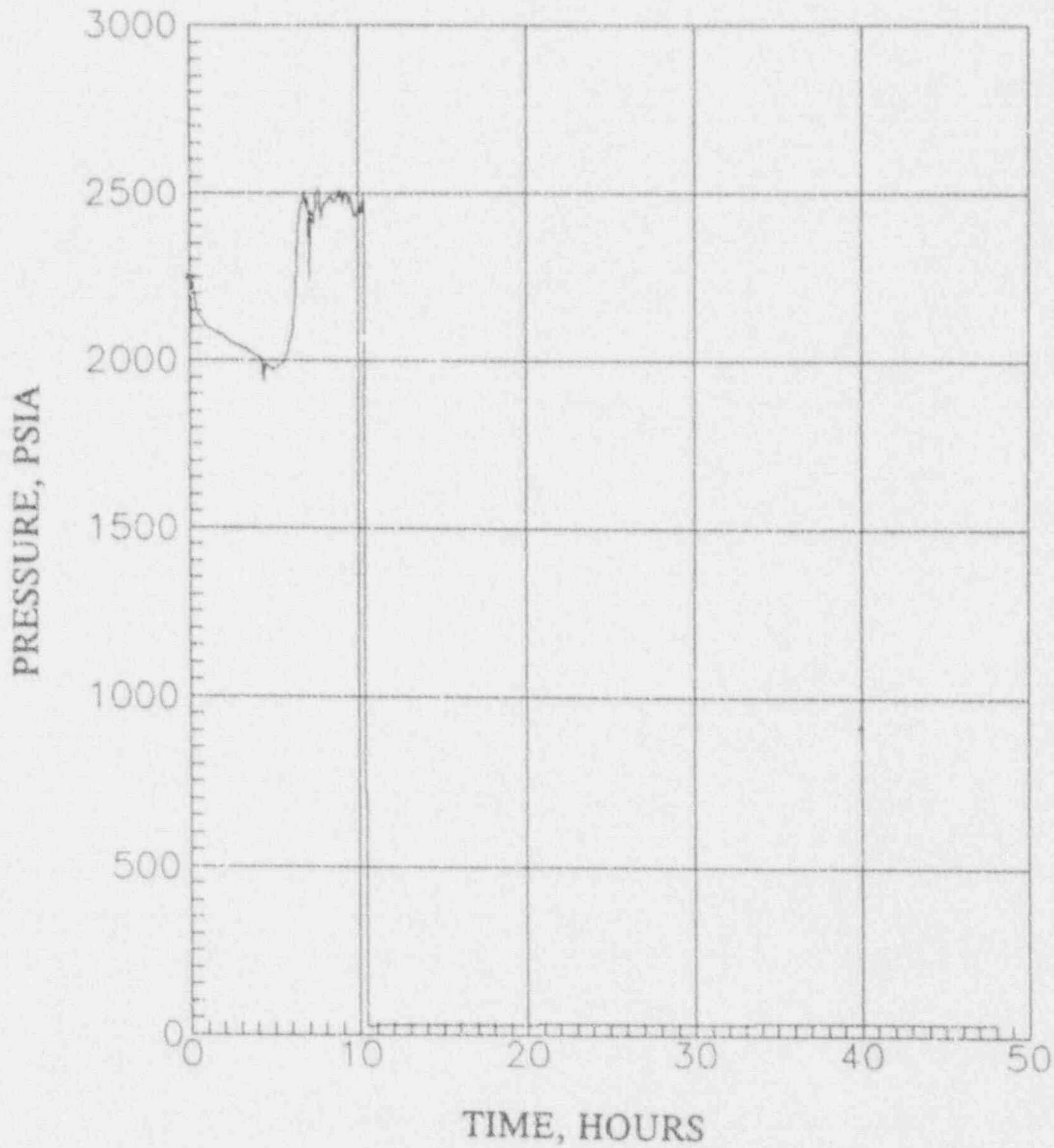


FIGURE 5.3.1-3

SYSTEM 80+ : STATION BLACKOUT WITH "WET" CAVITY

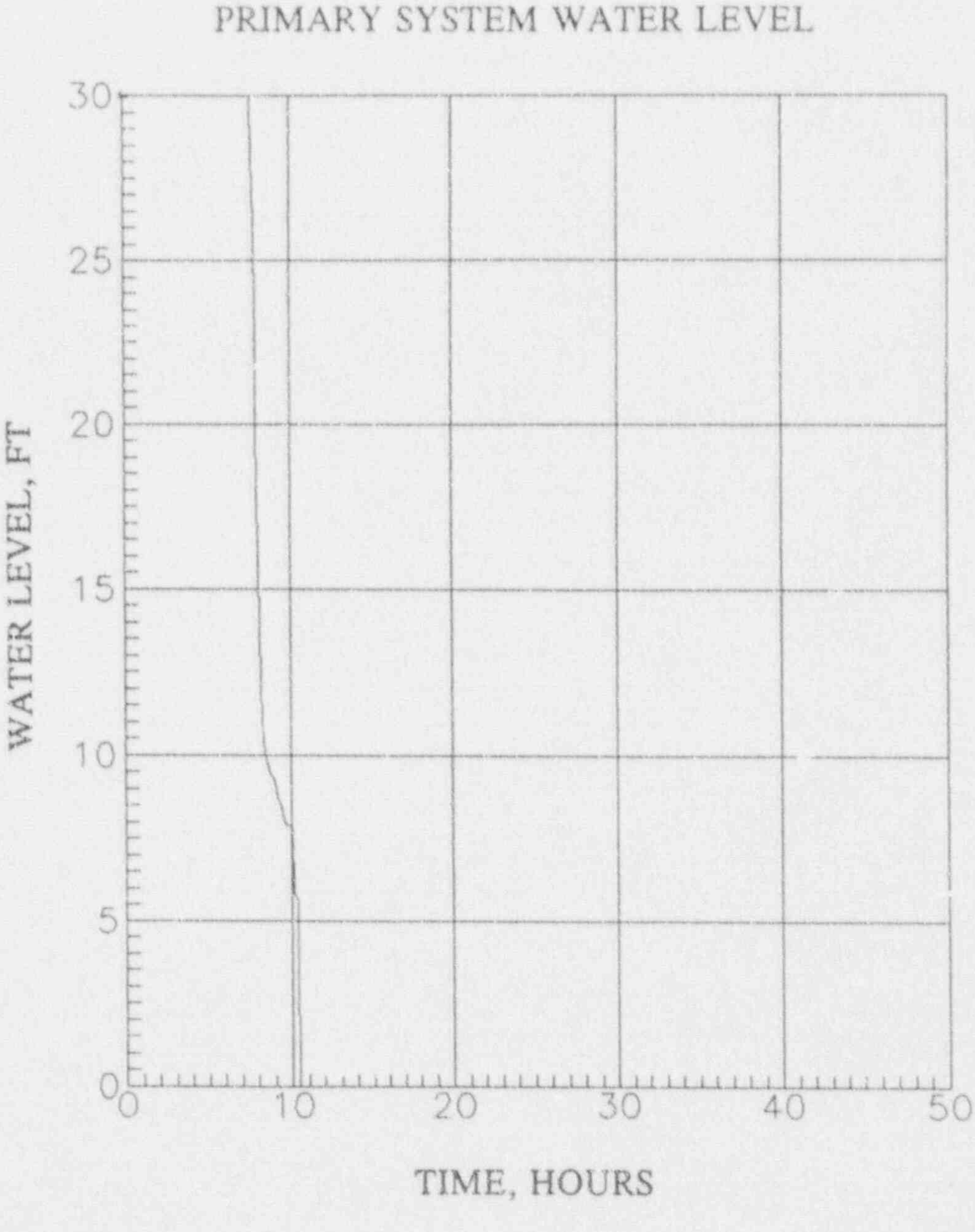


FIGURE 5.3.1-4

SYSTEM 80+ : STATION BLACKOUT WITH "WET" CAVITY

CONTAINMENT TEMPERATURE

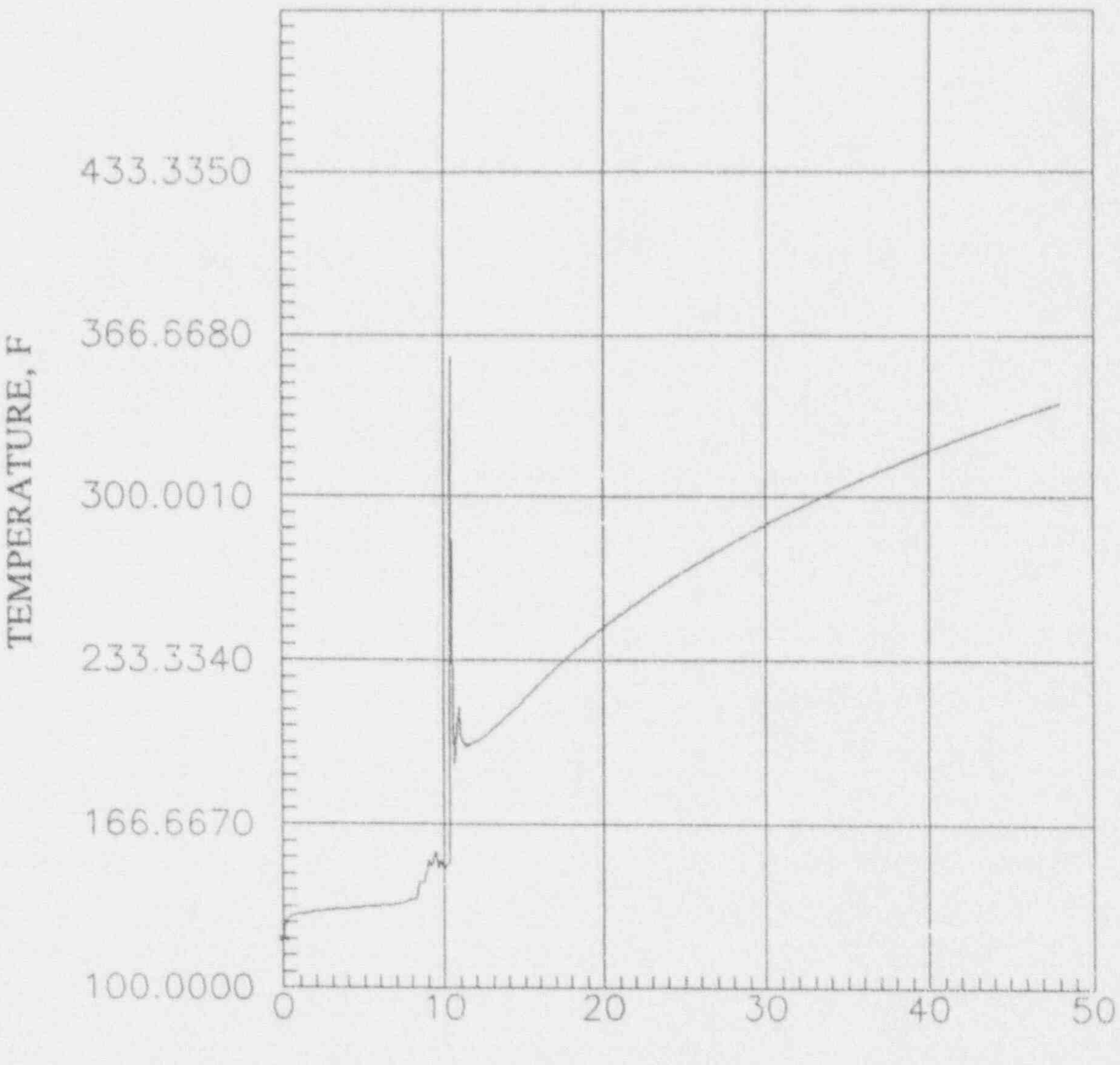


FIGURE 5.3.1-5

SYSTEM 80+ : STATION BLACKOUT WITH "WET" CAVITY

CONTAINMENT PRESSURE

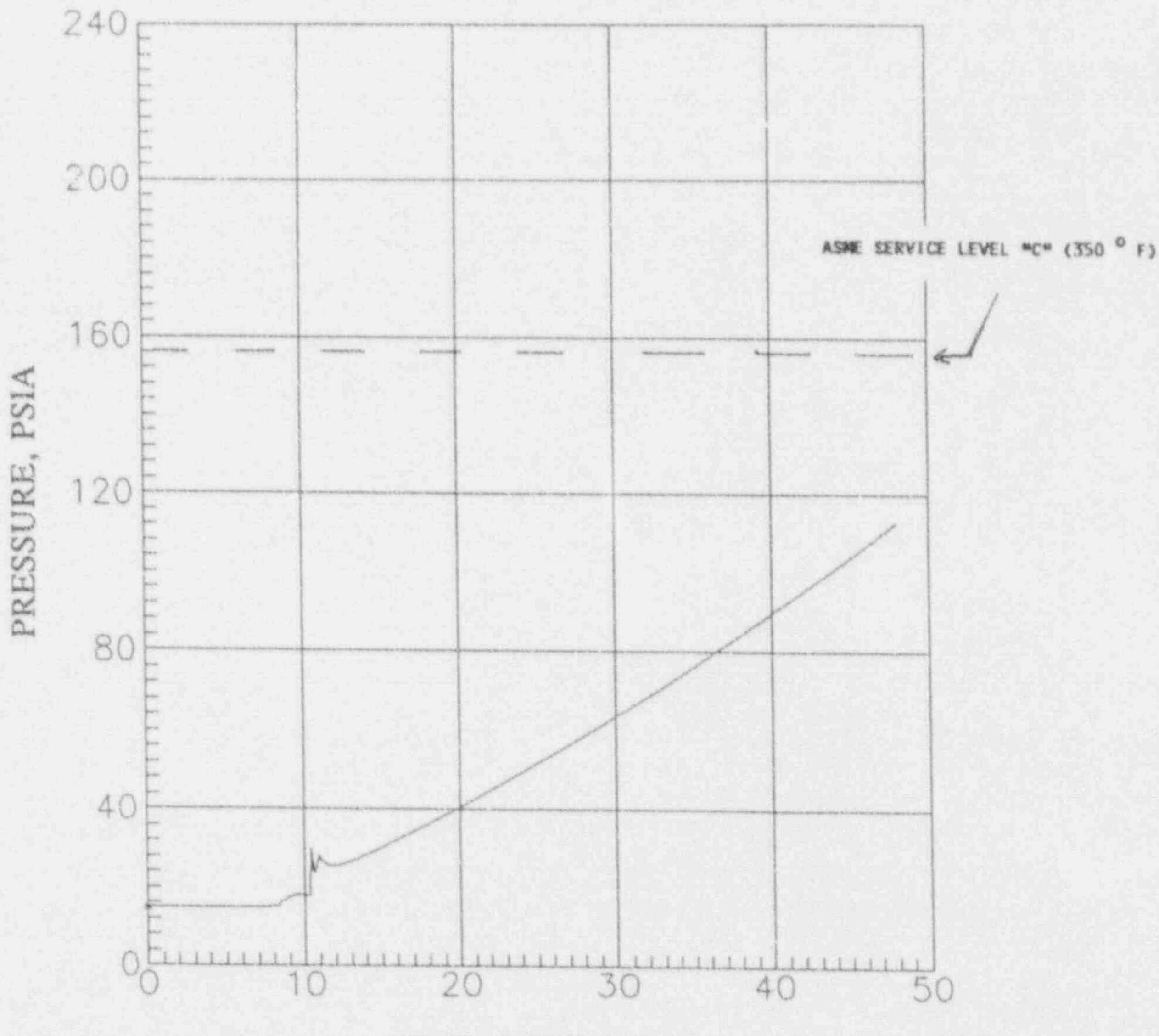


FIGURE 5.3.2-1

SYSTEM 80+ : STATION BLACKOUT WITH "DRY" CAVITY

WATER INVENTORY IN THE REACTOR CAVITY

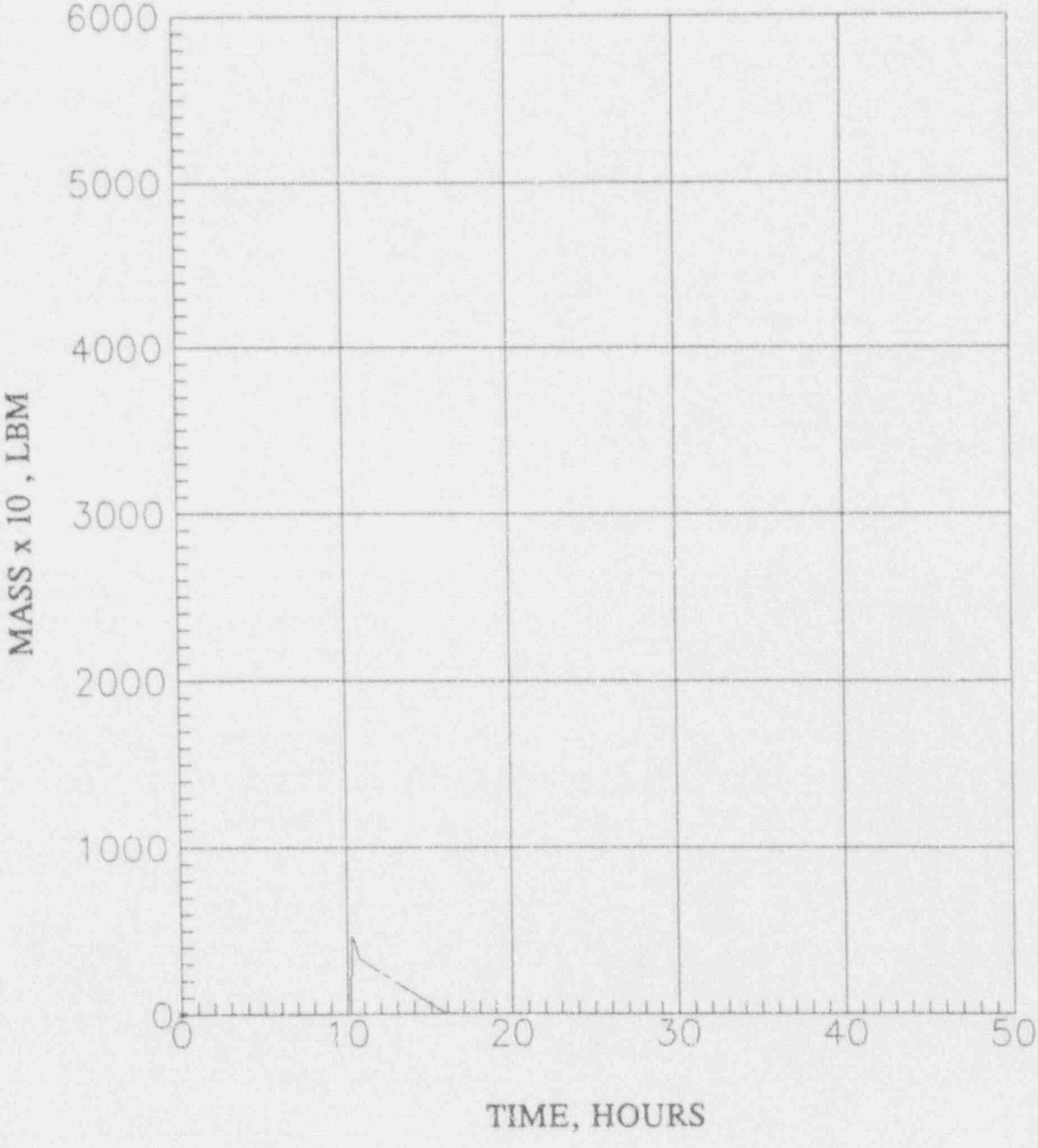


FIGURE 5.3.2-2

SYSTEM 80+ : STATION BLACKOUT WITH "DRY" CAVITY

BASEMAT EROSION

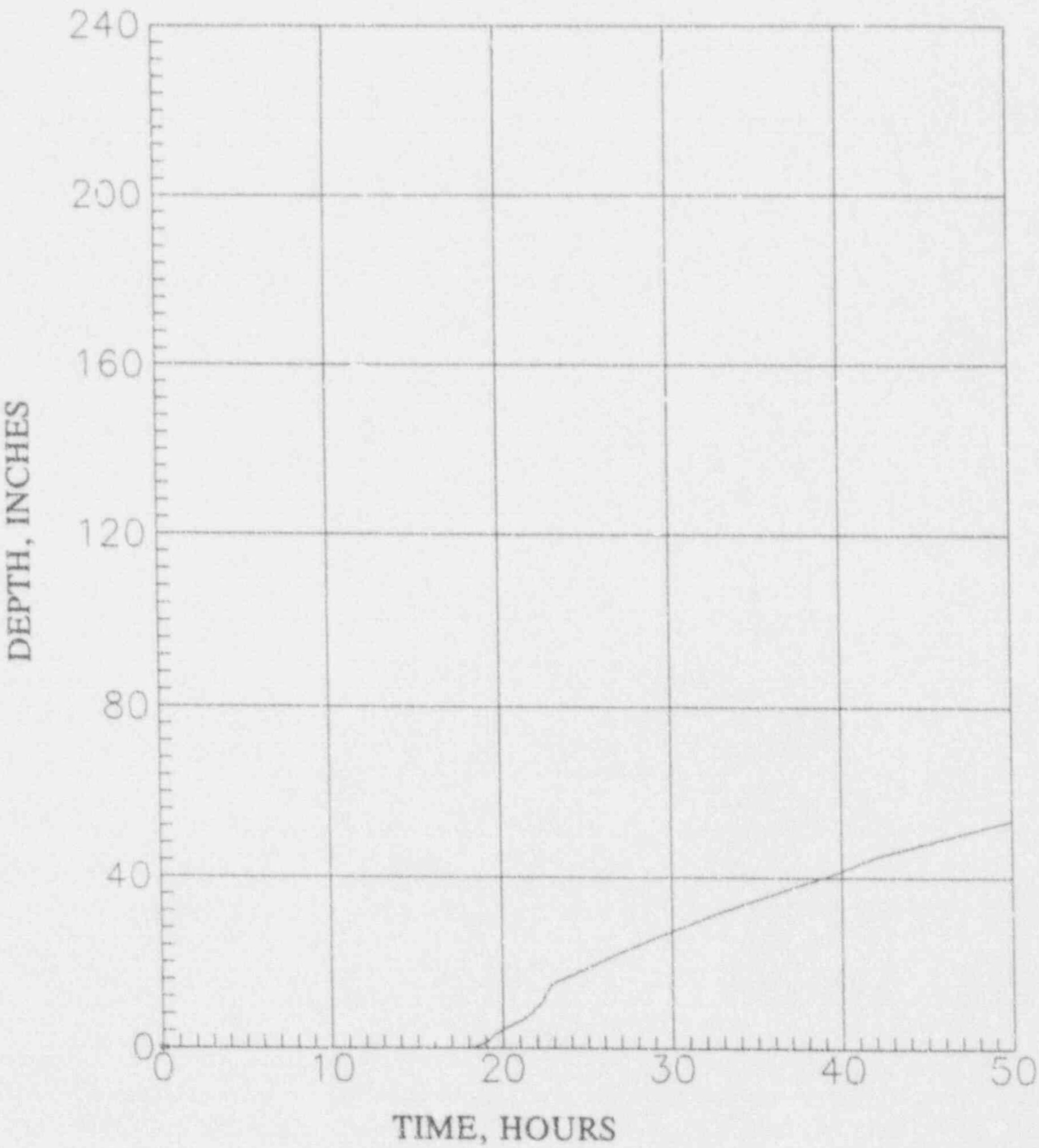


FIGURE 5.3.2-3

SYSTEM 80+ : STATION BLACKOUT WITH "DRY" CAVITY

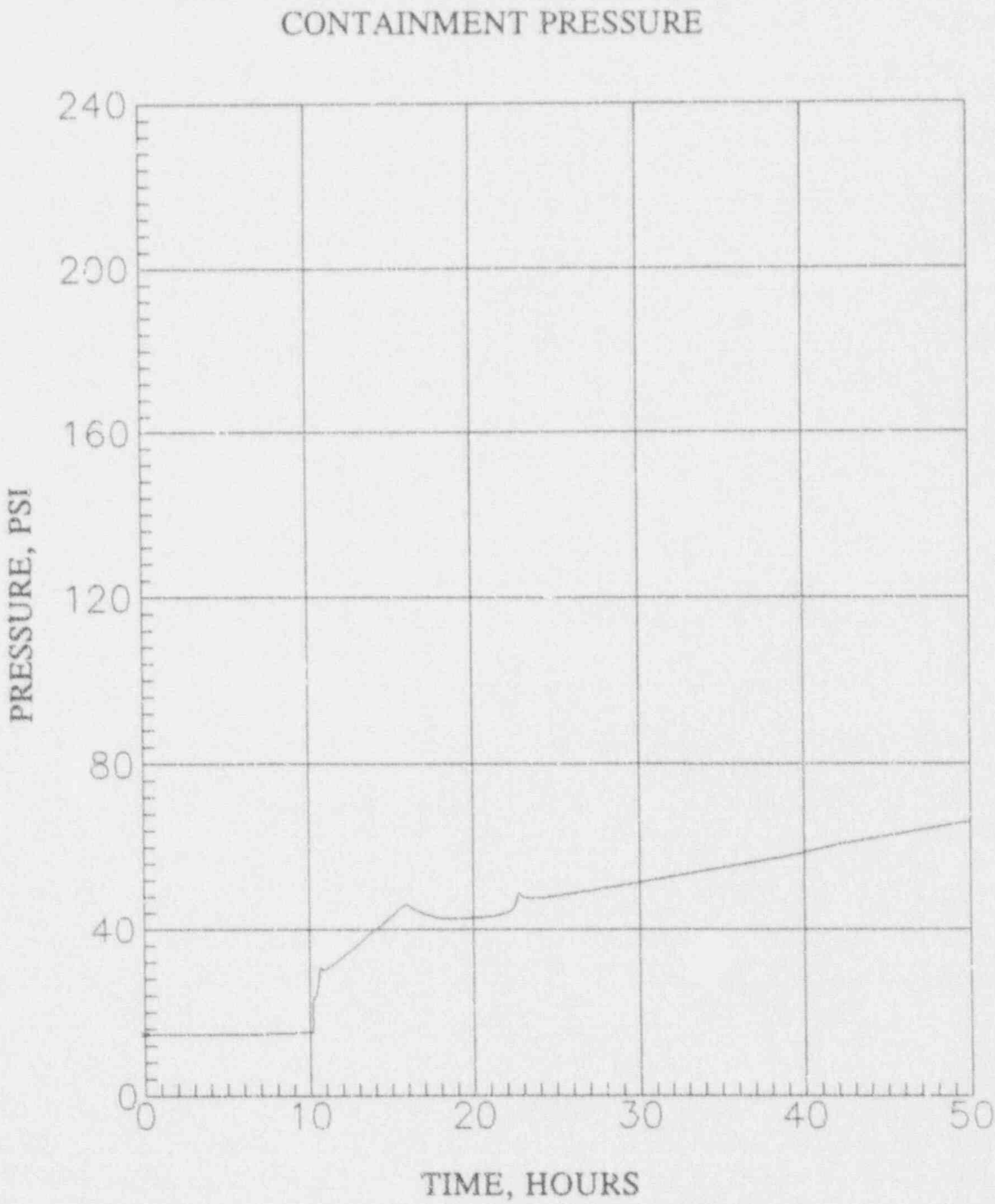
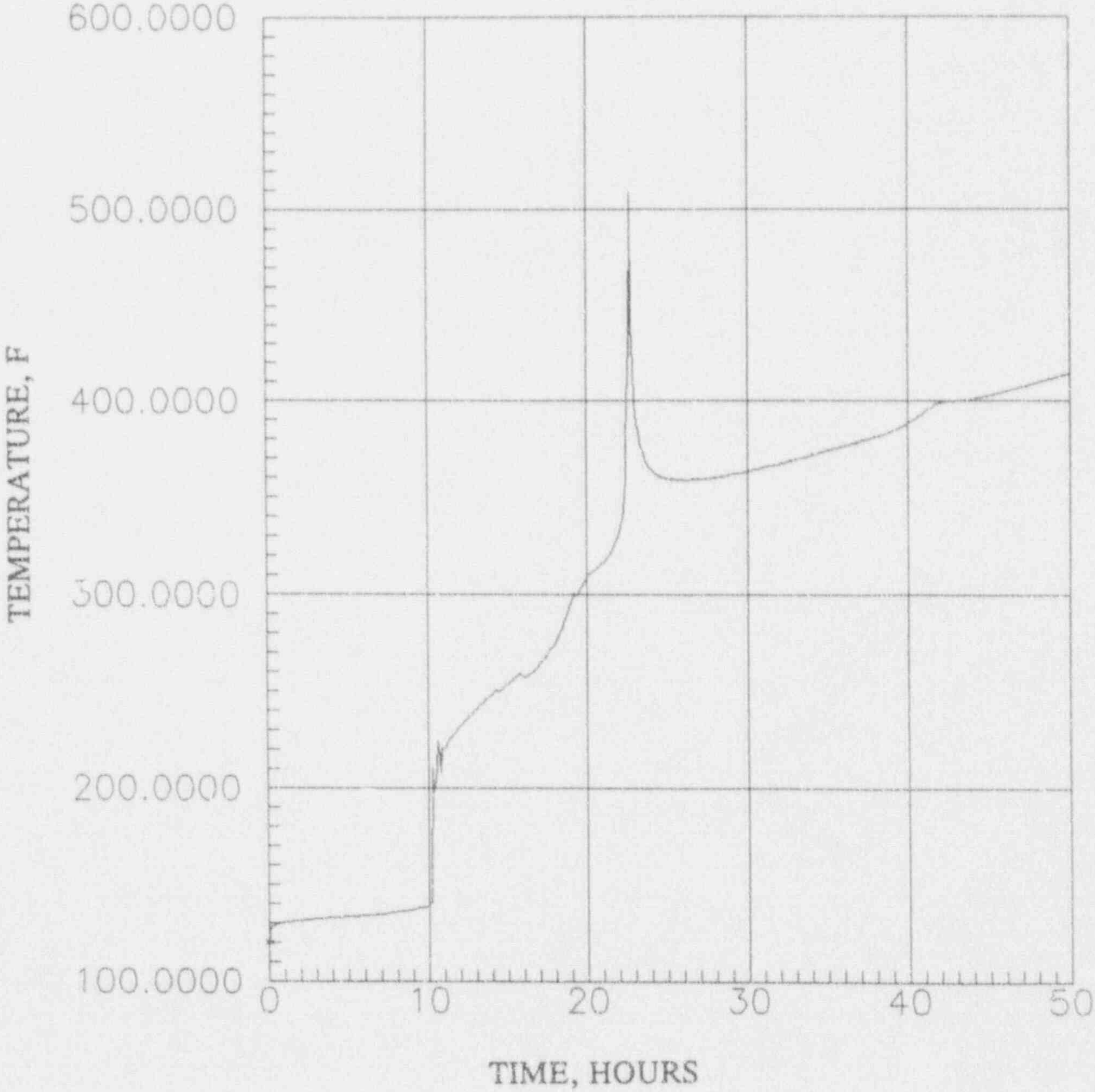


FIGURE 5.3.2-4

SYSTEM 80+ : STATION BLACKOUT WITH "DRY" CAVITY

CONTAINMENT TEMPERATURE



6.0 SUMMARY AND CONCLUSIONS

The severe accident mitigation features of the System 80+ design have been described along with their impact on the phenomenological response of the plant to beyond design basis accidents. Where applicable, conformance of these features with the requirements of the EPRI ALWR Utility Requirements Document has been highlighted. The application of the phenomenological response of the plant to the PRA is discussed in the context of the System 80+ PRA.

Analyses of severe accidents using the MAAP code were carried out to demonstrate the mitigative capabilities of the design features and to show that the containment pressure remains below the ASME Level "C" Pressure limit for the "best estimate" severe accident scenario for up to 48 hours.

It is concluded that by considering severe accident prevention and mitigation early in the design process, System 80+ represents a robust plant design that has both low core damage frequencies and low conditional containment failure probabilities.