Docket Nos.: 50-443 and 50-444

SEP 19 1985

DISTRIBUTION
Docket File 50-443/444
NRC PDR
LPDR
NSIC
PRC System
LB#3 Reading
JLee
VNerses
HLThompson/FMiraglia
TMNovak
GWKnighton
RPerlis, ELD
EJordan
JPartlow
BGrimes
WLyons
JRosenthal
BSheron

Mr. Robert J. Harrison
President & Chief Executive Officer
Public Service Company of New Hampshire
Post Office Box 330
Manchester, New Hampshire 03105

Dear Mr. Harrison:

Subject: Seabrook Probabilistic Safety Assessment (PSA) Review

In my letter of April 4,1985 to you, I provided you a status at that time of our PSA review effort and our decision to terminate this review. As an enclosure to that letter, I provided you a draft review report from one of our contractors, Lawrence Livermore Laboratories (LLNL). The LLNL draft report constituted the Phase I review (review of the sequences leading up to core melt) of the Seabrook PSA.

As noted in my letter, because your support of our review had to be withdrawn, it impeded our efforts to complete the review. In your letter of July 12, 1985, you notified us of your recent contract with your PSA consultants, and you were prepared again to provide our staff any necessary support of their PSA review.

Subsequently, we completed the Phase II review (review of the severe accident response and consequences portion of the PSA), and for your information, enclosed is a copy of the draft report from our Phase II contractor, Brookhaven National Laboratory.

We intend to meet with your staff to discuss our Phase II review and this draft report.

Sincerely,

/s/

George W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing

Enclosure: As stated

cc: See next page

DL:LB#3
VNerses/yt
9/1/85

DL:LB#3
GWKnighton
9/19/85

Mr. Robert J. Harrison
Public Service Company of New Hampshire

Seabrook Nuclear Power Station

cc:
Thomas Dignan, Esq.
John A. Ritscher, Esq.
Ropes and Gray
225 Franklin Street
Boston, Massachusetts 02110

Mr. Bruce B. Beckley, Project Manager
Public Service Company of New Hampshire
Post Office Box 330
Manchester, New Hampshire 03105

Dr. Mauray Tye, President
Sun Valley Association
209 Summer Street
Haverhill, Massachusetts 01839

Robert A. Backus, Esq.
O'Neil, Backus and Spielman
116 Lowell Street
Manchester, New Hampshire 03105

Ms. Beverly A. Hollingworth
7 A Street
Hampton Beach, New Hampshire 03842

William S. Jordan, III
Diane Curran
Harmon, Weiss & Jordan
20001 S Street, NW
Suite 430
Washington, DC 20009

Jo Ann Shotwell, Esq.
Office of the Assistant Attorney General
Environmental Protection Division
One Ashburton Place
Boston, Massachusetts 02108

D. Pierre G. Cameron, Jr., Esq.
General Counsel
Public Service Company of New Hampshire
Post Office Box 330
Manchester, New Hampshire 03105

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, Pennsylvania 19406

E. Tupper Kinder, Esq.
G. Dana Bisbee, Esq.
Assistant Attorney General
Office of Attorney General
208 State Hosue Annex
Concord, New Hampshire 03301

Resident Inspector
Seabrook Nuclear Power Station
c/o US Nuclear Regulatory Commission
Post Office Box 700
Seabrook, New Hampshire 03874

Mr. John DeVincentis, Director
Engineering and Licensing
Yankee Atomic Electric Company
1671 Worchester Road
Framingham, Massachusetts 01701

Mr. A. M. Ebner, Project Manager
United Engineers & Constructors
30 South 17th Street
Post Office Box 8223
Philadelphia, Pennsylvania 19101

Mr. Philip Ahrens, Esq.
Assistant Attorney General
State House, Station #6
Augusta, Maine 04333

Mr. Warren Hall
Public Service Company of
New Hampshire
Post Office Box 330
Seabrook, New Hampshire 03874

Seacoast Anti-Pollution League
Ms. Jane Doughty
5 Market Street
Portsmouth, New Hampshire 03801

Mr. Diana P. Randall
70 Collins Street
Seabrook, New Hampshire 03874

cc:
Mr. Calvin A. Canney, City Manager
City Hall
126 Daniel Street
Portsmouth, New Hampshire 03801

Ms. Letty Hett
Town of Brentwood
RFD Dalton Road
Brentwood, New Hampshire 03833

Ms. Roberta C. Pevear
Town of Hampton Falls, New Hampshire
Drinkwater Road
Hampton Falls, New Hampshire 03844

Ms. Sandra Gavutis
Town of Kensington, New Hampshire
RDF 1
East Kingston, New Hampshire 03827

Chairman, Board of Selectmen
Town Hall
South Hampton, New Hampshire 03827

Mr. Angie Machiros, Chairman
Board of Selectmen
for the Town of Newbury
Newbury, Massachusetts 01950

Ms. Cashman, Chairman
Board of Selectmen
Town of Amesbury
Town Hall
Amesbury, Massachusetts 01913

Honorable Richard E. Sullivan
Mayor, City of Newburyport
Office of the Mayor
City Hall
Newburyport, Massachusetts 01950

Mr. Donald E. Chick, Town Manager
Town of Exeter
10 Front Street
Exeter, New Hampshire 03823

Mr. William B. Derrickson
Senior Vice President
Public Service Company of
New Hampshire
Post Office Box 700, Route 1
Seabrook, New Hampshire 03874

Mr. Alfred V. Sargent,
Chairman
Board of Selectmen
Town of Salisbury, MA 01950

Senator Gordon J. Humphrey
U. S. Senate
Washington, DC 20510
(Attn: Tom Burack)

Senator Gordan J. Humphrey
1 Pillsbury Street
Concord, New Hampshire 03301
(Attn: Herb Boynton)

Mr. Owen B. Durgin, Chairman
Durham Board of Selectmen
Town of Durham
Durham, New Hampshire 03824

Charles Cross, Esq.
Shaines, Mardrigan and
McEaschern
25 Maplewood Avenue
Post Office Box 366
Portsmouth, NH 03801

Mr. Guy Chichester, Chaiman
Rye Nuclear Intervention
Committee
c/o Rye Town Hall
10 Central Road
Rye, New Hampshire 03870

Jane Spector
Federal Energy Regulatory
Commission
825 North Capital Street, NE
Room 8105
Washington, D. C. 20426

Mr. R. Sweeney
New Hampshire Yankee Division
Public Service of New Hampshire
Company
7910 Woodmont Avenue
Bethesda, Maryland 20814

DRAFT

BNL/NUREG-
NUREG/CR-

A REVIEW OF THE SEABROOK STATION PROBABILISTIC SAFETY ASSESSMENT: CONTAINMENT FAILURE MODES AND RADIOLOGICAL SOURCE TERMS

M. Khatib-Rahbar, A. K. Agrawal, H. Ludewig
and W. T. Pratt

September 1986

Department of Nuclear Energy
Brookhaven National Laboratory
Upton, New York 11973

ABSTRACT

A technical review and evaluation of the Seabrook Station Probabilistic Safety Assessment has been performed. It is determined that (1) containment response to severe core melt accidents is judged to be an important factor in mitigating the consequences, (2) there is negligible probability of prompt containment failure or failure to isolate, (3) failure during the first few hours after core melt is also unlikely, (4) the point-estimate radiological releases are comparable in magnitude to those used in WASH-1400, and (5) the energy of release is somewhat higher than for the previously reviewed studies.

ACKNOWLEDGMENT

CONTENTS

LIST OF TABLES

LIST OF FIGURES

1. INTRODUCTION

1.1 Background

Probabilistic Risk Asessment (PRA) studies have been undertaken by a number of utilities (as exemplified by Refs. 1-4) and submitted to the Nuclear Regulatory Commission (NRC) for review. Brookhaven National Laboratory (BNL) under contract to the NRC, has been involved in reviewing core melt phenomenology, containment response and site consequence aspects of the PRAs.

This report presents a review and evaluation of the containment failure modes and the radiological release characteristics of the Seabrook Station Probabilistic Safety Assessment (SSPSA), which was completed by Pickard, Lowe and Garrick, Inc. (PLG) for the Public Service Company of New Hampshire and Yankee Atomic Electric Company in December 1983.[5]

1.2 Objective and Scope

The objective of this report is to provide a perspective on severe accident propagation, containment response and failure modes together with radiological source term characteristics for the Seabrook Station. Accident initiation and propagation into core damage and meltdown sequences were reviewed by the Lawrence Livermore National Laboratory (LLNL) as reported in an incomplete report [6] prepared for the Reliability and Risk Assessment Branch of NRC.

? In the present report, principal containment design features are discussed and compared with those of Zion, Indian Point and Millstone-3 designs. Those portions of the SPSS related to severe accident phenomena, containment response and radiological source terms are described and evaluated. Numerical adjustments to the SPSS estimates are documented and justified.

1.3 Organization of the Report

At brief review of the Seabrook plant features important to severe accident analysis is presented in Chapter 2 along with comparisons to Zion, Indian Point and Millstone-3 plant designs. Chapter 3 contains the assessment of containment performance. Specifically, definition of containment response classes and plant damage states, analytical mehods, containment failure model, containment event tree and accident phenomenology and the containment matrix are reviewed. Chapter 4 addresses the accident source terms together with justifications for adjustment where necessary. The results of this review are summarized in Chapter 5.

methods

2. PLANT DESIGN AND FEATURES IMPORTANT TO SEVERE ACCIDENT ANALYSIS

In this section, those plant design features that may be important to an assessment of degraded and core melt scenarios and containment analysis are reviewed. These important features are then compared with the Zion, Indian Point and Millstone-3 facilities to identify commonalities for benchmark comparisons.

2.1 Assessment of Plant Design

The Seabrook Station is comprised of two nuclear units each having an identical Nuclear Steam Supply System (NSSS) and turbine generator. The units are arranged using a "sling-along" concept which results in Unit 2 being arranged similar to Unit 1 but moved some 500 feet west. Each unit is a 1150 MWe (3650 MWt), 4-loop, Westinghouse PWR plant. The turbine-generators are supplied by the General Electric Company and the balance of the plant is designed by United Engineers and Constructors.

Each containment completely encloses an NSSS, and is a seismic Category I reinforced concrete structure in the form of a right vertical cylinder with a hemispherical top dome and flat foundation mat built on bedrock. The inside face is lined with a welded carbon steel plate, providing a high degree of leak tightness. A protective 4 ft. thick concrete mat, which forms the floor of the containment, protects the liner over the foundation mat. The containment structure provides biological shielding for normal and accident conditions. The approximate dimensions of the containment are:

Inside diameter	140 ft.
Inside height	219 ft.
Vertical wall thickness	4 ft. 6 in. and 4 ft. 7 1/2 in.
Dome thickness	3 ft. 6 1/8 in.
Foundation mat thickness	10 ft.

Containment penetrations are provided in the lower portion of the structure, and consist of a personnel lock and an equipment hatch/personnel lock, a fuel transfer tube, electrical, instrumentation, and ventilation penetrations.

Each containment enclosure (also known as secondary containment) surrounds a containment and is designed in a similar configuration as a vertical right cylindrical seismic Category I, reinforced concrete structure with dome and ring base. The approximate dimensions of the structure are: inside diameter, 158 ft; vertical wall thickness, varies from 1 ft, 3 in. to 3 ft; and dome thickness, 1 ft, 3 in.

? The containment enclosure is designed to collects any leakage from the containment structure other than leakage associated with piping, electrical and access passage penetration and discharge to the filtration system of containment. To accomplish this, the space between the containment enclosure and the containment structure, as well as the penetration and safeguards pump areas, are maintained at a negative pressure following a design basis accident by fans which take suction from the containment enclosure and exhaust to atmosphere through charcoal filters. To ensure air tightness for the negative pressure, leakage through all joints and penetrations has been minimized.

A containment spray system is utilized for post accident containment heat removal. The containment spray system is designed to spray water containing boron and sodium hydroxide into the containment atmosphere after a major accident to cool it and remove iodine. The pumps initially take suction from the refueling water storage tank and deliver water to the containment atmosphere through the spray headers located in the containment dome. After a prescribed amount of water is removed from the tank, the pump suction is transferred to the containment sump, and cooling is continued by recirculating sump water through the spray heat exchangers and back through the spray headers.

The spray is actuated by a containment spray actuation signal which is generated at a designated containment pressure. The system is completely redundant and is designed to withstand any single failure.

The containment isolation system establishes and/or maintains isolation of the containment from the outside environment in order to prevent the release of fission products. Automatic trip isolation signals actuate the appropriate valves to a closed position whenever automatic safety injection occurs or high containment pressure is experienced. Low capacity thermal electric hydrogen recombiners are provided.

The emergency core cooling system (ECCS) injects borated water into the reactor coolant system following accidents to limit core damage, metal-water reactions and fission product release, and to assure adequate shutdown margin. The ECCS also provides continuous long-term post-accident cooling of the core by recirculating borated water between the containment sump and the reactor core.

The ECCS consists of two centrifugal charging pumps, two high pressure safety injection pumps, two residual heat removal pumps and heat exchangers, and four safety injection accumulators. The system is completely redundant, and will assure flow to the core in the event of any single failure.

The control building contains the building services necessary for continuous occupancy of the control room complex by operating personnel during all operating conditions. These building services include: HVAC services, air purification and iodine removal, fresh air intakes, fire protection, emergency breathing apparatus, communications and meteorological equipment, lighting, and housekeeping facilities.

Engineered Safety Feature (ESF) filter systems required to perform a safety-related function following a design basis accident are discussed below:

a. The containment enclosure exhaust filter system for each unit collects, filters and discharges any containment leakage. The system is not normally in operation, but in the event of an accident, it is placed in operation and keeps the containment enclosure and the building volumes associated with the penetration tunnel and the ESF equipment cubicles under negative pressure to ensure all leakage from the containment structure is collected and filtered before discharge to the plant vent.

b. One of two redundant charcoal filter exhaust trains is placed in operation in the fuel storage building whenever irradiated fuel not in a cask is being handled. These filter units together with dampers and controls will maintain the building at a negative pressure.

The emergency feedwater system supplies demineralized water from the condensate water storage tank to the four steam generators upon loss of normal feedwater flow to remove heat from the reactor coolant system. Operation of the system will continue until the reactor coolant system pressure is reduced to a value at which the residual heat removal system can be operated. The combination of one turbine-driven and one motor-driven emergency feedwater pump provides a diversity of power sources to assure delivery of condensate under emergency conditions.

The two units of the facility are interconnected to off-site power via three 345 kilovolt lines of the transmission system for the New England states. The normal preferred source of power for each unit is its own main turbine generator. The redundant safety feature buses of each unit are powered by two unit auxiliary transformers. A highly reliable generator breaker is provided to isolate the generator from the unit auxiliary transformers in the event of a generator trip, thereby obviating the need for a bus transfer upon loss of turbine generator power. In the event that the unit auxiliary transformers are not available, the redundant safety feature buses of each unit are powered by two reserve auxiliary transformers. Upon loss of off-site power, each unit is supplied with adequate power by either of two fast-starting, diesel-engine generators. Either diesel-engine generator and its associated safety feature bus is capable of providing adequate power for a safe shutdown under accident conditions with a concurrent loss of off-site power. A constant supply of power to vital instruments and controls of each unit is assured through the redundant 125 volt direct current buses and their associated battery banks, battery chargers and inverters.

2.2 Comparison with Other Plants

Table 2.1 sets forth the design characteristics of the Zion, Indian Point-2, and Millstone-3 facilities as they compare to the Seabrook station.

It is seen that the containment characteristics are quite similar with the exception of containment operating pressure for Millstone-3 (subatmospheric design), and the use of fan coolers in Zion and Indian Point for post-accident containment cooling, the lower reactor cavity configuration, and chemical composition of the concrete mix. The primary system designs are nearly identical between the four units.

The Seabrook containment building basemat and the internal concrete structures are composed of basaltic-based concrete. As concrete is heated, water vapor and other gases are released. The initial gas consists largely of carbon dioxide, the quantity of which depends on the amount of calcium carbonate in the concrete mix. Limestone concrete can contain up to 80% calcium carbonate by weight, which could yield up to 53 lb of carbon dioxide per cubic foot of concrete. However, basaltic-based concrete contains very little calcium carbonate (3.43 w% for Seabrook) and would not release a substantial amount of carbon dioxide.[5] Thus, pressurization of the containment as a

Table 2.1 Comparison of Selected Design Characteristics

Design Parameters		Zion Unit 1[1]	Indian Point Unit 2[3]	Millstone Unit 3[4,7]	Seabrook Unit 1,2[5]
Reactor Power	[MW(t)]	3,250	3,030	3,411	3,650
Containment Building:					
Free Volume	(ft^3)	2.73×10^6	2.61×10^6	2.3×10^6	2.7×10^6
Design Pressure	(psia)	62	62	59.7	67.7
Initial Pressure	(psia)	15	14.7	12.7/9.1	15.2
Initial Temperature	(°F)	120	120	120/80	120
Primary System:					
Water Volume	(ft^3)	12,710	11,347	11,671	13,140
Steam Volume	(ft^3)	720	720	?	2,012
Mass of UO_2 in Core	(lb)	216,600	216,600	222,739	222,739
Mass of Steel in Core	(lb)	21,000	20,407	?	19,000
Mass of Zr in Core	(lb)	44,500	44,600	45,296	45,234
Mass of Bottom Head	(lb)	87,000	78,130	87,000	87,000
Bottom Head Diameter	(ft)	14.4	14.7	14.4	14.4
Bottom Head Thickness	(ft)	0.45	0.44	0.45	0.45
Containment Building Coolers:					
Sprays		yes	yes	yes	yes
Fans (with safety function)		yes	yes	no	no
Accumulator Tanks:					
Total Mass of Water	(lb)	200,000	173,000	348,000	213,000
Initial Pressure	(psia)	665	665	600	615
Temperature	(°F)	150	150	80	?
Refueling Water Storage Tank:					
Total Mass of Water	(lb)	2.89×10^6	2.89×10^6	10^7	2.89×10^6*
Temperature	(°F)	100	120	50	86
Reactor Cavity:					
Configuration		Wet	Wet	Dry	Dry/Wet
Concrete Material		Limestone	Basaltic	Basaltic	Basaltic

*Minimum (Maximum Capacity = 3.9×10^6 lb) result of corium/concrete interactions would be expected to take a very long time.

3. ASSESSMENT OF CONTAINMENT PERFORMANCE

In this chapter, the review of containment responce to severe accidents is described. Analytical techniques used to analyze core meltdown phenomena and containment response are reviewed, containment failure model is assessed and plant damage states and containment failure modes are evaluated. Parallels between this study and other PRAs are set forth. Finally, the relevance and validity of the conclusions is addressed.

3.1 Containment Analysis Methods

A brief description of the computer codes used to perform the transient degraded, core meltdown and containment response analyses is provided in this section.

The MARCH[8] computer code is used to model the core and primary system transient behavior and to obtain mass and energy releases from the primary system until reactor vessel failure. These mass and energy releases are then used as input to the other computer codes for analysis of containment response.

For sequences in which the reactor coolant system remains at an elevated pressure until the vessel failure ("time-phased dispersal"), the MODMESH[5] computer code is used. This code calculates the steam and hydrogen blowdown from the reactor vessel using an isothermal ideal gas model. The water level boil-off from the reactor cavity floor is modeled using a saturated critical heat flux correlation. Additionally, the accumulator discharge following depressurization caused by the vessel failure is also considered.

A modified version of the CORCON[9] code is used to replace the INTER[8] subroutine of the MARCH code. CORCON models the core-concrete interaction after the occurrence of dryout in the reactor cavity. The mass and energy releases from the core-concrete interaction are transferred to the MODMESH code for proper sequencing and integration into the overall mass and energy input to COCOCLASS9[5] code.

COCOCLASS9, a modified version of the Westinghouse COCO computer code utilizes the mass and energy inputs to the containment as computed by MARCH to model the containment building pressurization and hydrogen combustion phenomena. This code replaces the MACE subroutine of the MARCH code. The code also models heat transfer to the containment structures and capability for containment heat removal through containment sprays and sump recirculation.

Fission product transport and consequence calculations are performed using the CORRAL-II and the PLG proprietary CRACIT[5] computer codes, respectively.

The analytical methods used to carry out the core and containment thermal hydraulics, and fission product transport calculations are identical to those used for MPSS-3.[7]

3.2 Containment Failure

3.2.1 Background

In order to assess the risk of the Seabrook-1 plant, radiological source terms have to be calculated. Many steps are involved in such calculations. These are schematically shown in Fig. 3.1. The mode and time of containment failure directly impact on the radioactivity release categories. These, when coupled with the status of reactor cavity and the spray system, determine the source terms. This section deals with the mode and time of containment failure.

3.2.2 Design Description

The primary containment of the Seabrook plant is a seismic Category I reinforced concrete dry structure. It consists of an upright cylinder topped with a hemispherical dome. The inside diameter of the cylinder is 140 feet and the inside height from the top of the basemat to the apex of the dome is approximately 219 feet. The cylindrical wall is 4'6" thick above elevation 5' and 4'7-1/2 " thick below that evaluation. The dome is 3'6-1/8" thick and 69'11-7/8" in radius. The cylinder is thickened to provide room for additional reinforcing steel around the openings for the equipment hatch and the personnel airlock. The net free volume of the containment is approximately 2.7 x 10^6 ft^3.

? The inside of the containment is welded with a steel liner. The liner plate is the cylinder is 3/8" thick in all areas except penetration and the junction of the basemat and cylinder where it is 3/4" thick. This liner serves as a leak-tight membrane. Welds that are embedded in the concrete and not readily accessible are covered by a leak chase system which permits leak testing of these welds throughout the life of the plant. The dome liner is 1/2" thick and flush with the outside face of the cylindrical liner. The operating and the design parameters of containment are noted in Table 3.1.

The containment building is surrounded by an enclosure. The containment enclosure is a reinforced concrete cylindrical structure with a hemispherical dome. The inside diameter of the cylinder is 158 feet. The vertical wall varies in thickness from 36 inches to 15 inches; the dome is 15 inches thick. The inside of the dome is 5'6" above the top of the containment dome. Located at the outside of the enclosure building is the plant vent stack, consisting of a light steel frame with steel plates varying in cross-section. The stack carries exhaust air from various buildings.

The containment enclosure is designed to control any leakage from the containment structure. To accomplish this, the space between the containment and the enclosure building (approximately 4'6" wide) is maintained at a slight negative pressure (-0.25" water gauge) during accident conditions by fans which take suction from the containment enclosure and exhaust to atmosphere through charcoal filters.

There are a number of containment penetrations which are steel components that resist pressure. These penetrations are not backed by structural concrete and include the following:

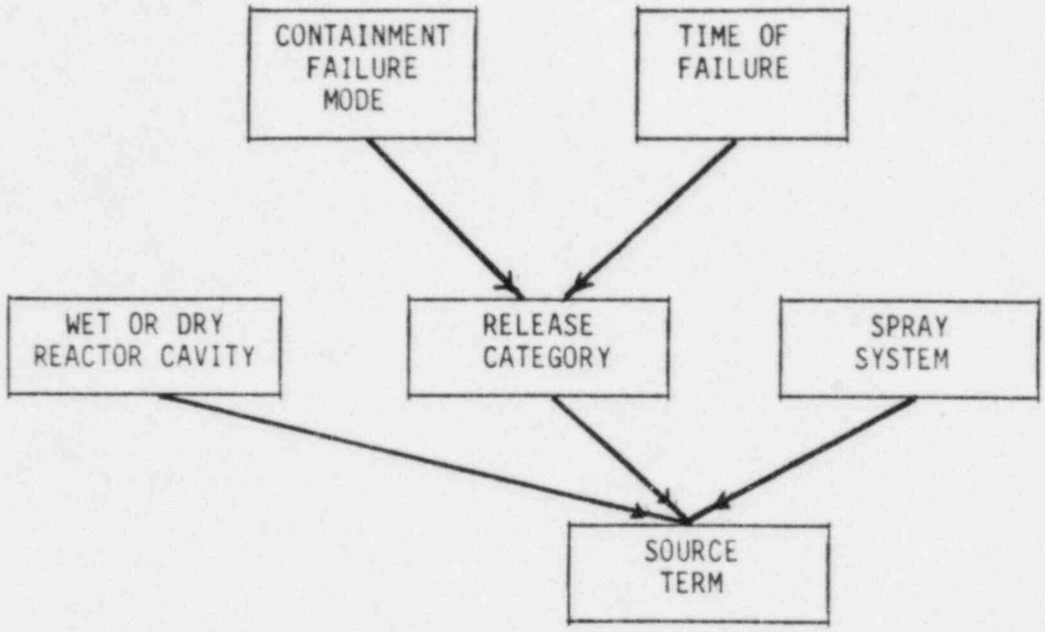


Figure 3.1 A schematic representation of source term calculation.

Table 3.1 Containment Operating and Design Parameters

Parameter	Value
Normal Operation	
Pressure , psig	0.5
Inside Temperature , F	120
Outside Temperature , F	90
Relative Humidity , %	45
Service Water Temperature , F	80
Refueling Water Temperature , F	86
Spray Water Temperature , F	88
Containment Enclosure Pressure , inches w.g.	-0.25
Design Conditions	
Pressure , psig	52.0
Temperature , F	296
Free Volume , ft^3	2.7×10^6
Leak Rate , % mass/day	0.2
Containment Enclosure Pressure , psig	-3.5

1. Equipment hatch,
2. Personnel air lock,
3. Piping penetrations,
4. Electrical penetrations,
5. Fuel transfer tube assembly,
6. Instrumentation penetrations, and
7. Ventilation penetrations.

These components penetrate the containment and containment enclosure shells to provide access, anchor piping, or furnish some other operational requirement. All penetrations are anchored to sleeves (or to barrels) which are embedded in the concrete containment wall.

Equipment Hatch

The equipment hatch (Fig. 3.2) consists of the barrel, the spherical dished cover plate with flange, and the air lock mounting sleeve. The centerline of the hatch is located at elevation 37'1/2" and an azimuth of 150°. The hatch opening has an inside diameter of 27'5". A sleeve for a personnel air lock, the inside diameter of which is 9'10", is provided at centerline elevation 30'6". Thicknesses of the primary components are as follows:

Component	Thickness (inches)
Barrel	3 1/2
Spherical	1 3/8
Flange	5 3/8
Air lock mounting sleeve	1 1/2

The equipment hatch cover is fitted with two seals that enclose a space which can be pressurized to 52.0 psig. The flange of the cover plate is attached to the hatch barrel with 32 swing bolts, 1-3/8 inch in diameter. The barrel, which is also the sleeve for the equipment hatch, is embedded in the shell of the concrete containment. The equipment hatch cover can be lifted to clear the opening.

Inserted into the mounting sleeve through the equipment hatch cover is a personnel air lock consisting of two air lock doors, two air lock bulkheads, and the air lock barrel. Significant dimensions of the air lock are as follows:

Parameter	Dimension
Inside Diameter of Barrel	9'6"
Barrel Thickness	1/2"
Door Opening	6'8" x 3'6"
Door Thickness	3/4"
Bulkhead Thickness	1-1/8"

Each door is locked by a set of six latch pin assemblies, and is designed to withstand the design pressure from inside the containment. To resist the test pressure, each door is fitted with a set of cast clamps. The doors are hinged and both swing into the containment. Each door is fitted with two seals that

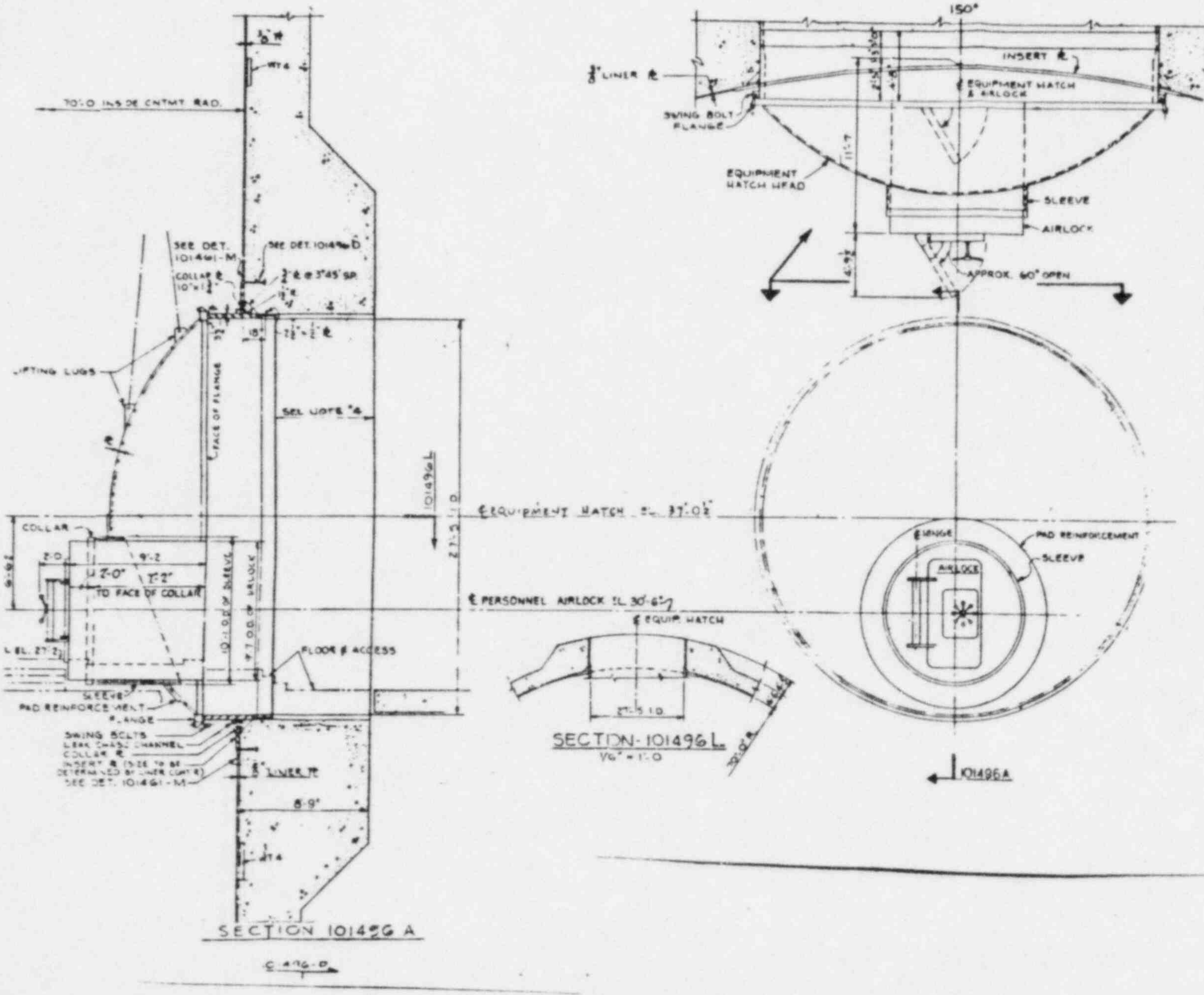


Figure 3.2 Equipment hatch with personnel airlock.

are located such that the area between doors can be pressurized to 52.0 psig. The doors are mechanistically interlocked so that only one door can be opened at a time. The capability exists for bypassing this interlock to equalize the pressure by use of special tools. The doors may be operated mechanically.

Personnel Air Lock

The personnel air lock (Fig. 3.3) consists of the air lock doors (2) and the lock barrel. The barrel, which is also the sleeve for the personnel air lock, is imbedded in the shell of the concrete containment. The centerline of the barrel is located at elevation 29'6" and an azimuth of 315°. Significant dimensions are as follows:

Parameter	Dimensions
Clear Opening	7'0"
O.D. of Flange on Door	7' 9 1/8"
Barrel Thickness	5/8"
Cover Thickness	5/8"

The air lock barrel has a door on each end, each of which is designed to withstand the design pressure from inside the containment. The doors are hinged and swing away from the air lock barrel. Each door is fitted with two seals that are located such that the area between doors can be pressurized to 52.0 psig. The locking device for the doors is a rotating, third ring, breach-type mechanism. These doors are also mechanically interlocked so that only one door can be opened at a time. The capability exists for bypassing this interlock and relieving the internal pressure by use of special tools. The doors may be operated mechanically.

Piping Penetrations

There are two types of piping penetrations: moderate energy and high energy. Moderate energy piping penetrations are used for process pipes in which both the pressure is less than or equal to 275 psi, and the temperature of the process fluid is less than or equal to 200°F. High energy piping penetrations are used for that piping in which the pressure or temperature exceeds these values.

High energy piping penetrations (Fig. 3.4) consist of a section of process pipe with an integrally-forged fluid head, a containment penetration sleeve and, where a pipe whip restraint is not provided, a penetration sliding support inside the containment. The sliding support provides shear restraint while permitting relative motion between the pipe and the support. The annular space between the process pipe and the sleeve is completely filled with fiberglass thermal insulation. The pipe and the fluid head, are classified as ASME III Safety Class 2 (NC), whereas the sleeve is classified as part of the concrete containment, ASME III (CC). The sliding support inside the containment is classified as an ASME Safety Class 2 component support (NF).

Moderate energy piping penetrations (Fig. 3.5) consist of one or more process pipes, the containment penetration sleeve, and a flat circular end-plate. The pipe is classified as ASME III Safety Class 2 (NC). The sleeve is classified as ASME III Div. 2 (CC). The end-plate is classified as Class MC.

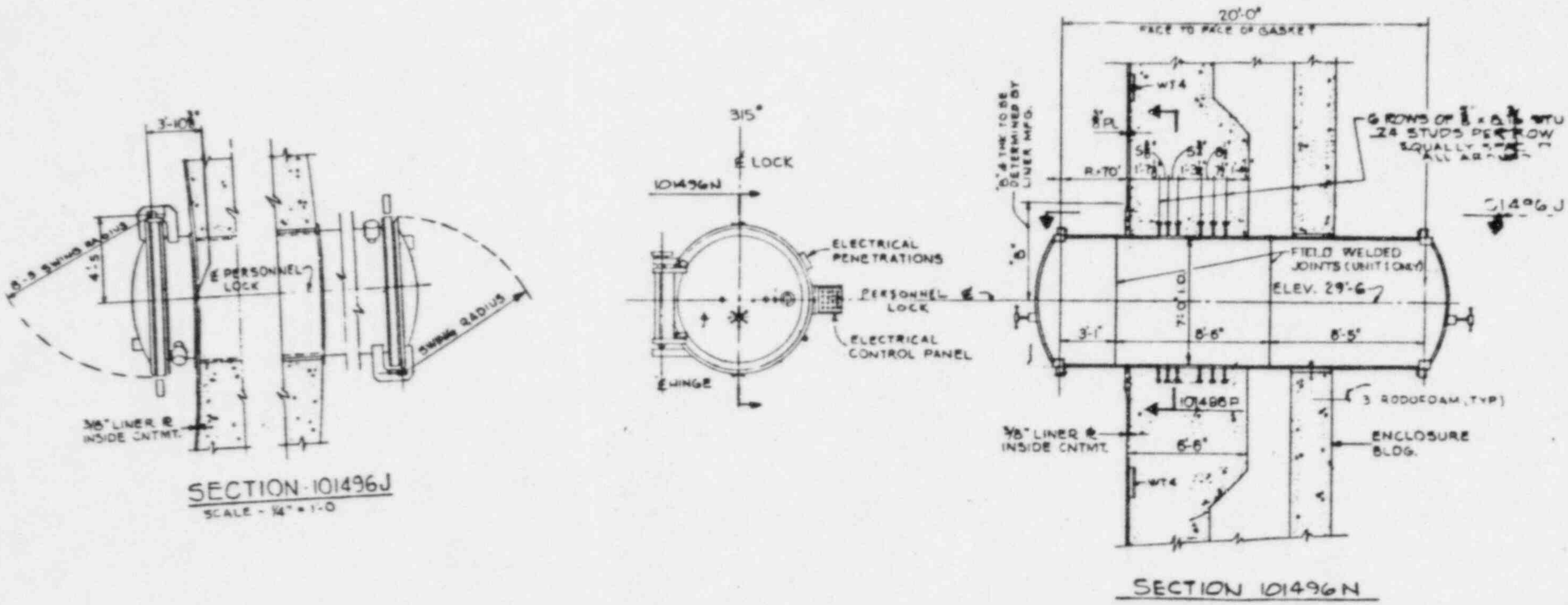


Figure 3.3 Personnel airlock.

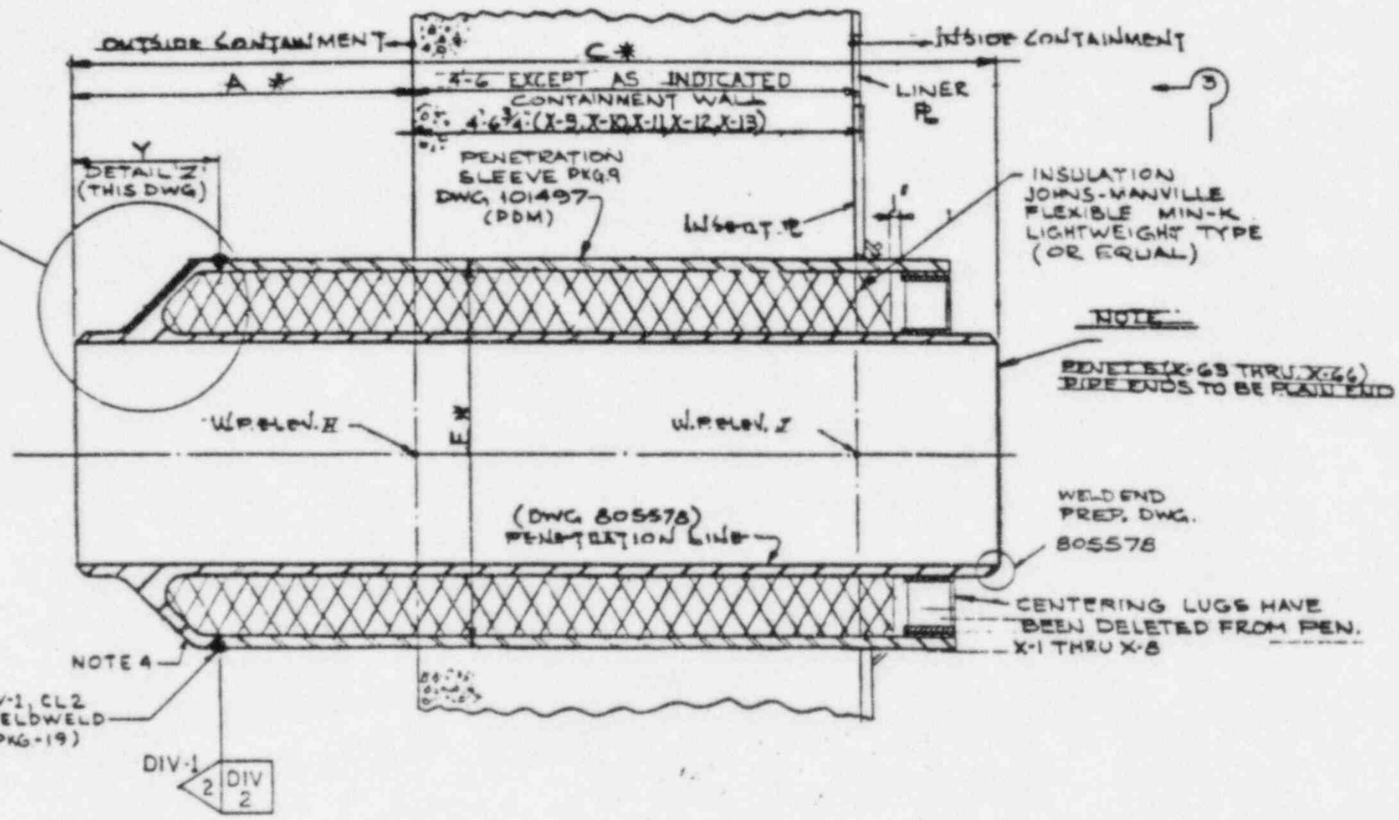
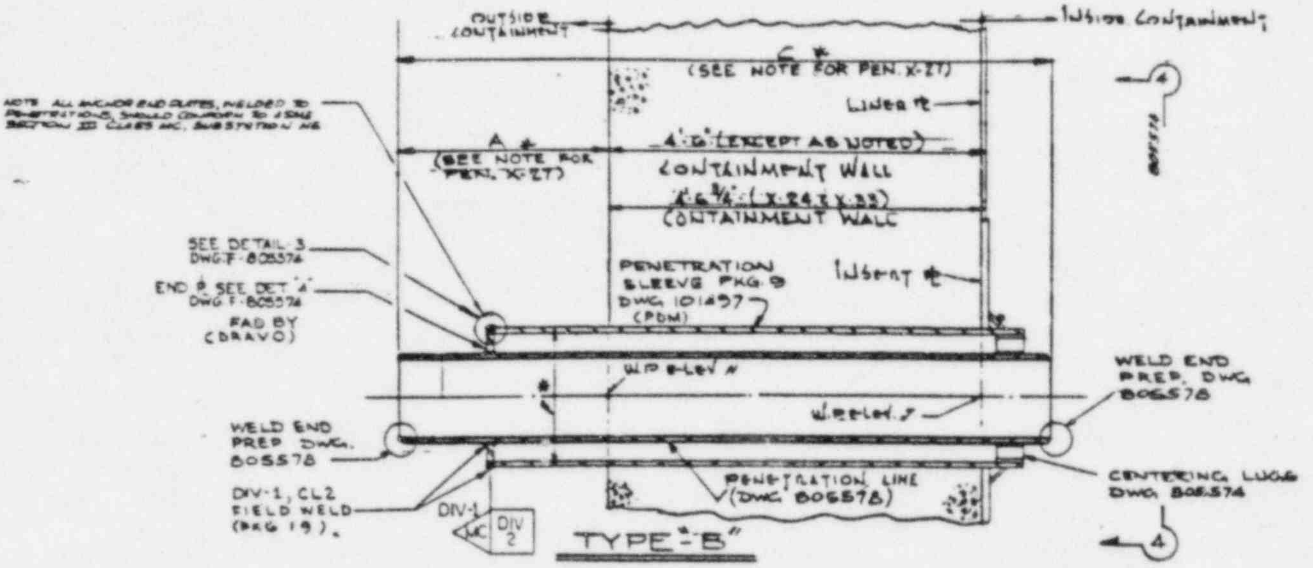


Figure 3.4 Typical high energy piping penetration.



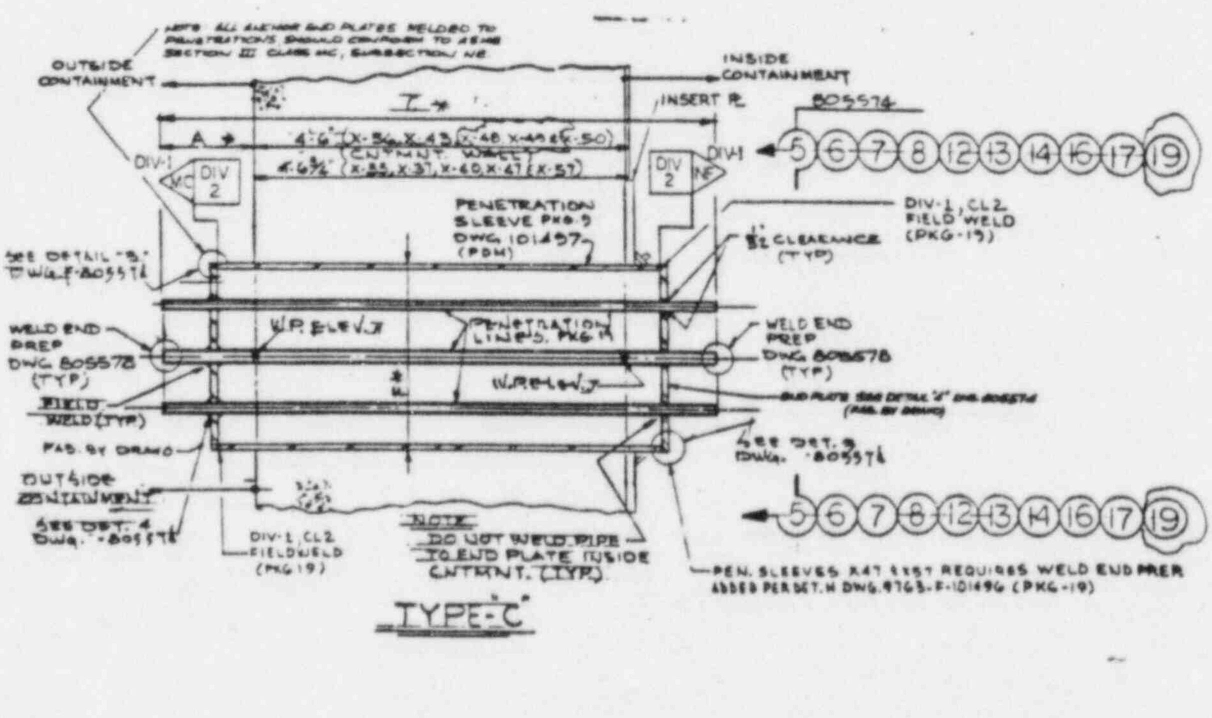


Figure 3.5 Typical moderate energy piping penetration.

Table 3.2 gives a list of the containment piping penetrations. Included in this table is the penetration size. All of these piping penetrations are in the lower portion of the structure.

Electrical and Instrumentation Penetrations

Electrical penetrations (Fig. 3.6) consist of a stainless steel header plate with an attached terminal box, electrical modules which are clamped to the header plate, and a carbon steel weld ring which is welded to the header plate and to the sleeve. The metallic pressure resisting parts, the sleeve, stainless steel header plate and carbon steel weld ring were designed as ASME III Safety Class MC components (NE); that portion of the sleeve which is backed by concrete was designed as part of the concrete containment, ASME III (CC).

Double silicone and Hypalon O-rings provide a seal with a cavity for leakage monitoring between the header plate and the modules. The header plate is provided with a hole on the outside of the containment to allow for pressurization of the penetration assembly for leakage monitoring.

There are a total of 64 electrical penetrations out of which 14 are spare and 8 are unused. All of these electrical penetrations are below the grade.

Instrumentation penetrations are of two types -- electrical and fluid. The electrical type is similar in construction to the other electrical penetrations. The fluid penetrations are similar in construction to the moderate energy piping penetrations.

Fuel Transfer Tube Assembly

The fuel transfer tube assembly consists of the fuel transfer tube, the penetration sleeve, the fixed saddle on the reactor side, and the sliding saddle in the fuel storage building. The fuel transfer tube centerline is at elevation (-)9'4-1/4" and it has approximately 20" inner diameter. The fuel transfer tube wall penetration sleeve, which is embedded in the concrete, has an inside diameter of about 25".

Ventilation Penetrations

There are two types of ventilation penetrations -- the containment air purge penetrations (HVAC-1 and HVAC-2) and the containment on-line penetrations (X-16 and X-18). The containment air purge penetrations (Fig. 3.7) each consist of a pipe sleeve (a rolled and welded pipe section, 36" outer diameter by 1/2" wall thickness) which is flanged at each end with 36" weld neck flanges and, attached to these flanges, the inner and outer isolation valves. Together with the pipe, these valves form a part of the containment pressure boundary. The valves are 36" diameter butterfly valves with fail-safe pneumatic operators. The weld between the pipe and the containment liner is equipped with a leak chase for pressure testing.

The containment on-line purge penetrations each consist of a pipe sleeve (a rolled and welded pipe section, 8" o.d. by 1/2" wall thickness). A short section of pipe with a nipple is welded to the sleeve on the outside of the containment, and a 3/4" valve and test connection is attached to it. The

Table 3.2 Containment Liner Penetrations

Penetration Numbers	Service	Penetration Size
X-1 to X-4	Main steam line	30"
X-5 to X-8	Main feedwater	18"
X-9, X-10	RHR pump suction	12"
X-11 to X-13	RHR to safety injection	8"
X-14 to X-15	Containment building spray	8"
X-16, X-18	Containment on-line purge	8"
X-17	Hydrogenated vent header	2"
X-20 to X-23	CCW supply and return	12"
X-24 to X-27	Safety injection	4"
X-28 to X-31	CVCS to pump seal injection	2"
X-32, X-34	Drain line	3",2"
X-33, X-37	CVCS	3"
X-35, X-36, X-40 X-52, X-71, X-72	RCS test/sample control	1" or smaller
X-38	Combustible gas control	10"
X-39	Spent fuel pool cooling	2"
X-43, X-47, X-50 X-57	Instrumentation lines	?
X-60, X-61	From containment recirculation sump	16"
X-62	Fuel transfer tube	20"
X-63 to X-66	Steam generator blowdown	3"
X-67	Service air	2"
HVAC-1,2	Containment purge supply/exhaust lines	36"
X-19, X-41, X-42 X-44 to X-46, X-48 X-49, X-51, X-58 X-59, X-68 to X-70	Spare	?

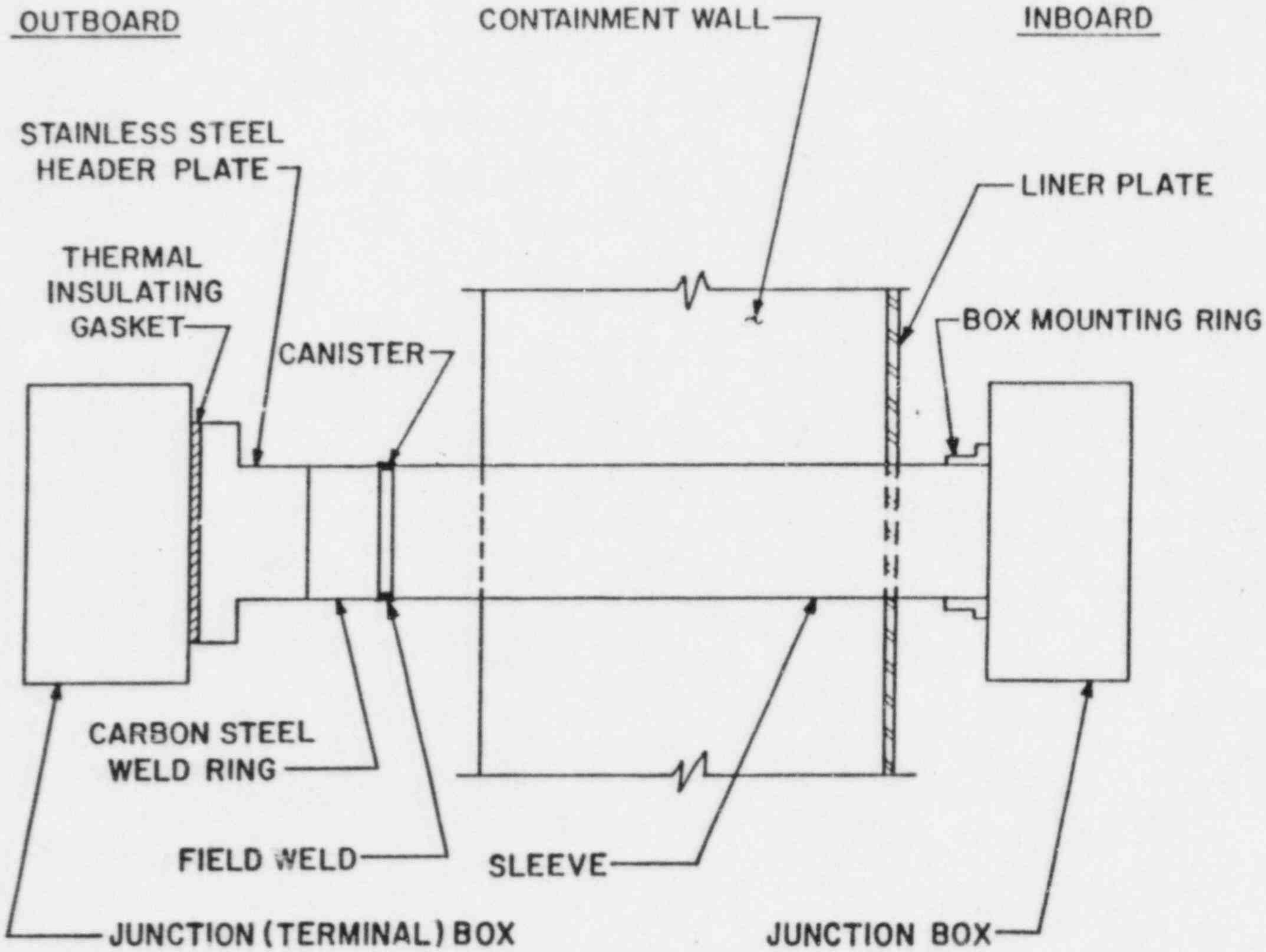


Figure 3.6 Tyical electrical penetration.

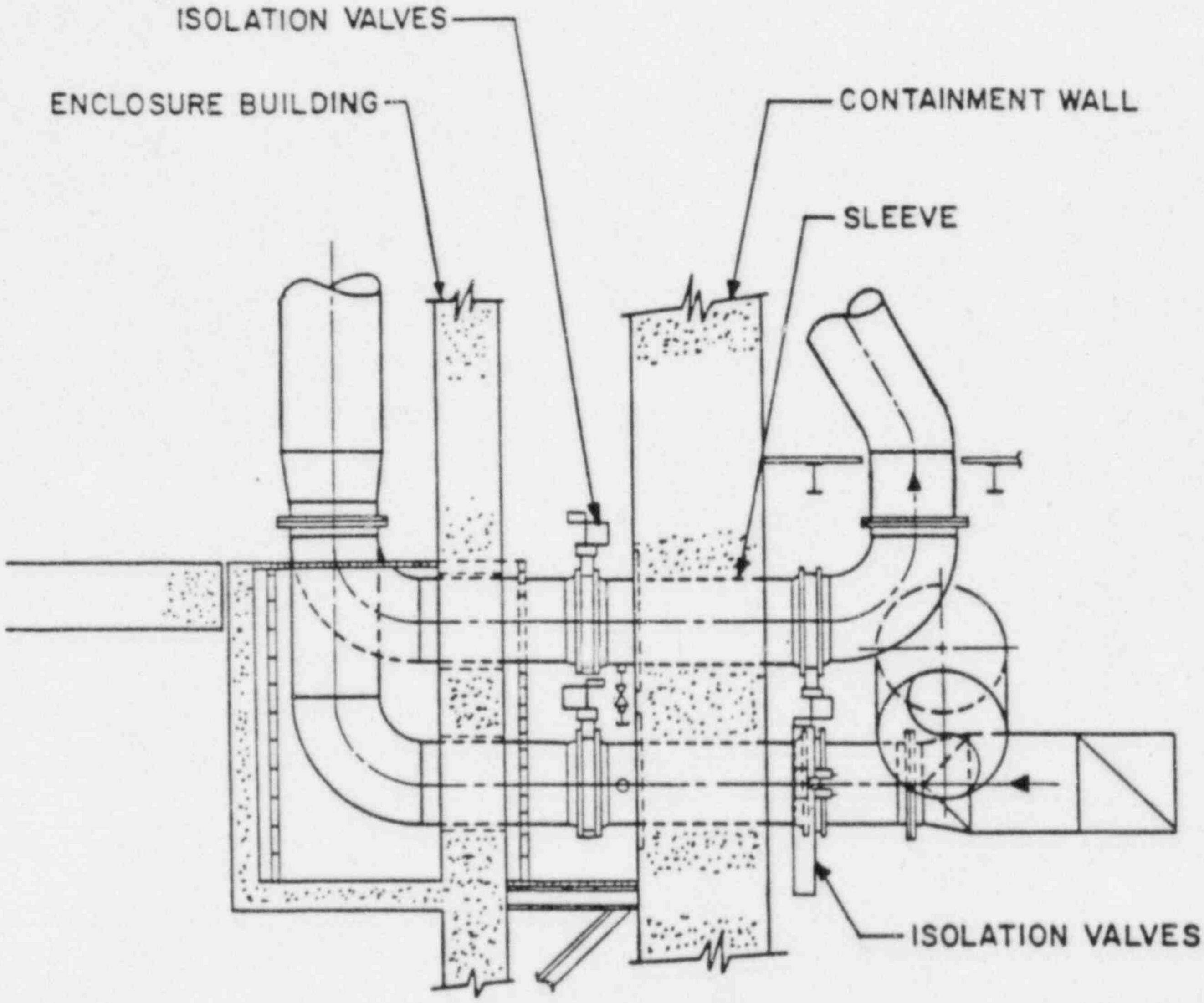


Figure 3.7 Typical ventilation penetration.

ends of this resulting assembly are welded to 8" weld neck flanges which are through-bolted to the inner and outer isolation valves. These valves are 8" diameter butterfly valves having fail-safe pneumatic operators. The weld between the pipe sleeve and the containment liner is equipped with a leak chase for pressure testing. These on-line purge penetrations are very similar to those for 36" lines shown earlier.

3.2.3 Leakage Rate Calculation

Under severe accident conditions the pressure inside the containment quickly builds up in the range of 75 to 200 psi. At these pressures, any leakage through the containment holes will essentially be choked. The leakage under choked flow condition is given as (Ref. 10):

$$W = \sqrt{k\left(\frac{2}{k+1}\right)^{\frac{k+1}{k-1}}}\, A\sqrt{\rho P} \qquad (1)$$

where

W = discharge rate (kg/s),
A = leak area (m^2),
P = absolute pressure (N/m^2),
ρ = mixture density (kg/m^3), and
k = ratio of specific heat at constant pressure to that at constant volume.

For air and water vapor mixture, $k \approx 1.3$. If the mixture density is expressed by perfect gas law

$$\rho = \frac{P}{RT} \qquad (2)$$

where

R = gas constant, and
T = the absolute temperature,

Then Eq.(1) becomes

$$W = \sqrt{k\left(\frac{2}{k+1}\right)^{\frac{k+1}{k-1}}}\, A\sqrt{\frac{P}{RT}} \qquad (3)$$

The mass of mixture can be written as

$$M = V\rho$$

or,

$$M = \frac{V}{RT}P \qquad (4)$$

where V is the free mixture volume in the containment. Equations (3) and (4) can be combined to get the leakage rate, in terms of mass fraction, as

$$\frac{W}{M} = \sqrt{k\left(\frac{2}{k+1}\right)^{\frac{k+1}{k-1}}}\ \frac{1}{V}\sqrt{RT}\ A \qquad (5)$$

Note that the leakage rate, when expressed in terms of mass fraction, depends only on the leakage area.

For Seabrook-1, using $V = 2.704 \times 10^6$ ft^3 and T = 296 F, Eq.(5) gives

$$\text{Leakage Rate} = 0.721\ A_{in} \quad \text{w/o per hour} \qquad (6)$$

where A_{in} is the leakage area in in^2. Alternately,

$$\text{Leakage Rate} = 17.3\ A_{in} \quad \text{w/o per day.} \qquad (7)$$

The essentially intact design basis containment leakage of 0.2 w/o per day, thus, corresponds to an equivalent leakage area of 0.012 in^2 (or, an equivalent hole of 1/8-in diameter). A leakage area of 4 to 10 in^2 would correspond to the leakage rate of 2.9 to 7.2 w/o per hour. In other words, it will take about 14 hours to leak the entire content to the environment through a 10-in^2 hole.

3.2.4 Containment Failure Model

3.2.4.1 Leak-Before-Failure

During accident sequences involving core damage, the containment structure will be exposed to pressures and temperatures beyond those used in the design basis accident (DBA). Response of the containment building to these severe conditions is evaluated in SSPSA by employing, for the first time, a leak-before-failure model. In this model allowance is made for continuous leakage from the containment to the surroundings. This mode of containment failure is termed local failure. The containment leakage can occur at many locations and discontinuities such as mechanical and electrical penetrations, personnel lock, equipment hatch, fuel transfer tube, welds, and in between the liner and concrete. Depending upon the size of leakage area and the accident sequence, local failures may gradually relieve pressure, thereby gross containment failure may be averted.

The leak-before-failure model is a realistic one. The extent of leakage and the health consequences must, however, be carefully studied. In order to explain this issue, it is observed that traditionally probabilistic risk assessment is made by using what is termed a threshold model. In the threshold model, the containment is considered intact until the internal loading equals or exceeds a pressure threshold (which may also be temperature dependent), at which it is deemed to have suffered a failure (gross). If the internal loading is below this threshold value, the containment is considered intact and hence the risk is quite low. In the leak-before-failure model, the release of activity, which is considerably small compared with that for the gross failure mode, must be considered in health consequences. However, such leakages can potentially prevent the internal pressure from approaching the threshold value and thus a catastrophic or gross failure may be avoided.

3.2.4.2 Classification of Failure

The SSPSA report has classified containment failures in three categories:

- Containment Failure Category A. Includes containment failures that develop a small leak that is substantially larger than the leak acceptable from an intact containment, but not large enough to arrest the pressure rise in the containment. Category A failures thus cause an early increase in the rate of leakage of radionuclides over the design basis leak rate but pressurization of the containment continues until either a category B or C containment failure occurs.

  The intact containment is defined as the one in which leakage is limited by the Technical Specification value. For Seabrook-1, this value is 0.2 w/o per day at the calculated peak accident pressure of approximately 47 psig. Note that the SSPSA study has used 0.1 volume percent per day for this leakage, although prior to the most recent amendment dated August 1984, the FSAR has cited both 0.1 volume percent and 0.1 w/o per day. The 10CFR50, Appendix J mandates the allowable leakage to be quoted as w/o per day. The higher value noted here is based on Amendment 53, August 1984.*

- Containment Failure Category B. Includes failure modes that develop a large enough leak area so that the pressure in the containment no longer increases. The time during which a substantial fraction of the radionuclide source term is released is longer than approximately 1 to 2 hours. Category B failures include self-regulating failure modes where the leak area is initially small but increases with pressure so that it becomes sufficient to terminate the pressure rise before a category C containment failure occurs.

  The definition of "substantial" fraction is unclear.

- Containment Failure Category C. Includes those containment failure modes that develop a large leak area. A large fraction of the total radionuclide source term is released over a period of less than 1 hour. All gross failure modes are included in category C.

Mathematically, these three failure categories can be expressed in terms of leakage areas as follows:

Should these be different? See above definitions

$$A_{DBA} < A_A < A_{NP} \qquad \underline{\text{Type A}}$$
$$A_{NP} < A_B < A_P \qquad \underline{\text{Type B}} \qquad (8)$$
$$A_C < A_P \qquad \underline{\text{Type C}}$$

where

A_{DBA} = leakage area corresponding to the technical specification limit for containment leakage,

*There appears to be substantial update/changes in the Engineered Safety Features flow diagram, including arrangements of motor operated valves and bypass lines, which may substantially change the frequency of events. BNL, however, is not reviewing this part of SSPSA.

A_{NP} = leakage area not large enough to arrest pressurization, and

A_P = leakage area sufficient to release 100 w/o in one hour.

The leakage area required to release a substantial fraction of the radionuclide source term in approximately an hour and can be computed using Eq. (6). Assuming one-hundred percent turnover as a substantial fraction in one hour, Eq. (6) gives the required leakage area to be equal to 138 in^2 or about 1 ft^2. Therefore, any containment leak area in excess of 1 ft^2 will be defined as a gross containment failure (Category C). This estimate of the lead area is a factor two too high from the value stated in SSPSA.

The leakage area required to arrest containment pressurization is in the range of 4 to 10 square inches, the lower value being more representative of wet sequences and the upper value is representative of dry sequences. A leak area of about 6 square inches will result in the release of about 100 w/o of activity in a day (see Eq. 7). The upper bound leak area for Type A failure is taken as 4 in^2. This corresponds to release of the radioactive source term (100% turnover) in about 36 hours. The Category B leak area is, thus, in the range of 4 in^2 to 1 ft^2. Figure 3.8 is a pictorial representation of these leakage categories.

3.2.5 Containment Pressure Capacity

3.2.5.1 Concrete Containment

The Seabrook PSA has examined failure modes for the containment structure itself, the steel liner, all penetrations, equipment and personnel lock hatches, and the secondary containment. The containment structure includes the cylindrical wall, the hemispherical dome, the base slab and the base slab and containment wall junction. The most critical membrane tension was found to occur in the cylinder in the hoop direction. The median pressure which causes yield of both the liner steel and the reinforcing bars was found to be approximately 157 psi, with a coefficient of variation of 0.084. The ultimate hoop load in cylinder is 216 psig. The containment wall is, thus, assumed to fail at this pressure. At pressures beyond this, very large irreversible deformations occur which will cause cracks in the reinforced concrete but the loss of integrity of the pressure boundary may not occur until the liner tears. The compiled radial deformations of the containment wall are shown in Figure 3.9. Note that the radial strain at the expected failure pressure of 216 psi is 4.7% ($\Delta r/r$).

The hemispherical dome was calculated to yield at a slightly higher pressure (163 psig). The failure pressure is predicted at 223 psig.

The median pressure for flexural failure of the base slab is 400 psig, with a logarithmic standard deviation of 0.25. However, the shear mode of failure is more restrictive. For this mode, the median failure pressure is estimated in SSPSA as 323 psig, with a logarithmic standard deviation of 0.23. Although the uncertainty for failure of the base slab is large, the probability of failure is small because the median capacities are high. Thus, failure of the base slab is not considered to be a critical failure mode and

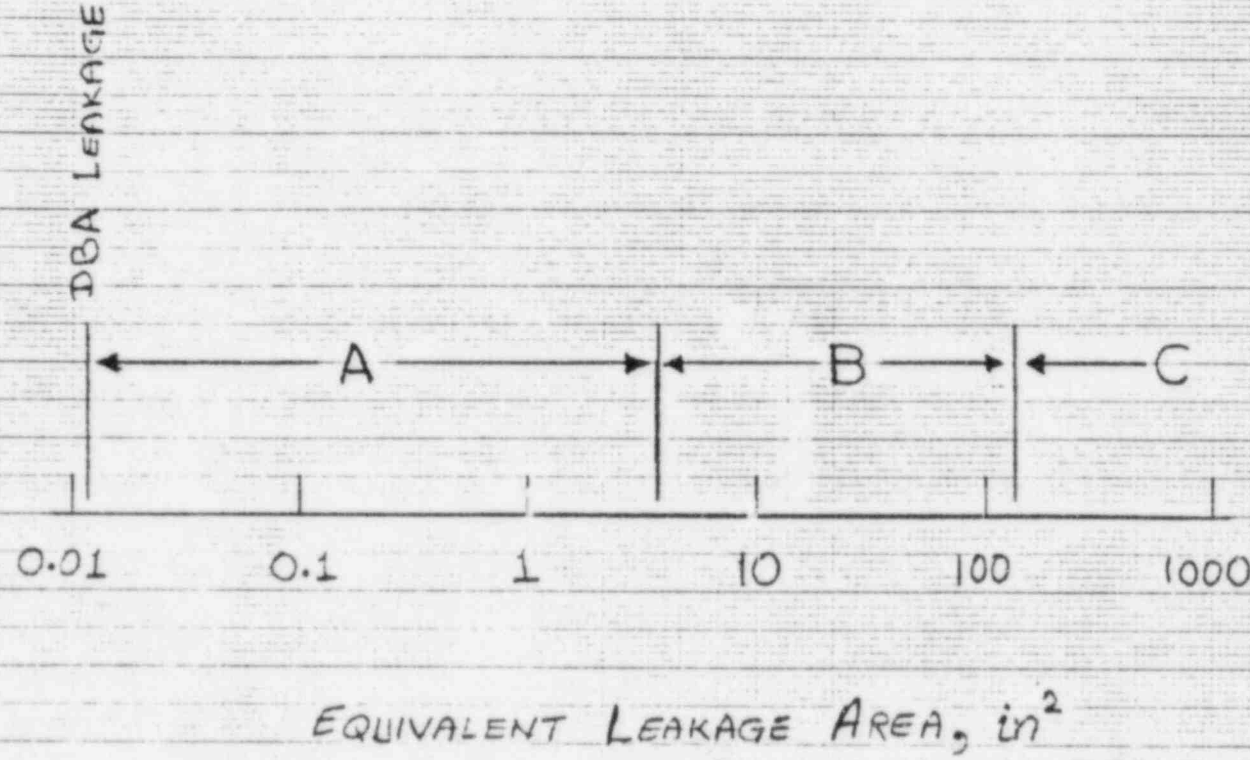


Figure 3.8 A pictorial representation of leakage categories.

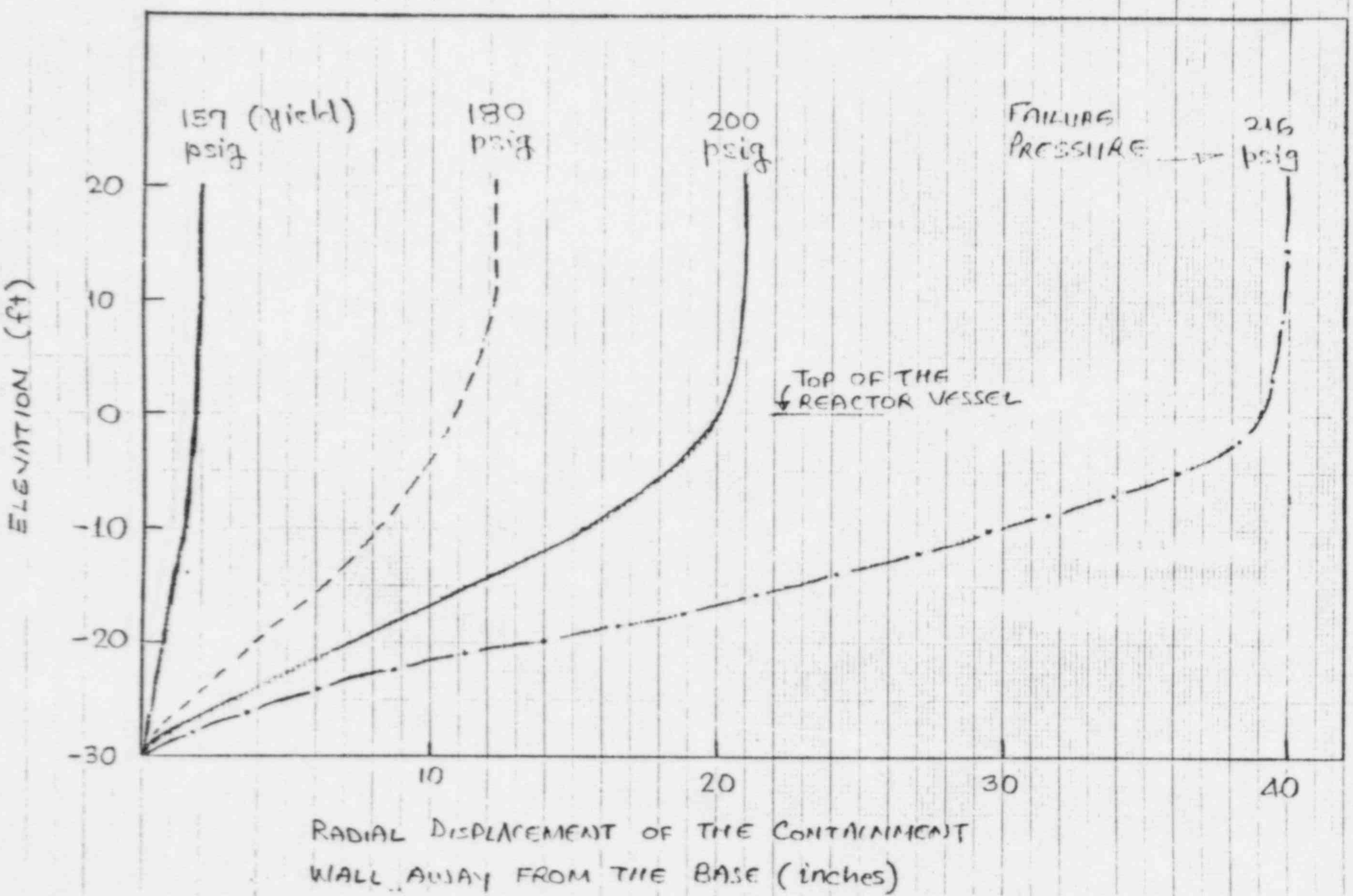


Figure 3.9 Estimated radial displacement of containment wall.

an estimation of leak areas was, therefore, not considered for this mode of failure.

Secondary stresses in the cylindrical portion of the containment occur at discontinuity such as at the base slab containment wall junction, at the springline, and where the amount of reinforcing changes. The flexural yield at the base of the cylinder occurs at 175 psi. At higher pressures, a plastic hinge forms with considerable cracking of the concrete. These cracks, however, are small enough so as not to threaten the integrity of the liner. The loss of integrity of the liner is not expected until a median pressure of 408 psi is reached. Thus, the failure of the base slab and containment wall junction is not limiting.

In summary, the containment wall is expected to undergo significant deformation ($\approx 4.7\%$ $\Delta r/r$) prior to its failure at 216 psig. At this pressure, Type C (i.e., gross) failure occurs.

3.2.5.2 Liner

The elongation capacity of the steel liner is computed by neglecting the friction forces between the liner and the concrete. The possibility that the liner stresses and strains could be different between two different pairs of tees was, however, considered. The SSPSA computed an elongation of 8.1 percent under uniaxial conditions, or an elongation of 4.7 percent under plane strain conditions can be achieved without fracture. This would ensure integrity of the liner until fracture of the reinforcing bars. Additionally, the leakage of the containment at penetrations is considered likely before hoop failure of the liner occurs.

3.2.5.3 Penetrations

At all major penetrations, the containment wall is thickened and additional reinforcement is provided to resist stress concentrations. None of the meridional or hoop reinforcing bars are terminated at penetrations. Instead, they are continued around the penetrations, thus ensuring that excess hoop and meridional capacity is available. Table 3.2 lists all piping penetrations.

As the containment pressure increases beyond its yield value (157 psi), large radial deformations begin to occur. This induces stresses in the pipes by relative displacements between the containment wall and the pipe whip restraints. Therefore, the most critical penetrations are the areas where the pipe is supported close to the penetration. Also, stronger and stiffer pipes develop higher forces at the penetrations for a given relative displacement. The SSPSA study selected the following penetrations for investigation as being among the lines most likely to fail:

Penetration X-23 (also X-20 to X-22 by similarity)	12" schedule 40 carbon steel
Penetration X-26 (also X-24, X-25, X-27)	4" schedule 160, stainless steel

Penetration X-71 (also X-72 and possibly others)	1" - multiple pipe penetration
Penetration X-8 (also X-5 to X-7)	18" main feedwater schedule 100, carbon steel
Fuel Transfer Tube	Convoluted Bellows

The probability of failure at these penetrations was computed by (a) establishing a pressure-displacement relation, (b) estimating the failure probability as a function of radial displacement and then (c) combining the two. The radial displacements for the containment wall were shown earlier (Fig. 3.9). The vertical displacement due to meridional strains is small (less than 3 inches) and hence its impact on the penetrations was ignored. Since most of these penetrations are in the lower part of the containment, the radial displacements experienced by them due to plastic deformation of containment would also be small.

The multiple penetration (X-71 and X-72) would not fail even for the most unfavorable forces which these pipes could sustain. For penetrations X-23 and X-26, the most likely location for failure is at the partial penetration fillet welds which join the pipe to the end plate. When failure of this weld occurs, the pipe remains in the hole provided in the end plate. The gap between the pipe and the end plate is likely to remain small unless the pipe wall buckles. Exact gap size is hard to compute. The SSPSA appears to use a uniform gap size of 0.04 in., and 0.10 in. as median and upper estimates, respectively. The corresponding leak areas for X-23 (as well as X-20 to X-22) and X-26 (as well as X-24, X-25, and X-27) penetrations are shown in Table 3.3. The median failure pressure for X-23 penetration, at which the leak areas shown in this table, is higher than the hoop failure pressure (216 psig) of the containment wall. These leak areas, therefore, are not expected to develop.

Penetration X-26 is expected to fail at a median pressure of 166 psig. The combined leak area for all safety injection penetrations is obtained by independently adding individual median leak area of 0.5 in^2.

Penetrations X-71 and X-72 are not likely to contribute to the overall leak area, as stated earlier.

The main feedwater lines (penetrations X-5 to X-8) are 18-in. diameter, Schedule 100 pipes. The failure mode of most concern is failure of the flued head due to axial loads in the pipe at the penetration. At a median pressure off 180 psig, each one of these penetrations is likely to result in a leak area of 50 in^2 each. Since all four of these can fail independent of each other, the total leak area is 200 in^2. Although the failure of a single such penetration can be considered as Type B failure, if all four main feedwater penetrations were to fail simultaneously the resulting leakage will be of Type C.

The fuel transfer tube is fixed to an elevated floor inside the containment. As the pressure in the containment increases, the containment wall moves outwards and thereby exerts pressure on the bellows. The most pertinent

Table 3.3 Leak Area Estimates for Mechanical Penetrations

Penetration	Median Leak Area in^2	Median Failure Pressure psig	Line Size
X-20 to X-23 CCW Supply and Return	6.0	>216	12"
X-24 to X-27 Safety Injection	2.0	166	4"
X-71 and X-72 Sample/Control	Negligible		$\leq$ 1"
X-5 to X-8 Main Feedwater	200	180	18"
Fuel Transfer Tube	3	172	--
X-16, X-18 On-line Purge	See Text		8"
HVAC-1,2 Containment Purge	See Text		36"

bellows from the viewpoint of containment leakage is the one inside the containment (EP-2). Three potential failure modes, in their order of decreasing probability of failure, considered are (a) failure due to overall buckling of the bellows, (b) failure due to local buckling within the convolute, and (c) failure due to meridional bending strains. The SSPSA has estimated median leak area of about 3 in^2 at a pressure of about 172 psig. This is a Type A failure.

There are two sets of containment penetrations which are open to the containment atmosphere on the inside. The on-line penetrations (X-16 and X-18) are the 8-inch purge suction and discharge lines and containment purge suction and discharge lines (HVAC-1 and 2) are the 36-inch lines. Each one of these four lines has two containment isolation valves, one inside and one outside the containment. All eight valves are pneumatically operated butterfly valves. At elevated temperatures, the seal material (usually ethylene propylene) on these valves may deteriorate and lose its sealing function. Any deposition of radioactive aerosols could further deteriorate the sealant material. Considering sealant degradation due to temperature alone, ethylene propylene seal life (Ref. 10) is 5 hours, 40 mts, or 20 mts if exposed to 400, 500 or 600 F, respectively.

In the event of the failure of the sealant material, a narrow crack leak path may develop and containment atmosphere may being to leak into the space between the two isolation valves. Since the isolation valves are closed from the containment isolation signal system, the leakage of containment atmosphere to the environment can occur only if the sealant of the outer containment isolation butterfly valve also fails. The time duration elapsed before this happens can be significantly long (of the order of hours). The SSPSA has estimated it to be long compared to the containment failure by other causes. The SSPSA study, therefore, has disregarded this release path.

The available leakage area due to sealant degradation has been estimated (Ref. 10) by assuming an equivalent clearance of 1/16 inch between valve disc and body for 'low' and 1/8 inch for 'high' estimates. This gives a total leakage area of 17 in^2 as low value and 34 in^2 as high value. As noted earlier, the outer butterfly valves must also experience high temperatures prior to a through release path. This leak area is of Category B. The SSPSA study has argued that such a leak path is not likely to result prior to a gross containment failure (Category C).

Electrical penetrations can fail primarily due to overheating of the potting compound. The SSPSA study has concluded that the failure of electrical penetrations is not expected to make a significant contribution to containment failure for any accident sequence. This conclusion, appears justified for the wet case, but, for the dry case, it is based on their estimate of slow overheating of the potting compound. A careful thermal conduction calculation should be made to check this assessment. Such a calculation, similar to the problem of vent/purge line butterfly isolation valve failure, is beyond the scope of this work and hence it was not done.

The equipment hatch and personnel lock penetrations can fail either due to pressure loading or degradation of the sealant material (generally silicone). The structural failure, prior to containment failure, appears unlikely. The sealant material can degrade at high temperatures typical of a severe accident. According to the O-Ring Handbook (see Ref. 10), silicone can survive for twenty hours when exposed to 500 F temperature. Furthermore, the personnel air lock is a double door system so even if the sealant around one door were to become ineffective, substantial time delay would be required to make the second sealant also ineffective. It, thus, appears that the equipment hatch and personnel lock penetrations do not contribute significantly to Type B failure.

3.2.5.4 Containment Failure Probability

The calculation of the probability of containment failure as a function of the pressure is quite involved. The method used and results reported in the SSPSA study seem reasonable except for the impact of all four main feedwater lines failure. The SSPSA has categorized the failure of X-8 (one of the four main feedwater lines) penetration as Type B since anticipated leak area is 50 in^2. It appears to us that when one such penetration fails, the remaining three will also fail at nearly the same pressure of 180 psig (195 psia). Any depressurization due to a 50-in^2 hole is not likely to be fast enough to reduce the containment pressure substantially prior to the failure of the three remaining penetrations. Assuming that all four main feedwater lines fail at 180 psig, an equivalent leak area of 200 in^2 will result. This failure, therefore, should be classified as Type C. The impact of this change on the containment failure probability numbers will be to reduce the rate for Type B with a corresponding increase in Type C. The total failure rate is not likely to change. Estimated containment failure fractions are compared with the SSPSA results in Fig. 3.10.

3.2.5.5 Containment Enclosure

The containment enclosure building is designed to withstand 3.5 psi pressure difference between the enclosure and the environment. During normal operation, the internal pressure is about -0.25 inches of water gauge. The SSPSA study has calculated its pressure capacity to range from more than 1 psid to 10 psid. In view of relatively strong primary containment, the role of the secondary containment is important primarily for Type B failures of the primary containment. In the event of Type C failure, the secondary enclosure building might not play any significant role as far as the source term calculation is concerned.

3.3 Definition of Plant Damage States and Containment Response Classes

The grouping of accident sequences into plant damage states proceeds from the premise that the broad spectrum of many plant failure scenarios can be discretized into a manageable number of representative categories for which a single assessment of core and containment response will represent the response of all the individual scenarios in that category.

The plant damage states classify events in accordance to the following three parameters:

1. Initiating Events
   "A" - Large Loss of Coolant Accident
   "S" - Small Loss of Coolant Accident
   "T" - Transient

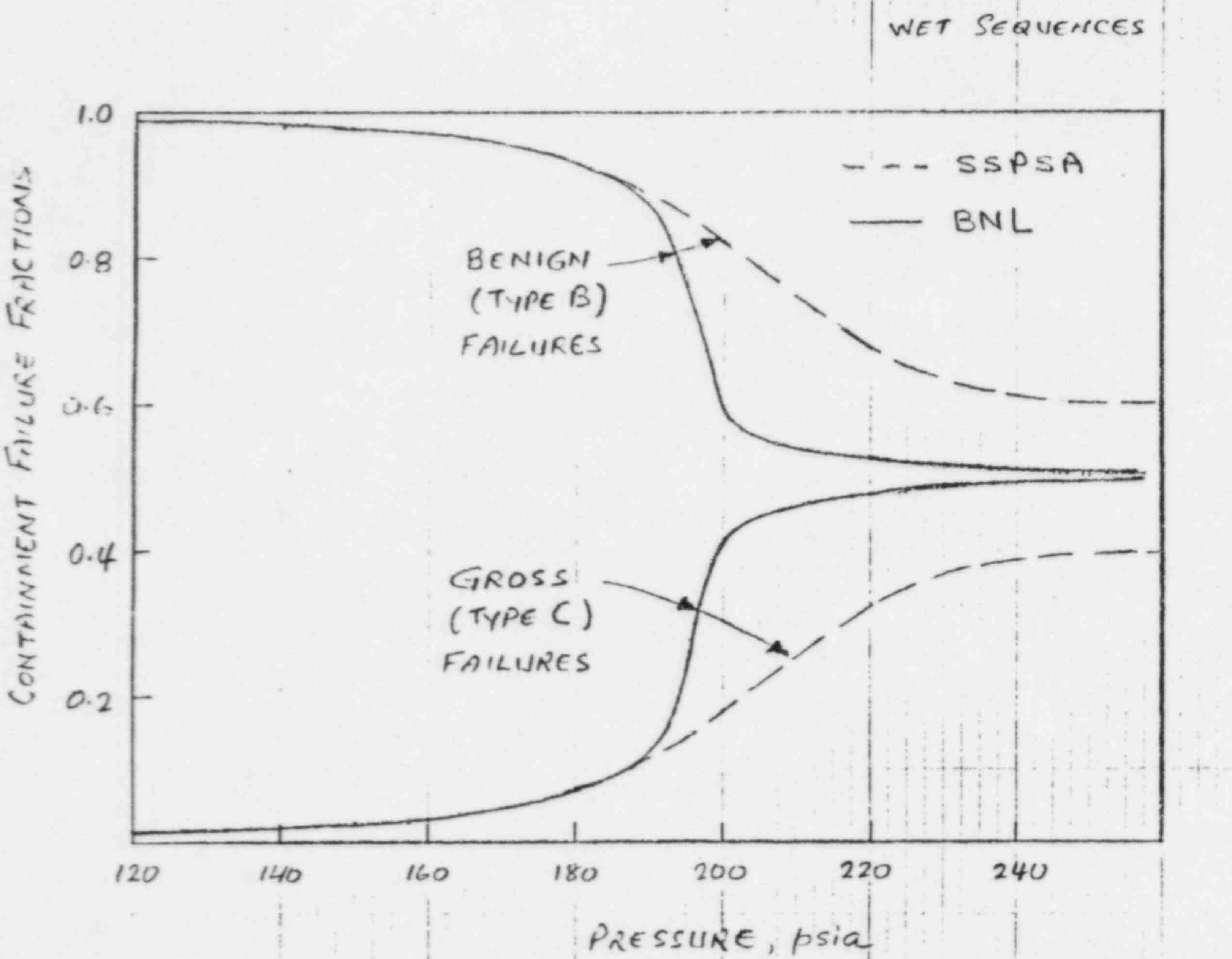


Figure 3.10 Estimated containment failure fraction.

2. Timing of Core Melt and Conditions at Vessel Failure
   "E" - No RWST Injection to RCS
   "L" - With RWST Injection to RCS
   "-" - No Emergency Feedwater
   "FW" - No Emergency Feedwater

3. Availability of Containment Systems
   "C" - Long-Term Containment Spray Cooling
   "4" - Long-Term Spray Recirculation, No Cooling
   "I" - Isolation Failure or Bypass

Figure 3.11 gives the definition of the plant damage states and their respective frequencies listed in Table 3.4 as used in the SSPSA risk model. These damage states are categorized in a matrix of eight physical conditions in the containment (numerals (1) to (8)) and six combinations of containment safety function availability (letters A to F) for a total of 48 potential plant damage states. A ninth damage state type has been defined for accident sequences involving steam generator tube ruptures. Figure 3.11 indicates that only 39 plant damage states can be identified as credible sequences.

From the viewpoint of containment response, many of the plant damage states can be grouped into containment classes. The classes defined in Table 3.5 are differentiated primarily according to spray availability. The frequency of each containment class is the sum of the frequencies of the plant states included therein.

Annual plant state frequencies calculated by the applicant[5] for both internal and external events were reviewed by the Lawrence Livermore National Laboratory[6] and were found acceptable. Table 3.6 presents the calculated containment class frequency estimates for internal events, fires, floods and truck crashes; moderate and severe seismic events.

In order to comprehensively assess the risk from seismic events, it is necessary to make separate consequence calculations for those accidents which are initiated by earthquakes severe enough to impair evacuation. For this purpose, the seismic frequency estimates are divided into two categories in Table 3.6. The seismic events with instrument peak ground acceleration below 0.5g can be binned with internal events, fires, floods and truck crashes. Seismic events with acceleration greater than 0.50g are judged to impair evacuation, and must be treated separately in the consequence analysis.

These containment response classes (or plant damage states) are the starting point for the containment event three analysis and they define the link or interfaces with the plant analysis.

3.4 Containment Event Tree and Accident Phenomenology

An important step towards the development of the containment matrix involves the quantification of branch point probabilities in the containment event tree. These probabilities depend heavily on the analyses of degraded and core melt phenomenology and the containment building response described in Appendix H of the SSPSA.[5]

CORE MELT TIME SINCE REACTOR SHUTDOWN	CONDITIONS AT TIME OF REACTOR VESSEL MELT-THROUGH		CONTAINMENT INTACT AT TIME OF CORE MELT START							
			YES				NO			
			CONTAINMENT FUNCTIONS AVAILABLE				ACTIVITY RELEASE			
								UNFILTERED		
	PRESSURE IN REACTOR VESSEL (≷ 300 PSIA)	RWST INJECTION INITIATED	HEAT AND FISSION PRODUCT REMOVAL (A)	HEAT REMOVAL ONLY (B)	FISSION PRODUCT REMOVAL ONLY (C)	NONE (D)	FILTERED (E)	OPENING > 3 INCH DIAMETER (F)	OPENING < 3 INCH DIAMETER (FP)	AIRCRAFT CRASH (FA)
EARLY < 8 HOURS	LOW	NO (1)				1D		1F	1FP	1FA
		YES (2)	2A		2C	2D	2E	2F	2FP	2FA
	HIGH	NO (3)				3D		3F	3FP	
		YES (4)	4A		4C	4D	4E	4F	4FP	
LATE > 8 HOURS	LOW	NO (5)								
		YES (6)	6A		6C	6D	6E	6F	6FP	6FA
	HIGH	NO (7)				7D		7F	7FP	
		YES (8)	8A		8C	8D	8E	8F	8FP	
CORE MELT WITH NONISOLATED STEAM GENERATOR TUBE RUPTURE (9)			9A		9C	9D				

PRECLUDED BY SEABROOK STATION DESIGN FEATURES

NOT USED BECAUSE OF UNCERTAINTIES IN FAN COOLER CAPABILITY

PHYSICALLY POSSIBLE BUT NOT USED IN SSPSA RISK MODEL

Plant State	Represents
2A	AEC
4A	TEC,SEC
2C/6C	AE4
4C/8C	TE4
1D	AE
2D/6D	AL
3D/7D	SE,TE/TEFW
4D/8D	SL,TL
2E/6E	AECI
4E/8E	TECI
1F	V
2F/6F	AEI
3F/7F	SEI
4F/8F	SLI

Figure 3.11 Definitions of the plant damage states used in SPSS.

Table 3.4 Frequencies of Occurrence of the Plant Damage States

Plant Damage State	Frequency (events per reactor year)	Plant Damage State	Frequency (events per reactor year)
1D	3.03(-7)	6A	3.41(-7)
1F	1.89(-6)	6C	3.57(-10)
1FA	6.10(-11)	6D	2.49(-7)
1FP	8.52(-7)	6E	5.30(-14)
2A	1.85(-6)	6F	2.08(-16)
2C	1.91(-9)	6FA	1.11(-11)
2D	2.53(-7)	6FP	1.34(-12)
2E	1.40(-13)	7D	7.06(-5)
2F	1.06(-13)	7F	3.55(-8)
2FA	3.10(-11)	7FP	1.09(-5)
2FP	1.58(-10)	8A	4.50(-5)
3D	1.94(-5)	8C	4.29(-8)
3F	5.00(-7)	8D	5.51(-5)
3FP	6.21(-6)	8E	5.02(-11)
4A	1.28(-5)	8F	1.02(-10)
4C	1.65(-7)	8FP	1.95(-7)
4D	2.79(-6)	9A	7.51(-10)
4E	2.24(-11)	9C	3.62(-13)
4F	2.25(-13)	9D	9.09(-9)
4FP	1.18(-7)		
		TOTAL	2.30(-4)

NOTE: Exponential notation is indicated in abbreviated form; i.e., 3.03(-7) = 3.03×10^{-7}.

Table 3.5 Containment Response Class Definitions

Class	Plant State	Represents
1	1D	AE
2	2A/6A, 4A/8A	AEC, TEC, SEC
3	2C/6C, 4C/8C	AE4, TE4, SE4
4	3D/7D	SE, TE, TEFW
5	2D/6D, 4D/8D	AL, SL, TL
6	1F, 2F, 3F, 4F, 6F, 7F, 8F	V
7	2E/6E, 4E/8E	AECI, TECI
8	1FP, 3FP/7FP	Small leaks w/o RWST
9	2FP/6FP, 4FP/8FP	Small leaks w/ RWST
10	1FA, 2FA/6FA	Aircraft crashes
11	9A	V2 (SGTR)
12	9C	V2 (SGTR)
13	9D	V2 (SGTR)

Table 3.6 Containment Class Mean Frequencies†

Containment Response Class	Frequency (per reactor year)				
	Internal, Fires, Floods and Truck Crashes	Seismic <0.5g	Seismic >0.5g	Total Seismic	Internal and External
1	1.08E-7	-	1.95E-7	1.95E-7	3.03E-7
2	5.70E-5	1.54E-6	1.24E-6	2.78E-6	6.0E-5
3	1.80E-7	1.91E-8	*	1.91E-8	1.99E-7
4	8.60E-5	1.85E-6	2.27E-6	4.12E-6	9.0E-5
5	5.50E-5	1.10E-6	1.76E-6	2.86E-6	5.8E-5
6	1.80E-6	1.66E-7	3.93E-7	5.59E-7	2.4E-6
7	*	*	*	*	*
8	*	5.29E-6	1.25E-5	1.79E-5	1.79E-5
9	*	1.12E-7	2.40E-7	3.52E-7	3.52E-7
10	*	-	-	-	*
11	*	-	-	-	*
12	*	-	-	-	*
13	*	-	-	-	*

†Reference [5] Tables 5.1-3 and 9.2-9.

*Indicates frequencies less than 10^{-8} yr^{-1}.

The SSPSA containment event tree uses the twelve top events identified in Table 3.7 as major phenomenological phases which could occur with respect to the formation and location of core debris. These processes are grouped into four phases following and accident initiation (1) phenomena occurring while the core is still in place; (2) phenomena occurring while the core is located below the lower grid plate but is still in the reactor vessel; (3) ~~phenomena~~ occurring with the core debris located in the reactor cavity and onthe containment floor; and (4) the phenomena involving long-term cooling ~~of the~~ containment and/or basemat penetration.

3.5 Containment Matrix (C-Matrix)

The twelve top events in the Seabrook containment event tree are summarized in Table 3.7. A negative response at any of the five nodes (4, 8, 10, 11, and 12) in the containment event tree results in the failure of the containment building by a variety of failure modes. Each of these failure modes results in a particular radiological release category. For those paths that do not have a negative response at any of the five nodes, the path will eventually result in no failure of the containment. The containment event tree thus links the plant damage states to a range of possible containment failure modes via the various paths through the tree. For a given tree, each path ends in a conditional probability (CP) of occurrence, and these CPs should sum to unity. The quantification of an event tree is the process by which all the paths are combined to give the conditional probabilities of the various release categories. In SSPSA, fourteen release categories are used for the quantification as summarized in Table 3.8. Note that two of these release categories (namely, S5 and $\hat{S}5$) correspond to intact/isolated containment. Fission product release for this category would, therefore, be via normal leakage paths in the containment (and enclosure) building, which can be different depending on availability of the enclosure building ventilation/ filteration system.

Table 3.9 sets forth a simplified containment matrix (C-matrix) for the Seabrook plant using the containment response class definitions discussed in Section 3.3, and the release category definitions given in Table 3.8. In arriving at the C-matrix of Table 3.9 all of the very low probability values were disregarded. This is shown[7] to be insignificant to the risk estimate.

The present assessment of containment response for Seabrook plant is not based upon independent confirmatory calculations of accident progression and containment response. Instead, the knowledge gained from review of similiar risk studies for other[1,3,4] pressurized water reactors with large dry containments is used to guide this assessment.

The mode and timing of containment failure cannot be calculated with a great degree of accuracy. Judgements must be made about the nature of the dominant phenomena and about the magnitude of several important parameters. Furthermore, the codes and methods used for these calculations are approximate and do not model all of the detailed phenomena. Fortunately, risk measured in personal exposure is not sensitive to minor variations in failure mode and timing. It is important, however, to properly characterize the major attributes of failure mechanisms; (1) whether the failure is early or late, (2) whether it is by overpressurization, bypass, or basemat melt-through and, (3) whether or not radionuclide removal systems are effective.

Table 3.7 Accident Phase and Top Events for the Seabrook Containment Event Tree

Accident Phase	Top Event	
Initiator	1	Plant State
Debris in Vessel	2 3 4	Debris Cooled in Place No H_2 Burn Containment Intact
Debris in Reactor Cavity	5 6 7 8	Debris Dispersed from Cavity Debris Cooled No H_2 Burn Containment Intact
Long-Term Behavior	9 10 11	No Late Burn Containment Shell Intact Basemat Intact
Failure Mode	12	Benign Containment Failure (Small Leak)

Table 3.8 Release Categories Employed in the Seabrook Station Risk Model

Release Category Group	Release* Category	Definition
Containment Intact/Isolated	S5	Containment intact/isolated with enclosure air handling filtration working.
	$\hat{S5}$	Same as S5 but with enclosure air handling filtration not working.
Long-Term Containment Failure	S2	Early containment leakage with late overpressurization failure and containment building sprays working.
	$\overline{S2}$	Same as S2, but with containment building spray not working.
	$\overline{S2V}$	Same as $\overline{S2}$, but with an additional vaporization component of the source term.
	S3	Late overpressurization failure of the containment with no ear'y leakage and containment building sprays working.
	$\overline{S3}$	Same as S3, but with containment building sprays not working.
	$\overline{S3V}$	Same as $\overline{S3}$, but with an additional vaporization component of the source term.
	S4	Basemat penetration failure, sprays operating
	$\overline{S4V}$	Containment basemat penetration failure with containment building sprays not working and additional vaporization component of the source term.
Early Containment Failure/Bypass	S6	Containment bypass or isolation failure with containment building sprays working.
	$\overline{S6V}$	Same as S6, but with containment building sprays not working and an additional vaporization component of the source term.
	S1	Early containment failure due to steam explosion or hydrogen burn with containment building sprays working.
	$\overline{S1}$	Same as S1, but with containment building sprays not working.

*S denotes applicability to Seabrook Station; number corresponds with containment failure mode; bar denotes nonfunctioning of containment building sprays; and V denotes achievement of sustained elevated core debris temperatures and associated vaporization release.

Table 3.9 Simplified Containment Matrix for Seabrook

Containment Class	Release Category										
	S1	S2	S3	S5	S6	$\overline{S2}$	$\overline{S3}$	$\overline{S2V}$	$\overline{S3V}$	$\overline{S4V}$	$\overline{S6V}$	$\overline{S6V}$-d
1									0.60	0.40		
2			0.01	0.99								
3			1.0									
4									0.89	0.11		
5							1.0					
6											1.0	
7					1.0							
8								1.0				
9						1.0						
10	1.0											
11		1.0										
12												
13												1.0

The assessment of the containment response and failure mechanisms is based on the general understanding of that accident phenomenology and the containment design characteristics discussed earlier. The phenomena of interest may be summarized as follows:

Early Failure (S1, $\overline{S1}$) which can result from a steam explosion or an early hydrogen burn is believed to be unlikely. Although explosions in the reactor vessel lower plenum are highly probable, the resulting mechanical energy would be limited by the fraction of the core which could participate in a single explosion and by the efficiency of the process. In recent PRA reviews,[4,7] we have assigned a conditional probability of 10^{-4} to steam explosion induced containment failure. This probability leads to the conclusion that steam explosions would have a negligible effect on risk, and consequently, the applicants 5×10^{-4} value is not included in the simplified C-matrix.

The conditional probability for an early containment failure due to external events (i.e., aircraft crashes) is assigned 1 in the SSPSA as shown in Table 3.6. This simply indicates that an aircraft crash into the containment is assumed to fail the containment structure with certainty.

Early containment failure could also conceivably result from direct heating due to a rapid dispersal of the core debris throughout containment in the form of aerosols. The dispersal could only be caused by the high primary system pressures that may exist at vessel failure for a number of transient sequences (recent calculations[11] indicate that there exists a propensity for establishment of natural convection pattern inside the reactor vessel and the hot leg; which can cause rapid heatup of the RCS boundaries possibly leading to failure and depressurization prior to bottom head melt through, thus eliminating, high pressure ejection sequences). The aerosols could rapidly pressurize containment by direct heat exchange and concomitant chemical reactions. Scoping calculations performed by the Containment Loads Working Group (CLWG) showed that a very severe challenge to the containment integrity could result provided 25 percent of the core mass were converted to aerosols.[12] However, no consensus could be reached among the CLWG analysts as to the credibility of this parameter value, and this failure mode is still speculative. Furthermore, the configuration of the Seabrook lower cavity would tend to reduce the dispersal of core debris beyond the reactor cavity boundaries.

For the reasons outlined above as well as the high containment failure pressure for Seabrook, it is concluded that early overpressure failure has a very low likelihood.

?

Early Containment Leakage (S2, $\overline{S2}$, $\overline{S2V}$) without gross failure of containment building is expected to occur for nonisolated steam generator tube rupture event with containment sprays available (S2), for large break LOCA sequences with RWST injection in the absence of sprays ($\overline{S2}$), and for dry cavity sequences with a vaporization release ($\overline{S2V}$).

There seems to be a basic inconsistency in assigning plant damage states to this failure mode as defined in the C-matrix. Specifically, large break LOCA sequences with RWST injection in the absence of containment sprays are expected to lead to an $\overline{S3}$ failure mode with 100% probability (see $\overline{S3}$ below);

while they are also assigned to $\overline{S2}$ with 100% probability. This can be correct only if the initiator and the sequences are indeed different, but at this time we cannot resolve the inconsistency.

Similarly, the significance of containment functions on steam generator tube rupture sequences is not at all obvious.

<u>Late Overpressurization Failure</u> (S3, $\overline{S3}$, $\overline{S3V}$) can occur due to steam production in a wet cavity or noncondensable gas production as a result of core-concrete interaction for a dry cavity situation. For sequences in which early and intermediate failure is not expected to occur, and for which containment sprays are inoperable, failure is expected to be a certainty.

The conditional probability for a late overpressurization failure with a vaporization release (dry cavity) is shown to be 0.60. This results from the relative competition between the late overpressure failure and the basemat penetration ($\overline{S4V}$) for accident sequences without the containment sprays.

The failure time for the late overpressurization failure mode is much longer than previously calculated for other large dry containment.[1,3,4] This is as a result of the very high failure pressure for the Seabrook containment. As a consequence of this high containment failure pressure (median pressure of 211 for wet and 187 psia for dry* sequences) it is difficult to challenge the containment integrity by any conceivable event.

Hydrogen deflagration early in the accident sequence or later after vessel failure when steam condensation occurring as a result of reactivation of sprays (due to regaining of ac power), or other natural heat sink mechanisms,[7] which can produce a deinerted atmosphere is not expected to challenge the containment integrity.

The impact of changes in the containment failure distribution discussed in 3.2.5.4 is not significant for late failures.

<u>Basemat Penetration Failure</u> (S4, $\overline{S4V}$) can only result in the absence of containment heat removal system (sprays) for a dry cavity. A 26-inch high curb surrounds the reactor cavity that prevents the entry of water into the cavity unless the full RWST has been injected. The conditional probability of the basemat melt though is always less than the late overpressurization failure, particularly for Seabrook with the natural bed rock formation directly under the basemat foundation. Therefore, the basemat penetration failure probabilities are conservatively assigned.

<u>No Failure (S5, $\overline{S5}$)</u> would result for all sequences with full spray operation. The radiological releases are thus limited to the design basis leakage with essentially negligible off-site consequences.

<u>Containment Isolation Failure (S6, $\overline{S6V}$)</u> is represented by an 8-inch diameter purge line. The accident sequences where the containment is either not

*For dry sequences, only primary system water inventory is available in the containment. In this case, the containment atmosphere becomes superheated and, at failure, the temperature can exceed 700°F.

isolated or bypassed (Event V) are assigned a conditional probability of unity to this release category.

An interfacing systems LOCA (V sequence) results from valve disc rupture or disc failing open for series check valves that normally separate the high pressure system. This event results in a LOCA in which the reactor coolant bypasses the containment and results in a loss-of-coolant outside the containment. Furthermore, the concurrent assumed loss of RHR and coolant make-up capability leads to severe core damage. In the SPSS, three possible interfacing systems LOCA sequences have been found and discussed. These are

1. Disc rupture of the check valve in the cold-leg injection lines of the RHR.
2. Disc rupture of the two series motor-operated valves in the normal RHR hot-leg suction.
3. Disc rupture of the motor-operated valve equipped with a steam mounted limit switch and "disc failing open while indicated closed" in the other motor-operated valve in the normal RHR hot-leg suction.

For the V-sequence, the core melts early with a low RCS pressure and a dry reactor cavity at vessel melt-through. The containment sump remains dry and recirculation is not possible.

The core and containment phenomenology used to arrive at the split fractions for the containment event tree and thus the C-matrix are in general agreement with the other previously NRC reviewed studies[1,3,4] for PWRs with large dry containments. Furthermore, the claimed unusually high strength of the Seabrook containment reduces the impact of sensitivity caused by uncertainties in the severe accident progression. However, should the claimed strength of the containment be reduced to levels comparable to some of the other large dry containments, the impact of uncertainties may become significantly more pronounced, as discussed in our review of the MPSS-3.[7]

3.6 Release Category Frequencies

Based on the containment class frequencies in Table 3.6 and the containment failure matrix of Table 3.9, the release frequencies were computed and are summarized in Table 3.10.

Table 3.10 indicates that only light of the release categories dominate the total release frequency.

Tables 3.11 and 3.12 set forth the contribution to core melt frequency from the various containment response classes and release categories, respectively. It is seen that containment classes 2, 4, and 5 dominate the core melt frequency while, the release categories S5 (containment intact), $\overline{S3}$ and $\overline{S3V}$ dominate the source term frequency.

Table 3.10 Frequency of Dominant Release Categories (yr^{-1})

Category	Internal, Fires, Floods and Truck Crashes	Seismic <0.5g	Seismic >0.5g	Internal and External
S3	7.50E-7	3.45E-8	2.69E-7	1.05E-6
S5	5.64E-5	1.52E-6	1.23E-6	5.92E-5
$\overline{S2}$	*	1.12E-7	2.40E-7	3.52E-7
$\overline{S3}$	5.50E-5	1.10E-6	1.76E-6	5.79E-5
$\overline{S2V}$	*	5.29E-6	1.25E-5	1.78E-5
$\overline{S3V}$	7.66E-5	1.65E-6	2.14E-6	8.04E-5
$\overline{S4V}$	9.50E-6	2.04E-7	3.27E-7	1.0E-5
$\overline{S6V}$	1.80E-6	1.66E-7	3.93E-7	2.36E-6

Table 3.11 Contribution of Containment Response Classes to the Total Core Melt Frequency

Containment Class	Internal, Fires, Floods and Truck Crashes	Seismic <0.5g	Seismic >0.5g	Total Seismic	Internal and External
1	-	-	-	-	<0.01
2	0.25	<0.01	<0.01	0.01	0.26
3	-	-	-	-	<0.01
4	0.37	0.01	0.01	0.02	0.39
5	0.24	-	0.01	0.01	0.25
6	0.01	-	-	-	0.01
7	*	*	*	*	*
8	*	0.025	0.055	0.08	0.08
9-13	*	*	*	*	*

Table 3.12 Release Category Frequency as a Fraction of Core Melt Frequency

Release Category	Internal, Fires, Floods and Truck Crashes	Seismic <0.5g	Seismic >0.5g	Total Seismic	Internal and External
S3	<0.01	<0.01	<0.01	<0.01	<0.01
S5	0.25	<0.01	<0.01	0.01	0.26
$\overline{S2}$	*	<0.01	<0.01	<0.01	<0.01
$\overline{S3}$	0.24	<0.01	<0.01	0.01	0.25
$\overline{S2V}$	*	0.03	0.05	0.08	0.08
$\overline{S3V}$	0.33	0.01	0.01	0.02	0.35
$\overline{S4V}$	0.04	<0.01	<0.01	<0.01	0.04
$\overline{S6V}$	0.01	<0.01	<0.01	<0.01	0.01

4. ACCIDENT SOURCE TERMS

In this chapter the approach utilized in the SSPSA to determine the fraction of fission products originally in the core which can leak to the outside environment will be outlined. The fission product source to the environment as calculated by this approach for each release category will also be discussed.

4.1 Assessment of Severe Accident Source Terms

As in the Reactor Safety STudy (RSS)[13] the CORRAL-II code was used in the SSPSA for determining fission product leakage to the environment. This code takes input from the thermal-hydraulic analysis carried out for the containment atmosphere. In addition, it needs the time-dependent emission of fission products. The fission products were assumed to be released in distinct phases as suggested in the RSS, namely, the Gap, Melt, and Vaporization phases. The time dependence of these phases is determined by the timing of core heatup, primary system failure, and core/concrete interaction. The methods used in the SSPSA differ from the RSS methods in the following ways:

1) The treatment of iodine was changed and iodine was treated as cesium iodide. This was accomplished by merely using the same fraction of core inventory released for both the cesium group and the iodine group,

2) Leakage releases are represented by a multi-puff model,

3) An uncertainty analysis was carried out in which it was attempted to account for shortcomings in the RSS methods.

In general, the net result of the SSPSA analysis was to reduce the fractional release of particulate fission products. This will be discussed in more detail later. In all, fourteen releases were determined ranging from containment bypass sequence to the no-fail sequence as shown in Table 3.8.

These release categories were evaluated by considering the containment failure mode, the availability of the spray system, and whether or not the cavity was wet or dry. Table 4.1 shows the point-estimate releases as determined by the methods outlined above. Containment failure mode S1 corresponds to a gross failure of the containment, resulting from a steam explosion, early pressure spikes, or early hydrogen burns. Failure mode S2 represents a loss of containment function early in the accident sequence. This loss of function takes the form of an increase in the leak rate to 40% per day where it stays until the containment fails due to overpressurization. Failure mode S3 represents a late overpressurization failure of the containment driven by decay heat or late hydrogen burns. Failure mode S4 represents a basemat meltthrough, S5 represents no containment failure and the leak rate is limited to the containment design basis leak rate. Finally, failure mode S6 represents sequences where the containment is failed or bypassed as part of the initiating event.

The second parameter considered in defining the source term is the availability of sprays. This is determined by the plant damage states. Those

Table 4.1 Seabrook Point-Estimate Release Categories

Seabrook Release Category	Accident Sequence	Time of Initiation of Release (hours)	Release Duration (hours)	Warning Time (hours)*	Energy Release 10^6 cal/sec	Release Height (meters)	Release Fractions by Group						
							Xe	I-2**	Cs	Te	Ba	Ru	La
S1	AEC	1.9	0.5	0.35	< 10.0	10.	.94	.023	.023	.24	.0033	.41	9.8-5
S2	AEC	2.6	1.0	1.9	< 10.0	10.	.89	2.1-5	2.1-5	4.4-6	2.9-6	8.8-7	8.8-8
S3	TE4	66.1	0.5	62.5	210.0	10.	.90	1.0-7	1.0-7	1.9-8	1.3-8	3.8-9	3.8-10
S5	AEC	1.9	24.	0.35	< 10.0	10.	.0091	3.5-8	3.5-8	6.1-9	4.0-9	1.2-9	1.2-10
S6	TEC1	4.5	4.0	4.0	< 10.0	10.	.90	.0036	.0036	.00067	.00044	.00013	1.3-5
$\overline{S1}$	AL	1.4	0.5	0.3	210.0	10.	.94	.75	.75	.39	.093	.46	.0028
$\overline{S2}$-1		7.3	9.1	6.2	< 10.0	10.	.15	.092	.092	.017	.011	.0034	.00034
$\overline{S2}$-2		20.3	17.	19.2	< 10.0	10.	.24	.093	.093	.017	.012	.0034	.00034
$\overline{S2}$-3		29.3	1.2	28.2	< 10.0	10.	.51	.12	.12	.023	.015	.0046	.00046
$\overline{S2}$ Total	AL	7.3	27.3	6.2	--	10.	.90	.31	.31	.057	.038	.011	.0011
$\overline{S3}$	AL	27.2	0.5	26.4	210.0	10.	.90	.122	.122	.022	.015	.0044	.00044
S5	TEC	4.3	24.	0.6	< 10.0	10.	.014	5.2-7	5.2-7	9.5-8	6.3-8	1.9-8	1.9-9
$\overline{S2V}$-1		2.2	3.5	1.9	< 10.0	10.	.05	.037	.037	.0069	.0045	.0014	.00014
$\overline{S2V}$-2		6.2	7.2	5.9	< 10.0	10.	.10	.072	.072	.0080	.0079	.0062	.0010
$\overline{S2V}$-3		35.2	78.0	34.9	< 10.0	10.	.85	.20	.20	.30	.022	.018	.0030
$\overline{S2V}$ Total	AE	2.2	88.7	1.9	--	10.	1.0	.31	.31	.32	.034	.025	.0042
$\overline{S3V}$	TE	81.5	0.5	76.2	210.0	10.	1.0	.024	.024	.030	.0026	.0023	.00039
$\overline{S4V}$	AE	50.0	0.5	49.6	210.0	10.	1.0	.058	.058	.072	.0062	.0054	.00091
$\overline{S6V}$-1		2.2	1.0	1.7	.35	10.	.15	.11	.11	.020	.014	.0041	.00041
$\overline{S6V}$-2		4.2	3.0	3.7	.33	10.	.31	.14	.14	.026	.017	.0052	.00051
$\overline{S6V}$-3		11.2	10.0	10.7	.24	10.	.51	.18	.18	.36	.017	.024	.0044
$\overline{S6V}$ Total	SEI	2.2	14.0	1.7	.26	10.	.97	.43	.43	.40	.048	.033	.0053

NOTE: Exponential notation is shown in abbreviated form; i.e., 2.1-5 = 2.1×10^{-5}.

*Based on time of gap release except for S6 and $\overline{S6V}$ where it is based on 0.5 hours after accident initiation.
**Elemental iodine - not used, all iodine is treated as CsI.

release categories with operating sprays systems are designated S1 to S6, while those with spray systems not operating are designated $\overline{S1}$ to $\overline{S6}$.

The third and final parameter considered in differentiating between source terms distinguishes between wet and dry cavities. In the case of dry cavities a vaporization release due to core/concrete interactions will occur, while for wet cavities the core debris is assumed to be quenched or the water in the cavity will scrub the vaporization release thus effectively reducing the release to zero. The release categories which include a vaporization release include a "V" in their designation as shown in Table 3.8.

From the point of view of risk it was found[5] that $\overline{S2V}$, $\overline{S3}$, $\overline{S3V}$, and $\overline{S6V}$ were dominant either for acute or latent health effects. In view of this result these four categories will be considered in more detail.

Release categories $\overline{S3}$ and $\overline{S3V}$ have late overpressurization failure modes, with no spray systems operating and differ only in the omission or inclusion of a vaporization release, respectively. The containment at Seabrook is calculated to fail at a median pressure of 211 psia for wet sequences and 187 psia for dry sequences. At this pressure a gross failure is expected resulting in a puff release of approximately 0.5 hr release duration. From Table 4.1 it is seen that the $\overline{S3}$ and $\overline{S3V}$ sequences fail at 27.2 hrs and 91.5 hrs, respectively. These failure times are several hours later than was calculated for Indian Point, Zion, and Millstone-3. The primary reason for the later failure in this case is due to the superior strength of the containment structure. Table 4.2 compares the $\overline{S3}$, $\overline{S3V}$ release parameters with similar parameters for the other three reactors mentioned above. Note that a fair comparison should set (OI+I) equal to (Cs-Rb), since iodine was treated as CsI. It is seen that I, Cs, and Ba groups for $\overline{S3}$ are approximately half the other releases, while the Te, Ru, and La groups are low by approximately an order of magnitude. This difference is due to the latter failure time, allowing more time for settling and the absence of a vaporization release, which dominates the release of Te, Ru, and La. A similar comparison for the $\overline{S3V}$ release indicates a uniform reduction of approximately an order of magnitude for all species. The reduction is entirely due to the late failure time for this sequence.

Another important consideration is the increased rate of release due to an increase in the leak area prior to attaining gross failure conditions. This can also impact the radionuclide transport mechanisms inside the containment due to changes in the containment thermal hydraulic conditions.

Release category $\overline{S2V}$ is associated with early containment failure in which the containment function is compromised by increasing the leakage area in such a way that the leak rate increases from 0.1% per day to 40% per day. This release rate is not enough to prevent an ultimate overpressurization failure. This release is modeled as a multi-puff release. The first puff corresponds to the release up to the time when vaporization starts (melt+gap). The second puss includes the period of vaporization release and the third puff is equivalent to an overpressurization failure at the time of catastrophic containment failure. In this model the duration of the melt

Table 4.2 Late Overpressurization Failure Comparison

	Seabrook[5]		Millstone-3[7]	Zion/Indian[14] Point Study	Indian[3] Point
	$\overline{S3}$	$\overline{S3V}$	M-7	TMLB'	2RW
Xe	9.0(-1)	1.0	9 (-1)	9.6(-1)	1.0
IO+I	1.2(-1)	2.4(-2)	1.5(-1)	1.05(-1)	9.3(-2)
Cs-Rb	1.2(-1)	2.4(-2)	3.0(-1)	3.4(-1)	2.6(-1)
Te-Sb	2.2(-2)	3.0(-2)	3.0(-1)	3.8(-1)	4.4(-1)
Ba-Sr	1.5(-2)	2.6(-3)	3.0(-2)	3.7(-2)	2.5(-2)
Ru	4.4(-3)	2.3(-3)	2.0(-2)	2.9(-2)	2.9(-2)
La	4.4(-4)	3.9(-4)	4.0(-3)	4.9(-3)	1.0(-2)
T (release) (hrs)	27.2	81.5	20		
T (duration) (hrs)	0.50	0.50	0.50	0.50	
Energy (Btu/hr)	300E7	300E7	540E6	150E6	

release is seen to be 3.5 hours, vaporization release 7.2 hours and the remaining release 78.0 hours. It is not clear that the melt release in this case is 5.3 hours, however, it does not seem to be unreasonable. A 7.2 hour duration for the vaporization release is not consistent with the RSS,[13] which only allows 2 hours for this phase. Finally, it is not clear how the 78 hours for the last phase was determined. The release duration for a single puff, which is the sum of the above three phases leads to a release time of 88.7 hours which seems extraordinarily long. Our recommendation would be to reduce these times to be more consistent with RSS methods (see Table 4.7).

The total release of fission products from the sequences can be compared to the M-4 release determined for the Millstone-3 study. This comparison is made in Table 4.3. It is seen that, once adjustments are made for the different ways in which iodine is treated, the $\overline{S2V}$ release is approximately half the M-4 release. Without the benefit of a calculation, it is difficult to judge whether the differences are reasonable. However, a possible reason for this reduction is the credit taken for the enclosure building surrounding the actual containment building. This feature is unique to the Seabrook containment structure.

Release category $\overline{S6V}$ has binned into it an isolation failure corresponding to an 8" diameter breach in containment and the interfacing LOCA (V-sequence). This sequence is also represented by a multi-puff release. In this case as in the previous case, the total release time is long compared to acceptable limits of the RSS[13] consequence model. Our recommendation would be to reduce these times to more reasonable values (see Table 4.7).

The release fraction can be compared (Table 4.3) to the M-4 release from Millstone-3, PWR-2 for the RSS and the V-sequence from the RSSMAP study for Surry.[15] Except for the iodine group, it is seen that the release fractions are comparable. If the iodine group were set equal to the cesium group value, it is seen that the value for $\overline{S6V}$ would be the lowest release fraction.

4.2 Source Term Uncertainty Analysis

In this section we will briefly describe the uncertainty analysis carried out for the four dominant accident sequences and, where possible compare the fission product leakage to the environment to more mechanistic determinations. There are two contributors to the uncertainty in release characterization. First, uncertainty in time parameters which are influenced by:

1) Prediction of key event times, and

2) The mix of accident sequences binned into a release category.

Second, uncertainties in release fractions, which are influenced by:

1) Analysis methods and data, and

2) Uncertainties in timing of key events.

Table 4.3 Comparison of Releases for Failure to Isolate Containment and the By-Pass Sequence

	Seabrook[5]		Millstone-3[7]	RSS[13]*	RSSMAP[15]
	$\overline{S2V}$	$\overline{S6V}$	M-4	PWR-2	V-Sequence
Xe	1.0	9.7(-1)	9.0(-1)	1.0	1.0
OI+I	3.1(-1)	4.3(-1)	2.0(-1)	7.0(-1)	4.8(-1)
Cs-Rb	3.1(-1)	4.3(-1)	6.0(-1)	5.0(-1)	7.9(-1)
Te-Sb	3.2(-1)	4.0(-1)	5.0(-1)	3.0(-1)	4.4(-1)
Ba-Sr	3.4(-2)	4.8(-2)	7.0(-2)	6.0(-2)	9.0(-2)
Ru	2.5(-2)	3.3(-2)	5.0(-2)	2.0(-2)	4.0(-2)
La	4.2(-3)	5.3(-3)	7.0(-3)	4.0(-3)	6.0(-3)
T (release) (hrs)	2.2	2.2	0.20	2.5	2.5
T (duration) (hrs)	88.7	14	2.0	1.0	1.0
Energy (Btu/hr)	140E6	4E6	70E6	20E6	0.5E6

*The same as M1A release category in Millstone-3.[7]

The above principles were used to determine source term multipliers which would give a range of fission product leakage to the environment. A probability is associated with each source term, and for later overpressurization failure modes ($\overline{S3}$, $\overline{S3V}$, and $\overline{S2V}$) the following discrete probability distribution is used, i.e.,

Subcategory	Probability
$\overline{S3}$-a	.02
$\overline{S3}$-b	.08
$\overline{S3}$-c	.30
$\overline{S3}$-d	.60

This indicates, for example that there is an 8% confidence level that $\overline{S3}$-b correctly defines the source term for the $\overline{S3}$ release category.

The results of this analysis for the overpressurization failure modes is:

	Particulate Release Factor (multiplier)		
Probability	$\overline{S3}$	$\overline{S3V}$	$\overline{S2V}$
.02	.22	.63	.17
.08	.071	.22	.07
.30	.024	.065	.02
.60	.0071	.021	.007

From this table it is seen that for the most likely release, i.e., "d", the reduction factors of the source term are substantial.

The first two releases can be compared to releases published in BMI-2104 Volume V (Surry) for the TMLB'-ε and AB-ε sequences. These two sequences correspond to late containment failures and are both binned into $\overline{S3}$ and $\overline{S3V}$ sequences. A comparison of these sequences is shown on Table 4.4. From this table it is evident that for the volatile species, Xe, Cs, and I, the release categories $\overline{S3}$ and $\overline{S3V}$ bracket or exceed the mechanistic estimates carried out in BMI-2104 for both TM:B' and AB sequences. However, for the less volatile species Te, Ba, Ru, and La, the release of the AB sequence is the only one bracketed or superseded by the $\overline{S3}$ and $\overline{S3V}$ releases. The release fraction determined for the TMLB' sequence is higher than all the $\overline{S3}$ and $\overline{S3V}$ releases. This discrepancy is primarily due to the comparatively early failure time. It is felt that agglomeration and settling would reduce the source for the TMLB' sequence to values close to those reported for $\overline{S3}$ and $\overline{S3V}$. No comparative sequence for $\overline{S2V}$ was analyzed in BMI-2104.

In the case of the $\overline{S6V}$ release category a different probability distribution was used. This change reflects the break location, which initiates the

Table 4.4 Comparison of AB-ε and TMLB'-ε (BMI-2104) to $\overline{S3V}$ and $\overline{S3}$

Release Category	Probability	Release Time (hrs)	Release Fractions Xe	Cs	I	Te	Ba	Ru	La
$\overline{S3V}$-a	.02	28	1.0	1.5(-2)	1.5(-2)	1.9(-2)	1.6(-3)	1.5(-3)	2.5(-4)
$\overline{S3V}$-b	.08	36	9.0(-1)	5.3(-3)	5.3(-3)	6.6(-3)	5.7(-4)	5.1(-4)	8.6(-5)
$\overline{S3V}$-c	.30	54	8.0(-1)	1.6(-3)	1.6(-3)	2.0(-3)	1.7(-4)	1.5(-4)	2.5(-5)
$\overline{S3V}$-d	.60	89	7.0(-1)	5.0(-4)	5.0(-4)	6.3(-4)	5.5(-5)	4.8(-5)	8.2(-6)
$\overline{S3}$-a	.02	22	1.0	2.6(-2)	2.6(-2)	4.9(-3)	3.3(-3)	9.7(-4)	9.7(-5)
$\overline{S3}$-b	.08	28	9.0(-1)	8.5(-3)	8.5(-3)	1.6(-3)	1.1(-3)	3.1(-4)	3.1(-5)
$\overline{S3}$-c	.30	34	8.0(-1)	2.9(-3)	2.9(-3)	5.3(-4)	3.6(-4)	1.1(-4)	1.1(-5)
$\overline{S3}$-d	.60	53	7.0(-1)	8.5(-4)	8.5(-4)	1.6(-4)	1.1(-4)	3.1(-5)	3.1(-6)
TMLB'-ε	-	12	1.0	2.8(-3)	6.0(-4)	8.5(-2)	1.7(-2)	2.4(-5)	4.3(-4)
AB-ε	-	24	1.0	4.8(-5)	4.7(-5)	4.0(-5)	4.9(-5)	2.4(-7)	3.6(-5)

V-sequence. This break could be either in the hot-leg (b release subcategory) or the cold-leg (c release subcategory). This sequence is modeled as multi-puff release and each puff is treated separately. In this comparison only the sum of the release will be considered, since no adequate method of analyzing a multi-puff release is readily available. Table 4.5 shows a comparison between the totals of the various $\overline{S6V}$ releases and two V-sequence releases computed for Surry and published in BMI-2104. One of the V-sequences is "dry," implying no water in the path of the release and the other is "wet," implying that the release passes through 3 feet of water before entering the atmosphere. From this comparison it can be seen that all the releases, except Cs for the "dry" V-sequence, are bracketed by the $\overline{S6V}$ releases.

4.3 Recommended Source Terms

The severe accident source terms used in the Seabrook Probabilistic Safety Study reviewed in the previous sections, are aimed at the multi-puff consequence model present in the CRACIT computer code. In order to make these source terms useful to the NRC staff for evaluation with the CRAC code, total releases must be used as summarized in Table 4.6. Furthermore, the suggested source terms of Table 4.6 together with their release category characteristics given in Table 4.7 have been adjusted to more closely represent our assessment of the severe accidents based upon the RSS methodology.

It must also be noted that the suggested source term for the Steam Generator Tube Rupture (SGTR) sequence is assumed to be one-tenth of the source term for the event V ($\overline{S6V}$). This is believed to be a conservative estimate and can be used in the absence of a more specific mechanistic calculation.

The suggested source terms can be used to estimate the number of health and economic effects (consequences) in the population surrounding the Seabrook Station due to radioactive atmospheric releases as a result of core melt accidents.

The resulting consequences together with the frequency of radiological releases will enable the establishment of the severe accident risk at the Seabrook site considering the double-reactor unit effect.

Table 4.5 Comparison of $\overline{S6V}$ (sum) to V-sequence (Surry)

Release Category	Probability	Release Fractions Xe	Cs	I	Te	Ba	Ru	La
$\overline{S6V}$-a	.02	.97	4.3(-1)	4.3(-1)	4.06(-1)	4.2(-2)	3.32(-2)	5.3(-3)
$\overline{S6V}$-b	.45	.97	2.95(-1)	2.95(-1)	1.36(-1)	3.53(-2)	1.52(-2)	2.0(-3)
$\overline{S6V}$-c	.45	.97	1.295(-1)	1.295(-1)	3.2(-2)	1.593(-2)	5.2(-3)	5.3(-4)
$\overline{S6V}$-d	.08	.97	5.2(-2)	5.2(-2)	1.3(-2)	6.6(-3)	2.0(-3)	2.2(-4)
V (dry)	-	1.0	5.52(-1)	1.99(-1)	1.2(-1)	*	*	*
V (submerged)	-	1.0	1.04(-1)	3.84(-2)	2.5(-2)	*	*	*

*Individually not reported.

Table 4.6 BNL-Suggested Source Terms

Release Category	Xe	OI	I-2*	Cs	Te	Ba	Ru	La
S1	0.94	-	0.023	0.023	0.24	0.0033	0.41	9.8E-5
S2	0.89	-	2.1E-5	2.1E-5	4.4E-6	2.9E-6	8.8E-7	8.8E-8
S3	0.90	7E-3	1.E-7	1.E-7	1.9E-8	1.3E-8	3.8E-9	3.8E-10
S5	0.0091	-	3.5E-8	3.5E-8	6.1E-9	4.0E-9	1.2E-9	1.2E-10
S6	0.90	-	3.6E-3	3.6E-3	6.7E-4	4.4E-4	1.3E-4	1.3E-5
$\overline{S1}$	0.94	-	0.75	0.75	0.39	0.093	0.46	2.8E-3
$\overline{S2}$	0.90	-	0.31	0.31	0.057	0.038	0.011	1.1E-3
$\overline{S2V}$	1.0	-	0.31	0.31	0.32	0.034	0.025	4.2E-3
$\overline{S3}$	0.90	-	0.12	0.12	0.022	0.015	4.4E-3	4.4E-4
$\overline{S3V}$	1.0	-	0.024	0.024	0.030	2.6E-3	2.3E-3	3.9E-4
$\overline{S4V}$	1.0	-	0.058	0.058	0.072	6.2E-3	5.4E-3	9.1E-4
S5	0.014	7E-4	5.2E-7	5.2E-7	9.5E-8	6.3E-8	1.9E-8	1.9E-9
$\overline{S6V}$	0.97	-	0.43	0.43	0.40	0.048	0.033	5.3E-3
$\overline{S6V}$-d**	0.90	-	0.043	0.043	0.040	4.8E-3	3.3E-3	5.3E-4

*Elemental iodine not used, all iodine treated as CsI.
**$\overline{S6V}$-d release is 1/10th of the $\overline{S6V}$ values.

Table 4.7 BNL-Suggested Release Characteristics for Seabrook

Release Category	Release Time (hr)	Release Duration (hr)	Release Energy (Btu/hr)	Release Height (m)	Warning* Time (hr)
S1	1.9	0.5	140E6	10	0.35
S2	2.6	1.0	0.5E6	10	1.05
S3	66.0	0.5	250E6	10	63
S5	1.9	10	n/a	10	0.35
S6	4.5	4	0.5E6	10	0.50
$\overline{S1}$	1.4	0.5	520E6	10	0.30
$\overline{S2}$	27	10	10E6	10	26
$\overline{S2V}$	35	10	25E6	10	35
$\overline{S3}$	27	0.5	250E6	10	26
$\overline{S3V}$	81	0.5	450E6	10	76
$\overline{S4V}$	50	0.5	250E6	0	49
S5	4.3	24	10E6	10	0.30
$\overline{S6V}$	2.5	1.0	0.5E6	10	1.0
$\overline{S6V}$-d	2.5	1.0	0.5E6	10	1.0

*Warning time is defined as the time after core melt starts to the time of radiological release.

5. SUMMARY AND CONCLUSIONS

The purpose of this report is to describe the technical review of the Seabrook Station Probabilistic Safety Assessment and to present an assessment of containment performance, and radiological source term estimates for severe core melt accidents.

The containment response to severe accidents is judged to be an important factor in mitigating the severe accident risk. There is negligible probability of prompt containment failure or failure to isolate. Failure during the first few hours after core melt is also unlikely. Most core melt accidents would be effectively mitigated by containment spray operation.

Our assessment of the containment failure characteristics indicate that, there is indeed a tendency to fail containment through a realistic benign mode compared with the traditional gross failures.

The point-estimate release fractions used in the SSPSA are comparable in magnitude to those used in the RSS. In those cases where comparisons can be made to the more mechanistic source term study carried out by the Accident Source Term Program Office (ASTPO) and reported in BMI-2104 it was found that the SSPSA releases were either higher than or for the most part similar to the recent release fractions. It was also found that the energy of release was somewhat higher in the SSPSA than for other existing studies.

6. REFERENCES

1. "Zion Probabilistic Safety Study," Commonwealth Edison Company (September 1981).

2. "Limerick Probabilistic Safety Study," Philadelphia Electric Co. (September 1982).

3. "Indian Point Probabilistic Safety Study," Power Authority of the State of New York and Consolidated Edison Company (March 1982).

4. "Millstone Unit 3 Probabilistic Safety Study," Northeast Utilities (August 1983).

5. B. J. Garrick, et al., "Seabrook Station Probabilistic Safety Assessment," Pickard, Lowe and Garrick, Inc., PLG-0300 (December 1983).

6. A. A. Garcia, et al., "A Review of the Seabrook Station Probabilistic Safety Assessment," Lawrence Livermore National Laboratory Report (Dec. 12, 1984).

7. M. Khatib-Rahbar, et al., "Review and Evaluation of the Millstone Unit 3 Probabilistic Safety Study: Containment Failure Modes, Radiological Source Terms and Off-Site Consequences," NUREG/CR-4143 (report to be published).

8. R. O. Wooten and H. Avei, "MARCH: Meltdown Accident Response Characteristics - Code Description and User's Manual," BMI-2064, NUREG/CR-1711 (1980).

9. J. F. Muis, et al., "CORCON-Modl: An Improved Model for Molten Core/Concrete Interactions," SAND80-2415 (1981).

10. B. E. Miller, A. K. Agrawal, and R. E. Hall, "An Estimation of Pre-Existing Containment Leakage Areas and Purge and Vent Valve Leakage Areas Resulting from Severe Accident Conditions," A-3741, 11/15/84 (Draft report dated August 1984) transmitted via letter to V. Noonon, June 29, 1984.

11. W. Lyon (organizer), "RCS Pressure Boundary Heating During Severe Accidents," USNRC Meeting, Bethesda, Maryland (May 14, 1984).

12. "Estimates of Early Containment Loads From Core Melt Accidents," Containment Loads Working Group, NUREG-1079 (Draft 1985).

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

14. "Preliminary Assessment of Core Melt Accidents at the Zion and Indian Point Nuclear Power Plants and Strategies for Mitigating Their Effects," NUREG-0850, Vol. 1 (November 1981).

15. G. S. Kolb, et al., "Reactor Safety Study Methodology Application Program: Oconee #3 PWR Plant," NUREG/CR-1659/2 of 4.