NUCL 402 Engineering of Nuclear Power Systems

Lecture 35: Reactor Safety

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Reduce probability of accidents

-Escapes cape of radioactivity from plant

1Reactor System Design – operational disturbances

2Multiple barrier to escape

3Detailed analysis of various events to evaluate safety

4Safety systems for various reasons (human errors) (natural & unnatural)

NRC – Statutory Responsibility

Basic Design Philosophy – Defense in Depth

3 Levels of Safety

- Level 1 – Design of reactor and other component – high degree (quality standards) of reliability.

- Level 2 – Provide protection system in case of abnormalities

- Level 3 – Engineered safety features – for highly unlikely events – Design Basis Accidents

Inherent Reactor Stability – As Negative Temperature Coefficient – Fuel Dopple Coefficient (Expansion Of Coolant-Moderator)

Quality Assurance – Codes and Standard – provide adequate confidence in component structure of system.

ASME boiler and pressure vessel code

Redundancy And Diversity – Parallel Systems- Common Made Failure

Barriers To The Escape of Radioactivity
(LWR)

- First: Ceramic $U0_2$ Sintered Fuel Pin -
- Second – Zr Cladding On Fuel Pin
- Third-Primary Coolant Boundary
- Fourth - Containment Structure

Reactor Protection System

Reactor Our Power Capability – 112 To 120% -If More Than This – Trip – By Protection System.

Transient – Significant Deviation from Normal.
Severe Transient – Activate Protection System
TRIPs PWR - Control Rod Drop – Held By Electromagnetic Clutches
Borated Water Can Be Injected With CVCS

BWR - Central Rods Are Pushed Up By Hydrostatic Pressure
\- Recirculation Pumps Are Stopped
To Avoid False Trip – 3 or More Independent Detectors For Trip Signal Used.

Trip signals

1Rapid Increase In Neutron Flux -During In Startup

2High Neutron Flux - During Operation

3Abnormal P/T

4Loss Of Coolant Flow, e.g. Pump Failure

5High Steam Flow

6Closure Of Steam Isolation Valve (BWR)

7Turbine Generator Trip

8Loss Of Power Supplies To Instruments

9High Water Level In Pressurized (PWR)

1Low Water Level In Reactor Vessel (BWR)

1Low Water Level In Steam Generator (PWR)

1High Radioactivity In The Steam From a BWR

Shut Down Cooling

Sensible Heat, Delayed Fission Heat ½ minute

Decay Heat – Important 7% At Shutdown

1.3 % 1 Hr.

0.5% 1 Day

0.3% 1 Week

Steam Generated – Bypasses The Turbine - Goes To Condenser
Once the Temperature Low – Shutdown Cooling System
-Separate Heat Exchanger
-Cooled By Ultimate Heat Sink

If Steam Isolation Valves Are Closed –

In PWR – Large condensate tank – Auxiliary supply of fuel water to steam generator for 8 hour cooling

In BWR – Pressure relief valve provide release of steam to steam suppression pool.

If Electric Power is Lost:

Release steam from safety valve
Feed water to steam generator (PWR)
or reactor vessel (BWR)
by auxiliary system until diesel generator are started.

Reactor Safety Analysis

Abnormal Events: Transients

1Operational Faults (Human or Instrumental)

2Mechanical Failure

Safety Analysis Report (SAR) for permit to construct and license to operate

Abnormal

1 Events Of Moderate Frequency -from operational occurrences

Release of radioactive - Meeting criteria of normal operation

2 Events Of Low Probability – some release of small radioactivity

– arise due to most mechanical failure

3 Potentially Sever Accidents Of Low Probability – Design Basis Accidents – Serious Mechanical Failure.

1) Events Of Moderate Frequency

Imbalance between heat generation and cooling rates

Increase in Thermal Power:

1Coolant Temp Decrease – Increase In P & T

2) Control Material Removed

3) System Pressure Increase

Decrease in Cooling Effectiveness: Reduction in coolant flow rate or Decrease in CHF due to Pressure decrease.

2) Events of Low Probability – (Mechanical Failure)

1Small Break LOCA

2Power Failure – Less of Coolant Flow

Small Pipe Breaks – PWR – Inlet Feed Line – Serious

For Small Break – Makeup Available Systems Like CVCS

For Large Break – ECCS – If Fuel Damage - Containment Isolation.

BWR – If Small Break – Auxiliary Fuel Water System

For Large Break – Water From Steam Suppression Pool

Loss of Flow Accident

After Power Failure – 30 seconds- Power from Diesel

Meanwhile – Reactor Tripped – Loss of Flow Signal

Steam Dumped From Turbine

Some Power -Onsite -Due To Steam Dumping

Pump Coast down + natural Circulation ~ 30 sec.

3) Design Basis Accidents: (Most Sever)

– Double Ended Break of Large Pipe

-LOCA

Earthquakes, Tornadoes & Flooding

Seismic Activity – Ground Motion -Safe Shut Down

Earthquake - Maximum Horizontal Acceleration 0.3 of Gravity

in US Operating Basis

Control Element Ejection

Spent Fuel Handling Accident

During Refueling - Decay Heat -Core Criticality - Spent Fuel Transfer - Radioactivity

Loss Of Coolant Accident (LOCA)

Guillotine Break (Double -Ended) In Cold Leg of PWR or Recirculation From Intake of BWR

ECCS Sub System Provide Cooling
Extensive Studies Carried – To Predict Through Codes

PWR Events
1 Blow Down – Water Level Down
2 Refill – Emergency Coolant
3 Re-Flood – Water Level Up

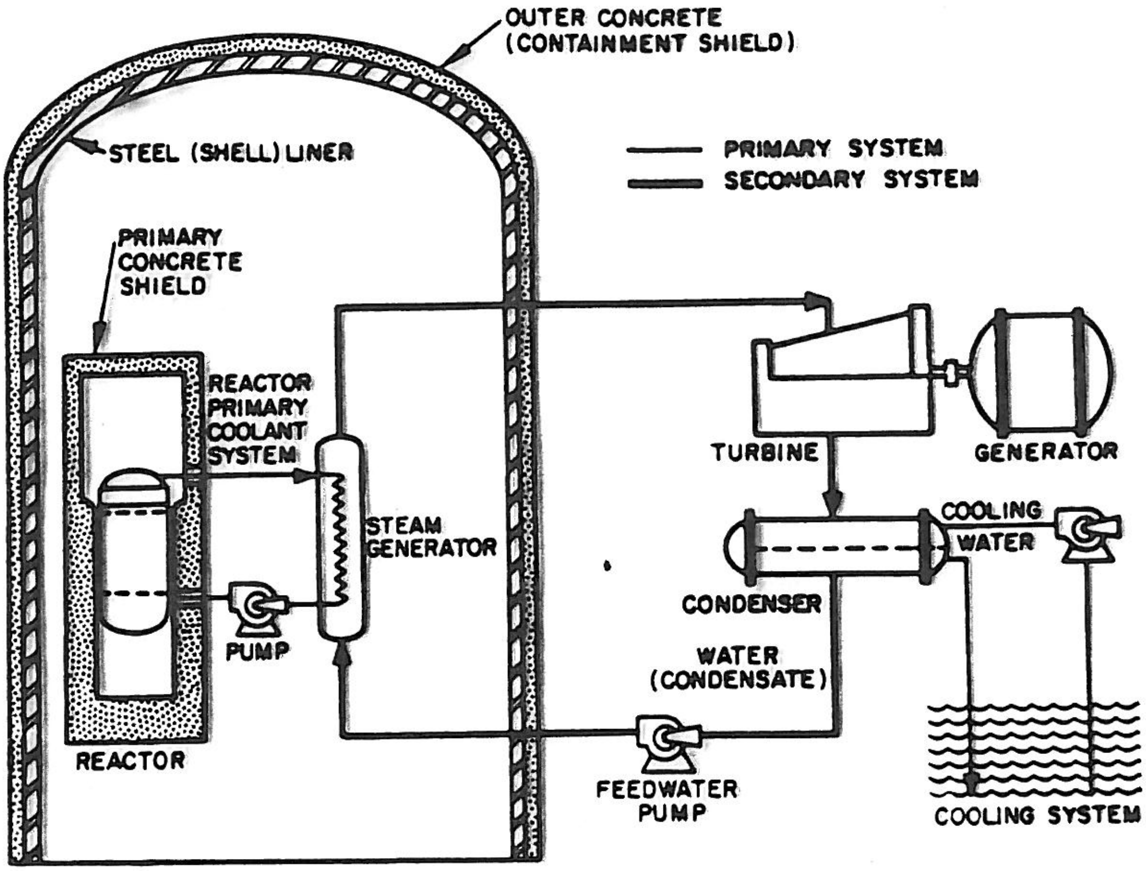


Figure 5-1. SCHEMATIC PRESSURIZED-WATER REACTOR POWER PLANT.
The primary reactor system is enclosed in a steel-lined concrete containment building. Steam generated within the building flows to the turbine-generator system (outside the building), after which it is condensed and returned to the steam generators. (Figure reproduced from ERDA-1541.)

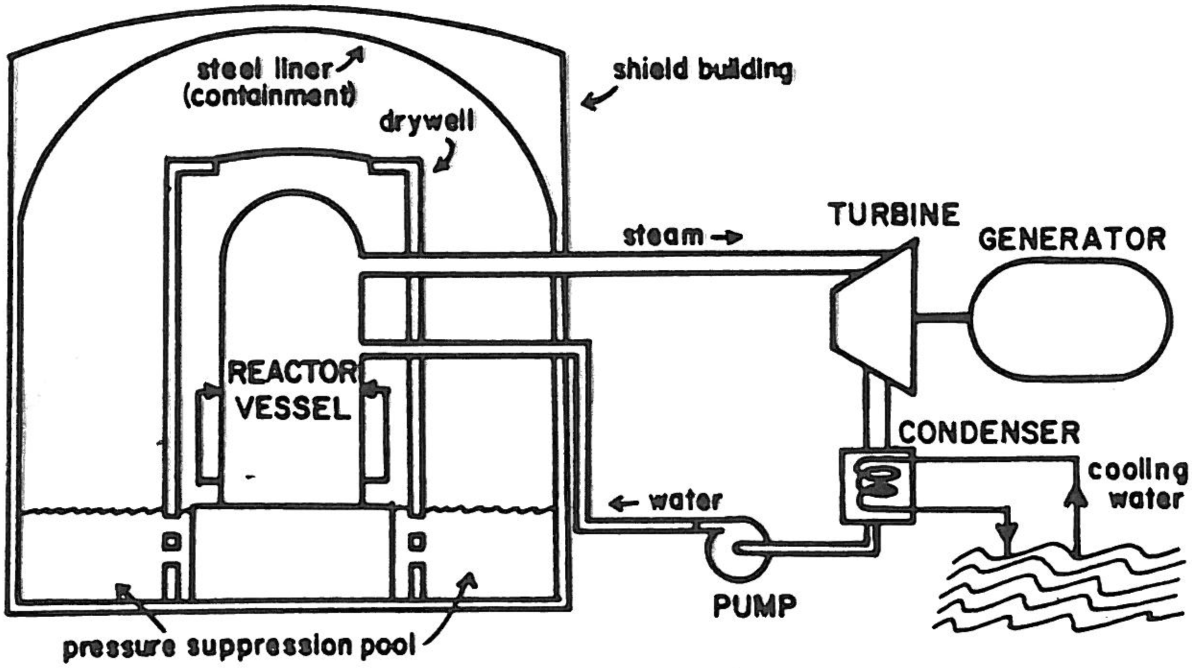


Figure 6-2. SCHEMATIC OF BOILING-WATER REACTOR POWER PLANT.
Steam from a BWR reactor vessel flows to the turbogenerator, after which it is condensed and returned as feedwater to the reactor vessel. The reactor vessel is contained in a dry well which, in turn, is within a reactor building.

(1) Blow Down – Subcooled –depressurization- Propagation of Wave-
Damage
Saturated (15-20 seconds) Void – Two-Phase
Choked Flow. Temperature of Core Increase after CHF
(Figure)

(2) Refill – ECCS

(3) Reflood – Complete Re-Flood Within 2 Minutes
Steam Binding – Flow Oscillations

BWR – For Design Basis LOCA.

Figures

Figures

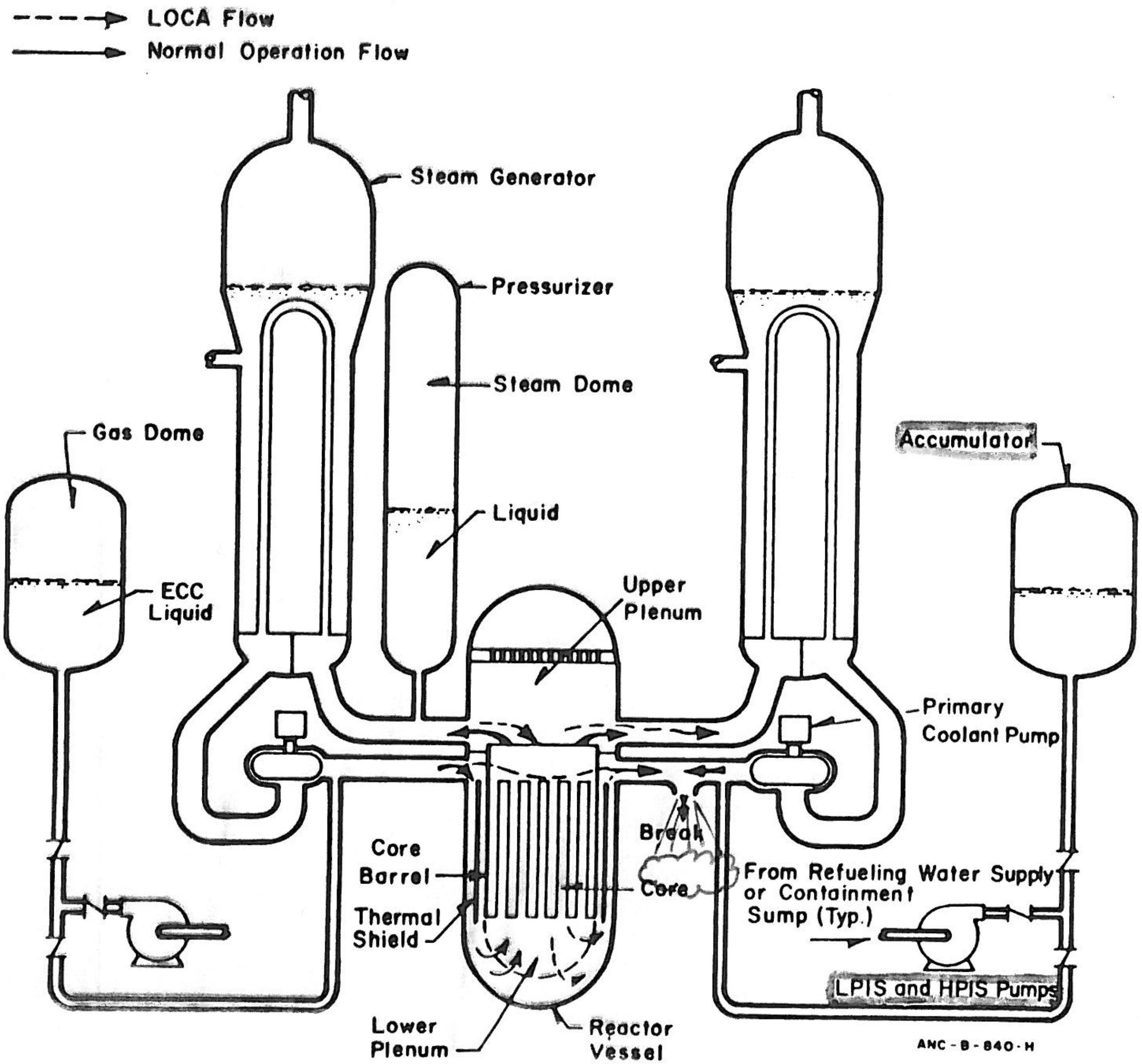
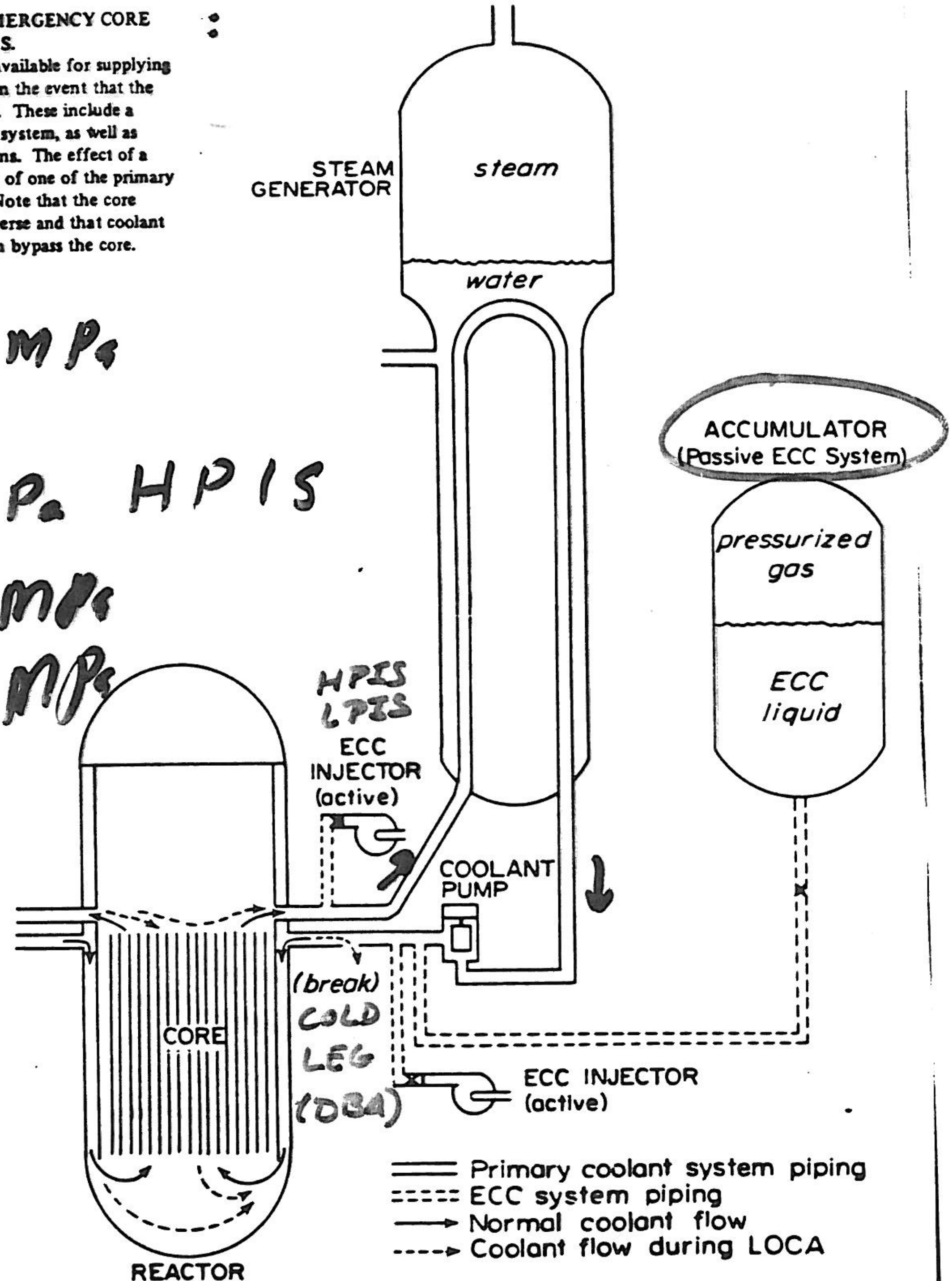

LOCA Flow
Normal Operation Flow
Steam Generator
Pressurizer
Steam Dome
Gas Dome
Accumulator
Liquid
ECC
Liquid
Upper
Plenum
Primary
Coolant Pump
Break
Core
Barrel
Core
From Refueling Water Supply
or Containment
Sump (Typ.)
Thermal
Shield
LPIS and HPIS Pumps
Lower
Plenum
Reactor
Vessel
ANC-B-840-H

Figure 5-8. PWR EMERGENCY CORE COOLING SYSTEMS.
Several systems are available for supplying coolant to the core in the event that the primary system fails. These include a passive accumulator system, as well as active injector systems. The effect of a break in the cold leg of one of the primary loops is indicated. Note that the core coolant flow can reverse and that coolant from other loops can bypass the core.

15.5 MPa
↓
11. MPa HPIS
4.1 MPa
1.4 MPa
↓
LPIS



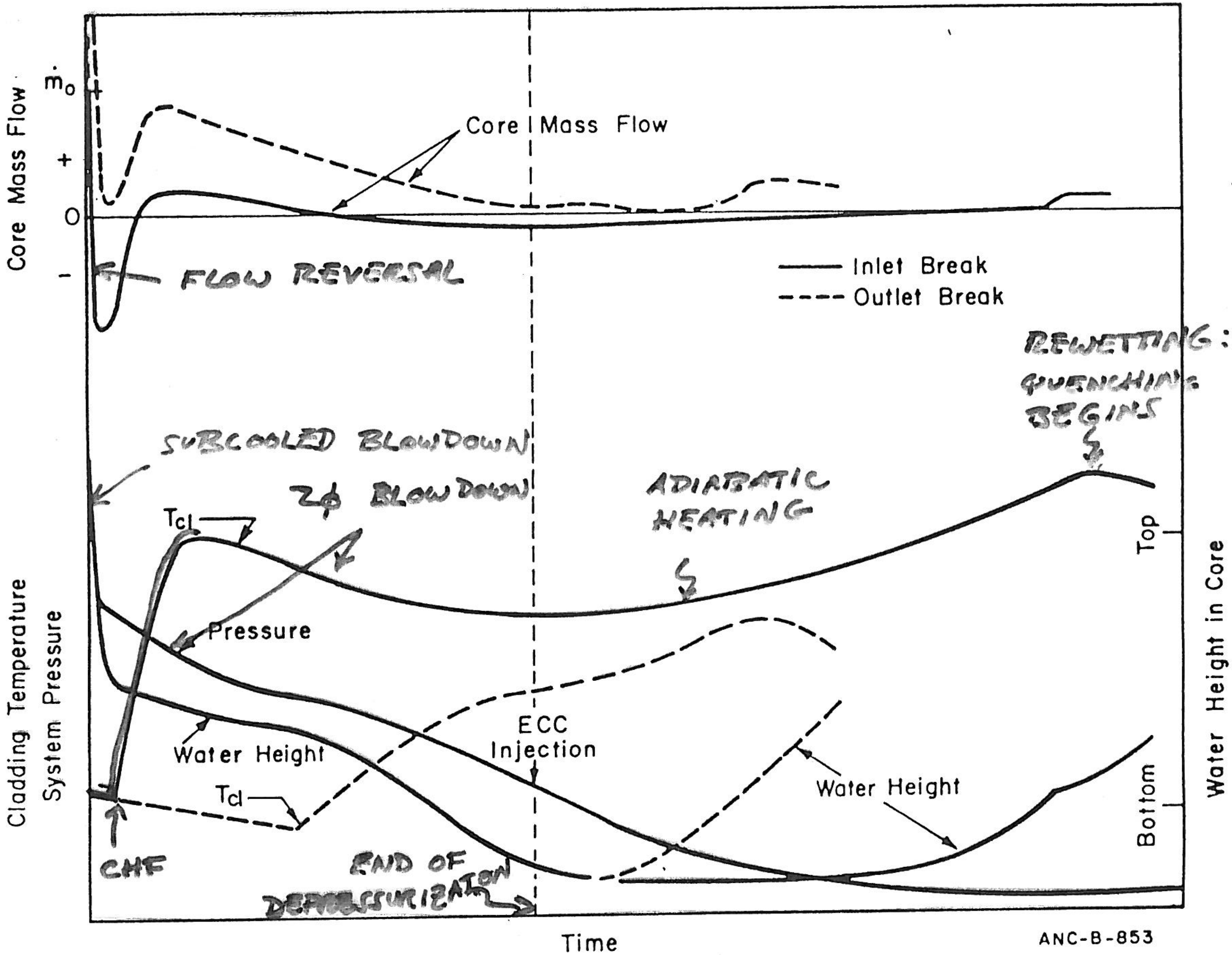


FIGURE 8-23 Generalized loss-of-coolant behavior for large pipe breaks in a PWR. Adapted from G. F. Brockett, R. W. Shumway, J. O. Zane and R. W. Griebe, "Loss of Coolant, Control of Consequences by Emergency Core Cooling," *Proc. 1972 Conf. on Nuclear Solutions to World Energy Problems,* Washington, D.C., November 1972. Used

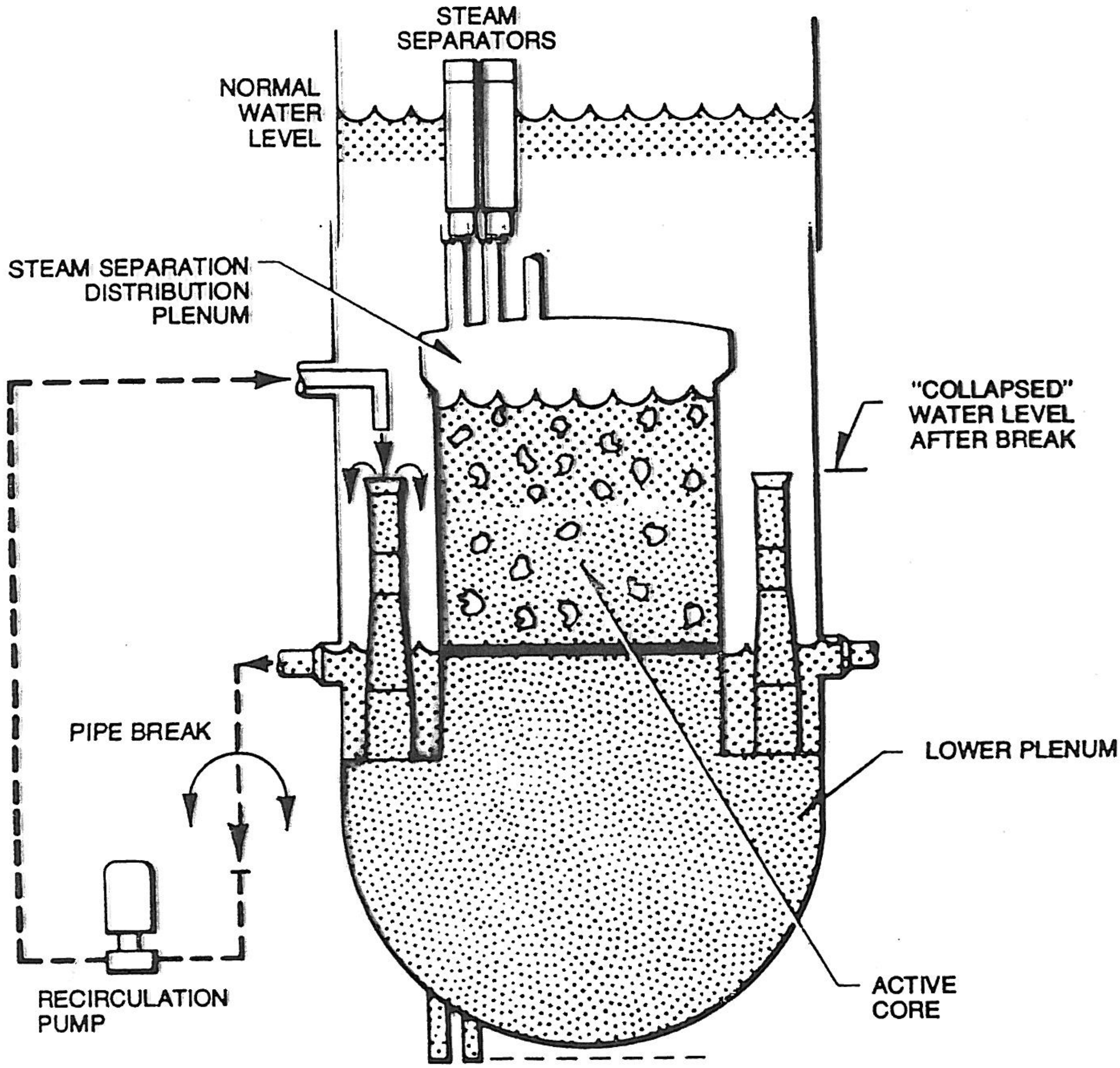
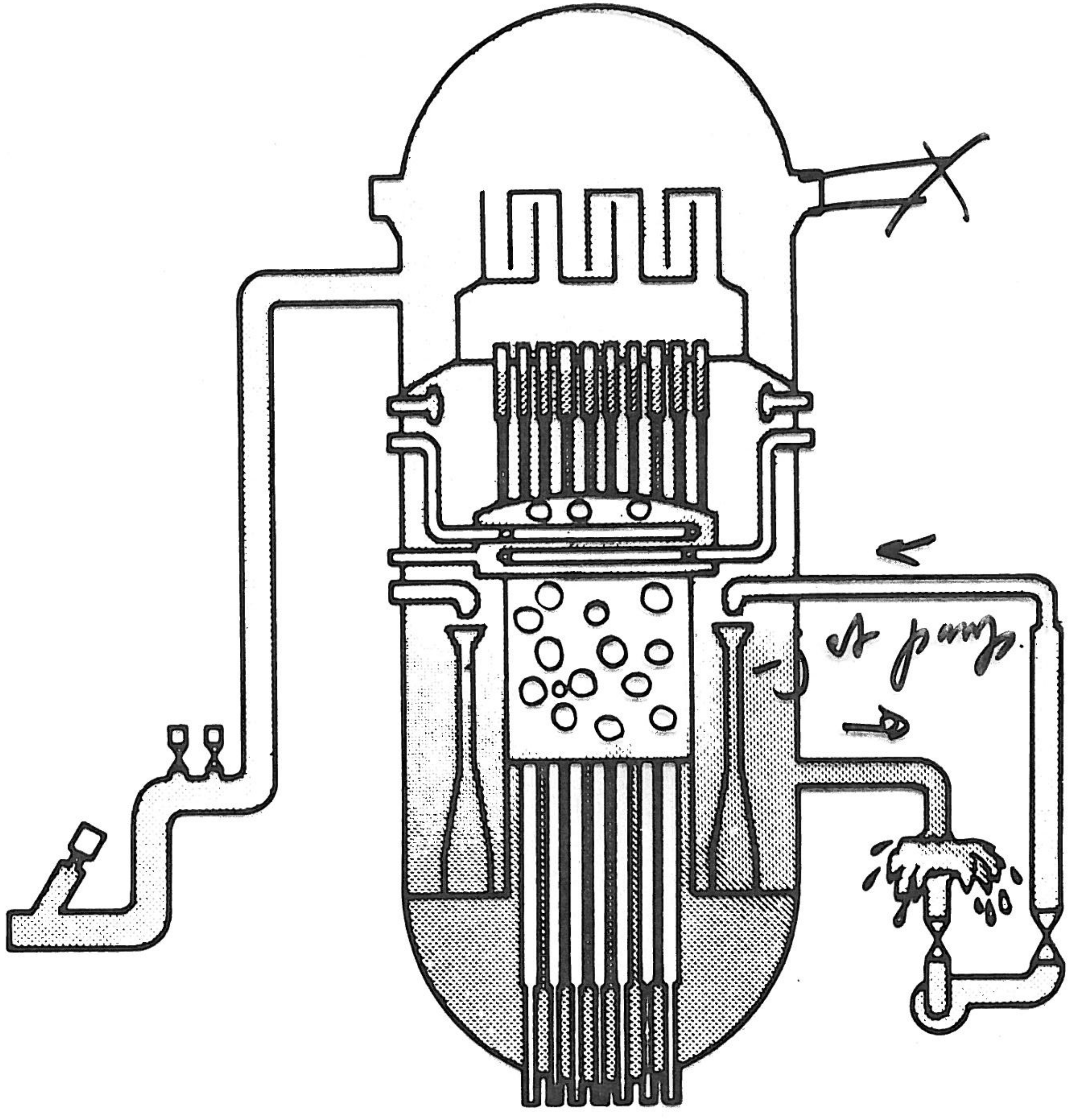


Fig. 2-7. Core submersion capability of jet pump system.


Jet pump

Blowdown.

Refill

Reflood

10-7MPa

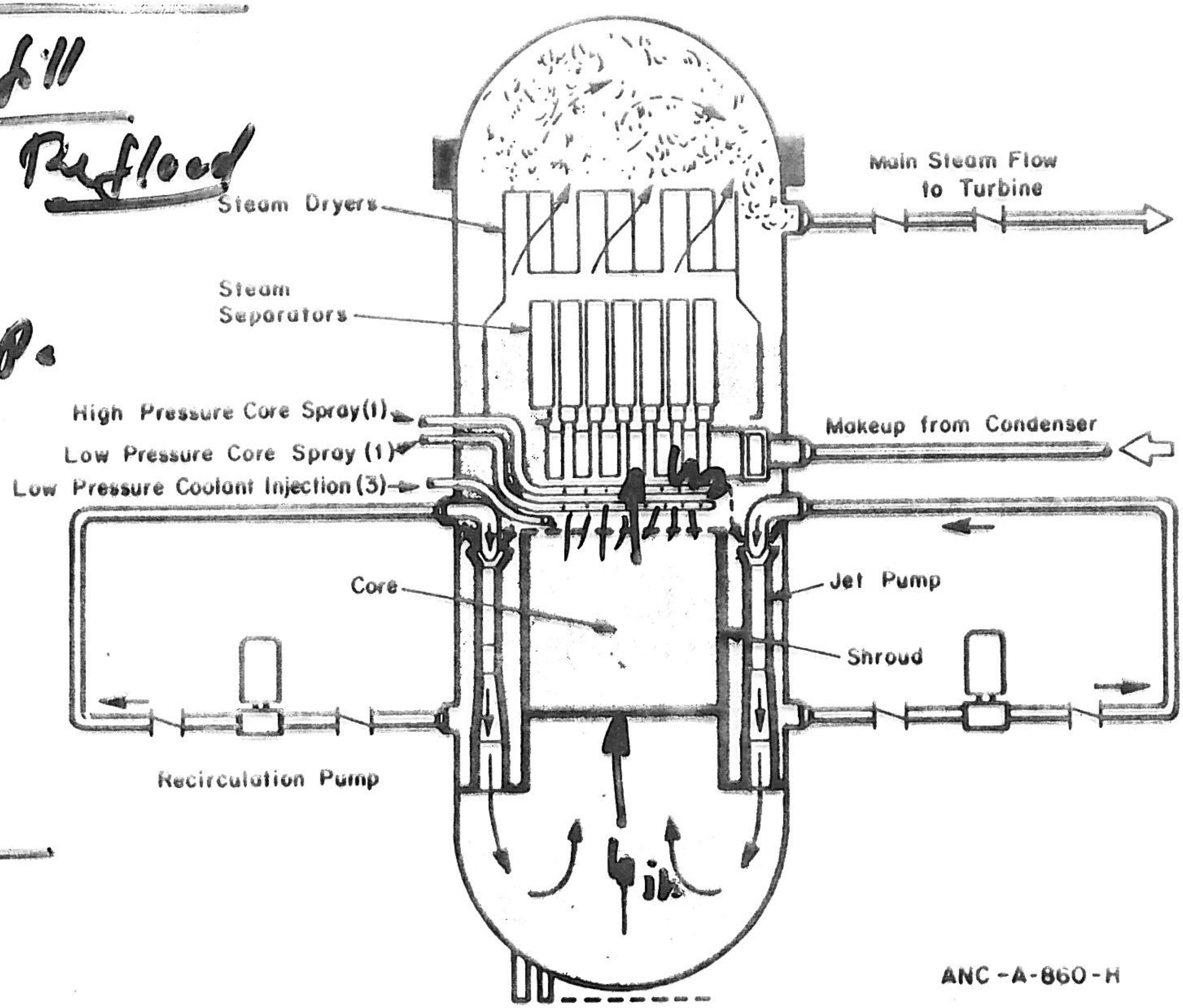
7.24MPa

HPCS

↓

LPCS

LPCI



ANC-A-860-H

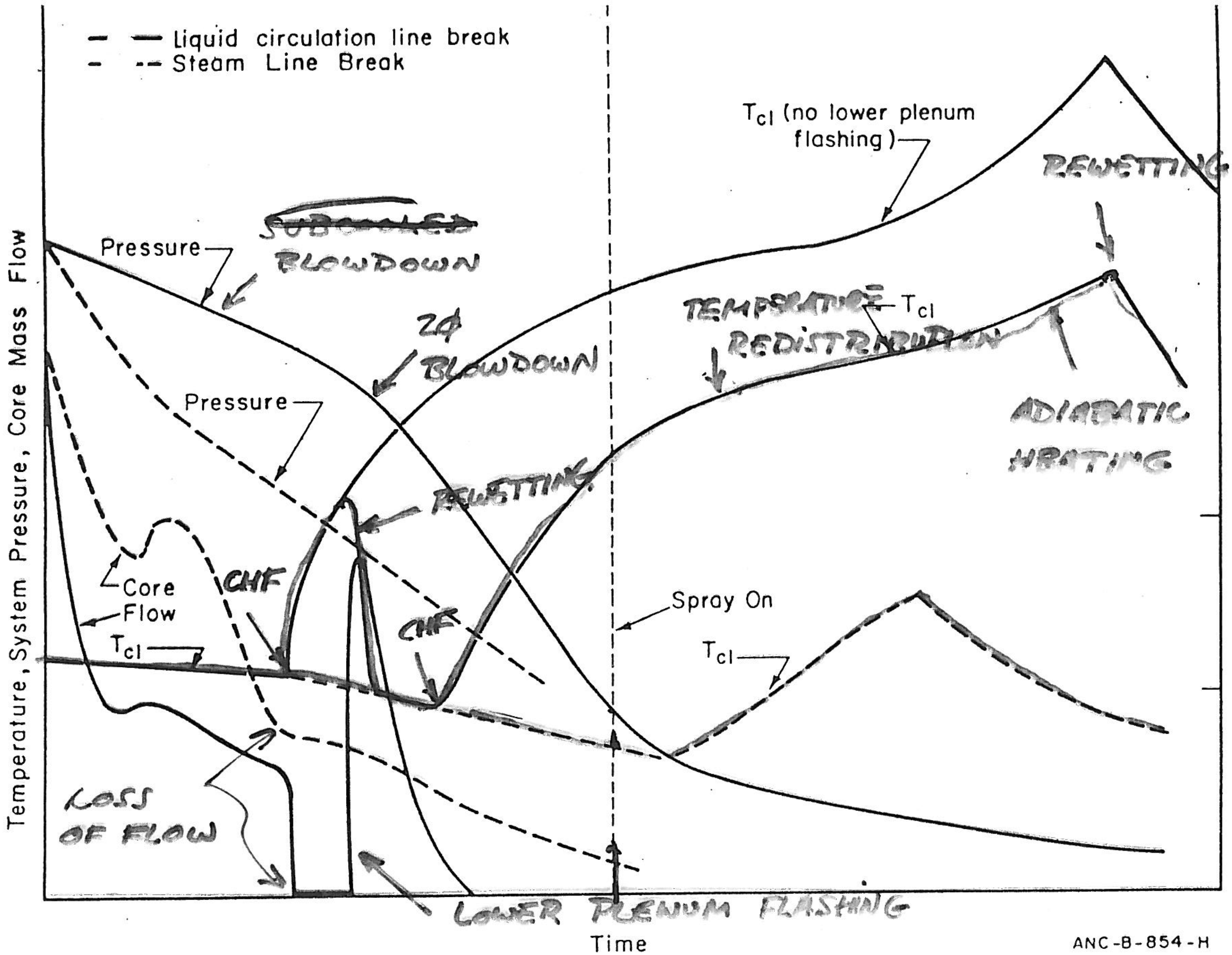


FIGURE 8-25 Generalized loss-of-coolant behavior for large pipe breaks in a BWR. Adapted from G. F. Brockett, R. W. Shumway, J. O. Zane and R. W. Griebe, "Loss of Coolant, Control of Consequences by Emergency Core Cooling," *Proc. of the 1972 Conference on Nuclear Solutions to World Energy Problems,* Washington, D.C.,

Emergency Core Cooling Criteria

NRC

1. $T_{Clad\ Max}$ < 1204 C (2200 F)
2. Zr Oxidation < 0.17 times the Cladding Thickness

   $Zr+2H_2O \rightarrow ZrO2+ 2H_2$ (Exothermic)
3. Generation of H2 <1% of hypothetical H2 generation if all cladding is oxidized.
4. Geometry Changes – hamper Core Cooling
5. ECCS – Provide Core Temp – At Accepted Low Level – Extended Period

Conservative Assumption –

1Stored Heat 102% of Peak Power, Low Thermal Conductivity of Fuel Pallet

2 Heat Transfer Coefficients

3 Decay Heat 20% Larger than ANS Standard Decay heat

41200 C at the Hottest Channel

(Thermal-Hydraulic Calculation

Evaluation Models

1Heat Source

2Hydraulic Parameter

3Heat Transfer Mechanism

Types of Codes:

System Codes –Entire Primary Sytem

RELAP – Fluid, Control (nodal) Volume

Mass Momentum and Energy

2 phase Thermal-Hydraulic Transients

2 Fluid, Two-Velocity, Two-Temp Model

RETRAN -1D

TRAC, TRACE -2Fluid Model

COBRA – Subchannel Analysis

Engineered Safety Features

When Reactor Protection System Fails To Prevent Or Limit The Escape Of Radioactivity To The Environment – Then Engineered Safety Features Should Take Care Of That.

These Are: Emergency Core Cooling System (ECCS)
Containment Vessel
Containment Cleanup System
Hydrogen Control

ECCS – Becomes Operative When Primary Pressure Drops Include – Subsystems – Act In Sequence.

PWR – (Small Break Or Failure To Close Pressure Relief Valve) Normal Operating Pressure 15.5 MP

→ 11.4 MPa High Pressure Injection System
Then from 1.4 to 4.1 Accumulator Injection System
Further Reduction In Pressure – Low-Pressure Injection System (LPIS)

Figure

Large Break – 20-25 Seconds,

BWR

Large Break – Design Basis Accident

Operating Range 10-7 MPa

Depressurization Rate Less Than PWR.

High Pressure Core-Spray (HPCS) System 7.24 MPa

Low Pressure Core-Spray (LPCS) For Rapid Depressurization

Low Pressure Injection System (LPIS) For Low Pressure - Large Amount Of Water.

Containment Systems

During LOCA if all Water all the Steam Generated and Radioactivity Must be contained – Design Pressure 50 psi

- ECCS Functions To Some Level

PWR: 37 m Dia X 61 m High – Cylindrical Structure

1.07 m Thick Concrete + 3.8cm Steel Liner

If all Coolant Flashes → Maximum Pressure 280 kPa (g)

Containment Design 310 kPa (g)

Tested at 350 kPa (g)

Containment Spray - Contain – Sodium Hydroxide

Alkaline Sodium Thiosulphate

-Reduce Radio-Iodide

Plus Particular Filter - Air Circulation

Use Of Borated Ice Sheet Between Steel Liner And Concrete

BWR – Primary -Secondary Structure→Drywell + Wet Well

Concrete + Steel Liner Design Pressure 350 kPa(g)

Leakage < 0.5 % Volume per day from Primary to Secondary

Recent -Mark III Drywell -Wet Well -Pressure Suppression -90% Steam condensed

Core-Spray + ECCS System Supply + Iodine Absorber In Wet Well

Hydrogen Concentration – Control

Sources:

1 Zr oxidation

2 NaOH + Al → H_2

3Radioanalytical Decomposition

Accumulation of 4% Volume of air can Ignite H_2.

As a general rule – takes longer time to accumulate substantial concentration so that circulation of air reduces the concentration or use

H_2 O_2 recombiners.