Operational Reactor Safety

22.091/22.903

Professor Andrew C. Kadak
Professor of the Practice

Safety Analysis Report and LOCA

Lecture 10

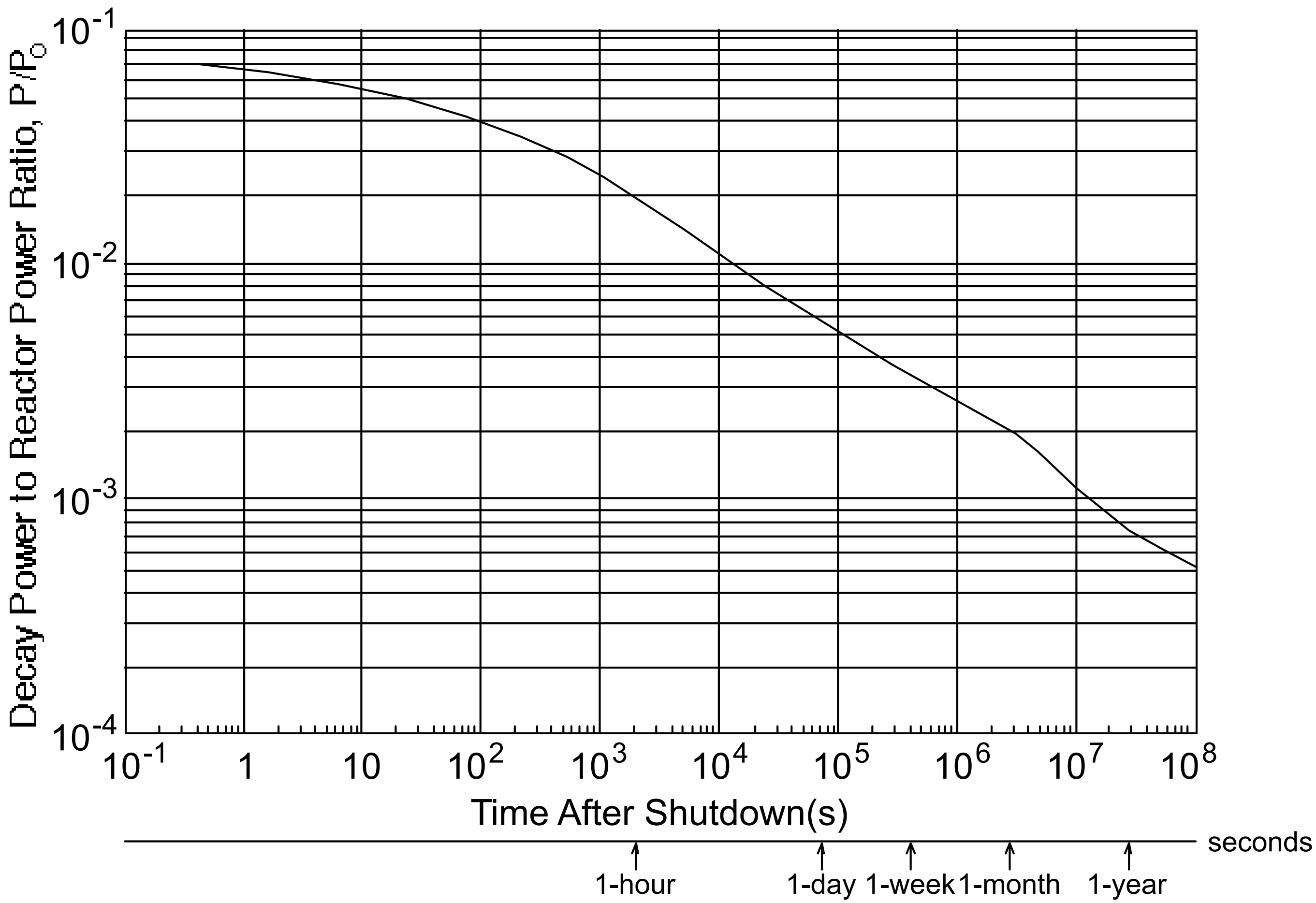
Topics to be Covered

- Safety Analysis Report
  - Contents
- Chapter 15
  - Transients and Accidents Analyzed
- Loss of Coolant Accident Example

The Hazard (some fission-product isotopes)

Isotope	Half-Life	Volatility	Health Hazard
^{131}I	8 d	Gaseous	External whole-body radiation; internal irradiation of thyroid; high toxicity
^{89}Sr	54 y	Moderately volatile	Bones and lungs
^{106}Ru	1 y	Highly volatile	Kidneys
^{137}Cs	33 y	Highly volatile	Internal hazard to whole body

Decay Heat



Source: Todreas & Kazimi, Vol. 1

CRITICAL SAFETY FUNCTIONS

KEEP FISSION PRODUCTS WITHIN THE FUEL

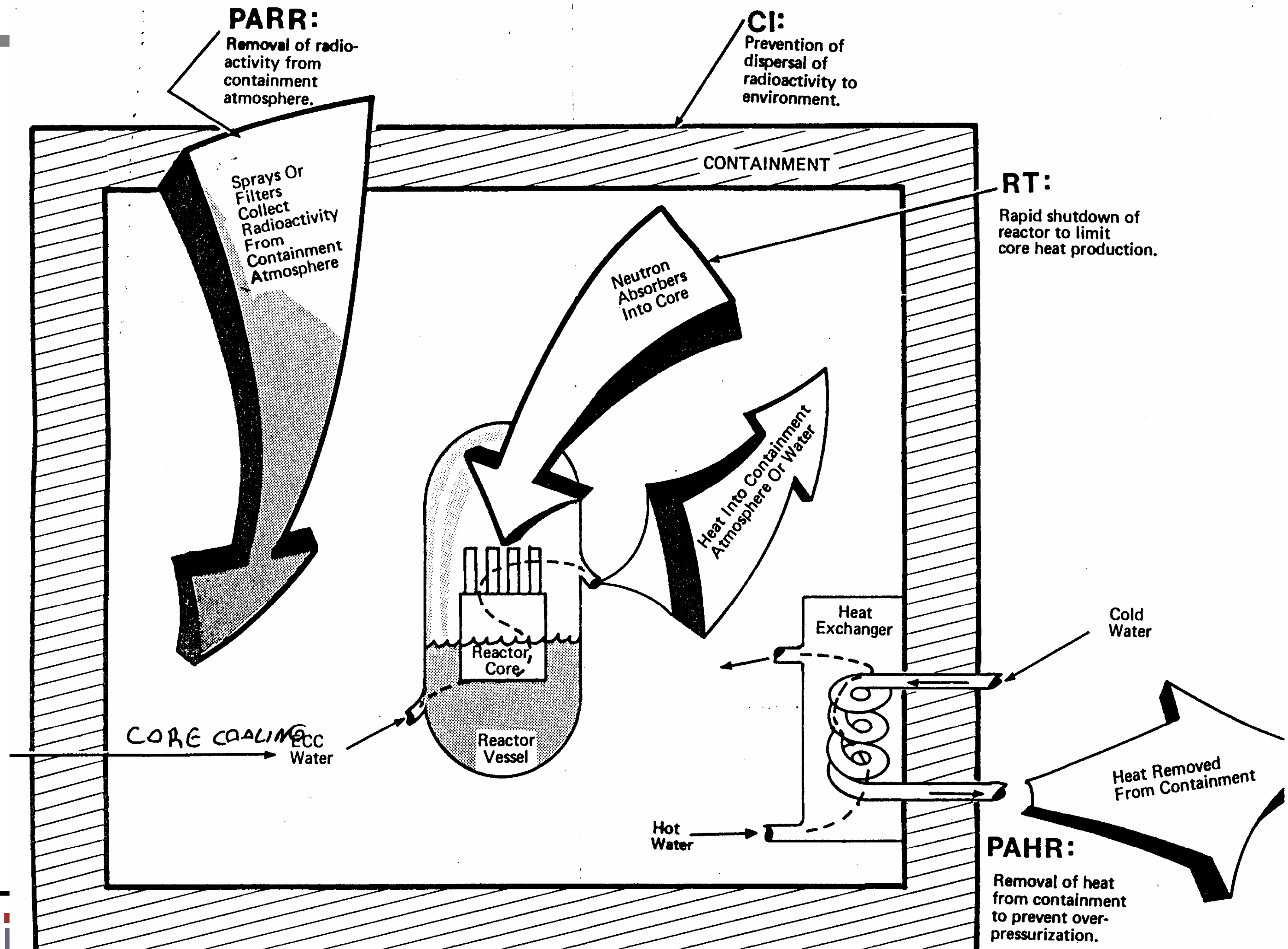
- Control Reactor Power
  - Control reactivity additions
  - Shutdown reliably
- Cool the Reactor and Spent Fuel
  - Maintain coolant inventory
  - Maintain coolant flow
  - Maintain coolant heat sinks

KEEP RADIOACTIVE MATERIAL OUT OF THE BIOSPHERE

- Maintain Containment Integrity
  - Prevent over-pressurization
  - Prevent over-heating
  - Prevent containment bypass
- Capture Material Within Containment
  - Scrubbing
  - Deposition
  - Chemical capture

SHIELD PERSONNEL FROM RADIATION

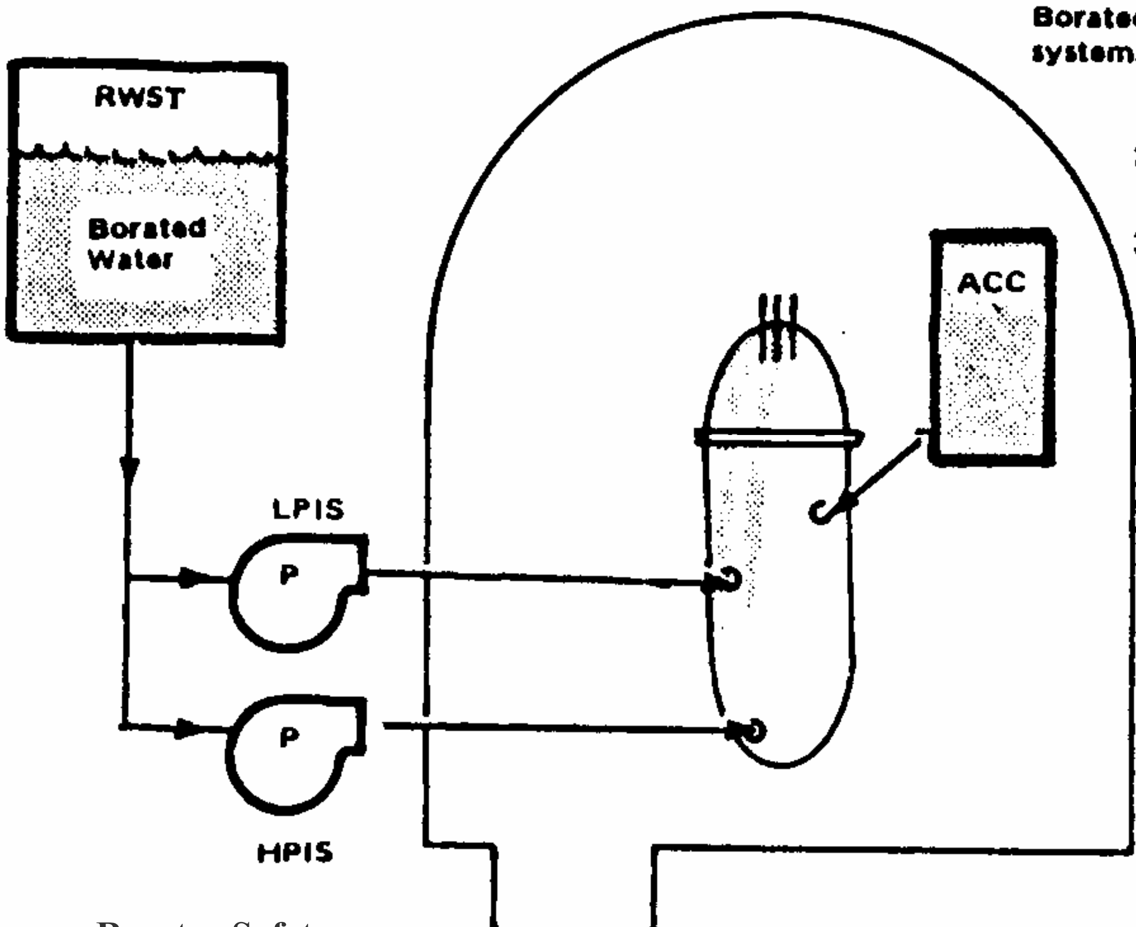
Emergency Safety Functions



Study, WASH-1400

PWR SYSTEMS USED TO PERFORM EMERGENCY FUNCTIONS

Emergency Coolant Injection



ECI

Borated water is furnished to cool the core by three systems:

1) Accumulators,
2) the Low Pressure Injection System (LPIS), and
3) the High Pressure Injection System (HPIS).

Reactor Safety Study, WASH-1400

PWR SYSTEMS USED TO PERFORM EMERGENCY FUNCTIONS

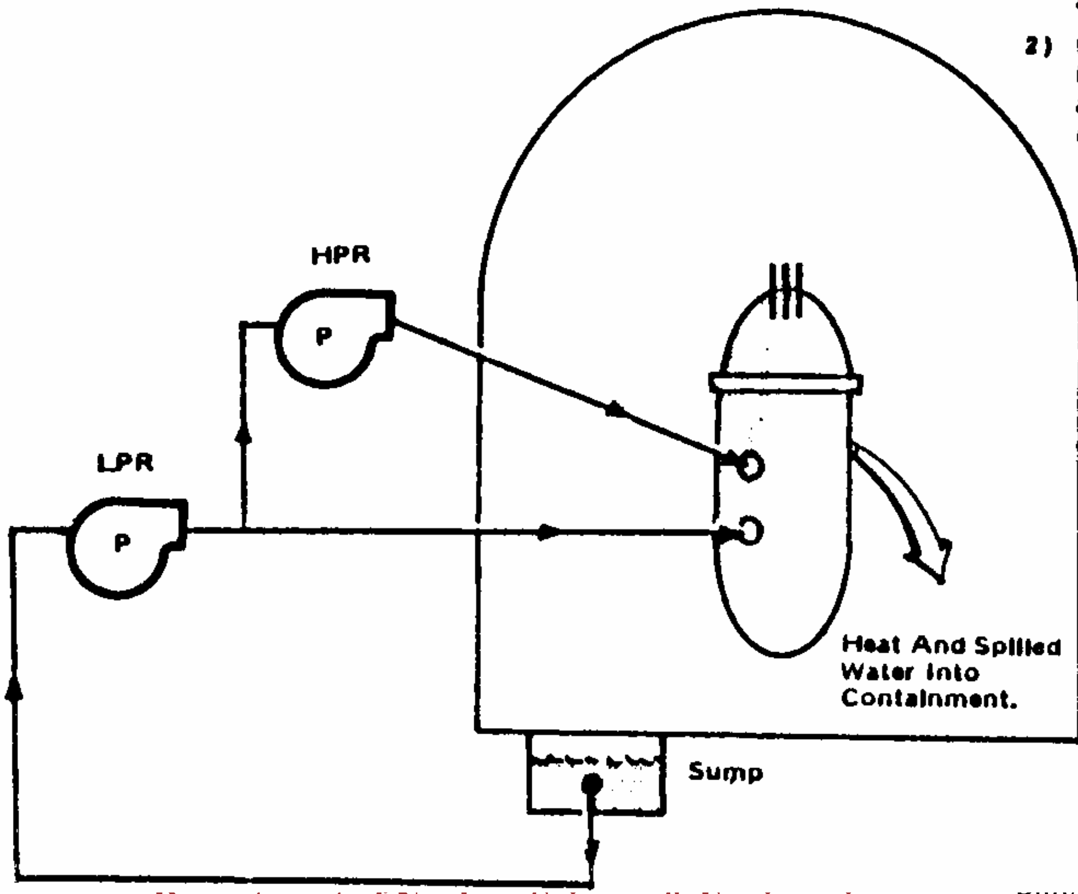
Emergency Coolant Recirculation

ECR

The core is cooled by heat being transferred to containment by two systems:

1) the Low Pressure Recirculation System (LPRS), and

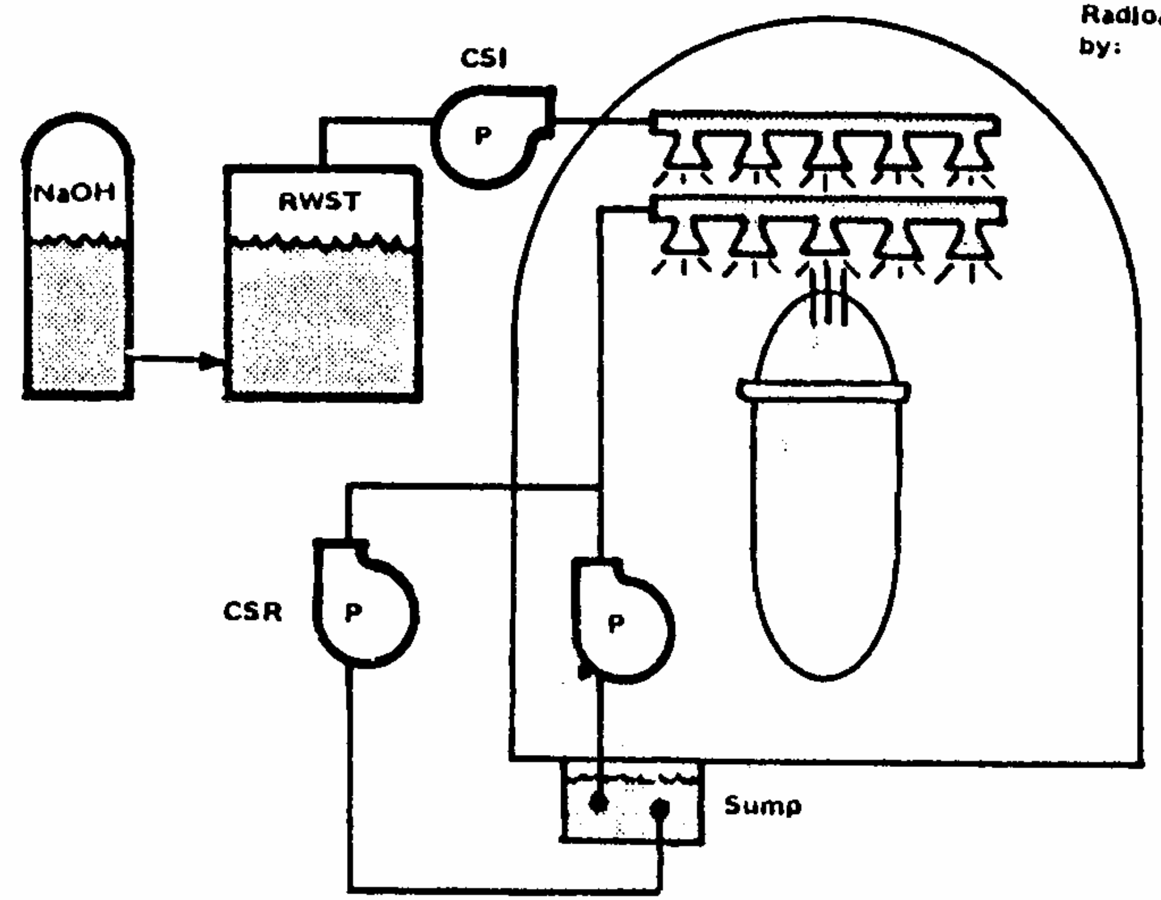
2) the High Pressure Recirculation System (HPRS). Both systems, using injection pumps aligned to a recirculation mode, pump water from a containment sump into the core.



Reactor Safety Study, WASH-1400

PWR SYSTEMS USED TO PERFORM EMERGENCY FUNCTIONS

Post Accident Radioactivity Removal



PARR

Radioactivity is collected from the containment atmosphere by:

1) the Containment Spray Injection System (CSIS),
2) the Containment Spray Recirculation System (CSRS), and
3) Sodium Hydroxide Addition (SHA) to spray water.

Reactor Safety Study, WASH-1400

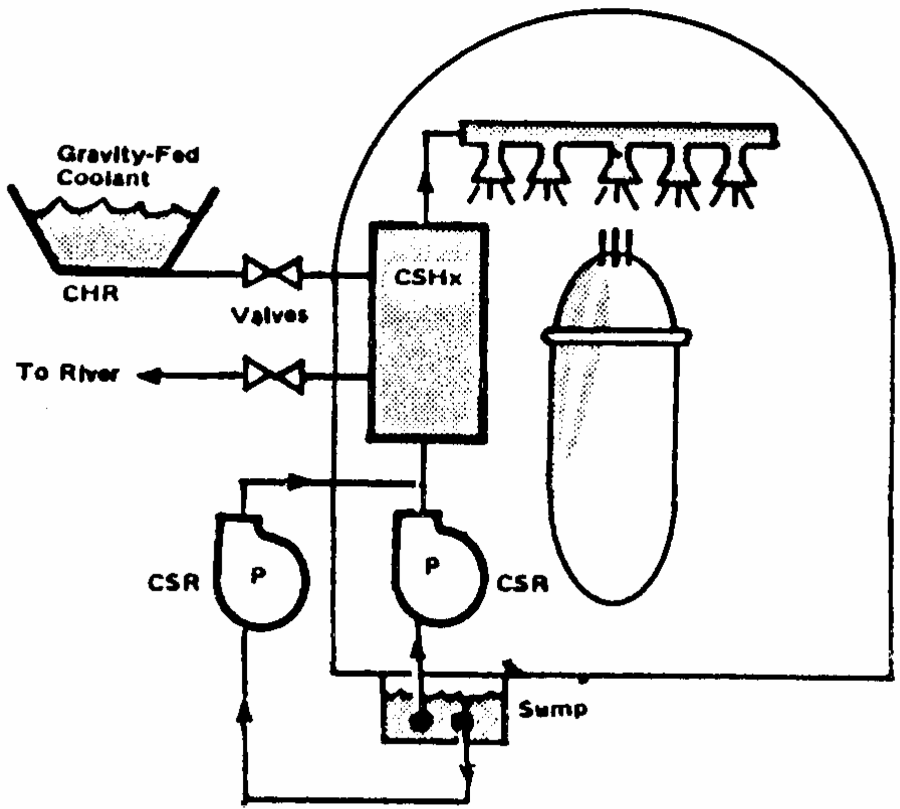
PWR SYSTEMS USED TO PERFORM EMERGENCY FUNCTIONS

Post Accident Heat Removal

PAHR

Heat is removed from containment by heat exchangers that involve two systems:

1) the Containment Spray Recirculation System, and
2) the Containment Heat Removal System (CHRS).



Reactor Safety Study, WASH-1400

PWR Engineered Safety Systems

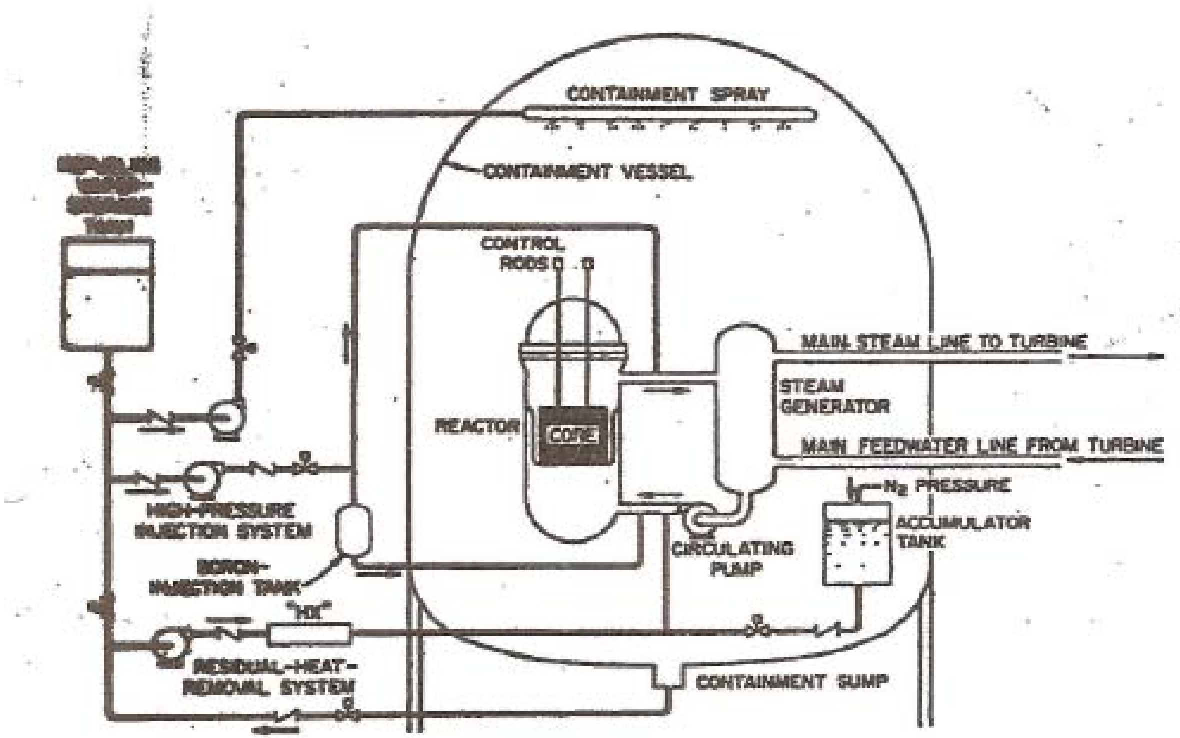


FIGURE 14-2
Engineered safety systems for a PWR. (From W. B. Cottrell, "The ECCS Rule-Making Hearing," *Nuclear Safety*, vol. 15, no. 1, Jan.–Feb. 1974.)

BWR Early Engineered Safety Systems

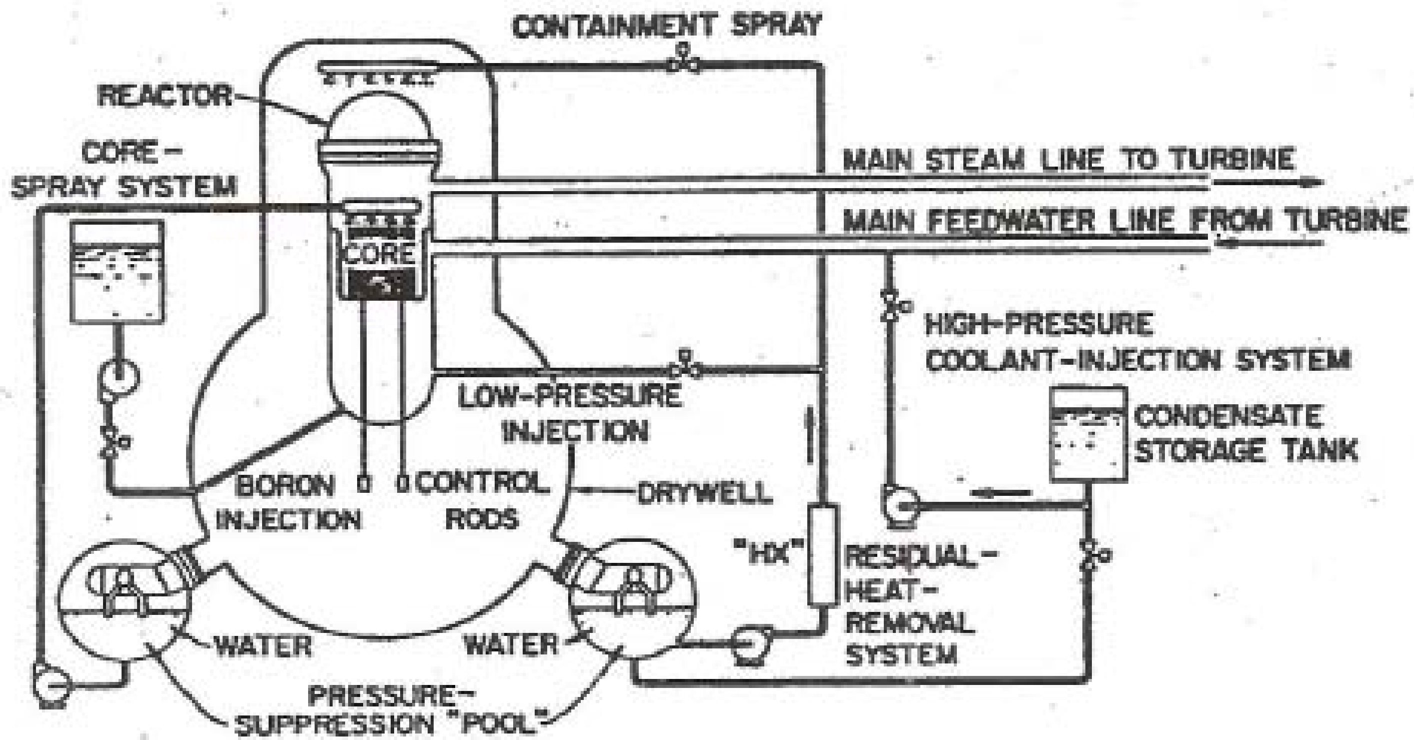


FIGURE 14-6
Engineered safety systems for an early BWR. (From W. B. Cottrell, "The ECCS Rule-Making Hearing," *Nuclear Safety*, vol. 15, no. 1, Jan.–Feb. 1974.)

Siting Criteria (10 CFR 100)

- Consideration of:
  - Characteristics of reactor design
  - Population characteristics, exclusion area, low population zone, population center distance
    - ✓ Assume a bounding fission product release based on a major accident
    - ✓ Define an exclusion area of such size that an individual located at any point on its boundary for two hours immediately following the accident would not receive a total radiation dose to the whole body in excess of 25 rem (250 mSv) or a total radiation dose in excess of 300 rem (3000 mSv) to the thyroid from iodine exposure.
    - ✓ Define a low population zone of such size that an individual located at any point on its outer boundary who is exposed to the radioactive cloud during the entire period of its passage would not receive a total radiation dose to the whole body in excess of 25 rem (250 mSv) or a total radiation dose in excess of 300 rem (3000 mSv) to the thyroid from iodine exposure.
    - ✓ A population center distance of at least 1.33 times the distance from the reactor to the outer boundary of the population center distance
  - Seismology, meteorology, geology, hydrology.

General Design Criteria (10 CFR 50 Appendix A)

http://www.nrc.gov/reading-rm/doc-collections/cfr/part050/

- The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is, structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.
- Six major categories:
  - Overall requirements
  - Protection by multiple fission product barriers
  - Protection and reactivity control systems
  - Fluid systems
  - Reactor containment
  - Fuel and reactivity control

The Single-Failure Criterion

- "Fluid and electric systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming passive components function properly) nor (2) a single failure of a passive component (assuming active components function properly), results in a loss of the capability of the system to perform its safety functions."
- The intent is to achieve high reliability (probability of success) without quantifying it.
- Looking for the worst possible single failure leads to better system understanding.

GDC 10 and 11

- *Criterion 10--Reactor design*. The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

- *Criterion 11--Reactor inherent protection*. The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

GDC 35

- An ECCS must be designed to withstand the following postulated LOCA: a double-ended break of the largest reactor coolant line, the concurrent loss of offsite power, and a single failure of an active ECCS component in the worst possible place.

Defense in Depth

"Defense-in-Depth is an element of the Nuclear Regulatory Commission's safety philosophy that employs successive compensatory measures to prevent accidents or mitigate damage if a malfunction, accident, or naturally caused event occurs at a nuclear facility."

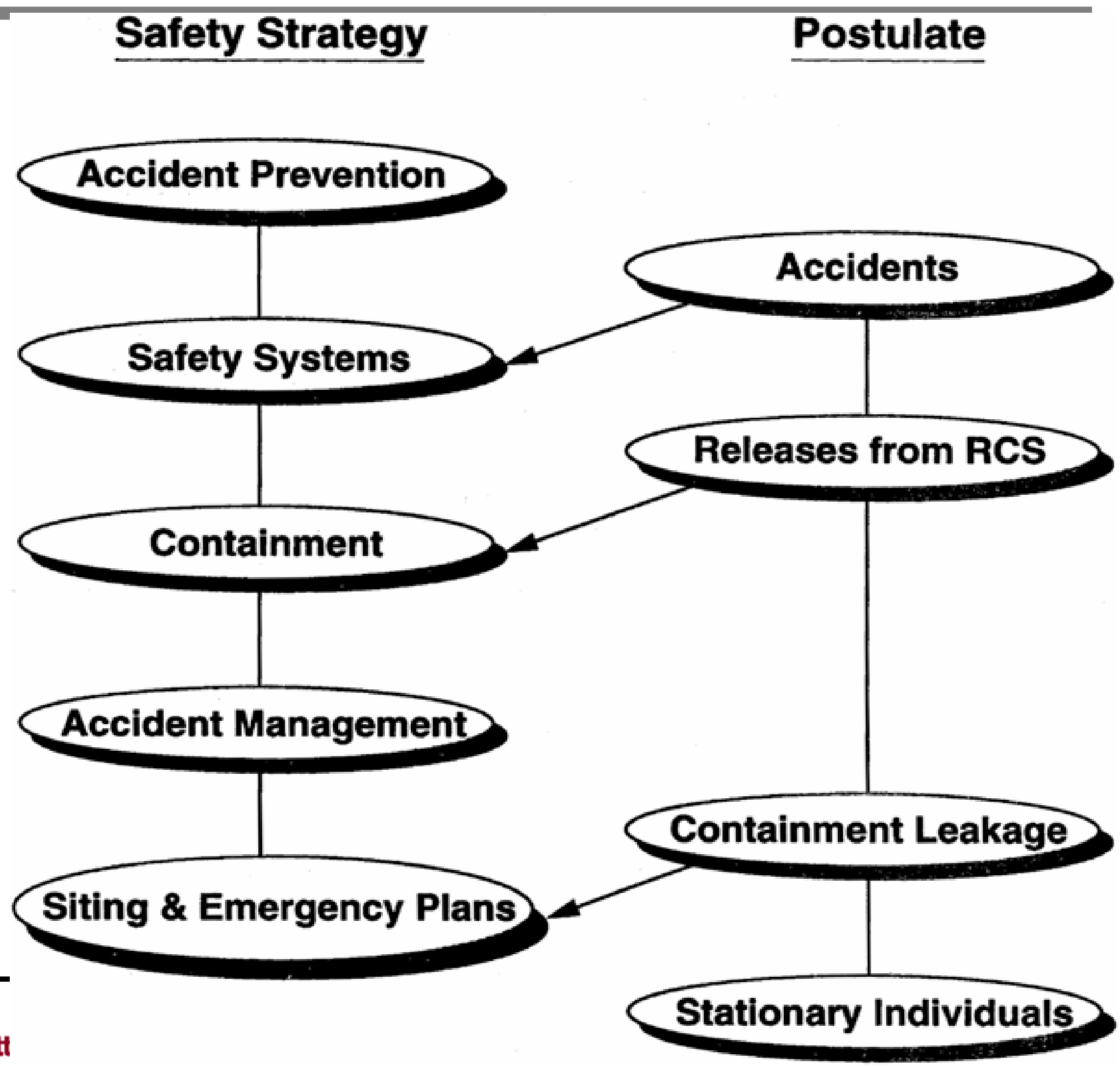
[Commission's White Paper, USNRC, 1999]

DEFENSE-IN-DEPTH MULTILAYER PROTECTION FROM FISSION PRODUCTS

Barrier or Layer	Function
1. Ceramic fuel pellets	Only a fraction of the gaseous and volatile fission products is released from the pellets.
2. Metal cladding	The cladding tubes contain the fission products released from the pellets. During the life of the fuel, less than 0.5 percent of the tubes may develop pinhole sized leaks through which some fission products escape.
3. Reactor vessel and piping	The 8- to 10-inch (20- to 25-cm) thick steel vessel and 3- to 4-inch (7.6- to 10.2-cm) thick steel piping contain the reactor cooling water. A portion of the circulating water is continuously passed through filters to keep the radioactivity low.
4. Containment	The nuclear steam supply system is enclosed in a containment building strong enough to withstand the rupture of any pipe in the reactor coolant system.
5. Exclusion area	A designated area around each plant separates the plant from the public. Entrance is restricted.
6. Low population zone, evacuation plan	Residents in the low population zone are protected by emergency evacuation plans.
7. Population center distance	Plants are located at a distance from population centers.

NUREG/CR-6042, 19

DEFENSE-IN-DEPTH, SAFETY STRATEGIES



NUREG/CR-6042, 1994

Safety Analysis Report Contents

Chapter 1	Introduction and General Description of Plant
Chapter 2	Site Characteristics
Chapter 3	Design of Structures, Components, Equipment, and Systems
Chapter 4	Reactor
Chapter 5	Reactor Coolant Systems and Connected Systems
Chapter 6	Engineered Safety Features
Chapter 7	Instrumentation and Controls
Chapter 8	Electric Power
Chapter 9	Auxiliary Systems
Chapter 10	Steam and Power Conversion System
Chapter 11	Radioactive Waste Management
Chapter 12	Radiation Protection
Chapter 13	Conduct of Operations
Chapter 14	Initial Test Program
Chapter 15	Accident Analysis
Chapter 16	Technical Specifications
Chapter 17	Quality Assurance

NUREG/CR-6042, 1994.

Design Basis Accidents

- A DBA is a postulated accident that a facility is designed and built to withstand without exceeding the offsite exposure guidelines of the NRC's siting regulation (10 CFR Part 100).

- Each DBA includes at least one significant failure of a component. In general, failures beyond those consistent with the single-failure criterion are not required (unlike in PRAs).

NUREG/CR-6042, USNRC, 1994.

POSTULATED ACCIDENTS AND OCCURRENCES

Class Number	Description	Example(s)
1	Trivial incidents	Small spills Small leaks inside containment
2	Miscellaneous small releases outside containment	Spills Leaks and pipe breaks
3	Radwaste system failures	Equipment failure Serious malfunction or human error
4	Events that release radioactivity into the primary system	Fuel defects during normal operation Transients outside expected range of variables
5	Events that release radioactivity into the secondary system	Class 4 and heat exchanger leak
6	Refueling accidents inside containment	Drop fuel element Drop heavy object onto fuel Mechanical malfunction or loss of cooling in transfer tube
7	Accidents to spent fuel outside containment	Drop fuel element Drop heavy object onto fuel Drop shielding cask—loss of cooling to cask, transportation incident on site
8	Accident initiation events considered in design basis evaluation in the safety analysis report	Reactivity transient Rupture of primary piping Flow decrease—steamline break
9	Hypothetical sequences of failures more severe than Class 8	Successive failures of multiple barriers normally provided and maintained

U.S. Atomic Energy Commission, 1973.

REPRESENTATIVE INITIATING EVENTS

TO BE ANALYZED IN SECTION 15.X.X OF THE SAR

1. **Increase in Heat Removal by the Secondary System**
   - 1.1 Feedwater system malfunctions that results in a decrease in feedwater temperature.
   - 1.2 Feedwater system malfunctions that result in an increase in feedwater flow.
   - 1.3 Steam pressure regulator malfunction or failure that results in increasing steam flow.
   - 1.4 Inadvertent opening of a steam generator relief or safety valve.
   - 1.5 Spectrum of steam system piping failures inside and outside of containment in a PWR.
2. **Decrease in Heat Removal by the Secondary System**
   - 2.1 Steam pressures regulator malfunction or failure that results in decreasing steam flow.
   - 2.2 Loss of external electric load.
   - 2.3 Turbine trip (stop valve closure).
   - 2.4 Inadvertent closure of main steam isolation valves.
   - 2.5 Loss of condenser vacuum.
   - 2.6 Coincident loss of onsite and external (offsite) a.c. power to the station.
   - 2.7 Loss of normal feedwater flow.
   - 2.8 Feedwater piping break.
3. **Decrease in Reactor Coolant System Flow Rate**
   - 3.1 Single and multiple reactor coolant pump trips.
   - 3.2 BWR recirculation loop controller malfunctions that result in decreasing flow rate.
   - 3.3 Reactor coolant pump shaft seizure.
   - 3.4 Reactor coolant pump shaft break.

NUREG/CR-6042, USNRC, 1994.

REPRESENTATIVE INITIATING EVENTS

TO BE ANALYZED IN SECTION 15.X.X OF THE SAR (cont.)

4. **Reactivity and Power Distribution Anomalies**
   - 4.1 Uncontrolled control rod assembly withdraws from a subcritical or low power startup condition (assuming the most unfavorable reactivity conditions of the core and reactor coolant system), including control rod or temporary control device removal error during refueling.
   - 4.2 Uncontrolled control rod assembly withdraws at the particular power level (assuming the most unfavorable reactivity conditions of the core and reactor coolant system) that yields the most severe results (low power to full power).
   - 4.3 Control rod maloperation (system malfunction or operator error), including maloperation of part length control rods.
   - 4.4 A malfunction or failure of the flow controller in BWR loop that results in an incorrect temperature.
   - 4.5 A malfunction or failure of the flow controller in BWR loop that results in an increased reactor coolant flow rate.
   - 4.6 Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant of a PWR.
   - 4.7 Inadvertent loading and operation of a fuel assembly in an improper position.
   - 4.8 Spectrum of rod ejection accidents in a PWR.
   - 4.9 Spectrum of rod drop accidents in a BWR.
5. **Increase in Reactor Coolant Inventory**
   - 5.1 Inadvertent operation of ECCS during power operations.
   - 5.2 Chemical and volume control system malfunction (or operator error) that increases reactor coolant inventory
   - 5.3 A number of BWR transients, including items 2.1 through 2.6 and item 1.2.

NUREG/CR-6042, USNRC, 1994.

REPRESENTATIVE INITIATING EVENTS

TO BE ANALYZED IN SECTION 15.X.X OF THE SAR (cont.)

6. Decrease in Reactor Coolant Inventory

6.1 Inadvertent opening of a pressurizer safety or relief valve in a PWR or a safety or relief valve in a BWR.

6.2 Break in instrument line or other lines from reactor coolant pressure boundary that penetrate containment.

6.3 Steam generator tube failure.

6.4 Spectrum of BWR steam system piping failures outside of containment.

6.5 Loss-of-coolant accidents resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary, including steam line breaks inside of containment in a BWR.

6.6 A number of BWR transients, including items 2.7, 2.8, and 1.3.

7. Radioactive Release from a Subsystem or Component

7.1 Radioactive gas waste system leak or failure.

7.2 Radioactive liquid waste system leak or failure.

7.3 Postulated radioactive releases due to liquid tank failures.

7.4 Design basis fuel handling accidents in the containment and spent fuel storage buildings.

7.5 Spent fuel cask drop accidents.

NUREG/CR-6042, USNRC, 1994.

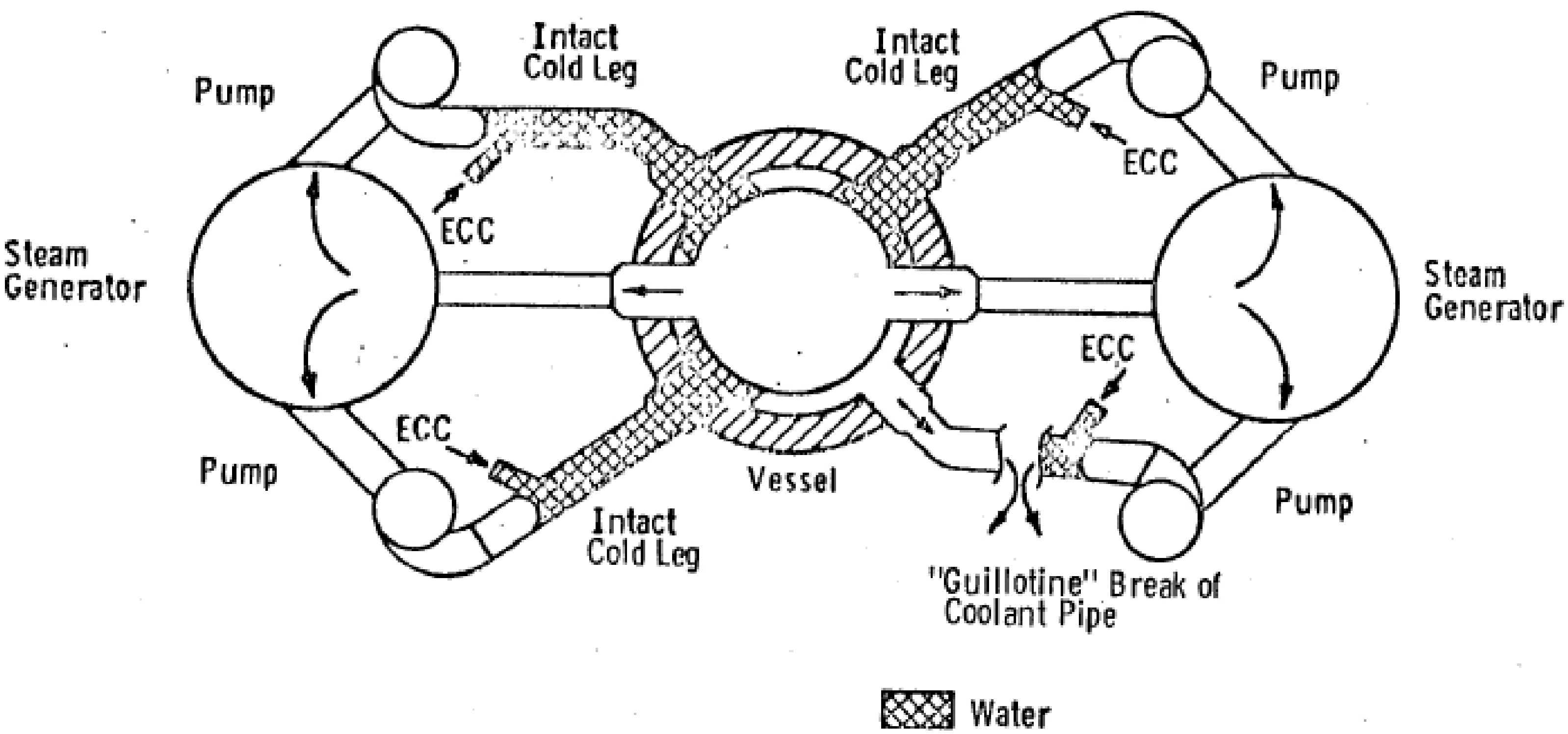
Emergency Core Cooling System (ECCS) (January 1974, 10 CFR 50.46)

- Postulate several LOCAs of different sizes and locations to provide assurance that the most severe LOCAs are considered.
- Postulate concurrent loss of offsite or onsite power and the most damaging single failure of ECCS equipment (GDC 35).
- Acceptance Criteria
  - Peak cladding temperature cannot exceed 2200 ºF (1204 ºC)
  - Oxidation cannot exceed 17% of cladding thickness
  - Hydrogen generation from hot cladding-steam interaction cannot exceed 1% of its potential
  - Core geometry must be coolable
  - Long-term cooling must be provided

Double Ended Guillotine Break

Figure 2

PLAN VIEW OF REACTOR AND PRIMARY LOOPS



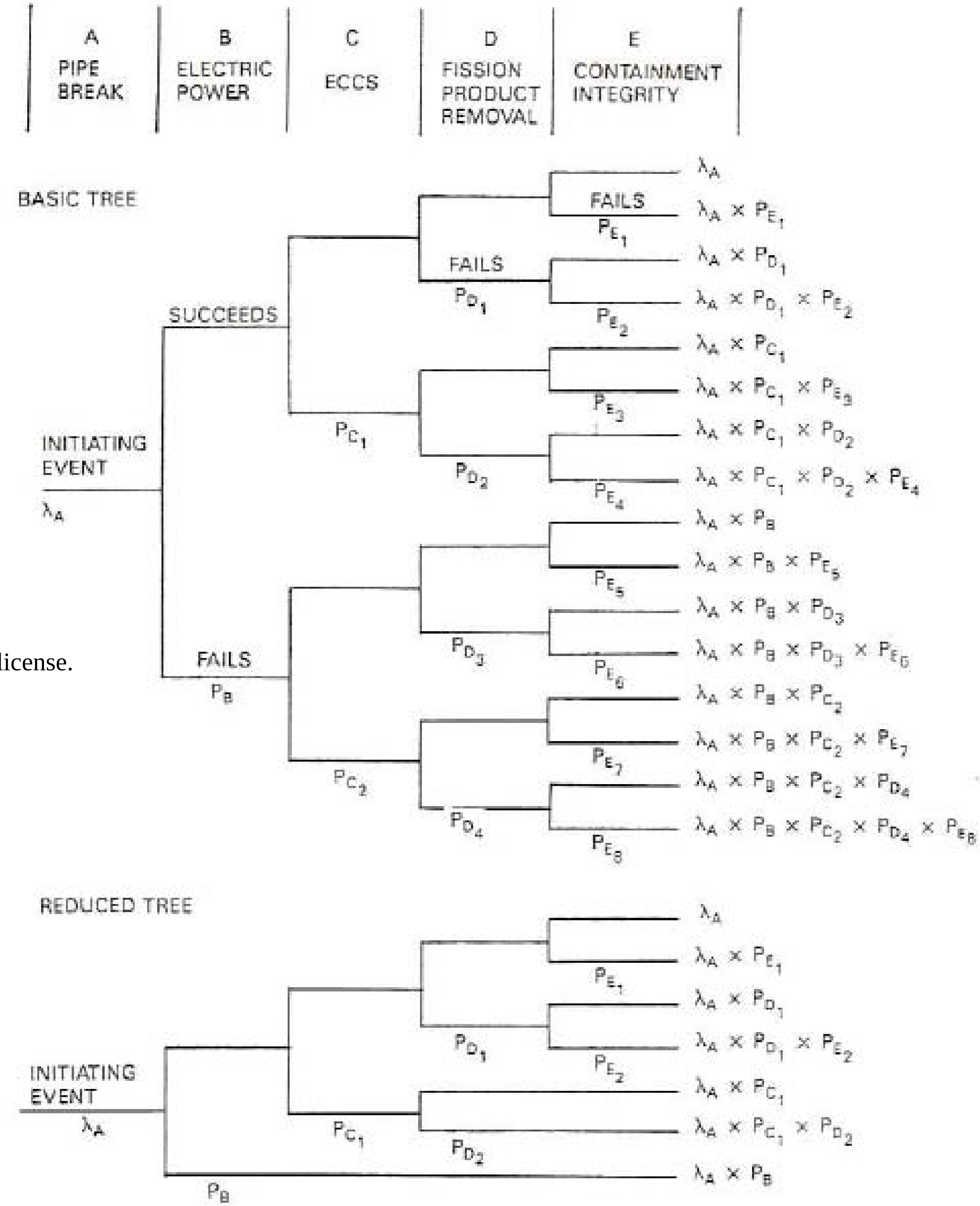


FIGURE 14-15
Simplified event tree logic diagrams for a design-basis LOCA in an LWR. (Adapted from WASH-1400, 1975.)

Table 2.3.1

Reactor Protective System Trip Settings

	Four Reactor Coolant Pumps Operating	Three Reactor Coolant Pumps Operating	Two Reactor Coolant Pumps Operating Same Loop	Two Reactor Coolant Pumps Operating Opposite Loops
1. High Power Level (1)	≤106.5% of Rated Power	≤80% of Rated Power	≤47% of Rated Power	≤51% of Rated Power
2. Low Reactor Coolant Flow (2)	≤95% of Reactor Coolant Flow with 4 Pumps Operating	≤73% of Reactor Coolant Flow with 4 Pumps Operating	≤47% of Reactor Coolant Flow with 4 Pumps Operating	≤50% of Reactor Coolant Flow with 4 Pumps Operating
3. High Pressurizer Pressure	≤2400 Psia	≤2400 Psia	≤2400 Psia	≤2400 Psia
4. Thermal Margin/Low Pressure (2)	Trip Point Set at Applicable Limits to Satisfy Figure 2-4	Trip Point Set at Applicable Limits to Satisfy Figure 2-3	Trip Point Set at Applicable Limits to Satisfy Figure 2-1	Trip Point Set at Applicable Limits to Satisfy Figure 2-2
5. Low Steam Generator Water Level	50" Below Normal Water Level	50" Below Normal Water Level	50" Below Normal Water Level	50" Below Normal Water Level
6. Low Steam Generator Pressure (3)	≤500 Psia	≤500 Psia	≤500 Psia	≤500 Psia
7. Containment High Pressure	≤4 Psig	≤4 Psig	≤4 Psig	≤4 Psig

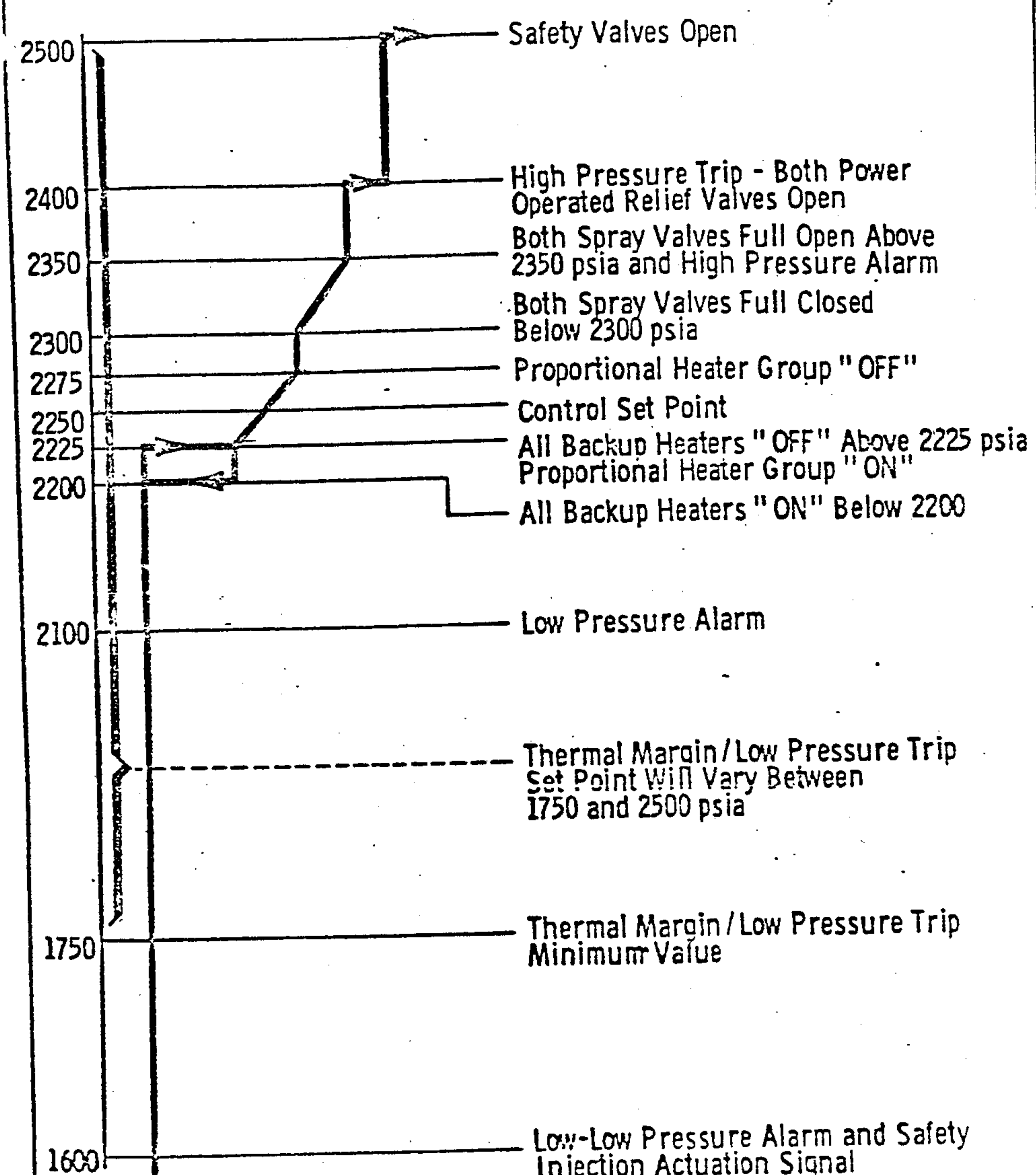
(1) Below 5% rated power, the trip setting may be manually reduced by a factor of 10.
(2) May be bypassed below 10^{-4}% of rated power provided auto bypass removal circuitry is operable.
(3) Manual inhibit permitted below 600 psig: automatically removed above 650 psig.

Additional trips not credited in safety analysis:

8. High rate of change of 2.6 DPM, functional between 10^{-4}% and 15% power

9. Loss of turbine load, bypassed when <15% power

10. Manual (2 locations)



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	Pressure Control Program	Figure 7-13

TABLE 15.4-1c

LARGE BREAK
TIME SEQUENCE OF EVENTS

	DECL (Sec)	0.6 DECL (Sec)	0.4 DECL (Sec)	0.8 DECL (Sec)
START	0.	0.	0.	0.
Rx Trip Signal	1.04	1.10	1.16	1.06
S. I. Signal	0.94*	1.10	1.16	1.02*
Acc. Injection	12.5	14.6	19.4	13.0
End of Blowdown	26.5	23.3	36.9	20.7
Bottom of Core Recovery	38.8	37.6	48.8	34.4
Acc. Empty	55.0	57.4	63.9	55.4
Pump Injection	25.94	26.10	26.16	26.02
End of Bypass	23.4	23.2	32.9	20.5

* From Containment Pressure Signal.

N10

TABLE 15.4-1d

LARGE BREAK
CONTAINMENT DATA - NEP CONTAINMENT

Net Free Volume	2.987×10^6 ft^3
Initial Conditions	
Pressure	14.7 psia
Temperature	90°F
RWST temperature	40°F
Raw water temperature	NA
Outside temperature	0°F
Relative humidity	99%
Spray System	
Number of pumps operating	2
Runout flow rate (total)	6600 gpm
Actuation time	35 sec

Structural Heat Sinks

Item	Thickness (ft)	Area (ft^2)
Containment Cylinder	.0313 steel, 4.5 concrete	70151
Containment Dome	.0417 steel, 3.5 concrete	33867
Containment Floor	4.0 concrete, .0208 steel, 9.0 concrete	13820
Containment Sump	9.0 concrete	972
Miscellaneous Concrete	1.0 concrete	160,000
Miscellaneous Steel	0.2 steel	5,000
	0.05 steel	65,000
	0.03125 steel	90,000
	0.030 steel	100,000
	0.020 steel	70,000
	0.0057 steel	45,000

N10

TABLE 15.4-1a

LARGE BREAK

	DECL	0.6 DECL	0.4 DECL	0.8 DECL
Results				
Peak Clad Temp. ^oF	2148	2137	1790	2144
Peak Clad Location Ft.	7.5	7.5	9.0	7.5
Local Zr/H_2O Rxn (max) %	6.7	6.7	2.1	6.8
Local Zr/H_2O Location Ft.	7.5	7.5	8.0	6.0
Total Zr/H_2O Rxn %	<0.3	<0.3	<0.3	<0.3
Hot Rod Burst Time sec	21.0	23.2	84.8	20.2
Hot Rod Burst Location Ft.	5.75	5.75	7.0	6.0

Calculation	
NSSS Power Mwt 102% of	3425
Peak Linear Power kw/ft 102% of	12.6
Peaking Factor (At License Rating)	2.32
Accumulator Water Volume (Cubic Feet)	950

Fuel region + cycle analyzed	Cycle	Region
UNIT 1	1	All

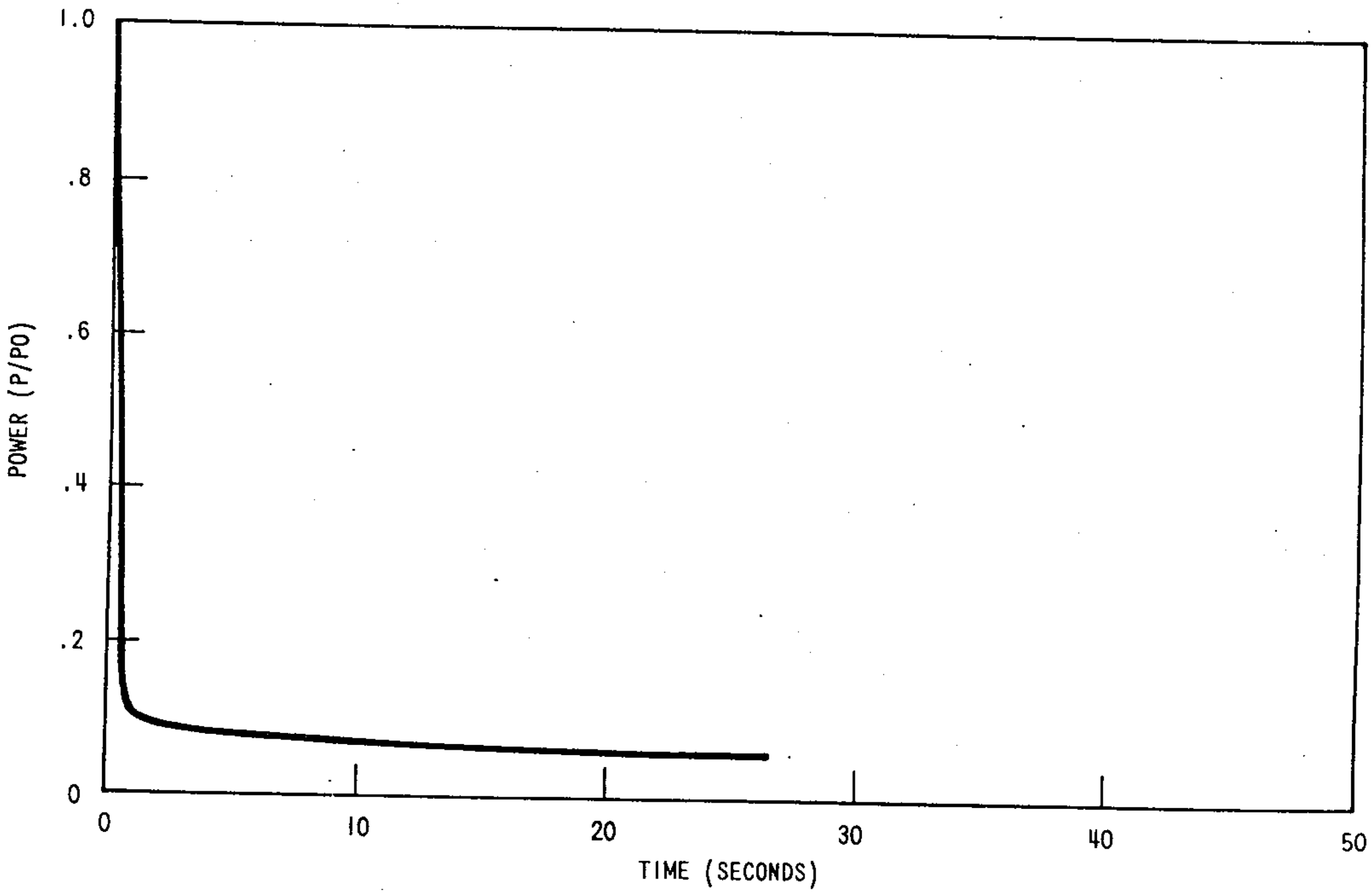
N10

NEW ENGLAND POWER COMPANY
NEP 1 & 2
Preliminary Safety Analysis Report

CORE POWER TRANSIENT – DECLG (C_D = 1.0)

FIG. 15.4.1-14A

AMENDMENT N10
MARCH 1978

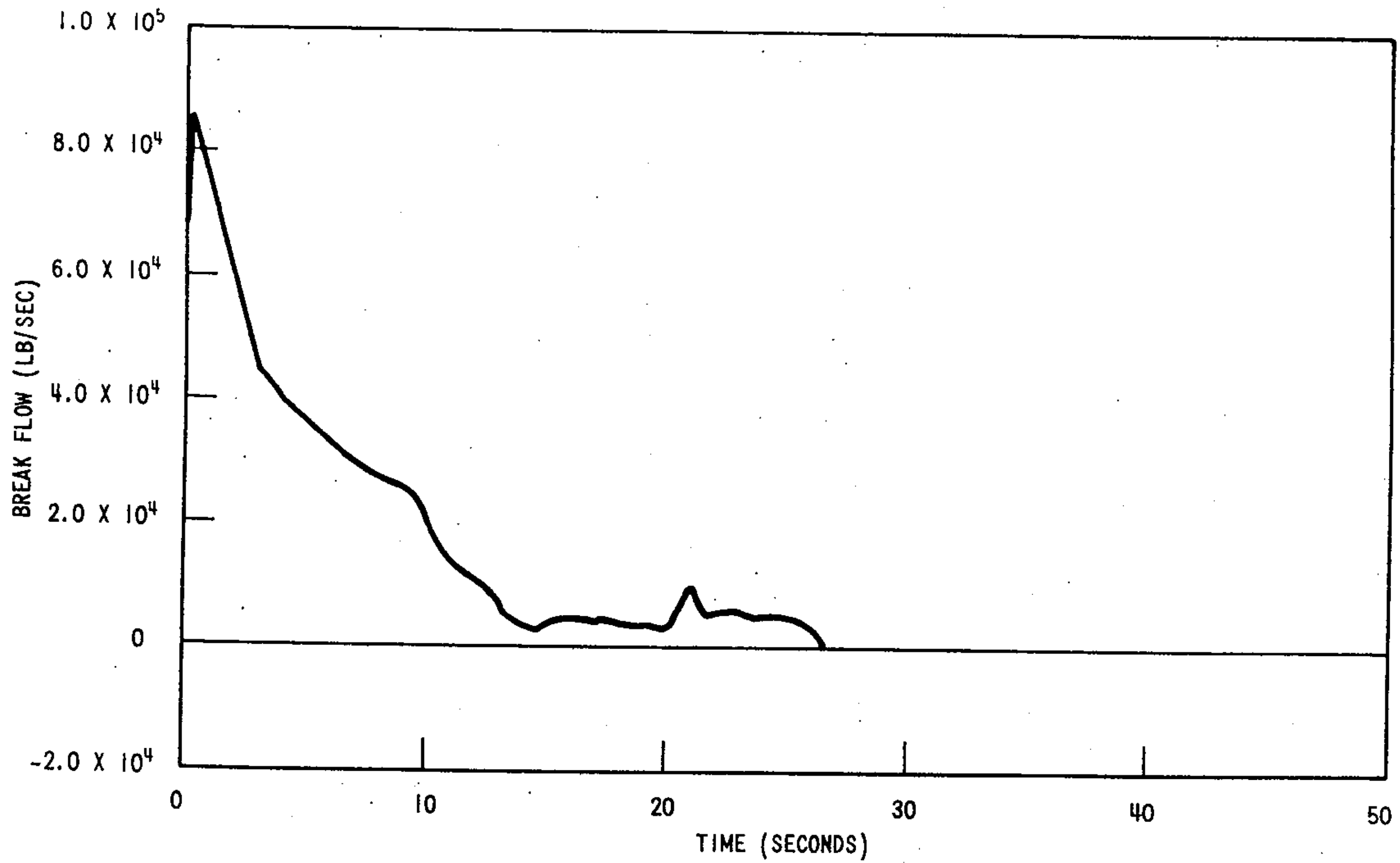


NEW ENGLAND POWER COMPANY
NEP 1 & 2
Preliminary Safety Analysis Report

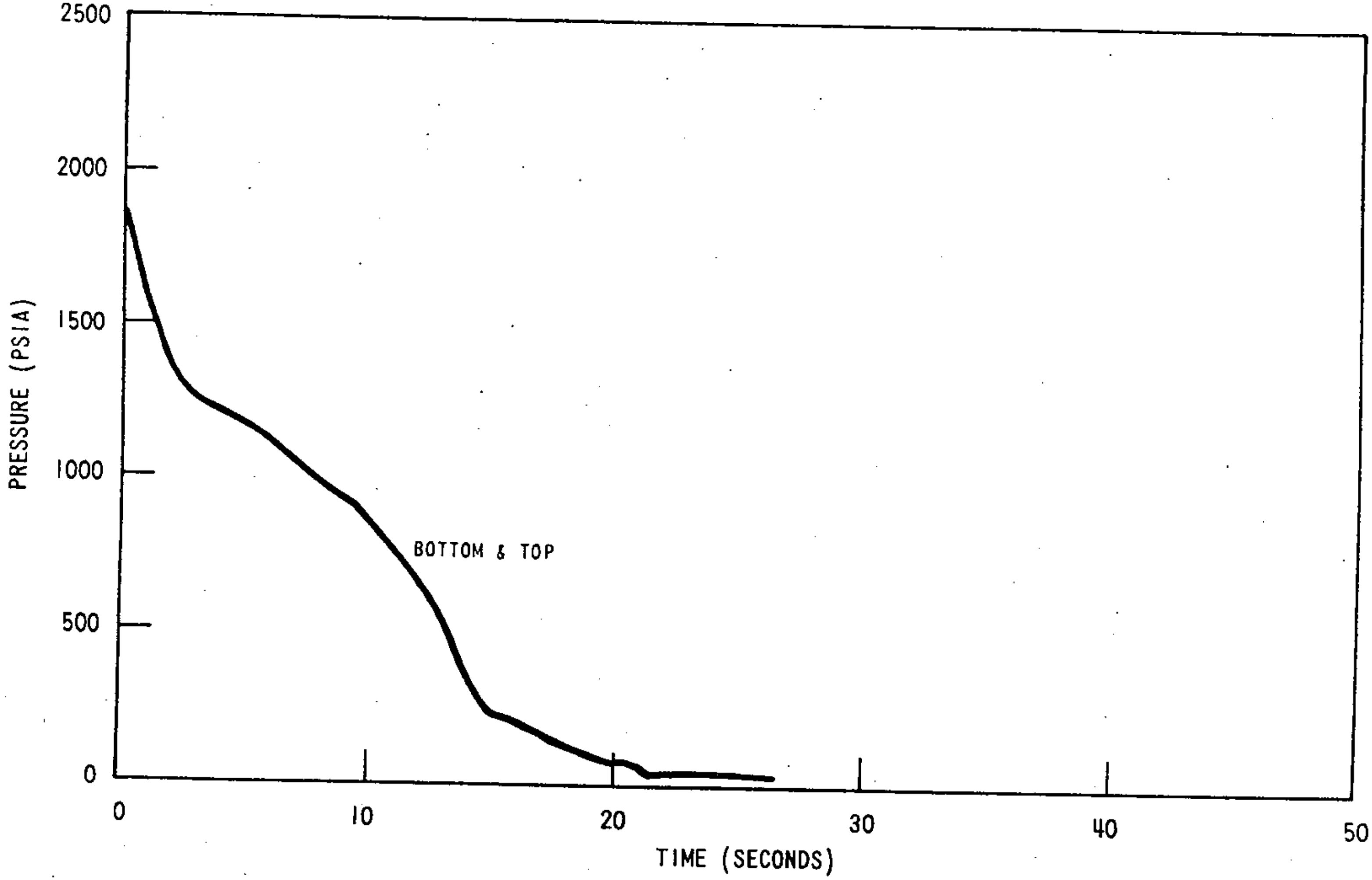
BREAK FLOW RATE – DECLG (C_D = 1.0)

FIG. 15.4.1-5A

AMENDMENT N10
MARCH 1978



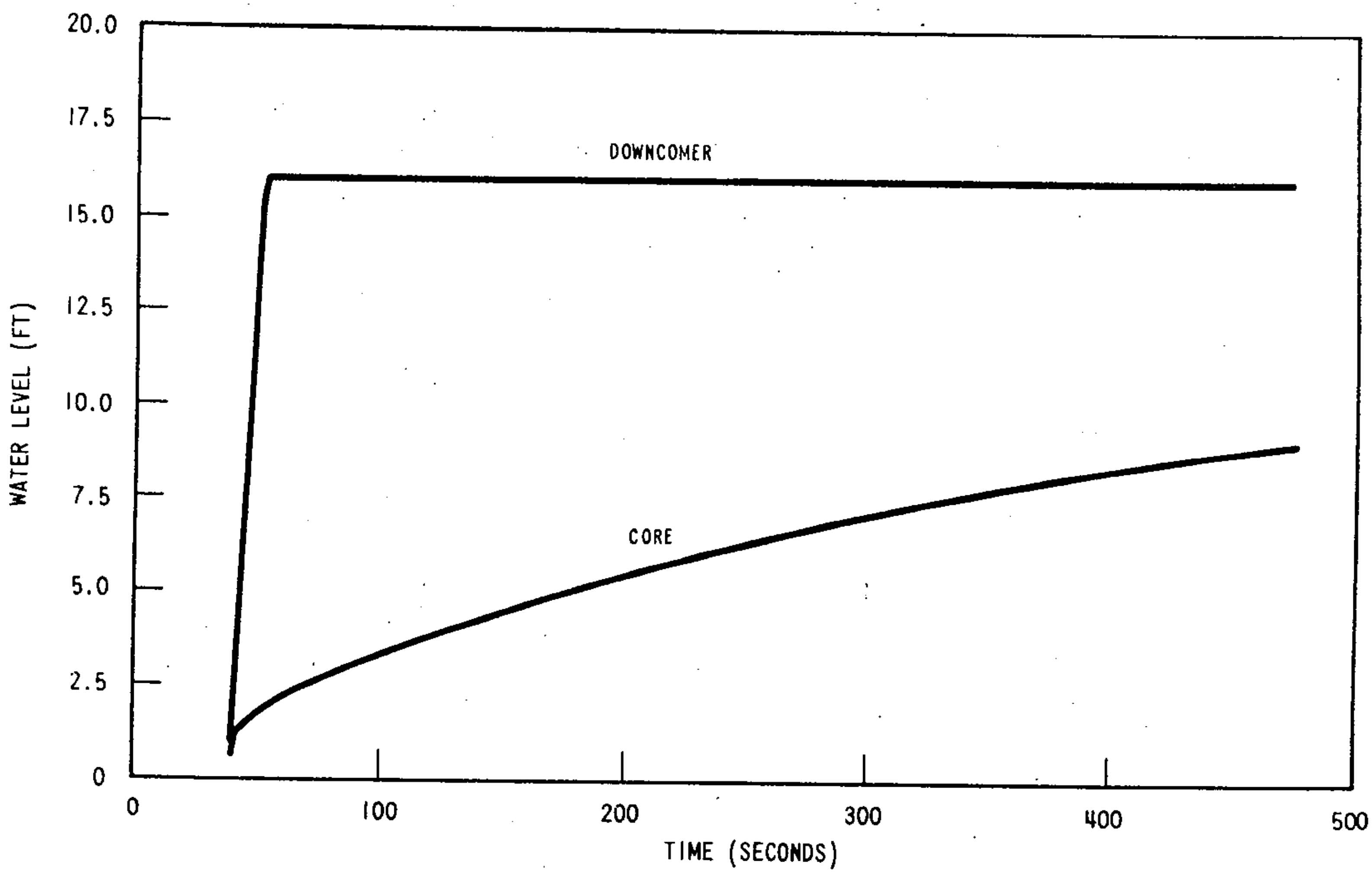
NEW ENGLAND POWER COMPANY
N E P 1 & 2
Preliminary Safety Analysis Report



CORE PRESSURE – DECLG (C_D = 1.0)

FIG. 15.4.1-4A

AMENDMENT N10
MARCH 1978



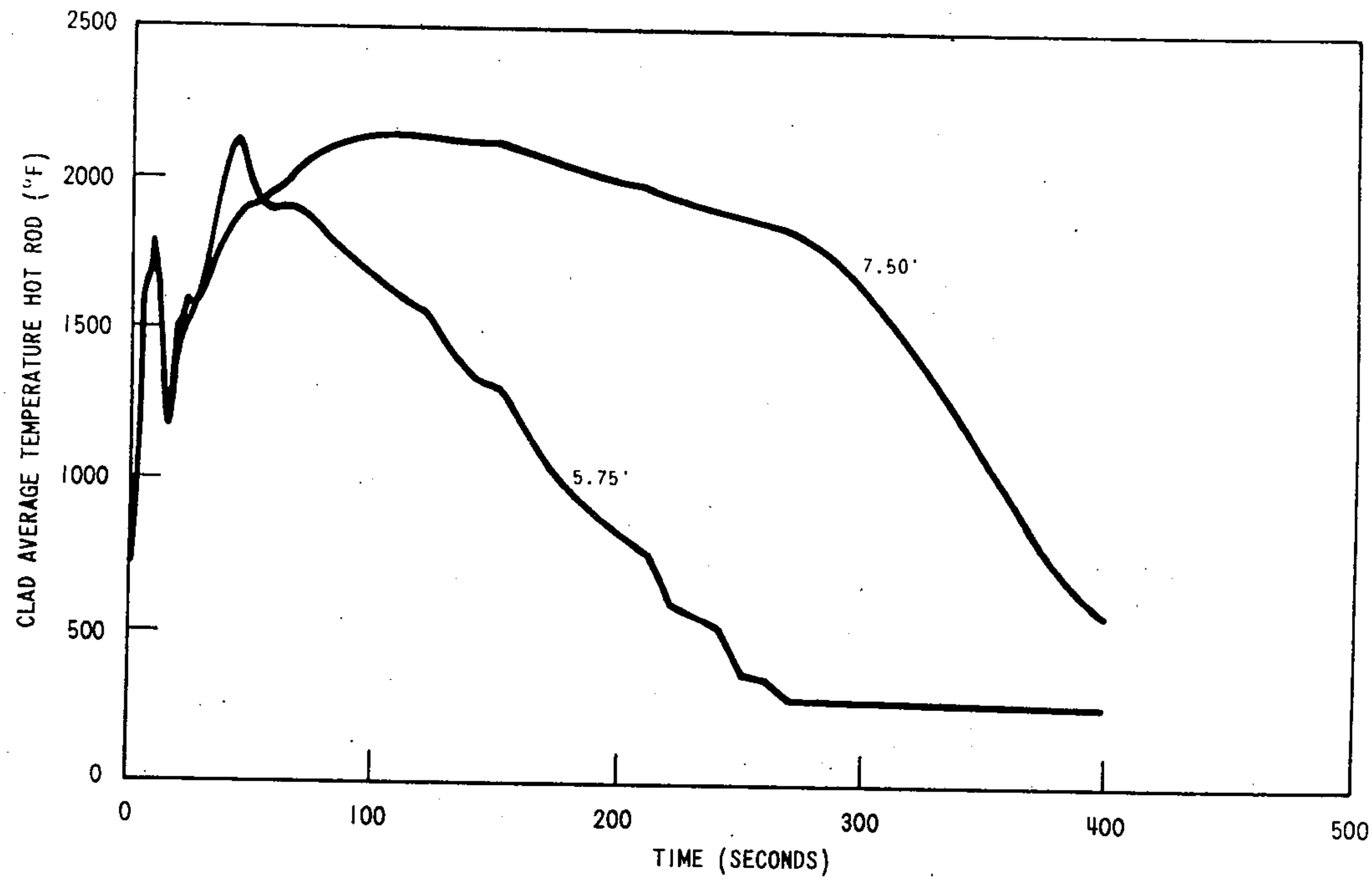
NEW ENGLAND POWER COMPANY
NEP 1 & 2
Preliminary Safety Analysis Report

REFLOOD TRANSIENT – DECLG (C_D = 1.0)
DOWNCOMER AND CORE WATER LEVELS

FIG. 15.4.1-10A

AMENDMENT N10
MARCH 1978

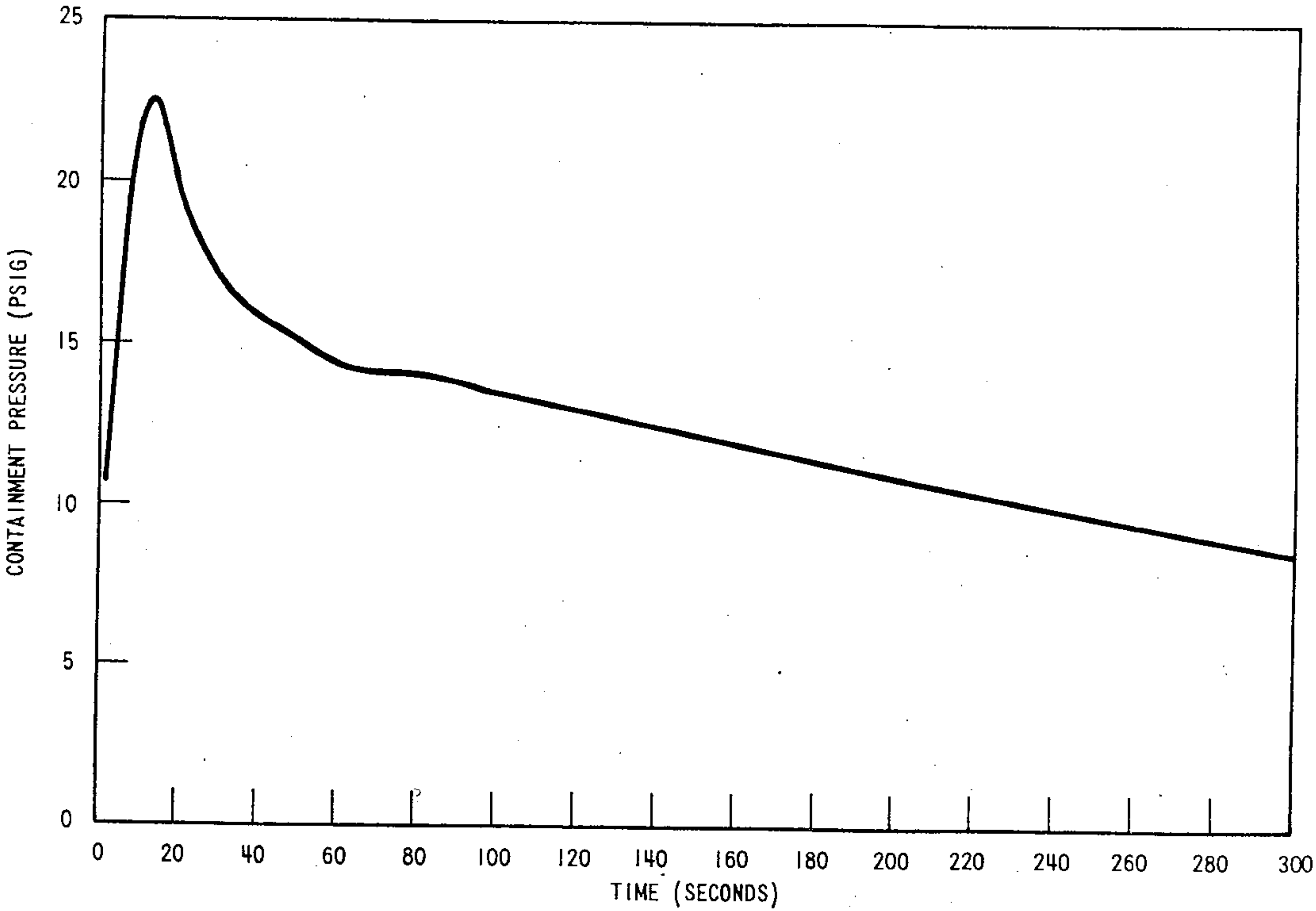
NEW ENGLAND POWER COMPANY
NEP 1 & 2
Preliminary Safety Analysis Report



PEAK CLAD TEMPERATURE — DECLG (C_D = 1.0)

FIG. 15.4.1-7A

AMENDMENT N10
MARCH 1978

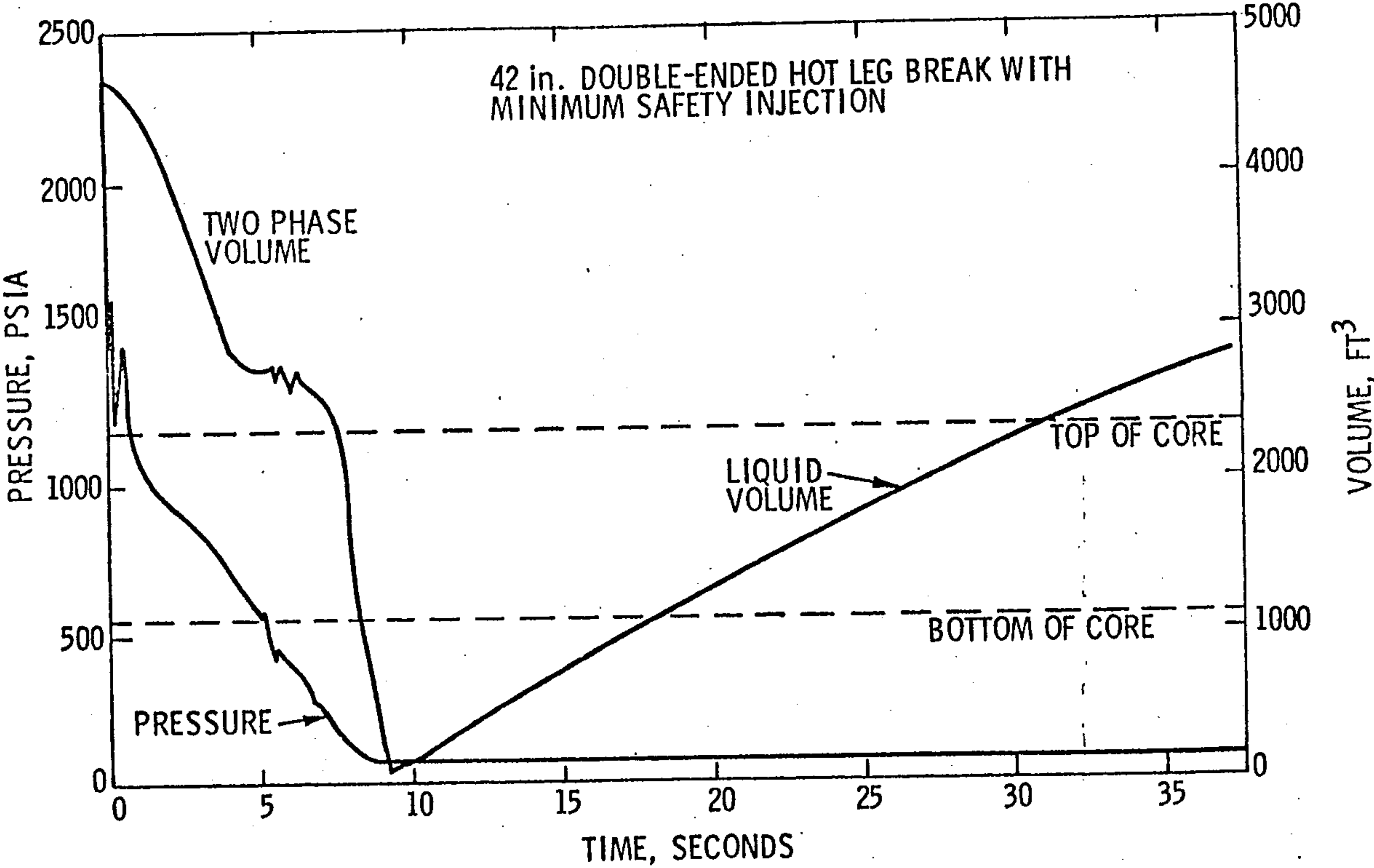


NEW ENGLAND POWER COMPANY
NEP 1 & 2
Preliminary Safety Analysis Report

CONTAINMENT PRESSURE-DECLG (C_D = 1.0)

FIG. 15.4.1-13A

AMENDMENT N10
MARCH 1978



Calvert Cliffs

Loss of Coolant Incident
Pressure and Volume vs Time

Figure 14.15-3

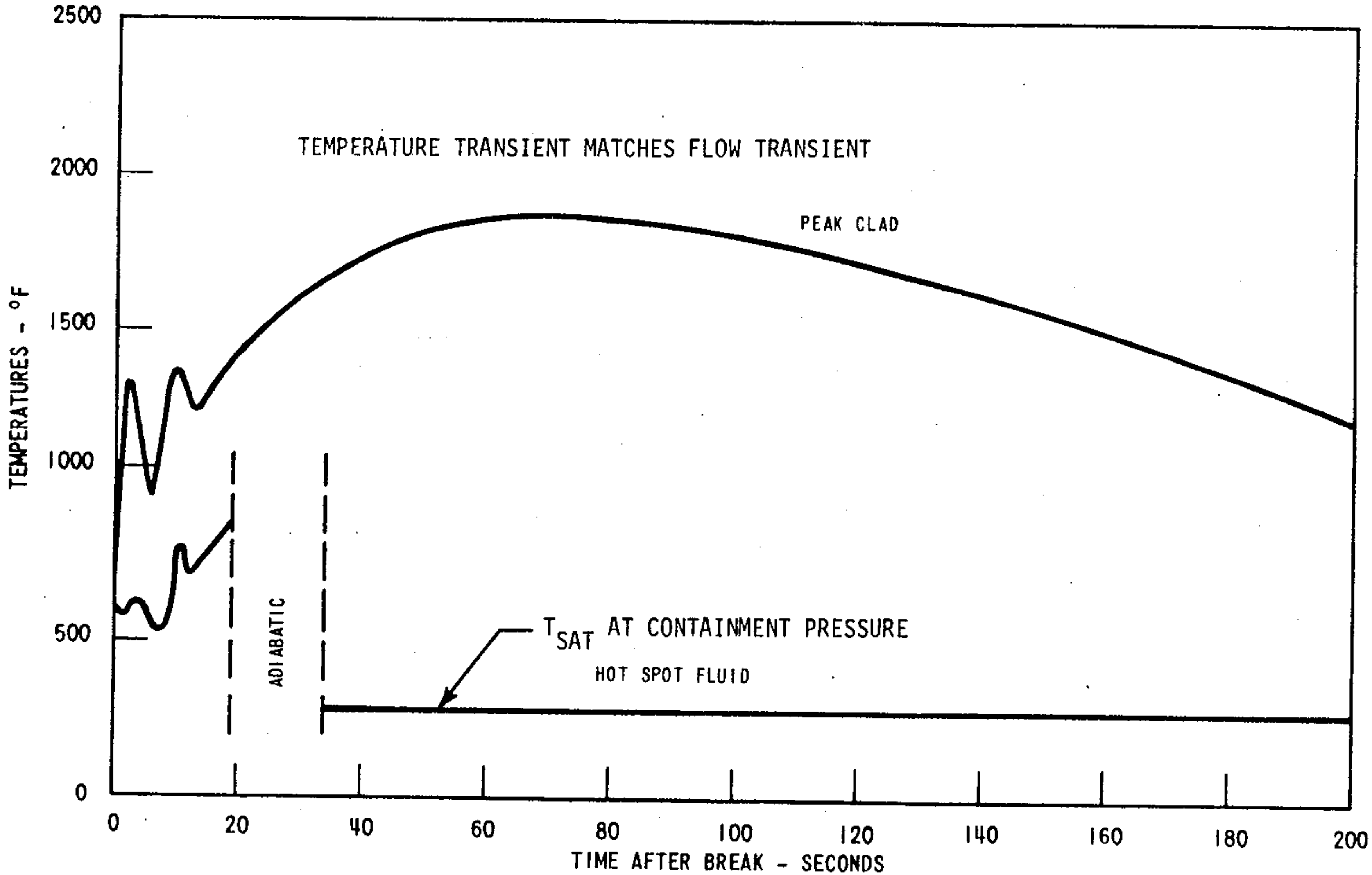


Figure 15.4-16. Double Ended Cold Leg Break (Guillotine)

15.4-106

6274-105

15.4 CONDITION IV - LIMITING FAULTS

Refer to RESAR-3 (4-loop, without loop stop valves) Section 15.4, with the following modifications:

15.4.1 Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident)

The analysis specified by 10 CFR Part 50.46(1) Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors is presented in this section. The results of the LOCA analyses are shown in Table 15.4-1a and show compliance with the Acceptance Criteria. The analytical techniques used are in compliance with Appendix K of 10 CFR Part 50, and are described in Reference (2). The results for the small break LOCA are presented in subsection 15.3.1 of the PSAR and are in conformance with 10 CFR Part 50.46 and Appendix K of 10 CFR Part 50.

The boundary considered for the LOCA as related to connecting piping is defined in RESAR-3, Section 3.6.

Should a major break occur, depressurization of the reactor coolant system results in a pressure decrease in the pressurizer. Reactor trip signal occurs when the pressurizer low-pressure trip set point is reached. A safety injection system signal is actuated when the appropriate set point is reached. These countermeasures will limit the consequences of the accident in two ways:

a. Reactor trip and borated water injection complement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat.

b. Injection of borated water provides heat transfer from the core and prevents excessive clad temperatures.

At the beginning of the blowdown phase, the entire reactor coolant system contains subcooled liquid which transfers heat from the core by forced convection with some fully developed nucleate boiling. After the break develops, the time to departure from nucleate boiling is calculated, consistent with Appendix K of 10 CFR Part 50. Thereafter, the core heat transfer is based on local conditions with transition boiling and forced convection to steam as the major heat transfer mechanisms. During the refill period rod-to-rod radiation is the only heat transfer mechanism.

N10

When the reactor coolant system pressure falls below 600 psia, the accumulators begin to inject borated water. The conservative assumption is made that accumulator water injected bypasses the core and goes out through the break until the termination of bypass. The conservatism is again consistent with Appendix K of 10 CFR Part 50.

15.4.1.1 Thermal Analysis

a. Westinghouse Performance Criteria for Emergency Core Cooling System

The reactor is designed to withstand thermal effects caused by a LOCA, including the double-ended severance of the largest reactor coolant system pipe. The reactor core and internals, together with the emergency core cooling system, are designed so that the reactor can be safely shut down and the essential heat transfer geometry of the core preserved following the accident.

The emergency core cooling system, even when operating during the injection mode with the most severe single active failure, is designed to meet the Acceptance Criteria.

b. Method of Thermal Analysis

The description of the various aspects of the LOCA analysis is given in Reference (2). This document describes the major phenomena modeled, the interfaces among the computer codes, and features of the codes which maintain compliance with the Acceptance Criteria. The individual codes are described in detail in References (3) through (6). The containment parameters used in the containment analysis code, Reference (6), to determine the emergency core cooling system backpressure are presented in Table 15.4-1b.

The analysis presented here was performed using the October, 1975 version of the Westinghouse Evaluation Model. This version includes the modifications to the models, referenced above, as specified by the NRC in Reference (7) and complies with Appendix K of 10 CFR Part 50. The October, 1975 Westinghouse Evaluation Model is documented in References (8) through (10).

The analysis was performed using the conservative assumption that the fluid temperature in the upper head of the reactor vessel is equal to the reactor vessel outlet temperature. The effect of upper head temperature on ECCS performance is discussed in References (13) and (14).

The time sequence of events for all breaks analyzed is shown in Table 15.4-1c.

N10

The analysis was performed using a reference containment which has internal steel and concrete structural heat sinks which conform to the guidelines of Branch Technical Position CSB 6-1.

The containment initial conditions of 90°F and 14.7 psia are representatively low values anticipated during normal full-power operation. The initial relative humidity was conservatively assumed to be 98.8 percent.

The condensing heat transfer coefficient used for heat transfer to the steel containment structures for the limiting break is given in Figure 15.4.1-16.

The containment temperature response is presented in Figure 15.4.1-17 for the limiting break.

The containment sump temperature does not affect the analysis because the maximum peak cladding temperature occurs prior to initiation of the recirculation mode for containment spray system.

The mass and energy releases used in the containment backpressure calculation for the limiting break are presented in Table 15.4-1e.

These results can be demonstrated to be conservative for NEP as follows:

Table 15.4-1b lists the reference containment parameters used in the calculation that yielded a peak clad temperature of 2148°F. Table 15.4-1d lists the NEP containment parameters. Figure 15.4.1-18 shows containment pressure versus time for the limiting break for each set of containment parameters. The figure demonstrates that the NEP back pressure is at all times higher than that of the containment used in the ECCS performance calculation. As a result, the core flooding rates for NEP 1 & 2 exceed the calculated flooding rates; the higher flooding rates will yield a peak clad temperature lower than 2148°F.

c. <u>Results</u>

Table 15.4-1a presents the peak clad temperatures, hot spot metal reaction, and other key results for a range of break sizes. The range of break sizes was determined to include the limiting case for peak clad temperature from sensitivity studies reported in References (11) and (12).

The SATAN VI analysis of the LOCA is performed at 102 percent of Engineered Safeguards Design Rating. The peak linear power and core power used in the analyses are given in Table 15.4-1a. The equivalent core parameter at the license application power level are also shown in Table 15.4-1a. Since there is margin between the value of the peak linear power density used in this analysis and the value expected in operation, a lower peak clad temperature would be obtained by using the peak linear power density expected during operation. N10

For the results discussed below, the hot spot is defined to be the location of maximum peak clad temperature. This location is given in Table 15.4-1a for each break size analyzed.

Figures 15.4.1-1A through 15.4.1-16 present the transients for the principal parameters for the break sizes analyzed. The following items are noted:

Figures 15.4.1-1A through 15.4.1-3D	The following quantities are presented at the clad burst location and at the hot spot (location of maximum clad temperature), both on the hottest fuel rod (hot rod): a. Fluid quality b. Mass velocity c. Heat transfer coefficient The heat transfer coefficient shown is calculated by the LOCTA IV Code.
Figures 15.4.1-4A through 15.4.1-6D	The system pressure shown is the calculated pressure in the core. The flow rate out the break is plotted as the sum of both ends for the guillotine break cases. The core pressure drop shown is from the lower plenum, near the core, to the upper plenum at the core outlet.
Figures 15.4.1-7A through 15.4.1-9D	These figures show the hot spot clad temperature transient and the clad temperature transient at the burst location. The fluid temperature shown is also for the hot spot and burst location. The core flow (top and bottom) is also shown.
Figures 15.4.1-10A through 15.4.1-10H	These figures present the core reflood transient.
Figures 15.4.1-11A through 15.4.1-12D	These figures show the emergency core cooling system flow for all cases analyzed. As described earlier, the accumulator delivery during blowdown is discarded until the end of bypass is calculated. Accumulator flow, however, is established in refill-reflood calculations. The accumulator flow assumed is the sum of that injected in the intact cold legs.
Figures 15.4.1-13A through 15.4.1-13D	These figures show the containment pressure transient.

N10

Figures 15.4.1-14A through 15.4.1-14D — These figures show the core power transient.

Figure 15.4.1-15 — This figure shows the break energy released to the containment during blowdown.

Figure 15.4.1-16 — This figure provides the containment wall condensing heat transfer coefficient.

In addition to the above, Tables 15.4-1e and 15.4-1f present the reflood mass and energy releases to the containment and the broken loop accumulator mass and energy flowrate to the containment, respectively.

The clad temperature analysis is based on a total peaking factor of 2.32. The hot spot metal water reaction reached is 6.7 percent, which is well below the embrittlement limit of 17 percent, as required by 10 CFR Part 50.46. In addition, the total core metal water reaction is less than 0.3 percent for all breaks as compared with the 1 percent criterion of 10 CFR Part 50.46.

The results of several sensitivity studies are reported in Reference (12). These results are for conditions which are not limiting in nature and hence are reported on a generic basis.

Conclusions - Thermal Analysis

For breaks up to and including the double-ended severance of a reactor coolant pipe, the emergency core cooling system will meet the Acceptance Criteria as presented in 10 CFR PART 50.46. That is:

a. The calculated peak fuel element clad temperature provides margin to the requirement of 2200°F even with containment parameters as conservative as those presented in Table 15.4-1b.

b. The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1 percent of the total amount of Zircaloy in the reactor.

c. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. The cladding oxidation limits of 17 percent are not exceeded during or after quenching.

d. The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long-lived radioactivity remaining in the core.

N10

Reading and Homework Assignment

1. Read Knief Chapter 14
2. Problems: 14.9, 11, 12, 21, 23