Large and Small Break LOCA Analysis/Result

Course 22.39, Lecture 15

11/1/06

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Event Categories
(from Lillington, UK)

Category	Description	Frequency/Reactor
1	Conditions that occur regularly in normal operation	~10
2	Faults that are expected during the life of the plant: Anticipated moderately frequent events requiring safety response	~1
3	Faults not expected during the life of a particular plant: Anticipated infrequent events requiring safety response	$\sim 10^{-2}$
4	Improbable events not expected to occur in the nuclear industry but provided for by the design	$\sim 10^{-4}$
5	Extremely improbable events not provided for in the design of the plant	$\sim 10^{-6}$

Figure by MIT OCW. After Lillington, 1995.

Example Events
(from Lillington, UK)

Events	Categories
Bringing the Reactor to Full Power	1
Loss of External Grid Loss of Feedwater Loss of Reactor Coolant Pump	2
Small LOCA Valves Open	3
Large LOCA Main Steam Line Break	4
LOCAs without ECCS Transients with Total Loss of ON- and Off-Site Power	5

Figure by MIT OCW. After Lillington, 1995.

Energy Outflows as Steam and Water

Bleed and Feed

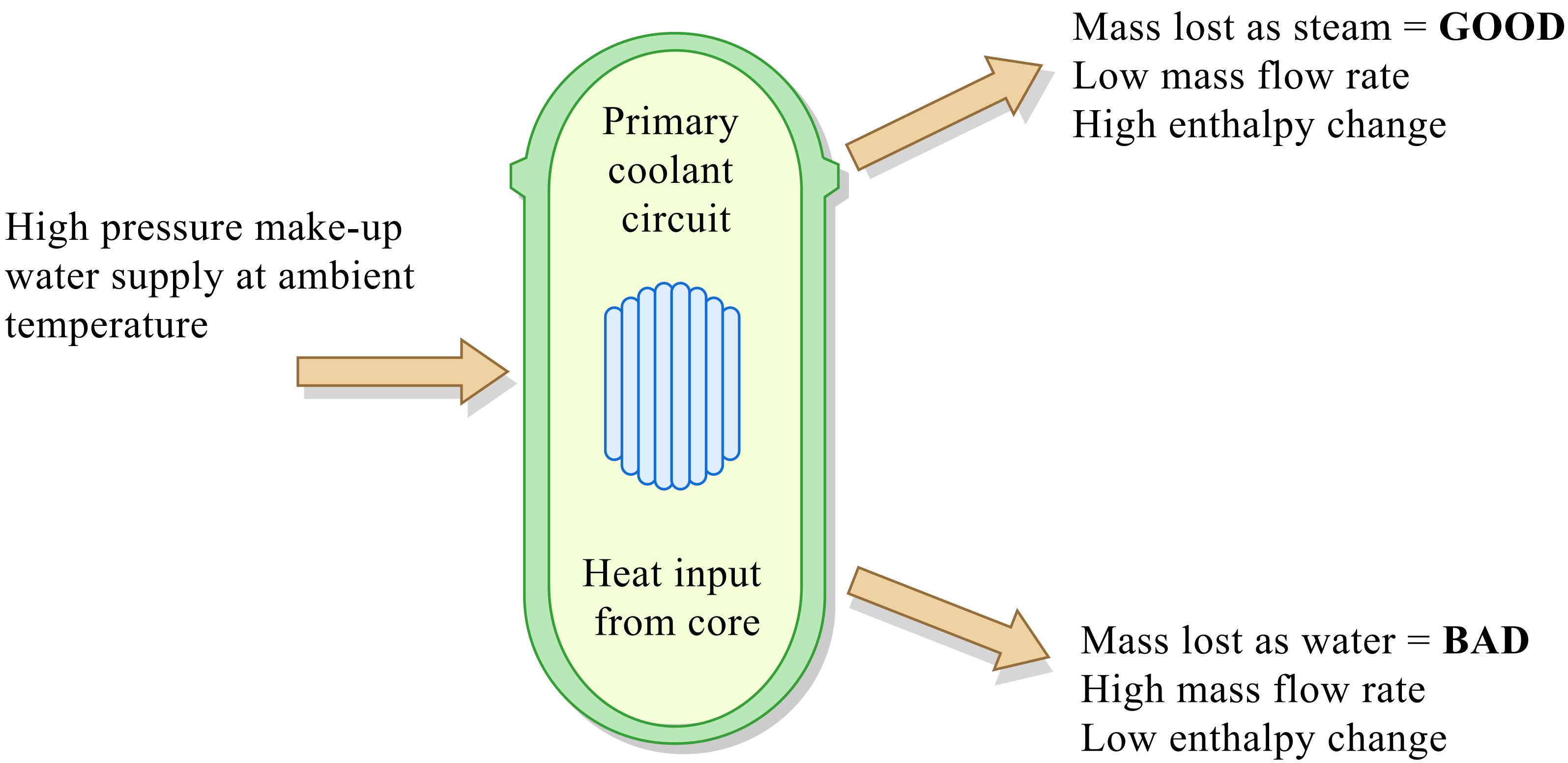
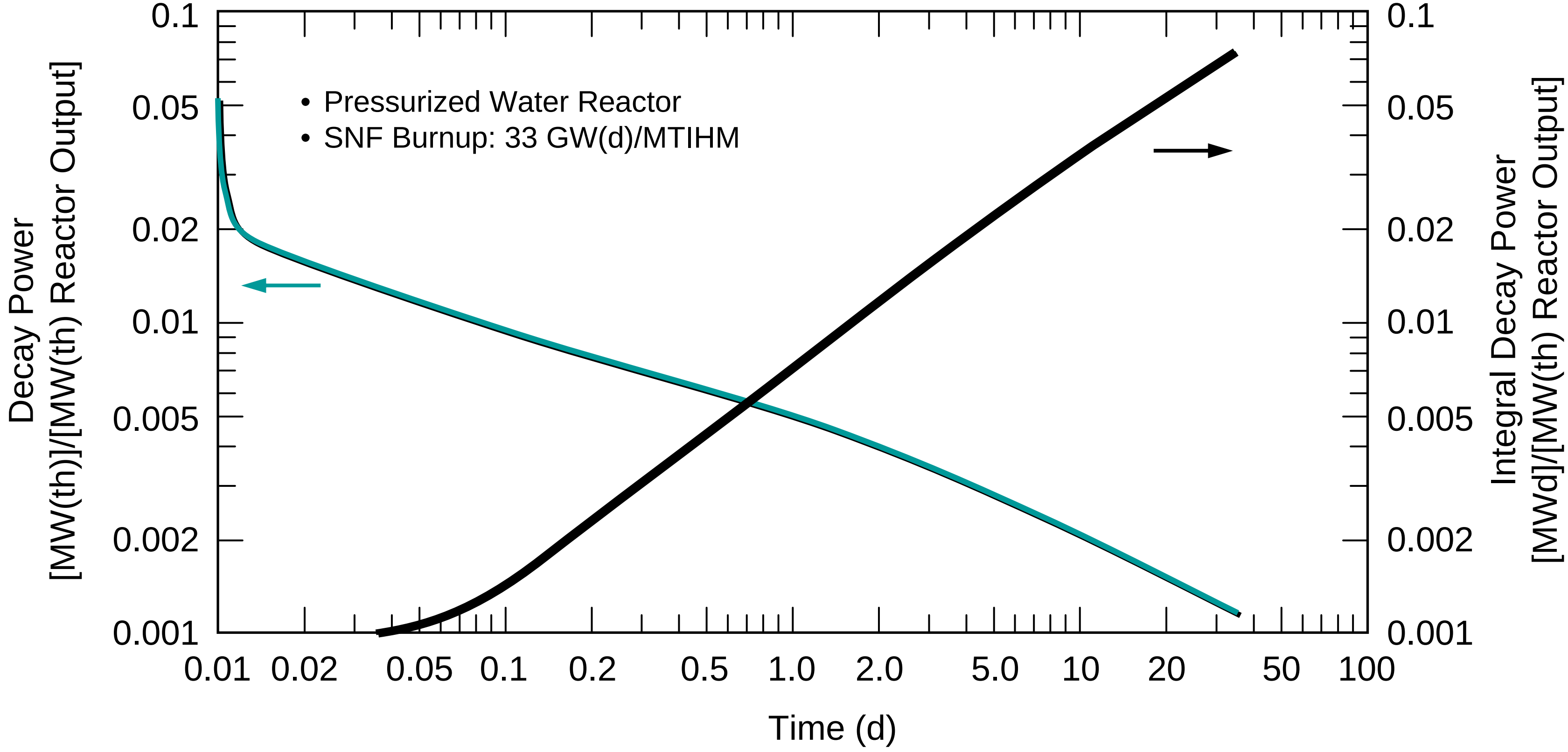


Figure by MIT OCW.

Decay Power and Integral Decay Power As A Function of Time

By Charles Forsberg. Courtesy of Oak Ridge National Laboratory.



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UT-BATTELLE

Diagrammatic representation of PWR primary and secondary circuits and the emergency cooling systems

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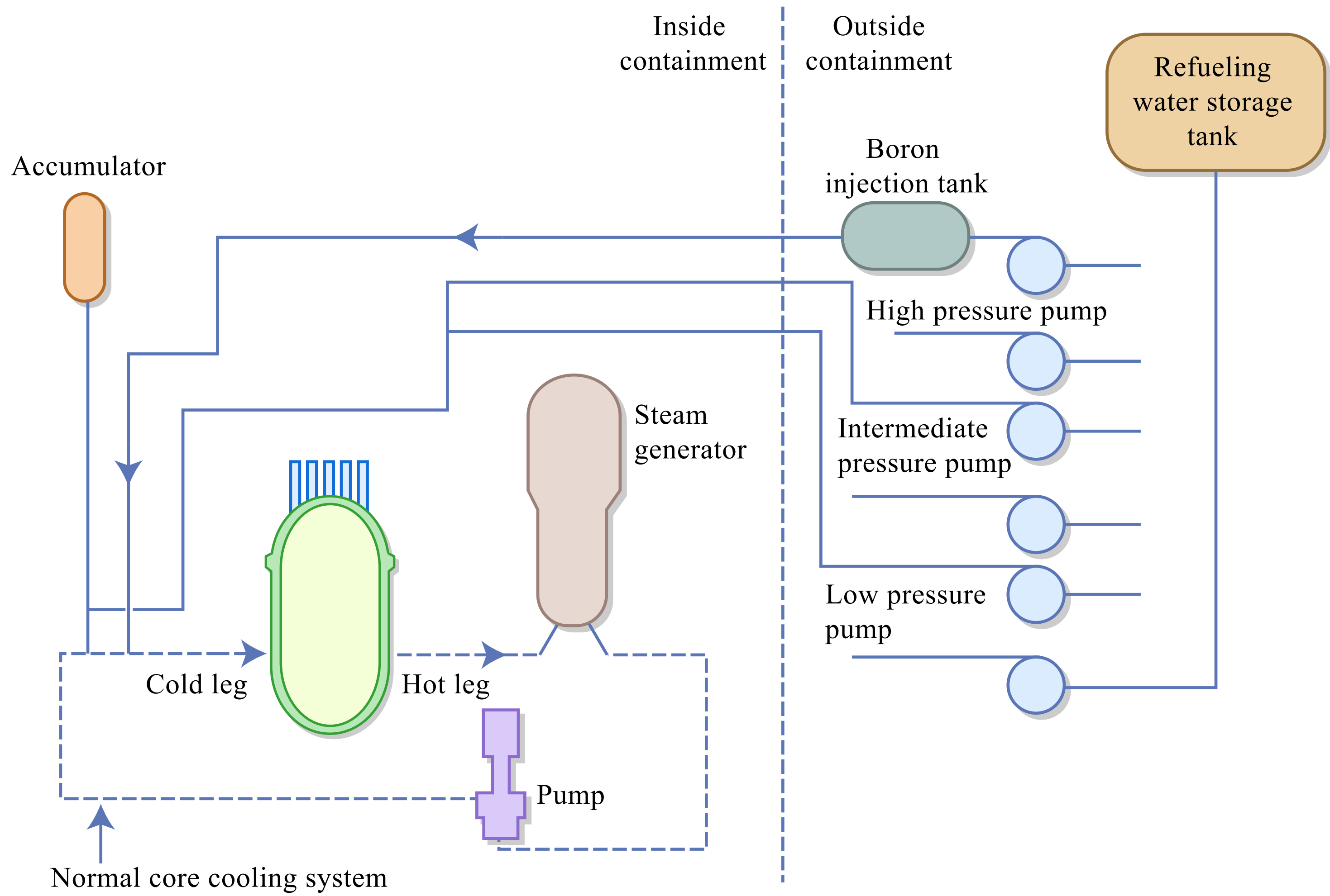
Figure 4.4 in Collier, J. G., and G. F. Hewitt. *Introduction to Nuclear Power*.
Washington, DC: Hemisphere Publishing, 1987.



EMERGENCY CORE COOLING SYSTEM (SCHEMATIC)

Figure by MIT OCW.

Connection pipe diameter/cross section/percentage spectrum of a PWR. (Solid lines) Primary loop system; (dashed lines) pressurizer

Appendix C: Basic Assumptions of the LOCA

1. The reactor has been operating for an infinite period of time at an assumed slight overpower condition. No power or other transient preceeds the accident.
2. Peak core power density or linear power generation is at the maximum allowable value.
3. A double-ended rupture of <u>one</u> primary coolant loop is assumed (largest existing pipe)

   PWR: rupture of cold leg imposes most severe conditions

   BWR: rupture of a recirculation loop

   Remaining intact loops continue their operation as dictated by available electric supply or stored rotational energy.
4. Off-site power is lost upon initiation of the accident and is restored after several days.

(continued)

5. Reactor scram systems need not contribute to the nuclear shut down because voiding of the core provides sufficient negative reactivity for shutdown.

6. The reactor is isolated after the initiation of the accident, i.e. the regular heat sink is removed.

   PWR: Upon initiation of the accident the steam generators are isolated on the secondary side by closing the steam supply valves and the feedwater valves.

   BWR: Upon receipt of a reactor-vessel low-water signal the main steam isolation valves close within 10 seconds. Feedwater flow ramps to zero within four seconds.

(continued)

7. EECS are actuated automatically by appropriate signals. No corrective operator action is assumed for the first 10 minutes following initiation of the accident.

8. A single failure criterion is applied to the reactor system whereby an additional fault is postulated which may render inoperative any <u>one</u> of the following:

   - Mechanical active components (e.g. pump)
   - Active or passive electrical components

Events in the reactor pressure vessel during a large-break LOCA. a) Normal operation; b) blowdown phase; c) refill phase; d) reflood phase

PWR operating conditions

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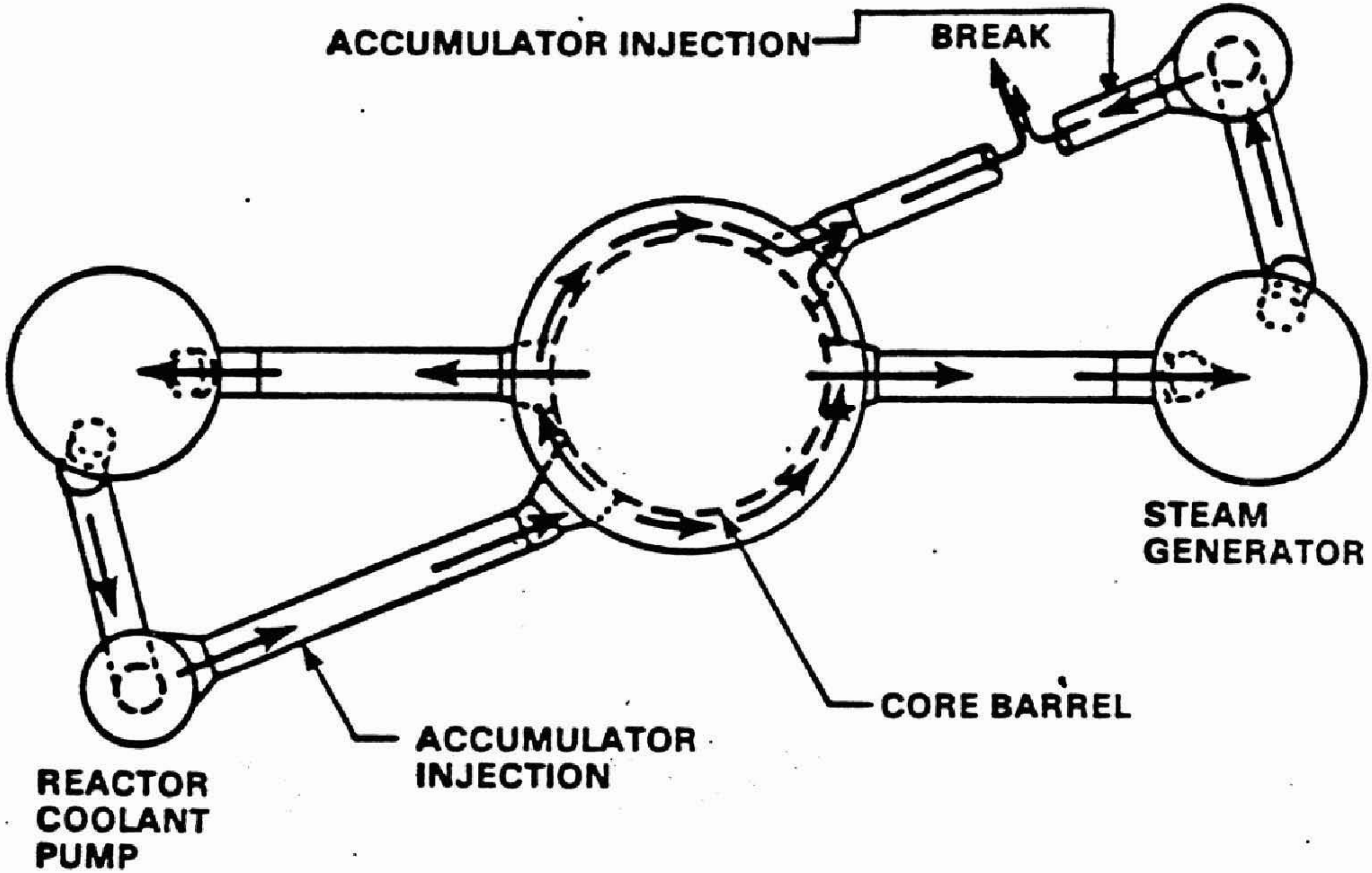
Figure 4.5 in Collier, J. G., and G. F. Hewitt. *Introduction to Nuclear Power*. Washington, DC: Hemisphere Publishing, 1987.

Cold Leg Break Steam Flow Path



Courtesy of Westinghouse. Used with permission.

Adiabatic heat-up for PWR fuel (17 x 17)

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Figure 4.1 in Collier, J. G., and G. F. Hewitt. *Introduction to Nuclear Power*.
Washington, DC: Hemisphere Publishing, 1987.

NRC Appendix K Criteria

1) <u>Peak Cladding Temperature</u>. The calculated maximum fuel element cladding temperature shall not exceed 2200° F.
2) <u>Maximum Cladding Oxidation</u>. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.
3) <u>Maximum Hydrogen Generation</u>. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.
4) <u>Coolable Geometry</u>. Calculated changes in core geometry shall be such that the core remains amenable to cooling.
5) <u>Long-Term Cooling</u>. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptable low value and decay heat removed for the extended period of time required by the long-lived radioactivity remaining in the core.

Variation of peak clad temperature with time for a large-break LOCA

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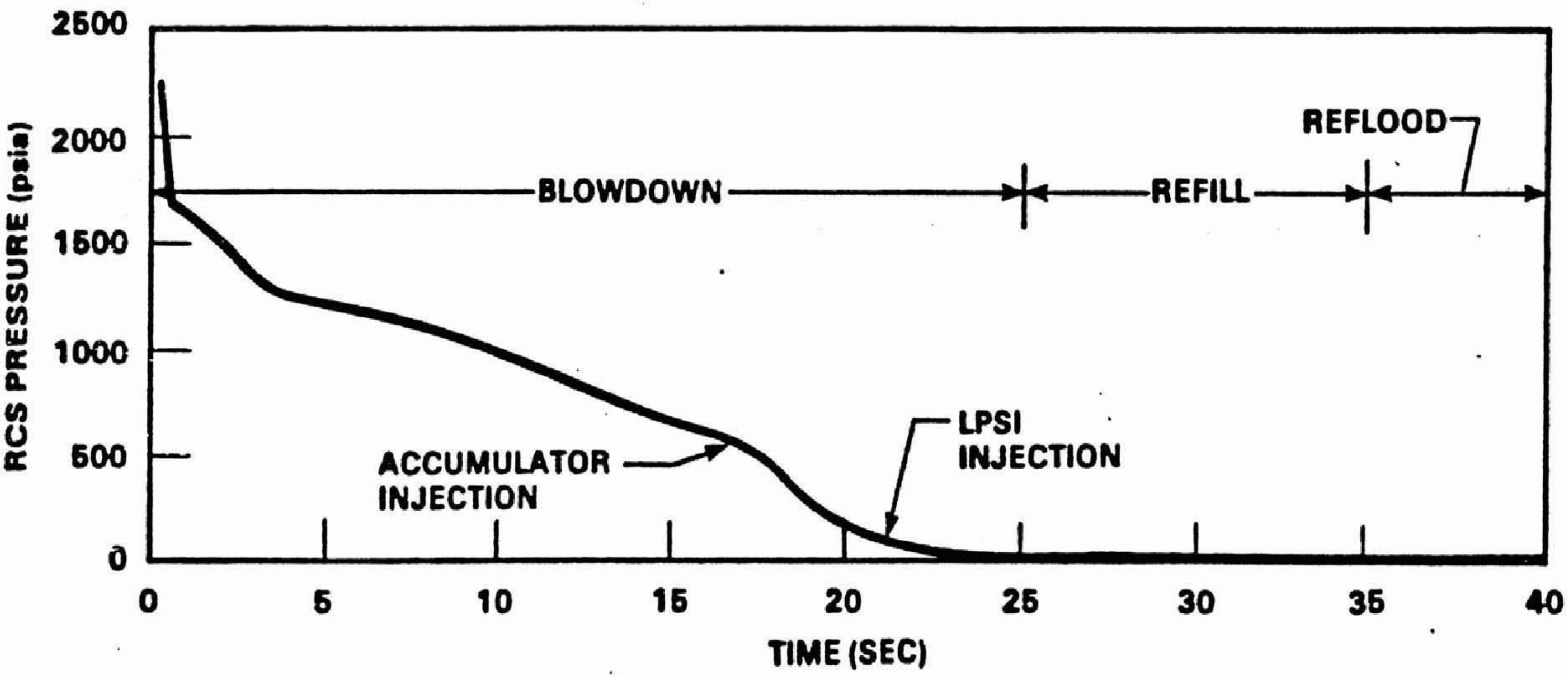
Figure 4.19 in Collier, J. G., and G. F. Hewitt. *Introduction to Nuclear Power*. Washington, DC: Hemisphere Publishing, 1987.

Schematic calculated fuel clad temperatures for a PWR LOCA

PWR Large Break LOCA Phases

Phase	Time(s)
Bypass	20-30
Refill	30-40
Reflood	40-250
Long term cooling	250-

LOCA Transient Periods



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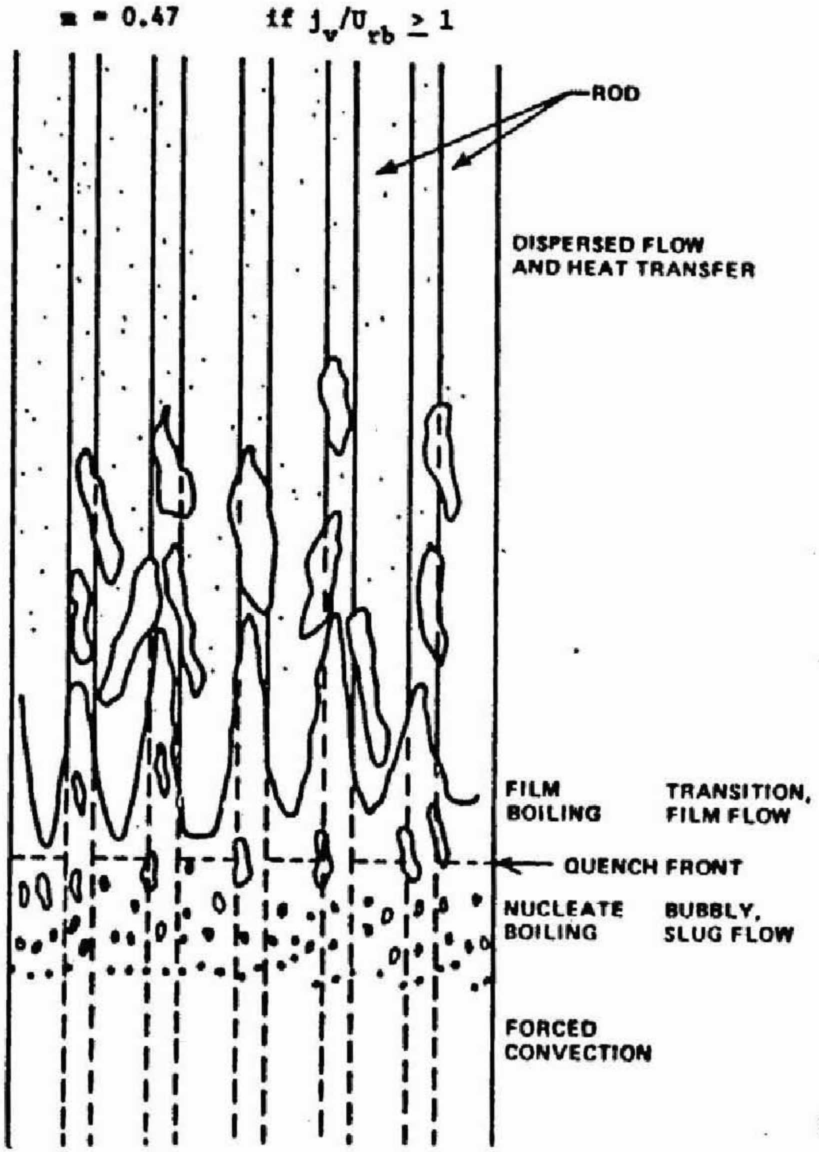
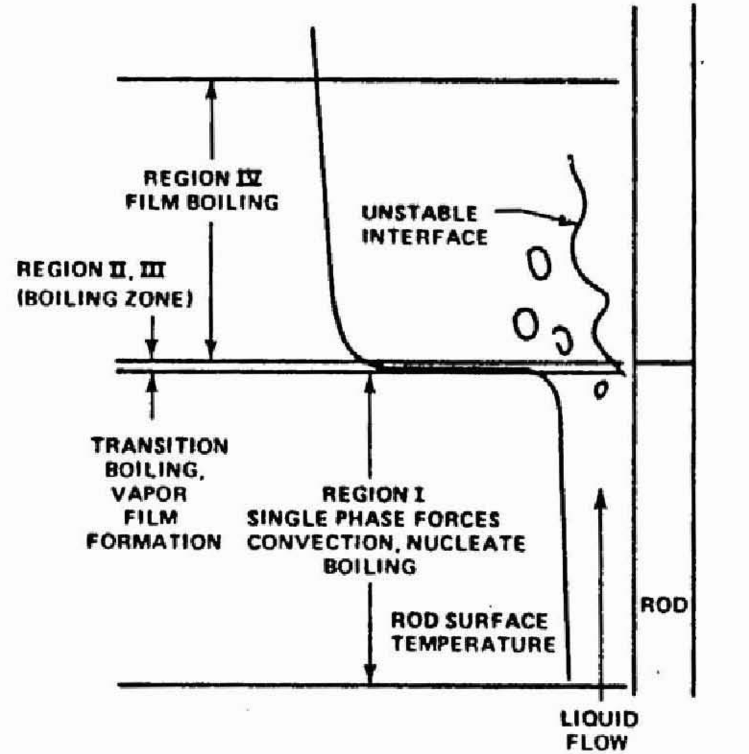
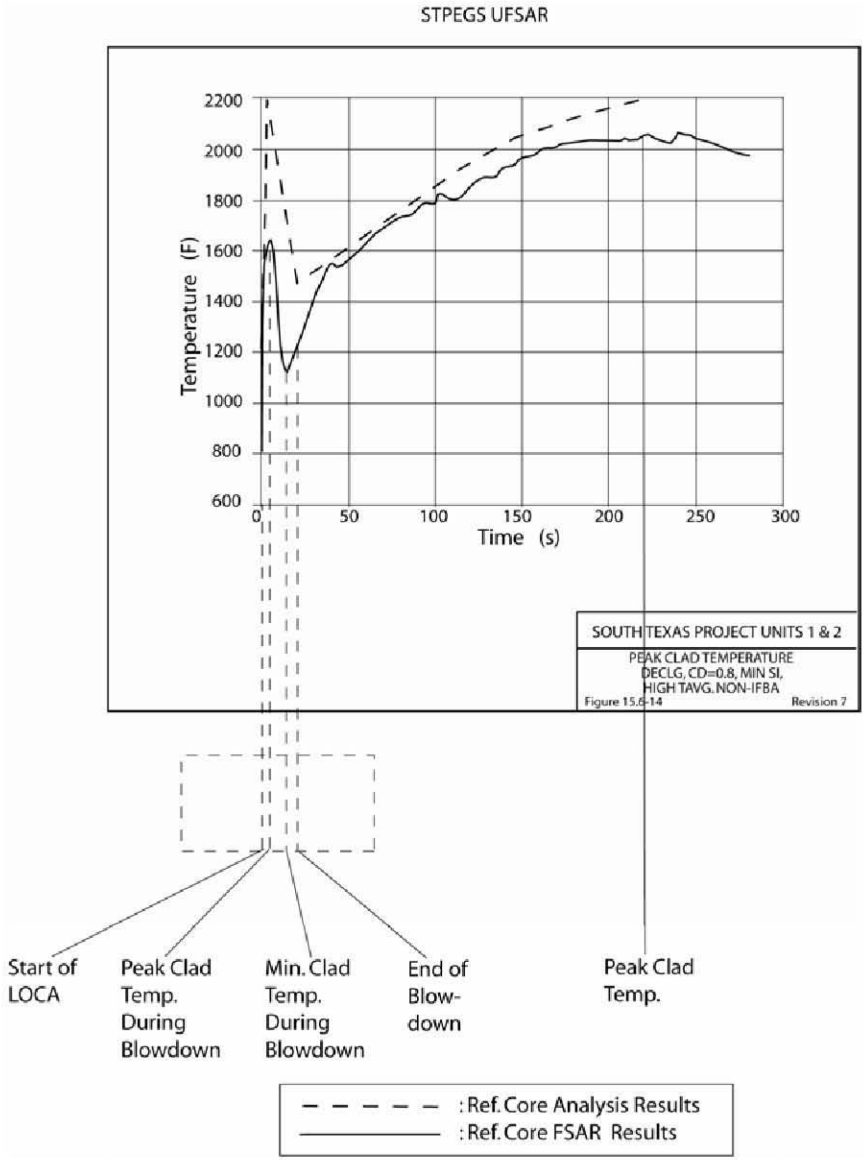
Typical Conditions in Rod Bundle During Reflood





Hydrodynamic and Heat Transfer Regions Near Quench Front

South Texas FSAR Limiting Clad Temperature for LBLOCA



Source: Shuffler, C., J. Trant, N.E. Todreas, and A. Romano, *Thermal Hydraulic and Economic Analysis of Grid-Supported, Hydride-Fueled PWRs* (MIT-NFC-PR-077). Cambridge, MA: MIT CANES, January 2006

Analytic Estimate of PWR LBLOCA PCT

(Catton, I. et al, "Quantifying Reactor Safety Margines, Part 6, A Physically Based Method of Estimating PWR LBLOCA PCT," *NED* 119 (1990) 109-117)

T_{PCT} =	T_i + Blowdown Heating from Decay heat & Stored energy	- Blowdown convective cooling	+ Refill decay heatup	+ Reflood decay heatup and cooling	
=	$\underbrace{580 + 570}$	- 164	+ 134	+ (70 – 350) = 840°F	
Components σ_{i95} =	± 364	+ 60 / - 82	+ 16 / - 0	+ 230 / -55	

$$\sigma_{95} = \left(\sum_i^n \sigma_{i95}^2 \right)^{1/2} = 435°F/377°F$$

Mean T_{PCT} = 840°F ; 95th T_{PCT} = 1217 to 1275°F

Assumptions: $q_i^1 = 9.5$ kw/ft

Reflood rate = 4 inches/sec

Blowdown peak minus minimum clad temperature times = 20 sec

Source: Shuffler, C., J. Trant, N.E. Todreas, and A. Romano, *Thermal Hydraulic and Economic Analysis of Grid-Supported, Hydride-Fueled PWRs* (MIT-NFC-PR-077). Cambridge, MA: MIT CANES, January 2006

Temperature at which significant phenomena occur during core heat-up

Temperature (^oC)	Significant Phenomenon
350	Approximate cladding temperature during power operation.
800-150	Rod internal gas pressure in the post-accident environment causes cladding to perforate or swell Some fission gases release Solid reactions begin stainless steels and Zircaloy Clad swelling may block some flow channels
1450-1500	Zircaloy steam reaction may produce energy in excess of decay heat Gas absorption embrittles Zircaloy Hydrogen formed Steel alloy melts
1550-1650	Zircaloy-steam reaction may be autocatalytic unless Zircaloy is quenched by immersion.
1900	Zircaloy melts
2150	Increasingly significant fission product release from UO_2
2700	UO_2 and ZrO_2 melt

Figure by MIT OCW. After Hewitt and Collier.

Best Estimate Large Break LOCA Analysis Results

BLOWDOWN	0 sec.	Break occurs
		Reactor trip (pressurizer pressure or high containment pressure)
		Pumped SI signal (pressurizer pressure or high containment pressure)
		Accumulator injection begins
		Pumped ECCS injection begins (offsite power available)
		Containment heat removal system starts (offsite power available)
	20-25 sec.	End of bypass
		End of blowdown
REFILL		Pumped ECCS injection begins (loss of offsite power)
		Containment heat removal system starts (loss of offsite power)
	35-40 sec.	Bottom of core recovery
REFLOOD		Accumulators empty
	5 min.	Core quenched
LONG TERM COOLING		Switch to cold leg recirculation on RWST low level alarm
		Switch to hot leg/cold leg recirculation

BEST ESTIMATE LARGE BREAK LOCA TIME SEQUENCE OF EVENTS

BEST ESTIMATE LARGE BREAK LOCA ANALYSIS RESULTS

Component	Blowdown Peak	First Reflood Peak	Second Reflood Peak
$PCT^{average}$	<1508°F	<1681°F	<1384°F
$PCT^{95\%}$	<1760°F	<1976°F	<1964°F
Maximum Oxidation		<11%	
Total Oxidation		<0.89%	

Research Program to Address Appendix K Conservatisms

Appendix K Requirement	Research Program
Decay heat	ANS Decay Heat Standard Committee ANSI/ANS-5.1-1979
Zirconium-water oxidation	ORNL - Reaction Rate Studies
ECC bypass	BCL 1/15, 2/15 Scale Tests, Creare 1/15 Scale Tests
Post CHF heat transfer	ORNL film boiling tests, semiscale tests, University of Lehigh single tube tests
Reflood heat transfer	Full length emergency core heat transfer (FLECHT) low flooding rate tests, FLECHT-SEASET (separate effects and systems effects test) tests
Steam binding during reflood	FLECHT-SET phase B systems tests, semiscale, FLECHT-SEASET steam generator tests, cylindrical core test facility reflood systems tests
Flow blockage heat transfer at low reflood rates	FLECHT-SEASET 21-rod and 163-rod bundle tests, FEBA tests, NRU rod burst tests
No rewet after CHF until reflood	Semiscale, loss of fluid test (LOFT) systems tests, Westinghouse film boiling tests
Moody break flow model	Marviken full scale critical flow tests, LOFT, semiscale

Small Break LOCA Strategy

1100 Mwe (3400 Mwt) $\Rightarrow$ ~ 200 Mwt immediate decay heat generation

LPSI
But if pressure too high use
HPSI and/or accumulators

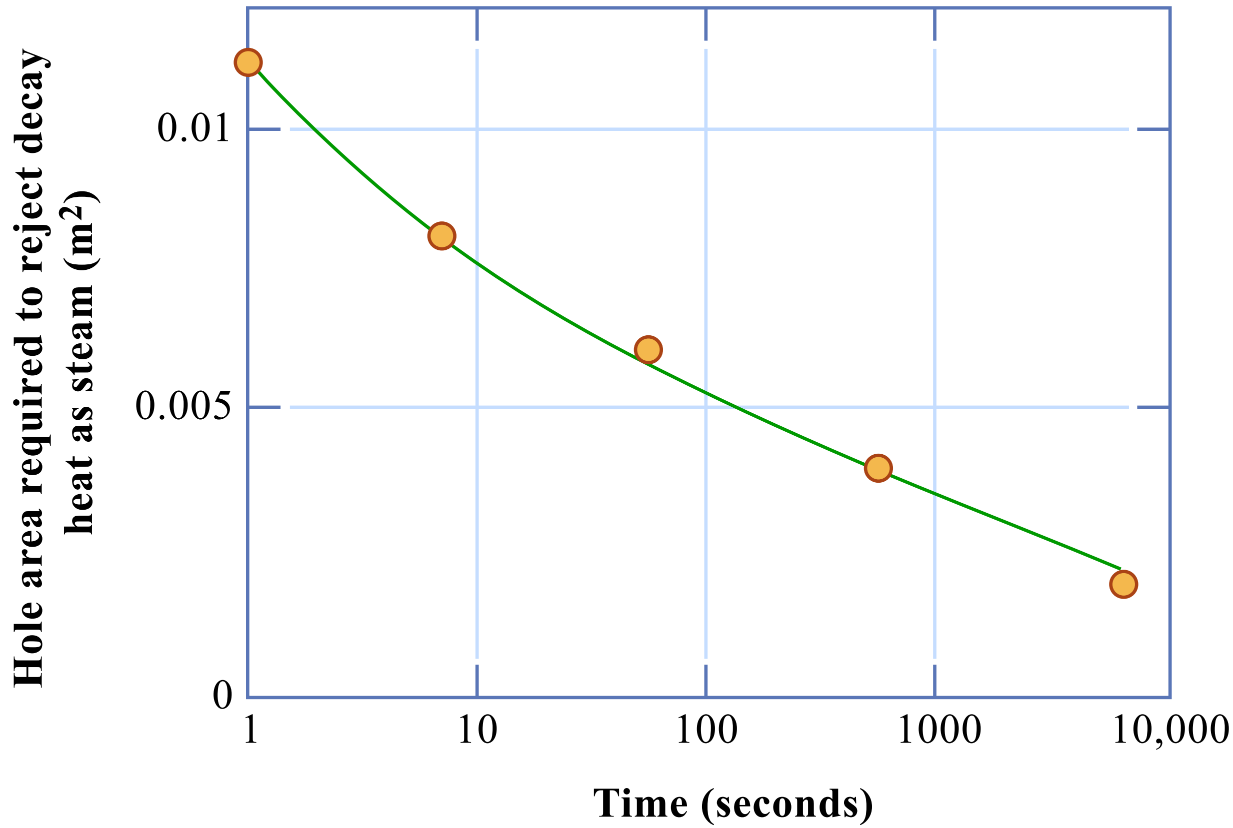
For pressure and inventory control

Steam generator if
Feedwater available

	Steam Discharge	Water Discharge
Mass Flowrate	low	high
Enthalpy Loss	high	low

TMI
- No auxillary feedwater
- PORV stack open

If break too small or water discharged
increase break area
- PORVs – top of pressurizer ($0.002 m^2$/valve)
- Safety relief valves



Hole Size to Remove Decay Heat as Steam

Figure by MIT OCW.

PWR Small Break LOCA Phases

Phase	Time(s)
Initial stage	0-10
Reflux condensation	10-220
Potential for first core uncovery	220-280
Loop seal clearing, potential for core uncovery	280-310
Long term cooling	310-

Primary pressure vs time for small-break LOCAs in a PWR. (○) Primary temperature 175°C; (□) reflood tank empty. (Two HPI pumps; reflood tanks 4 x 286 m^3; no LPI pumps)

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Figure 4.22 in Collier, J. G., and G. F. Hewitt. *Introduction to Nuclear Power*.
Washington, DC: Hemisphere Publishing, 1987.

Small-break LOCA; mixture level and clad temperatures

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Figure 4.27 in Collier, J. G., and G. F. Hewitt. *Introduction to Nuclear Power*. Washington, DC: Hemisphere Publishing, 1987.

Notrump Transient Results

Event Time (sec)	2 Inch	3 Inch	4 Inch	6 Inch
Break Initiation	0	0	0	0
Reactor Trip Signal	42.5	16.2	9.41	6.07
S-Signal	56.8	28.0	20.3	13.9
SI Delivered	83.8	55.0	47.3	40.6
Loop Seal Clearing*	960	416	210	42.5
Core Uncovery	1468	700	436	181.5
Accumulator Injection	N/A	1246	609	259.9
RWST Empty Time	3910	3878	N/A	N/A
PCT Time	2822.5	1482	764.2	317.7
Core Recovery**	> 6000	> 5000	2828	382.2

* Loop seal clearing is defined as break vapor flow > 1 lb/s

** For the 2 and 3 inch cases, where core recovery is > TMAX, basis for transient termination can be concluded based on the following arguments: (1) The RCS system pressure is decreasing which will increase SI flow, (2) Total RCS system mass is increasing, (3) Core mixture level has begun to increase and is expected to continue for the remainder of the accident

Beginning of Life (BOL) Rod Heatup Results

	2 Inch	3 Inch	4 Inch	6 Inch
PCT (°F)	1276	1829	1531	1418
PCT Time (s)	2822.5	1482.0	764.2	317.7
PCT Elevation (ft)	11.5	11.75	11.5	10.75
Burst Time (s)	N/A	N/A	N/A	N/A
Burst Elevation (ft)	N/A	N/A	N/A	N/A
Max. Local ZrO_2 (%)	0.23	3.98	0.49	0.16
Max. Local ZrO_2 Elev (ft)	11.5	11.75	11.5	11.0
Core-Wide Avg. ZrO_2 (%)	0.03	0.53	0.08	0.03