NUCLEAR SAFETY AND RELIABILITY

WEEK 8

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Loss of Primary Coolant Analysis

The objective of this analysis is to calculate the radiation dose to the public, given a fully defined accident sequence. Loss of coolant accidents (LOCA) are chosen for examination because (a) failure of the primary pressure boundary is a necessary part of any accident sequence which might result in radiation doses to the public due to radioactive material release from fuel, (b) all accident sequences are analyzed to show the level of protection against radiation doses to the public, so that they include LOCA implicitly or explicitly as an event tree branch at some level of probability, and (c) LOCA accident sequences of various kinds are useful to test many of the capabilities of the plant safety systems.

The major components of a LOCA analysis are:

(1) Plant analysis
(2) Containment analysis
(3) Radioactive materials dispersion and dose

(1) Plant analysis

The key questions that must be answered are:

1. Where are the potential leaks or breaks in the HT, and how large might they be?
2. Given a coolant leak, what is the coolant discharge rate as a function of time?
3. What is the voiding rate in the core, and the resulting reactivity as a function of time?
4. When is the reactor tripped by SDS1 or SDS2, and what is the subsequent power rundown as a function of time?
5. What is the coolant flow and inventory distribution around the HT system?
6. How much stored heat is removed from the fuel during blowdown?
7. When is the emergency coolant injection (ECI) flow initiated, and what is the flowrate as a function of time?
8. At what time do the headers, feeders, and channels refill with water?
9. How is the decay heat removed in the long term?
10. What is the reliability of shutdown, emergency cooling, and safety support systems (power, instrument air, circulating cooling water) under LOCA conditions?

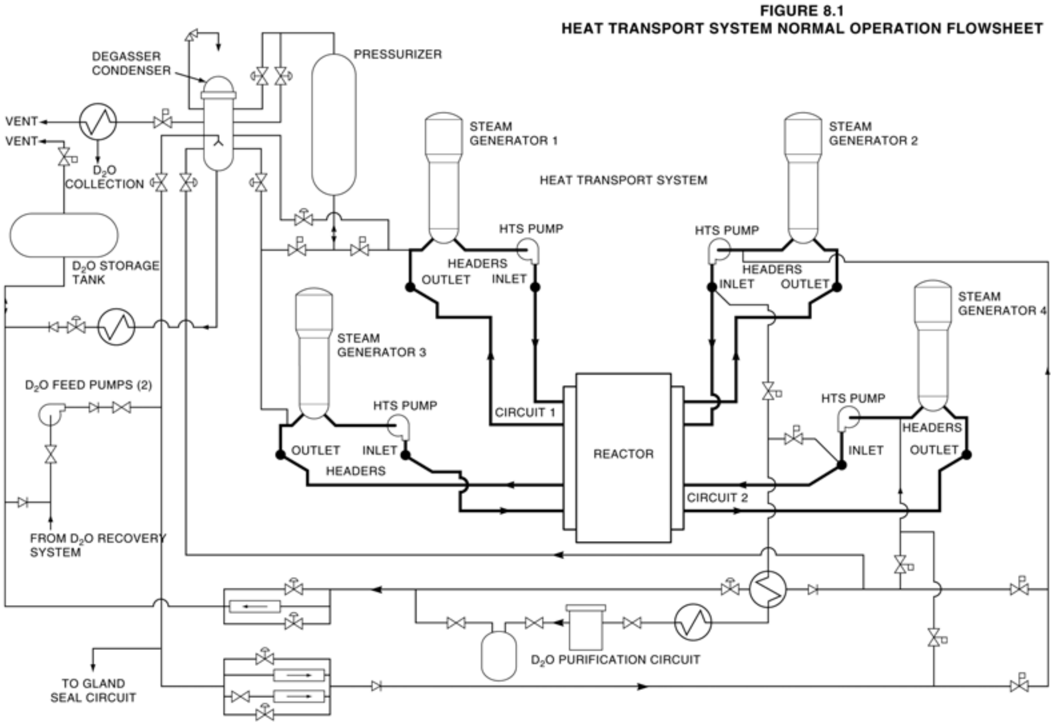
11. If ECI fails after a LOCA, what is the subsequent behavior of fuel channels, and the amount of radioactive material released from fuel to containment?
12. Is there hydrogen produced during the accident sequence due to Zr-water reaction? Can it reach flammable concentrations?
13. Can jet or broken-pipe reaction forces damage containment or other systems in containment?

Many other questions arise in the course of the analysis. In the course of the safety investigation, the designer or operator must address each of these questions.

Potential Leaks or Breaks

Figures 8.1 to 8.7 show the geometry of HT piping in the 600 MWe CANDU.

FIGURE 8.1
HEAT TRANSPORT SYSTEM NORMAL OPERATION FLOWSHEET



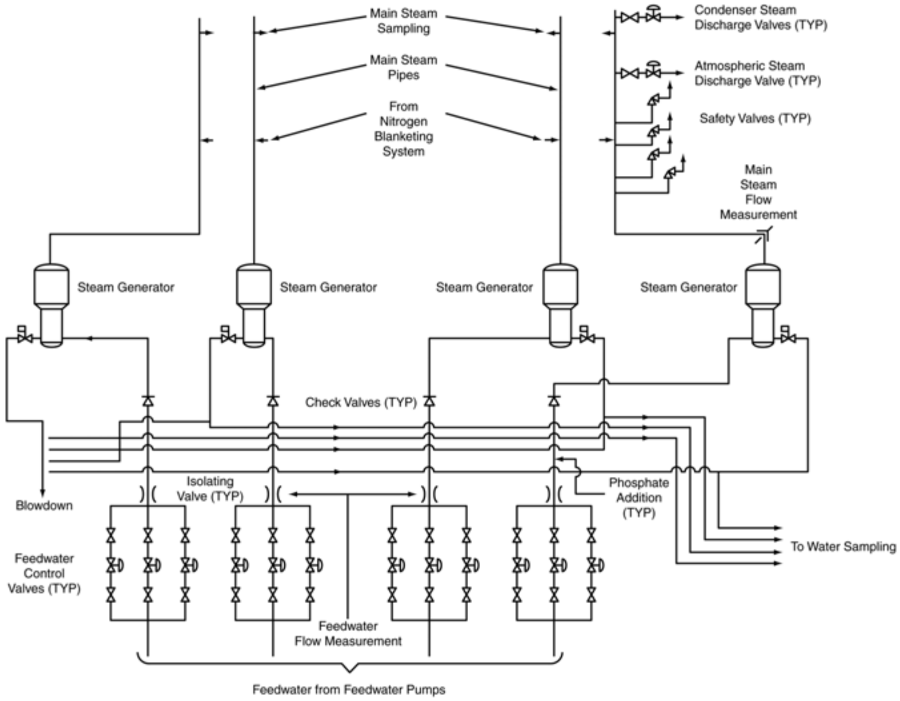


FIGURE 8.2
STEAM AND FEEDWATER SYSTEMS

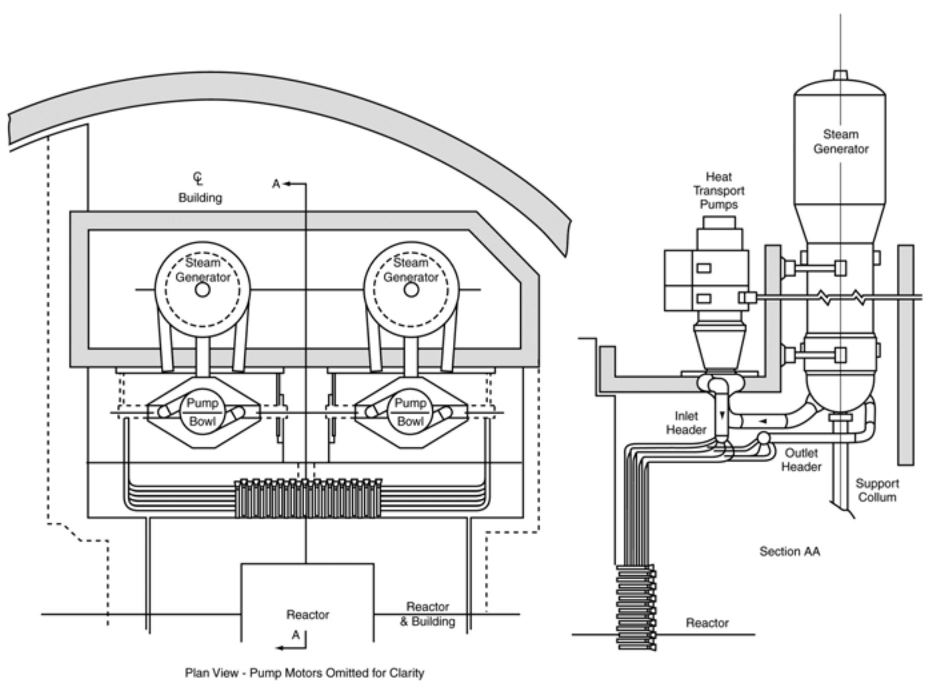


FIGURE 8.3
HEAT TRANSPORT SYSTEM ARRANGEMENT

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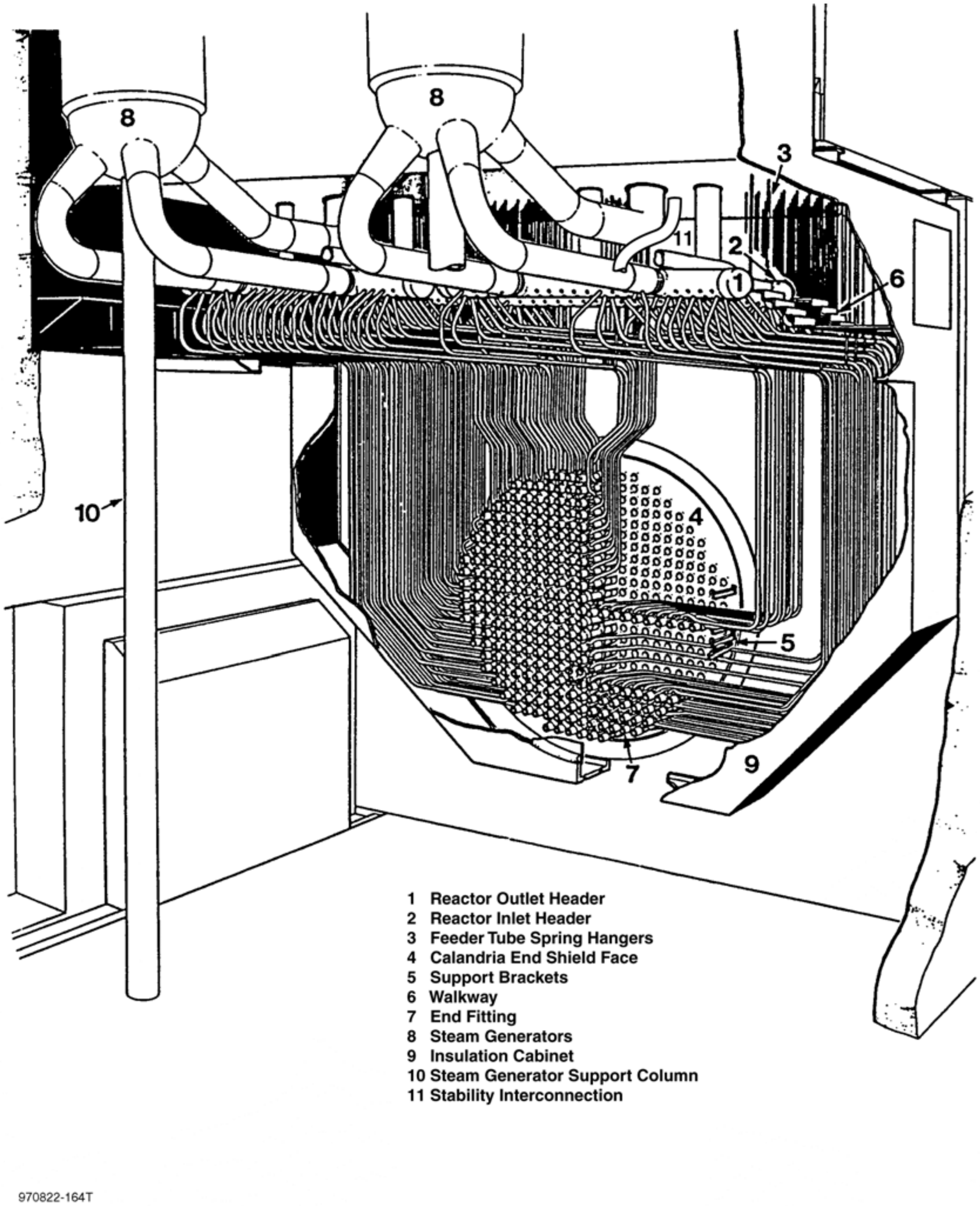

FIGURE 8.4 HEAT TRANSPORT SYSTEM TYPICAL FEEDER ARRANGEMENT

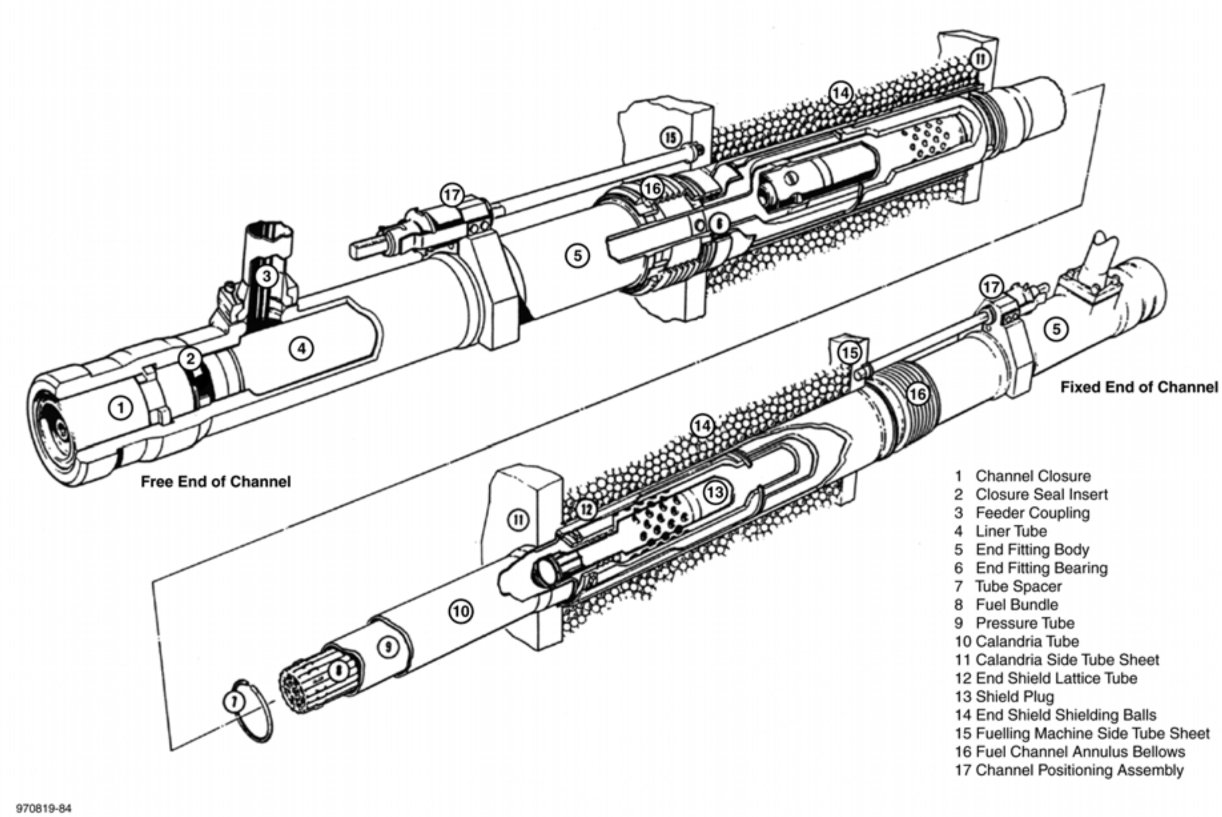

FIGURE 8.5 FUEL CHANNEL ASSEMBLY

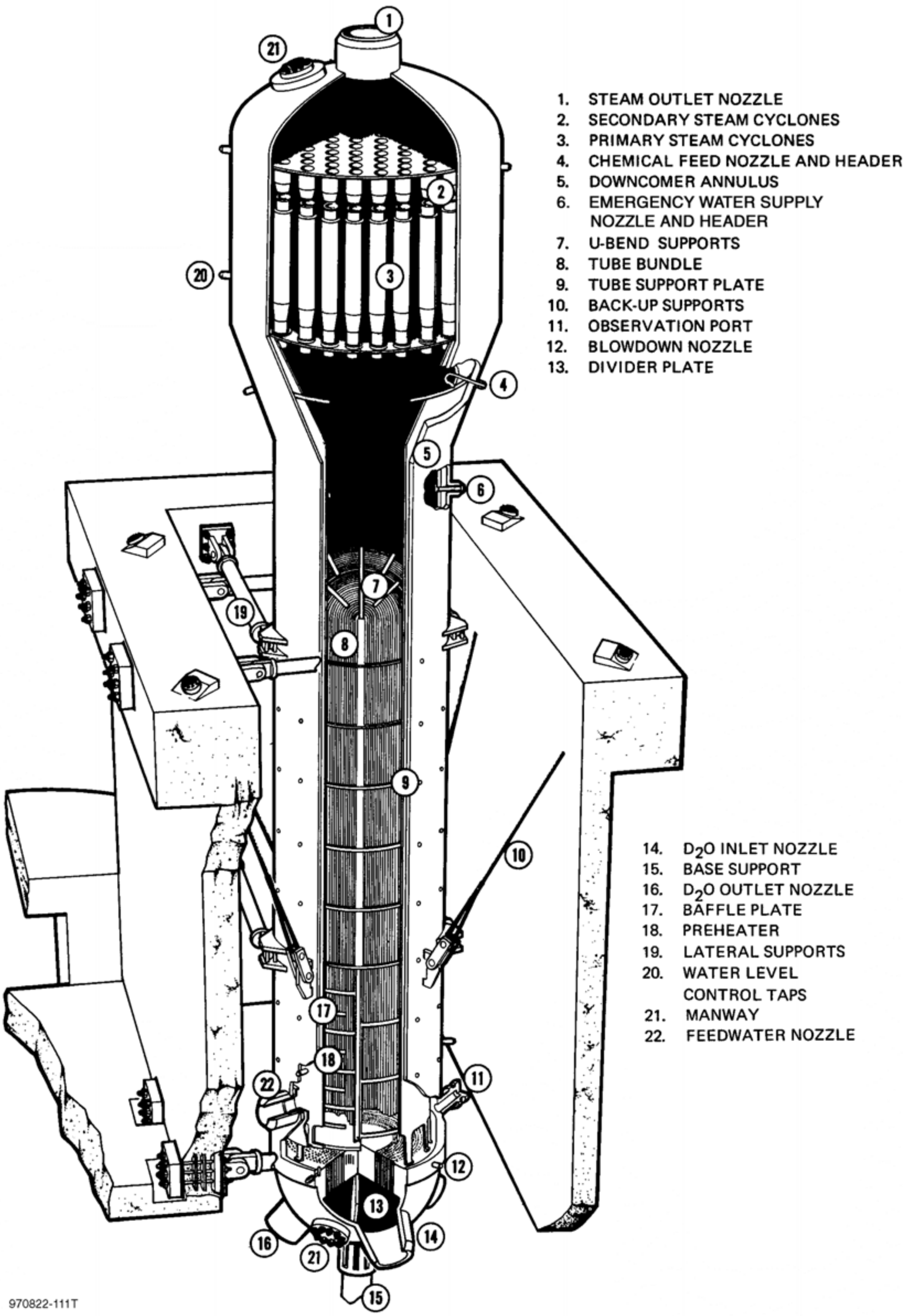

FIGURE 8.6 STEAM GENERATOR FOR 600 MW(e) N.P.S.

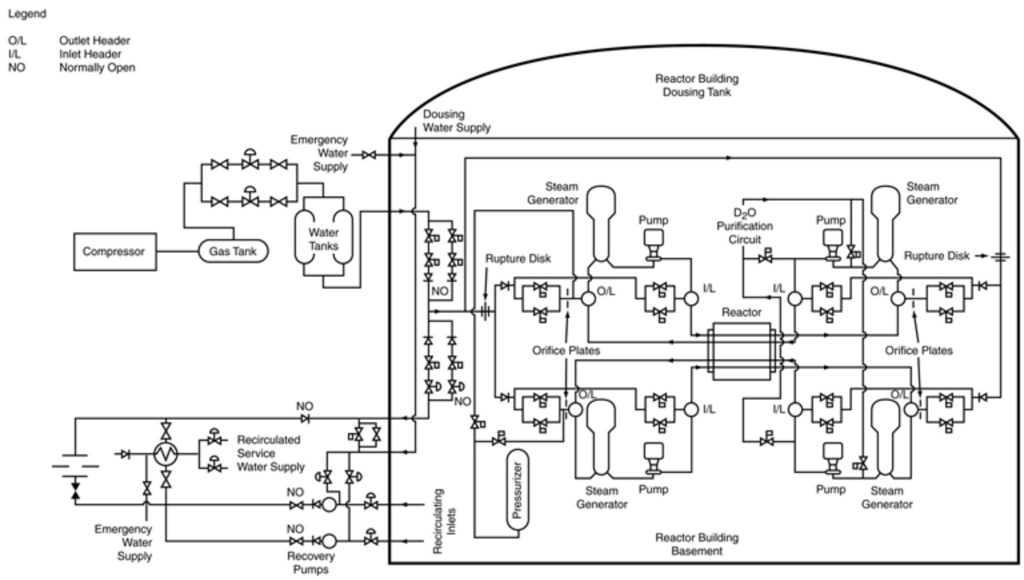

FIGURE 8.7
EMERGENCY CORE COOLING SYSTEM

Potential leak pathways for HT coolant escape are:

(a) Feed/bleed/relief systems - these are active systems which control inventory and pressure in the HT system. Malfunctions could result in coolant loss through control or relief valves associated with the degasser/condenser or through rupture of small lines.

(b) Fueling machines - fueling operations take place about twice a day; during the sequence the HT boundary must be opened at two locations and then resealed.

(c) Pump seals - HT water escapes steadily, at a low rate, through each HT pump shaft seal. There are three seals in series. Massive seal failure (possibly initiated by shaft vibration) would result in a LOCA.

(d) Instrument tube rupture - small diameter, susceptible to vibration.

(e) Steam generator tubes - there are thousands of tubes in each SG; leakage is detected by presence of heavy water in secondary side light water. Several tubes would have to fail simultaneously to produce significant HT water loss, but resulting LOCA would breach the containment boundary. Also, ECI water would not return to the recovery sump.

(f) Pressure tube failure (without calandria tube rupture) - discharge through end fitting bearing spaces ruptures annulus gas bellows.

(g) Feeder pipes or couplings - there are 760 feeders and Grayloc couplings in each unit. Each is subjected to thermal stress and fluid flow vibration. An inlet break can lead to flow stagnation.

(h) End fittings - pressure tube guillotine rupture would result in end load being carried by the yoke on end fitting. Failure of this yoke would eject end fitting and fuel into the vault. Closure plug failure also would eject fuel.

(i) Pressure tube and calandria tube failure - discharge from HT system is into calandria tank -- potential for tank overpressure, reaction forces damaging shutoff rod guide tubes and neighboring channels. Possible fuel discharge into calandria.

(j) Headers and large HT piping, pump casings - largest HT breaks considered in licensing analysis; potential for damage of adjacent equipment or impairment of containment integrity.

(k) Emergency coolant injection lines - penetrate containment boundary, but have multiple isolation valves in series. Would result in LOCA outside containment, with no return of ECI water for long-term makeup.

Potential Causes of LOCA

The following are listed as immediate causes. The root cause of any accident can be traced to human failure of some kind in management, design, manufacture, construction, or operation.

i) Control Malfunction - pressure and inventory control, fueling machine, reactor control, heat removal control (leading to overpressure rupture) or manual intervention.

ii) Material Degradation - material flaws, either "built in" or due to aging (erosion, corrosion, wear, fatigue) of components. For example -- pressure tubes, steam generator tubes, weld areas in HT piping, piping materials, valves, and seals.

iii) Unusual Stresses - thermal (flow oscillations, flow blockage), mechanical (F/M loads applied to end fittings, pipe hanger restrictions), hydraulic (pump cavitation, ECI injection) or external (earthquake, explosion).

LOCA Classifications

By Location: i) Inside core - pressure tube+calandria tube
ii) Inside reactor vault or boiler room - P/T and annulus gas bellows, piping, feeders, end fittings, instrument tubes, relief valves, etc.
iii) Outside containment (SG tubes, ECI lines)

By Size:	i)	$<$	D_2O feed system capacity
	ii)	$\leq$	largest feeder pipe
	iii)	$\leq$	largest HT pipe

Each of these leak classifications introduces some unique challenges to safety systems capability. Figure 8.8 shows typical total length distributions by piping size for a CANDU unit.

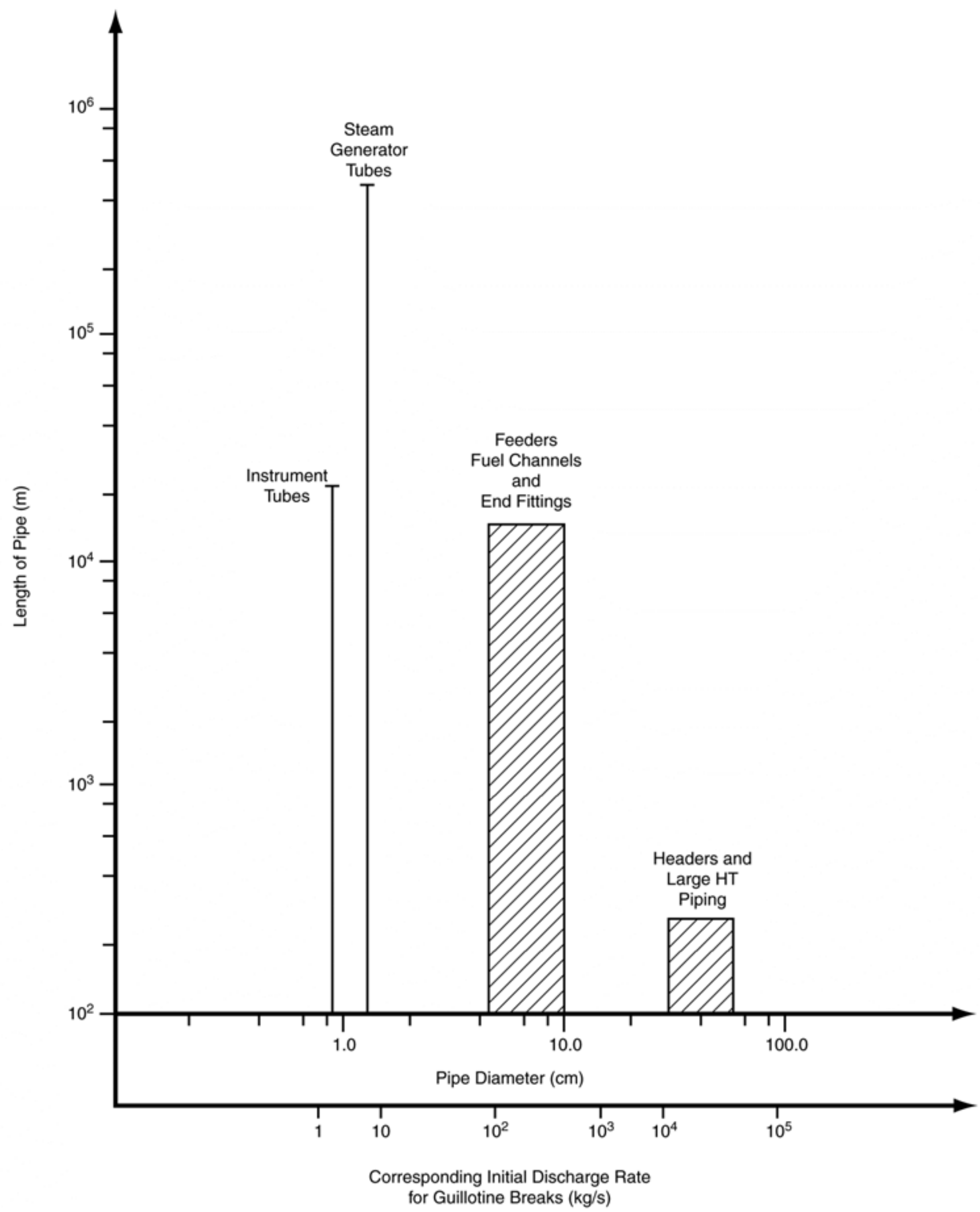

FIGURE 8.8
APPROXIMATE LENGTH OF VARIOUS DIAMETER PIPES
IN HT SYSTEM OF ONE REACTOR UNIT

If one presumes that the break probability is proportional to the total piping length (a somewhat questionable assumption) it can be seen that the occurrence frequency of small LOCA is expected to be much larger than that of large LOCA. Single instrument tube or SG tube failures result in discharge rates which are much less than the flow capability of the HT feed system at full pressure, which is about 35 kg/s for two feed pumps. Rupture of the largest feeder pipe results in a discharge rate much lower than the flow capacity of the ECI system at its normal delivery pressure of around 5 MPa. At this pressure the HT pumps can be operated in low temperature conditions without cavitation at their inlet. Very large breaks result in rapid system depressurization because their discharge rate is far beyond the flow capacity of the ECI system.

Components of LOCA Analysis

The major sub-models required in LOCA analysis are shown in Figure 8.9. Each of these sub-models involves one or more computer codes of varying complexity and sophistication. Each of these codes must be verified against plant data or experiments. The type of approximation used is dictated by the use for which the calculation is intended and the detailed characteristics of the transient being simulated.

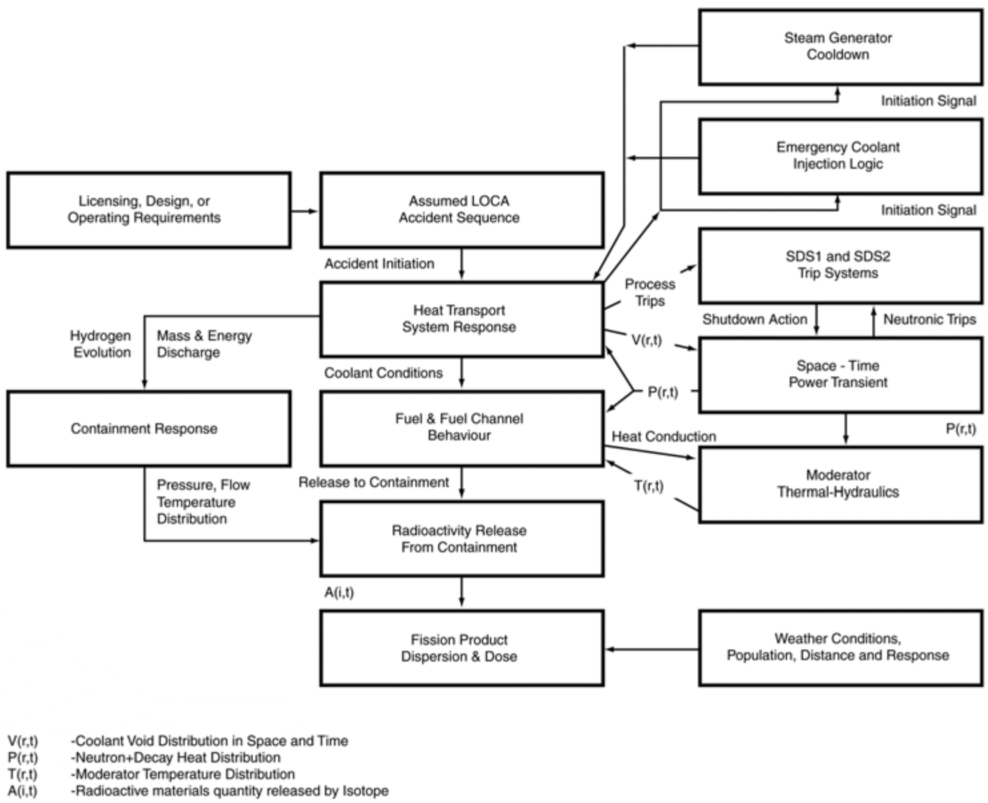

FIGURE 8.9
COMPONENTS OF LOCA ANALYSIS

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Assumptions: These are determined by the requirements of design, operator information, or plant licensing. The information requirements for design are closely related to eventual operating license requirements because the design must be satisfactory to licensing authorities. Within limits, virtually any level of safety can be designed for (not necessarily achieved), but the cost rises rapidly if requirements are excessive. Usually, licensing calculations must be conservative rather than realistic; the results often are not closely related to what one would expect to happen under the defined accident conditions. Their purpose is merely to satisfy regulatory authorities that public protection is assured during plant operation. In contrast, operations requirements are strictly for realistic estimates - transient conditions which the plant operators might expect to experience, and for which they must be trained. The two sets of calculations might produce entirely different results, and must never be confused one for the other.

Once the assumptions are decided, the accident sequence is prepared as input to computer codes used in the analysis.

Heat Transport System Response: The assumed leak initiates an inventory and pressure transient in the HT system. Pressure and flow transients are detected both by the plant process control systems and the process trip parameters on SDS1 and SDS2. Coolant voiding in the reactor produces reactivity changes which increase reactor power and change its spatial distribution. Control and shutdown actions also influence the power level and distribution. Calculation of power changes requires 3-dimensional neutronics models; the neutronic equations are coupled non-linearly to those describing fluid flow in the HT system. Fortunately, the coupling between neutronic transients and coolant behavior is weak, so the equations can be decoupled.

Emergency coolant injection and steam generator cooldown are initiated by off-normal process parameters. Injection and cooldown influence the HT system response by altering coolant inventory, temperature, and heat removal rate.

Fuel and Fuel Channel Behaviour: Fuel behavior is influenced primarily by the net heat input to fuel during the transient (stored + decay heat - heat removed by the discharging steam-water mixture and emergency coolant). Fuel burnup prior to the accident determines the total inventory enclosed inside the fuel sheath. Calculated sheath temperature, stress, and strain conditions of each bundle are compared with sheath failure criteria measured experimentally. Post-failure fission product releases also are determined from experimental measurements. Under some severe accident conditions bundle slumping, heat transfer to the pressure tube, pressure tube distortion, and heat conduction through the calandria tube to moderator water also influence the key release parameter, which is fuel temperature. Entry of cool water from the ECI system can add to releases because of the resultant shattering of fuel particles.

Containment Response: The short term pressure-time dependence of the containment space is determined by the steam-water discharge from the HT system, dousing water flow, and vault cooler heat removal. Containment ventilation pathways are isolated on detection of either high pressure or high radiation level. Hydrogen produced from the zirconium-water reaction enters the containment volume, and can reach combustible concentration under certain severe accident conditions. Hydrogen burning increases containment pressure, which is the driving force for release of airborne radioactive species. Long-term pressure history is determined by the balance of heat production and removal, instrument air inflow, and by activation of the filtered-vent discharge system.

Radioactive Materials Release from Containment: Determined by the timing, amount, and chemical form of each isotope released through the break in the heat transport system, as well as by the pressure history and leakage pathways. The balance between airborne and water-borne inventories is an important quantity, because water-borne isotopes are effectively trapped inside the containment space.

Radioactive Materials Dispersion and Dose: This calculation is determined by the amount, timing, half-life, chemical form, biological uptake, and relative biological effectiveness of each isotope released from containment. Weather conditions (wind, temperature profile, precipitation) and population distribution are important parameters. Dilution within the exclusion zone of 1 km between station and the nearest residence reduces the maximum individual dose. The response of people to the accident (stay-in, evacuation) also is an important factor in total dose accumulation.

Timing: Characteristic time scales of the different phenomena that must be modeled vary over a wide range, from milliseconds in the case of rapid neutron power changes and some fluid flow behavior to months or years in the case of fission product retention in containment. Detection and shutdown take place within 1 second to a few minutes, heat removal from fuel must continue for weeks to months, and containment requirements continue until the recovery and cleanup phases are complete. In the event of a severe accident this could be several years (e.g. Three Mile Island Unit 2). This wide time scale poses a considerable challenge to the analyst - he must ensure accurate prediction of important short-term phenomena within a manageable calculation cost over the long term.

Computer Models Used in Current CANDU Analysis: Table 8.1 lists the major codes now used for LOCA analysis of CANDU stations. Many other supporting codes and models are required.

TABLE 8.1 - Computer Codes Used in CANDU LOCA Analysis

System Modeled	Calculation Model	Equations Solved
Heat transport system response	SOPHT, FIREBIRD CATHENA	1-D transient 2-phase flow and heat transfer homogeneous equilibrium or two-fluid)
Space-time power transient	CERBERUS, SMOKIN	3-D transient neutron flux distribution
Fuel & Fuel Channel Behavior	CHAN-II	3-D radiative, conductive, convective heat transfer in fuel channel
Containment response	PRESCON2, PATRIC, VENT	1-D compressible flow, heat transfer, condensation, evaporation, hydrogen deflagration
Moderator thermal-hydraulics	MODCIR, 2DMOTH	Transient forced and free-convective flow, heat transfer in clandria tank
Radioactive Material Behaviour in Containment	FI SSCON-II	Transport, deposition, resuspension, chemical kinetics in containment
Radioactive Material Dispersion and Dose	CSA-N288.2	Modified Gaussian plume dispersion, deposition, human uptake, cumulative dose

Some further details of these computer models are given below.

Heat Transport System Response: The thermal-hydraulic transient following a LOCA is modeled by the SOPHT code or equivalent. The main sub-models are:

(1) fluid flow network - HT, feed/bleed/relief, secondary side, ECI, break
(2) heat production & transport - fuel, piping, heat exchangers
(3) network components - pumps, tanks, valves
(4) controllers - process controls, trip systems, digital control computer

A number of constitutive equations including the equations of state for water and various heat transfer correlations for the various flow regimes also are included in the model. The steady-state (pre-accident) equations are solved to provide the initial condition for each transient.

(1) Fluid Flow Network

The steam-water mixture is represented by a one-dimensional homogeneous equilibrium model, in which the steam and water phases are assumed to have equal velocity and temperature. The piping is segmented into a number of nodes. These nodes are coupled by links, as shown in the following diagram. Mass and energy conservation equations are written for the nodes, and the momentum equation is written for the links.

Node i Node j

$$\boxed{\begin{matrix} M_i,\ U_i,\ P_i \\ \rho_{i,}\ A_i,\ Q_i \end{matrix}} \xrightarrow{\text{Link k}} \boxed{\begin{matrix} M_j,\ U_j,\ P_j \\ \rho_j,\ A_j,\ Q_j \end{matrix}}$$

$W_k,\ A_k,\ L_k$

$D_k,\ K_k,\ \phi^2\Omega_k$

Mass: $$\frac{d}{dt}(M_i) = \sum_{v,i\rightarrow} W_v - \sum_{v,\rightarrow i} W_v$$

Energy: $$\frac{d}{dt}(U_i) = \sum_{v,i\rightarrow} W_v h_v - \sum_{v,\ \rightarrow i} W_v$$

Momentum: $$\frac{d}{dt}(W_k) = \frac{A_k}{L_k}\left[P_i - P_j - \left\{\frac{fL_k}{D_k} + K_k\right\}\phi\Omega_k \frac{|W_k|W_k}{2\rho_{I_k}A_k^2} - g\Delta H_k\rho_i\right.$$

$$\sum_{v,\rightarrow i}$$

$$\left. + \frac{W_k^2}{2}\left\{\frac{1}{A_i} + \frac{1}{A_j}\right\}\left\{\frac{1}{A_i\rho_i} - \frac{1}{A_j\rho_j}\right\} + \Delta P_{pump}\ \right]$$

Definitions:

A_k	=	flow area of link k
D_k	=	equivalent diameter of link k
F	=	Darcy friction factor
G	=	gravitational acceleration constant
ΔH_k	=	elevation change of link k – up is positive direction
h_v	=	specific enthalpy entering node i
h^*_v	=	specific enthalpy leaving node i
K_k	=	fitting losses
L_k	=	length of link k
M_i	=	total mass in node i
P_i, P_j	=	pressure in node i, j
ΔP_{pump}	=	pressure change due to pump
Q_i	=	energy entering fluid in node i (due to heat transfer)
U_i	=	total internal energy in node i
W_k	=	mass flowrate in link k
ρ_i	=	average density in node I
ρ_{li}	=	liquid density in node i
ρ_i'	=	effective density
	=	$\alpha\rho_v + (1-\alpha)\rho_l$ for homogenous flow model
	=	$\left[\frac{\chi^2}{\alpha\rho_v} + \frac{(1-\chi)^2}{(1-\alpha)\rho_l}\right]^{-1}$ for slip flow model
α	=	void fraction
χ	=	flow quality
$\phi^2\Omega_k$	=	two-phase flow pressure drop multipliers
$\sum\limits_{v,i\rightarrow}$	=	sum over all links that initiate from node I
$\sum\limits_{v,\rightarrow i}$	=	sum over all links that terminate at node i

The mass, energy, and momentum equations are coupled through the water equation of state:

$$P=S(M,U)$$

The time derivatives are replaced by their implicit difference form. They are solved at each time step using a block iterative inversion procedure on the matrix that represents the finite-difference form of the equations. Figures 8.10, 8.11, and 8.12 show typical node-link layouts used in CANDU LOCA analysis.

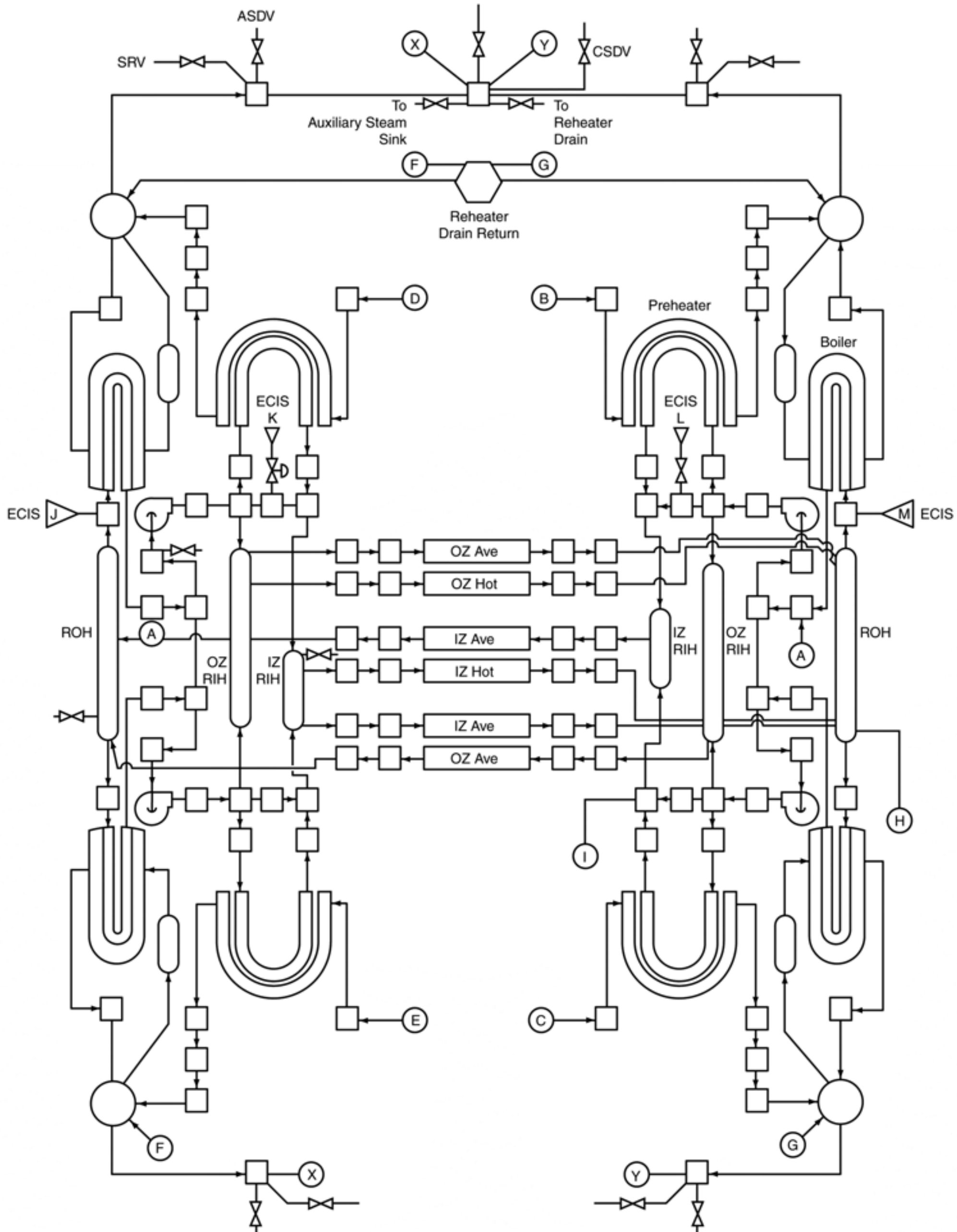


FIGURE 8.10
SOPHT NODE-LINK DIAGRAM FOR BRUCE B HEAT TRANSPORT SYSTEM AND SECONDARY SIDE

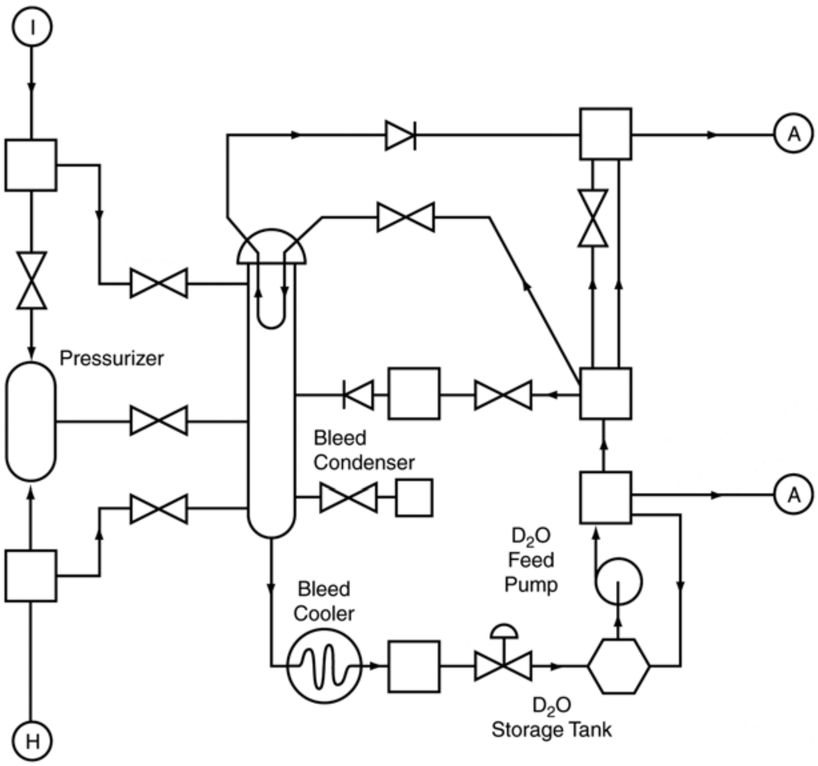


FIGURE 8.11
HEAT TRANSPORT PRESSURE AND INVENTORY CONTROL SYSTEM NODE-LINK DIAGRAM

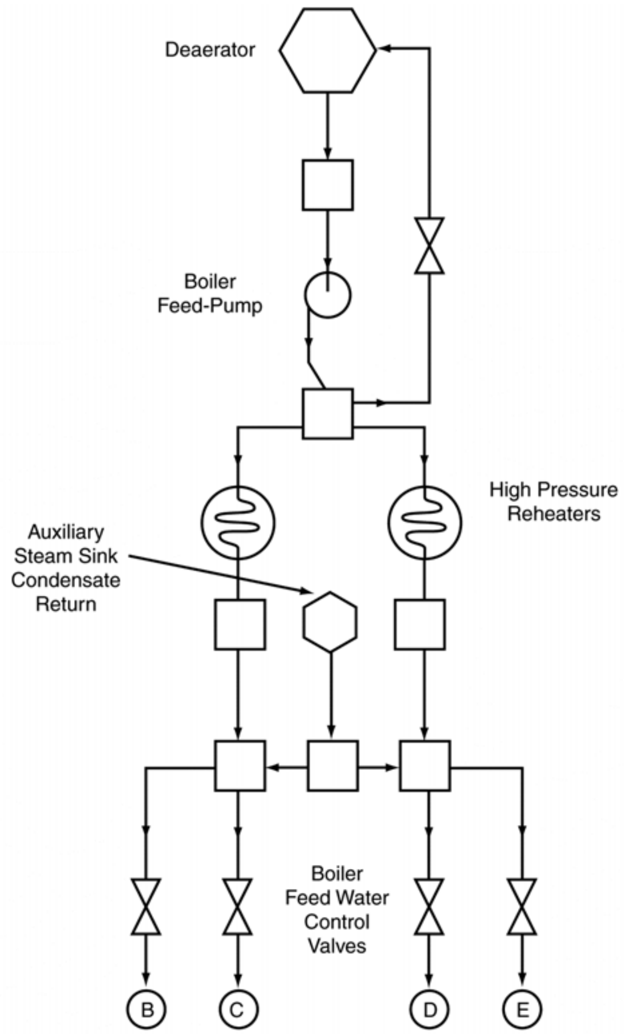
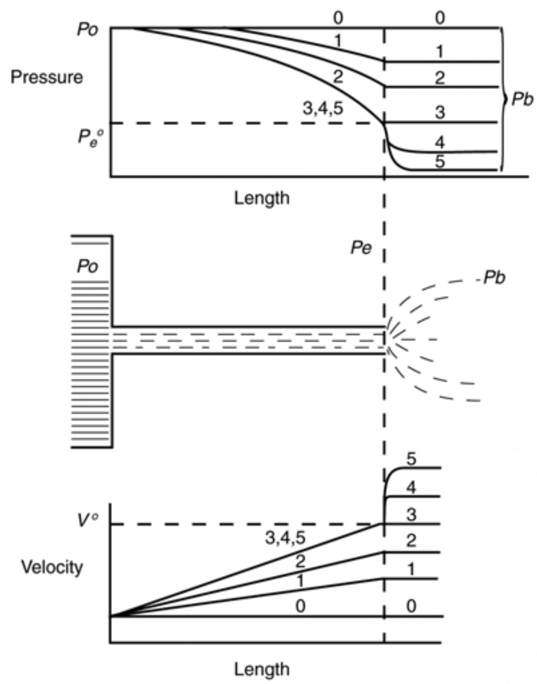


FIGURE 8.12
FEEDWATER TRAIN NODE-LINK DIAGRAM

The break-discharge link is modeled as an orifice passing either single-phase or two-phase critical flow. Figures 8.13, 8.14, and 8.15 illustrate some characteristics of critical flow; basically, the rate of discharge is limited to the sonic velocity at the exit or "choking" plane; the depressurization wave travels upstream at sonic velocity, so an abrupt pressure decrease occurs at the exit plane that limits the flow.



CRITICAL FLOW MODEL

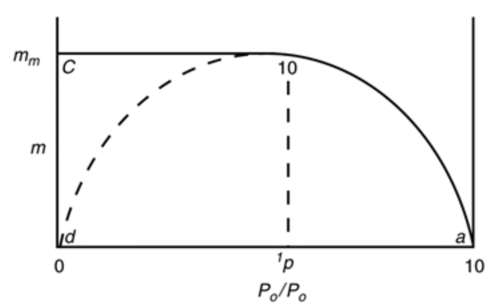
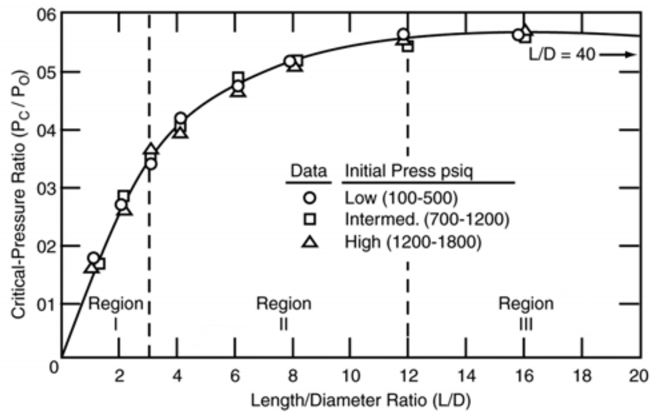


FIGURE 8.13
EFFECT OF PRESSURE RATIO ON MASS-FLOW RATE



EXPERIMENTAL CRITICAL PRESSURE RATIO DATA AS A FUNCTION OF LENGTH/ DIAMETER RATIO

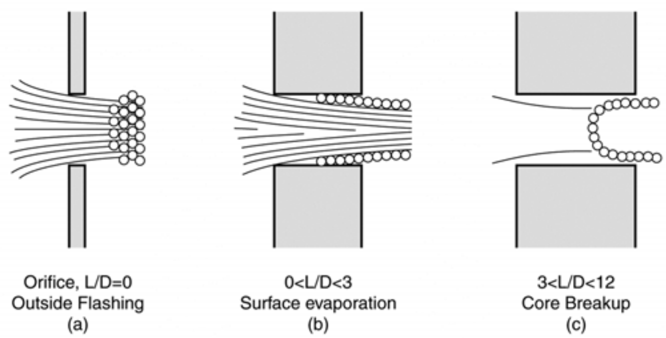


FIGURE 8.14
TWO-PHASE CRITICAL FLOW IN ORIFICES AND SHORT CHANNELS

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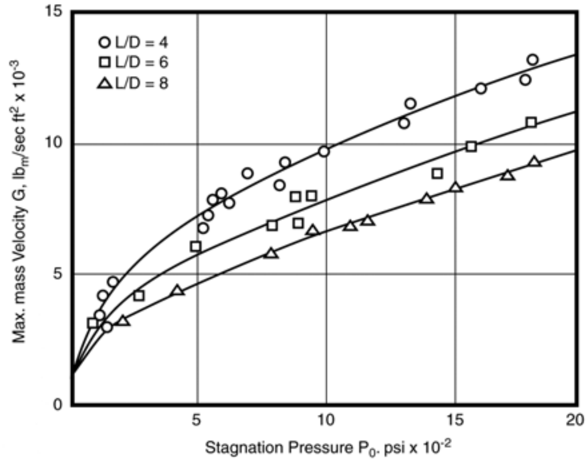

EXPERIMENTAL TWO-PHASE CRITICAL FLOW RATES FOR REGION II

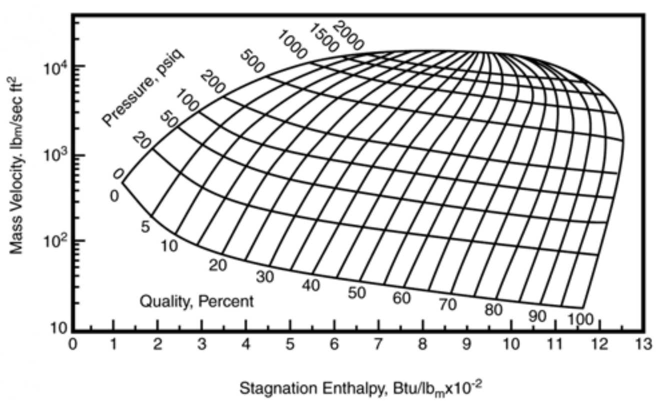

FIGURE 8.15
PREDICTIONS OF CRITICAL STEAM-WATER FLOW RATES WITH SLIP EQUILIBRIUM MODEL

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In spite of the apparent complexity of this model, it is the simplest form that can approximate the actual transient two-phase flow field. The primary strengths of the SOPHT code are (a) its relative simplicity, (b) all controllers important to the HT system response are modeled, and (c) it has been successfully checked against actual plant transients in a number of cases, as well as against a number of LOCA simulation transients in experimental facilities. Its primary disadvantage is that it is incapable of modeling steam-liquid separation which occurs under low-flow conditions in some cases, particularly in horizontal fuel channels.

Figures 8.16 to 8.21 show some typical distributions of liquid, two-phase mixture, and dry steam as a function of time following a large HT pipe break - the term "35% Break" refers to a break with cross-sectional area equal to 35 percent of the total break area of the pipe for a guillotine rupture. This area is twice the physical cross-sectional area of the pipe.

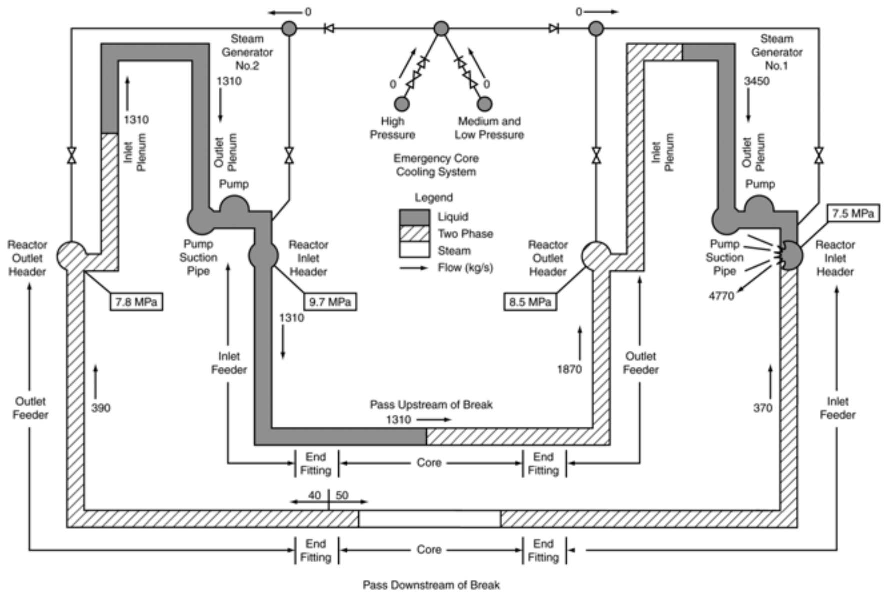

FIGURE 8.16
VOID AND FLOW DISTRIBUTION AT TWO SECONDS AFTER A 35 PERCENT REACTOR INLET HEADER BREAK

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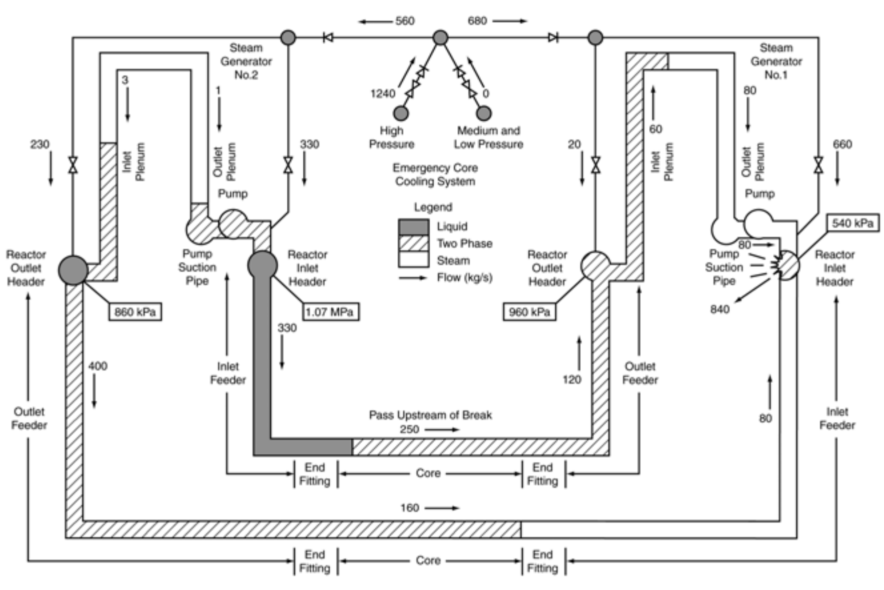

FIGURE 8.17
VOID AND FLOW DISTRIBUTION AT 40 SECONDS AFTER A 35 PERCENT REACTOR INLET HEADER BREAK

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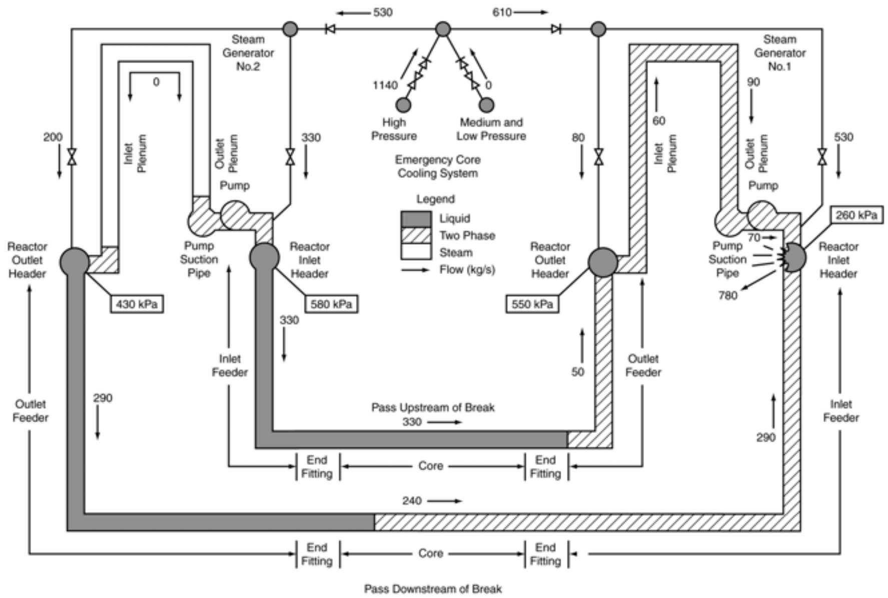

FIGURE 8.18
VOID AND FLOW DISTRIBUTION AT 66 SECONDS AFTER A 35 PERCENT REACTOR INLET HEADER BREAK

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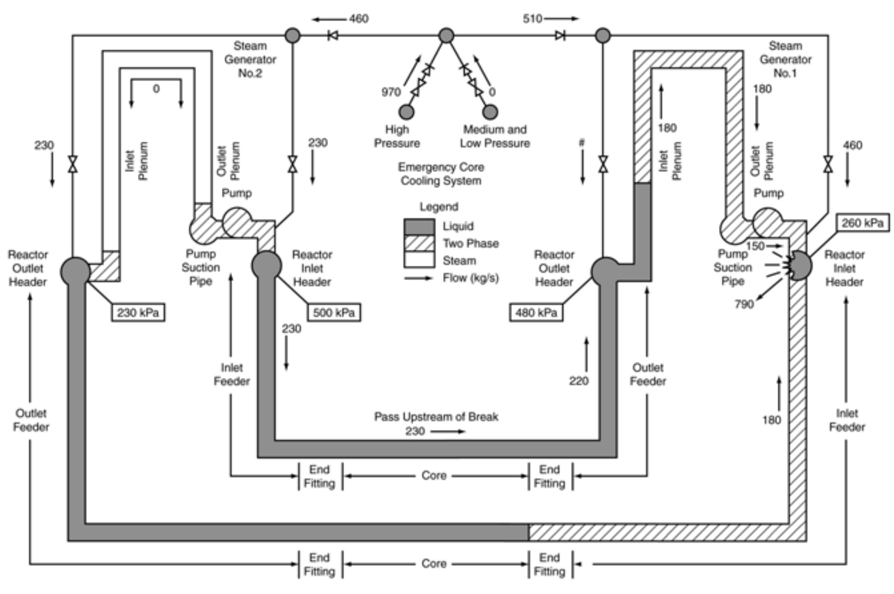

FIGURE 8.19
VOID AND FLOW DISTRIBUTION AT 100 SECONDS AFTER A 35 PERCENT REACTOR INLET HEADER BREAK

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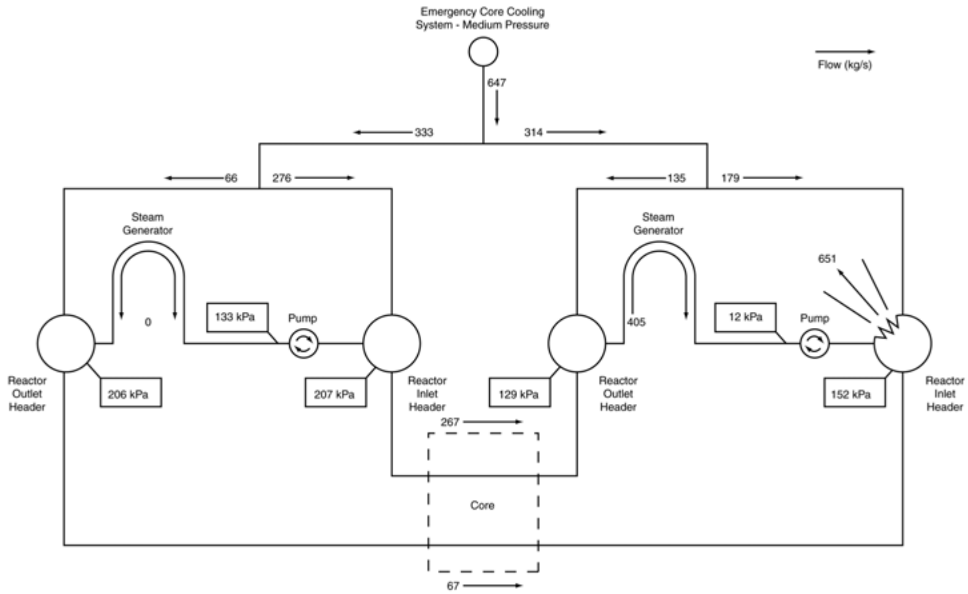

FIGURE 8.20
FLOW DISTRIBUTION AT 315 SECONDS AFTER A 35 PERCENT REACTOR INLET HEADER BREAK

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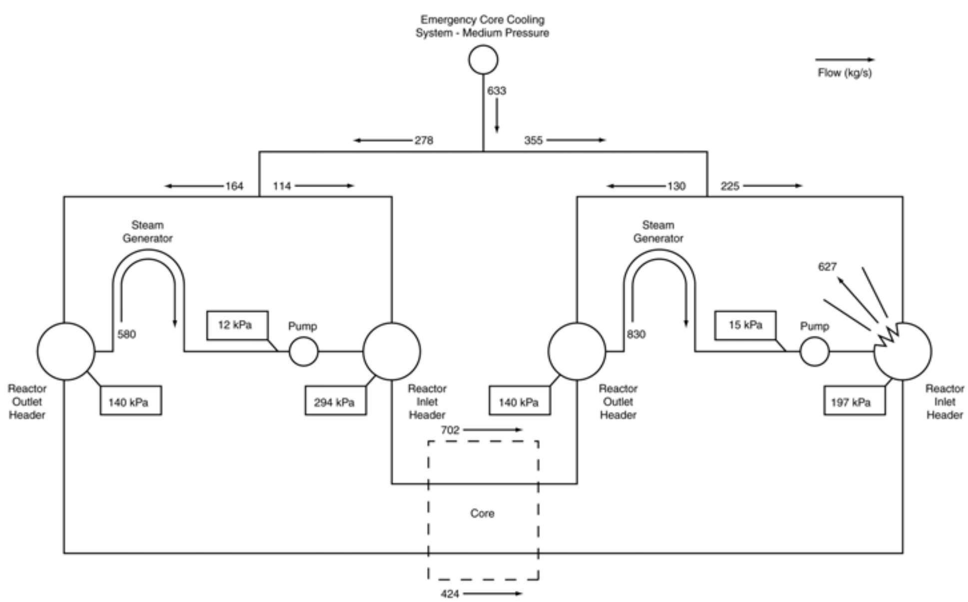

FIGURE 8.21
FLOW DISTRIBUTION AT 900 SECONDS AFTER A 35 PERCENT REACTOR INLET HEADER BREAK

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Figures 8.22 and 8.23 show short-term pressure transients and medium-term flow transients for the 35 percent inlet header break. It can be seen that the pressure continues to decrease even though the emergency injection flow initiates at about 10 seconds. In the medium term, a flow reversal occurs in one of the core passes; the long term flow pattern remains in the forward direction for this break as long as the HT pumps are on.

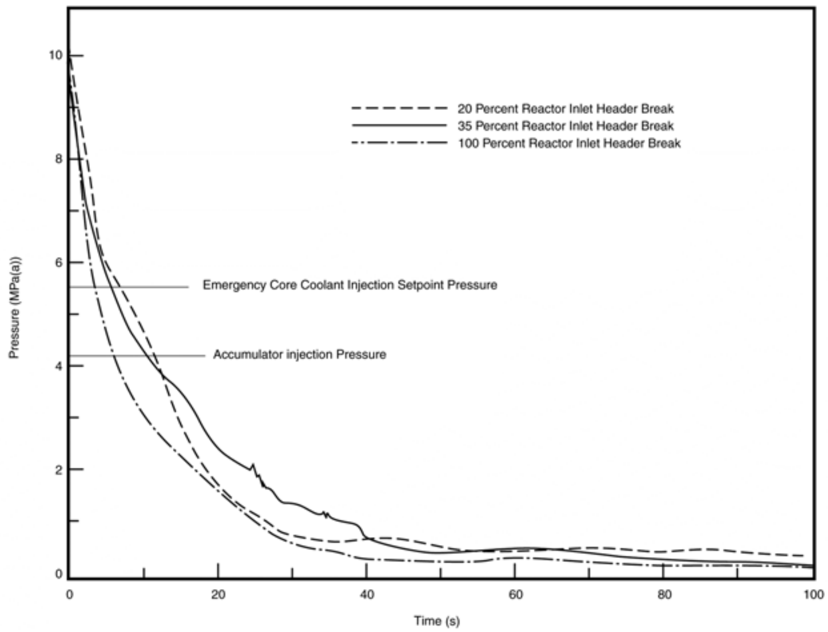

FIGURE 8.22
COOLANT PRESSURE TRANSIENTS FOR REACTOR INLET HEADER BREAKS

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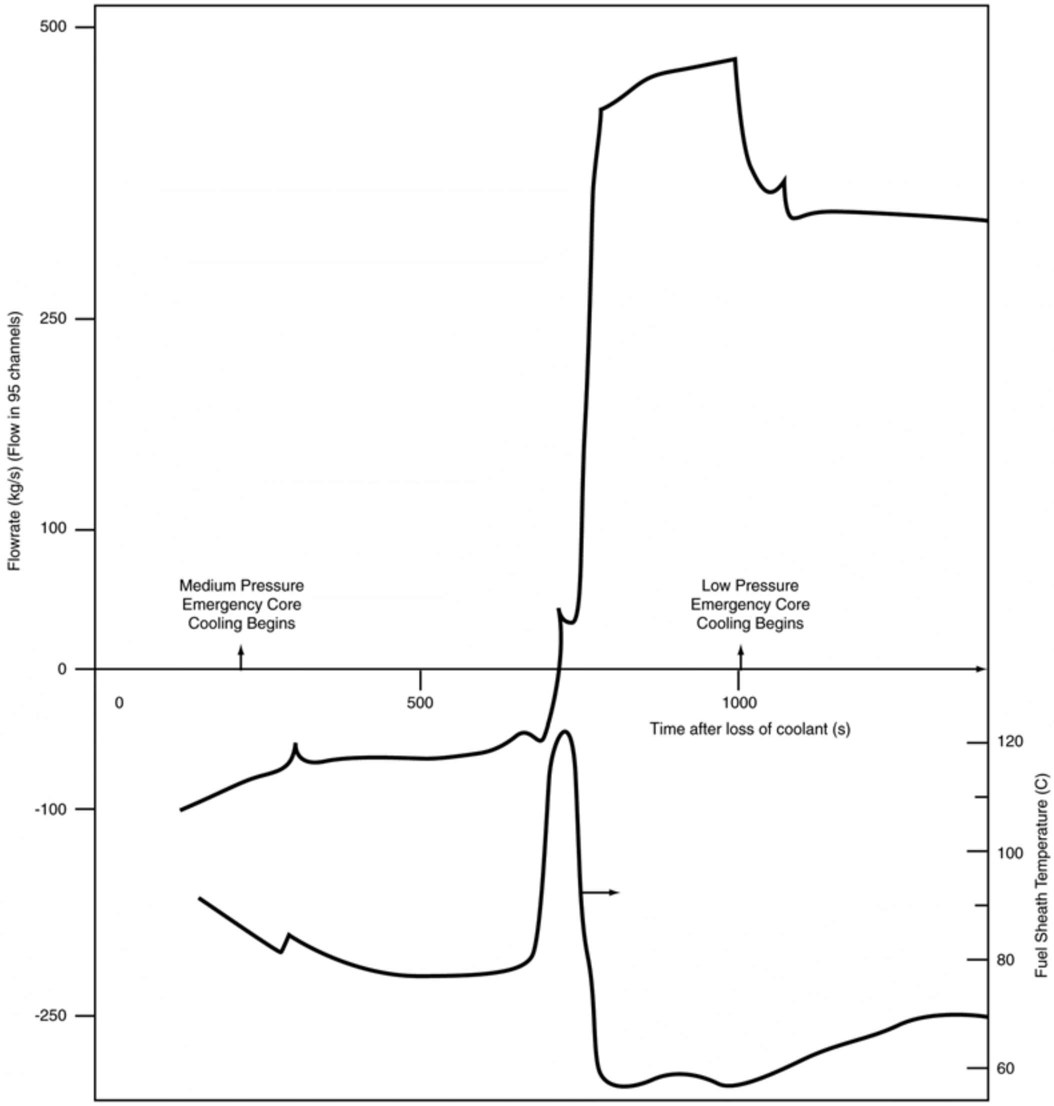

FIGURE 8.23
FLOW AND SHEATH TEMPERATURE TRANSIENT IN CORE PASS DOWNSTREAM OF A 35 PERCENT REACTOR INLET HEADER BREAK

PPI649 8-23

Verification of LOCA flow predictions cannot, of course, be done by comparison with the real thing, so several experimental facilities have been built by AECL-WNRE to test these models. In order of increasing scale these facilities are RD4, RD12, and RD14, the last of which is still in operation. RD14 is a figure-of-eight loop with two full-scale channels containing fuel bundle simulators heated by electricity. Feeders, headers, pumps, and steam generators are placed at the same vertical elevations as the 600 MWe reactor; components are scaled to approximately match the channel heat production. Blowdown, injection, and refill tests are done for comparison with calculated values. To date, the general conclusions are (a) depressurization, sheath temperature rise, and inventory during the blowdown phase are quite well predicted in high channel flow conditions, (b) stratification of flow leads to poor prediction of peak sheath temperatures, and (c) refill of feeders and channels under low flow conditions requires separate predictive models based on experiments. The Canadian Westinghouse Injection Test (CWIT) facility, a full-scale model of headers, feeders, and fuel channels, has provided the data for semi-empirical refill models that are reasonably good at prediction of refilling parameters.

The refill model is applied when the SOPHT model breaks down during a simulation. From separate tests, it has been found that SOPHT results are unreliable after one of the following limit criteria is reached:

(a) < 50% mass inventory in the HT system
(b) > 50% void in upstream reactor header
(c) < 250 $kg/m^2/sec$ mass flux in 2-phase channels

Summary: SOPHT or FIREBIRD models give reasonably good results under high flow conditions, but are unreliable when low flow develops. Unfortunately, they are very expensive to run. There is scope for further development of simpler models that can give reasonably accurate results for limited ranges of accident conditions.