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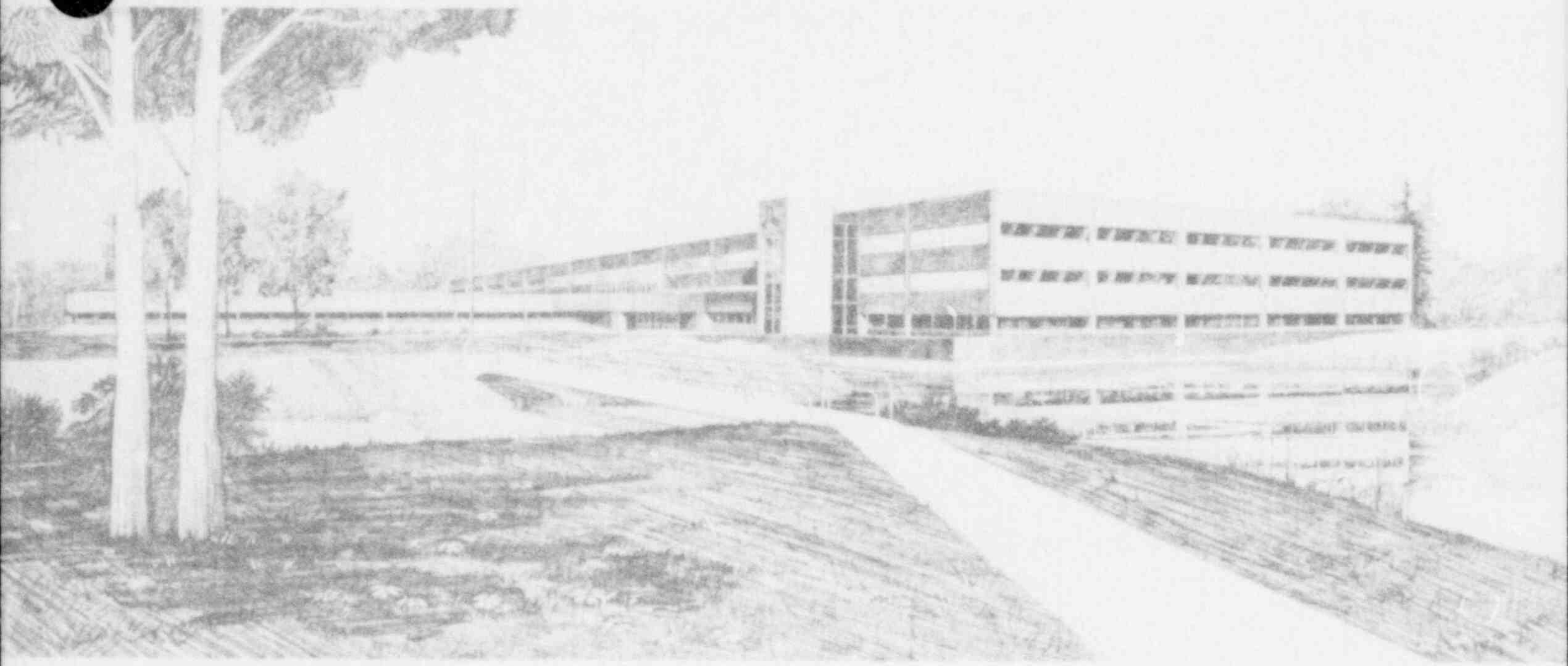

Idaho National Engineering Laboratory
Operated by the U.S. Department of Energy

System Analysis Handbook

Jay R. Larson

November 1985

Prepared for the
U.S. Nuclear Regulatory Commission
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SYSTEM ANALYSIS HANDBOOK

Jay R. Larson

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Idaho Falls, Idaho 83415

Prepared for the
Division of Accident Evaluation
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
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FIN No. A6336

ABSTRACT

This handbook provides simple procedures for calculating the behavior of light water reactors during a variety of incidents. It provides an additional tool for assessment of ongoing and postincident behavior. The handbook consists of a main body describing generic procedures, an appendix providing specific design data for a limited number of plants for application with the procedures, and an appendix listing existing and planned BWR and PWR plants by containment types and thermal-hydraulic parameters. The procedures are currently limited to break flow rate, decay heat power and integrated power, steam generation from decay heat, mass balance, shutdown margin, natural circulation, noncondensable gas generation, dose estimates, DNB evaluation, void formation in the upper head, and torus heatup.

FIN No. A6336—AEOD Thermal Hydraulic Handbook

SYSTEM ANALYSIS HANDBOOK

EXECUTIVE SUMMARY

The purpose of this handbook is to provide an additional tool for assessing the behavior of light water nuclear reactors that have exhibited abnormal behavior as the result of an unplanned incident. The intent is that this tool will be easy to apply to scope system behavior without relying on complex computer codes. This handbook provides simplified calculational procedures for evaluating a variety of nuclear plant thermal-hydraulic phenomena and conditions specifically identified by the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Division of Accident Evaluation.

This handbook is divided into a main body and appendixes. The main body describes generic analysis procedures for evaluating each identified item. The appendixes provide supporting design and calculational data for specific nuclear plants. The identified items in the main body include break flow rate, decay heat power and integrated power, steam generation from decay heat, mass balance, shutdown margin, natural circulation, noncondensable gas generation, dose estimates, DNB evaluation, void formation in the upper head, and torus heatup. The purpose, methodology, assumptions, and limitations of each procedure are stated and examples of applying the procedures are included. Each procedure contains sample appendix information. Values of the parameters necessary to perform the procedure are listed and also designated according to procedure nomenclature to expedite performance.

Appendix A contains a list of existing and planned BWR and PWR plants categorized by containment types and thermal-hydraulic parameters.

Appendix B contains specific plant design and characteristic information for the Bellefonte, Browns Ferry Unit 1, McGuire, and Rancho Seco Unit 1 plants.[a] The Appendix B data are grouped by plant name and by corresponding items in the main body for each plant. Due to limited funding for this initial handbook, only limited information is provided for the Browns Ferry, McGuire, and Rancho Seco plants.

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a. A proprietary version of this document exists which contains additional plant data. Copies can be obtained by request through J. Hopenfeld, NRC.

CONTENTS

SYSTEM ANALYSIS HANDBOOK

1. INTRODUCTION

In carrying out its responsibilities, the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Division of Accident Evaluation has identified the need for a handbook that would provide (a) simple, manual analytical techniques for analyzing and interpreting the behavior of water reactor systems resulting from incidents, and (b) specific characteristic information for individual or plant groups necessary to apply the techniques.

Several guidelines were specified for the development of the handbook as follows:

- The prime usage would be for postevent evaluation requiring only "ballpark" accuracy.
- The handbook would be a compilation of readily available information only.
- The main body of the handbook would contain generic information with specific plant data included in appendixes.
- The British engineering system of units would be used.

The items identified for inclusion in the handbook as analytical techniques are breakflow rate, decay heat power and integrated power, steam generation from decay heat, mass balance, shutdown margin, natural circulation, noncondensable gas generation, dose estimates, DNB evaluation, void formation in the upper head, and torus heatup.

The items listed are found in the main body and are described in terms of generic analysis procedures. The purpose, methodology, assumptions, and limitations of each procedure are stated and examples of applying the procedures are included. Each procedure contains sample appendix information. Values of the parameters necessary to perform the procedure are listed and also designated according to procedure nomenclature to expedite performance.

Appendix A contains a list of existing and planned BWR and PWR plants categorized by containment types and thermal-hydraulic parameters.

Appendix B contains specific plant design and characteristic information for the Bellefonte, Browns Ferry Unit 1, McGuire, and Rancho Seco Unit 1 plants. The Appendix B data are grouped by plant name and by corresponding items in the main body for each plant. Due to limited funding for this initial handbook, only limited information is provided for the Browns Ferry, McGuire, and Rancho Seco plants.

SYSTEM ANALYSIS HANDBOOK

2. BREAK FLOW RATE

This section provides a technique for estimating the coolant mass flow rate through a leak in the reactor system permitting (a) primary or secondary coolant flow to the containment resulting from a pipe break or stuck open relief valve, or (b) primary coolant flow to the secondary side of a PWR steam generator resulting from the rupture of one or more steam generator tubes.

Estimating Break Flow Rate to Containment

For a break or opening connecting the primary or secondary system to the containment, the mass flow rate through the break can be estimated from the expression

$$\dot{M} = A\, G_c\, C_D \qquad (2\text{-}1)$$

where

$\dot{M}$ = mass flow rate (lbm/s)

A = estimated break flow area (ft^2)

G_c = critical or choked flow mass flux [$lbm/(s\text{-}ft^2)$]

C_D = a discharge coefficient (≤ 1.0).

Fluid Conditions. Fluid conditions at a leak location can be either saturated or subcooled, depending on the temperature and pressure conditions. The flow can be either choked or unchoked, depending on the pressure differential across the leak area and the critical pressure ratio. The critical pressure ratio is determined by the homogeneous equilibrium model (HEM)[1] for saturated fluid, or the Henry-Fauske model (HF)[2] for subcooled fluid upstream of the break.

Choked Flow. The flow out of the leak is considered to be choked if the pressure ratio, $P_{downstream}/P_{upstream}$, is equal to or less than the critical pressure ratio. The critical pressure ratio is plotted in Figure 2-1 as a function of upstream pressure for both saturated and subcooled upstream fluids, with upstream fluid temperature (subcooled) or quality (saturated) as parameters.

If the upstream fluid is saturated, the quality (χ) of the fluid must be estimated ($0 < \chi \leq 1.0$) and Figure 2-2 (HEM results[3]) used to estimate the choked flow mass flux. Figure 2-2a relates the quality, void fraction, and pressure for homogeneous flow and may be helpful in estimating the quality. Figure 2-2b provides the choked volumetric flow rate based on the upstream conditions obtained from the homogeneous model. If the upstream fluid is subcooled, or $\chi = 0$, Figure 2-3 (Henry-Fauske model results[3]) or Figure 2-3a (Chen[4] model results) must be used to estimate the choked flow mass flux. The Chen model has been compared to recent data with excellent results in the metastable region designated Zone 1. The model is not recommended for Zone 2 beneath the dashed line locus. Figure 2-3b provides the choked volumetric flow rate based on the upstream conditions obtained from the Henry-Fauske model.

The discharge coefficient is assumed to be unity for piping ruptures. Values for safety and relief valves are tabulated in Appendix B, "Mass Balance."

Estimating Steam Generator Primary-to-Secondary Break Flow Rate

For single-phase liquid flow through the steam generator tubes to the secondary side where the flow is not choked, the expression in the rearranged Darcy Equation (2-2) can be used to estimate the leakage as follows:

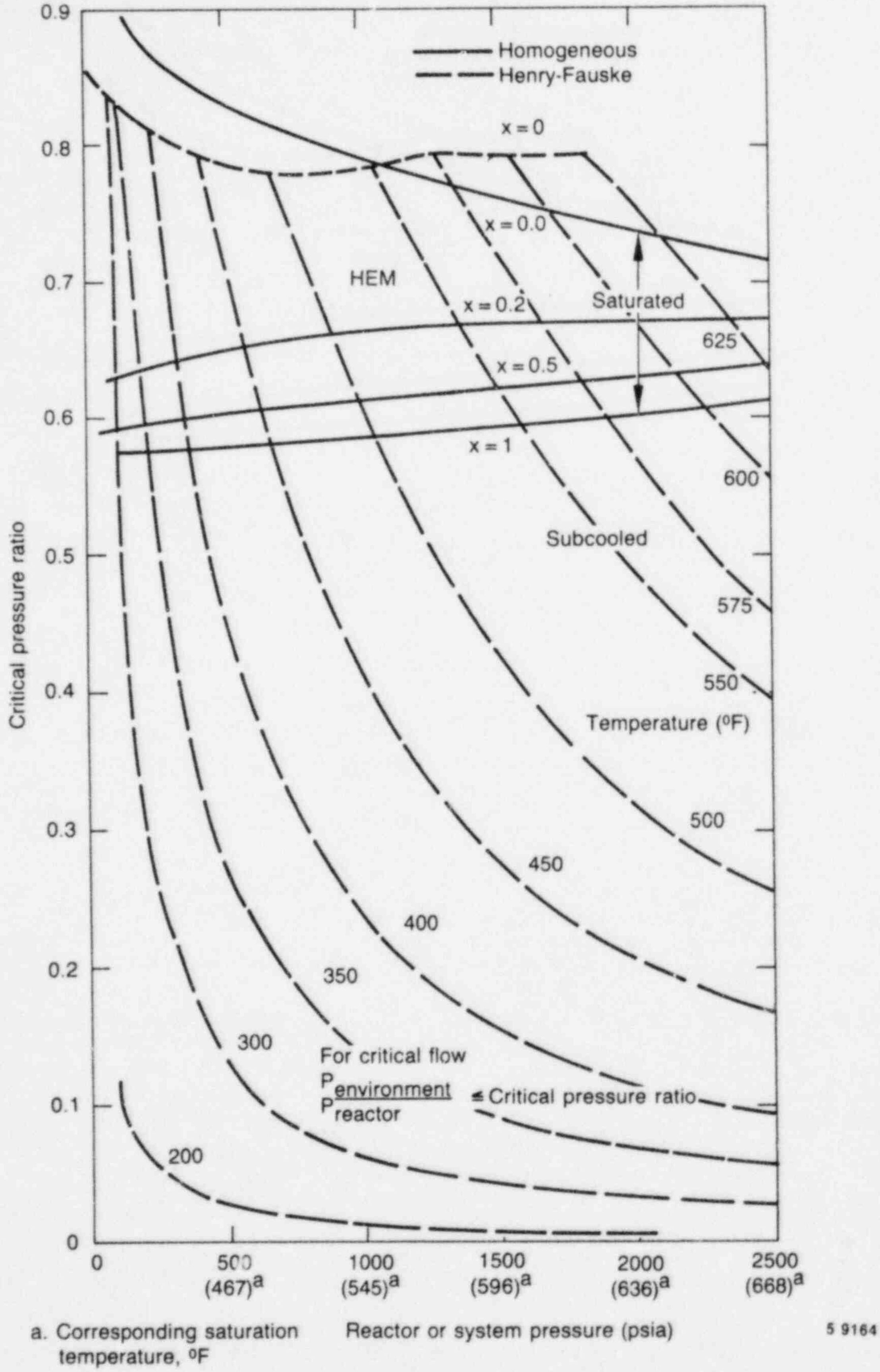


Figure 2-1. Critical pressure ratio for saturated fluid (HEM) and subcooled liquid (HF).

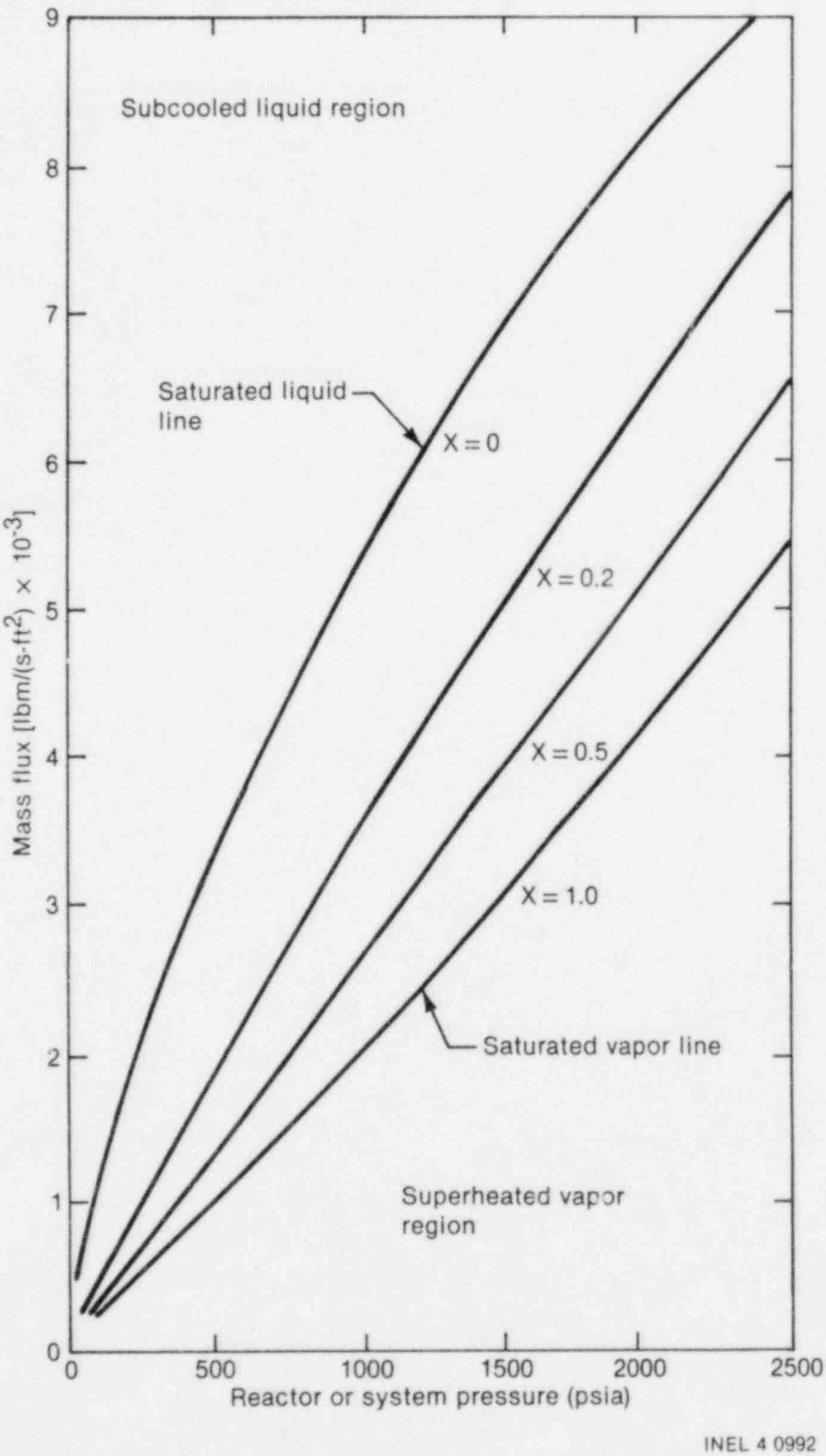


Figure 2-2. Critical flow rate for saturated liquid (HEM).

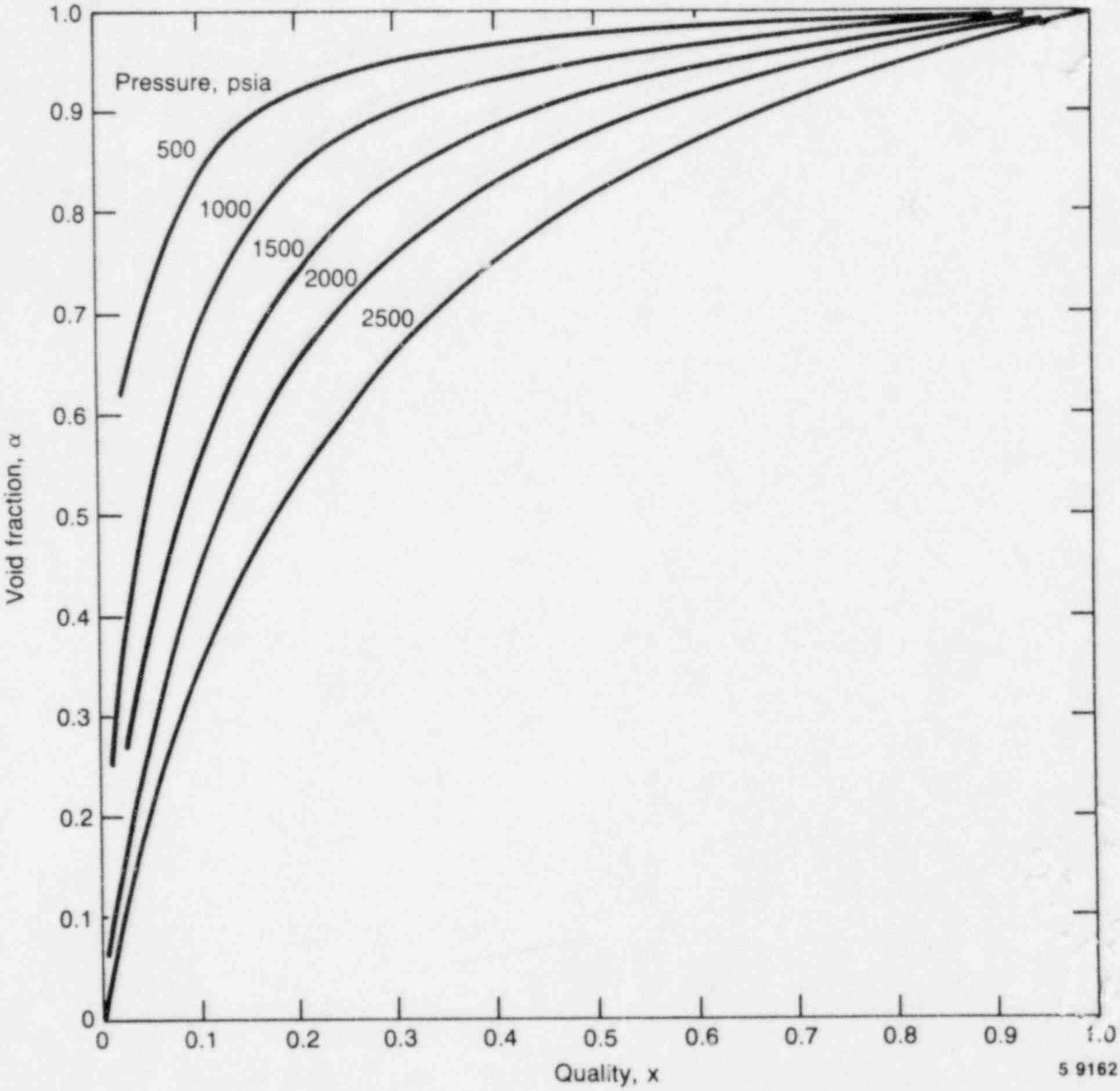


Figure 2-2a. Void fraction as a function of pressure and quality for homogeneous flow.

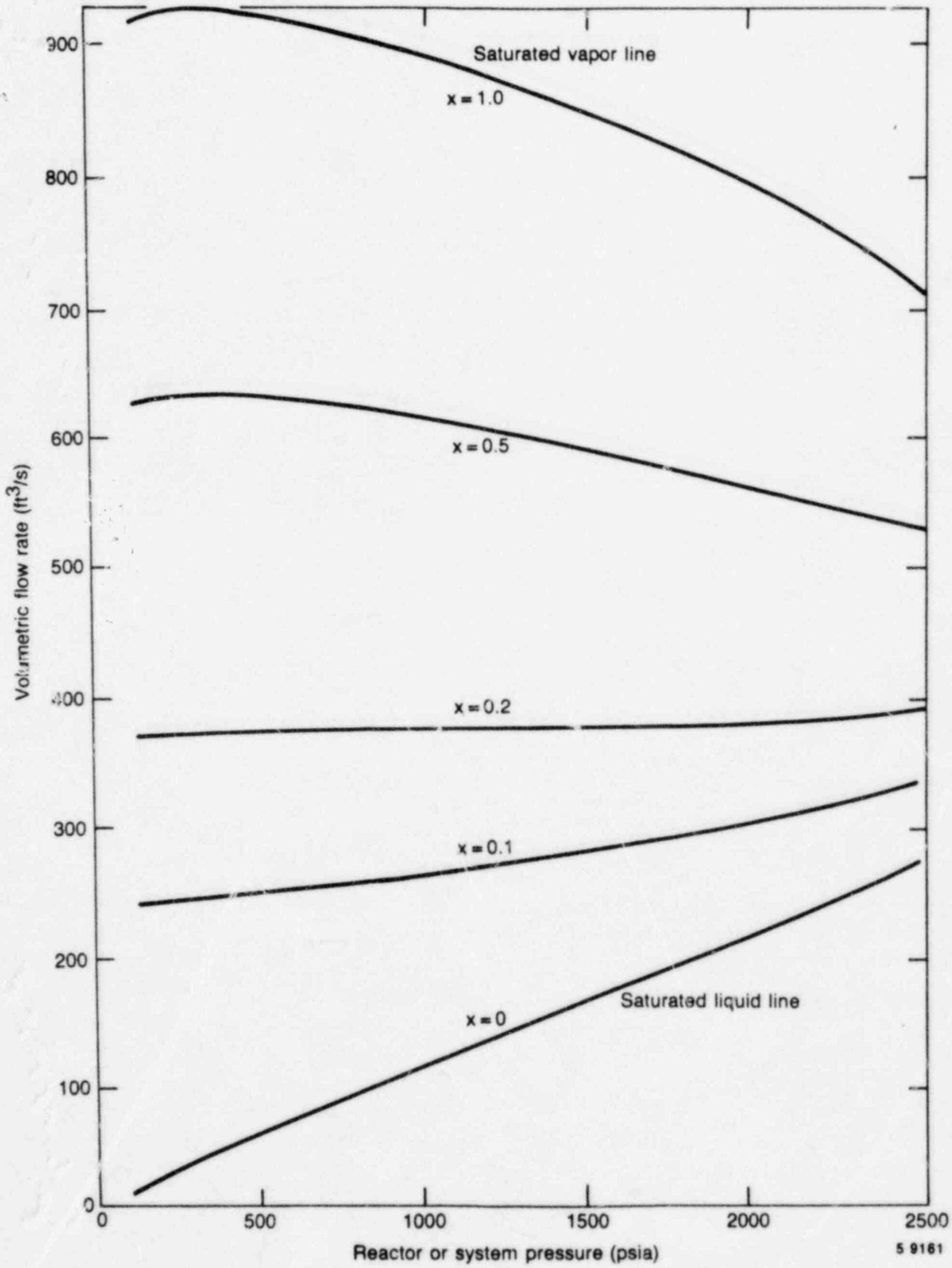


Figure 2-2b. Source volumetric flow rate for saturated mixture.

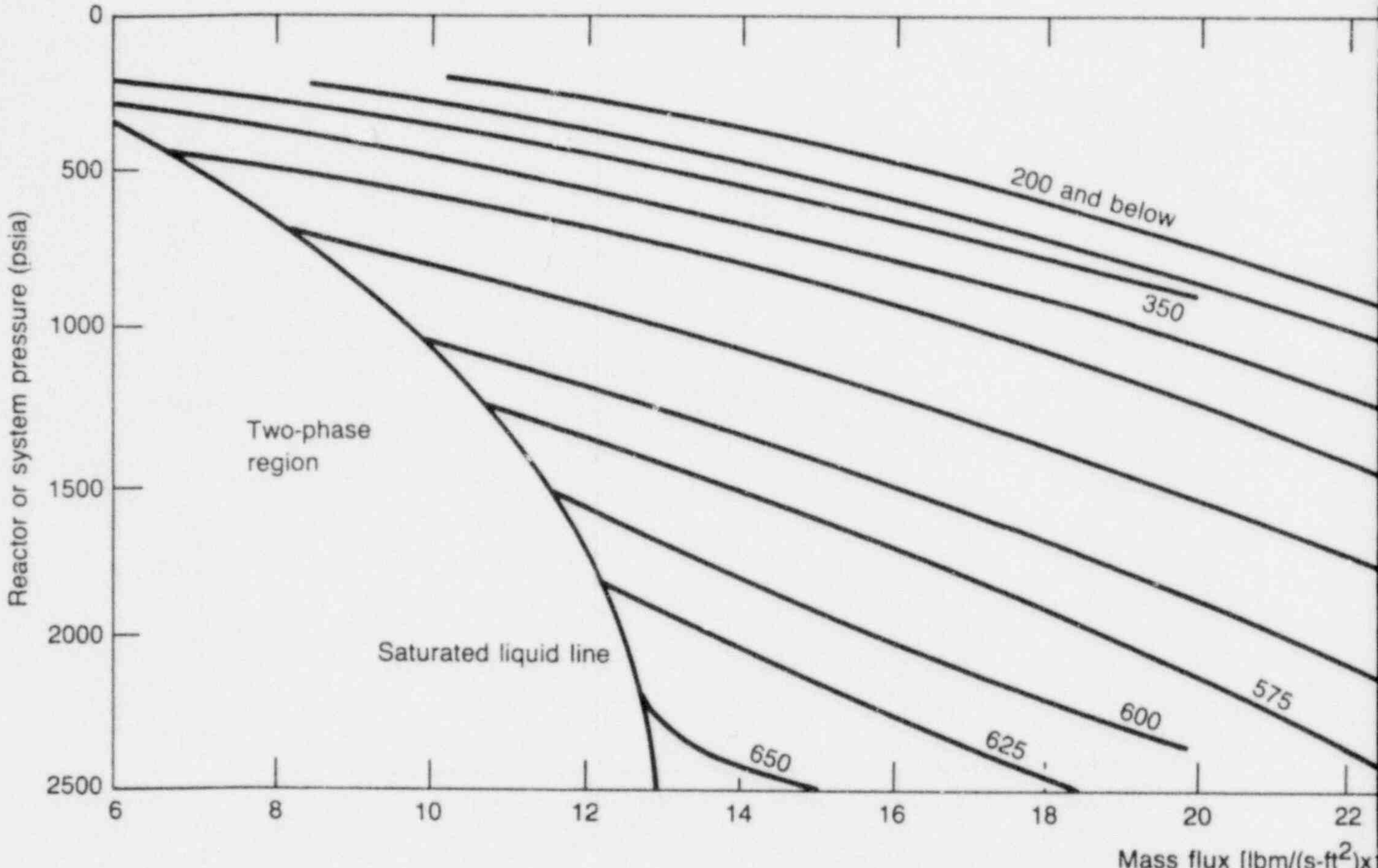
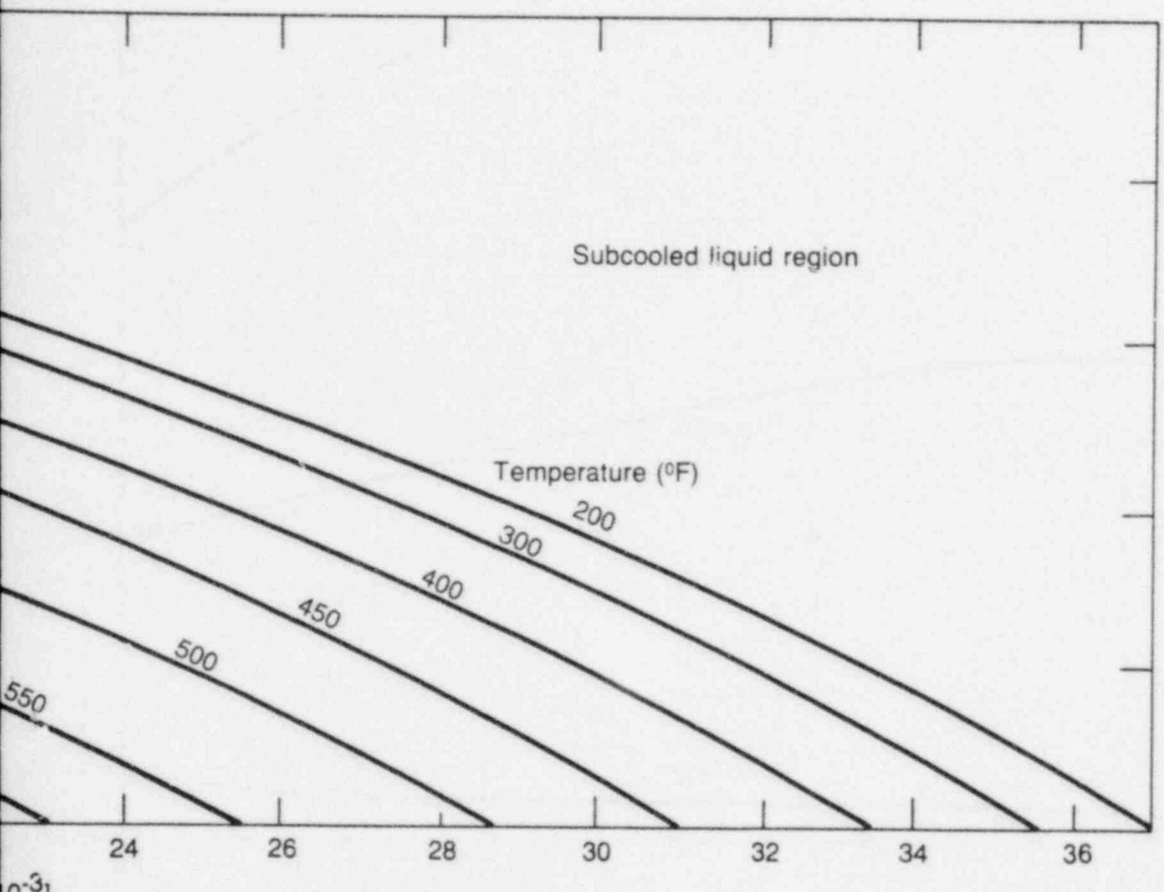


Figure 2-3. Critical flow rate for subco



oled liquid (HF).

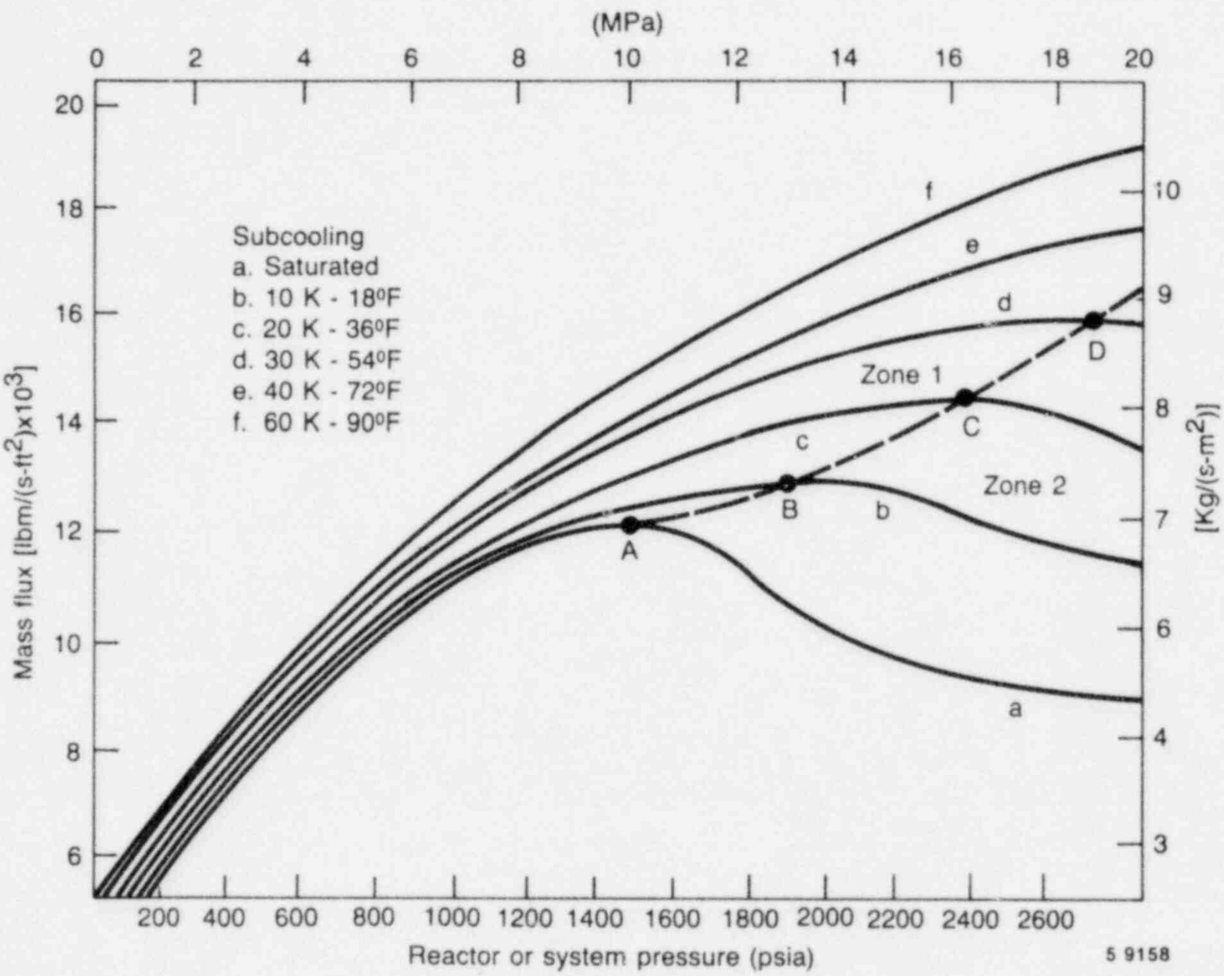


Figure 2-3a. Critical flow rate for subcooled liquid (Chen).

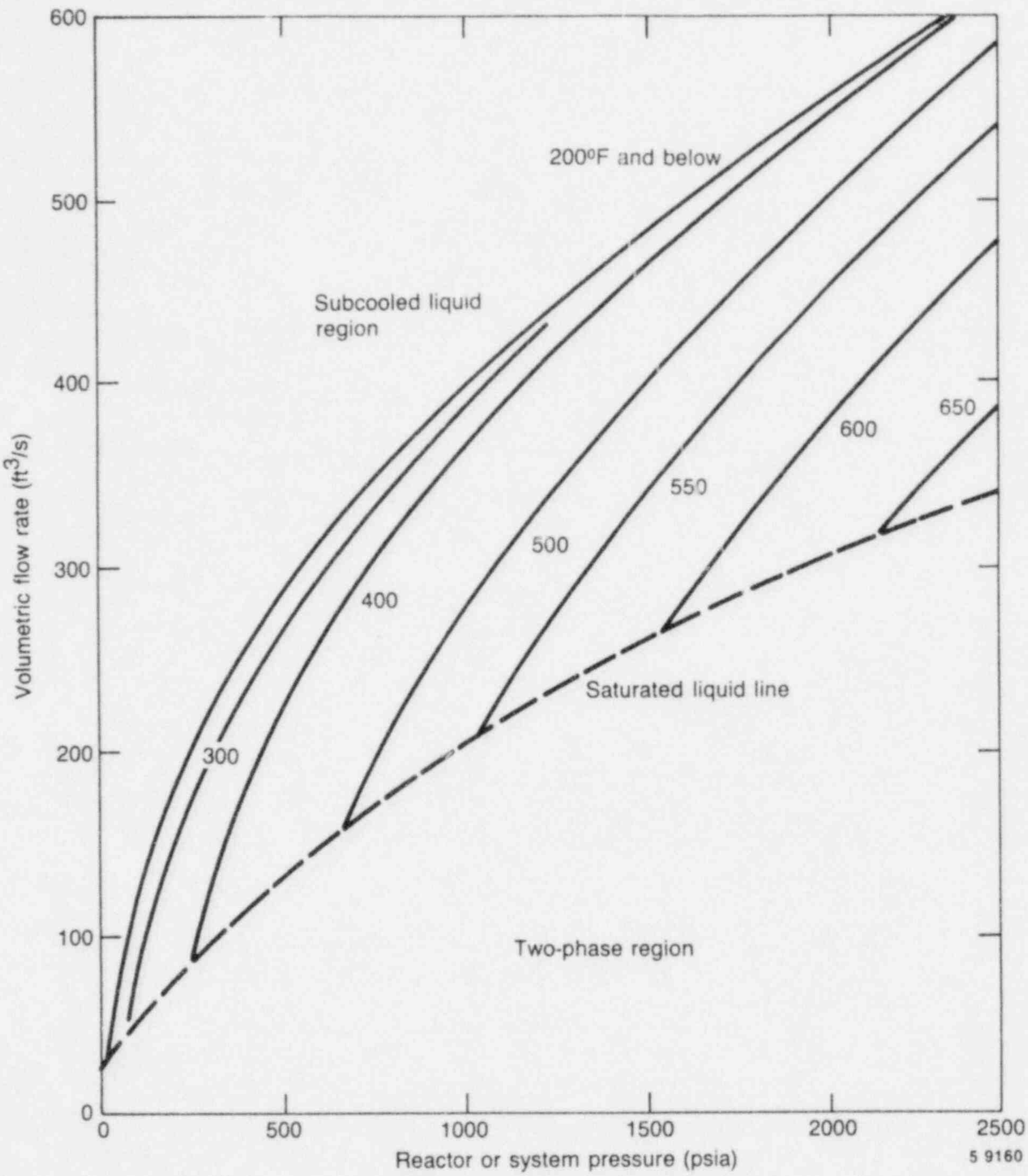


Figure 2-3b. Source volumetric flow rate for subcooled liquid (HF).

$$G = \left[\frac{a\,\Delta P}{fL/D + K} \cdot 2\varrho g_c\right]^{1/2} \tag{2-2}$$

where

G = mass flux [lbm/(s-ft^2)]

a = unit conversion constant (144 in.^{2}/ft^2)

ΔP = pressure difference between fluid in primary side and fluid in secondary side (lbf/in.^{2})

f = friction factor, dimensionless

L = assumed tube length from steam generator plenum to rupture location (ft)

D = tube inside diameter (ft)

K = sum of inlet and exit form loss coefficients, dimensionless

ϱ = fluid density (lbm/ft^3)

g_c = unit conversion factor = 32.174 (lbm-ft)/(lbf-s^2).

The equation must be solved for the path from each steam generator plenum to the rupture location. The mass flux determined for each path is multiplied by the tube flow area to obtain the total leakage

$$\dot{m}_T = (G_1 + G_2)\,A \tag{2-3}$$

where

$\dot{m}_T$ = total mass leakage (lbm/s)

$G_{1,2}$ = mass fluxes for each path of single ruptured tube [lbm/(s ft^2)]

A = tube flow area (ft^2).

Figure 2-4 shows a range of solutions for Equation (2-2) where mass flux is plotted as a function of pressure difference, with flow resistance as a parameter. For steam generators included in Appendix B, an approximate solution for a tube rupture near the tube sheet is shown by the dashed lines.

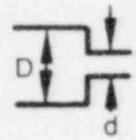
Assumptions

Flow through the rupture does not choke.[5] Flow through the steam generator tubes (Figure 2-4) is based on a liquid density of 45.455 lbm/ft^3 (P = 1300 psia, T = 560°F). For a reasonable temperature and pressure range, the use of actual liquid density would make little difference. The curve values could be corrected by multiplying the density ratio $(\varrho_{actual}/45.455)^{1/2}$.

Measured pressures inside the primary system are approximately equal to the stagnation pressure.

It is further assumed that steam generator tube ruptures result in fully developed turbulent flow with a friction factor = 0.012 for drawn tubing with an inside diameter near 0.6 in.

Suggested values		Orifice with full turbulence[6]	
Geometry	K	$\frac{d}{D}$	K
Slightly rounded inlet	0.23		
Sharp inlet[6]	0.5		
Full exit[6]	1.0	0.8	~0.8
Rupture exit	3.0 to 5.0	0.6	~0.65
		0.4	~0.62



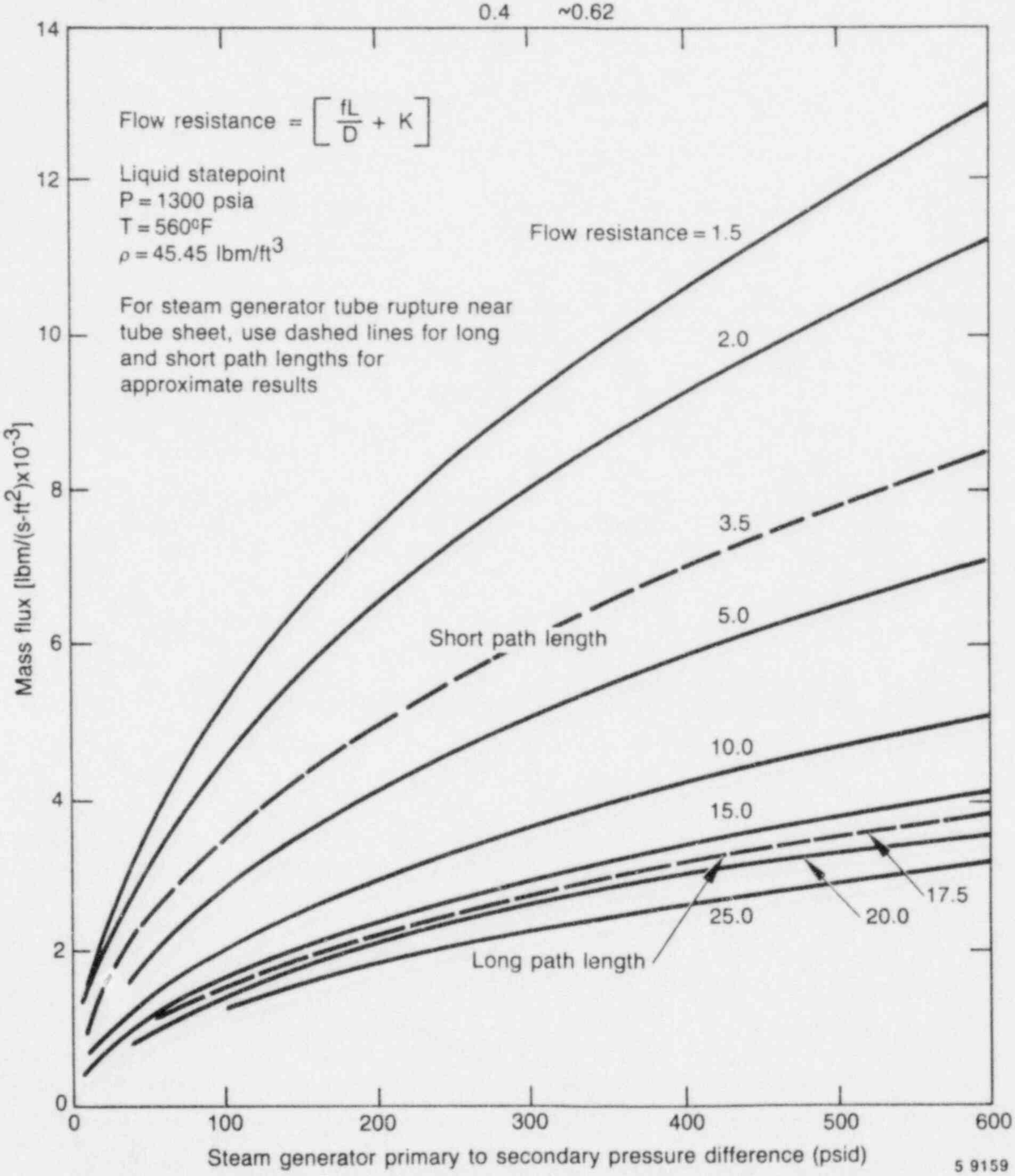


Figure 2-4. Unchoked single-phase liquid mass flux as a function of flow resistance and pressure difference for steam generator tube rupture.

A steam generator tube rupture area will be equal to or larger than the flow area for each path. (The tube rupture area for the Ginna plant was about five times the flow area.[7]) The form loss coefficient for the rupture is assumed to be no less than 3.0 (Reference 5). If the rupture area is known to be less than the tube flow area, the form loss coefficient for an orifice, as shown on Figure 2-4, may be added.

The critical mass flux or discharge coefficient is not a function of size.[8]

For many steam generators, the tube nlet form loss coefficient may be taken as 0.23. The outlet form coefficient may be taken as 3.0, making K = 3.23 for both paths.

Limitations

The critical mass flow rates determined from the Figures 2-1, 2-2, and 2-3 are based on a knowledge of the upstream stagnation pressure. Exact stagnation conditions will not be available.

Critical mass flow rates for saturated fluid conditions can vary by ~50%, based on the potential range in fluid quality.

For saturated liquid ($\chi = 0$), the HF model gives better results.

The break or rupture flow area will not be known.

Examples of Technique Application

1. Fluid is flowing out of a ruptured ECC line connected to the cold leg piping into the containment at a pressure of 50 psia. The measured vessel fluid pressure is 1500 psia and cold leg temperature is 500°F.

   From Figure 2-1, it is determined that the ratio of the containment pressure to vessel pressure is less than the critical pressure ratio based on fluid conditions upstream of the break, i.e.,

$$\frac{P_{containment}}{P_{vessel}} = \frac{50.0}{1500.0} \leq \frac{P_{critical}}{P_{stagnation}} = 0.42;$$

   thus, the flow is choked at the break. Figure 2-3 (HF model) yields the subcooled liquid mass flux of 19.9×10^3 lbm/s-ft^2. Figure 2-3a [Chen (Reference 4) model] yields an approximate value of 16×10^3 lbm/(s-ft^2) for the subcooling value of 96°F.

2. A steam generator tube rupture is suspected. The vessel fluid conditions are 1200 psia saturated liquid. What is the leakage mass flow rate for a rupture location near the tube sheet? The tube sheet is about 1 ft thick; the tube inside diameter (D) is 0.606 in.; the inside flow area (A) is 2.003×10^{-3} ft^2; the average active tube length (L_{ave}) is 57.6 ft, and the tube inlet is slightly rounded. The steam generator secondary side pressure is 1000 psia.

   It is assumed that the flow is not choked and Figure 2-4 applies. For the short flow path length, the flow resistance term is fL/D + K or

$$\frac{0.012 \times 1.0 \text{ ft}}{\dfrac{0.606 \text{ in.}}{12 \text{ in./ft}}} + 3.23 \simeq 3.5.$$

For the long path length, the term is

$$\frac{0.012 \times (1.0 + 57.6) \text{ ft}}{\frac{0.606 \text{ in.}}{12 \text{ in./ft}}} + 3.23 = 17.3 \simeq 15.0.$$

The mass flux for the short path length (G_1) is $\sim 4.9 \times 10^3$ lbm/(s-ft^2), and the mass flux for the long path length (G_2) is $\sim 2.2 \times 10^3$ lbm/(s-ft^2).

The total leakage rate is then found from Equation (2-3)

$$\dot{m}_T = 2.003 \times 10^{-3} \text{ ft}^2 \times (4.9 + 2.2) \times 10^3 \text{ lbm/(s-ft}^2) = 14.2 \text{ lbm/s}.$$

References

1. Hall, D. G. and L. S. Czapary, *Tables of Homogeneous Equilibrium Critical Flow Parameters for Water in SI units*, EGG-2056, September 1980.

2. Hutcherson, M. N., *Numerical Evaluation of the Henry Fauske Critical Flow Model*, MDC-N9654-100, McDonnell Douglas Automation Company, Nuclear and Electrical Engineering Department, July 1980.

3. Hall, D. G., MASFLO, EG&G Idaho, Inc.

4. Chen, Y. S., *Critical Flow Calculation of Subcooled Water from Orifices*, Office of Nuclear Regulatory Research, Nuclear Regulatory Commission, June 1984

5. Kashiwa, B. A. and R. C. Mjolsness, *Analytic Studies Pertaining to Steam Generator Tube Rupture Accidents*, NUREG/CR-4079, March 1985.

6. Roberson, J. A. and C. T. Crowe, *Engineering Fluid Mechanics*, Boston: Houghton, 1975.

7. Martin, T. T., *NRC Report on the January 25, 1982, Steam Generator Tube Rupture at R. E. Ginna Nuclear Power Plant*, NUREG-0909, April 1982.

8. Jarrell, D. B. and D. G. Hall, *Determination of the Scale Effect on Subcooled Critical Flow*, NUREG/CR-2498, February 1982.

Appendix Information

Specific values for equations:

L_{ave} = 57.6 ft, average steam generator tube length

D = 0.606 in., steam generator tube inside diameter

A = 2.003×10^{-3} ft^2, steam generator tube flow area.

3. DECAY HEAT POWER AND INTEGRATED POWER

This section provides a technique for estimating the decay heat power and integrated power (total energy released) after reactor shutdown.

Estimating Decay Heat and Integrated Power

The decay heat power and integrated power generated following shutdown of a light water reactor that has operated for an infinite period (10^{13} s) or finite operating time as a function of time after scram are shown in Figures 3-1 and 3-2, respectively.

The decay heat and integrated decay power for an arbitrary power history can be estimated by linearizing the power history into time intervals of constant power. Then, the terms can be computed for each interval and summed to obtain the total values.

Figure 3-3 provides an optional correction factor which accounts for neutron capture in fission products. It is an upper bound based on four years operating time with a typical light water reactor neutron flux spectrum. For shorter operating times, the correction factor is overestimated.

Figures 3-4 and 3-5 provide additional actinide correction factors that can optionally be applied if the production rate of $^{239}U/^{235}U$ fission is known. The figures are based on a production rate of 0.7 atoms/fission. It is thought that most light water reactors operate with factors near 0.6 to 0.7. The actinide decay energy is not very sensitive over this limited range of production factor.

Assumptions

The calculations for the decay heat power presented herein are based on representations of the ANSI/ANS 5.1 standard of 1979.[1] The decay heat power presented is the combined energy release from ^{235}U fission products and actinide (^{239}U and ^{239}Np) decay. The constants for the actinides and 23 fission groups from ANSI/ANS 5.1 have been incorporated into RELAP5/MOD1.6,[2] which was used to obtain the decay power and integrated decay power.

The energy release per fission is assumed to be 200 MeV.

Limitations

The presented data do not include the effect of thermal fission of ^{239}Pu or fast fission of ^{238}U. The fast fission of ^{238}U should not add more than 2% to power presented for light water reactors.

Spacial effects are not included.

The effect of fuel burnup on the distribution of fission product sources is not included. The one sigma uncertainty for ^{235}U decay power is no more than 2%.

Fission energy release from delayed neutron induced fission depends on the negative reactivity at shutdown and is not included within the scope of this work. The magnitude could be 1 or 2 (MW-s)/MW of initial power.

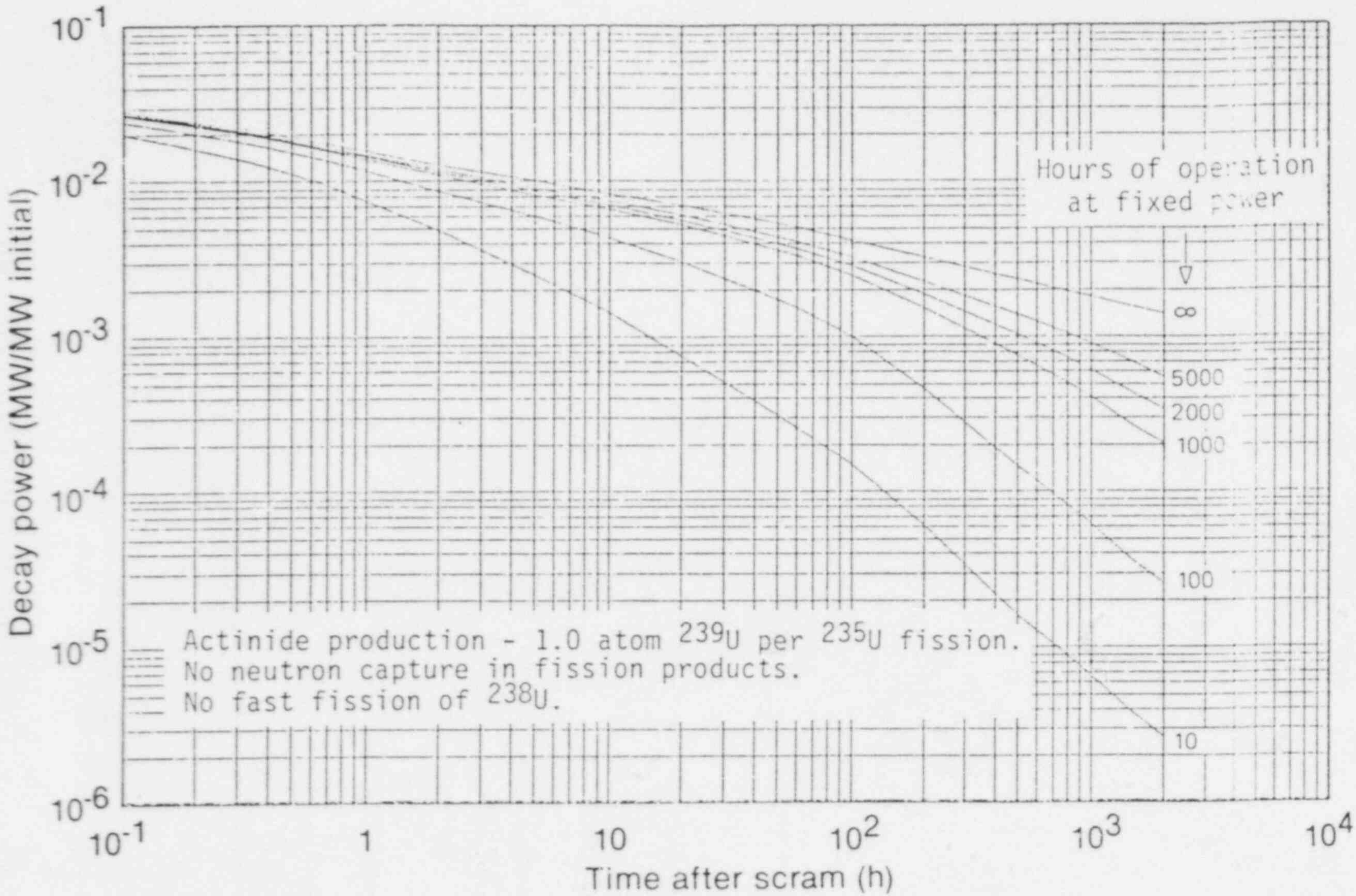


Figure 3-1. Decay power of ^{235}U as a function of operating time and time after scram.

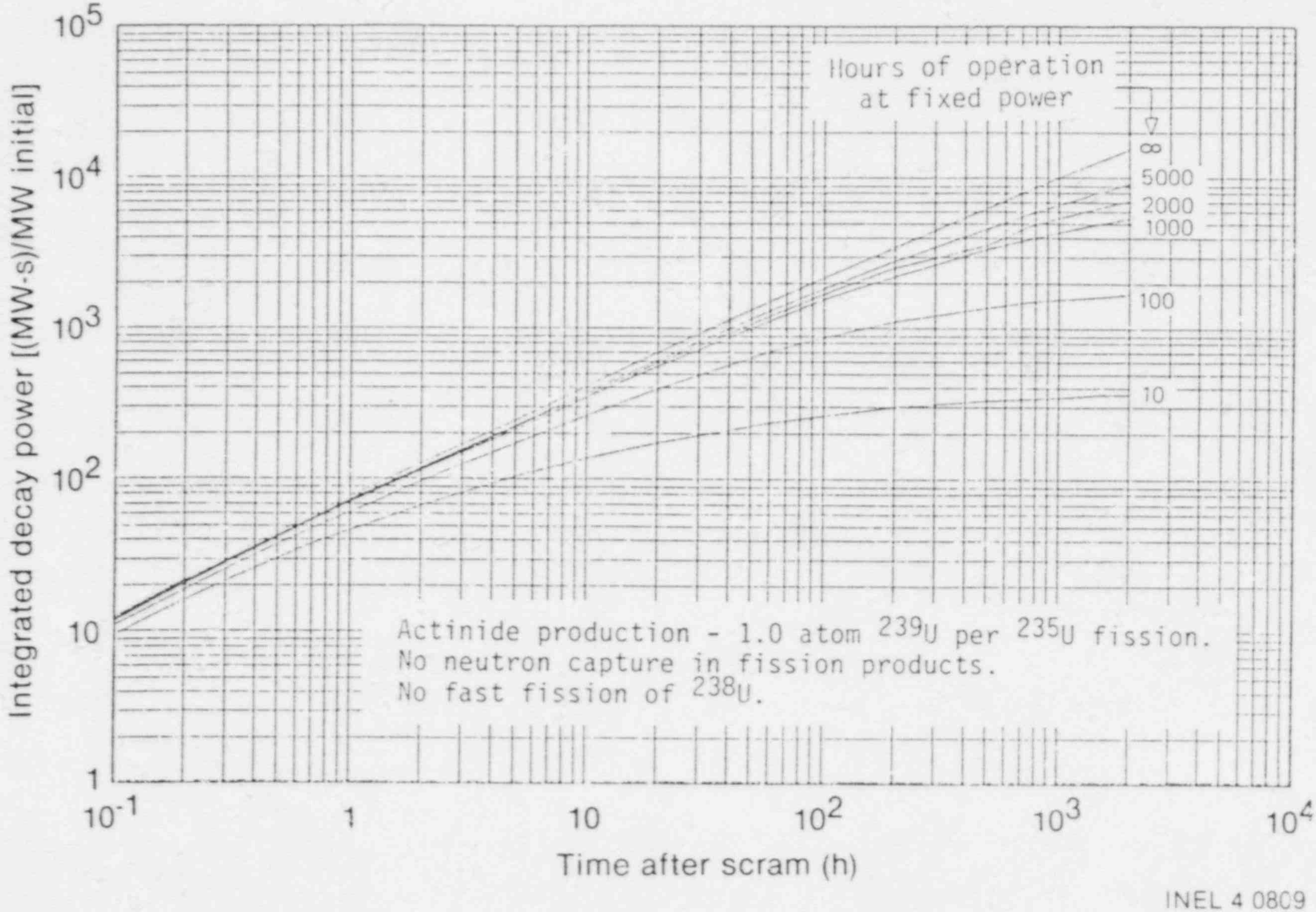


Figure 3-2. Integrated decay power of ^{235}U as a function of operating time and time after scram.

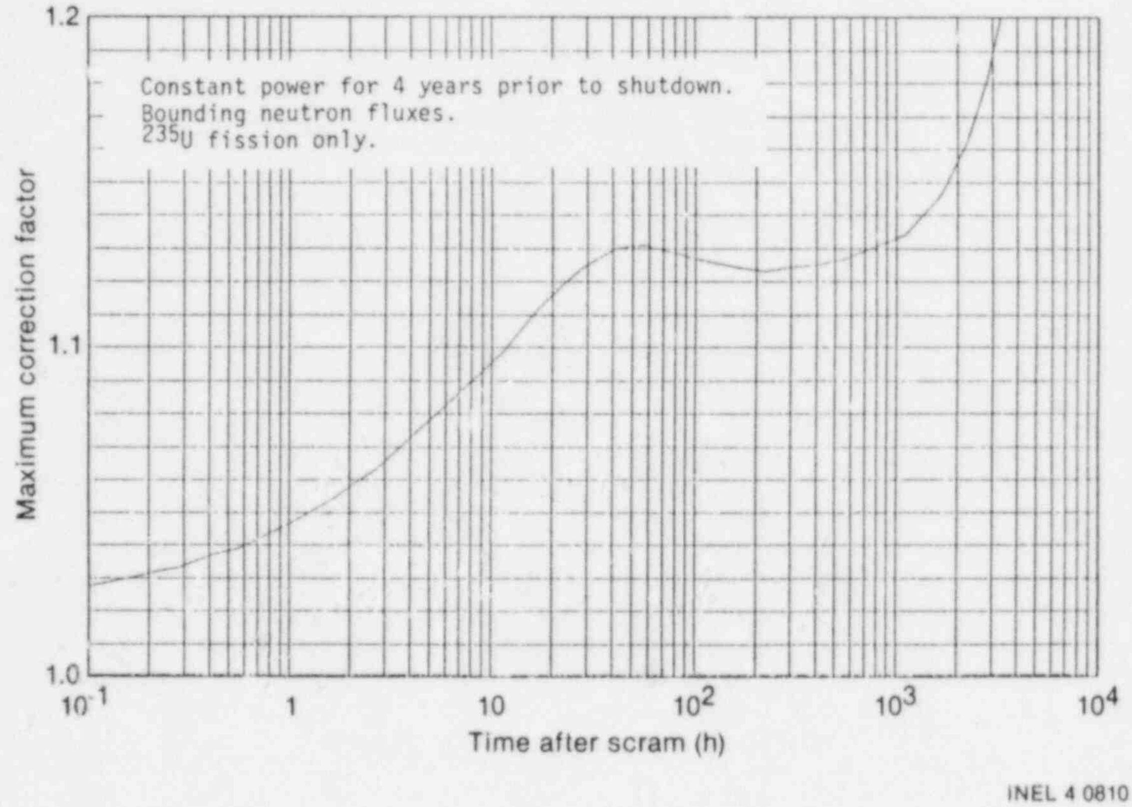


Figure 3-3. Optional correction factor for maximum ratio of decay power including neutron capture in fission products to decay power without neutron capture in fission products.

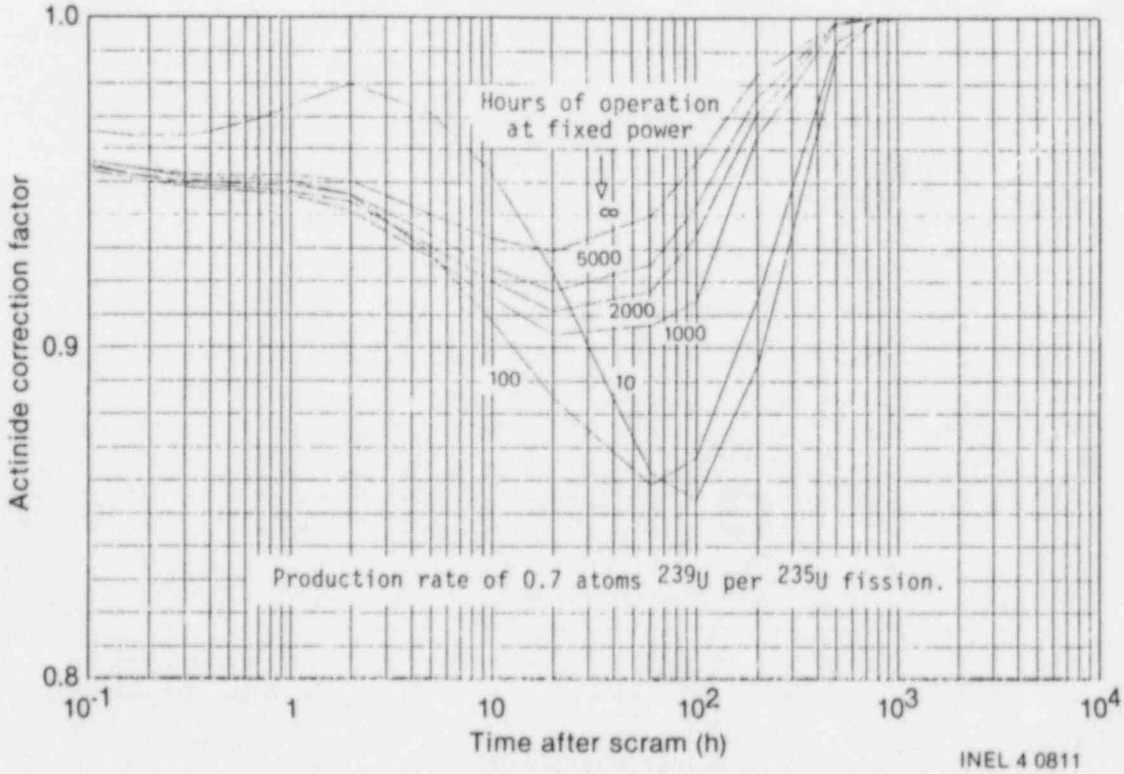


Figure 3-4. Decay power correction factor for a non-unity actinide production factor.

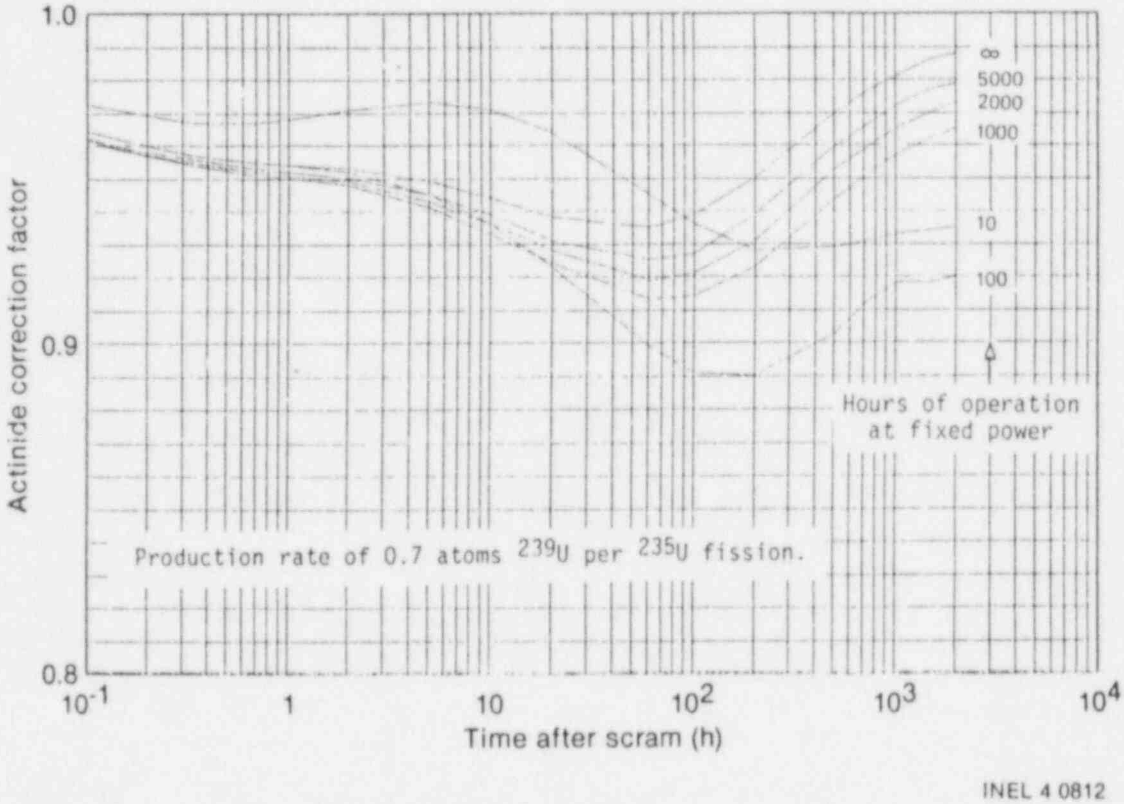


Figure 3-5. Integrated power correction factor for a non-unity actinide production factor.

Example of Technique Application

A reactor was operated for 100 h at a steady-state power of 1000 MW, shut down for 80 h, and then operated again for 100 h at a steady-state power of 500 MW before an incident resulted in scram. The production rate for actinide decay is 0.7 atoms per fission of ^{235}U. What is the decay power 20 h after the scram?

The contr:bution to the decay power from the first operating interval is determined as follows from Figures 3-1, 3-3, and 3-4: 1000 MW (steady power level) x 0.000456 MW/MW steady power (decay power for 100 h operation—200 h after first shutdown) x 1.123 (maximum fission product absorption correction factor for 200 h after shutdown) x 0.915 (actinide correction factor for 100 hours operation and 200 h after shutdown) = 0.468 MW.

For the second operating interval, the decay power is 500 MW x 0.00306 MW/MW x 1.116 x 0.885 = 1.511 MW.

The total decay power is the sum of the contribution from each interval or 0.468 + 1.511 = 1.979 MW.

References

1. American National Standards Institute, *American National Standard for Decay Heat Power in Light Water Reactors*, ANSI/ANS 5.1, August 1979.

2. Ransom, V. H. et al., *RELAP5/MOD1 Code Manual Volume 1: System Models and Numerical Methods (DRAFT)*, EGG-2070, November 1980.

4. STEAM GENERATION FROM DECAY HEAT

This section provides a technique for estimating the elapsed time following shutdown for the core to uncover based on the pump heat and decay heat generated in the core. The technique applies directly to a station blackout situation. The technique divides the process into three periods: (a) a period where the steam generator secondary side becomes dry; (b) a period where the coolant heats up to a temperature equivalent to the safety valve relief pressure; and (c) a period where the coolant boils off.

Steam Generator Dryout

The approximate elapsed time and integrated decay energy required for a steam generator secondary side to become dry can be estimated from the expression

$$ {}_0Pe_1 + \int_0^{t_1} \dot{q}_{dh}\, dt = a\, M_{ls}\, h_{fg} \qquad (4\text{-}1)$$

where

${}_0Pe_1$ = pump energy input to coolant from zero time up to time (t_1) (assumed zero for station blackout) (MW-s)

t, t_1 = time, time at which secondary is dry (s)

$\dot{q}_{dh}$ = decay heat rate (MW)

a = unit conversion constant = [1.055×10^{-3} (MW-s)/Btu]

M_{ls} = total mass of liquid contained in the secondary side of all steam generators (lbm)

h_{fg} = latent heat of vaporization at the pressure of the secondary side (Btu/lbm).

This expression equates the integrated decay heat energy, plus any pump heat input to the primary system, with the energy needed to evaporate the liquid mass contained in the secondary sides of the steam generators.

Coolant Heatup Period

The heatup period will include the effects of (a) heating the coolant and structural material from an average coolant temperature to saturation temperature corresponding to the safety valve set pressure, (b) the coolant displacement or loss from the primary system caused by coolant thermal expansion, and (c) the energy transferred from the fuel to the coolant as the fuel cools from its steady state average value to a value closer to the coolant temperature. The expression for the heatup period is

$$ {}_1Pe_2 + \int_{t_1}^{t_2} \dot{q}_{dh}\, dt = a\Big[M_m\, C_{pm}\, (T_2 - T_1) - M_f\, C_{pf}\, (T_{f1} - T_{f2}) $$

$$+ \Delta M_l \frac{(h_2 - h_1)}{2} \quad \text{(assumed energy absorbed by liquid pushed out primary loop system)}$$

$$\left. + M_{l2} (h_2 - h_1) \right] \quad \text{(assumed energy absorbed by liquid remaining in primary loop system)}$$

$$\Delta M_l = V_{lc} (\varrho_1 - \varrho_2) \tag{4-2}$$

where

${}_1Pe_2$	=	pump energy input from time (t_1) to time (t_2) (MW-s)
t_2	=	time when pressure relief occurs (heatup phase complete) (s)
M_m	=	metal mass of structural components, core vessel internals; piping walls, steam generator internals, core vessel wall, pressurizer, and surge line are excluded (lbm)
$C_{pm,f}$	=	specific heat for components, steel and UO_2 = 0.11 Btu/(lbm-°F) (steel); 0.072 Btu/(lbm-°F) (UO_2)
$T_2 - T_1$	=	change in system coolant average temperature (°F)
M_f	=	mass of UO_2 (lbm)
$T_{f1} - T_{f2}$	=	change in average fuel temperature (°F)
ΔM_l	=	mass of liquid pushed out of primary loop system during the heatup period (lbm)
$h_2 - h_1$	=	liquid enthalpy difference between statepoint 1 and 2 (Btu/lbm)
M_{l2}	=	mass of liquid remaining in primary loop system after volume expansion during the heatup period, excludes pressurizer, surge line, and accumulator lines (lbm)
V_{lc}	=	volume of primary system excluding pressurizer, surge line, and accumulator lines (ft^3)
$\varrho_{1,2}$	=	liquid density at initial statepoint 1 and statepoint 2 (lbm/ft^3).

Coolant Boiloff Period

The boiloff period accounts for (a) liquid displacement caused by void formation, (b) energy required to heat liquid contained in the pressurizer to saturation, (c) energy required to vaporize the liquid in the loop system including pressurizer, and (d) liquid remaining in the pressurizer because of a loop seal design of the surge line. The expression is

$${}_2PE_3 + \int_{t_2}^{t_3} \dot{q}_{dh}\, dt = a\left[\left(V_{1c} - V_v + V_{P+S} - V_{PH} - V_R\right) \varrho_2 h_{fg} + V_{P+S}\, \varrho_{ave} \frac{(h_2 - h_1)}{2}\right] \tag{4-3}$$

where

$_2Pe_3$ = pump energy input from time (t_2) to time (t_3) (MW-s).

t_3 = time when core will uncover (s).

V_v = volume of liquid in primary system displaced by initial void generation—volume voided is sum of following:

- upper head
- upper plenum above hot leg pipe centerline
- 0.2 x upper plenum below hot leg pipe centerline
- 0.2 x core
- 0.2 x hot leg

V_{P+S} = combined volume of pressurizer and surge line (ft^3)

V_{PH} = volume of pressurizer and surge line below elevation of surge line connection to hot leg (ft^3)

V_R = volume of liquid in primary system below the top of the core and downcomer inlet (ft^3)

h_{fg} = heat of vaporization at statepoint 2 (Btu/lbm)

ϱ_{ave} = average density = $(\varrho_1 + \varrho_2)/2$ (lbm/ft^3).

Assumptions

All energy generated in the core (decay heat) is transferred to the primary coolant and then to the steam generator secondary side until the secondary side becomes dry. The steam generator secondary side is assumed to remain at near constant pressure during the dryout process with no feedwater being injected and with no liquid mass loss out of relief or safety valves.

No rate-related effects are considered except for the decay heat. The processes are assumed to be quasi-steady-state. No heat losses to the containment are considered. After steam generator dryout, the decay heat is assumed to heat the system (fluid and metal) uniformly by means of circulation to a new system temperature and pressure limited by safety valve set points. Fluid thermal expansion and voiding of volume in the primary system displaces liquid which is forced into the pressurizer and perhaps out the pressurizer safety valves. Liquid displaced from the system loop during the thermal-expansion phase absorbs about half of the specific energy absorbed by liquid remaining in the loop.

After safety valve set points are reached, decay heat goes to heat the liquid in the pressurizer to saturation, and to evaporate the liquid remaining in the primary loop system above the core (including pressurizer and surge line) that can fall back to the core.

During the boiloff process, the hot legs, the upper plenum below the hot leg centerline, and the core are voided to a value of 0.2. The upper head and upper plenum above the hot leg centerline are completely voided. The void distribution is based on RELAP5 calculations for Bellefonte.

Mass and energy gains or losses from injection or letdown systems are not considered. No energy is added from pressurizer heaters.

Limitations

The procedure was compared to RELAP5 calculations for the Bellefonte PWR, with the results as tabulated below.

Event	RELAP5 (s)	Handbook (s)
Steam generator dryout	300	356
System heatup	900	1188
Core uncovery begins	1650	1500

The treatment of the heat capacity effect of the structural components of the system is approximate. Specific consideration of thermal resistance of masses, surface areas, and heat transfer coefficients is needed for a more accurate representation.

Heat loss to the containment is expected to have an effect of increasing times by about 2% based on RELAP4 calculations performed with a model of the Zion PWR.[1] Coolant thermal expansion and liquid displacement by vapor generation are approximately treated.

Example of Steam Generator Dryout and Core Uncovery

An incident occurs at a large four-loop PWR causing a loss of all electrical power. Estimate the time for the steam generator secondary to dry out and the core to uncover. Natural circulation transfers energy to the steam generator. The initial conditions of the plant are: liquid mass per steam generator, $M_{ls}/4$ = 98,660 lbm, steam generator liquid state, saturated at 1000 psia; h_{fg} = 650.4 Btu/lbm; primary coolant average temperature = 588°F; primary coolant average pressure = 2250 psia; plant power = 3425 MW. The plant has operated for several years.

The time for the secondary side of the steam generator to boil dry is found as follows.[a] The integrated decay heat energy needed to vaporize the mass of liquid found from Equation (4-1) is

$$\int_0^{t_1} \dot{q}_{dh}\, dt = 394{,}600 \text{ lbm} \times 650.4 \text{ Btu/lbm (at 1000 psia)}$$

$$\times\ 1.055 \times 10^{-3} \text{ (MW-s)/Btu} = 2.706 \times 10^{5} \text{ MW-s.}$$

The time necessary for the core to generate the energy is found from Figure 3-2, assuming infinite operating time with no optional correction factors as follows. The integrated energy per MW initial power is needed and is found from (2.706×10^5 MW-s)/3425 MW initial = 79.01 (MW-s)/MW.

This value in conjunction with the figure yields time for steam generator dryout of about 4000 s.

After the steam generators boil dry on the secondary side, the primary fluid and system structural mass is assumed to heat up to a temperature corresponding to a pressure of 2500 psia at which time the safety valves maintain pressure at 2500 psia until sufficient coolant leaves the primary system to uncover the core.

The integrated decay heat energy necessary to raise system pressure during the heatup period is found from Equation (4-2) as applied below.

a. Precalculated values for the terms found in Equations (4-1), (4-2), and (4-3) are listed in "Appendix Information."

The primary system steel mass (M_m) in lbm consists of the listed components and associated mass obtained from the Appendix:

- Core cladding (54,197 lbm)
- Core barrel (94,200 lbm)
- Remaining vessel internals (269,540 lbm)
- Hot, intermediate, cold leg piping (4) (258,800 lbm)
- Pump (4) (200,000 lbm)
- Steam generator tubes, tube plate, channel head (4) (1,014,800 lbm)

The total = 1,891,537 lbm = M_m. The vessel wall, steam generator shell, pressurizer, surge line, and accumulator lines are excluded.

The UO_2 mass is 250,300 lbm = M_f.

The energy needed to heat the primary system coolant from the initial average temperature (588°F) to the saturation temperature corresponding to 2500 psia depends on the volume expansion of the coolant. The coolant mass (tabulated in volume, ft^3) that is heated (V_{1c}) is taken as the sum of the liquid in the primary system excluding liquid contained in the pressurizer, surge line, and accumulator lines:

- Vessel total (4586.77 ft^3)
- Piping total (4) (1778.36 ft^3)
- Steam generator primary (4) (3848 ft^3)
- Total volume = 10,213.13 ft^3 = V_{1c}.

The initial density is taken at the average temperature of 588°F and is 44.15 lbm/ft^3 = ϱ_1. The final density is that of saturated liquid at 2500 psia, ϱ_2 = 34.98 lbm/ft^3.

The mass pushed out of the primary system components specified by thermal expansion is

$$\Delta M_l = V_{1c}(\varrho_1 - \varrho_2) = 10{,}213.13 \text{ ft}^3 \, (44.15 - 34.98) \text{ lbm/ft}^3 = 93{,}654 \text{ lbm.}$$

This mass (ΔM_l) expands into the surge line and pressurizer displacing the fluid originally contained. (Using an average density of 39.57 lbm/ft^3 = $(\varrho_1 + \varrho_2)/2$, the displaced mass is 93,654 lbm and occupies a volume of 93,654 lbm/39.57 lbm/ft^3 = 2367 ft^3, a volume in excess of the pressurizer and surge line volume, 1853.2 ft^3.) The mass pushed out absorbs about half of the specific energy compared to the mass remaining in the loop components.

The mass remaining in the loop components is

$$M_{12} = V_{1c}\,\varrho_2 = 10{,}213.13 \text{ ft}^3 \times 34.98 \text{ lbm/ft}^3 = 357{,}255 \text{ lbm.}$$

Substituting into Equation (4-2), the energy needed to heat the system uniformly to 668°F is then

$$[1{,}891{,}537 \text{ lbm steel} \times 0.11 \text{ Btu/(lbm-°F) steel} \times (668\text{-}588)\text{°F}$$

$$+ 250{,}300 \text{ lbm } UO_2 \times 0.072 \text{ Btu/(lbm-°F) } UO_2 \times (1100\text{-}668)\text{°F}$$

$$+ 357{,}255.9 \text{ lbm } H_2O \times (731.7 - 596.7) \text{ Btu/lbm}$$

$$+ 93{,}654 \text{ lbm } H_2O \left(\frac{731.7 - 596.7}{2}\right) \text{Btu/lbm}] \; 1.055 \times 10^{-3} \text{ (MW-s)/Btu}$$

$$= 66{,}900 \text{ MW-s.}$$

Dividing by the steady state operating power of 3425 MW so the Figure 3-2 can be be used, the number for figure usage equals 19.53 (MW-s)/MW initial.

The energy required to boil off sufficient liquid to uncover the core is determined in a similar manner using Equation (4-3). The vapor first produced collects in the system and displaces additional liquid from the primary system. The voided volume (V_v) in ft^3 is the volume of the following components:

- Upper head (350.9 ft^3)
- Upper plenum above hot leg centerline (264.4 ft^3)
- Upper plenum below hot leg centerline [(0.2)(858) = 171.6 ft^3]
- Core [(0.2)(685.4) = 137.1 ft^3]
- Hot legs [(0.2)(427.6) = 85.5 ft^3]
- Total volume of liquid displaced = 1009.5 ft^3 = V_v.

The volume in the pressurizer and surge line is V_{P+S} = 1853.2 ft^3 as given in the Appendix.

As the surge line and pressurizer are completely above the hot leg connection, $V_{PH} = 0$.

The liquid volume remaining in the system following core uncovery is as follows:

- Core [(0.8)(685.4) = 548.32 ft^3]
- Core bypass = 289.65 ft^3
- Downcomer below inlet = 696.11 ft^3
- Intermediate piping leg below downcomer inlet (4) = 381.0 ft^3
- Lower plenum - 897.21 ft^3

Total volume = 2812.29 ft^3 = V_R.

The volume of liquid to be boiled off is

$$V_{lc} - V_v + V_{P+S} - V_{PH} - V_R = 10{,}213.13 - 1009.5 + (1853.2 - 0.0) - 2812.29 = 8244.5\ ft^3.$$

The energy needed to evaporate the liquid mass plus heat the liquid contained in the pressurizer and surge line to saturated conditions from Equation (4-3) is

$$\int_{t_2}^{t_3} \dot{q}_{dh}\, dt = [8244.5\ ft^3 \times 34.98\ lbm/ft^3 \times 361.6\ Btu/lbm(h_{fg})$$

$$+\ 1853.2\ ft^3\ \frac{(44.15 + 34.98)}{2}\ lbm/ft^3 \times \frac{(731.7 - 596.7)}{2}\ Btu/lbm]$$

$$\times\ 1.055 \times 10^{-3}\ (MW\text{-}s)/Btu = 115{,}339\ MW\text{-}s.$$

Converting to the energy per initial MW of operating power for use in Figure 3-2, gives 115,339 (MW-s)/3425 MW initial = 33.65 (MW-s)/MW initial.

The total energy needed to this point in time is

$$\int_0^{t_1} \dot{q}_{dh}\, dt + \int_{t_1}^{t_2} \dot{q}_{dh}\, dt + \int_{t_2}^{t_3} \dot{q}_{dh}\, dt = 79.01 + 19.53 + 33.65 = 132.2\ (MW\text{-}s)/MW,$$

which yields a time of about 7700 s. At this time, an error of 1 (MW-s)/MW results in an error of about 60 s.

Reference

1. Fletcher, C.D., *A Revised Summary of PWR Loss of Off-Site Power Calculations*, EGG-CAAD-5553, September 1981.

Appendix Information

1. Specific values for equations:

M_{ls} = 394,640 lbm, total nominal liquid in secondary side of all steam generators

h_{fg} = 650.4 Btu/lbm, nominal latent heat of vaporization of liquid in steam generator secondary side

M_m = 1,791,340 lbm, total mass of miscellaneous vessel internals, primary system piping, pumps, steam generator tubes, tube plate and channel head, cladding

M_f = 250,300 lbm, total mass of UO_2

V_{1c} = 10,213.13 ft^3, volume of primary system excluding pressurizer, surge line, and accumulator lines

T_1 = 588°F, initial average primary system coolant

T_2 = 688°F, saturation temperature at 2500 psia, statepoint 2

ϱ_1 = 44.15 lbm/ft^3, primary coolant liquid density at initial average statepoint 1

ϱ_2 = 34.98 lbm/ft^3, saturated liquid density at 2500 psia, statepoint 2

h_1 = 596.7 Btu/lbm, primary coolant enthalpy at statepoint 1

h_2 = 731.7 Btu/lbm, saturated liquid enthalpy at 2500 psia, statepoint 2

T_{f1} = 1100°F, nominal initial average UO_2 temperature

T_{f2} = 668°F, final average UO_2 temperature

V_v = 1009.5 ft^3, primary system coolant displaced by initial void generation

V_{P+S} = 1853.2 ft^3, volume of pressurizer and surge line

V_{PH} = 0.0 ft^3, volume of surge line and pressurizer below elevation of surge line connection to hot leg

V_R = 2812.29 ft^3, liquid volume remaining in primary system below top of core and downcomer inlet

2. Coolant volumes (ft^3):

Vessel

Lower plenum	897.21
Core	685.4
Core bypass	289.65
Upper plenum	1122.4
Upper plenum above hot leg centerline	264.4
Upper plenum below hot leg centerline	858.0
Guide tubes	247.4
Upper head	350.9
Downcomer above inlet	297.7
Downcomer below inlet	696.11
Vessel total	4586.77

Piping

Hot leg	106.9
Cold leg	109.79
Intermediate leg	146.9
Intermediate leg below downcomer inlet	95.25
Pump	81.0
Piping total	444.59 (hot leg, cold leg, intermediate leg, pump)
Surge line	53.2
Pressurizer	1066 liquid, 734 vapor,
Steam generator primary	962
Steam generator secondary	2118 liquid, 3788 vapor 98,660 lbm liquid
Accumulator	1000 liquid, 350 gas
Accumulator line	31.09

3. Structural masses (lbm):

Vessel

Vessel shell	662,374
Vessel head	165,150
Vessel studs, nuts, washers	37,152

Vessel Internals

Core cladding	54,197
Core UO_2	250,300
Core barrel	~94,200
Lower plenum internals	94,890
Core baffle assembly	28,760
Control rods	7,150
Upper plenum internals	109,190
Upper head internals	3,360
Downcomer neutron plate assembly	26,190

Piping

Hot leg	13,500
Intermediate leg	23,080
Cold leg	13,450
Surge line	14,690
Pressurizer	
Pump casing	50,000
Steam generator total	715,000
Steam generator lower shell	95,300
Steam generator tubes	102,000
Steam generator tubeplate	91,000
Steam generator channel head	60,700

4. Temperature:

Average fuel temperature, full power	1100°F
Average primary coolant temperature	588°F

SYSTEM ANALYSIS HANDBOOK

5. MASS BALANCE

This section provides a procedure for (a) determining the net rate of gain or loss of coolant mass contained in the primary system, or (b) determining the amount of mass in the system at present or at some future time.

Determining Primary System Coolant Mass

The mass contained in the primary system, net change, or rate of change can be determined by suitable application of an equation expressing the conservation of mass for the primary system. The equation can be written as

$$\text{Mass}_{(\text{at time, } t)} = \text{Mass}_{(\text{initial})} + \text{gains} - \text{losses}. \tag{5-1}$$

Mass gains may be due to operation of the following systems (injection rates are given in Appendix B under "Mass Balance"):

- High pressure injection (HPI)
- Low pressure injection (LPI)
- Charging
- Accumulator activation.

Mass losses may be due to operation of the following systems:

- Safety valves
- Power-operated relief valves
- Piping breaks
- Pump seal leakage
- Letdown system.

Primary system losses to the containment where flow choking is controlling and to the steam generator secondary may be determined by the methodology described in Section 2.

Assumptions

The process to which the reactor system is exposed may be represented by quasisteady-state assumptions over a period of time. System breaks must be relatively small so that break flow rates do not change rapidly.

Pumps will be on or off with no degraded flow rates considered.

Limitations

The result will be limited by estimations of the injection system rates and estimation of the effective break area.

Example of Technique Application

An incident occurs in a large PWR, which results in one of two power-operated relief valves sticking completely open. The pressurizer pressure (2250 psia) is seemingly unaffected by the open valve as the top of the pressurizer is steam filled. The high pressure charging and letdown systems are fully operational at the time. By closing the letdown system, will the charging system provide sufficient mass to maintain operation?

From information in Appendix B, "Mass Balance," it is determined that the PORV is a Copes Vulcan valve and has a discharge coefficient (C_D) of 0.46 for liquid flow and 0.66 for vapor flow[1] based on the nozzle flow area of 2.90 in.^{2} The mass loss through the valve is determined from the relationship expressed by Equation (2-1) where the mass flux is taken from Figure 2-2 for saturated vapor at 2250 psia, and is determined to be $G_c = 4.87 \times 10^3$ lbm/(s-ft^2).

$$\dot{m} = A\, G_c\, C_D = \frac{2.90 \text{ in.}^2}{144 \text{ in.}^2/\text{ft}^2} \times 4.87 \times \frac{10^3 \text{ lbm}}{\text{s-ft}^2} \times 0.66 = 65.0 \text{ lbm/s.}$$

By quickly isolating the letdown system, the only other flow path is the charging system for which the input is taken from the plot of maximum cooling injection rate as a function of system pressure (Figure 5-1). The charging flow rate = 57.0 lbm/s at 2250 psia. The mass change rate is 57.0-65.0 = -8.0 lbm/s.

The effect of this mass loss on the system pressure may be estimated by making an energy balance on the pressurizer based on the following assumptions:

- Only saturated vapor at 2250 psia exits through the PORV
- The liquid injected by the charging system displaces hot leg coolant which enters the pressurizer
- The pressurizer heaters are operating
- Initial liquid and vapor volumes in the pressurizer are known and are initially saturated and remain so.

A heat balance yields the rate of heat loss which may be used with the initial stored energy and mass balance over a specified time interval to find the enthalpy and specific volume which fixes the final pressure.

For this example, the pressurizer volume is 1800 ft^3, which initially consists of 1066 ft^3 of liquid and 734 ft^3 of vapor.

The initial energy content is

$$\frac{1066 \text{ ft}^3 \text{ (liquid)}}{0.02697 \text{ ft}^3/\text{lbm (saturated liquid)}} \times 690.1 \text{ Btu/lbm (internal energy of saturated liquid)}$$

$$+ \frac{734 \text{ ft (vapor)}}{0.15692 \text{ ft}^3/\text{lbm (saturated vapor)}} \times 1052.4 \text{ Btu/lbm (internal energy of saturated vapor)}$$

$$= 3.2199 \times 10^7 \text{ Btu.}$$

The initial mass contained in the pressurizer is

$$\frac{1066\ ft^3}{0.02697\ ft^3/lbm}\ (liquid) + \frac{734\ ft^3}{0.15692\ ft^3/lbm}\ (vapor) = 44{,}202.8\ lbm.$$

The energy lost from the pressurizer in 1 s is

65 lbm/s (vapor lost) x 1117.7 Btu/lbm (vapor enthalpy)

$$- \frac{1800\ kW}{1.055\ (kW\text{-}s)/Btu}\ \text{(heater capacity)} - 57\ lbm/s\ \text{(hot leg liquid entering)}$$

x 622.61 Btu/lbm (liquid enthalpy) = 35,455.2 Btu/s.

The average internal energy of coolant per pound of fluid at 1 min is

$$\frac{3.2119 \times 10^7\ Btu - 35{,}455.2\ Btu/s \times 60\ s/min}{44{,}202.8\ lbm - 8\ lb/s \times 60\ s/min} = 685.95\ Btu/lbm.$$

The specific volume of the coolant treated as a uniform mixture at 60 s is

$$\frac{1800\ ft^3}{44{,}202.8\ lbm - 8\ lbm/s \times 60\ s} = 0.041168\ ft^3/lbm.$$

The pressure at 60 s (1870 psia) may be found in steam tables by fixing u, the internal energy, then iterating on pressure to satisfy specific volume.

The initial rate of depressurization is then

$$\frac{2250\ psia - 1870\ psia}{60\ s} = 6.33\ \frac{psi}{s}.$$

The conclusion then is that the charging system will not provide sufficient mass to maintain operation indefinitely.

Reference

1. Abdollahian, D., *Critical Flow Predictions Through Safety and Relief Valves*, EPRI Research Project V102-5, S. Levy Inc., June 1982.

Appendix Information

1. Injection system capacity:
   See Figure 5-1.

2. Accumulator:

Number	4
Capacity	1000 ft^3 liquid each

3. Letdown system:

Capacity	75 gal/min (7.72 lbm/s) nominal
Capacity	120 gal/min (12.35 lbm/s) maximum

4. Pump seal water:

Supply capacity, total	32 gal/min (4.43 lbm/s) nominal
Return capacity, total	12 gal/min (1.65 lbm/s) nominal

5. Pressurizer safety valve:

Type	Crosby HP-BP-6-6M6
Number	3
Throat area	3.644 $in.^2$
Discharge coefficient	0.84 vapor, 0.9 liquid

6. Pressurizer PORV:

Type	Copes Vulcan 3 in. 1500
Number	2
Throat area	2.90 $in.^2$
Discharge coefficient	0.66 vapor, 0.46 liquid

7. Steam generator safety valve:

Type	Unknown
Number	5 per main steam line
Set points	1185 psia (lowest) 1255 psia (highest)
Flow rate	893,200 lbm/h @ 1185 psia 945,300 lbm/h @ 1255 psia

8. Steam generator PORV:

Type	Unknown
Number	1 per main steam line
Flow rate	400,000 lbm/h @ 1135 psia

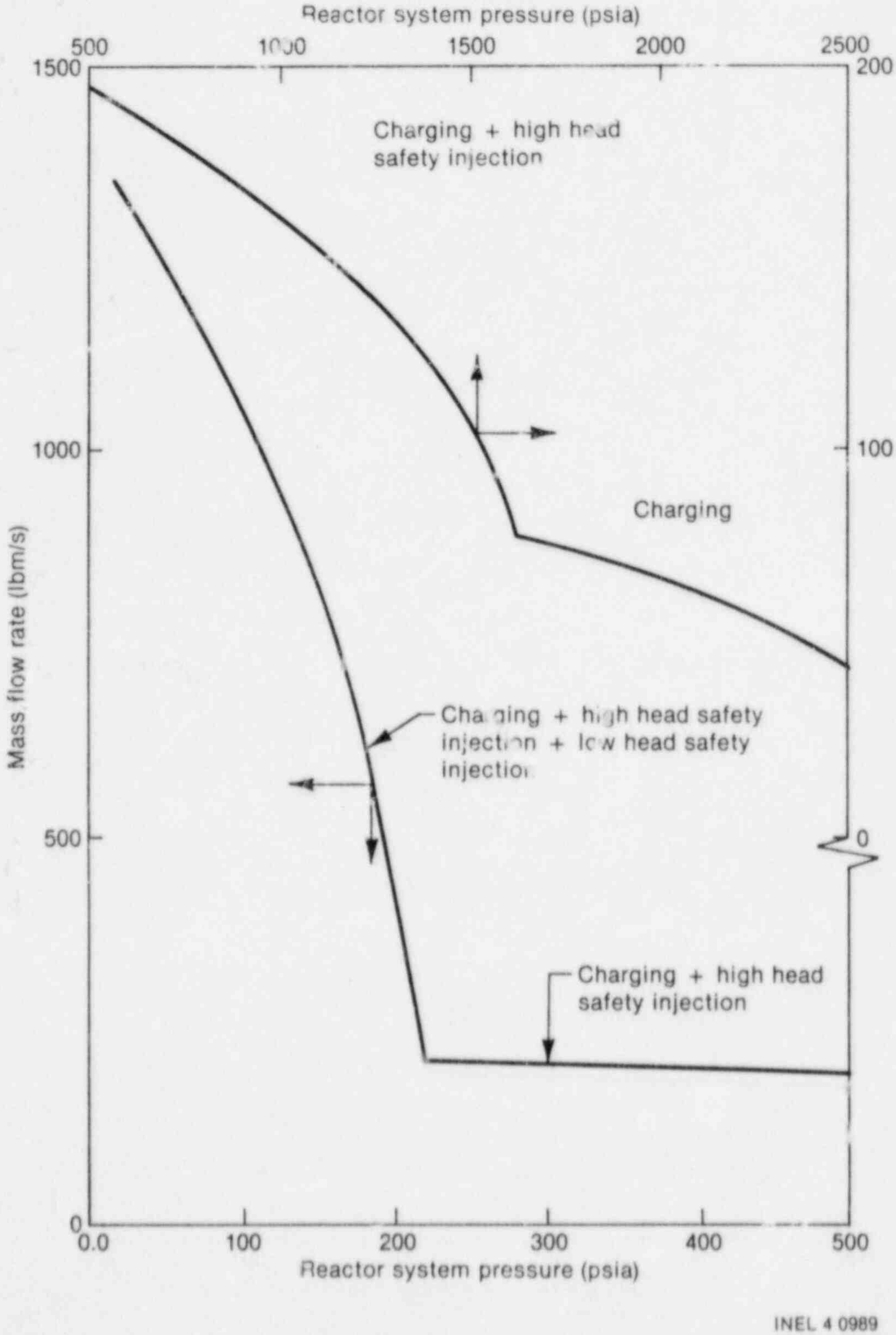


Figure 5-1. ECC injection capacity (best estimate assumption) for all systems operating and discharging into reactor coolant system.

6. SHUTDOWN MARGIN

This section provides a technique for estimating the immediately-available shutdown margin during normal reactor operation. The method depends upon the plant-specific information available for determining the power defect and the control-rod reactivity worths.

Estimating Power Defect and Control-Rod Worths

The power defect and available control-rod worths are estimated from available plant-specific information. These quantities are adjusted for the plant conditions and allowance is made for stuck rods. The shutdown margin is then simply determined by

$$SM = \Delta\varrho_{POWER} + \Delta\varrho_{CRA} \tag{6-1}$$

where

SM = shutdown margin. The amount of negative reactivity available in the control-rod assemblies (CRAs) over and above that required to compensate for Doppler and moderator temperature effects in bringing the reactor from any power level to hot standby. This includes the conservative assumption that the most reactive CRA is stuck and will not be inserted in the event of a reactor scram. It must be demonstrated that the shutdown margin is sufficient to show the reactor to be subcritical by an amount greater than some prescribed safety margin. The axial-power-shaping rods do not change position during a scram; therefore, no consideration is given to their worth in determining the shutdown margin.

$\Delta\varrho_{POWER}$ = reactivity change as the fuel and moderator conditions change from those at power to the conditions at hot standby. This reactivity quantity is the negative of the power defect and is usually positive.

$\Delta\varrho_{CRA}$ = change in reactivity resulting from the insertion of all available full-length control rods. This quantity is always negative.

These reactivity values should be obtained from data sources that are appropriate to reactor conditions or should be corrected to account for discrepancies such as power level, rod positions, redistribution (effect on reactivity due to changes in the neutron spatial distribution), boron concentrations, burnup in fuel and poison sections, and any anomaly in core conditions.

Available control-rod worth ($\Delta\varrho_{CRA}$) should not include any inoperable rods and should allow for the possibility of the most-reactive control rod or control-rod cluster (all the rods on driving mechanism) failing (stuck rod). Adjustments for inoperable rods would depend upon where the inoperable rods are currently positioned (e.g., if fully inserted, use the "dropped rod worth"; if fully withdrawn, use "stuck rod" worth; if partially inserted, the worth should be prorated).

The following steps should be taken:

1. Determine the value of $\Delta\rho_{POWER}$.

   1.a Determine the full-power power defect.

   1.b Correct to the actual power and change the sign (The power defect will carry a negative sign unless it has been omitted in the data, or unless this is a very unusual situation.)

2. Determine the value of $\Delta\rho_{CRA}$.

   2.a Determine the total worth of the safety-rod groups.

   2.b Determine the worth of the regulating CRAs.

   2.b.1 Determine the total worth of the CRAs (always a negative quantity).

   2.b.2 Adjust 2.b.1 to account for partial insertion of the CRAs. (This adjustment will cause the CRA worth to become less negative.)

   2.c Determine the number of inoperable rods and resulting worths. The "stuck rod" data may be used; however, if the rod is fully inserted the "dropped rod" data may be used or the worth may be prorated if the rod is partially inserted.

   2.d Determine the worth of the most-reactive "stuck" rod.

   2.e Determine the effect of redistribution (difference in actual control-rod worth from published data due to neutron-flux effects)

   $\Delta\rho_{CRA}$ = (2.a + 2.b) + (2.c + 2.d + 2.e).

   NOTE: The items in the first parenthetical field (2.a + 2.b) are both negative quantities. The adjustments in the second parenthetical field (2.c + 2.d + 2.e) are all positive corrections. The resultant $\Delta\rho_{CRA}$ is a negative quantity.

3. Apply Equation (6-1).

Assumptions

The following assumptions are made:

- Reactivity effects are separable and are additive.

- Reactivity effects due to changes in boron concentrations (expressed as weight fractions), burnup and Xe, Sm, and other fission products are negligible over the short period of time from normal operation at power to hot standby conditions.

Limitations

This procedure is limited to determination of the capability to return to hot standby from normal operating conditions and retain an adequate shutdown margin. This procedure is not used to determine shutdown margin after hot standby is obtained, nor is it to be used during borating and cooling procedures.

Examples of Technique Application

1. A PWR plant is operating at 80% of full power 240 effective full power days (EFPD) into Cycle 5 with negligible xenon

   1.a The power deficit at full power is estimated to be -2.83% $\Delta k/k$ by Figure 6-1

   1.b The fraction (f) of the power deficit at 80% of full power is estimated (using Figure 6-2) to be

$$f = 0.88 + \frac{240.0 \text{ EFPD (actual plant burnup) - 4.0 EFPD (lower curve)}}{(305.0 - 4.0) \text{ EFPD (difference between curves)}} \times (0.9 - 0.88) = 0.896,$$

assuming a linear relation between f and EFPD. Therefore,

$$\Delta\varrho_{POWER} = -(0.896)(-2.83\% \ \Delta k/k) = 2.54\% \ \Delta k/k.$$

2. The best data available for rod worths for this example are at $\Delta\varrho$(% $\Delta k/k$) beginning of cycle (BOC). These data are slightly conservative for all the rod groups except Groups 6 and 7, but final results will be conservative.

   2.a Safety control-rod assembly worth taken from Table 6-1 (sum of Groups 1-4) -5.12

   2.b Regulating CRAs

      2.b.1 Total worth of CRA Groups 5, 6, 7 (taken from Table 6-1) -3.48

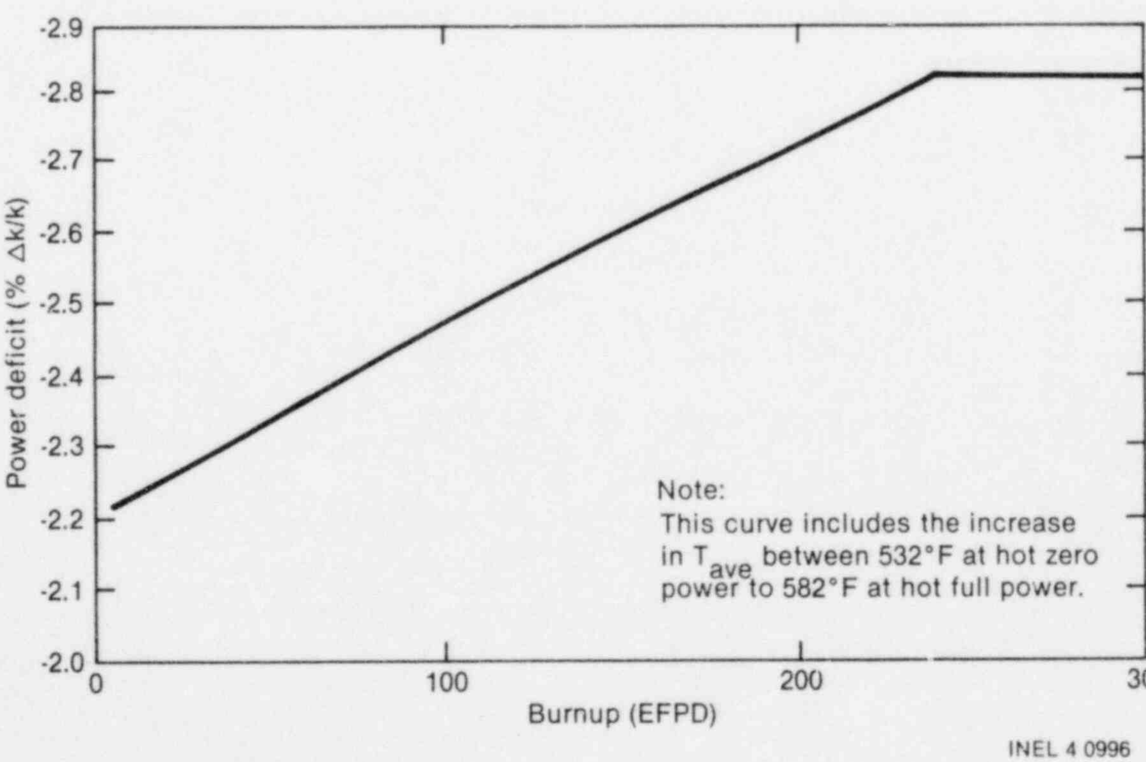


Figure 6-1. Power deficit at 100% full power for Cycle 5 of Rancho Seco Unit 1 as a function of burnup in effective full power days.[1]

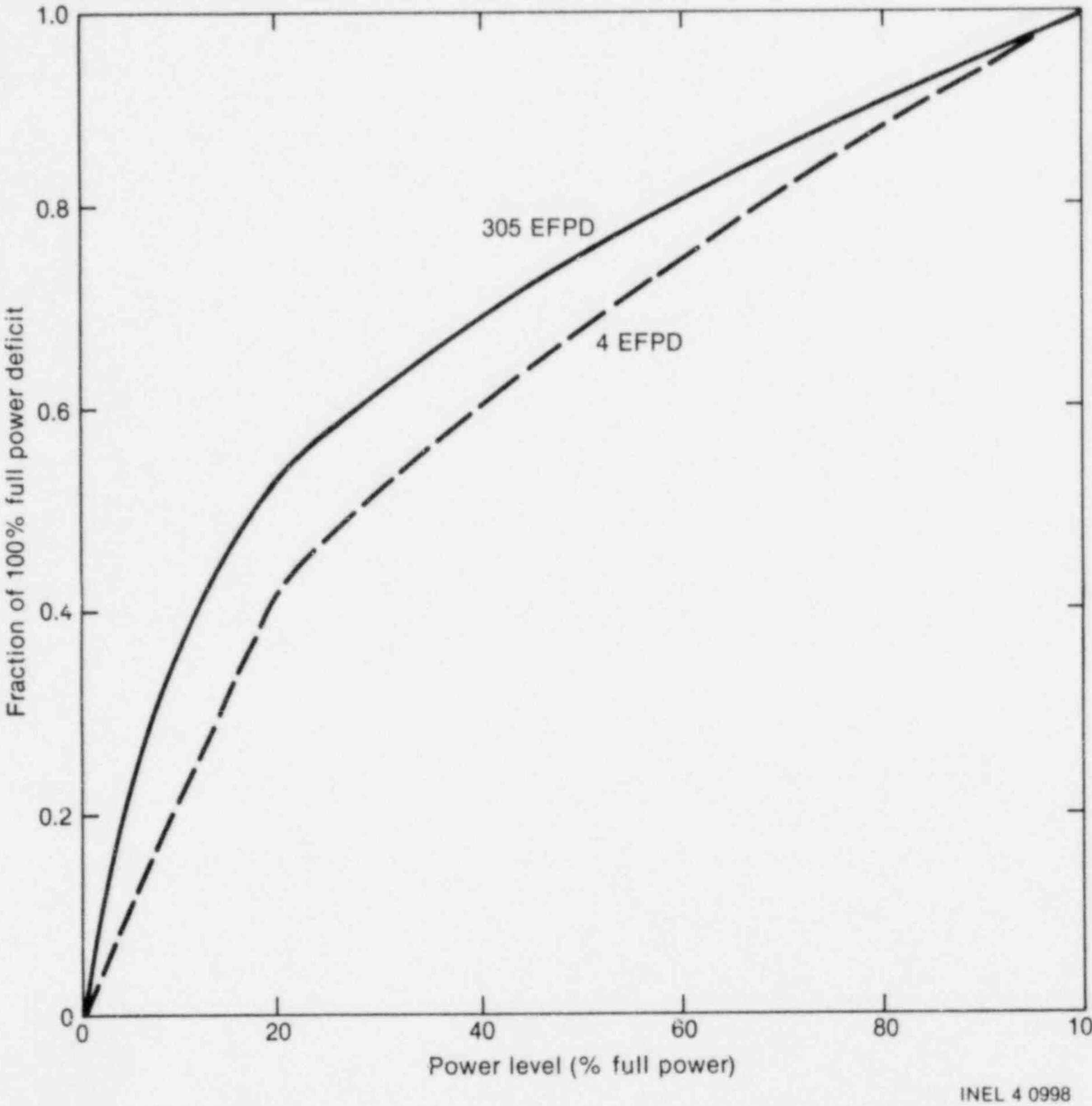


Figure 6-2. Fraction of 100% full power deficit versus power level for Cycle 5, Rancho Seco Unit 1 (Reference 1).

2.b.2	Less "bite" from Figure 6-3 (CRA Group 7 40% withdrawn)	-0.83
2.b.3	Total available regulating CRA worth (2.b.1 - 2.b.2)	-3.48 - (-0.83) = -2.65

2.c Inoperable Rod

For this example, let us assume that Group 6 is "stuck" in the fully-withdrawn position. Subtract a negative 0.96 (taken from Table 6-1). +0.96

Table 6-1. Zero power controlled rod group worths for T_{ave} = 532°F, cycles of PWR

Group	Number of CRAs	BOC-5 Worth (% Δk/k)	EOC-5 Predicted Worth (% Δk/k)	Purpose
1	8	1.302	5.304[b]	Safety
2	8	3.818[a]	—	Safety
3	5	—	—	Safety
4	12	—	—	Safety
5	8	1.096	1.259	Control
6	8	0.964	0.962	Control
7	12	1.422	1.407	Control
Total:		8.602	8.932	
8	8	0.331	NA	APSRs (minimum imbalance position)
TOTAL:	69	8.933	8.932	

a. Sum for Groups 2, 3, and 4.

b. Sum for Groups 1, 2, 3, and 4.

2.d "Stuck Out" CRA

(Taken from Figure 6-4) +1.83

2.e Maximum effect of redistribution

(Assumed for this example—although believed to be representative value, a positive value reduces available control-rod worth) +0.85

Total available control-rod worth (sum of Items 2.a through 2.e)

$\Delta\varrho_{CRA}$ = [-5.12 + (-2.65)] + (+0.96 + 1.83 + 0.85) = -4.13

3. Apply equation (6-1) as follows:

SM = 2.54 + (-4.13) = -1.59% Δk/k.

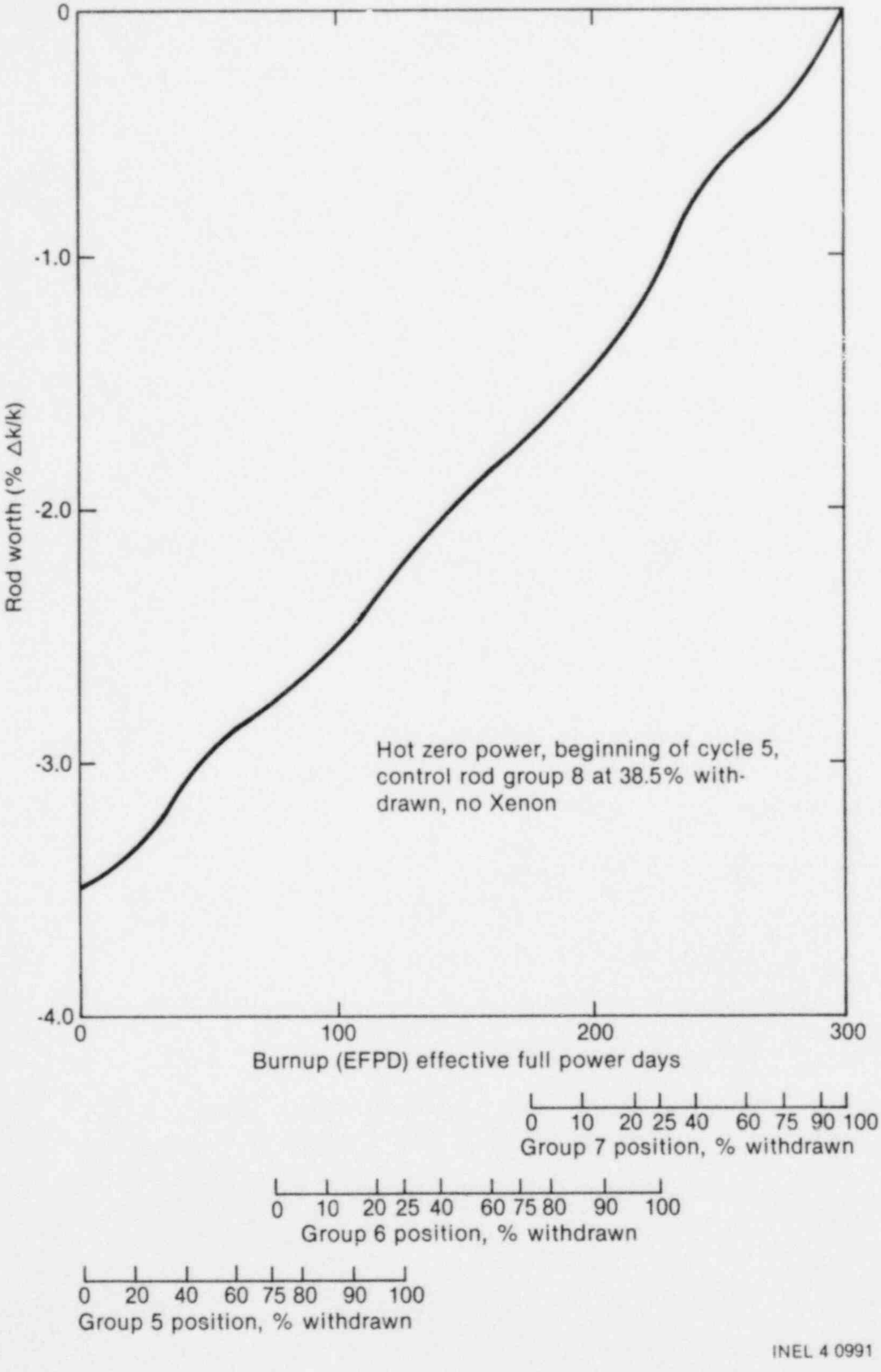


Figure 6-3. Controlling group rod worths Cycle 5, Rancho Seco Unit 1 (Reference 1).

This value of the shutdown margin is to be compared with the established safety margins. If, for example, the safety margin is -1.0% Δk/k, then the plant, in this example, is within the operational envelope and can safely remain at power.

Reference

1. *Rancho Seco Unit 1 Operations Manual*, Procedure B.6-Reactivity Balance Calculations, Rev. 21, July 20, 1982.

Appendix Information

Power Deficit and Control-Rod Worths:

Figure 6-1, Power Deficit at 100% FP
Figure 6-2, Fraction of 100% FP Power Deficit
Figure 6-3, Controlling Group Rod Worths
Figure 6-4, Stuck Rod Worth Versus Burnup
Table 6-1, Zero Power Control Rod Group Worths

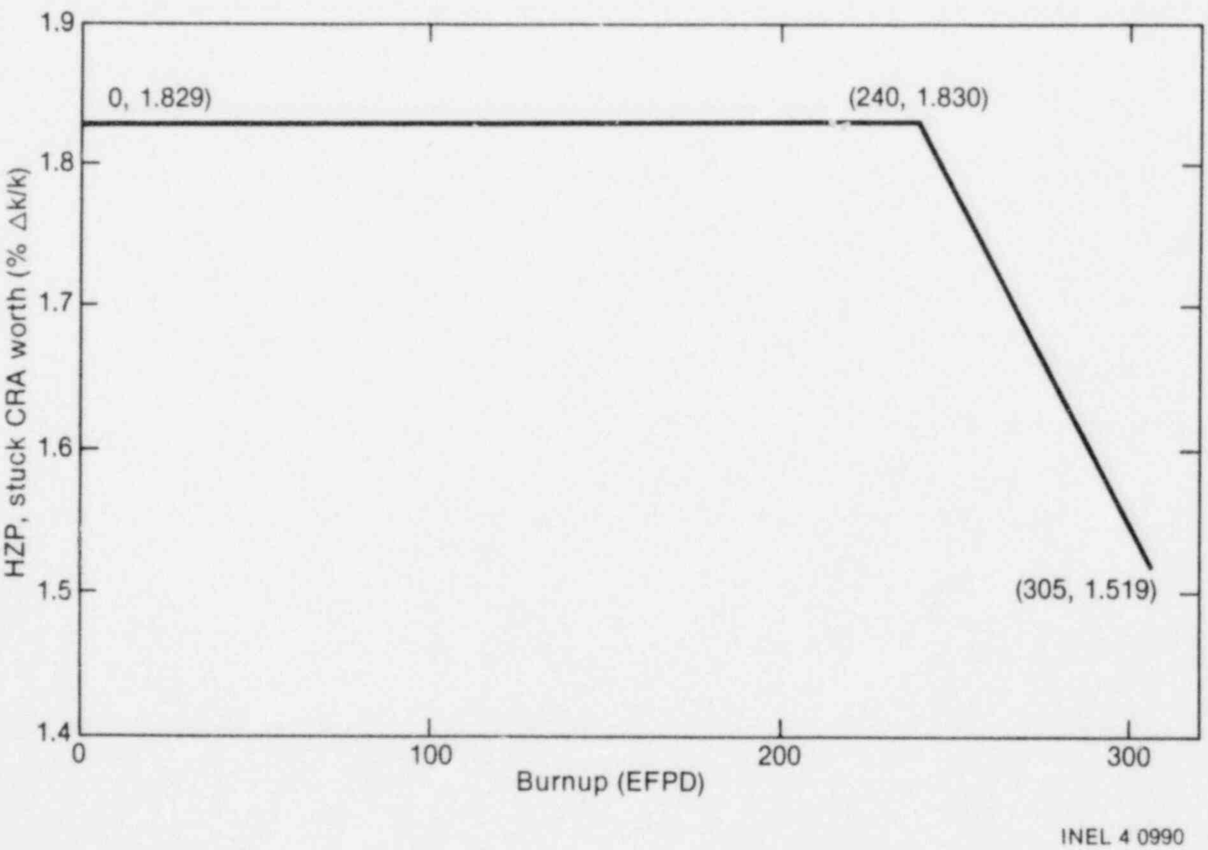


Figure 6-4. Stuck rod worth versus burnup for Cycle 5, Rancho Seco Unit 1 (Reference 1).

SYSTEM ANALYSIS HANDBOOK

7. NATURAL CIRCULATION

This section provides a technique for estimating the flow rate through a PWR or BWR core due to natural circulation. The core coolant temperature rise derived from the technique can be compared to measurements to aid confirmation of the assumed process.

Estimating Flow Rate Through a PWR Core

The natural circulation volumetric flow rate through a PWR core for single-phse fluid conditions has been correlated by the following expression[1]

$$\dot{Q} = a\left[\frac{2\beta g\ \Delta z\ P}{\varrho_o\ C_p\ R_e}\right]^{1/3} \tag{7-1}$$

where

$\dot{Q}$ = volumetric flow rate (ft^3/s)

a = unit conversion constant = 9.822 $[Btu/(MW\text{-}s)]^{1/3}$

β = temperature coefficient for volume expansion at average coolant temperature (1/°F)

g = acceleration of gravity = 32.17 ft/s^2

Δz = for U-tube steam generator, the elevation difference between axial midplane of core and axial midplane of active tube bundle; for once-through steam generator, the elevation difference between axial midplane of core and the axial location of the thermal center[2] (location of average temperature) (ft)

P = core power (MW)

ϱ_o = coolant density at average coolant temperature (lbm/ft^3)

C_p = coolant specific heat at constant pressure at average coolant temperature [Btu/(lbm-°F)]

R_e = equivalent resistance factor for coolant flow path ($1/ft^4$)

The resistance factor (R_e) is defined as follows:

$$\Delta P_N = \frac{1}{2}\frac{\varrho_o}{g_c} R_e\, \dot{Q}_N^2 \tag{7-2}$$

where

ΔP_N = total coolant pressure loss through system at nominal conditions (equal to pump head) (lbf/ft^2)

g_c = unit conversion factor = 32.17 $lbm/lbf\ ft/s^2$

$\dot{Q}_N$ = total volumetric flow rate at nominal conditions (ft^3/s)

The resistance factor can be determined for a multiloop PWR by evaluating resistance subfactors for the vessel and loop individually as follows:

$$\Delta P_N = \Delta P_v + \Delta P_L = \frac{\varrho_o}{2 g_c} R_v \dot{Q}_N^2 + \frac{\varrho_o}{2 g_c} R_L \left(\frac{\dot{Q}_N}{n} \right)^2 \tag{7-3}$$

where

$\Delta P_{v,L}$ = coolant pressure loss through vessel and loops at nominal conditions (lbf/ft^2)

$R_{v,L}$ = resistance subfactor for vessel and loops at nominal conditions

n = number of loops operating at nominal conditions.

The resistance factor (R_e) then for a condition where natural circulation is occurring through one or more loops is as follows:

$$R_e = 1/L^2 \left[R_L + K/A^2 \right] + R_v \tag{7-4}$$

where

L = number of loops through which natural circulation is occurring

K = equivalent hydraulic resistance of locked pump rotor[a]

A = pipe area consistent with K (ft^2).

Equation (7-4) is derived for a system with multiple complete loops, i.e., each loop containing a steam generator and a pump. For a system where a loop consists of two pumps in parallel that are also in series with a steam generator, the effective loop resistance of a locked rotor pump (K/A^2) must be halved ($K/2A^2$) for inclusion in Equation (7-4).

The average temperature rise across the core due to natural circulation can be determined from an energy balance

$$P = \varrho_o \dot{Q} C_p \Delta T/b \tag{7-5}$$

where

b = unit conversion factor = 948.1 Btu/(MW-s).

An equivalent expression to (7-1) for two-phase natural circulation[3] is

$$\dot{Q} = a \left[2 g \Delta z P/(\varrho_o h_{fg} R_e) \right]^{1/3} \tag{7-6}$$

a. The form loss coefficient (K) for a pump with flow in the normal direction can be derived from the HVN homologous curve.

where

h_{fg} = latent heat of vaporization (Btu/lbm).

Estimating Flow Rate Through a BWR Core

A simple correlation has not been developed for BWR application. Reasonable results have been obtained from a simple computer code representing the BWR system.[4] Figure 7-1 presents a comparison of data with calculations for several BWR systems (BWR/1, BWR/2, BWR/3 and BWR/4). The curve presents the average natural circulation flow rate plotted against the average power per fuel bundle with the normal inventory of coolant. Ranges of uncertainty in power and flow rate are included on the plot. The curves apply to any plant within the system category.

For application of the data presented to a specific plant system, the average power per fuel bundle must be determined from

$$\bar{P}_b = \frac{P}{N_b} \tag{7-7}$$

where

$\bar{P}_b$ = is the core average bundle power (MW)

N_b = number of bundles in core.

The total core power (P) is either the operating power level or the decay heat power obtained with the methodology detailed in Section 3.

For a condition of decreasing coolant inventory, calculations for natural circulation have been performed for a BWR/4, Peach Bottom. The results are shown in Figures 7-2 and 7-3. The results shown in Figure 7-2 show the percent of rated core flow as a function of the downcomer water level and core power. To apply this curve to a BWR/3 or 4, a measured level is needed or it can be estimated using the mass balance technique of Section 5.

For the results shown in Figure 7-3, the system is isolated. Pressure is maintained with mass exiting the safety valves. The reactor is scrammed, the recirculation pump is tripped, but neither the reactor coolant inventory control system nor the emergency core cooling systems are activated. Figure 7-3 shows the downcomer water level as a function of the time after scram. Superimposed on the figure are the times when the water level reaches the top of the fuel bundles (∼1000 s) causing circulation through the upper plenum to cease. Core uncovery begins about ∼1500 s later, and overheating of the hottest fuel rod begins in another 20 min.

Assumptions

A heat sink (steam generator secondary for PWR) is available to reject heat. Average quantities may be used to represent the parameters needed in the equations correlating the flow.

Flow resistances computed for forced flow conditions are assumed applicable to the natural circulation phenomenon. However, as the flow velocity profiles are different at the wall for the two flow regimes, flow resistances computed from forced convection correlations do not strictly apply.

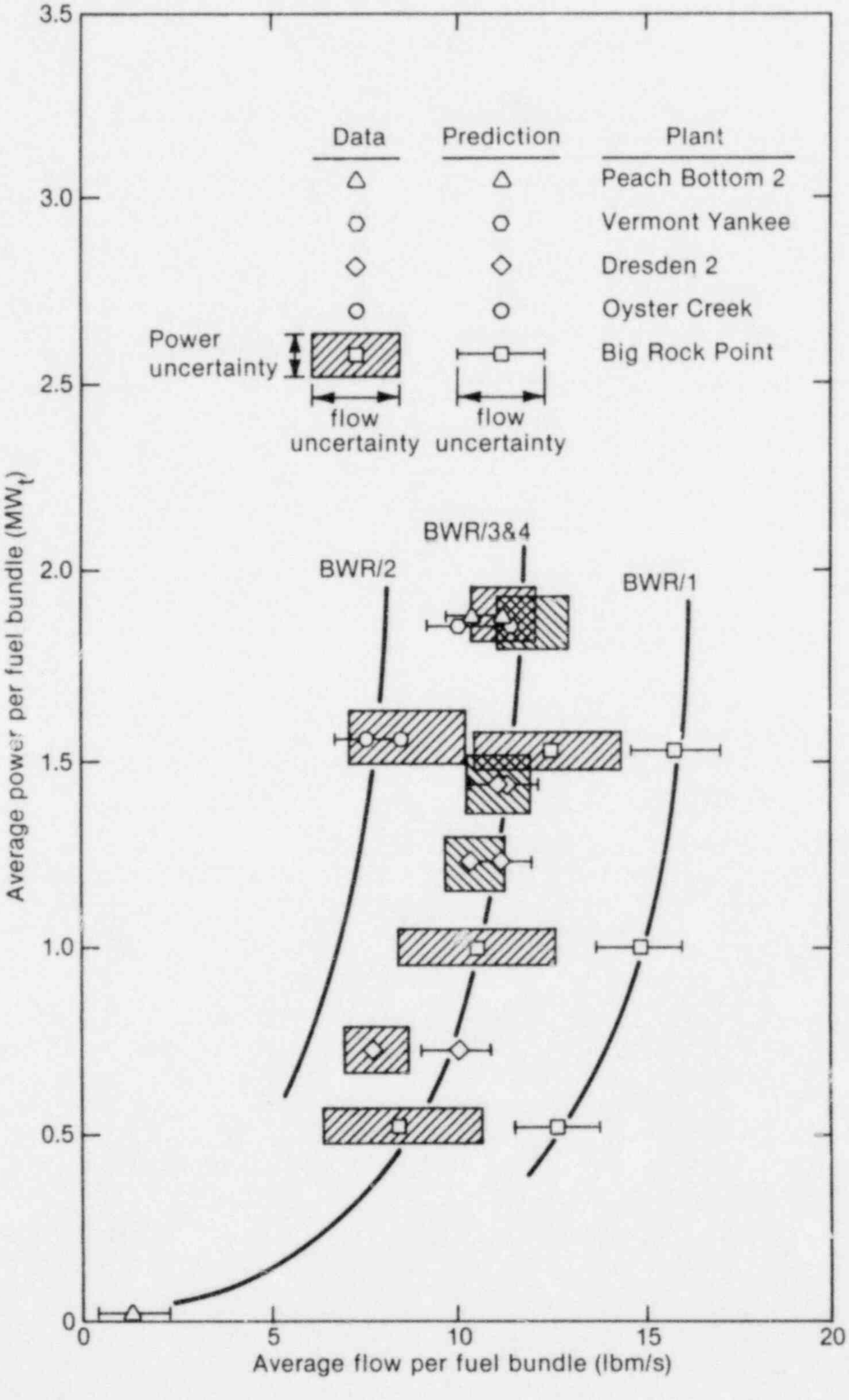


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Figure 7-1. Comparison of data and predictions of natural circulation for various BWRs.

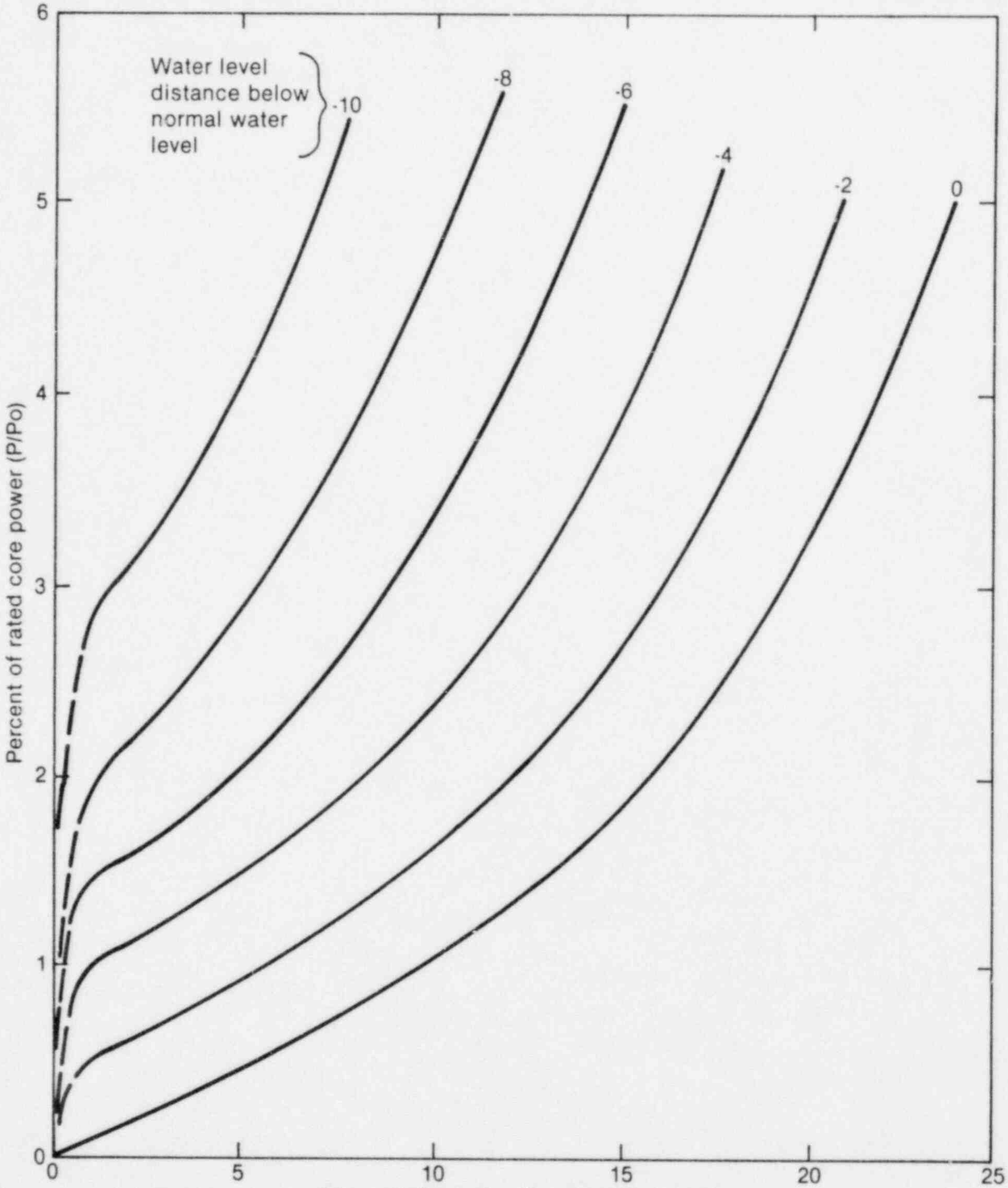

Figure 7-2. Natural circulation flow at various water levels active coolant core flow (% rated) versus core power (% rated).

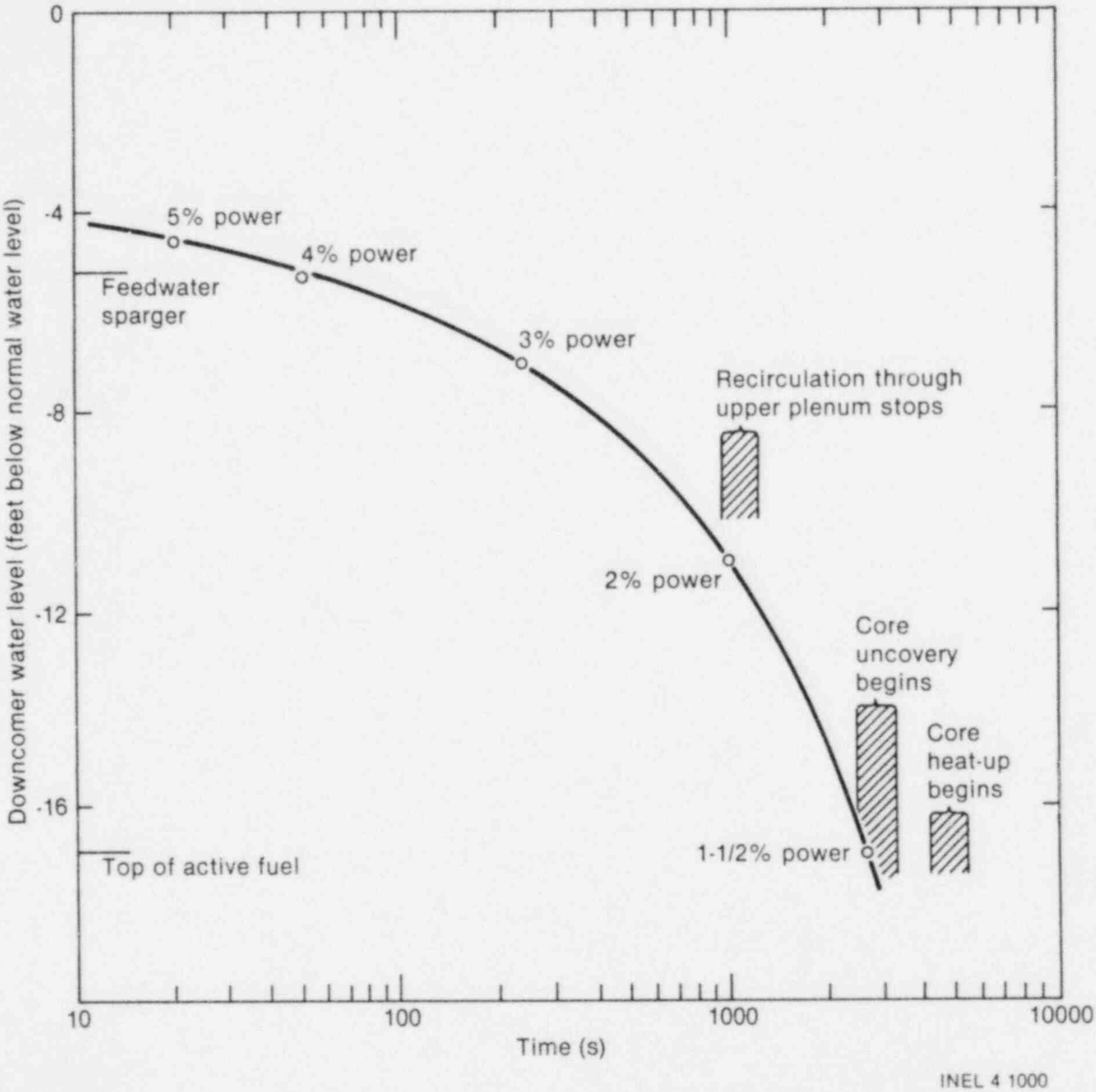

Figure 7-3. Estimated Peach Bottom 2 downcomer level history.

Limitations

For single-phase natural circulation flow, the technique presented compares to data from a variety of loops and PWRs within 30%.

For two-phase natural circulation flow, very imited data have been compared to the technique presented. It has been noted that effective natural circulation has occurred with maximum void fractions up to 0.55 in a PWR and at higher void fractions when a heat sink exists in the steam generator.[5]

Example of Technique Application

A large four-loop PWR undergoes an incident resulting in scram, shutdown of all pumps, and termination of all auxiliary coolant sources and sinks. The reactor had been operating for a very long time. The nominal full power conditions of interest were as follows:

- Loop pressure drop (ΔP_L) = 46.5 psia
- Vessel pressure drop (ΔP_v) = 46.2 psia
- Total nominal mass flow rate = 1.403 x 10^8 lbm/h
- Pump form loss (K) = 11.49, based on inlet area of 5.22 ft^2
- Power (P) = 3425 MW
- Average coolant temperature = 588°F
- Nominal pressure = 2250 psia
- Elevation between core and steam generator midplanes (Δz) = 33.0 ft

What is the expected temperature difference across the core 1 h after scram?

The vessel and loop flow resistance subfactors are calculated first using Equation (7-3)[a]

$$R_v = \frac{2\Delta P_v\ g_c}{\varrho_o\ \dot{Q}_N^2}$$

$$= \frac{2.0 \times 46.2\ \text{lbf/in.}^2 \times 144\ \text{in.}^2/\text{ft}^2 \times 32.17\ \text{lbm/lbf ft/s}^2}{44.15\ \text{lbm/ft}^3\ (\rho_o \text{ at 2250 psia, 588°F}) \times \left[\dfrac{1.403 \times 10^8\ \text{lbm/h}}{3600\ \text{s/h}} \times \dfrac{1}{44.15\ \text{lbm/ft}^3}\right]^2}$$

$$= 0.01244\ 1/\text{ft}^4$$

and

$$R_L = \frac{2\Delta P_L\ g_c\ n^2}{\varrho_o\ \dot{Q}_N^2}$$

$$= \frac{2.0 \times 46.5\ \text{lbf/in.}^2 \times 144\ \text{in.}^2/\text{ft}^2 \times 32.17\ \text{lbm/lbf ft/s}^2 \times 16.0}{44.15\ \text{lbm/ft}^3\ (\rho_o \text{ at 2250 psia, 588°F}) \times \left[\dfrac{1.403 \times 10^8\ \text{lbm/h}}{3600\ \text{s/h}} \quad \dfrac{1}{44.15\ \text{lbm/ft}^3}\right]^2}$$

$$= 0.2037\ 1/\text{ft}^4$$

a. Precalculated values for the equations are found in the appendix information.

Then (R_e) is calculated as follows from Equation (7-4)

$$R_e = 1/L^2 \, [R_L + K/A^2] + R_v$$

$$= 1/16 \, [0.2037 \; 1/\text{ft}^4 + 11.49/(5.22 \; \text{ft}^2)] + 0.01244 \; 1/\text{ft}^4$$

$$= 0.0513 \; 1/\text{ft}^4.$$

Next, the decay heat power is needed. Infinite operation is assumed and a time of 60 min is entered in Figure 3-1 to obtain the decay heat fraction of 0.01558 MW per MW of the steady-state operating level. Thus,

$$P = 3425 \text{ MW} \times 0.01558 \text{ MW/MW} = 54.39 \text{ MW} .$$

Equation (7-1) can now be solved for the volumetric flow rate:

$$\dot{Q} = 9.822 \left[\frac{2 \times 11.76 \times 10^{-4} \; 1/^\circ\text{F} \times 32.17 \; \text{ft/s}^2 \times 33.0 \; \text{ft} \times 54.39 \; \text{MW}}{44.15 \; \text{lbm/ft}^3 \times 1.39 \; \text{Btu/(lbm-}^\circ\text{F)} \times 0.0513 \; 1/\text{ft}^4} \right]^{1/3} = 34.46 \; \text{ft}^3/\text{s}$$

The average temperature rise across the core is then found from the heat balance expression (7-5)

$$\Delta T = bP/\varrho_o \, \dot{Q} \, C_p$$

$$= \frac{948.1 \; \text{Btu/(MW-s)} \times 54.39 \; \text{MW}}{44.15 \; \text{lbm/ft}^3 \times 34.46 \; \text{ft}^3\text{/s} \times 1.39 \; \text{Btu/(lbm-}^\circ\text{F)}} = 24.4^\circ\text{F} .$$

References

1. Zvirin, Y., *A Review of Natural Circulation Loops in Pressurized Water Reactors and Other Systems*, Electric Power Research Institute, EPRI NP-1676-SR, January 1981.

2. Babcock and Wilcox, *Abnormal Transient Operator Guidelines, Bellefonte Plant 1 and 2, Part 1, Procedural Guidelines*, 74-1135402-00, June 1976.

3. Adams, J. P., G. E. McCreery, and V. T. Berta, "Natural Circulation Cooling Characteristics During PWR Accident Simulations," *Thermal-Hydraulics of Nuclear Reactors, Volume II, Proceedings of Second International Topical Meeting on Nuclear Reactor Thermal-Hydraulics, Santa Barbara, California, January 11-14, 1983*.

4. Healzer, J. M. and P. G. Bailey, "National Circulation in Boiling Water Reactors," *Thermal-Hydraulics of Nuclear Reactors, Volume II, Proceedings of Second International Topical Meeting on Nuclear Reactor Thermal-Hydraulics, Santa Barbara, California, January 11-14, 1983*.

5. Loomis, G. G. and K. Soda, *Results of the Semiscale MOD-2A Natural Circulation Experiments*, NUREG/CR-2336, September 1982.

Appendix Information

Specific values for equations:

β = 11.76×10^{-4} 1/°F, nominal temperature coefficient for volume expansion at average coolant temperature of 588°F

Δz = 33.0 ft, elevation difference between midplane or core and steam generator

ϱ_o = 44.15 lbm/ft^3, coolant density at nominal average temperature of 588°F

C_p = 1.39 Btu/lbm-°F, coolant specific heat at constant pressure at average temperature of 588°F

R_ℓ = 0.0513 1/ft^4, equivalent resistance for coolant flow path with all loops operating

Q_N = 882.72 ft^3/s, total volumetric flow rate at nominal conditions

R_L = 0.2037 1/ft^4, resistance subfactor for loops

R_V = 0.01244 1/ft^4, resistance subfactor for vessel

K = 11.49, flow coefficient for locked rotor pump based on inlet pipe area, A = 5.22 ft^2

ΔP_L = 46.5 psia, coolant pressure loss through loops at nominal conditions

ΔP_V = 46.2 psia, coolant pressure loss through vessel at nominal conditions

8. NONCONDENSABLE GAS GENERATION

This section describes a technique for estimating the amount of hydrogen generated in the primary system of a light water reactor due to (a) metal-water reaction, (b) radiolysis of water about the core, and (c) hydrogen off-gassing due to depressurization.

Metal-Water Reaction

The mass of hydrogen generated by a Zircaloy surface reacting with water or steam at a constant, uniform temperature is described by the expression[1]:

$$m_H = A_s \left[2A_r \exp\left(-B/RT_{R\text{-}Zr}\right)t\right]^{1/2} \tag{8-1}$$

where

m_H = mass of hydrogen generated (lbm)

A_s = total surface area of Zircaloy at temperature ($T_{R\text{-}Zr}$) exposed to water (ft^2)

A_r = rate constant for release of hydrogen[a]

= $0.0112\ lbm^2/(ft^4\text{-}s)$ for $T_R \leq 3335$ R (2875°F)

= $0.00362\ lbm^2/(ft^4\text{-}s)$ for $T_R > 3335$ R (2875°F)

B/R = activation energy/universal gas constant

= 3.611×10^4 R for $T_R \leq 3335$ R (2875°F)

= 2.990×10^4 R for $T_R > 3335$ R (2875°F)

$T_{R\text{-}Zr}$ = absolute temperature of Zircaloy surface [R (°F + 460)]

t = time at temperature (s).

The volume occupied by the mass of hydrogen generated is estimated by application of the perfect gas expression

$$V = m_H\ (IR/m)\ T_g/dP \tag{8-2}$$

where

V = total volume of hydrogen gas generated (ft^3)

a. The rate constants are derived by multiplying the constants for weight gain (EGG-CDD-5647[2]) by the square of the molecular weight ratio of $2H_2/O_2$ and converting to engineering units.

IR/m = universal gas constant/molecular weight of hydrogen

= 766.6 (ft-lbf)/(lbm-R)

T_g = absolute average temperature of hydrogen gas (R)

d = conversion constant = 144 in.^{2}/ft^2

P = absolute pressure of hydrogen gas (lbf/in.^{2} or psia)

The mass of hydrogen generated per lbm of zirconium reacting with water is 0.0439 lbm H_2/lbm Zr.

Radiolysis Method

The mass of hydrogen generated due to the radiolytic decomposition of reactor coolant in the primary system may be estimated by application of the information contained in Figure 8-1. This plot is for a 3300 MW(t) reactor using the net rate constant for pure water, and assuming all decay radiation reacts with the coolant.

The results presented may be used to estimate the hydrogen generated for other plants as the hydrogen generated is directly proportional to integrated power.

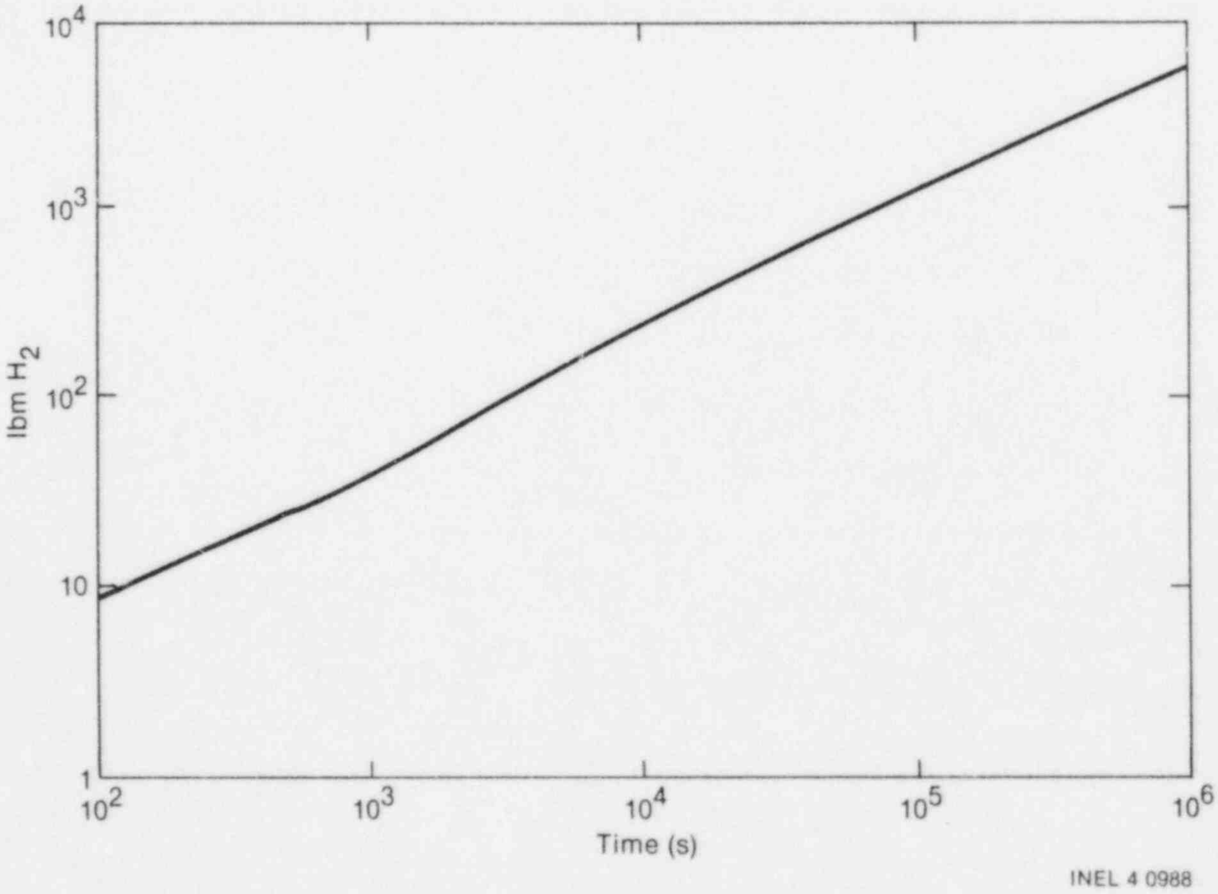

Figure 8-1. Conservative calculation (Reference 1) for radiolysis yield of hydrogen for a 3300 MW(t) reactor.

Depressurization Off-Gassing

The mass of hydrogen released from solution with the primary coolant of a PWR during a change in state conditions may be estimated by application of the data presented in Figure 8-2. The mass released from solution at a specific temperature and pressure change may be obtained as follows (Reference 1):

$$m_H = C\left[DHS_{(1)} - DHS_{(2)}\right]M_W \tag{8-3}$$

where

$DHS_{(1,2)}$ = dissolved hydrogen at saturation expressed as a volume at standard conditions at statepoints 1 and 2

$$\left[\frac{cm^3}{kg}\ \frac{(hydrogen)}{(water)}\right]$$

C = conversion constant

$$= \frac{0.0824\ \ kg/m^3}{10^6\ \ cm^3\ /m^3}\ \text{(density of hydrogen at standard conditions)}$$

$$= 0.824 \times 10^{-7}\ kg/cm^3$$

M_W = mass of water in primary system (lbm).

Assumptions

Unlimited steam is available for the Zircaloy-wat_r reaction. Hydrogen gas released does not blanket the Zircaloy and inhibit the reaction. Zircaloy surface temperatures must exceed 1600°F before reaction rates are significant.

During a reactor depressurization, the primary system is an open system. That is, hydrogen released from the water coolant is not limited by the solubility limits of the gas in water. Boiling or agitation of the coolant exists such that the hydrogen and oxygen generated are released from the coolant and do not recombine to form water. The effect of impurities on the solubility of hydrogen gas in liquid water is small and is neglected.

A PWR system maintains the primary coolant in a state where the coolant is saturated with hydrogen during normal operation.

BWR primary systems continually remove generated hydrogen and oxygen gas and, therefore, little hydrogen is dissolved in the primary system liquid during normal operation.

Limitations

The time and spacial distribution of the Zircaloy cladding and structural material surface temperatures must be lumped into averag^s for application of Equation (8-1). The Zircaloy-water reaction equations are not valid above ∿3400°F, where a geometry change would occur with melting.

The effect of impurities, boron, and fission products may change the chemical reaction rates for hydrogen generated by radiologic decomposition of water. These effects are not understood and are not considered.

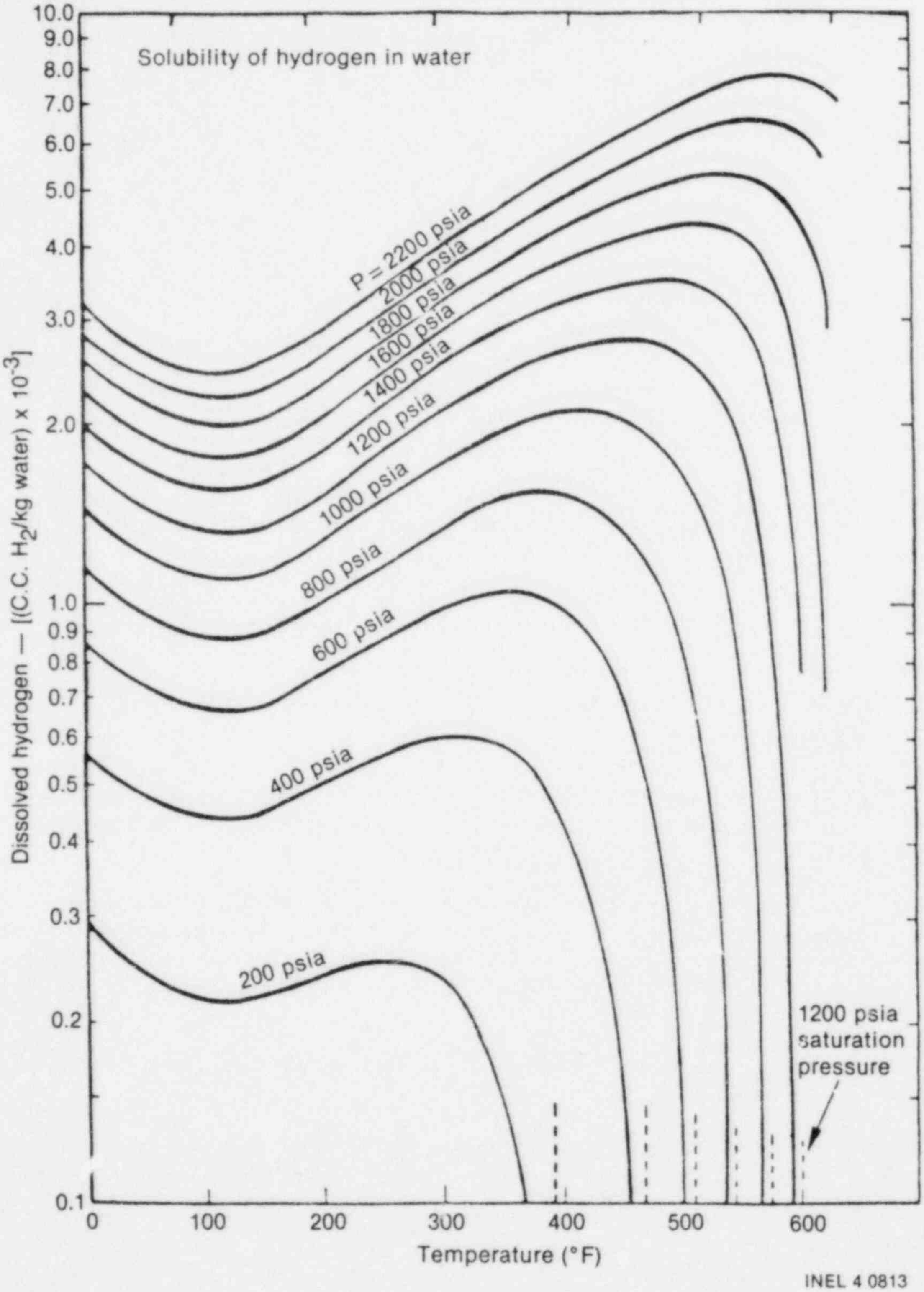

Figure 8-2. Hydrogen concentration in water under various conditions (Reference 1).

The generation of hydrogen gas by radiolytic decomposition presented in Figure 8-1 is likely conservative but the degree of conservatism cannot be determined.

The rate equation for the Zircaloy-water reaction equation up to 2875°F agrees with the data within 3 to 4% to one standard deviation.

Example of Technique Application

A large four-loop PWR experiences an incident during which the pressure decreases abruptly to 800 psia and after 30 min, coolant is lost from the core. The cladding surface temperature for the middle third of the core increases to a constant 2000°F while the cladding surface temperature of the outer two-thirds increases to a constant 1800°F for another 30 min. The reactor was operating at 3425 MW for several years prior to the incident. Scram occurs with the incident initiation. How much hydrogen gas was released at the end of the 60-min period? Does it fill the upper head and plenum volumes? How much more hydrogen could be released?

Hydrogen is first released from the primary system coolant volume due to the change in state from the initial pressure (∿2200 psia) and temperature (∿588°F average) to the final state point (P = 800 psia, T = 518°F) corresponding to saturated liquid.

Equation (8-3) is applied as follows (From the "Appendix Information," the primary system liquid volume is computed excluding the accumulator lines and accumulators.):

• Vessel	4586.11
• Piping(4)	1778.36
• Steam generator(4)	3848.0
• Pressurizer	1066.0
• Surge line	53.2
• Total volume	11,331.6 ft^3

The primary system coolant mass is next estimated using an average density corresponding to the average coolant temperature as follows:

M_W = 44.15 lbm/ft^3 (average density) x 11,331.6 ft^3 = 500,290 lbm.

The dissolved hydrogen is next found at the statepoints from Figure 8-2 as follows:

$DHS_{(1)} \cong 7.4 \times 10^3\ cm^3/kg$ (volume of hydrogen per unit mass of liquid water at

P = 2200 psia, T = 588°F).

$DHS_{(2)}$ = ∿0.0 (P = 800 psia, T = 518°F).

Thus, the mass of hydrogen released by depressurization is

$$m_H = 0.824 \times 10^{-7}\ kg/cm^3\ [(7.4 - 0.0) \times 10^3 cm^3/kg]\ 500{,}290\ lbm\ (water)$$

$$= 305.0\ lbm\ (hydrogen)\ .$$

The mass of hydrogen generated by radiolysis during the 30-min period when the core was covered is determined from Figure 8-1 to be ~70 lbm. Correcting by the ratio of steady-state full power, the gas mass is

$$m_H = \frac{3425 \text{ MW (plant power)}}{3300 \text{ MW (figure basis)}} \times 70 \text{ lbm} = 72.6 \text{ lbm} .$$

The hydrogen released due to Zircaloy-water reaction is determined from Equation (8-1). The additional information needed is the exposed Zircaloy surface area at each temperature.

The total Zircaloy surface area exposed to the steam environment (fuel rods and outside surface of guide thimbles) is

$$A_s \text{ (fuel rods)} = 50{,}952 \text{ (number of fuel rods)}$$

$$\times \pi \times 0.374 \text{ in./12 in./ft (fuel rod diameter)}$$

$$\times 151.6 \frac{\text{in.}}{12 \text{ in./ft}} \text{(fuel rod length)} = 63{,}026 \text{ ft}^2$$

$$A_s \text{ (guide thimbles)} = 193 \text{ (number of fuel bundles)} \times 25 \left(\frac{\text{guide thimbles}}{\text{bundle}}\right)$$

$$\times \pi \times 0.482 \frac{\text{in.}}{12 \text{ in./ft}} \text{ (outside diameter of guide thimble)}$$

$$\times 151.6 \frac{\text{in. (length)}}{12 \text{ in./ft}} = 7691 \text{ ft}^2 .$$

The mass of hydrogen from the third of the core at 2000°F (assuming guide thimbles and cladding are at the same temperature) is

$$m_H = \left[\frac{63{,}026 \text{ (cladding)} + 7691 \text{ (guide thimble)}}{3}\right] \text{ft}^2$$

$$\times \left[2 \times 0.0112 \frac{\text{lbm}^2(H_2)}{\text{ft}^4\text{-s}} \exp -\left(\frac{3.611 \times 10^4 \text{R}}{2460 \text{ R}}\right) 1800 \text{ s}\right]^{1/2} = 97.2 \text{ lbm.}$$

The mass of hydrogen from the two-thirds of the core at 1800°F is

$$m_H = \frac{2}{3}(63{,}026 + 7{,}691) \times \left[2 \times 0.0112 \exp -\left(\frac{3.611 \times 10^4}{2260}\right) 1800\right]^{1/2}$$

$$= 101.5 \text{ lbm.}$$

The total hydrogen gas generated to this point in time is

$$m_{H(total)} = (97.2 + 101.5) \text{ lbm (from Zircaloy reaction)}$$

$$+ 305.0 \text{ lbm (from depressurization)}$$

$$+ 72.6 \text{ lbm (from radiolysis)}$$

$$= 576.3 \text{ lbm .}$$

The volume occupied by this mass at 800 psia is estimated from Equation (8-2) as

$$V = 576.3 \text{ lbm (hydrogen)} \times 766.6 \text{ Btu/(lbm - R)}$$

$$\times \frac{2460 \text{ R (estimated gas temperature)}}{144 \text{ in.}^2/\text{ft}^2 \times 800 \text{ lbf/in.}^2} = 9434 \text{ ft}^3 .$$

The generated hydrogen mass would fill the upper plenum and upper head whose combined volume (found in "Appendix Information") is

$$V_{\text{(upper head)}} + V_{\text{(upper plenum)}} = 1573 \text{ ft}^3 .$$

Additional hydrogen that could be released would come from the Zircaloy-water reaction since for the assumed incident all hydrogen initially contained in the liquid has been released and little water remains in the vessel to react with the core radiation. The maximum hydrogen that could be released from the zirconium is shown as

$$m_H \text{ (maximum from Zircaloy)} = m_{Zr} \text{ (mass Zircaloy)} \times C \text{ (conversion constant)}$$

The mass of Zircaloy in the vessel is shown as

$$m_{Zr} = 46{,}030 \text{ lbm (fuel rods)} + 3{,}966 \text{ lbm (guide thimbles)} = 49{,}996 \text{ lbm}.$$

The total hydrogen available from the Zircaloy reaction is

$$m_H = 49{,}966 \text{ lbm (Zircaloy)} \times 0.0439 \text{ lbm } H_2\text{/lbm Zr} = 2194 \text{ lbm (hydrogen)} .$$

Thus,

$$2194 \text{ lbm (total available)} - (97.2 + 101.5) \text{ lbm (generated)}$$

$$= 1995.3 \text{ lbm (hydrogen that could be generated).}$$

References

1. Camp, Allen L. et al., *Light Water Reactor Hydrogen Manual*, NUREG/CR-2726, SAND 82-1137, August 1983.

2. Hampton, N. L. and D. L. Hagrman, *Cladding Oxidation (CORROS, COBILD, COXIDE, COXWTK, and COSTHK)*, EGG-CDD-5647, November 1981.

Appendix Information

1. Specific values for equations:

$A_{s(\text{fuel rods})}$	=	63,026 ft^2, Zircaloy surface area of fuel rods
$A_{s(\text{guide thimbles})}$	=	7691 ft^2, Zircaloy surface area of guide thimbles
M_W	=	500,290 lbm, mass of water in primary system.

2. Fuel rods:

Zircaloy mass in fuel rods	46,030 lbm
Number of fuel rods	193 bundles at 264 rods/bundle = 50,952

3. Guide thimbles:

Zircaloy mass	3966 lbm
Number	193 bundles @ 25 guide thimbles/bundle = 482

4. Coolant:

Nominal average coolant temperature	588°F

5. Volumes:

$V_{(\text{upper head})}$	=	878.1 ft^3
$V_{(\text{upper plenum})}$	=	695.2 ft^3

SYSTEM ANALYSIS HANDBOOK

9. DOSE CALCULATION

This section describes a technique for calculating a rough estimate of doses received at the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) following a release from a reactor.

Estimating Doses

The information required is as follows:

- Atmospheric dispersion factors, $\left(\frac{\chi}{Q}\right)_{EAB}$ and $\left(\frac{\chi}{Q}\right)_{LPZ}$ for the Exlusion Area Boundary and the Low Population Zone, respectively from Appendix B, Item 9.

- Release rate of radioactivity ($\dot{Q}$) in Ci/unit time, from plant or post-incident report.

- Duration of release (t_R) in a time unit consistent with the release rate, from plant or post-incident report.

The procedure is as follows:

- Obtain information from the sources described above.

- Calculate total activity release (Q) from the following:

$$Q = \dot{Q}\, t_R \text{(curies or Ci)} \tag{9-1}$$

- Insert parameters into the following equations to calculate the doses for the Exclusion Area Boundary and Low Population Zone, respectively:

$$D_\gamma = 0.127^a \left(\frac{\chi}{Q}\right) Q \quad \text{(Reference 1)} \tag{9-2}$$

$$D_s = 0.201 \left(\frac{\chi}{Q}\right) Q \tag{9-3}$$

$$D_t = 55.302 \left(\frac{\chi}{Q}\right) Q \tag{9-4}$$

where

D_γ = whole body gamma dose (rem)

D_s = skin dose (rem)

D_t = thyroid dose (rem).

The constants in the equations have the units of (rem-m^3)/(s-Ci).

a. Constants are activity-weighted dose conversion factors based on conservative isotope inventories within the core

Assumptions

Use of the atmospheric dispersion factors (χ/Q's) listed in the Appendix will be very conservative leading to higher-than-realistic doses. Actual (χ/Q's) from plant should be used when available to recalculate dose rates.

Isotopic releases are based on conservative reactor inventories which will actually change from plant to plant based on core life, power history, etc. It is assumed that the released radioactivity contains isotopes in the same proportions as found in the reactor.

Example of Technique Application

A release of radioactivity has occurred at a large four-loop PWR. A release rate of 10×10^{-6} Ci/s has occurred for 45 min. What approximate doses are received at the EAB and LPZ?

The atmospheric dispersion factors are taken from the Appendix:

$\left(\frac{\chi}{Q}\right)_{EAB} = 2.67 \times 10^{-4}$ s/m^3 (Reference 2)

$\left(\frac{\chi}{Q}\right)_{LPZ} = 1.31 \times 10^{-4}$ s/m^3

Then the reactivity released is calculated from Equation (9-1) as

$Q = 10 \times 10^{-6}$ Ci/s x 45 min x 60 s/min = 0.027 Ci released.

The doses at EAB are found from Equations (9-2, -3, -4) as

$D_\gamma = 0.127$ (rem-m^3)/(s-Ci) x 2.67×10^{-4} s/m^3 x 0.027 Ci = 0.915×10^{-6} rem for the whole body.

$D_s = 0.201$ (rem-m^3)/(s-Ci) x 2.67×10^{-4} s/m^3 x 0.027 Ci = 1.449×10^{-6} rem for the skin.

$D_t = 55.302$ (rem-m^3)/(s-Ci) x 2.67×10^{-4} s/m^3 x 0.027 Ci = 0.399×10^{-3} rem for the thyroid.

Then, the doses at the LPZ are found in a similar manner as

$D_\gamma = 0.127 \times 1.31 \times 10^{-4} \times 0.027 = 0.449 \times 10^{-6}$ rem

$D_s = 0.201 \times 1.31 \times 10^{-4} \times 0.027 = 0.711 \times 10^{-6}$ rem

$D_t = 55.302 \times 1.31 \times 10^{-4} \times 0.027 = 195.6 \times 10^{-6}$ rem.

References

1. Slade (ed.), *Meteorology and Atomic Energy*, TID-24190, 1968.

2. Westinghouse, Nuclear Power Division, Seabrook FSAR Chapter 15, Appendix 15A and B.

Appendix Information

Atmospheric dispersion factors:

$\left(\frac{\chi}{Q}\right)$ EAB = 2.67×10^{-4} s/m^3

$\left(\frac{\chi}{Q}\right)$ LPZ = 1.31×10^{-4} s/m^3.

10. DNB CALCULATION

This section describes a technique for estimating if departure from nucleate boiling (DNB) has occurred in the core.

Calculating Departure from Nucleate Boiling

The method for calculating DNB consists of (a) calculating an approximate fluid quality[a] as a function of axial position in an isolated channel by means of an energy balance, and then (b) comparing the heat flux predicted to cause DNB, which is a function of the local fluid quality, fluid void fraction, and other parameters, with the estimated actual heat flux at the same location. If the DNB heat flux is equal to or less than the actual value, DNB has occurred.

To obtain the local fluid quality, the enthalpy at a particular distance downstream the flow channel from the entrance is determined from the expression[1-3]

$$h_z = \frac{a\, q'_{ave} L}{\dot{m}} \int_o^z f(z)\, dz\, F_R^N + h_{in}. \tag{10-1}$$

where

$h_{z,in}$	=	enthalpy of fluid at distance z along the channel and at the inlet to the channel, respectively (Btu/lbm)
a	=	unit conversion factor = 0.9481 Btu/(kW-s)
q'_{ave}	=	average linear rod power (kW/ft)
L	=	active length of fuel rod (ft)
$\dot{m}$	=	mass flow rate through fluid channel associated with rod of interest (lbm/s)
$\int_o^z f(z)\, d_z$	=	fraction of rod power generated along rod length up to distance z normalized to 1
F_R^N	=	radial power distribution factor.

The fluid quality $[\chi_{(z)}]$, is then determined from the steam tables as a function of pressure and enthalpy.

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a. Quality is defined as the mass fraction of vapor in a two-phase mixture. For homogeneous flow (phase temperatures and velocities equal), the quality as defined is equivalent to a quality defined by the ratio of vapor mass flow rate to the mass flow rate of the mixture.

The channel mass flow rate, $\dot{m}$, is also obtained from the core mass flow rate, core flow area, and channel flow area by the relation

$$\dot{m} = \frac{b \dot{M} A_c}{144 \text{ in.}^2/\text{ft}^2 A_{core}} \tag{10-2}$$

where

b = fraction of average nominal flow rate in channel of interest

$\dot{M}$ = average core mass flow rate (lbm/s)

A_c = channel flow area (in.^{2})

A_{core} = core flow area (ft^2).

The channel mass flux is obtained from the expression

$$G = 144 \text{ in.}^2/\text{ft}^2 \, \dot{m}/A_c \tag{10-3}$$

where

G = mass flux [lbm/(ft^2-s)].

The variables P, pressure, G, mass flux, and x, quality, can now be used with Table 10-1 after conversion to SI units to determine a DNB heat flux value. The value so obtained must be corrected by multiplication by the correction factors defined in Table 10-1, and then converted back to common engineering units.

The actual heat flux at a location of interest is obtained from the expression

$$q'' = \frac{c \, q'_{ave} F_R^N F_Z^N}{\pi \, d} \tag{10-4}$$

where

q'' = heat flux [Btu/(h-ft^2)]

c = unit conversion factor = 3413 Btu/(kW-h) x 12 in./ft

F_Z^N = axial power distribution factor

d = fuel rod outside diameter (in.).

Table 10-1. Heat flux correction factors[4]

Factor	Form	Comment
K_1, Subchannel factor	For $0.079 < Dhy < 0.63$, $K_1 = \left(\frac{0.315}{Dhe}\right)^{1/3}$ For $Dhy > 0.63$, $K_1 = 0.79$	Dhy = channel hydraulic diameter (in.) Dhe = heated diameter (in.) Includes diameter effect on CHF
K_2, Bundle factor	K_2 = minimum of 0.8 or $0.8 \exp(-0.5\ \chi\ 1/3)$	χ = quality Corrects for quality
K_3, Grid spacer factor	$K_3 = 1.0 + AK\left(\frac{G}{204.8}\right)^{0.2}$ $x \exp\left(-0.025 \frac{Lsp}{Dhy}\right)$	K = flow coefficient for spacer grid A = 5 (Groeneveld) G = mass flux - [lbm/(ft^2-s)] Lsp = distance from location to spacer grid immediately upstream.
K_4, Heated length factor	For $\frac{L}{Dhy} > 5$, $K_4 = \exp\left(\frac{Dhy}{L} \exp 2\alpha\right)$	L = heated length (in.) α = void fraction based on homogeneous flow
K_5, Axial flux distribution factor	For $\chi > 0.0$, $K_5 = \frac{q_{BLA}}{q_{local}}$ For $\chi < 0.0$ $K_5 = 1$	q_{BLA} = average heat flux over boiling length (Btu/(h-ft^2)] q_{local} = local heat flux [Btu/(h-ft^2)]
K_6, Vertical flow factor	For $G < -82$, $G > 20$, or $\chi \leq 0$, $K_6 = 1.0$ For $-10 < G < 2$ $CHF = CHF_{(G=0,\ X=0)}\ (1\text{-}\alpha)\ C_1$ where if $\alpha < 0.8$, $C_1 = 1$. if $\alpha \geq 0.8$, $C_1 = \frac{0.8 + 0.2\ \varrho_\ell/\varrho_g}{\alpha + (1\text{-}\alpha)\ \varrho_\ell/\varrho_g}$ For $2 < G < 20$, and $-82 < G < -10$ CHF is found by linear interpolation on G between tabled value and CHF determined at $\chi = 0$, $G = 0$.	$\varrho_{\ell,g}$ = density of liquid and vapor

Assumptions

The quality can be estimated by assuming no fluid mixing. A quasi-steady-state homogeneous flow condition is assumed.

Data for critical heat flux in tubes can be applied to rod-bundle geometry after making certain corrections.

The tabulated values for CHF and correction factors were developed based on a correlation of local conditions with CHF. The procedures presented are valid for BWR and PWR rod bundle application.

Limitations

The technique calculates the onset of DNB only. The spatial spread will have to be estimated from the assumed power and flow distribution through repeated application of the procedure. The accuracy of the procedure will be limited by the uncertainty in power distribution, local mass flux, and local quality.

Errors exist in Table 10-2. Corrected values will likely be available in future publications.

Example of PWR DNB Calculation

A large PWR is experiencing an incident such that the average mass flux is reduced to 80% of the nominal full power value, while the pressure is maintained at 2000 psia. Has DNB occurred at the core hot spot?

Data available are as follows:

- Hydraulic diameter (Dhy) = 0.4635 in.
- Fuel rod outside diameter (d) (Dhe) = 0.374 in.
- Average linear heat rate (q'_{ave}) = 5.579 kW/ft
- Radial peak to average power -1.19
- Average nominal core flow rate ($\dot{M}$) = 36,746 lbm/s
- Core flow area (A_{core}) = 54.13 ft^2
- Channel flow area (A_c) = 0.1362 in.^{2}
- Active fuel length (L) = 143.7 in.
- Coolant inlet temperature = 557°F
- Core inlet flow distribution = uniform.

Observation of the tabulated values for axial power peaking factors indicates a rather flat axial power distribution. Thus, the procedure will be applied at the 9.64 ft elevation point. Equation (10-1) is applied after finding $\dot{m}$ from Equation (10-2) as

$$\dot{m} = \frac{0.8 \text{ (fraction of normal flow) x } 36746 \text{ lbm/s (total normal core flow rate)}}{54.13 \text{ ft}^2 \text{ (core flow area)}}$$

$$\times \frac{0.1362 \text{ in.}^2}{144 \text{ in.}^2/\text{ft}^2} \text{ (channel flow area)} = 0.5137 \text{ lbm/s.}$$

Table 10-2. Estimated CHF values[4]

PRESSURE (KPA)	G (KG/M2S)	QUALITY -.15	-.10	-.05	0.00	.05	.10	.15	.20	.25	.30	.40	.50	.60	.70	.80	.90	1.00
200	0	4820	3700	2774	1450	-	-	-	-	-	-							
200	100	5000	4100	3200	2600	2300	2100	2000	1900	1800	1700	1600	1363	1072	1096	800	315	0
200	200	5200	4300	3400	2800	2500	2300	2150	2000	1800	2000	1700	1357	1261	1142	800	315	0
200	300	5400	4450	3500	3000	2700	2400	2200	2000	1814	2355	1814	1500	1100	767	600	350	0
200	500	5600	4650	3700	3150	2900	2500	2250	2000	2000	2165	1649	1500	1031	632	450	300	0
200	750	5800	4850	3900	3300	3000	2650	2350	2118	2204	1826	1767	1600	1000	600	450	300	0
200	1000	6000	5000	4000	3400	3000	2600	2350	2154	2154	1800	1700	1600	1000	600	450	300	0
200	1500	6300	5300	4300	3500	3000	2600	2350	2150	2050	1800	1700	1500	1000	600	450	300	0
200	2000	6400	5550	4500	3700	3000	2600	2350	2150	2050	1800	1700	1400	1000	600	450	300	0
200	3000	7000	5800	4700	3900	3000	2600	2350	2150	2050	1800	1700	1400	1000	600	450	300	0
200	4000	8500	7000	5400	4100	3000	2500	2350	2150	2050	1800	1700	1400	1000	600	450	300	0
200	5000	9000	7500	6000	4600	3000	2600	2350	2150	2050	1800	1700	1400	1000	600	450	300	0
200	7500	10000	8500	7000	5000	3000	2600	2350	2150	2050	1800	1700	1400	1000	600	450	300	0
300	0	4650	3667	2683	1700	-	-	-	-	-								
300	100	5200	4500	3500	3600	3200	3000	2900	2800	2600	2400	2000	1365	1318	1287	1066	700	0
300	200	5600	4700	3600	3500	3200	2900	2800	2700	2600	2360	1900	1786	1527	1148	849	600	0
300	300	5800	4800	3700	3600	3200	2800	2700	2600	2500	2360	1755	1749	1365	900	624	500	0
300	500	5900	4800	3800	3700	3200	2700	2600	2500	2200	2200	2157	1676	1400	800	400	350	0
300	750	6000	5000	3900	3800	3200	2600	2500	2400	2149	2117	1836	1535	1396	700	400	350	0
300	1000	6200	5200	4000	3900	3200	2600	2500	2300	2178	2100	1800	1500	1200	700	400	350	0
300	1500	6400	5400	4700	4000	3200	2600	2500	2294	2200	2100	1800	1500	1200	700	400	350	0
300	2000	6400	5000	4400	4100	3200	2600	2500	2294	2200	2100	1800	1500	1200	700	400	350	0
300	3000	6800	5800	4600	4200	3200	2600	2500	2294	2200	2100	1800	1500	1200	700	400	350	0
300	4000	7000	6000	4800	4200	3200	2600	2500	2294	2200	2100	1800	1500	1200	700	400	350	0
300	5000	7200	6200	5000	4200	3200	2600	2500	2294	2200	2100	1800	1500	1200	700	400	350	0
300	7500	7600	6200	5000	4200	3200	2600	2500	2300	2200	2100	1800	1500	1200	700	400	350	0
450	0	4540	3687	2833	1980	-	-	-	-									
450	100	5200	4100	3400	2900	2800	2700	2600	2550	2500	2400	2400	2100	1800	1400	1047	538	0
450	200	5600	4150	3400	3000	2800	2600	2550	2500	2450	2600	2400	2008	1751	1224	885	525	0
450	300	5800	4200	3600	3000	2800	2600	2500	2450	2400	2300	2367	1841	1644	922	496	487	0
450	500	5900	4250	3400	3100	2800	2600	2400	2350	2200	2100	1912	1614	1028	594	500	392	0
450	750	6000	4300	3400	3100	2800	2600	2300	2150	2000	1900	1600	1012	876	500	400	350	0
450	1000	6200	4400	3400	3200	2800	2600	2100	2000	1700	1600	1300	1000	700	500	400	350	0
450	1500	6400	4600	3300	3200	2800	2600	2000	1800	1500	1300	1183	970	619	500	400	350	0
450	2000	6600	4800	3300	3300	2800	2600	2100	1800	1500	1300	1200	1000	700	500	400	350	0
450	3000	6800	5000	3700	3350	2800	2600	2200	1800	1500	1300	1200	1000	700	500	400	350	0
450	4000	7000	5200	3400	3400	2800	2600	2200	1800	1500	1300	1200	1000	700	500	400	350	0
450	5000	7200	5400	3600	3500	2800	2600	2700	1900	1500	1300	1200	1000	700	500	400	350	0
450	7500	7400	5700	4000	3600	2800	2600	2200	2000	1600	1300	1200	1000	700	500	400	350	0
700	0	4482	3761	3041	2320	-	-	-	-									
700	100	5200	4200	3500	3000	2800	2600	2300	2000	1700	1400	1200	1000	800	596	691	613	0
700	200	5600	4250	3600	3200	3000	2900	2700	2600	2500	2300	2200	2166	2163	1595	1063	595	0
700	300	5800	4300	3700	3400	3000	2800	2600	2500	2300	2100	2000	1898	1600	1280	743	495	0
700	500	5900	4600	3800	3500	3000	2700	2500	2400	2100	2000	1900	1500	1380	653	600	346	0
700	750	6000	4600	3900	3600	3000	2500	2300	2200	1900	1700	1685	1386	1000	600	500	336	0
700	1000	6200	4800	4000	3600	3000	2500	2200	2000	1700	1500	1578	958	712	564	400	350	0
700	1500	6400	5000	4200	3600	3000	2500	2200	1800	1500	1340	1342	900	700	523	450	350	0
700	2000	6600	5200	4400	3600	3000	2500	2200	1800	1500	1300	1300	900	700	550	500	400	0
700	3000	6800	5400	4600	3600	3000	2500	2200	1800	1500	1300	1200	900	700	550	500	450	0
700	4000	7000	5600	4800	3600	3000	2500	2200	1800	1500	1300	1200	900	700	550	500	450	0
700	5000	7200	5800	4900	3600	3000	2500	2200	1900	1500	1300	1200	900	700	550	500	500	0
700	7500	7400	6000	5000	3600	3000	2500	2200	2000	1600	1300	1200	900	700	550	500	500	0
1000	0	4480	3866	3240	2620	-	-	-	-									
1000	100	6000	5500	5000	4600	4400	4200	4000	3900	3600	3300	3000	2500	1400	1113	1114	550	0
1000	200	8000	7500	7000	6000	5800	5600	5400	5100	4700	4400	4300	4200	1778	1689	1139	702	0
1000	300	8000	7500	7000	6500	6300	5600	5400	5100	4700	4400	4400	3900	2202	1535	944	590	0
1000	500	8000	7500	7000	6500	6200	5600	5400	5100	4700	4400	4184	3500	2040	1039	667	459	0
1000	750	8000	7500	7000	6500	6100	5600	5400	5000	4800	4336	4061	2421	1220	731	617	500	0
1000	1000	8000	7500	7000	6500	6000	5600	5300	4957	4412	4390	3177	1689	1100	602	661	565	0
1000	1500	8000	7500	7000	6500	5900	5300	5000	4700	4084	3106	2378	1561	900	650	519	500	0
1000	2000	8000	7500	7000	6500	5700	5200	4800	4400	3405	3129	2200	1400	800	600	500	450	0
1000	3000	8000	7500	7000	6500	5500	5100	4700	4200	3530	3000	2000	1300	750	600	450	600	0
1000	4000	8000	7500	7000	6500	5300	5000	4500	4000	3400	2800	1850	1150	700	550	400	350	0
1000	5000	8000	7500	7000	6500	5400	4900	4600	3800	3300	2600	1650	1000	650	550	400	350	0
1000	7500	8000	7500	7000	6500	5500	4800	4300	3600	3200	2300	1500	900	600	500	400	350	0
1500	0	4525	4007	3488	2970	-	-	-										
1500	100	6200	5100	5400	5000	4600	4500	4300	4200	4000	3700	3300	3000	1450	1300	1131	548	0
1500	200	7100	6900	6600	6400	6200	6000	5700	5400	5000	4100	3600	3100	2131	2054	1314	779	0
1500	300	7400	7100	6700	6500	6300	6000	5700	5300	5200	4400	3800	3294	3696	2122	1334	827	0
1500	500	7700	7200	6700	6400	6100	5800	5500	5300	5100	5200	5300	4300	3396	1881	997	623	0
1500	750	8000	7200	6700	6300	5900	5600	5400	5200	5000	5100	5412	3231	1916	1130	594	366	0
1500	1000	8000	7200	6700	6250	5700	5400	5300	5156	4906	4959	3917	2367	1379	900	518	565	0
1500	1500	8000	7200	6700	6250	5500	5300	5150	5074	4603	4406	2863	1675	1080	800	600	500	0
1500	2000	8000	7200	6700	6200	5500	5200	4900	4758	3707	3555	2159	2400	1000	800	600	400	0
1500	3000	8000	7200	6700	6130	5600	5100	4800	4000	4119	3337	2100	2100	1000	800	600	400	0
1500	4000	8000	7200	6700	6100	5400	5000	4700	3800	3800	3100	1900	1800	1000	800	600	400	0
1500	5000	8500	7500	6900	6300	5600	4800	4400	3400	3500	2600	1700	1500	1000	800	600	400	0
1500	7500	9000	8000	7100	6300	5500	4700	4100	3600	3300	2350	1600	1200	1000	800	600	450	0

Table 10-2. (continued)

1.00	PRESSURE (ATA)	G (KG/M2S)	-.15	-.10	-.05	0.00	.05	.10	.15	.20	.25	.30	.40	.50	.60	.70	.80	.90	1.00
									QUALITY										
	2000	0	4581	4130	3680	3230	-	-	-	-									
0	2000	100	6300	5850	5400	5200	4900	4800	4600	4500	4300	4000	3600	3500	1500	1187	1191	556	0
0	2000	200	7000	6900	6800	6700	6600	6300	6000	5600	5300	4800	4400	4000	2151	1687	1311	824	0
0	2000	300	7600	7350	7200	6900	6600	6600	5900	5700	5500	5200	4950	4700	4363	2499	1500	869	0
0	2000	500	8200	7800	7500	7200	6600	6600	5800	5800	5000	4900	4750	4500	3637	2222	1274	831	0
0	2000	750	8400	8000	7600	7200	6600	6400	5700	5600	4737	4721	4737	3650	2184	1693	954	586	0
0	2000	1000	8700	8400	7800	7250	6700	6200	5600	5200	4800	4700	4079	2813	1221	1197	607	479	0
0	2000	1500	8800	8400	7800	7200	6600	6000	5500	5020	4650	4613	3194	2600	1600	1160	849	450	0
0	2000	2000	9200	8500	7700	7100	6500	5900	5078	4866	4606	4248	2172	1800	1264	1129	600	450	0
0	2000	3000	9400	8600	7800	7000	6200	5600	5506	4918	4779	4423	2000	1800	1700	1000	600	450	0
0	2000	4000	9600	8700	7800	6900	5900	5000	509[illegible]	4042	3961	3540	1780	1600	1200	900	600	450	0
0	2000	5000	10000	9500	7000	5800	4600	4600	4400	3581	3823	3200	1500	1400	1200	900	500	500	0
0	2000	7500	10300	8500	7200	6100	4235	4000	3800	3600	3400	2400	1700	1500	1300	1100	800	600	0
	3000	0	4650	4290	3930	3570	-	-	-										
0	3000	100	6500	6000	5500	5300	5200	5100	5000	4800	4600	4300	3900	3500	3000	1100	831	588	0
0	3000	200	8000	7500	7000	6600	7000	6700	6300	6000	5700	5400	4800	4000	3600	1461	1425	637	0
0	3000	300	9000	8500	8000	7600	7900	7700	7300	6970	6600	6260	5500	4600	4300	2614	1664	884	0
0	3000	500	9600	9300	8800	8400	8000	7700	7300	6970	6600	6260	5500	4084	3719	2522	1769	1035	0
0	3000	750	9800	9300	8700	8410	8050	7690	7340	6970	6600	6260	5519	3622	2825	1996	1281	663	0
0	3000	1000	9800	9050	8450	8050	7600	7170	6750	6242	5184	5387	4143	3131	2251	1823	1074	1006	0
0	3000	1500	9800	9000	8400	7730	7100	6590	5302	5482	4771	4353	3432	2474	1712	1638	1016	1265	0
0	3000	2000	9900	9000	7800	6735	6735	6045	5082	4814	4374	4076	2778	2029	1700	1350	1000	600	0
0	3000	3000	10000	9000	7600	6134	6134	5026	4912	4722	4195	3402	1895	1900	1758	1350	1000	600	0
0	3000	4000	10500	8900	7600	5800	4766	4757	4700	4436	3754	2645	2168	1900	1750	1350	1000	600	0
0	3000	5000	10800	8900	7250	5900	4900	4400	4692	3795	3281	2500	2150	1900	1758	1350	1000	600	0
0	3000	7500	11200	8800	7200	6000	5365	4242	3571	3719	3569	2200	2150	2000	1800	1500	1200	700	0
	4500	0	4676	4400	4125	3850	-	-	-										
0	4500	100	6000	5600	5300	5000	4800	4500	4400	4100	3600	2700	2000	1500	1000	800	746	666	0
0	4500	200	7900	7300	6700	6000	5800	4900	4700	4600	4200	3700	3200	2800	2500	1698	1419	954	0
0	4500	300	8100	7600	7100	7000	6300	5400	5300	5200	4700	4200	3700	3365	3880	2240	1620	1049	0
0	4500	500	8200	7800	7500	6700	5900	5100	4939	4785	4298	3609	3214	3067	2319	2358	1691	1103	0
0	4500	750	8300	8000	7500	6600	5700	4858	4784	4175	4112	4000	3923	2704	2602	2301	1631	965	0
0	4500	1000	8700	8000	7400	6500	5091	4571	4151	4680	4360	4336	3731	3032	2554	2284	1400	875	0
0	4500	1500	8900	8000	7300	6300	5500	5539	5169	4511	4083	3974	3178	2663	1976	1263	1200	800	0
0	4500	2000	8900	8000	7100	6165	6165	5215	4563	4204	3906	3898	2812	2208	1454	1300	1000	800	0
0	4500	3000	9000	7900	6700	5479	5479	4757	4744	4161	3743	3318	2218	1547	1514	1300	1000	800	0
0	4500	4000	9100	7500	6400	5023	4600	4527	3976	3796	3366	2472	1700	1600	1500	1300	1000	800	0
0	4500	5000	9200	7500	6500	5071	5071	4995	4755	3766	2580	2216	1700	1600	1500	1300	1100	900	0
0	4500	7500	9400	7500	6200	5282	4896	5000	4939	4600	4027	3500	2600	2200	1800	1500	1300	1000	0
	7000	0	4538	4348	4160	3970	-	-	-	-									
0	7000	100	5800	5400	5000	4600	4400	4100	3900	3700	3200	2400	1800	1600	1100	900	668	660	0
0	7000	200	6500	6000	5500	5091	4800	4600	4500	4400	4000	3500	3000	2300	1600	1384	1687	1165	0
0	7000	300	7000	6300	5600	5122	5000	4800	4700	5100	4100	4000	3500	2630	2568	2007	1521	1125	0
0	7000	500	7250	6500	5800	5200	5000	4900	4934	4785	4293	3589	3069	2819	2235	2048	1587	1083	0
0	7000	750	7500	7000	6200	5100	5000	4858	4819	4322	4090	3555	2839	2618	2292	1937	1497	1046	0
0	7000	1000	7700	7000	6200	4916	4916	4416	4170	3747	3400	3215	2853	2557	2743	1460	1009	1061	0
0	7000	1500	7900	7400	6200	4676	4676	4320	3967	3469	3206	2950	2456	1538	1482	842	600	400	0
0	7000	2000	8000	7200	6300	4823	4453	3942	3257	3182	2878	2697	1874	1420	763	268	177	110	0
0	7000	3000	8200	7200	6200	4203	4203	3644	3330	3050	2853	2080	1600	800	314	235	177	110	0
0	7000	4000	8400	7700	6200	4889	4056	3332	2959	2453	2114	1776	1421	600	255	232	180	110	0
0	7000	5000	8600	7700	6200	4686	4686	3817	3179	2953	2872	1730	1440	600	400	300	250	150	0
0	7000	7500	9100	7200	6200	6146	6146	5417	4031	4607	4492	1953	1700	700	500	400	300	200	0
	10000	0	4144	4019	3894	3770	-	-	-										
0	10000	100	4600	4300	4000	4100	3900	3400	3000	2600	2200	1800	1400	1100	800	499	568	660	0
0	10000	200	5000	4600	4700	4400	4000	3700	3400	3700	2800	2400	1800	1500	1200	998	1298	1370	0
0	10000	300	5100	4750	4600	4500	4200	4000	3800	3500	3300	3100	2800	2594	2072	1740	1285	868	0
0	10000	500	5200	4800	4600	4400	4200	4000	3800	3500	3200	2900	2611	2197	1726	1215	772	500	0
0	10000	750	5300	5000	4600	4340	4100	3800	3500	3182	2600	2600	2100	1655	1170	760	600	531	0
0	10000	1000	5400	5000	4600	4780	3843	3621	3497	3100	2985	2329	1853	1211	689	600	600	463	0
0	10000	1500	5700	5500	5478	3830	3838	3574	3299	2848	2561	1967	1106	900	750	500	500	462	0
0	10000	2000	5900	5023	4783	4336	3631	3191	2495	2127	2083	1309	1000	800	614	616	178	160	0
0	10000	3000	6400	5472	5174	3641	3641	3006	2518	2239	2093	1958	1012	700	415	497	177	160	0
0	10000	4000	7100	6120	5200	3388	3388	3797	3090	2000	2281	1700	1158	700	500	400	177	160	0
0	10000	5000	8600	7350	5400	3492	3492	4041	1629	1695	2518	1477	1300	800	500	400	177	160	0
0	10000	7500	9400	8000	5400	4538	4538	4000	1500	3000	2500	1800	1600	1000	500	400	170	160	0
	15000	0	3103	3046	2987	2930	-	-											
	15000	100	3200	3100	3000	2900	2600	2300	2200	2000	1900	1800	1700	900	800	603	543	560	0
0	15000	200	3200	3200	3000	2900	2600	2300	2200	2000	1750	1700	1700	900	811	422	678	504	0
0	15000	300	3300	3200	3000	2800	2600	2300	2200	2000	1601	1612	1601	935	840	620	513	493	0
0	15000	500	3300	3200	3000	2800	2615	2245	2278	2025	1828	1533	1163	812	463	346	513	404	0
0	15000	750	3400	3400	3100	2700	2543	2510	2116	1906	1601	1445	825	509	397	384	513	404	0
0	15000	1000	3451	3677	3508	2800	2527	2050	1903	1757	1548	1337	661	500	500	500	513	404	0
0	15000	1500	3432	2533	2864	2500	2207	2132	1931	1896	1640	1742	700	500	500	500	513	404	0
0	15000	2000	3819	3595	3107	2579	2336	2260	2151	2000	1632	1111	887	468	500	500	513	404	0
0	15000	3000	4400	3847	3513	2800	2534	2275	2100	2097	1500	1300	1200	500	500	500	513	404	0
0	15000	4000	5100	4200	3600	2808	2701	2653	2400	2279	1540	1500	1164	700	500	500	500	400	0
0	15000	5000	5600	4900	3700	3500	3081	2972	2500	2572	1600	1500	1300	1000	750	700	700	600	0
0	15000	7500	6100	5300	4100	4000	4030	3500	3000	2500	1800	1500	1300	1100	950	900	900	700	0
0																			

From Equation (10-1),

$$h_z = \frac{0.9481\ \text{Btu/(kW-s)}}{0.5137\ \text{lbm/s}} \times 5.579\ \text{kW/ft (average linear power)} \times \frac{143.7\ \text{in.}}{12\ \text{in./ft}}\ \text{(active length)}$$

$$\times\ 0.848\ \text{(integrated axial rod power fraction)} \times 1.41\ \text{(radial power factor)} + 582.5\ \text{Btu/lbm (enthalpy in.)};$$

thus,

$h_z = 730.2$ Btu/lbm.

The quality (χ) = 0.125, may be found from the relationship $h_z = h_f$ (saturated liquid enthalpy) + χh_{fg} (enthalpy increment due to evaporation) where $h_f = 672.1$ Btu/lbm and $h_{fg} = 466.2$ Btu/lbm at 2000 psia. The void fraction (α) = 0.512, may be found from the relationship

$\frac{1-\chi}{\chi} = \frac{\upsilon_g}{\upsilon_f}\frac{1-\alpha}{\alpha}$, where $\upsilon_g = 0.18831\ \text{ft}^3/\text{lbm}$ and $\upsilon_f = 0.02565\ \text{ft}^3/\text{lbm}$ at 2000 psia.

The mass flux is found from Equation (10-3)

$$G = 144\ \text{in.}^2/\text{ft}^2\ \frac{0.5137\ \text{lbm/s}}{0.1362\ \text{in.}^2} = 543.1\ \text{lbm/(ft}^2\text{-s)}$$

The parametric values for entering Table 10-2 are determined by converting the units as

$$P = 2000\ \text{psia} \times \frac{\text{kPa}}{0.145038\ \text{psia}} = 13789\ \text{kPa},$$

$$G = 543.1\ \text{lbm/(ft}^2\text{-s)} \times \frac{\text{kg/(m}^2\text{-s)}}{0.20482\ \text{lbm/(ft}^2\text{-s)}} = 2651\ \text{kg/(m}^2\text{-s)}$$

Bracketing entries of the table are extracted and linear interpolation is performed first on the quality, then on mass flux, and finally on the pressure.

The bracketing entries are shown in Table 10-3 along with the interpolation results. The bracketing values for pressure are 10,000 and 15,000 kPa; for mass flux are 2000 and 3000 kg/(m^2-s); and for quality are 0.10 and 0.15. The first step finds the heat flux values at $\chi = 0.125$ for the listed pressure and mass flux. The values are shown in single parenthesis. Step 2 finds the heat flux values for the desired mass flux with pressure and quality held constant. The values are shown in double parentheses. Step 3 finds the final heat flux value (underlined) at the desired pressure with the mass flux and quality held constant.

The correction factors of Table 10-1 are evaluated next:

$$K_1 = \left(\frac{0.315\ \text{in.}}{0.374\ \text{in.}}\right)^{1/3} = 0.944\ ,$$

$$K_2 = 0.8\ \exp\left(-0.5 \times 0.125^{1/3}\right) = 0.623\ ,$$

$$K_3 = 1.0 + 5.0 \times 0.066 \left[\frac{543.0\ \text{lbm/(s-ft}^2)}{204.8\ \text{lbm/(s-ft}^2)}\right]^{0.2} \exp\left(-0.025 \times \frac{7.25\ \text{in.}}{0.4635\ \text{in.}}\right) = 1.271\ .$$

Table 10-3. Extracted DNB heat flux values

	P(kPa)	G[kg/(m^2-s)]	χ 0.10	χ 0.125	χ 0.15	
table→	10000	2000	3191	(2857)	2495	
		2651		[(2802)]		←Step 2
table→	10000	3000	3006	(2772)	2518	
	13789	2651		2344		←Step 3 Answer
table→	15000	2000	2260	(2008)	2151	
		2651		[(2197)]		←Step 2
table→	15000	3000	2275	(2191) ↑ Step 1	2100	

The length downstream of a spacer grid (Lsp) is 9.64 ft x 12 in./ft - 108.43 in = 7.25 in.

$$K_4 = \exp\left[\frac{0.4635 \text{ in.}}{115.7 \text{ in.}} \times \exp(2.0 \times 0.512)\right] = 1.011 .$$

$K_5 \cong 1.0.$

Since the axial power profile is very flat as shown in Table 10-4, the heat flux over the boiling length will be assumed equal to the heat flux at the location of interest.

$K_6 = 1.0.$

Taking the product of the correction factors and the critical heat flux value from the table,

$$\text{CHF} = 0.944 \times 0.623 \times 1.271 \times 1.011 \times 1.0 \times 1.0 \times 2344 \text{ kW/m}^2$$

$$\times\ 317 \frac{\frac{\text{Btu}}{\text{h-ft}^2}}{\text{kW/m}^2} = 561{,}000 \frac{\text{Btu}}{\text{h-ft}^2} .$$

The actual heat flux at the location is

5.579 kW/ft (core average) x 1.41 (radial peaking factor)

x 1.096 (axial peaking factor at 9.64 ft) x 3413 Btu/(kW-h) x 12 in./ft^2

÷ π 0.374 in. (rod diameter) = 300,500 Btu/(h - ft^2)

Table 10-4. Best estimates reactor core axial power distribution

	Axial Power Distribution	
Elevation (ft)	Normalized Integrated Power to Elevation $\left(\int_0^Z f(z)\,dz\right)$	Peak to Average Value $\left(F_N^Z\right)$
0.0	0.0	0.0
1.246	0.0604	0.966
2.46	0.164	1.131
3.64	0.275	1.19
4.85	0.393	1.19
6.04	0.508	1.19
7.25	0.625	1.178
8.43	0.737	1.143
9.64	0.848	1.096
10.82	0.958	0.907
12.00	1.00	0.0

So, the DNB ratio $= \dfrac{561{,}000}{300{,}500} = 1.87$ and DNB has not occurred.

References

1. Collier, J. G., *Convective Boiling and Condensation*, New York: McGraw Hill, 1972.

2. Lamarch, J. R., *Introduction to Nuclear Engineering*, Reading: Addison-Wesley, 1975.

3. Tong, L. S. and J. Weisman, *Thermal Analysis of Pressurized Water Reactors*, American Nuclear Society, 1970.

4. Groeneveld, D. C. and C. W. Snoek, "A Review of Current Heat Transfer Correlations Used in Reactor Safety Assessment," *Thermal Hydraulics of Nuclear Reactors, Volume I, Proceedings of Second International Topical Meeting on Nuclear Reactor Thermal Hydraulics, Santa Barbara, California, January 11-14, 1983.*

Appendix Information

1. Specific values for equations:

q'_{ave}	=	5.579 kW/ft, fuel rod average linear heat rate
L	=	143.7 in., active fuel length
$h_{in.}$	=	582.5 Btu/lbm, core coolant inlet enthalpy at 577°F inlet temperature
M	=	36,746 lbm/s, nominal average core mass flow rate
A_c	=	0.1362 in.^{2}, nominal channel flow area
A_{core}	=	54.13 ft^2, core flow area
d,Dhe	=	0.374 in., fuel rod outside diameter
Dhy	=	0.4635 in., channel hydraulic diameter
K	=	0.066, spacer grid flow coefficient.

2. Power conditions:

Axial peak to average power	1.19
Radial peak to average power	1.41

3. Flow conditions:

Coolant inlet temperature	577°F
Coolant nominal pressure	2250 psia
Core inlet flow distribution	uniform

4. Axial grid spacing:

Distance above bottom of active length (in.)

4.37
26.23
46.83
67.33
87.93
108.43
129.03
149.63

11. VOID FORMATION IN UPPER HEAD

This section describes a technique for estimating void formation and volume in the upper head of a PWR.

Determining Potential Void Volume

The method for determining the potential for void formation in the upper head is based on a comparison of the system pressure with the saturation pressure of the upper head fluid based on the average temperature of the fluid. If the system pressure decreases below the saturation pressure of the upper head fluid volume, expansion or voiding occurs. Considering only the initial saturation pressure and a representative end state pressure, the potential void volume for an isenthalpic process can be determined from the following expression[a]

$$V_o = V_{uh}\left(\nu_2/\nu_1\right)\alpha_2 = V_{uh}\,\chi_2\,\nu_{g2}/\nu_1 \tag{11-1}$$

where

V_o = potential void (ft^3)

V_{uh} = upper head volume (ft^3)

$\nu_{1,2,g2}$ = specific volume at initial and final statepoints and vapor specific volume at final statepoint (ft^3/lbm)

α_2 = void fraction at final statepoint

χ_2 = quality of final statepoint.

The potential void volume (V_o) can be compared to the upper head volume to estimate if the upper head was completely voided.

Also, for the upper head to become voided, there must be space elsewhere available for the displaced liquid to occupy. The space could be in the pressurizer where it could be measured or in the containment if a transport path exists. The volume of displaced liquid (V_{dl}) is given by

$$V_{dl} = V_{uh}\left[\left(\nu_2/\nu_1\right)-1\right] \tag{11-2}$$

Defining the process to be isenthalpic permits ready determination of the final statepoint.

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a. An equivalent void fraction for the mass of liquid initially contained in the upper head after depressurization may be defined as

$$\alpha_2 = \frac{\text{void volume}}{\text{total volume of fluid}} = \frac{V_o \text{ (potential void volume)}}{V_{uh}/\nu_1 \text{(mass initially in upper head)} \times \nu_2 \text{(specific volume)}}$$

For a homogeneous fluid $\chi\nu_g = \alpha\nu$, so the expression can be rewritten in terms of χ.

Assumptions

The method for determining potential void volume is based on an isenthalpic expansion process. Mixing of the fluid discharged from the upper head with fluid contained in the upper plenum is not considered.

Separation of vapor and liquid occurs with the vapor phase remaining in the upper head.

The expansion process is not rate limited or otherwise restricted because of flow resistance due to a baffle plate separating the upper head and upper plenum.

Heat transfer between the fluid and the structural components is not considered.

The upper head is supplied with coolant from the downcomer inlet annulus.

Limitations

The method presented is only an approximation.

Example of Technique Application

A large PWR experiences an incident during which considerable coolant is lost from the pressurizer and the system pressure declines to 1200 psia. At this point, a PORV is opened and the system pressure suddenly declines further to 1000 psia. After a moment, the valve is closed. Does the upper head void? The PWR had been operating with a cold leg temperature of 557°F, and the upper head coolant temperature was essentially equal to the cold leg temperature. The volume of the upper head was 878.1 ft^3.

The initial and final statepoint conditions are found as follows:

- P_1 = 1200 psia
- T_1 = 557°F
- h_1 = 558.3 Btu/lbm
- ν_1 = 0.02194 ft^3/lbm
- P_2 = 1000 psia
- h_2 = 558.3 Btu/lbm
- ν_2 = 0.03184 ft^3/lbm
- ν_{2g} = 0.4460 ft^3/lbm
- ν_{2f} = 0.02159 ft^3/lbm
- α_2 = 0.338

The volume of vapor found from Equation (11-1) is

$$V_o = 878.1 \text{ ft}^3 \times \frac{0.03184 \text{ ft}^3/\text{lbm}}{0.02194 \text{ ft}^3/\text{lbm}} \times 0.338 = 430.7 \text{ ft}^3$$

The extra volume of liquid that the system must contain is from Equation (11-2)

$$V_{dl} = 878.1 \text{ ft}^3 \times \left[\frac{0.03184 \text{ ft}^3/\text{lbm}}{0.02194 \text{ ft}^3/\text{lbm}} - 1\right] = 396.2 \text{ ft}^3$$

The water level in the pressurizer rises 16.0 ft. The 43.4 ft span between level taps corresponds to 1066 ft^3. The extra volume of liquid added to the pressurizer is

$$Vol_l = \frac{16.0 \text{ ft (level change)}}{43.4 \text{ ft (span)}} \times 1066 \text{ ft}^3 \text{(span volume)} = 393 \text{ ft}^3, \text{ liquid.}$$

Therefore, the upper head was about 45% voided.

Appendix Information

1. Specific values for equations:

   $V_{uh} = 878.1\ ft^3$.

2. Temperature:

Average upper head coolant temperature	557°F

3. Pressurizer:

Pressurizer level measurement span	43.4 ft
Pressurizer volume corresponding to span	1066 ft^3

12. TORUS HEATUP

This section describes a simple technique for estimating the elapsed time following an incident that the torus in a BWR would function satisfactorily as a heat sink for decay energy generated by the core.

Approximate Heat Balance Method

The method consists of an approximate heat balance considering the main components that act as heat sources and sinks in the process of transferring decay heat energy to the torus. An approximate m~~s balance is described with the condition that the operator, or automatic action, cause liquid from the condensate tank to enter the vessel.

The elapsed time that the torus is available to absorb energy can be estimated from the equation

$$1/a \int_0^{t_1} \dot{q}_{dh}\, dt = M_{coolr} (h_2 - h_1) + M_v C_p (T_{V2} - T_{V1})$$

$$+ M_{dw} C_p (T_{dw2} - T_{dw1}) + M_t C_p (T_{t2} - T_{t1})$$

$$+ (V_{sp}/\nu_{sp}) (h_{sp2} - h_{sp1}) + M_{cond} (h_{sp2} - h_{cond1})$$

$$+ RHR + DWC + Q_{loss} \tag{12-1}$$

where

a	=	unit conversion constant = 1.055×10^{-3} (MW-s)/Btu
$\dot{q}_{dh}$	=	decay power (MW)
M_{coolr}	=	mass of coolant in reactor vessel system (lbm)
$h_{1,2}$	=	average initial reactor system coolant enthalpy and final enthalpy at 100 psia.
$M_{v,dw,t}$	=	mass of steel in reactor vessel system, drywell, and torus (lbm)
C_p	=	specific heat of steel = 0.11 Btu/(lbm-°F)
$T_{V2,V1,dw2,dw1,t2,t1}$	=	temperature of steel in vessel system, drywell, and torus component at statepoint 2 (end of process or value when suppression pool can no longer function as heat sink) and statepoint 1 (initial condition) (°F)
V_{sp}	=	initial volume of liquid in suppression pool (ft^3)
ν_{sp}	=	initial specific volume of liquid in suppression pool (ft^3/lbm)

M_{cond}	=	mass from condensate tank assumed equal to mass addition to suppression pool (lbm)
$h_{sp1,2}$	=	enthalpy of suppression liquid pool at statepoint 1 and 2 (Btu/lbm)
h_{cond1}	=	enthalpy of liquid in condensate tank (Btu/lbm)
RHR	=	heat removal capacity of residual heat removal system (Btu/h)
DWC	=	heat removal capacity of drywell coolers (Btu/h)
Q_{loss}	=	heat loss to environment not specifically identified taken as zero (Btu/h).

The mass transferred from the vessel to the suppression pool is obtained from

$$\Delta M_{sp} = \frac{M_t C_p (T_{t2} - T_{t1}) + (V_{sp}/v_{sp})(h_{sp2} - h_{sp1})}{(h_g - h_{sp2})} \qquad (12\text{-}2)$$

where

ΔM_{sp} = M_{cond} (which assumes operator action maintains near constant coolant mass in reactor system)

h_g = enthalpy of saturated vapor leaving reactor through break or safety/relief valve (nearly independent of pressure) = 1190 Btu/lbm.

This method approximates the behavior for a station blackout-type incident. For a stuck open safety/relief valve, the method needs modification.

Assumptions

Heat is transferred to the drywell by conduction and radiation.

The net energy changes of the fluid contained in the vessel may be approximated by changes in the average conditions.

The mass of coolant and liquid level in the vessel are maintained constant. Liquid from the condensate tank transports energy from the vessel to the suppression pool and torus. Condensate is available during the process.

Exhaust steam from high pressure coolant injection (HPCI) and reactor core injection cooling (RCIC) turbines passed to the suppression pool is small compared to steam flow directly from the reactor and is neglected.

The suppression pool will function as a heat sink up to an average temperature of 190°F. The initial temperature of the suppression pool and torus is about 90°F.[1,2]

The operator will normally control the incident so that depressurization of the reactor coolant occurs to about 100 psia; thus, the drywell temperature will be limited to about 300°F.

Heat loss to the environment (surrounding building and concrete in drywell) is small and is neglected.

Limitations

The procedure was compared to results for a station blackout analysis to 5 h elapsed time performed by ORNL (NUREG/CR-2182[3]). The suppression pool temperature at that time was ~180°F. Using the end point temperature as a boundary condition, the method was exercised to determine the corresponding elapsed time. The method yielded a time of 4.7 h. It would be expected that the method would underestimate the time as heat losses to the environment (containment) are neglected.

A 3% error in the energy balance will change the elapsed time by 10 min.

Example of BWR/4 Station Blackout

A BWR/4 experiences a station blackout with loss of the residual heat removal (RHR) and drywell cooling (DWC) systems since they are electrically operated (see Figure 12-1). The RCIC system is operated to maintain vessel liquid level. The safety/relief valve is also operated to maintain the pressure initially and then to depressurize to 100 psia. How long will the suppression pool be available as an effective heat sink?[a] Nominal full power conditions existed prior to the incident as tabulated in the "Appendix Information" section.

The mass of coolant entering the suppression pool is initially computed from Equation (12-2).

$$\Delta M_{sp} = 629{,}777 \text{ lbm (torus steel)} \times 0.11 \text{ Btu/lbm (specific heat)}$$

$$\times [190°\text{F (final temperature)} - 90°\text{F (initial temperature)}]$$

$$+ \frac{123{,}000 \text{ ft}^3 \text{ (minimum pool volume)}}{0.016099 \text{ ft}^3\text{/lbm (initial pool specific volume)}} \quad [158.04 \text{ Btu/lbm (enthalpy at 190°F)}$$

$$- 58.018 \text{ Btu/lbm (enthalpy at 90°F)}]$$

$$\div [1190 \text{ Btu/lbm (enthalpy of vapor leaving reactor)}$$

$$- 158.04 \text{ Btu/lbm (final enthalpy at 190°F)}] = 7.4724 \times 10^5 \text{ lbm.}$$

In computing the mass (ΔM_{sp}) one finds that the heat capacity of the torus steel is about 1% of the heat capacity of the suppression pool liquid. The value obtained from Equation (12-2) is then input to Equation (12-1) to find the integrated decay energy and elapsed time with RHR, DWC, and $Q_{loss} = 0$, based on the assumptions and problem statements.

$$\frac{\int_0^{t_1} \dot{q}_{dh} \, dt}{1.055 \times 10^{-3} \text{ (MW - s)/Btu}} = 6.0921 \times 10^5 \text{ lbm (initial reactor coolant)}$$

$$\times [298.54 \text{ Btu/lbm (enthalpy of coolant at statepoint 2)}$$

a. Precalculated values are tabulated in "Appendix Information."

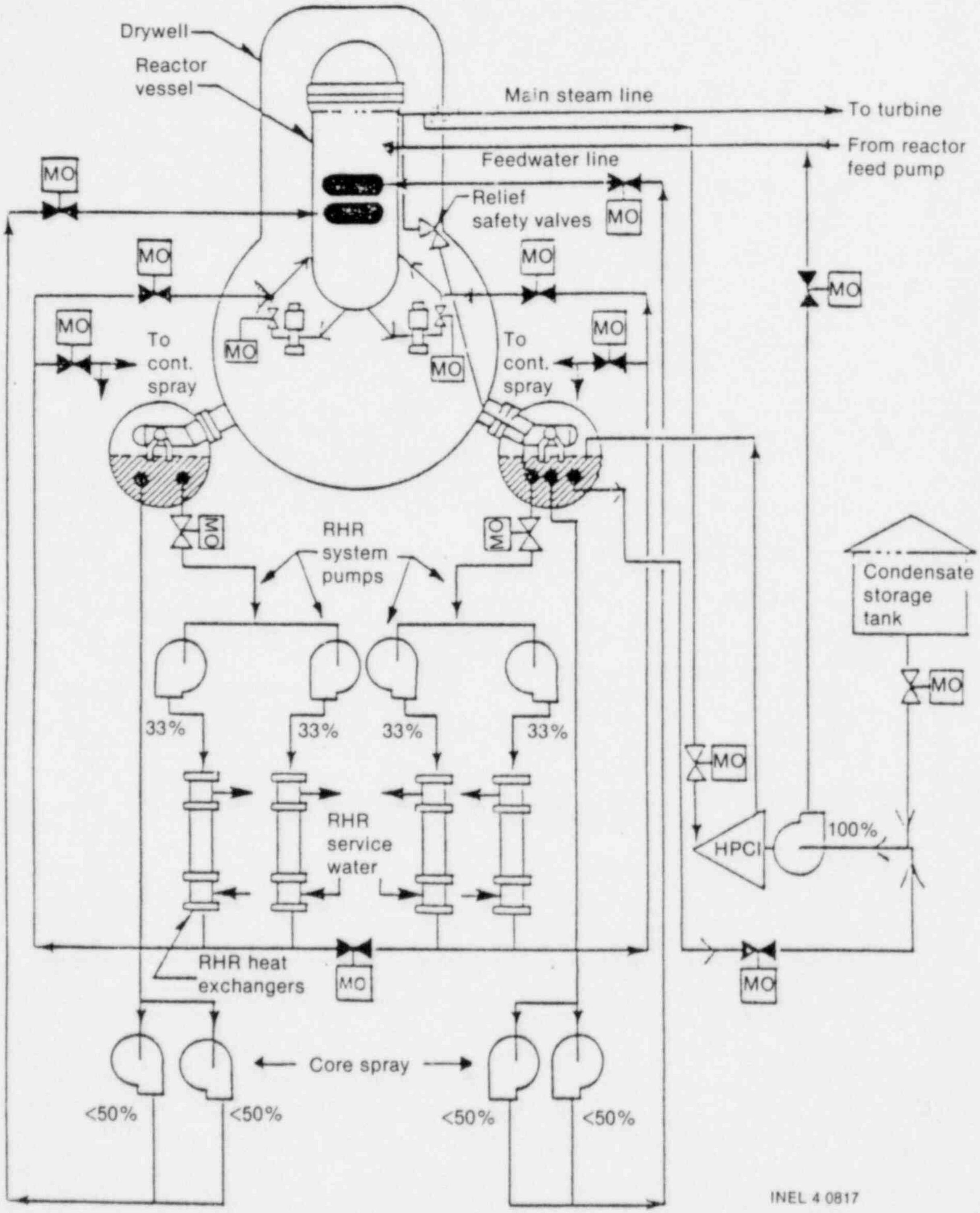

Figure 12-1. Emergency core cooling systems for a BWR/4 (Reference 2).

- 548.4 Btu/lbm (average initial coolant enthalpy)]

+ 1.501 x 10^6 lbm (vessel mass) x 0.11 Btu/lbm (specific heat)

x [328°F (saturation temperature at 100 psia)

- 550°F (approximate initial average vessel temperature)]

+ 8.0 x 10^5 lbm (drywell mass) x 0.11 Btu/lbm

x [300°F (maximum drywell temperature)

- 150°F (initial drywell temperature)]

+ 629,777 lbm (torus steel) x 0.11 Btu/lbm (specific heat)

x [190°F (final temperature) - 90°F (initial temperature)]

+ $\dfrac{123{,}000\ \text{ft}^3}{0.016099\ \text{ft}^3/\text{lbm}}$ (initial suppression pool mass) [158.04 Btu/lbm (final enthalpy)]

- 58.01 Btu/lbm (initial enthalpy)] + 7.4724

x 10^5 lbm (condensate tank liquid added to pool) x [158.04 Btu/lbm (initial enthalpy)]

= 1.522 x 10^8 Btu (heat given up by reactor coolant)

- 3.6654 x 10^7 Btu (heat given up by vessel)

+ 1.32 x 10^7 Btu (heat absorbed by drywell)

+ 6.235 x 10^6 Btu (heat absorbed by torus)

+ 7.642 x 10^8 Btu (heat absorbed by initial pool liquid)

+ 7.474 x 10^7 Btu (heat absorbed by condensate)

= 6.694 x 10^8 Btu.

Thus,

$$\int_0^{t_1} \dot{q}_{dh}\, dt = 7.062 \times 10^5 \text{ MW-s}.$$

Using Section 3 (Decay Heat Power and Integrated Power) to find the time (t_1), the integrated decay heat is divided by the initial reactor power, 3293 MW to obtain 214.5 (MW-s)/MW.

This corresponds to a time (t_1) of about 14,630 s or 4 h and 4 min.

The time is most sensitive to the initial volume and temperature of the suppression pool.

References

1. General Electric Company, *Analysis of Generic BWR Safety/Relief Valve Operability Test Results*, NEDE-24988-P, Class III, October 1981.

2. General Physics Corporation, *BWR Simulator Training*, 1979.

3. Cook, D. H. et al., *Station Blackout at Browns Ferry Unit One—Accident Sequence Analysis*, NUREG/CR-2182, ORNL/NUREG/TM-445/V1, November 1981.

Appendix Information

1. Specific values for equations:

M_{coolr}	=	609,210 lbm, mass of coolant in primary reactor system
M_v	=	1,501,000 lbm, mass of metal structure in reactor system
M_{dw}	=	800,000 lbm, mass of metal structure in drywell
M_t	=	629,777 lbm, mass of metal in torus (includes vent header and downcomers)
T_{v1}	=	550°F, nominal initial average vessel metal mass temperature
T_{V2}	=	328°F, nominal final average vessel metal mass temperature (saturation temperature at 100 psia)
T_{dW1}	=	150°F (135 to 200°F range), normal drywell temperature
T_{dW2}	=	300°F, maximum desirable drywell temperature
T_{t1}	=	90°F, nominal initial suppression pool and torus temperature
T_{t2}	=	190°F, maximum temperature for suppression pool to function
V_{sp}	=	123,000 to 128,400 ft^3 range, suppression pool volume
ν_{sp}	=	0.016099 ft^3/lbm, suppression pool initial specific volume
h_{sp1}	=	58.018 Btu/lbm, suppression pool initial enthalpy
h_{sp2}	=	158.04 Btu/lbm, suppression pool enthalpy at 190°F
h_1	=	548.3 Btu/lbm, nominal average initial reactor system coolant enthalpy
h_2	=	298.54 Btu/lbm, reactor system coolant enthalpy at saturation pressure of 100 psia
ΔM_{sp}	=	$M_{cond} = 7.4724 \times 10^5$ lbm, for conditions assumed
h_{cond1}	=	58.018 Btu/lbm, nominal initial condensate tank liquid enthalpy at 90°F.

2. Full power:

Full power	3293 MW

3. Reactor volumes and coolant masses:

Recirculation piping	1363.6 ft^3, 64,979 lbm
Jet pump	231.9 ft^3, 11,015 lbm
Lower plenum	2196.4 ft^3, 104,342 lbm
Core	1009.7 ft^3, 27,471 lbm
Core exit	126.8 ft^3, 3087.4 lbm
Control rod drive bypass	1277.7 ft^3, 61,623 lbm
Core bypass	940.0 ft^3, 44,497 lbm
Upper plenum	193.1 ft^3, 5117.5 lbm
Standpipes	1084.7 ft^3, 21,421 lbm
Separator	1063.6 ft^3, 20,100 lbm
Dryers	1206.6 ft^3, 2768.7 lbm
Separator bypass	370.5 ft^3, 6131.2 lbm
Steam dome	4896.0 ft^3, 11231.0 lbm
Separator liquid extractor	378.6 ft^3, 17,472 lbm
Downcomer	4300.5 ft^3, 204,243 lbm (2521.2 ft^3 above core)
Steam pipe and safety/relief valve header to isolation valve	1620.4 ft^3, 3864.0 lbm
Feedwater pipe	627.5 ft^3, 34,413 lbm
Total volume and mass in reactor system (excludes feedwater pipe)	22,260 ft^3, M_{coolr} = 609,210 lbm.

4. Safety/relief valve:

Type	Target Rock 2 Stage
Pressure set point	1105 psig (lowest)
Capacity	838,900 lbm/h at 1120 psia

5. System capacities:

RHR—Flow rate	36,000 gpm
RHR—Heat removal	116 x 10^6 Btu/h
RCIC—Flow rate	600 gpm
HPCI—Flow rate	5000 gpm
DWC—Heat removal	6.69 x 10^6 Btu/h

APPENDIX A

BWR AND PWR COMMERCIAL NUCLEAR POWER PLANTS

APPENDIX A

BWR AND PWR COMMERCIAL NUCLEAR POWER PLANTS

This appendix lists BWR (Table A-1) and PWR (Table A-2) commercial nuclear power plants based on containment type and thermal hydraulic parameters.

Reference

Silver, E.G., "Operating U.S. Power Reactors" and "Status of Power-Reactor Projects Undergoing Licensing Review," *Nuclear Safety, 6*, 3, May-June 1985.

Table A-1. United States BWRs grouped[a]

Group	Plant Name	BWR Mark	Containment[b] Mark	Vessel ID/ Number Fuel Bundles	Rated Power (MWt)	Expected or Operational Date	Comments
1	Browns Ferry 1	4	1	251/764	3293	8/74	—
	Browns Ferry 2	4	1	251/764	3293	3/75	—
	Browns Ferry 3	4	1	251/764	3293	3/77	—
	Peach Bottom 2	4	1	251/764	3293	7/74	—
	Peach Bottom 3	4	1	251/764	3293	12/74	—
	Fermi 2	4	1	251/764	3293	/85	—
	Hope Creek 1	4	1	251/764	3293	/86	—
	Limerick 1	4	2c	251/764	3293	/85	—
	Limerick 2	4	2c	251/764	3293	/89	—
	Susquehanna 1	4	2c	251/764	3293	7/82	—
	Susquehanna 2	4	2c	251/764	3293	/84	—
2	La Salle 1	5	2c	251/764	3323	4/82	—
	La Salle 2	5	2c	251/764	3323	/83	—
	Nine Mile Point 2	5	2c	251/764	3323	/86	—
	WNP 2	5	2	251/764	3323	/85	—
3	Clinton 1	6	3c	218/624	2894	/86	—
	Clinton 2	6	3	218/624	2894	?	—
	River Bend 1	6	3	218/624	2894	/85	Motor Driven Feed Pumps
4	Grand Gulf 1	6	3c	238/800	3800	6/82	—
	Grand Gulf 2	6	3c	238/800	3800	?	—
	Perry 1	6	3	238/732	3579	/87	—
	Perry 2	6	3	238/732	3579	/88	—

Table A-1. (continued)

Group	Plant Name	BWR Mark	Containment[b] Mark	Vessel ID/ Number Fuel Bundles	Rated Power (MWt)	Expected or Operational Date	Comments
5	Brunswick 1	4	1c	218/560	2436	3/77	—
	Brunswick 2	4	1c	218/560	2436	11/75	—
	Cooper	4	1	218/548	2831	7/74	—
	Fitzpatrick	4	1	218/560	2436	7/75	—
	Hatch 1	4	1	218/560	2436	12/75	—
	Hatch 2	4	1	218/560	2436	7/79	—
	Shoreham	4	2c	218/560	2436	/85	—
	Zimmer	5	2c	218/560	2436	/86	—
6A	Dresden 2	3	1	251/724	2527	8/70	Simulator
	Dresden 3	3	1	251/724	2527	10/71	—
6B	Quad Cities 1	3	1	251/724	2511	8/72	RHR
	Quad Cities 2	3	1	251/724	2511	10/72	—
7	Millstone 1	3	1	224/580	2011	12/70	—
	Pilgrim 1	3	1	224/580	1998	12/72	—
8	Monticello	3	1	205/484	1670	7/71	—
	Vermont Yankee	4	1	205/368	1593	11/72	—
	Duane Arnold	4	1	183/368	1658	5/74	—
9	Oyster Creek	2	1	213/560	1930	12/69	No Jet Pumps
	Nine Mile Point 1	2	1	213/532	1850	12/69	No Jet Pumps
10	"Others"						
10A	Dresden 1[c]	1	Steel Sphere	146/448	630	8/60	No Jet Pumps
10B	Big Rock Point	1	Steel Cylinder	106/84	240	5/64	No Jet Pumps
	Humbolt Bay	1	Steel Cylinder	120/184	240	8/63	No Jet Pumps
	LaCrosse	Allis	?	?	165	11/69	No Jet Pumps

a. Manufactured by General Electric except for LaCrosse manufactured by Allis Chalmers.

b. Containment Mark number.

1. Drywell with free standing torus.
1c. Drywell with concrete torus with steel liner.
2. Over/under with free standing steel pressure vessel.
2c. Over/under with steel surrounded by concrete.
3. Drywell with free-standing concrete containment.
3c. Drywell with steel-lined concrete containment.

c. Shutdown indefinitely.

Table A-2. United States PWRs grouped

Plant Name	Number of Loops	Number of Bundles	Power Level (MWt)	Loop Isolation Valves (Yes or No)	Porv Number	Porv Air or Electric	Main Feed Turbine or Motor (T)	Main Feed Turbine or Motor (M)	Auxiliary Feed Turbine or Motor (T)	Auxiliary Feed Turbine or Motor (M)	Expected or Operational Date
Group 1: Large high power four-loop Westinghouse with dry containments											
Group 1A											
South Texas 1	4	193	3800	N	2	A	0	3	1	3	/86
South Texas 2	4	193	3800	N	2	A	0	3	1	3	/88
Group 1B											
Braidwood 1	4	193	3425	Y	2	A	2	1	0	2	/85
Braidwood 2	4	193	3425	Y	2	A	2	1	0	2	/86
Byron 1	4	193	3425	Y	2	A	2	1	0	2	/85
Byron 2	4	193	3425	Y	2	A	2	1	0	2	/85
Callaway 1	4	193	3411	N	2	A	2	0	1	2	6/84
Catawba 1	4	193	3411	N	3	A	2	0	1	2	/85
Catawba 2	4	193	3411	N	3	A	2	0	1	2	/87
Comanche Peak 1	4	193	3411	N	2	A	2	0	1	2	/85
Comanche Peak 2	4	193	3411	N	2	A	2	0	1	2	?
Millstone 3	4	193	3411	Y	3	A	2	1[b]	1	2	/85
Seabrook 1	4	193	3411	N	2	A	2	0	0	2	/85
Seabrook 2	4	193	3411	N	2	A	2	0	0	2	?
Trojan	4	193	3411	N	2	A	2	0	1	1	5/76[c]
Vogtle 1	4	193	3411	Y	2	A	2	0	1	1	/86
Vogtle 2	4	193	3411	Y	2	A	2	0	1	1	/88
Wolf Creek	4	193	3411	N	2	A	2	0	1	2	/85
Diablo Canyon 1	4	193	3338	N	3	A	2	0	1	2	/85
Diablo Canyon 2	4	193	3411	N	3	A	2	0	1	2	/85
Group 2: Large high power four-loop Westinghouse with ice containments and ability to inject high GPM at safety valve set point											
Donald C. Cook 1	4	193	3250	N	3	A	2	0	1	2[d]	8/75
Donald C. Cook 2	4	193	3391	N	3	A	2	0	1	2[d]	6/78
McGuire 1	4	193	3411	N	3	A	2	0	1	2	1/81
McGuire 2	4	193	3411	N	3	A	2	0	1	2	3/83
Sequoyah 1	4	193	3423	N	2	A	2	0	1	2	10/80
Sequoyah 2	4	193	3423	N	2	A	2	0	1	2	6/81
Watts Bar 1	4	193	3411	N	2	A	2	1	1	2	/85
Watts Bar 2	4	193	3411	N	2	A	2	1	1	2	/86
Group 3: Medium power four-loop Westinghouse with dry containments											
Group 3A											
Salem 1	4	193	3423	N	2	A	2	0	1	2	12/76[c]
Salem 2	4	193	3423	N	2	A	2	0	1	2	4/80[c]
Zion 1	4	193	3250	Y	2	A	2	1	1	2	6/73[c]
Zion 2	4	193	3250	Y	2	A	2	1	1	2	12/73[c]
Indian Point 3	4	193	2760	N	2	A	2	0	1	2	8/76
Indian Point 2	4	193	2758	N	2	E	2	0	1	2	7/74
Group 3B											
Haddam Neck	4	157	1825	Y	2	A	0	2	1	0	1/68
Yankee Rowe	4	76	600	Y	3	E	0	3	1	0	6/61

Table A-2. (contin ed)

Plant Name	Number of Loops	Number of Bundles	Power Level (MWt)	Loop Isolation Valves (Yes or No)	Porv Number	Porv Air or Electric	Main Feed Turbine or Motor (T)	Main Feed Turbine or Motor (M)	Auxiliary Feed Turbine or Motor (T)	Auxiliary Feed Turbine or Motor (M)	Expected or Operational Date
Group 4: Medium power three-loop Westinghouse with subatmospheric containments and ability to inject high GPM at safety valve set point											
Beaver Valley 1	3	157	2652	Y	3	A	0	2	1	2	4/77
Beaver Valley 2	3	157	2660	Y	3	A	0	2	2[e]	0	/85
North Anna 1	3	157	2775	N	2	A	0	3	1	2	6/78
North Anna 2	3	157	2775	N	2	A	0	3	1	2	7/79
Surry 1	3	157	2441	Y	2	A	0	3	1	2	12/72
Surry 2	3	157	2441	Y	2	A	0	2	1	2	5/73
Group 5: Medium power three-loop Westinghouse with dry containments											
J. M. Farley 1	3	157	2652	N	2	A	2	0	1	2	12/77[c]
J. M. Farley 2	3	157	2652	N	2	A	2	0	1	2	10/80[c]
Shearon Harris 1[f]	3	157	2775	N	2	A	0	3	1	2	/85
Virgil C. Summer	3	157	2775	N	3	A	3	0	1	2	8/82[c]
Robinson 2	3	157	2200	N	2	A	0	2	1	2	3/71
Turkey Point 3	3	157	2200	N	2	A	0	2	3[d]	0	12/72
Turkey Point 4	3	157	2200	N	2	A	0	2	3[d]	0	9/73
San Onofre 1	3	157	1347	N	2	A	0	2	0	1	1/68
Group 6: Small low power two-loop Westinghouse with dry containments											
Norco 1 (Puerto Rico)	2	121	1785	N	2	A	0	3	1	2	Indef.
Kewaunee	2	121	1650	N	2	A	0	2	1	2	6/74
Prairie Island 1	2	121	1650	N	2	A	0	2	1	2[d]	12/73
Prairie Island 2	2	121	1650	N	2	A	0	2	1	2[d]	12/74
Point Beach 1	2	121	1518	N	2	A	0	2	1	2[d]	12/70
Point Beach 2	2	121	1518	N	2	E	0	2	1	2[d]	10/72
R. E. Ginna	2	121	1520	N	2	A	0	2	1	2	3/70
Group 7: Combustion Engineering with dry containments											
Group 7A											
Maine Yankee	3	217	2560	Y	2	A	0	2	1	2	12/72[c]
Group 7B											
Palo Verde 1	2	241[g]	3817	N	0		2	0	0	2	/85
Palo Verde 2	2	241[g]	3817	N	0		2	0	0	2	/85
Palo Verde 3	2	241[g]	3817	N	0		2	0	0	2	/87
WNP 3	2	241[g]	3800	N	0		2	1[h]	0	2	?
Group 7C											
San Onofre 2	2	217[g]	3410	N	0		2	0	1	1	2/82
San Onofre 3	2	217[g]	3410	N	0		2	0	1	1	11/82
Waterford 3	2	217[g]	3410	N	0		2	0	1	2	/85
Group 7D											
Calvert Cliffs 1	2	217	2560	N	2	E	2	0	2	0	5/75
Calvert Cliffs 2	2	217	2560	N	2	E	2	0	2	0	4/77
Millstone 2	2	217	2560	N	2	E	2	0	1	2	12/75
St. Lucie 1	2	217	2560	N	2	E	0	2	1	2	12/76
St. Lucie 2	2	217	2560	N	2	E	0	2	1	2	4/83

Table A-2. (continued)

Plant Name	Number of Loops	Number of Bundles	Power Level (MWt)	Loop Isolation Valves (Yes or No)	Porv		Main Feed Turbine or Motor		Auxiliary Feed Turbine or Motor		Expected or Operational Date
					Number	Air or Electric	(T)	(M)	(T)	(M)	
Group 7: Combustion Engineering with dry containments (continued)											
Group 7E											
Palisades	2	204	2200	N	2	E	2	0	1	0	12/71
Group 7F											
ANO-2	2	177[i]	2815	N	0		2	0	1	1	8/79
Group 7G											
Fort Calhoun	2	133	1420	N	2	E	0	3	1	1	6/73
Group 8: Babcock and Wilcox with dry containments and ability to inject high GPM at safety valve set point											
Group 8A[j]											
Bellefonte 1	2	205	3600	N	1	E	2	0	1	2	9/85
Bellefonte 2	2	205	3600	N	1	E	2	0	1	2	6/86
WNP-1	2	205	3600	N	1	E	0	3	1	2	?
Group 8B[k]											
Davis-Besse 1[j]	2	177	2772	N	1	E	2	1[m]	2	0	11/77
Crystal River 3	2	177	2560	N	1	E	2	1	1	1	3/77
Midland 1	2	177	2452	N	1	E	2	1	1	1	?
Midland 2	2	177	2452	N	1	E	2	1	1	1	?
ANO-1	2	177	2568	N	1	E	2	1	1	1	12/74
Rancho Seco	2	177	2772	N	1	E	2	1	1[n]	1	4/75
Oconee 1	2	177	2568	N	1	A	2	1	1	2	7/73
Oconee 2	2	177	2568	N	1	A	2	1	1	2	7/74
Oconee 3	2	177	2568	N	1	A	2	1	1	2	12/74
TMI-1	2	177	2535	N	1	E	2	1	1	2	9/74[l]
TMI-2	2	177	2772	N	1	E	2	1	1	2	12/78

a. One pump is diesel driven through a gear increaser.

b. On standby, uses gear increaser.

c. Ability to inject high gpm at safety valve set point.

d. Pumps shared by each unit.

e. These two pumps are actually dual drive; the turbine is priority.

f. Plants with large load decrease option; PORVs will accommodate a 95% loss of load.

g. 14 ft core.

h. Startup pump, 22% of flow.

i. 12.5 ft core.

j. High loop.

k. Low loop.

l. License suspended.

m. Used for startup and maybe for cooldown.

n. Dual drive, turbine on one end and motor on other, Rancho Seco is counting this as a turbine.

BELLEFONTE
GROUP 8A: BABCOCK AND WILCOX WITH DRY CONTAINMENT AND ABILITY TO INJECT HIGH GPM AT SAFETY VALVE SET POINT

2. BREAK FLOW RATE

1. Specific values for equations:

   L_{ave} = 52.104 ft, average steam generator tube length

   D = 0.5568, steam generator tube inside diameter

   A = 1.7609×10^{-3} ft^2, steam generator tube flow area

2. Tube sheet:

   Thickness: 1.865 ft

4. STEAM GENERATION FROM DECAY HEAT

1. Specific values for equations:

   M_{ls} = 65,220 lbm, total nominal liquid in secondary side of all steam generators

   h_{fg} = 639.0 Btu/lbm, nominal latent heat of vaporization of liquid in steam generator secondary side, P = 1060.

   m_m = 2,121,700 lbm, total mass of miscellaneous vessel internals, primary system piping, pumps, steam generator tubes, tube plate and channel head, cladding

   m_f = 239,074 lbm, total mass of UO_2

   V_{1c} = 11,515 ft^3, volume of primary system excluding pressurizer, surge line, and accumulator lines

   T_1 = 600.6°F, initial average primary system coolant

   T_2 = 688°F, saturation temperature at 2500 psia, statepoint 2

   ρ_1 = 43.12 lbm/ft^3, primary coolant liquid density at initial average statepoint, p = 2250, T = 600.6°F

   ρ_2 = 34.98 lbm/ft^3, saturated liquid density at 2500 psia, statepoint 2

   h_1 = 614.12 Btu/lbm, primary coolant enthalpy at statepoint 1

   h_2 = 731.7 Btu/lbm, saturated liquid enthalpy at 2500 psia, statepoint 2

   T_{f1} = 1520°F, nominal initial average UO_2 temperature

   T_{f2} = 668°F, final average UO_2 temperature

   V_v = 1951.2 ft^3, primary system coolant displaced by initial void generation

V_{P+S} = 2296.8 ft^3, volume of pressurizer and surge line

V_{PH} = 713.9 ft^3, volume of surge line and pressurizer below elevation of surge line connection to hot leg

V_R = 3376.3 ft^3, liquid volume remaining in primary system below top of core and downcomer inlet

2. Coolant volumes (ft^3):

Vessel

Lower plenum	768.4
Core	689.5
Core bypass	287.3
Upper plenum including guide tubes	1279.7
Upper plenum above hot leg centerline	760.2
Upper plenum below hot leg centerline	519.5
Upper head	641.0
Downcomer above inlet	134.7
Downcomer below inlet	789.1
Vessel total	4589.7

Piping

Hot leg total	1499.0
Cold leg total	228.8
Intermediate leg total	351.6
Intermediate leg and steam generator below downcomer inlet	955.4
Pump total	657.2
Piping total	2736.6 (hot leg, cold leg, intermediate leg, pump)
Surge line	46.8
Pressurizer	1100 liquid, 1150 vapor,
Steam generator primary total	4190
Steam generator secondary total	1422 liquid (12,776 lbm vapor, 65,220 lbm liquid)
Accumulator	Not available
Accumulator line	Not available

3. Structural masses (lbm):

Vessel

Vessel shell	760,300
Vessel head	206,300
Vessel studs, nuts, washers	48,100

Vessel Internals

Core cladding	51,762
Core UO_2	239,074
Core barrel	160,579
Lower plenum internals	81,580
Core baffle assembly	60,344

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