

VERMONT YANKEE NUCLEAR POWER CORPORATION

SEVENTY SEVEN GROVE STREET
RUTLAND, VERMONT 05701

2.C.2.1
FVY 82-50
REPLY TO:

ENGINEERING OFFICE
1671 WORCESTER ROAD
FRAMINGHAM, MASSACHUSETTS 01701
TELEPHONE 617-872-8100

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Office of Nuclear Reactor Regulation
Mr. Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing



References: a) Letter, USNRC to VYNPC, dated 4/2/82
b) NUREG 0737 Item Nos. I.A.2.1. and II.B.4
c) Letter, VYNPC to USNRC, dated 12/15/80

Dear Sir:

50-271

Subject: Upgraded SRO and RO Training and Training for Mitigating
Core Damage

Answers to the questions in Enclosure 1 of Reference (a) are submitted
as follows:

Item 1: Does the Initial Operator Licensing program described in procedure
AP 0710 teach the subjects of heat transfer, fluid flow and
thermodynamics at a level of detail which is comparable to that
presented in Enclosure 2 of Denton's March 28 letter?

Response: The program described in AP 0710 presents heat transfer,
fluid flow and thermodynamics at a level of detail at least
commensurate to that presented in Enclosure 2 of Denton's
March 28 letter.

Item 2: The Licensed Operator Retraining program described in AP 0711
includes lecture materials on heat transfer, fluid flow and
thermodynamics. Do these instructions also cover the range of
topics indicated in Enclosure 2 of Denton's March 28 letter?

Response: The retraining program described in AP 0711 includes lecture
materials on heat transfer, fluid flow and thermodynamics.
The topics covered are in Enclosures 1-5 which are handout
material. Other series of lectures cover pertinent heat
transfer topics.

8205100180 820505
PDR ADDCK 05000271
P PDR

A046
S
1/1

Item 3: Do the programs for Initial Operator Licensing and for Licensed Operator Retraining address the subject of using installed plant systems to control or mitigate an accident in which the core is severely damaged? This requirement is called out in Enclosure 1 of Denton's letter. Do the lectures in the programs address the topic at the level of detail specified in Enclosure 3 of Denton's letter?

Response: The Initial Operator Licensing program and the Licensed Operator Retraining Program covers the use of installed plant systems which are designed to control and mitigate an accident involving core damage. Topics covered are in the outline (Enclosure 6). The GE course on recognition of a degraded core condition is the basis of our training program. During retraining more detailed information on mitigating core damage is presented as part of Plant Safety and Emergency Systems.

Item 4: Does the training program have an increased emphasis on reactor and plant transients as called for in Enclosure 1 of Denton's March 28, 1980 letter? Does the program address both normal and abnormal (accident) transients?

Response: The training programs for Initial Operator Licensing, Licensed Operator Retraining and Shift Technical Advisors contain increased emphasis on both normal and abnormal (accident) reactor and plant transients. These events are covered in both classroom and in simulator training sessions.

Item 5: As called for in Denton's March 28, 1980 letter, are the training program instructors enrolled in a requalification program which addressed current operating history, problems and changes to procedures, and administrative limitations?

Response: Training program instructors are routed all material and included in training sessions related to operating history, problems with and changes to procedures and administrative limitations.

Item 6: Appendices A and B of the licenses operator retraining program lists certain control manipulations which are performed on an annual or semiannual basis. Appendix B is missing one control manipulation, "Reactor Trip," which corresponds to item (25) in Enclosure 4 of Denton's March 28, 1980, letter. Would you please explain this omission?

Response: Vermont Yankee's Licenses Operator Retraining procedure AP 0711 does not contain "Reactor Trip" in Appendix B. Fourteen of the twenty identified evolutions listed in Appendix B will result in a reactor trip. Since there are initiating events for the condition, the condition itself was not listed.

Item 7: Are the lectures and quizzes on the subject of accident mitigation given to shift technical advisors and operating personnel from the plant manager through the operations chain to the licensed operators? If they are, would you please provide the titles of the people who are trained and an organization chart which illustrates their position in the operations chain?

Response: Lectures and quizzes on the subject of accident mitigation are given to shift technical advisors and licensed operations personnel through the Operations Superintendent. An organization chart has been included in Enclosure 8. As stated in Reference (c), Training of the Plant Manager and the Assistant Plant Manager, has not been completed nor has it been completed for managers and technicians in the Instrument and Control and Chemistry and Health Physics departments. Vermont Yankee does not feel that this training is necessarily warranted based on their job functions and prior training.

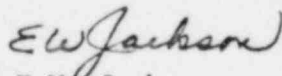
Item 8: Do the training and the requalification program elements which include heat transfer, fluid flow, thermodynamics and accident mitigation use 80 contact hours? (A contact hour of instruction is a one-hour period in which the course instructor is present or available for instructing or assisting students; lectures, seminars, discussions, problem-solving sessions, and examinations are considered contact periods under this definition.)

Response: Instruction in heat transfer, fluid flow, thermodynamics and accident mitigation for initial licensed operator training use 80 or more contact hours. A description of this program is provided in response to your inquiry for item II.B.4 of NUREG 0737. See Enclosure 7. The instructional program for requalification training in these specific program elements is at least 40 contact hours. Coincident material is covered during training in other subjects so a more precise estimate of contact hours is unavailable.

We trust this information is acceptable. However, should you have further questions regarding this submittal, please contact us.

Very truly yours,

VERMONT YANKEE NUCLEAR POWER CORPORATION



E.W. Jackson
Manager of Operations

EWJ/dm

cc: Dr. R.J. Liner
Science Applications, Inc.
1710 Goodridge Drive
McLean, Virginia 22102

THERMAL HYDRAULICS

Sufficient information is included in this section to provide you with a basic understanding of BWR thermal hydraulics. However, if you desire to obtain a more in-depth understanding of the subject you should study the following documents:

1. NEDO 10958 - GETAB
2. Nuclear Heat Transport - M.M. El Wakil
3. Boiling Crisis and Critical Heat Flux TID
25887 - L. S. Tong

Heat Transfer

In light water reactor operation we are concerned with the transfer of heat from the fuel center line (C_L) to the light water moderator which comes into contact with the outer fuel cladding surface. As you know, heat can be transferred by (1) Conduction, (2) Convection, and (3) Radiation. The first two listed modes of heat transfer are the one of primary interest in power plant operations.

Conduction

When heat is applied to a material, the kinetic energy of the atoms or molecules of the material is increased. Due to this increase in kinetic energy the particles mentioned will have a greater tendency to collide with each other. When these collisions occur, the particle with the least amount of energy (cold) will gain energy (heat).

For all practical purposes this is the process by which heat generated in the fuel pellet is transmitted to the outer clad surface. In relation to the fuel rod this conduction flow is in a horizontal plan

from the fuel center line to the cladding surface.

The rate at which heat is transferred by conduction is shown in the following equation:

$$\text{Conduction heat transfer rate} = \frac{\text{Thermal Conductivity} \times \text{Area} \times \text{Temperature Gradient}}{\text{material thickness}}$$

Convection

Convection is the process of transmitting heat by means of the movement of heated matter from one place to another. Convection takes place in liquids and gases. Our application deals with a fluid (moderator) flowing past a metallic fuel clad surface. Fluids have a tendency to adhere to solid surfaces resulting in the formation of a stagnant film on the surface. This film is normally very thin and heat is transferred across this film by a combination of conduction and convection.

After the heat penetrates the film, it is transferred rapidly through the remainder of the fluid. The resistance to temperature flow in this area becomes so low that there is virtually no temperature variation through the bulk of the fluid (moderator) at any given elevation along the fuel rod. It should be noted that the flow of heat in the fuel channel is in a vertical plane due to upward flow in the core.

The convection heat transfer rate from the heated surface, through the stagnant film and into the bulk fluid is governed by the same fundamental principles as conduction heat transfer rate, in that:

$$\text{Convection heat transfer rate} = \frac{\text{Thermal Conductivity} \times \text{Area} \times \text{Temperature Gradient}}{\text{stagnant film thickness}}$$

Heat Transfer in a BWR

In a BWR the primary heat transfer mechanism at the fuel cladding outer surface is convection; i.e., the mechanical migration of water molecules, from the stagnant water layer covering the clad surface, out into the bulk moderator. The temperature difference, ΔT , between the bulk coolant and the fuel clad surface is governed by:

$$\Delta T = \frac{Q}{hA}$$

where

Q = heat flow (btu/hr)

h = heat transfer coefficient (Btu/hr - ft² - °F)

A = surface area

The heat transfer coefficient (h) is also referred to as the convection or boiling heat transfer coefficient or the film coefficient.

Boiling Heat Transfer

Boiling is the evaporation of a liquid to vapor occurring within the body of the liquid by the mechanism of bubble formation. The very name "Boiling Water Reactor" implies that dependency is placed upon the boiling of water to dissipate heat that is generated in the reactor fuel.

When boiling occurs in a BWR, bubbles form on the fuel clad surface and then drift off into the bulk coolant giving off their energy to the cooler surrounding water. When these bubbles form, they tend to agitate or stir up the stagnant water film on the clad surface and improve the thermal conductivity of the film. Also each bubble carries off more energy than is possible by non-boiling convection heat transfer, thus increasing the total heat removal rate. Three types of boiling are of concern in the operation of the BWR. They

are nucleate boiling, transition boiling and film boiling.

Nucleate Boiling

By definition nucleate boiling means that small steam bubbles are formed at sites of nucleation, which are usually small imperfections on the heated surface. In a BWR nucleate boiling is characterized by a film of water clinging to the fuel rod surface with vigorous steam bubble formation and movement. Nucleate boiling is extremely important in reactor core heat transfer because, as shown in Figure 1.2-1 a great deal of heat energy can be transferred without extremely high surface temperature.

Transition Boiling

As power (heat flux) is increased, bubbles form at a faster rate. Eventually a point is reached where the cooling media cannot reach the heated surface as efficiently as before. This point is called departure from nucleate boiling (DNB) and is evidenced by a reduction in the amount of heat transfer for a given ΔT . DNB is characterized by local flow and temperature oscillations brought about by the changing steam bubble conditions. Experimentation has shown oscillations in fuel cladding temperatures of 50°F to 100°F at DNB. If a BWR is operated beyond DNB, there is the probability, due to excessive clad temperatures, that weakening and/or failure of the clad would occur. Excessive void formation will also cause deposition of impurities on the fuel clad (boiler scale) resulting in excessive corrosion film thickness and higher clad ΔT .

Film Boiling

Beyond transition boiling film boiling occurs. This is characterized by the surface of the cladding being blanketed with steam. Steam is an excellent insulator, therefore, the clad temperature would rise dramatically. In all probability, the clad would fail (burnout) if full film boiling occurred. Obviously, full film boiling is never permitted in an operating reactor under ordinary or transient conditions.

Fluid Flow

This section will deal with flow patterns from channel entry to exit. Figure 1.2-2 shows the different flow patterns that can exist on a single fuel rod during normal operation.

As can be seen from Figure 1.2-2, at a point along the rod the fluid loses its homogeneous character. At this point single phase flow changes to two phase flow. Two phase flow occurs from the area of bubble flow to annular flow. Note in particular that for high steam quality conditions, low moisture content in steam, the annular flow pattern occurs with a thin liquid film clinging to the fuel rods.

DNB and Two Phase Flow

As previously stated, a BWR operates below DNB. Yet Figure 1.2-2 clearly shows that DNB has been exceeded in the moderator channel. The statement is in reference to liquid on the clad surface and as long as a film of liquid exists on the clad surface, the requirement is satisfied.

Steam Volume Fraction and Slip Ratio

Steam volume (void) fraction is defined as the fraction of the flow area occupied by the steam, compared to the total flow area occupied

by steam plus water.

If homogeneous flow were assumed, the local steam volume fraction could be calculated based on the pressure and local steam quality and specific volumes for saturated steam and water found in the steam tables. Steam quality (x) being defined as the mass flow rate of steam (W_G) divided by the mass flow rate of steam plus the mass flow rate of water (W_L)

$$x = \frac{W_G}{W_G + W_L}$$

Frequently x is multiplied by 100 and expressed in percent.

Separation of phases occurs due to different buoyant effects and viscosity. The velocity of steam is not equal to the velocity of water in a boiling channel. The ratio of steam velocity to water velocity is defined as the Slip Ratio. Slip ratio increases as coolant goes up to the core. However, the rate of slip increase decreases. As bubbles accelerate, they tend to pull the liquid along. This effect increases the liquid velocity, hence the ratio increases at higher core elevations.

Fuel Assembly Pressure Drop

In a fuel assembly local pressure losses occur at the orifice, lower tie plate, spacers, and upper tie plate.

Further pressure losses would be incurred due to the effects of two phase flow. As flux increases, slip ratio increases, friction between the gas (steam) and liquid (film on cladding) increases and overall assembly pressure drop increases.

Fuel Channel Flow and Temperature Characteristics

Coolant enters the bottom core, flows upward around the fuel rods, absorbs energy from heat transfer originating from the nuclear process. Due to the peculiar characteristics of neutron caused fission reactions the average heat flux (Q/A) produced from fission in the core would assume a shape somewhat like that shown in Figure 1.2-3. The highest heat flux will be in the core interior, hence some fuel bundles will have a higher than average heat flux and some a lower than average.

The coolant temperature curve rises as heat is added, until temperature saturation occurs and coolant bulk boiling begins. From this point the coolant temperature remains constant all the way to core exit. Note the sharp temperature rise where there is a large value of heat flux. Because the coolant is changing phase, the coolant temperature profile is not altogether descriptive of coolant energy increase. A better description is obtained by plotting coolant enthalpy change.

The curve for fuel rod surface temperature rises and then levels at a constant value above coolant temperature. The initial rise is caused by the ΔT across the film required to accommodate the heat flux (Q/A). The point where the fuel rod temperature levels off is due to the inception of nucleate boiling. Nucleate boiling is an excellent heat transfer mode and therefore even though the heat flux is getting greater, the ΔT across the boiling film remains relatively constant.

The curve for fuel rod center line temperature will of course be above the fuel surface temperature. The amount that the center line

temperature is greater than surface temperature will depend directly on the heat flux. The beneficial effects of nucleate boiling on center line temperature can also be seen. As long as nucleate boiling is occurring on the fuel rod surface, the fuel rod surface temperature is only slightly greater than liquid temperature. This, in turn, automatically keeps the fuel center line temperature at a lower value than if non-boiling convection were the mode of heat transfer from surface to liquid. Figure 1.2-4 graphically illustrates a typical fuel temperature cross-section with nucleate boiling at a high heat flux.

Thermal Limits - Introduction

For normal and transient operation of the BWR it is necessary to limit peak fuel enthalpies in order to limit fuel pellet vapor pressure. A rapid melting and vaporization of fuel pellets and subsequent melting (burnout) and/or rupture of fuel cladding would represent the worst case in relation to failure to maintain thermal limits.

Thermal Limits - Terms

The following terms are used in conjunction with thermal limits:-

1. Plastic Strain - a plastic deformation of the fuel cladding due to excessive heat.
2. Linear Heat Generation Rate Limits (Kw/ft) - required to limit steady-state plastic strain.
3. Maximum Linear Heat Generation Rate - prescribed in order to maintain an adequate margin below a 1% clad plastic strain limit.

4. On set of Center Line (\bar{L}) Melt - the amount of heat required to begin melting the center of a fuel pellet.
5. Planar Linear Heat Generation Rate - the heat generation along a horizontal axial at any given elevation of a fuel bundle.
6. Average Planar Linear Heat Generation Rate (APLHGR) - the average LHGR of all fuel rods in a 6 inch node of a fuel bundle.
7. MAPLHGR - The Maximum APLHGR.
8. Total Peaking Factor - a combination of axial, radial and relative peaking factors.
9. Critical Power Ratio (CPR) - Critical Power Ratio (CPR) is the ratio of that assembly power which causes some point in the assembly to experience transitionboiling to the assembly power at the reactor condition of interest as calculated by application of the GEXL correlation.
10. Minimum Critical Power Ratio (MCPR) - the minimum CPR corresponding to the most limiting fuel assembly in the core.
11. Enthalpy - heat content; sensible heat; total heat; sum of internal energy of a system plus the product of the systems volume multiplied by the pressure exerted on the system by its surroundings.

Heat Generation, Fuel and Clad Limits

In maintaining thermal limits consideration must be given to two major areas; namely, limiting the deterioration of the heat transfer mechanism from clad to water and limiting the internal stresses from excessive heating of the fuel pellet. In relation to the latter

consideration maintaining the design peak LHGR (See Table 1.2-2) will ensure that overpowering of a fuel pellet does not occur.

Fuel Densification

As a reactor is taken from cold to power, the fuel densifies and expands. Densification is caused by annulation of pores found throughout the ceramic fuel pellet. This annulation occurs when the pellet is exposed to a neutron flux.

Expansion is caused by the temperature increase in the pellet. The net result of densification and expansion during ascension to power is an overall expansion. In setting thermal limits for fuel, thermal expansion is not taken into consideration, but the densification is.

If fuel densification is taken into consideration during heatup, then a factor that would have to be dealt with in any calculations to limit fuel cladding temperatures would be power spiking. As densification occurs, gaps are created between fuel pellets which have a positive reactivity effect due to the change in moderator to fuel ratio. Also increases in LHGRs result from axial shrinkage of the pellets. So, in the formula used to ensure that the design peak LHGR is not exceeded, a spiking penalty is incorporated.

Average Planar Linear Heat Generation Rate (APLHGR)

To ensure that the reactor fuel cladding does not reach the limit in temperature, designated as Peak Cladding Temperature (PCT) (i.e., 2200°F), on a design basis loss of coolant accident a limit on average planar linear heat generation rate is established.

The maximum clad temperature following a postulated loss of coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent, secondarily, on the rod to rod power distribution within the assembly. The PCT is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR corrected for densification.

Core Thermal Design

A comparison of BWR core parameters is made in Table 1.2-1. This listing will be used in the remainder of this chapter so that meaningful values can be associated with the subject matter.

Peaking Factors

Peaking factor can be defined as the power or flux of interest divided by the average power or flux of interest.

The end result of a discussion of peaking factors is determining a design limit called the total peaking factor. The Total Peaking Factor (TPF) tells you the maximum local heat flux in relation to the average power of the core. (See Table 1.2-1 for Design Total Peaking Factors.) Limiting safety system settings are based on the design total peaking factor, therefore, if the actual total peaking factor exceeds the design total peaking factor there is a possibility of fuel damage occurring on a transient. Technical specifications impose limitations to ensure that fuel damage does not occur under these circumstances.

In our discussion we will develop the process by which TPFs are determined and discuss the technical specification requirements in relation to this limit.

Total Peaking Factor (TPF)

The basic formula for TPF is:

$$\text{TPF} = \text{Axial Peaking Factor} \times \text{Local Peaking Factor} \times \text{Relative Peaking Factor}$$

$$\text{Axial Peaking Factor} = \frac{\text{Max. Fuel Rod Axial Heat Flux}}{\text{The Average Heat Flux for that Rod}}$$

$$\text{Local Peaking Factor} = \frac{\text{Average Heat Flux for Highest Powered Fuel Rod in Bundle}}{\text{Bundle Average Fuel Rod Heat Flux}}$$

$$\text{Relative Peaking Factor} = \frac{\text{The Bundle Power of Interest}}{\text{Average Core Bundle Power}}$$

This formula (TPF), when combined with the core average heat flux, will tell you the highest heat flux experienced in the core. A more understandable formula for TPF then might be:

$$\text{TPF} = \frac{\text{Maximum Heat Flux in Core}}{\text{Core Average Heat Flux}}$$

Technical specifications use a ratio, TPF design over actual maximum TPF, (MTPF), to reset the required scram and rod block trip points for the Average Power Range Monitors (APRM) when design values for TPFs are exceeded. However, this has been changed.

In Standard Technical Specifications (STS) the design TPF is defined as the ratio of the fuel rod design LHGR divided by the design average LHGR of that core. This TPF limit is specified for the type of fuel being used, i.e., for a specific BWR-4 the numbers are 2.63 (7 x 7 fuel assemblies) and 2.44 (8 x 8 fuel assemblies). MTPF is referred to in some technical specifications as Limiting Total Peaking Factor (LTPF).

A typical formula in technical specifications using these values to establish APRM settings in case of TPF exceeding the designed values would be:

$$\text{Scram Setting (\% power)} = (.66W + 54Z) \times \frac{\text{MFLPD}}{\text{FRP}}$$

$$\text{Rod Block Setting (\% power)} = (.66W + 42Z) \times \frac{\text{MFLPD}}{\text{FRP}}$$

W = Recirculation loop drive flow, the flow at the discharge of the recirculation pump.

NOTE: As long as $\frac{\text{MFLPD}}{\text{FRP}}$ is less than 1.0, the factor is ignored.

Only when it is greater than 1.0 is it used to lower APRM gain settings. Maximum Fraction Limiting Power Density 13.4 KW/ft² (8 x 8). Fracture of rated power 1593 MWt.

Fuel Cladding Integrity Safety Limit

The fuel cladding safety limit, as stated in the Standard Technical Specifications (STS), is a Minimum Critical Power Ratio (MCPR) of 1.07. This number can be broken down into two separate parts. The foundation for the safety limit is a MCPR of 1.0; that value where the onset of transition boiling occurs. Because of the uncertainties in the MCPR calculations, .07 is added for conservatism to bring the number up to 1.07. This number represents the actual safety limit. For limiting conditions for operations a third number is added. The different transients are analyzed to determine their effect on Critical Power Ratio (CPR). The ACPR for the transient having the greatest effect will be added to the safety limit. In a particular BWR/4 technical specifications this number is .2. It should be noted that the third number will vary from plant to plant, depending upon their own transient analysis package. We can now assemble the parts and come up with the limiting condition for operation on MCPR of 1.27.

Critical Power (CP)

CP is defined as that power where departure from nucleate boiling occurs. In the term Critical Power Ratio (CPR), the CP is in relations to a

single fuel bundle and to the state of boiling on the clad surfaces within that bundle. Therefore, if a fuel bundle reaches its critical power, then departure from nucleate boiling (transition boiling) has occurred at some point on the fuel cladding surface within the bundle. It should be noted that the occurrence of transition boiling does not mean that clad perforation will occur. At the time of transition boiling, the clad at that point would jump in temperature approximately 50°F and continue to increase in temperature to approximately 1100°F . 1100°F is below the perforation temperature of the cladding material. Test loops of fuel similar to that used in the BWR have operated above CP for 30 minutes without experiencing clad perforation.

Critical Power Ratio (CPR) Figure 1.2-5

CPR as defined and illustrated by Figure 1.2-5, CPR being a relationship between lines 1 & 3 on this figure. The axis on this figure are boiling length (L_B) and bundle average steam quality (X). Boiling length is defined as the distance from the point of initiation of bulk boiling to the point of initiation of transition boiling.

Figure 1.2-2 illustrates the fuel channel boiling conditions. Item C on this figure is labeled "bubble flow" and is that point in the fuel channel where the bulk coolant stream has reached saturation temperature. This is the point identified as "initiation of bulk boiling" used in the definition of L_B . "The boiling transition point" used in the definition of L_B refers to any point on the clad surface above that point where initiation of transition boiling occurs.

The units on the horizontal axis in Figure 1.2-5 would be measurements of length (feet, meters, etc.), with zero being at the extreme left of the axis.

The vertical axis on Figure 1.2-5 is the average steam quality in the coolant channel at the boiling length on interest. Steam quality being defined as the amount of moisture in a given volume of steam; the less moisture, the higher the quality. Steam quality is expressed in percent. The lines on Figure 1.2-5 are numbered, Line 1 represents a heat balance, integrated curve, for a single bundle in the core at a specific set of conditions; i.e., flow, flux, pressure, etc. Line 2 represents an integrated heat balance curve if the power in the bundle, represented by Line 1, were increased to CP. Line 3 is explained in the next paragraph.

The GEXL Correlation

Line 3 on Figure 1.2-5 is developed by an iterative process and represents the end results of approximately 1100 data points under various operating conditions. These operating conditions are specified below and they establish the boundaries under which this line is considered to be valid. This line is called the GEXL correlation line and represents a CP heat balance curve for any bundle at any boiling length. The point on Figure 1.2-5 where Line 2 becomes tangent to Line 3 represents the L_B where departure from nucleate boiling would occur in the bundle. A ratio of Line 3 (numerator) and Line 1 (denominator) is the CPR or in other words, the ratio of critical bundle power to operating bundle power. That fuel bundle in the core which has the minimum CPR (MCPR), closest to 1.0, is compared with the plant's technical specifications fuel clad integrity safety limit to determine if the plant's thermal limits are being violated.

The GEXL correlation is valid over a range of conditions used in the test data to develop the correlation. These conditions are:

Pressure:	800 to 1400 psia	
Mass Flux:	0.1 to 1.25 10^6 lb/hr-ft ²	
Inlet Subcooling:	0 to 100 Btu/lb	
Local Peaking:	1.61 at a corner rod to 1.47 at an interior rod	
Axial Peaking:	Shape	Max./Avg.
	Uniform	1.0
	Outlet Peaked	1.60
	Inlet Peaked	1.60
	Double Peaked	1.46 and 1.38
	Cosine	1.39
Rod Array:	16, 64 rods in an 8 x 8 array, 49 rods in a 7 x 7 array	

Core Thermal Power Limit (Reactor Pressure \leq 800 psia or Core Flow \leq 10% Rated)

The use of the GEXL correlation is not valid for the critical power calculations at pressures below 800 psia or core flows less than 10% of rated. For these conditions the fuel cladding integrity safety limits is established by other means. This is done by establishing a limiting condition of core thermal power operation with the following basis.

Since the pressure drop in the core bypass region, that region between fuel bundles, is 4.56 psi, due to elevation head, the core pressure drop at low power and all flows will always be greater than 4.56 psi.

Flow tests conducted outside a reactor core on a restricting area equal to that of a fuel bundle show that at 28×10^3 lbs/hr flow a pressure drop of 3.5 psi exists. Thus, fuel bundle flow with a 4.56 psi driving head will be greater than 28×10^3 lbs/hr. independent of total core flow and bundle power for the range of bundle powers of concern.

Full scale test data at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, the 3.35 MWt bundle power corresponds to a core thermal power of more than 50%. Therefore, a core thermal power limit of 25% for reactor pressures below 800 psia or core flow less than 10% is conservative.

Summary - Thermal Hydraulics

Classification - Thermal limits are technical specification and 10 CFR requirements.

Purpose - The purpose of having such limits is to ensure integrity of the fuel clad.

Terms - Plastic Strain, LHGR, APLHGR, MAPLHGR, TPF, CP, CPR, MCH

PARAMETER	BWR/3	BWR/4	BWR/5	BWR/6
1. RATED POWER (MWT)	2527	3203	3323	3570
2. NUMBER OF FUEL ASSEMBLIES	724	764	704	732
3. NUMBER OF CONTROL RODS	177	185	185	177
4. VESSEL SIZE	251	254	251	238
5. RATED CORE FLOW (LBS/HR)	98X10 ⁶	102.5X10 ⁶	108.5X10 ⁶	105X10 ⁶
6. RATED STEAM/FEEDWATER FLOW (LBS/HR)	9.8X10 ⁶	13.4X10 ⁶	14.3X10 ⁶	15.4X10 ⁶
7. AVERAGE POWER DENSITY (KW/LITER)	36.6	50.7	50.7	56.0
8. AVERAGE HEAT FLUX (BTU/HR-FT ²) OR (WATT/CM ²)	132,314 OR 41.7	164,410 OR 51.8	164,410 OR 51.8	159,580 OR 50.3
9. PEAK HEAT FLUX (BTU/HR-FT ²)	396,942	428,124	428,124	354,267
10. AVE. THERMAL POWER/BUNDLE (MWT)	3.49	4.31	4.35	4.89
11. TOTAL HEAT TRANSFER AREA (FT ²)	62,928	66,214	66,214	73,409
12. AVERAGE LHGR (KW/FT)	5.72	7.05	7.10	6.04
13. PEAK LHGR (KW/FT)	17.5	18.5	18.5	13.4
14. NUMBER OF BASIC ORIFICE ZONES	2	2	2	2
15. DESIGN OR MAX. TOTAL PEAKING FACTOR (FUEL TYPE DEPENDENT)	3.0	2.6	2.6	2.22
16. HEAT TRANSFER AREA/FUEL ROD (FT ²)	1,768	1,768	1,768	1,592

TABLE 1.2-1 COMPARISON OF BWR CORE PARAMETERS

TABLE 1.2-2

MAXIMUM LINEAR HEAT GENERATION RATE

The maximum LHGR in the core shall not exceed the value determined by the following relation:

$$\text{LHGF}_m = \text{LHGR}_L \left\{ 1.0 - \frac{\Delta P}{P}_{\max} \cdot \frac{L}{L_T} \right\}$$

LHGR_m = Maximum local linear heat generation rate.

LHGR_L = Local linear heat generation rate, license limit

L = Axial position from bottom core

L_T = Total core length

$(\Delta P/P)_{\max}$ = Maximum value of power spiking penalty

FACTORS THAT INFLUENCE CRITICAL QUALITY

MASS FLOW RATE
SYSTEM PRESSURE
BOILING LENGTH
HEATED LENGTH
THERMAL DIAMETER
LOCAL PEAKING FACTOR—R

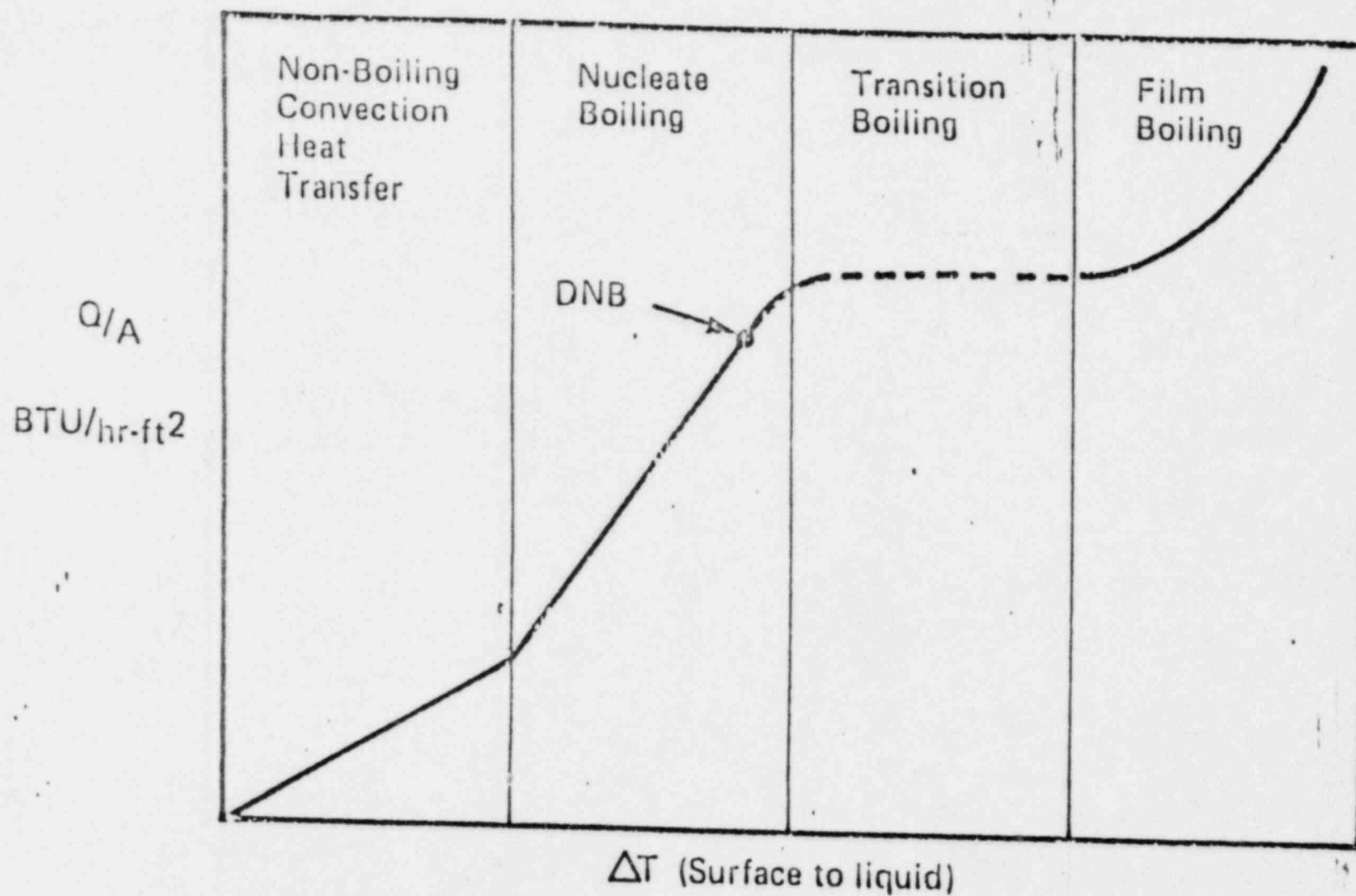
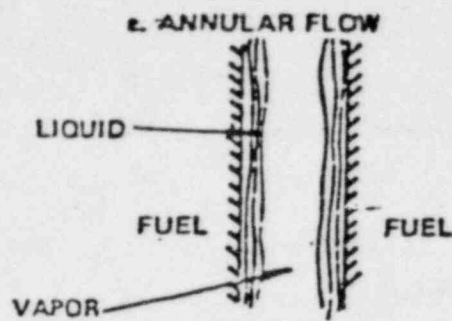
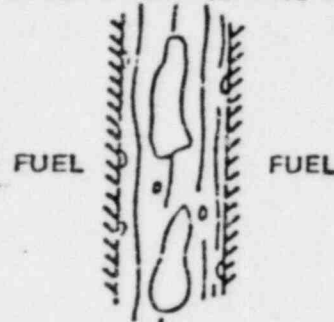


Figure 1.2-1 Boiling on the Surface of Cladding



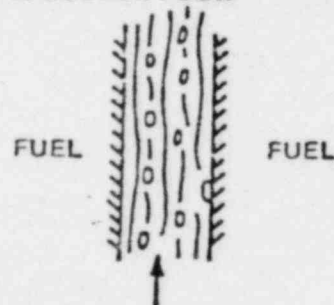
- 1) VAPOR FORMS A CONTINUOUS PHASE IN THE SPACE BETWEEN FUEL ELEMENTS (SATURATED VAPOR)
- 2) SLOWER MOVING LIQUID TRAVELS ALONG THE FUEL ELEMENT SURFACE
- 3) EXPERIENCED NEAR CHANNEL EXIT IN HIGH POWER CHANNELS

d. SLUG FLOW (LARGE VOIDS)



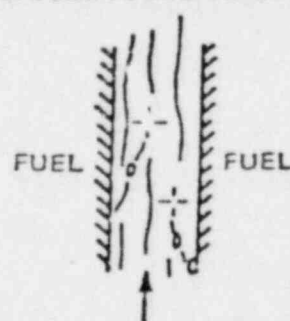
- 1) BUBBLES ARE FORMING ON WALL AND COALESCING TO FORM VAPOR "SLUGS" IN MIDCHANNEL
- 2) SIGNIFICANT VOID FRACTION
- 3) IN HIGH POWER CHANNELS SLUG FLOW BEGINS BELOW CORE MIDPLANE

c. BUBBLE FLOW



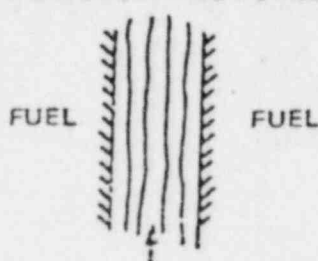
- 1) BUBBLES FORM ON THE WALL BUT DO NOT COLLAPSE IN COOLANT STREAM, I.E., "BULK BOILING"
- 2) BULK COOLANT TEMPERATURE IS AT SATURATION TEMPERATURE
- 3) BUBBLES ARE NOT YET COALESCING, LOW QUALITY

b. SUBCOOLED BOILING



- 1) BUBBLES FORM ON THE WALL BUT COLLAPSE IN THE BULK COOLANT STREAM
- 2) NO NET VAPOR GENERATION
- 3) A VERY SMALL VOID FRACTION
- 4) OCCURS ABOVE THE FORCED CONVECTION REGION

a. SINGLE PHASE FORCED CONVECTION



- 1) NO BUBBLES ARE FORMED, THE COOLANT REMAINS IN A SINGLE PHASE (LIQUID)
- 2) OCCURS AT VERY BOTTOM OF COOLANT CHANNEL

FIGURE 1.2-2 FUEL CHANNEL BOILING CONDITIONS

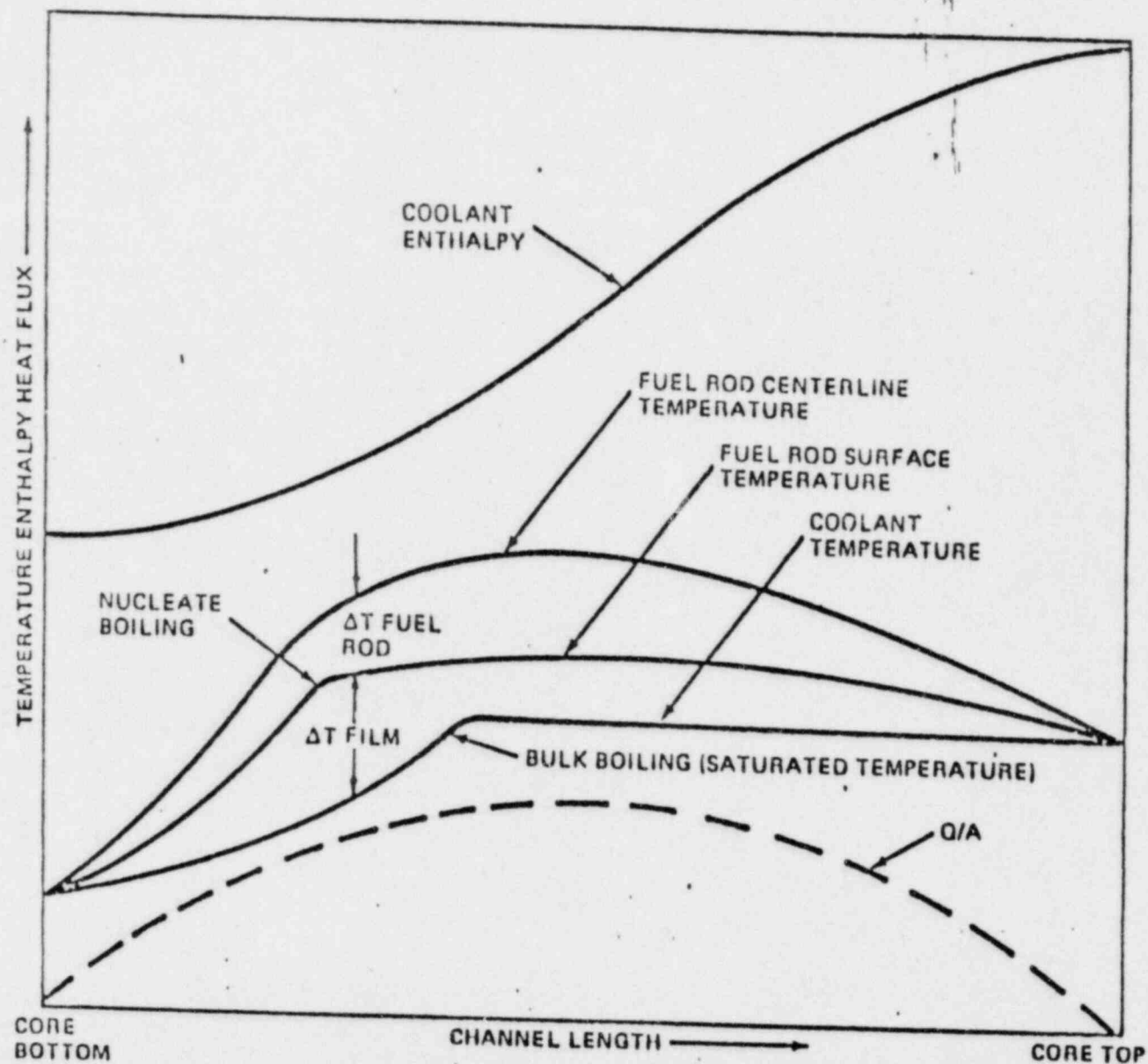


FIGURE 1.2-3 PLOT OF COOLANT AND FUEL BUNDLE TEMPERATURE vs FLOW PATH LENGTH

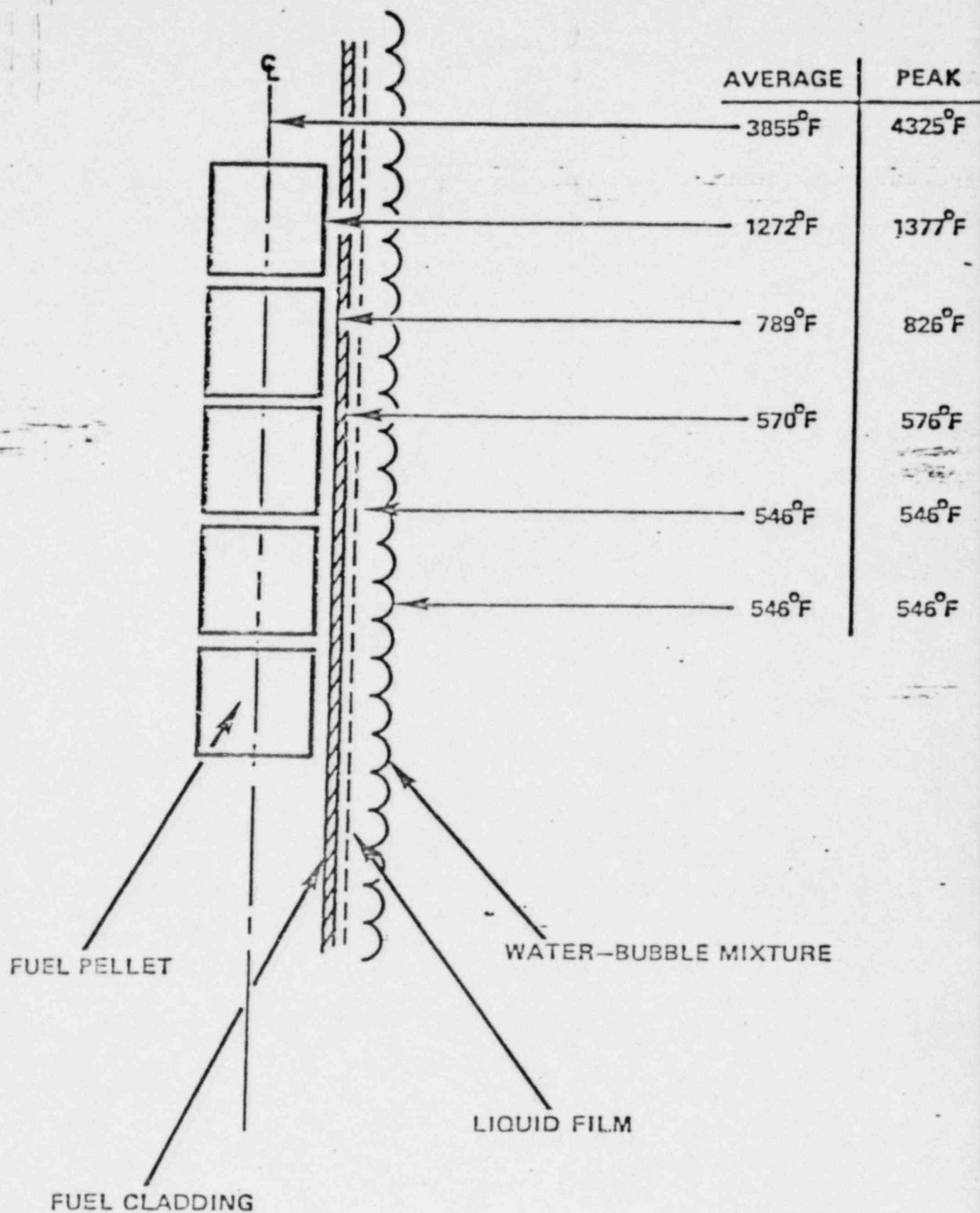


FIGURE 1.2-4 FUEL TEMPERATURE CROSS SECTION

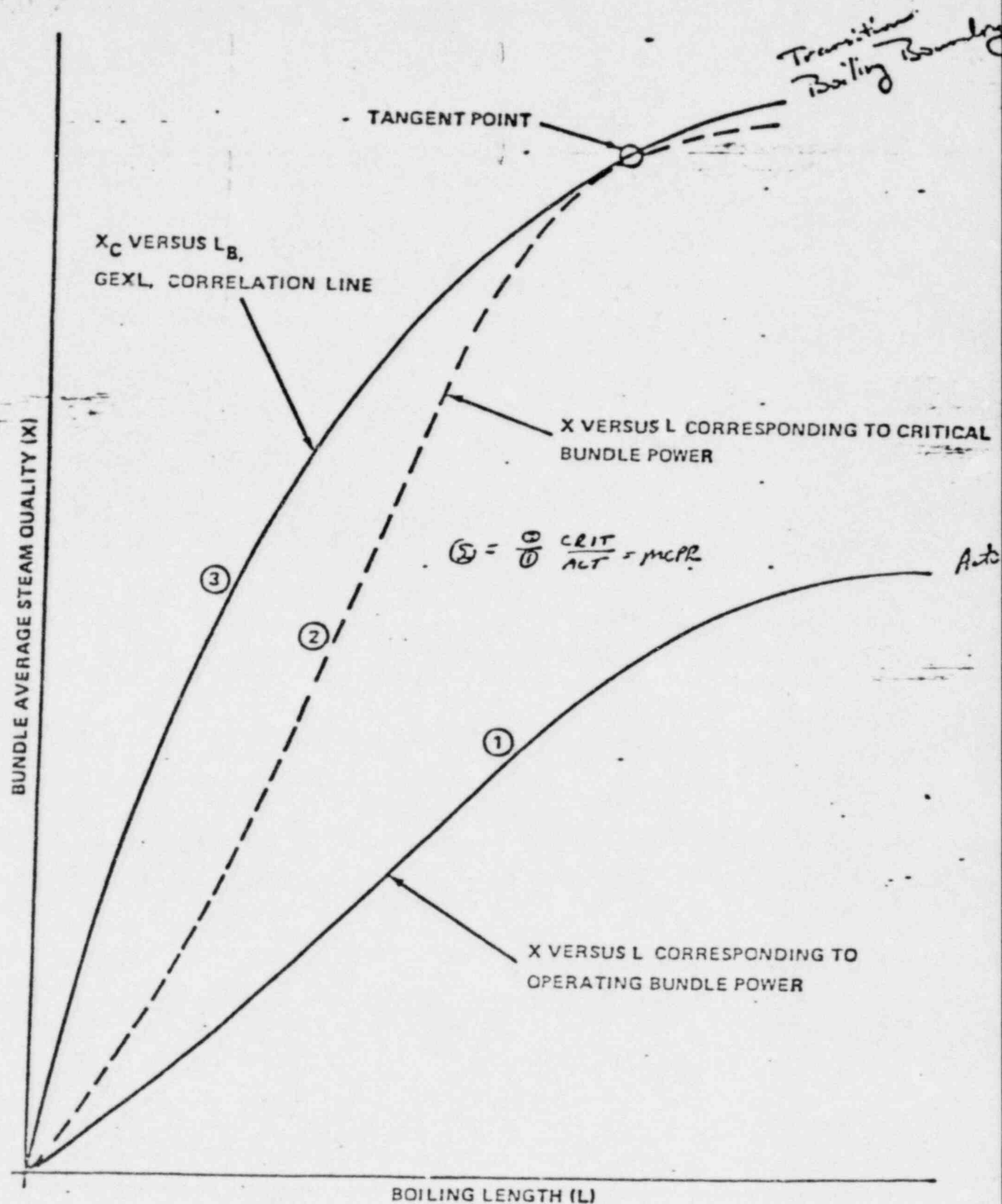


FIGURE 1.2-5 CRITICAL POWER RATIO

Introduction

The laws which govern heat transmission are very important to the engineer in design, construction, testing and operation of BWR systems and equipment. Heat is transferred in three different ways: Conduction, Convection and radiation. There are three different methods and are governed by three different laws. However, they share the common principal that temperature difference must exist for heat to be transmitted and heat is always transmitted in the direction of decreasing temperature.

TYPES OF HEAT TRANSFER

CONDUCTION

- o MOLECULAR INTERACTION - HIGHER ENERGY MOLECULE IMPARTS ENERGY TO LOWER ENERGY MOLECULE BY COLLISION.
- o TRANSFER BY FREE ELECTRONS - THE ABILITY OF SOLIDS TO CONDUCT VARIES DIRECTLY WITH THE CONCENTRATION OF FREE ELECTRONS.

CONVECTION

- o NATURAL CONVECTION - WARMER FLUID CAUSES CIRCULATION BECAUSE OF A DENSITY DIFFERENCE.
- o FORCED CONVECTION - FLUID IS MADE TO FLOW PAST THE SURFACE WHICH IS BEING COOLED.

RADIATION

- o DIRECT ENERGY TRANSFER FROM ONE BODY TO ANOTHER WITHOUT THE USE OF A MEDIUM FOR TRANSPORT.

CONDUCTIVE HEAT TRANSFER

$$\dot{Q} = \frac{KA(T_1 - T_2)}{X}$$

WHERE: \dot{Q} = HEAT TRANSFER RATE (BTU/HR)
 K = THERMAL CONDUCTIVITY (BTU/HR-FT-°F)
 A = AREA (FT²)
 T_1 = HIGHER TEMPERATURE (°F)
 T_2 = LOWER TEMPERATURE (°F)
 X = MATERIAL THICKNESS (FT)

NOTE:

- o A LARGE K RESULTS IN MORE HEAT TRANSFER FOR GIVEN ΔT
- o A LARGE ΔT RESULTS IN MORE HEAT TRANSFER FOR GIVEN K

SUBSTANCE	THERMAL CONDUCTIVITY, K (BTU/HR-FT-°F)
AIR	.0234
HELIUM	→ .1175 > AIR
HYDROGEN	.1580
STEAM	.0228
SATURATED WATER	.349
CARBON STEEL	30.0
STAINLESS STEEL	9.0
URANIUM	20.4
URANIUM DIOXIDE	2.9
ZIRACLOY-2	7.2
COPPER	228.0

CONDUCTION

The first method of heat transfer to be discussed is conduction. This method is characterized by a transfer of heat energy due to the transfer of kinetic energy from one molecule of the fluid to another by collision. For example, if a molecule is subjected to an increase in heat energy, the kinetic energy is increased. Some of this kinetic energy will be transferred to other molecules upon collision. Individual molecules are not heated but a fluid (or material) consisting of billions of molecules is subjected to some heat energy source. The molecules in contact with the source will have their kinetic energy increased and due to molecular motion, will collide with other molecules of the fluid, increasing their kinetic energy. Those molecules far from the heat source will experience less kinetic energy increases. Therefore, as the distance from the heat source varies the amount of increase in kinetic energy also varies. Since temperature is a measure of kinetic energy, it can be stated that molecules farther from the source will have a smaller temperature increase. Referring to Figure 3-1, it is seen that a temperature gradient across the material will exist. Factors affecting the transfer of heat energy from the source are 1) the thickness of the wall separating the source from the material (the thicker the wall, the less the heat transfer), 2) the total area of the wall over which heat transfer occurs (the greater the area, the more heat transfer that occurs) and 3) the temperature gradient (the greater the temperature gradient the greater the heat transfer rate). The above statements can be represented by

$$\text{heat transfer rate} \propto \frac{\text{area} \times \text{temperature gradient}}{\text{wall thickness}} \quad (3-1)$$

It is desirable to have an equation rather than a statement of proportionality. The term thermal conductivity is now introduced. The fluids ability to conduct heat is called thermal conductivity and has the units of BTU/hr.-ft.-°F. The above relation can be restated as:

$$\text{heat transfer rate} = \frac{\text{thermal conductivity} \times \text{area} \times \text{temperature gradient}}{\text{wall thickness}} \quad (3-2)$$

Symbolically, Equation 3-2 can be shown as

$$\dot{Q} = \frac{K \times A \times (T_1 - T_2)}{X} \quad (3-3)$$

where

\dot{Q} = heat transfer rate (BTU/hr.)

K = thermal conductivity (BTU/hr.-ft.-°F)

A = area (ft.²)

T₁ = higher temperature (°F)

T₂ = lower temperature (°F)

X = material thickness (ft.)

Heat transfer rate is normally dealt with on a per unit area basis.

Thus, heat transfer rate becomes heat transfer rate per square foot and is called heat flux (symbol \dot{q}). The equation for heat flux is

$$\dot{q} = \frac{\dot{Q}}{A} = \frac{K \times A \times (T_1 - T_2)}{A \times X} = \frac{K \times (T_1 - T_2)}{X} \quad (3-4)$$

Heat flux has the units BTU/hr.-ft.² or, if appropriately converted, watts/cm².

Conduction is the means by which the heat energy moves out from the fuel to the water. Heat energy is generated as a result of the fissioning of Uranium-235 into two smaller atoms (fission fragments). The major portion of the energy from fission occurs due to the kinetic energy of the fission fragments. Through conduction the kinetic energy is transmitted to other atoms within the UO₂ fuel rod. The atoms of UO₂ on the surface of the fuel pins transmit their increased kinetic energy to the He atoms which in turn transmit some of their kinetic energy to the Zr-2 cladding. The heat is finally conducted through the cladding to the water surrounding the fuel rod. Under normal conditions, the surface of the fuel rod has a layer of water a few molecules thick at zero velocity, through which heat is transferred by conduction. Further heat energy transmission is by convection.

CONVECTIVE HEAT TRANSFER

THE AMOUNT OF HEAT REMOVED FROM A SURFACE BY CONVECTION (\dot{q}) DEPENDS ON THE SURFACE CONDITIONS AND THE FLUID AND FLOW PROPERTIES (h) AND THE TEMPERATURE DIFFERENCE BETWEEN THE SURFACE AND THE FLUID (ΔT).

$$\dot{q} = h \Delta T$$

\dot{q} = HEAT FLUX FROM SURFACE, Btu/hr-ft²

h = CONVECTIVE HEAT TRANSFER COEFFICIENT, Btu/hr-ft²°F

ΔT = TEMPERATURE DIFFERENCE BETWEEN THE SURFACE AND THE FLUID, °F

MECHANISM	h , Btu/hr ft ² °F
FREE CONVECTION, AIR	1-10
FORCED CONVECTION, AIR	5-50
FORCED CONVECTION, WATER	50-3000
BOILING WATER	500-5000
FORCED CONVECTION, BOILING WATER	5000-8000

CONVECTION

Heat from a source reaches a greater number of molecules in a fluid if the fluid is circulating. This mechanism of heat transfer is known as CONVECTION. The fluid in a BWR is forced through the fuel region by recirculation pumps. The moving fluid accepts heat energy from the fuel rods via the stationary layer of the fluid surrounding the rod and transports the heat to the coolant by convection. Thus, as the fluid travels the length of the fuel rod, it continually accepts more heat energy (temperature and enthalpy increase).

For purposes of calculating the heat transfer rate for convection, an equation similar to Equation 3-2 is used. (See Figure 3-2 for illustration of the terms).

$$\dot{Q} = \frac{K_f A (T_s - T_f)}{x_f} \quad (3-5)$$

where

K_f = thermal conductivity of the stagnant film (BTU/hr.-ft.-°F)

T_s = surface temperature ($^{\circ}\text{F}$)

T_f = fluid temperature ($^{\circ}\text{F}$)

x_f = stagnant film thickness (ft.)

Also, heat flux (\dot{Q}/A) has a representation similar to Equation (3-3):

$$\dot{q} = \frac{\dot{Q}}{A} = \frac{k_f(T_s - T_f)}{x_f} \quad (3-6)$$

The value for x_f , the thickness of the stagnant film, is difficult to determine. It is not a fixed quantity, such as the thickness of a fuel rod. x_f will vary dependent on the rate of fluid flow, the fluid viscosity, the heat flux, the type of surface and the state of the fluid (liquid or vapor). Consider the ratio of k_f to x_f and call it the film heat transfer coefficient, h_f which will also vary due to system conditions. In equation form,

$$\frac{k_f}{x_f} = h_f \quad (3-7)$$

Equation 2-6 then becomes

$$\dot{q} = \dot{Q}/A = h_f (T_s - T_f) \quad (3-8)$$

RADIATIVE HEAT TRANSFER

THE AMOUNT OF HEAT TRANSFERRED BY RADIATION FROM ONE BODY TO ANOTHER (\dot{q}) DEPENDS ON THE MEDIUM BETWEEN THE TWO BODIES, THE SURFACE CONDITIONS OF THE BODIES (F) AND THE TEMPERATURE DIFFERENCE BETWEEN THE BODIES ($T_1 - T_2$).

$$\dot{q} = F\sigma (T_1^4 - T_2^4)$$

\dot{q} = HEAT FLUX FROM BODY 1 TO BODY 2, Btu/hr·ft²

F = SHAPE FACTOR ACCOUNTING FOR SURFACE CONDITIONS

σ = STEFAN-BOLTZMAN CONSTANT

$$= 0.1714 \times 10^{-8} \text{ Btu/hr} \cdot \text{ft}^2 \cdot ^\circ\text{R}^4$$

$T_{1,2}$ = TEMPERATURE, $^\circ\text{R}$

RADIATION

Heat transfer by radiation differs from heat flow by conduction in that the medium by which heat transmission takes place, does not become heated. For example, the earth receives radiant energy from the sun whereas the space above the earth does not become hot. When in winter, cars are parked close to a building, they receive energy from the building and remain warmer than they would be if parked further from structures.

Radiation heat transfer takes place in the following stages:

- 1) Conversion of thermal energy of heat source into an electromagnetic waves
- 2) Passing the electromagnetic waves through space
- 3) Reconversion of the electromagnetic waves into thermal energy at the cooler medium.

Finally this heat transfer is governed by the Stefan Boltzman equation namely

$$\dot{q} = F\sigma(T_1 - T_2)^4 \quad 3.6$$

where

\dot{q} = Heat Flux from body 1 at Temp T_1 to body 2 at Temp T_2 in
BTU/hr-ft²

F = Shape factor accounting for surface condition

σ = Stefan Boltzman Constant = $.1714 \times 10^{-8}$ BTU/hr.-ft²-R⁴)

This equation reveals that due to such a small Stefan Boltzman constant, for the heat transfer to be appreciable, ΔT , must be rather large.

BOILING HEAT TRANSFER

Boiling is the evaporation of a liquid to vapor occurring within the body of the liquid by the mechanism of bubble formation. The very term "Boiling Water Reactor" implies great dependency is placed on the boiling water as a primary mode of heat transfer. To become conversant in the terminology, it is convenient to use a simple example to demonstrate the fundamental principles of boiling heat transfer. The example selected, as shown in Figure 3-3, is a pan of water on an electric stove.

The process consists of a pan of water at room temperature and atmospheric pressure, resting on an electric heating coil. The coil and pan have a finite contact area "A". The amount of energy given off by the coil as heat can be varied by changing the position of the heat control knob (varying the electrical current). Assuming that the pan and coil are in good contact, heat will be transferred from the coil to the pan and then to the water. The heat transferred " \dot{Q} " divided by the contact area "A" then yields \dot{Q}/A or "heat flux" in BTU/hr.-ft.².

Turning on the electricity will supply a small amount of heat energy or heat flux. As the temperature of the pan increases, heat will be transferred to the water (via a ΔT) by convection heat transfer. As the water becomes hot at the pan surface, it becomes less dense and rises, being displaced by more dense, cooler water flowing in from the sides. This particular mode will be called Region 1 or "CONVECTION HEAT TRANSFER". This region is characterized by the phenomenon just described.

Next, increase the heat flux by turning up the heat control knob. Increasing

the heat flux increases the pan temperature (recall $\dot{Q}/A = UA(\Delta T)$). When the pan temperature is about 10°F higher than the bulk water saturation temperature (212°F @ atmospheric pressure), bubbles will begin to form on the bottom of the pan (Figure 3-4). As the bubbles get larger, their buoyance will eventually cause them to be detached from the surface of the pan. The bubbles will drift off into the bulk liquid. Here the bubbles will collapse and give up their energy to the surrounding cooler liquid. This is an excellent means of heat transfer from the pan to the water. The bubbles serve to "stir up" or "agitate" the stagnant fluid film, thus improving the thermal conductivity of the film. Additionally, each bubble carries off more energy than is possible by normal convection. Recall that as a fluid changes phase, its enthalpy increases; therefore each bubble, being vapor, is something akin to being a "little bundle of energy (enthalpy)" and the heat removal rate is greatly enhanced. The phenomenon just described is called NUCLEATE BOILING because bubbles are forming at sites of nucleation on the pan surface. NUCLEATE BOILING is extremely important in reactor core heat transfer because a great deal of heat energy can be transferred without extremely high surface temperatures.

The next area of discussion will require holding the heat flux right where it is for a while and allow the bulk temperature of the water to rise to saturation temperature. When the bulk liquid temperature reaches saturation temperature, the bubbles released from the pan surface will not collapse in the liquid

but travel all the way to the surface. This particular phenomenon is referred to as "BULK" or saturated boiling (Figure 3-5). NOTE: Nucleate boiling is still occurring because bubbles are forming at the pan surface! The fact that the bubbles are not collapsing is irrelevant to the nucleate boiling phenomenon.

BULK BOILING and SUBCOOLED BOILING are terms used to describe what happens to the bubbles AFTER the bubbles have left the surface. In both cases, nucleate boiling describes only the phenomenon occurring at the pan surface. In all of the vast nuclear power field there is no greater confusion existing than that which occurs when operators try to describe the terms, nucleate, subcooled and bulk boiling.

Return to the heat control knob and turn up the heat flux more and more, other phenomena will occur. As the heat flux is increased, more and more bubbles form until many of the spaces on the pan bottom are covered with bubbles. Although the bubbles are moving away from the surface, it appears vapor is on the surface for a longer time. The higher the heat flux, the more the heat transfer mechanism deteriorates until a steam blanket is formed. The onset of this phenomenon is called ONSET OF TRANSITION BOILING (abbreviated OTB) (Figure 3-6). This signals the end of nucleate boiling and the onset of heat transfer deterioration. As the heat flux continues to increase, bubbles continue to form at a faster and faster rate. Finally, in one portion of the pan, bubbles will form so fast that the liquid cannot displace the

bubbles and they combine to form a "bubble" or "vapor" blanketed region on the pan surface. This is a highly unstable condition. The bubbles may move away from the surface, allowing liquid to again wet the surface; or the bubble blanket may spread until the surface is vapor blanketed. This mode of boiling heat transfer is called TRANSITION or PARTIAL FILM BOILING and is an extremely unstable form of heat transfer (i.e., oscillating between vapor blanketing and nucleate boiling). Assume it is able to maintain the unstable partial film boiling mode for a moment. Now increase the heat flux just a little and see what happens (Figure 3-7).

With the first "tweak" of the heat control knob, the partial vapor blanket would spread over the entire pan surface. What is found is a thin layer of steam everywhere on the pan surface. As shown in a previous table of thermal conductivities, steam is an excellent insulator; hence, the pan surface now presents an extremely large resistance to heat transfer. Assume again the electric heater can maintain the desired heat flux, it is known from the equation:

$$Q/A = U(T_{\text{surf}} - T_{\text{liq}})$$

that if Q/A is held constant and U decreased drastically and T_{liq} is constant, then T_{surf} must increase rather substantially (on the order of 1,000°F).

If the pan surface was a fuel rod in the reactor core and full film boiling

were allowed to take place, the metal of the fuel rod might have melted, been seriously deformed, or cracked. Radioactivity would then leak from the damaged rod into the coolant with rather severe ramifications. OTB, transition boiling, and full film boiling are never permitted in an operating reactor under ordinary or transient conditions.

Plotting the results of the experimentation conducted thus far (Figure 3-8) yields a graphical representation of heat flux (\dot{Q}/A) versus ΔT ($T_{\text{surface}} - T_{\text{liquid}}$) with the Regions of heat transfer and the point of DNB annotated. The regions of heat transfer are described by the method of heat transfer as follows:

- Region I - Characterized by conduction and convection heat transfer
- No boiling
- Region II - Characterized by nucleate boiling
- Region III - Transition region - characterized by partial film boiling - unstable region
- Region IV - Film boiling region - continuous vapor blanket - radiation heat transfer

One further clarification is necessary concerning nucleate boiling, i.e., it is not necessary for the bulk liquid to be at saturation temperature for nucleate boiling to occur. Thus nucleate boiling as an effective heat transfer mechanism can be utilized in the BWR and yet maintain a substantial

margin to DNB by operating in the left hand portion of Region II in Figure 3-8. For a better idea of what is going on in the core, it is best to look at an average fuel bundle. Refer to Figure 3-9. Coolant enters the bottom, flows up around the fuel rods, and absorbs energy from the heat transferred by the fuel rods to the coolant. Notice the plot termed relative heat flux ("HEAT FLUX PROFILE"). The heat flux produced from fission in the core will assume a shape as shown. The highest heat flux (greatest fission rate) occurs in the interior of the core. The dotted line specifying "AVERAGE \dot{Q}/A " corresponds to the average heat flux for the bundle. An explanation of AVERAGE HEAT FLUX follows. The core may have a thermal output of 2420 MW. If there were 724 bundles in the core, the AVERAGE FUEL BUNDLE POWER would be $1/724$ of the core power if every fuel bundle was producing the same amount of energy. To obtain $1/724$ core power, each fuel bundle must operate at some average heat flux (\dot{Q}/A). As previously stated, the highest heat flux will be in the core interior. Therefore some portions of the fuel bundle have a higher than average heat flux and others a lower than average heat flux. The nuclear core heat flux profile also is useful to plot curves of coolant and fuel bundle temperatures versus flow path length up the core (Figure 3-10). The coolant temperature experiences a continuous rise as heat is added until the temperature reaches saturation temperature and begins to bulk boil. Coolant temperature remains constant from the point to the core exit. Notice the sharp temperature rise at the higher values of heat flux indicating energy is being absorbed at a high rate. The coolant temperature profile is not descriptive of coolant energy increase because the coolant is changing phase. A better description is obtained by plotting coolant enthalpy change. Notice again the steep rise in coolant enthalpy when the heat flux is maximum.

Examine the fuel rod surface temperature profile on Figure 3-10. The fuel rod surface temperature rises continuously and levels off at a rather constant value above the coolant temperature. The initial rise is caused by the ΔT across the film required to accommodate the heat flux, $\dot{Q}/A = h_{\text{film}} (T_{\text{surf}} - T_{\text{liq}})$. The fuel rod temperature levels off due to the beginning of nucleate boiling. Nucleate boiling is an excellent heat transfer mode and even though the heat flux is getting greater, the ΔT across the boiling film remains relatively constant.

Next, follow the fuel rod centerline temperature up the fuel bundle. The centerline temperature is above the fuel surface temperature. The amount the centerline temperature is greater than surface temperature will depend directly on the heat flux, $\dot{Q}/A = k_{\text{fuel}} (T_f - T_{\text{surf}})$. Note the beneficial effects of nucleate boiling on centerline temperature. As nucleate boiling is occurring on the fuel rod surface, the fuel rod surface temperature is only slightly greater than liquid temperature. This automatically keeps the fuel centerline temperature at a lower value than if convection were the mode of heat transfer from surface to liquid.

For a better perspective of the different modes of heat transfer, consider a single fuel rod in a bundle (Figure 3-11). The diagram graphically illustrates the fuel surface and centerline temperature profiles under three different conditions: 1) nucleate boiling with a low heat flux; 2) nucleate boiling with a heat flux four times that in Profile (1); and 3) film boiling with an even greater heat flux. The heat flux in the individual rod is a function of reactor power. The power for Profile (1) is less than Profile (2) since (1) has a lower heat flux. Nucleate boiling is the primary mode of heat transfer

at the fuel rod surfaces in both case (1) and (2). Notice that even with an increase by a factor of 4 in the heat flux, the rod surface temperature is only slightly increased.

Looking at Profile (3) where the heat flux is of sufficient magnitude to cause film boiling, the heat transfer mode at the rod is film boiling. The fuel rod surface is insulated by a steam blanket. Notice the temperature gradient between the fuel centerline and the rod surface is approximately 3000°F. Also note the significant increase in fuel rod surface temperature as compared to Profiles (1) and (2). It is for this reason that the reactor power is limited to below that power level which produces film boiling. Once the heat flux (power level) is sufficient to cause film boiling, all temperatures in the interior of the fuel rod and at rod surface significantly increase. This temperature increase can cause damage to the fuel (melting) and to the fuel rod (buckling or cracking).

TIME CONSTANT

When the reactor is operating at a steady state power level, there is a fixed temperature distribution from the center of the fuel pins to the moderator. The temperature distribution is governed by parameters such as the specific heat, density, volume and heat transfer coefficients of the fuel materials and the heat transfer area.

The response to a rapid change in the fuel temperature is measured in terms of the fuel thermal time constant, Using Figure 3-12.

$$T_2(t) - T_1 = (T_1' - T_1) (1 - e^{-t/\tau})$$

\downarrow
 when $t = \tau$
 \downarrow
 .63

$$T_2' - T_2 = (T_1' - T_1) (1 - e^{-t/\tau})$$

where

T_1 = initial fuel centerline temperature

T_2 = cladding surface temperature

T_1' = final fuel centerline temperature

t = time

τ = time constant

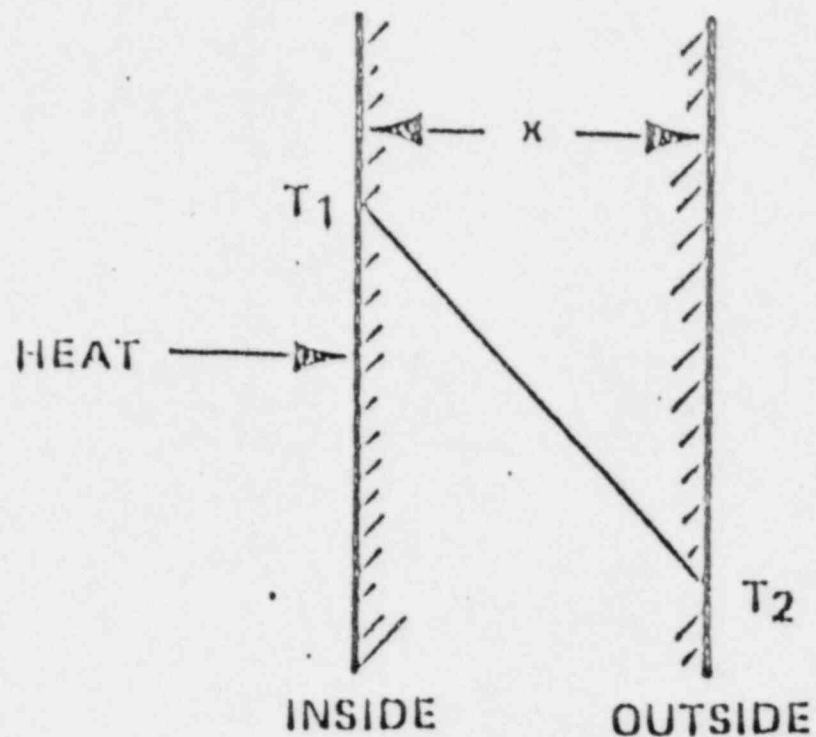
As one can see, a short time constant will result in a rapid response to power changes and a long time constant will result in a slow response.

For 8x8 fuel, the thermal time constant is about 5-6 sec. If the previous equation is used to calculate the cladding surface temperature after one time constant ($t = \tau$), it is seen that after one time constant, the cladding

surface temperature has reached 63.2% of its final temperature. Thus although the initial response to a power change is very rapid, the rate of temperature change is governed by the fuel thermal time constant.

HEAT TRANSFER IN TRANSIENT CONDITIONS

Temperature distributions in the fuel and moderator during transient conditions is a complete function of many variables. The interaction between conductive, convective, radiative and boiling heat transfer as well as pressure changes and local power variations affect the temperature response of the fuel during a transient. If one looks at the average surface heat flux being conducted from the clad in transient conditions, one can see that for slow transients the change in heat flux is slower than the change in power level, indicating a relatively long thermal time constant. For rapid transients involving large changes in power and pressure, the heat flux variation depends on variables such as the boiling rate and the rate of heat transfer to the surface of the clad.



- T_1 = INNER TEMPERATURE
- T_2 = OUTER TEMPERATURE
- x = MATERIAL THICKNESS
- A = AREA OF THE PLANE WALL

Figure 3-1 Section of a Plane Wall

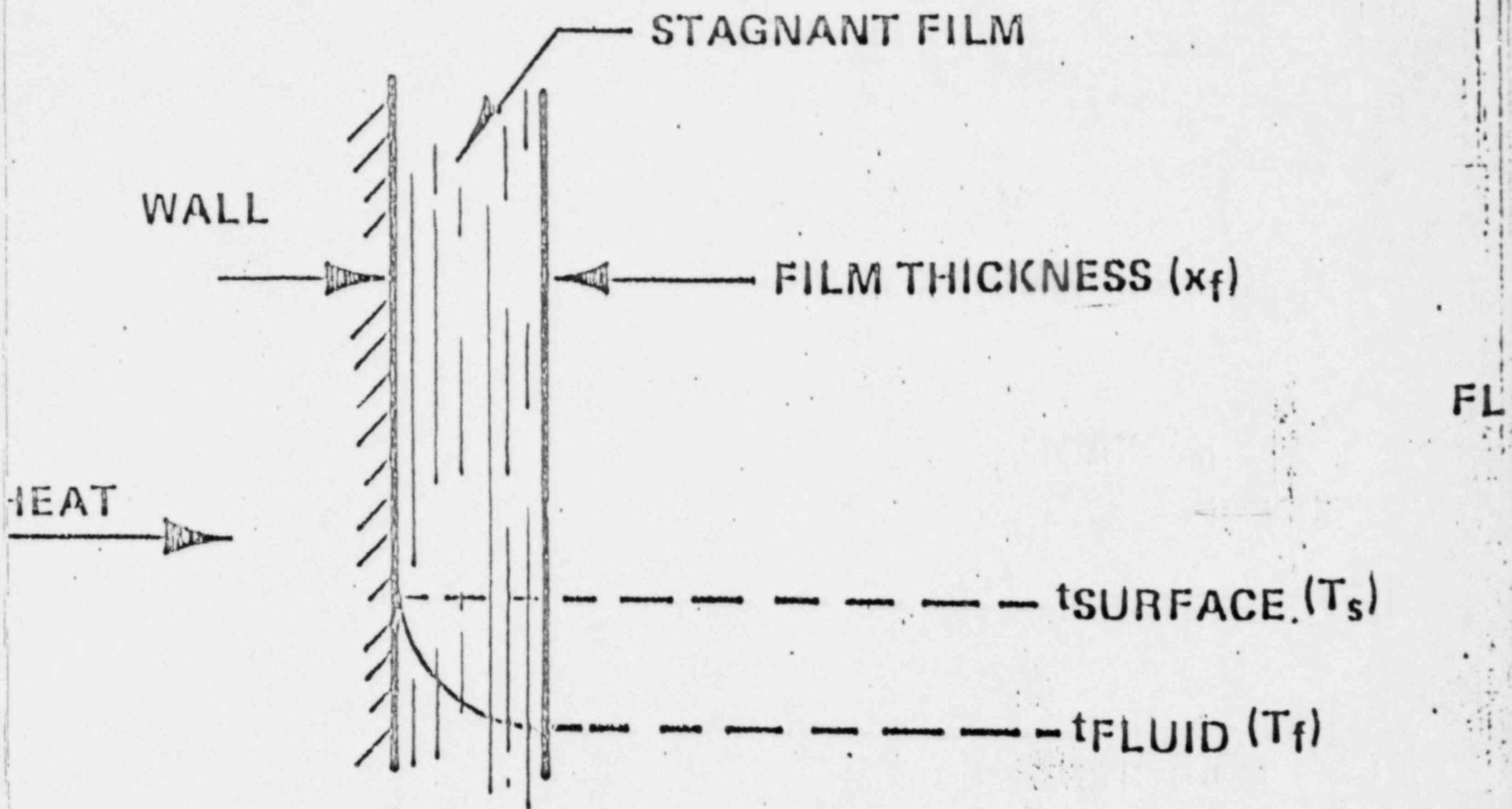


Figure 3-2 Convection Heat Transfer Process

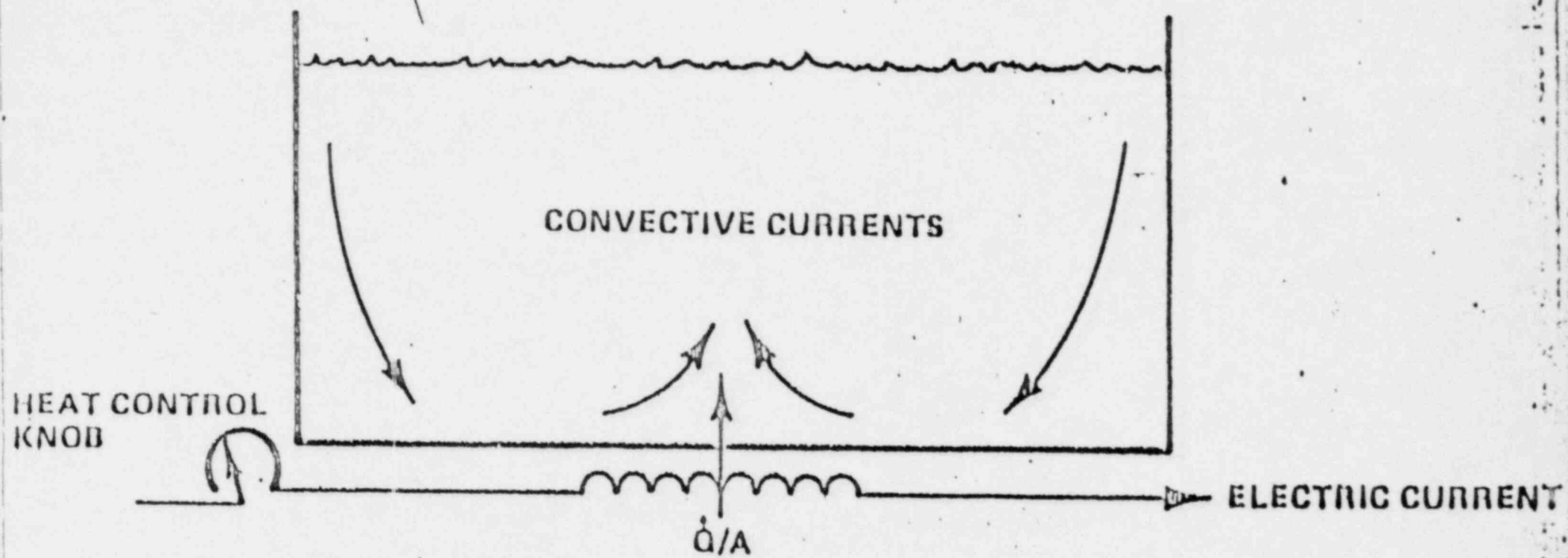


Figure 3-3 Region I: Convection Heat Transfer

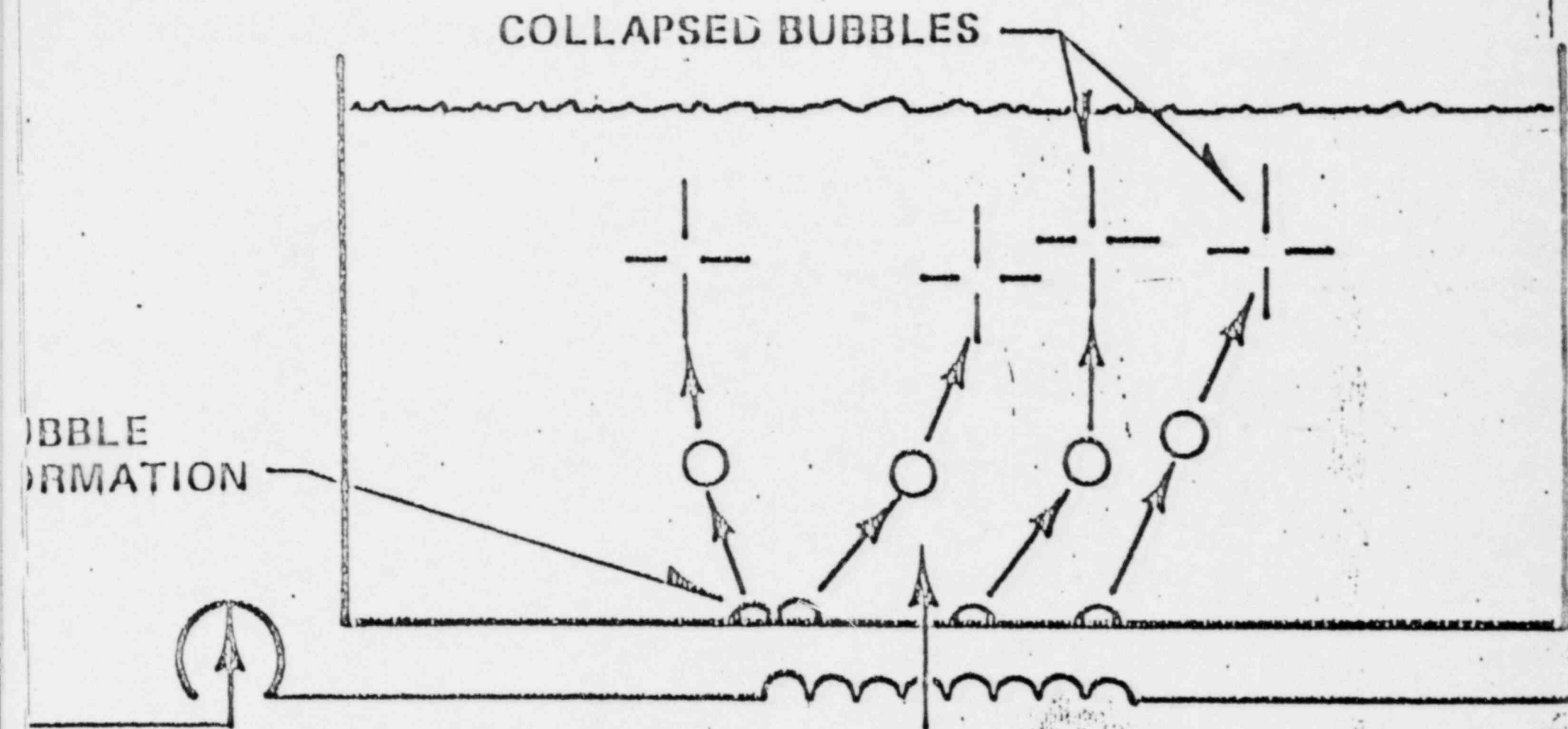


Figure 3-4. Region II: Nucleate Boiling, Subcooled Liquid

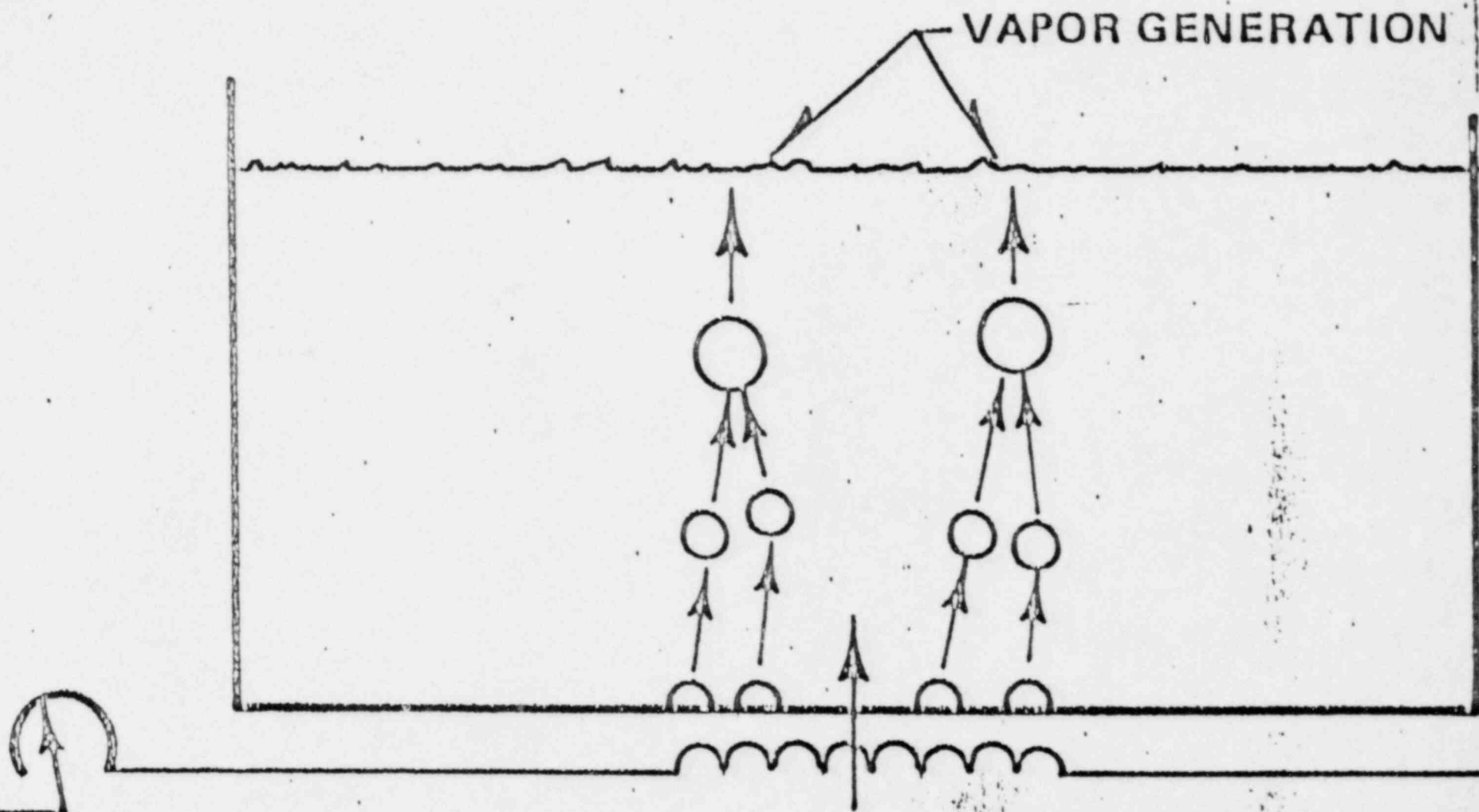


Figure 3-5 Region II: Nucleate Boiling, Bulk or Saturated Boiling

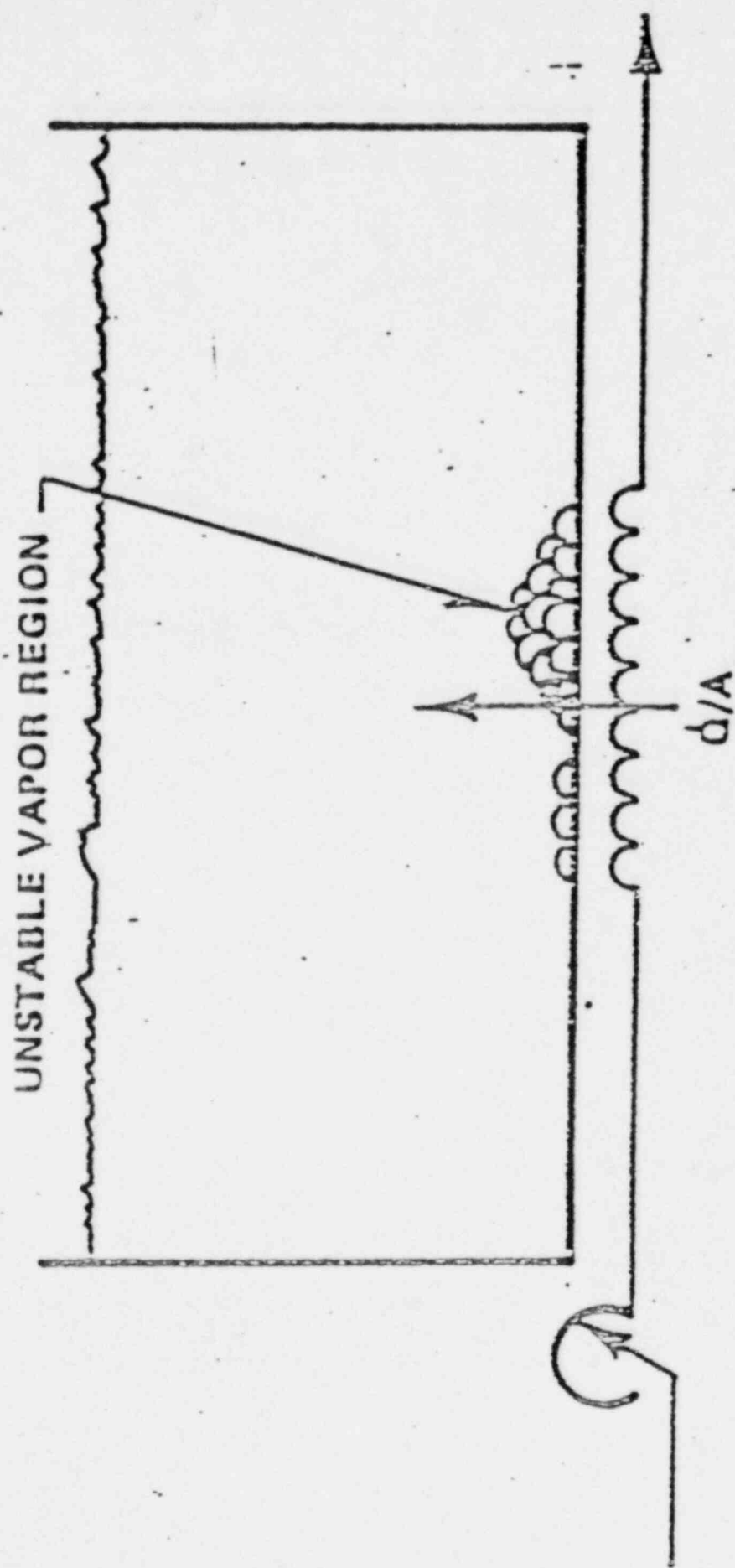


Figure 3-6 Region III: ONSET OF TRANSITION BOILING

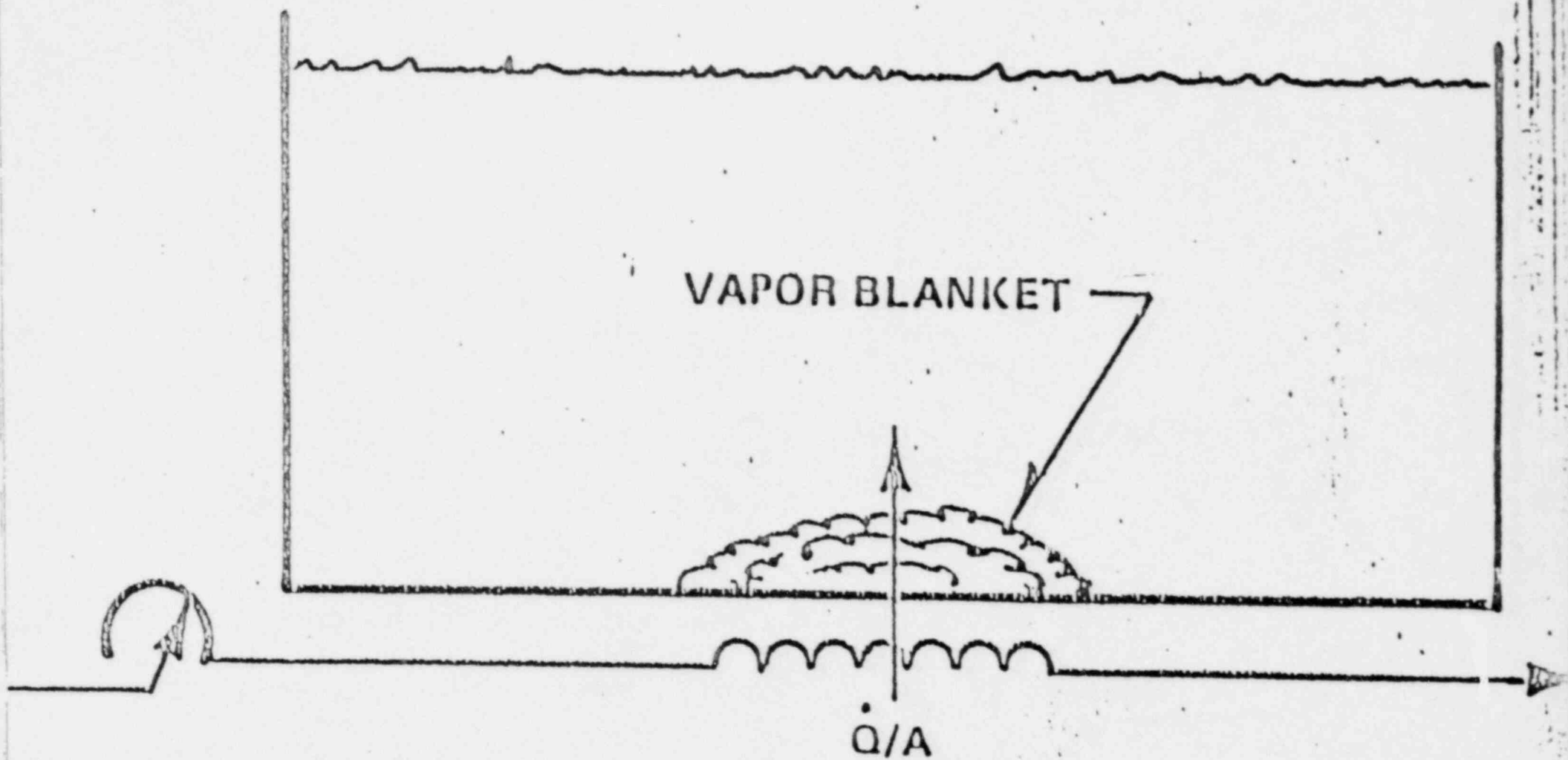


Figure 3-7 Region IV: Full Film Boiling

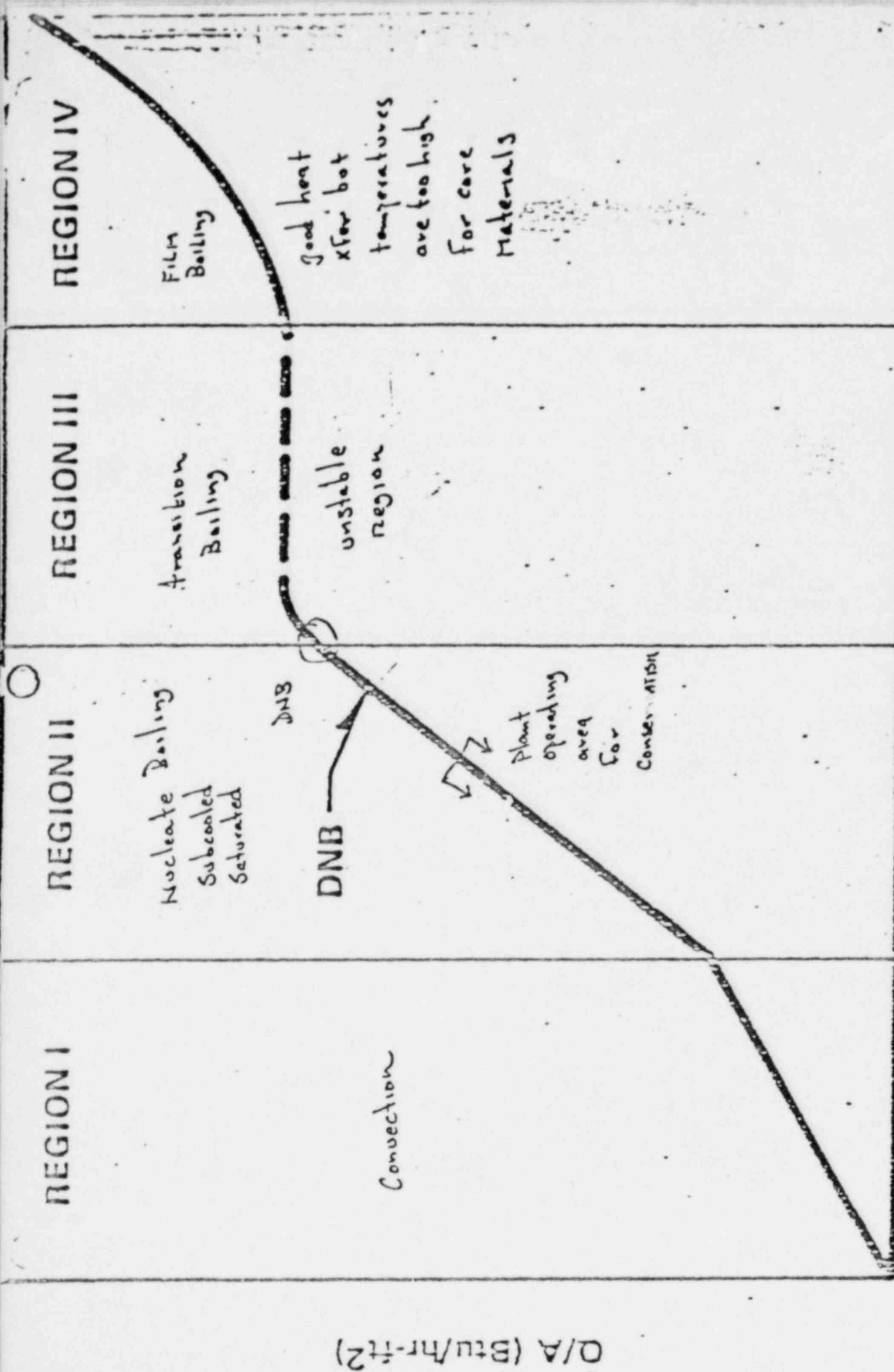


Figure 3-8 Plot of Heat Flux vs. Temperature

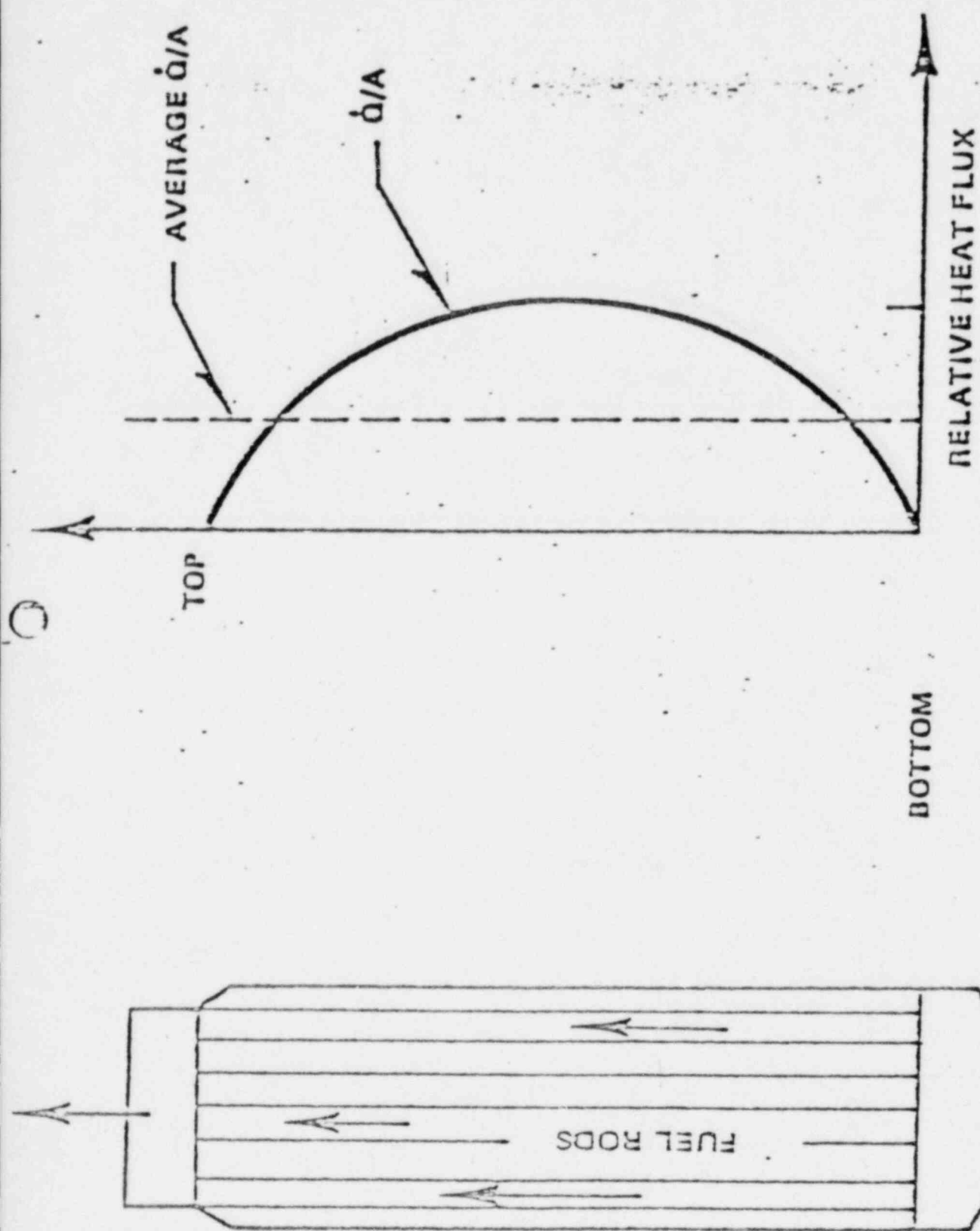
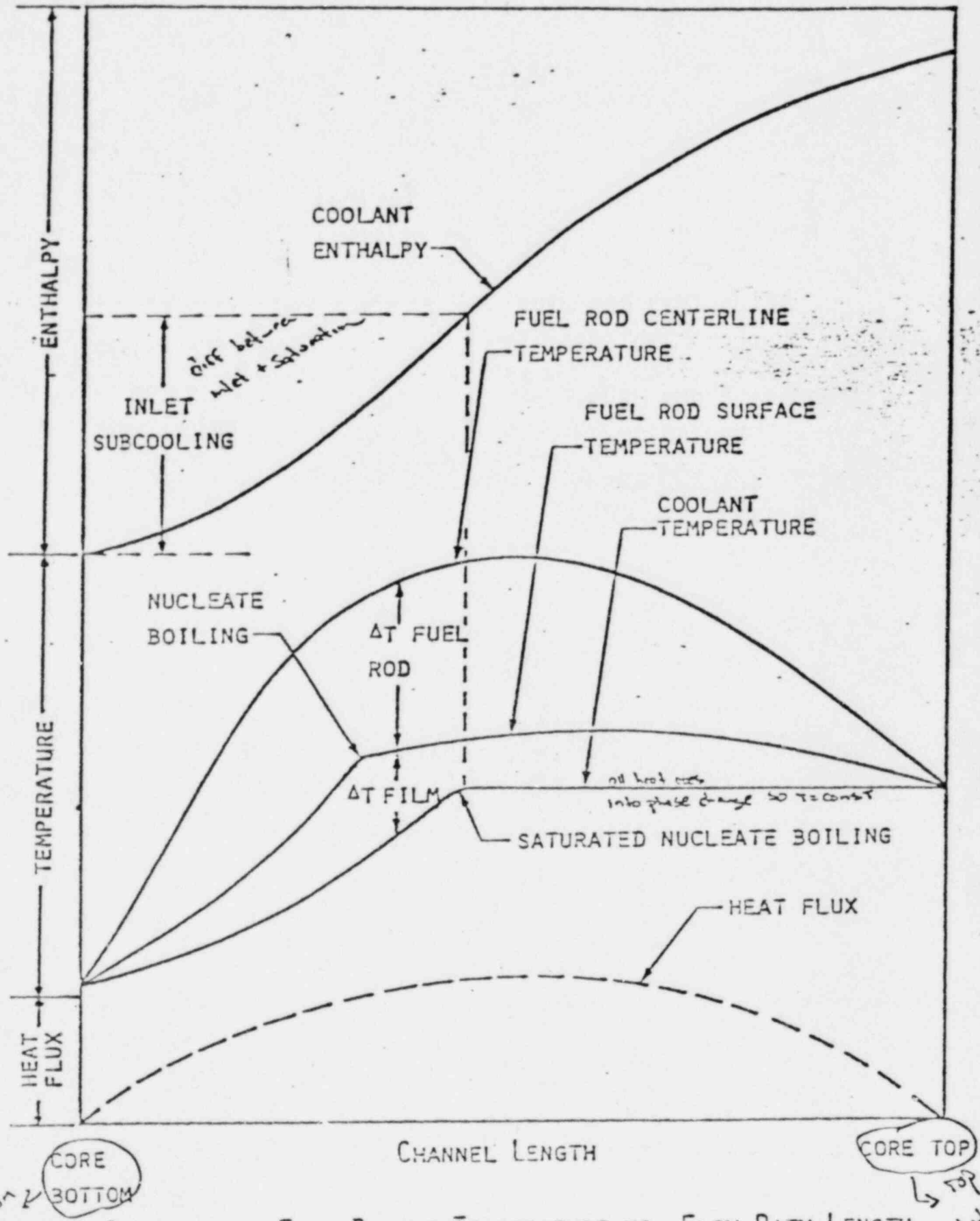


Figure 3-9 Simplified Diagram of an Average Fuel Bundle

center of fuel melting temp



COOLANT AND FUEL BUNDLE TEMPERATURE VS. FLOW PATH LENGTH

FIGURE 3-10

Bottom of active fuel

Top of active fuel

es
phair

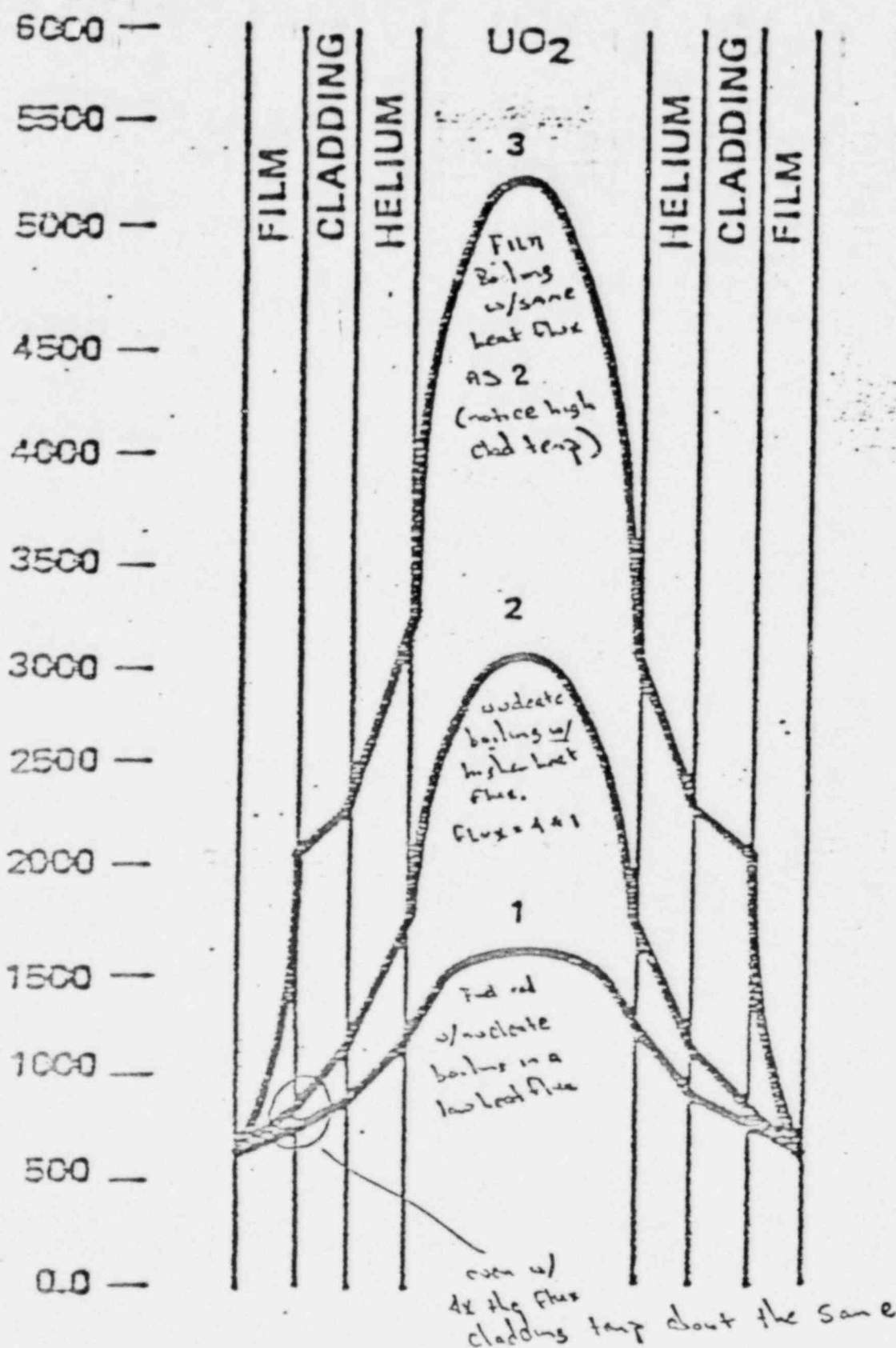


Figure 3-13 Effects of Heat Transfer Modes for a Single Fuel Rod

Center line of fuel
 Outside surface of clad
 OD of clad

TIME CONSTANT

$$\frac{T_1' - T_2}{T_1' - T_1} = e^{-\frac{t}{\tau}}$$

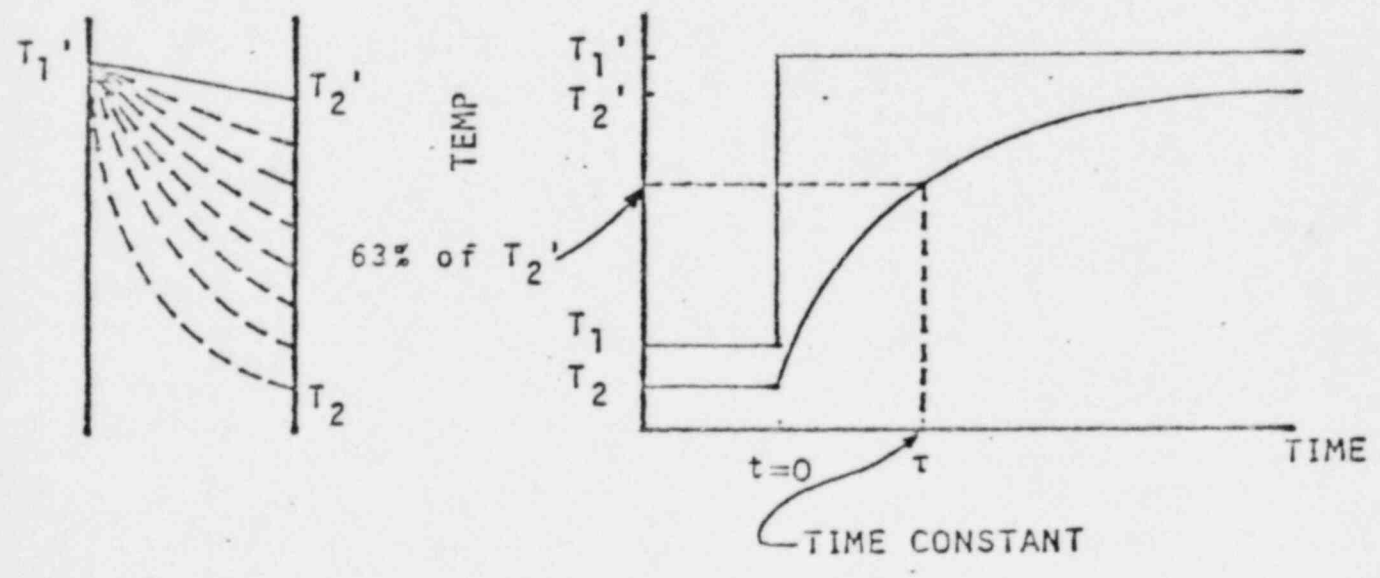
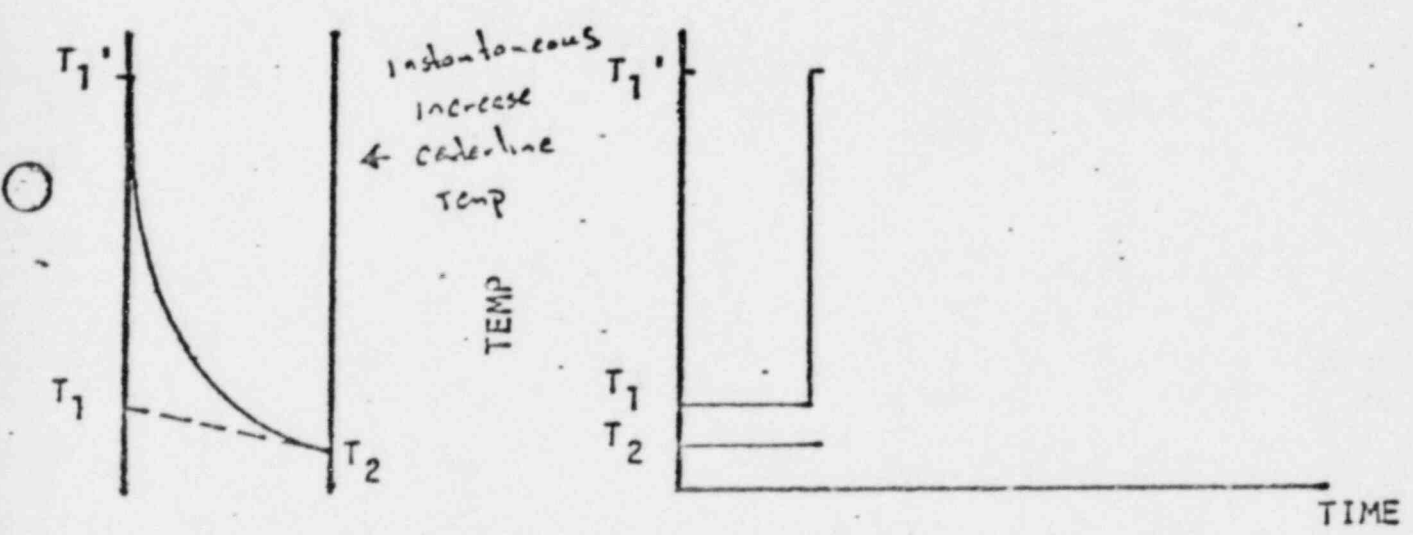
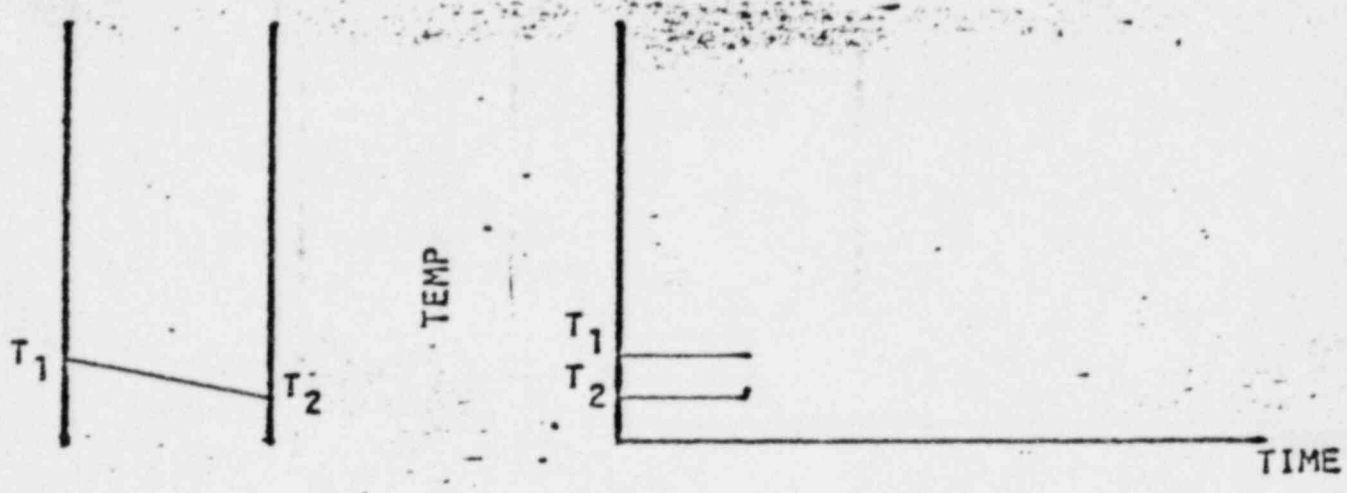


FIGURE 3-12

DEFINITION OF THERMODYNAMICS TERMS

ENCLOSURE 3
For Question 2

ENERGY - ABILITY TO DO WORK OR INTEGRATE USED POWER OVER TIME

ENTHALPY - MEASURE OF THE ENERGY CONTENT OF A SUBSTANCE

ENTROPY - MEASURE OF THE ENERGY CONTENT OF A SUBSTANCE WHICH IS
UNAVAILABLE FOR CONVERSION TO USEFUL WORK

HEAT FLUX - RATE OF HEAT TRANSFER THROUGH A SURFACE AREA

POWER - RATE OF TRANSFERRING ENERGY, OR ENERGY SPENT PER UNIT TIME

PRESSURE - THE FORCE EXERTED BY A FLUID ON A SURFACE

QUALITY - MASS FRACTION OF STEAM IN A STEAM-WATER MIXTURE

SATURATION - THE CONDITION OF TEMPERATURE AND PRESSURE WHERE THE
VAPOR PRESSURE OF THE FLUID EQUALS THE SYSTEM PRESSURE

SUBCOOLING - THE DIFFERENCE BETWEEN THE ENTHALPY OF THE COOLANT AND
SATURATION ENTHALPY

TEMPERATURE - MEASURE OF THE HOTNESS OF A SUBSTANCE

VOID FRACTION - VOLUME FRACTION OF STEAM IN A STEAM-WATER MIXTURE

THERMODYNAMICS

Any material which absorbs or transmits heat is known as a working material. The amount of heat energy can be quantified by measuring temperature, pressure and specific volume. These sources will result in enthalpy of the working material which delineates the state of the working substance. State of a working substance depends on knowing the thermodynamic properties of a working substance. The thermodynamic property of a working material depends on temperature, pressure and specific volume of the working material.

Temperature

Temperature is a measure of hotness or coldness of a substance. It is an indication of the energy content. On a molecular scale if the temperature of a substance is absolute zero, it is an indication of zero energy content since the atoms or molecules of the substance are immobile and unable to do any work. Absolute zero temperature has not been reached even in research laboratories even though the scientists have gotten very close to it: As temperature of a substance increases, the movement of the molecules increases and thus, their ability to do work increases.

Temperature is measured by instrumentation and there are four different scales which relate to temperature. Absolute zero is referred to 0°Kelvin or 0°Rankine . These are two temperature scales which scientists use to relate to absolute zero. The two common temperature measurement scales are, however, centigrade and Fahrenheit. The absolute zero on both of these two scales is -273°C and -460°F , respectively.

Centigrade is a metric scale which is calibrated to zero degrees as the melting point of ice and 100°C as boiling point of water, both at Standard Temperature and pressure at sea level elevation. The Farenheit scale's indication of some states is at 32°F and 212°F , respectively. See Figure 2-1 for clarity.

Conversion of these scales is as follows:

$$^{\circ}\text{K} = ^{\circ}\text{C} + 273$$

$$^{\circ}\text{R} = ^{\circ}\text{F} + 460$$

$$^{\circ}\text{F} = \frac{9}{5} ^{\circ}\text{C} + 32 \quad \text{or} \quad ^{\circ}\text{C} = \frac{5}{9} (^{\circ}\text{F} - 32)$$

Pressure

Pressure is measurement of force per unit area. Force is defined in terms of product of mass and acceleration. If either the mass or acceleration is increased, the force is likewise increased. On a molecular scale force by a substance is the physical contact of the molecules with their corresponding acceleration.

The higher the acceleration, the more movement and therefore the greater the temperature. Therefore, pressure and temperature are interrelated. This is the reason as temperature of a substance increases, its pressure increases. Pressure is increased if either force is increased for the same area, or if for the same force, the area is reduced.

If molecules or atoms have no movement, thus zero temperature, they do not have any force which relates to no pressure. This state is referred to as absolute zero pressure. This state is also referred to as perfect vacuum. This is maximum vacuum

Fahrenheit

Rankine

Centigrade

Kelvin

BOILING
POINT OF
WATER AT
1 atm

FREEZING
POINT OF
WATER

ABSOLUTE
ZERO

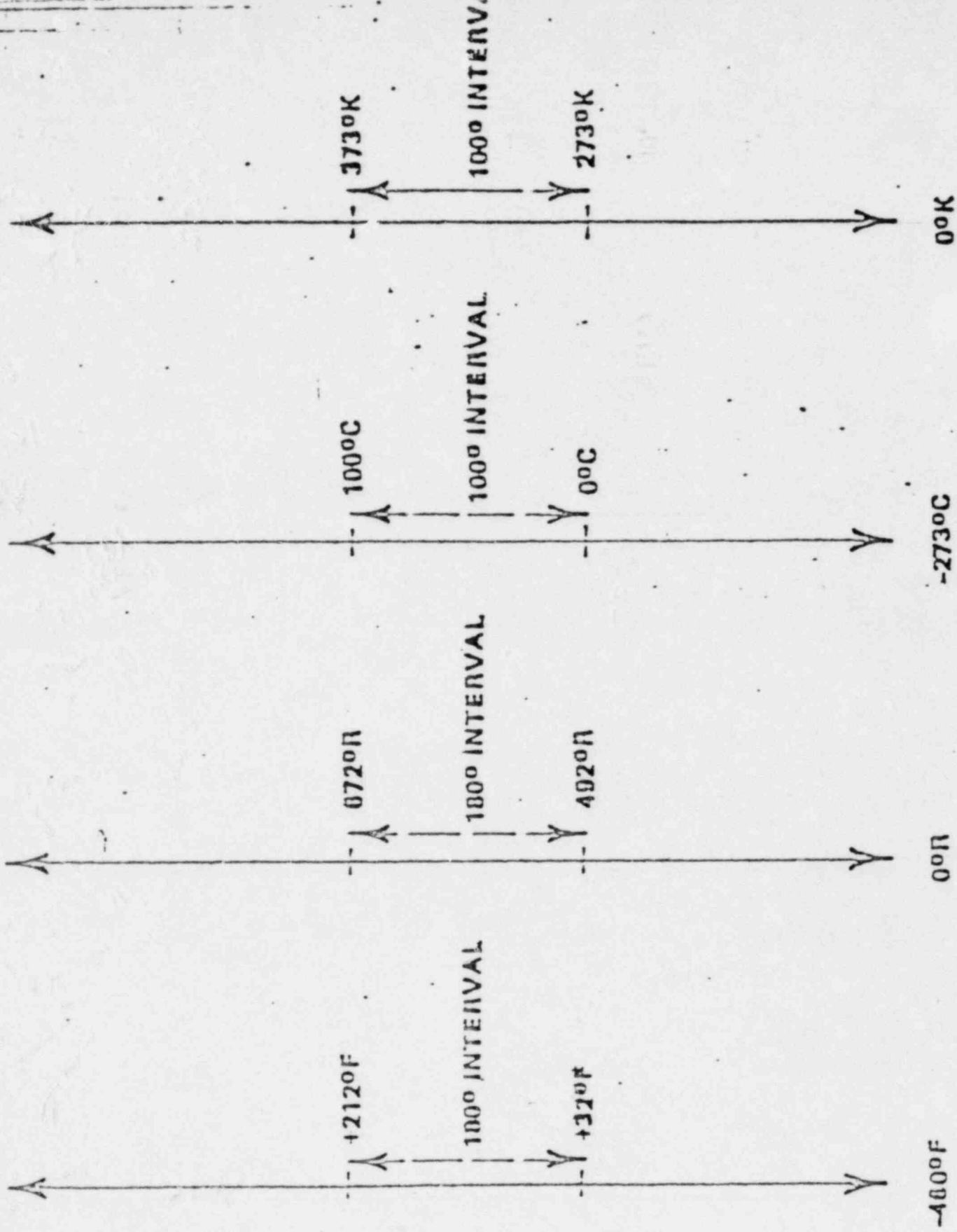


Figure 2-1 Temperature Scale Relationships

which can exist in universe i.e., when no atom moves with increase in molecular movement or temperature increase, pressure is increased and consequently vacuum is decreased. At atmospheric pressure and temperature, there is no vacuum and a pressure equal to 14.7 pounds force per a square inch is existed on everything. At this state, most pressure instrumentation show a zero pressure indication which is also referred to as zero gauge pressure. Any increase in pressure of a system is shown by the instruments as a positive gauge pressure indication. To determine the absolute pressure indication, the gauge value must be added to the 14.7 pounds per square inches. Pounds per square inch is one scale of measurement. Pressure can also be measured as a head in inches of mercury or inches (or sometimes feet) of water. Figure 2-2 exhibits different scales of pressure measurements. Note that atmospheric pressure indication in inches of mercury is the same as condenser back pressure as indicated by the instrumentation.

Enthalpy

Enthalpy is a measure of the energy content of a substance which can result in some type of work. The energy which is in a system is a combination of the internal energy and the amount of work which a substance can do. The capability to do work of a substance as measured by its enthalpy depends on the state of the substance. For a fluid, the enthalpy of a substance in a gaseous state is considerably higher than in liquid state. Tables have been developed for various working fluids which provide values of enthalpies in liquid and vapor state at various tabulated values of pressure and temperature.

COMPARISON OF PRESSURE MEASUREMENT SCALES

AND THEIR UNITS

GAUGE PRESSURE	ABSOLUTE PRESSURE	VACUUM	ATMOSPHERIC PRESSURE	HG COLUMN PRESS.	H ₂ O COLUMN PRESS
PSI GAUGE	PSI ABSOLUTE	INCHES Hg VAC.	INCHES Hg. ABS.	INCH - Hg PRESS.	FT - H ₂ O PRESS
100	114.7			204	231
1	15.7			2.04	2.31
.491	-15.2	N/A ↑	N/A ↑	1.0	.88
0	14.7	0	30	0	0
-.1	14.65	.1	29.9	↓ NOT USED	↓ NOT USED
-1	13.7	2.04	28		
-14.7	0	30"	0"		

FIGURE 2-2

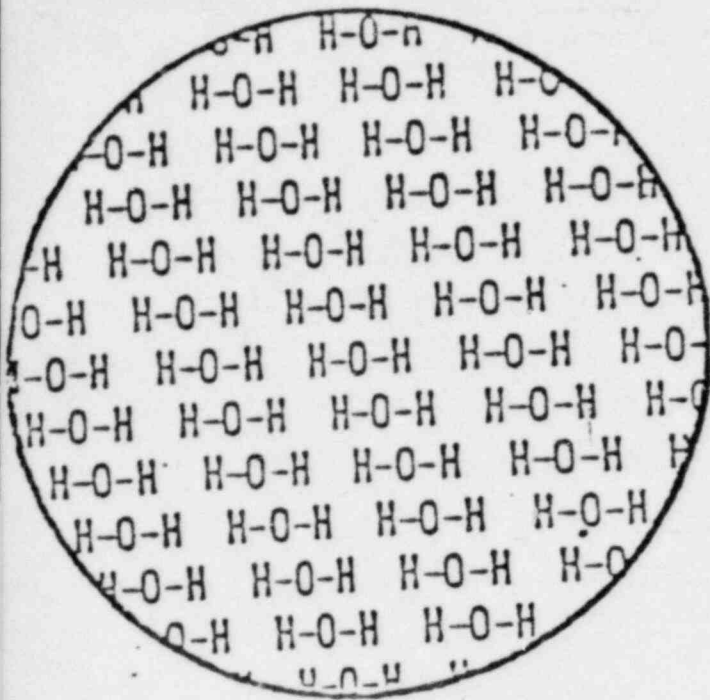
Entropy

Entropy is a measurement of the energy content of a substance which is untransferable or convertible to a useful work. Since, in a power plant we deal with workable substances, this is not a usable term or subject.

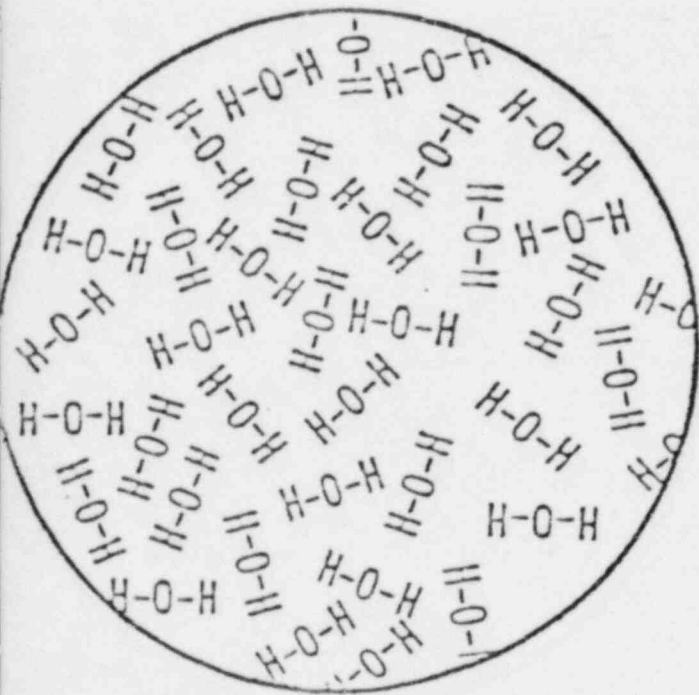
Saturation

This is referred to a state of a fluid where the pressure of the system in the liquid form and in its vapor form are the same. In normal atmospheric pressure, a coolant temperature such as water is increased by adding heat to it. As was previously discussed, as temperature increases, the pressure increases (on the molecular basis). If sufficient molecular pressure is increased, the molecules overcome the surface tension force and vaporize. At this point, the liquid pressure is the same as vapor pressure. At this point, the vapor pressure is 14.7 psia or zero gauge pressure. Clearly, if the pressure above the water is reduced, the molecules vaporize sooner at lower temperature. Likewise, if pressure above the liquid is increased, it will require higher energy or higher temperature to reach vaporization. Another term used for vaporization is boiling. In a BWR, the water at 1000 psig boils at 545°F and if pressure is reduced suddenly, considerable boiling or voiding will take place. Likewise, if pressure is suddenly increased, collapse of bubbles or transfer back to droplets will take place. If the temperature of a fluid in a vapor state is greater than saturation temperature at that pressure, it is referred to as a superheat steam. Similarly, if the fluid temperature in its liquid state is lower than the saturation temperature at that pressure, it is known as subcooled liquid. Subcooling is measured by the difference between the enthalpies of the saturated coolant and enthalpy of the coolant.

PHASE CHANGE



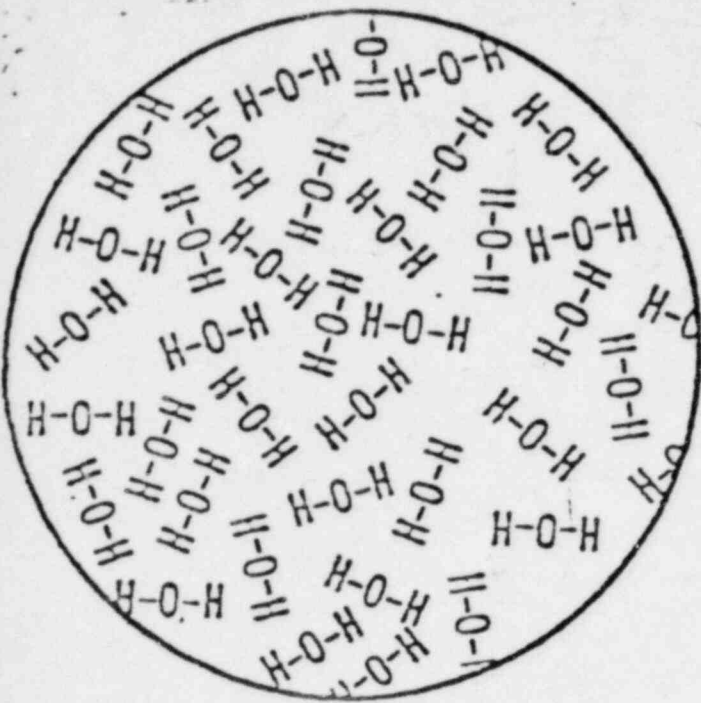
SOLID WATER



LIQUID WATER

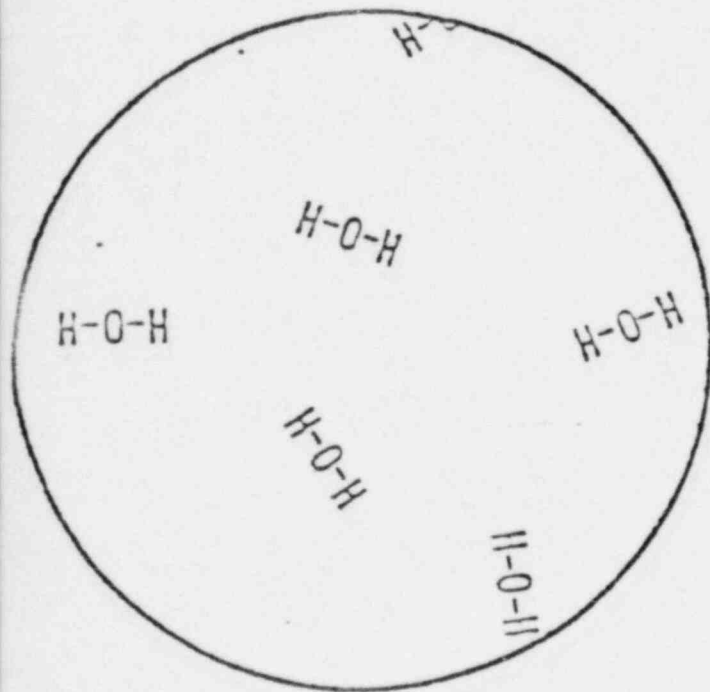
- SOLID WATER EXHIBITS VERY RIGID MOLECULAR ARRANGEMENT.
- THERE IS VIBRATIONAL MOTION OF THE MOLECULES, BUT THERE IS NO MOVEMENT OF MOLECULES FROM THEIR POSITIONS.
- AS HEAT IS ADDED AND THE TEMPERATURE RISES, THE VIBRATIONAL MOTION INCREASES.
- AT A SPECIFIC TEMPERATURE, IF MORE HEAT IS ADDED, THE SOLID CHANGES PHASE INTO A LIQUID.
- DURING THE PROCESS OF CHANGING FROM A SOLID TO A LIQUID, THE TEMPERATURE REMAINS CONSTANT BUT ENERGY IS BEING ADDED TO CHANGE THE PHASE.
- IN THE LIQUID PHASE, THE MOLECULES HAVE FREEDOM TO MOVE FROM ONE POSITION TO ANOTHER IN ADDITION TO VIBRATIONAL MOTION.

Atmospheric Pressure



LIQUID WATER

- AS THE TEMPERATURE OF LIQUID WATER INCREASES, THE VIBRATIONAL MOTION AS WELL AS THE MOVEMENT OF MOLECULES WITHIN THE LIQUID INCREASES.
- THE AMOUNT OF HEAT REQUIRED TO INCREASE THE TEMPERATURE OF ONE POUND OF WATER 1°F IS 1 BRITISH THERMAL UNIT (BTU).
- AT A SPECIFIC TEMPERATURE IF MORE HEAT IS ADDED, THE LIQUID CHANGES PHASE INTO A VAPOR.
- DURING THE PROCESS OF CHANGING FROM A LIQUID TO A VAPOR, THE TEMPERATURE REMAINS CONSTANT BUT ENERGY IS BEING ADDED TO CHANGE THE PHASE.
- THE AMOUNT OF HEAT REQUIRED TO CHANGE ONE POUND OF WATER AT 1000 PSIA INTO STEAM IS 650 BTU.



STEAM

Properties of Water

To understand properties of water, we need first to refer to temperature pressure variation.

At low temperatures and pressure, water molecules are tightly bound together and form a solid substance. At real low pressures, if temperature is increased, phase transfer will take place and water molecules will evaporate into a vapor state without going through liquid phase. This transfer is referred to as "sublimation".

At higher pressures, water from solid phase transfers to liquid phase before transferring to vapor phase. It takes one BTU of energy to raise 1 lb. of water 1°F . But, considerably larger quantity of energy is required to transfer 1 lb. of water into 1 lb. of steam. For example at 1000 psig, it takes one BTU to change 1 lb. of water from 544°F to 545°F . However, for the same 1 lb. of water to change from water to steam, it requires 650 BTU.

Thus, we can use 650 BTU of energy from fuel for every pound of water that boils in the reactor.

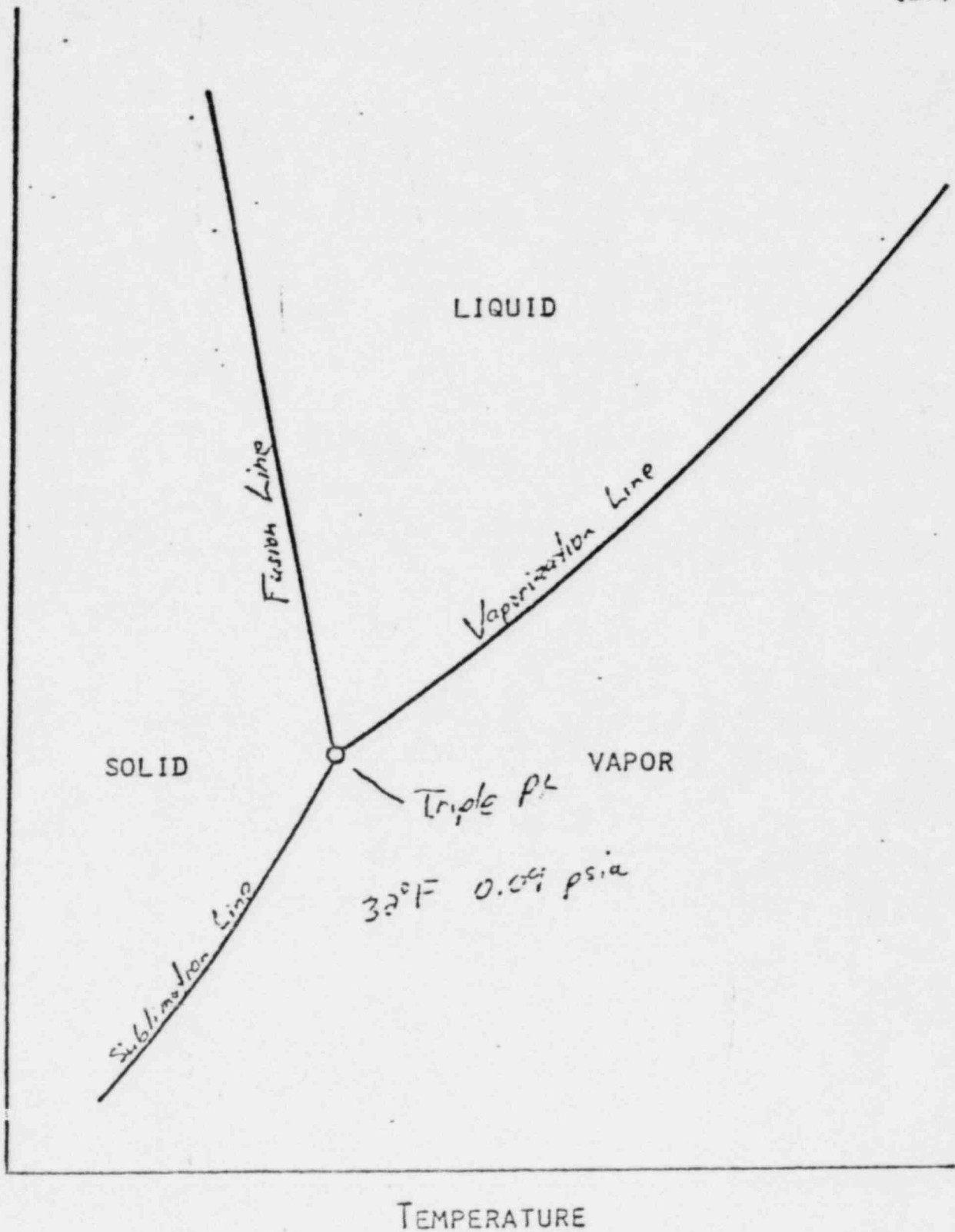
This energy requirement to change from liquid phase to vapor phase is referred to as Latent Heat of Vaporization and is given in the steam tables, Tables 1 and 2 under " h_{fg} " column for enthalpies. Note that when h_{fg} is added to the enthalpy of the liquid " h_f " at any pressure and temperature, it becomes equal to steam enthalpy " h_g ".

Temperature pressure of water is given in Figure 2-3. The triple point is a point where vapor, liquid and solid phase exist at one specific temperature and pressure.

Figure 2-4 shows the functional relationship between temperature and enthalpy for water. The two solid lines exhibit saturated water and saturated steam lines. The saturated steam lines partially curve to the inside.

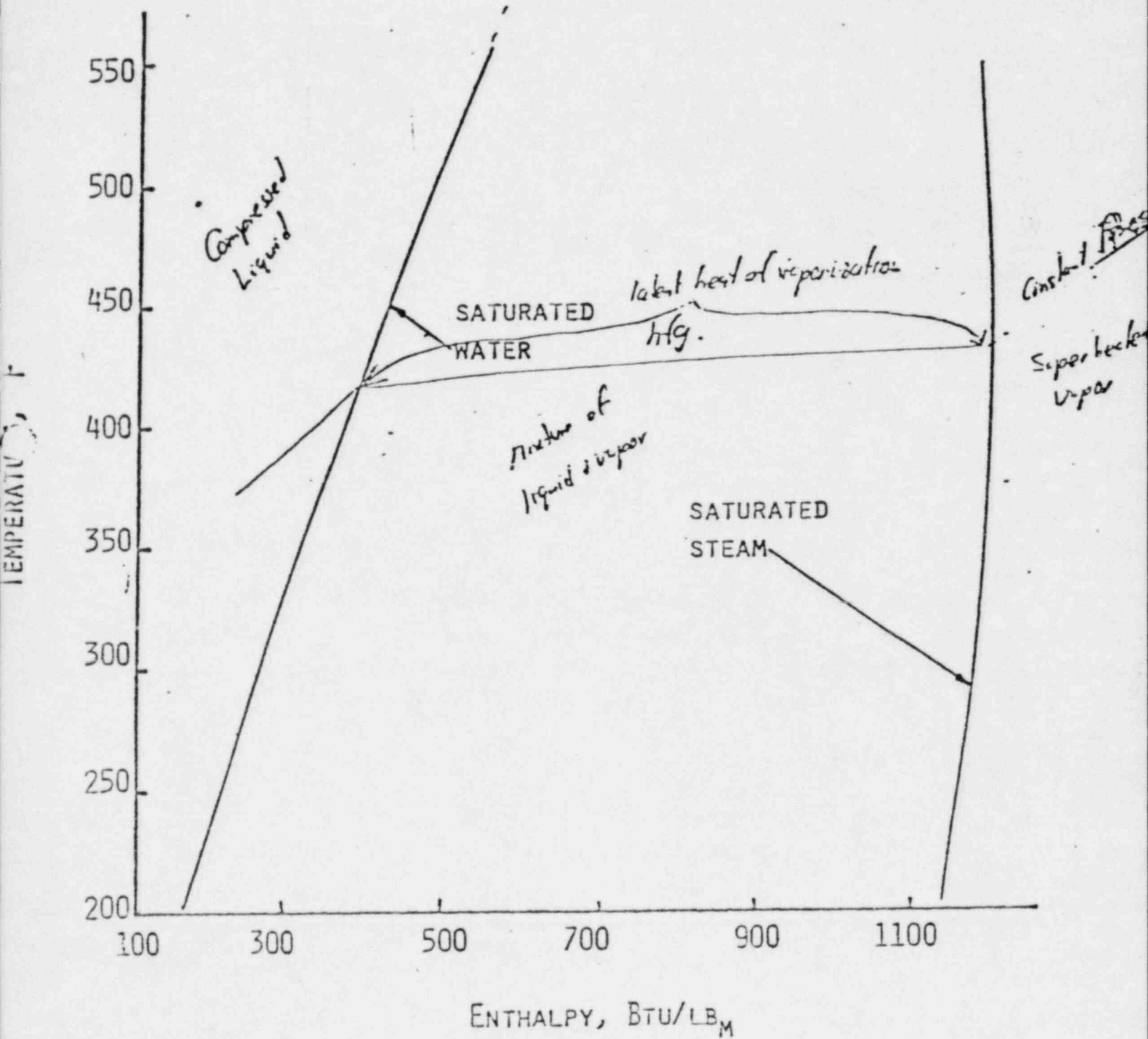
For one pound of water, at 300°F , energy equivalent to 270 BTU is required to reach saturation, where for the same pound of water to undergo phase transformation to steam at the same temperature, roughly 1200 BTU is required. It is noted that the differential energy requirement for phase transformation decreases as temperature increases. At sufficiently high temperature where liquid is in equilibrium with steam, there is no differential energy requirement.

PRESSURE



PRESSURE-TEMPERATURE DIAGRAM FOR WATER

FIGURE 2-3



TEMPERATURE VS. ENTHALPY FOR WATER

FIGURE 2-4

FLUID FLOW AND PRESSURE DROP

ENCLOSURE 4

For Question 2

FOR A GIVEN FLOW AREA AND A GIVEN FLUID, THE PRESSURE DROP ACROSS A LENGTH OF CHANNEL IS PROPORTIONAL TO THE SQUARE OF THE FLOW THROUGH THE LENGTH.

$$\Delta P = f \left(\frac{W^2}{2A^2 \rho g_c} \right) \text{ dynamic head}$$

ΔP = PRESSURE DROP ACROSS LENGTH, lb_f/ft^2

f = PROPORTIONALITY FACTOR ACCOUNTING FOR RESISTANCE TO FLOW

W = MASS FLOW RATE, lb_m/sec

A = AREA OF FLOW, ft^2

ρ = DENSITY OF THE FLUID, lb_m/ft^3

g_c = CONVERSION FACTOR TO CONVERT lb_m TO $\text{lb}_f = 32.2 \frac{\text{lb}_m \text{ft}}{\text{lb}_f \text{sec}^2}$

THIS EQUATION IS USED FOR DETERMINING PRESSURE DROP ASSOCIATED WITH A SPECIFIC FLOW. TO DETERMINE FLOW FROM A GIVEN PRESSURE DROP, REARRANGE THE EQUATION.

$$W = \left(\frac{\Delta P 2A^2 \rho g_c}{f} \right)^{\frac{1}{2}}$$

FACTORS AFFECTING RESISTANCE TO FLOW

- FRICTION - RESISTANCE TO FLOW CAUSED BY SURFACE FRICTION
 - ROUGHNESS - A SMOOTH PIPE HAS MUCH LESS FRICTIONAL RESISTANCE TO FLOW THAN A ROUGH PIPE.
 - VELOCITY - AS THE VELOCITY INCREASES, THE FRICTIONAL RESISTANCE TO FLOW ALSO INCREASES.
- ACCELERATION - THE MOMENTUM OF THE FLUID RESISTS CHANGES IN VELOCITY SO ANY TIME THE FLUID IS ACCELERATED, RESISTANCE TO FLOW INCREASES.
 - CHANGES IN AREA - FLUID WHICH FLOWS THROUGH A SMALLER DIAMETER FLOW AREA MUST BE ACCELERATED.
 - CHANGES IN VOLUME - AS THE FLUID BOILS, THE STEAM FORCES THE FLUID TO CHANGE VELOCITY.
- DENSITY DIFFERENCE - IF A FLUID IS HEATED, GRAVITY WILL TEND TO MAKE WARMER FLUID RISE AND BE DISPLACED BY COOLER FLUID.
 - SINGLE PHASE
 - 2 PHASE
- LOCAL RESTRICTIONS - WHEN A FLUID ENCOUNTERS A RESTRICTION IN THE FLOWING AREA, RESISTANCE TO FLOW INCREASES BECAUSE THE FLUID MUST CHANGE ITS DIRECTION OF FLOW TO CIRCUMVENT THE RESTRICTION.

FLOW THROUGH THE FUEL CHANNEL

SINGLE PHASE FLOW

- FLOW THROUGH THE CHANNEL IS GOVERNED BY THE FOLLOWING EQUATION:

$$W = \left(\frac{\Delta P 2 A^2 \rho g_c}{f} \right)^{\frac{1}{2}}$$

- f IS THE FACTOR ACCOUNTING FOR RESISTANCE TO FLOW
 - ADJUSTED FOR CRUD BUILDUP
 - MODIFIED FOR ACCELERATION THROUGH THE UPPER AND LOWER TIE PLATES
 - MODIFIED FOR THE GRID SPACERS AND THE INLET ORIFICE

TWO PHASE FLOW

- THE BOILING PROCESS CREATES ADDITIONAL RESISTANCE TO FLOW.
- THE RESISTANCE TERM, ~~THE RESISTANCE TERM, ϕ^2~~ , IS A VERY STRONG FUNCTION OF THE QUALITY OR VOID FRACTION.
- TWO PHASE FLOW CAN BE MODELED BY AN EQUATION SIMILAR TO SINGLE PHASE FLOW

$$W = \left(\frac{\Delta P 2 A^2 g_c \rho}{f \phi^2} \right)^{\frac{1}{2}} \quad \phi^2 \gg 1$$

- THE UNDESIRABLE EFFECT OF TWO PHASE FLOW FRICTIONAL LOSSES IS MINIMIZED BY THE INLET ORIFICES.

TWO-PHASE FRICTION FACTOR

Figure 4-1 is a typical experimentally derived two-phase friction factor plot. Note that at a quality of 0, the factor ϕ^2 is 1. This would simply indicate single phase flow. Also note that the factor goes down at very high qualities. For typical BWR operations, this friction factor might have values of 5 to 10.

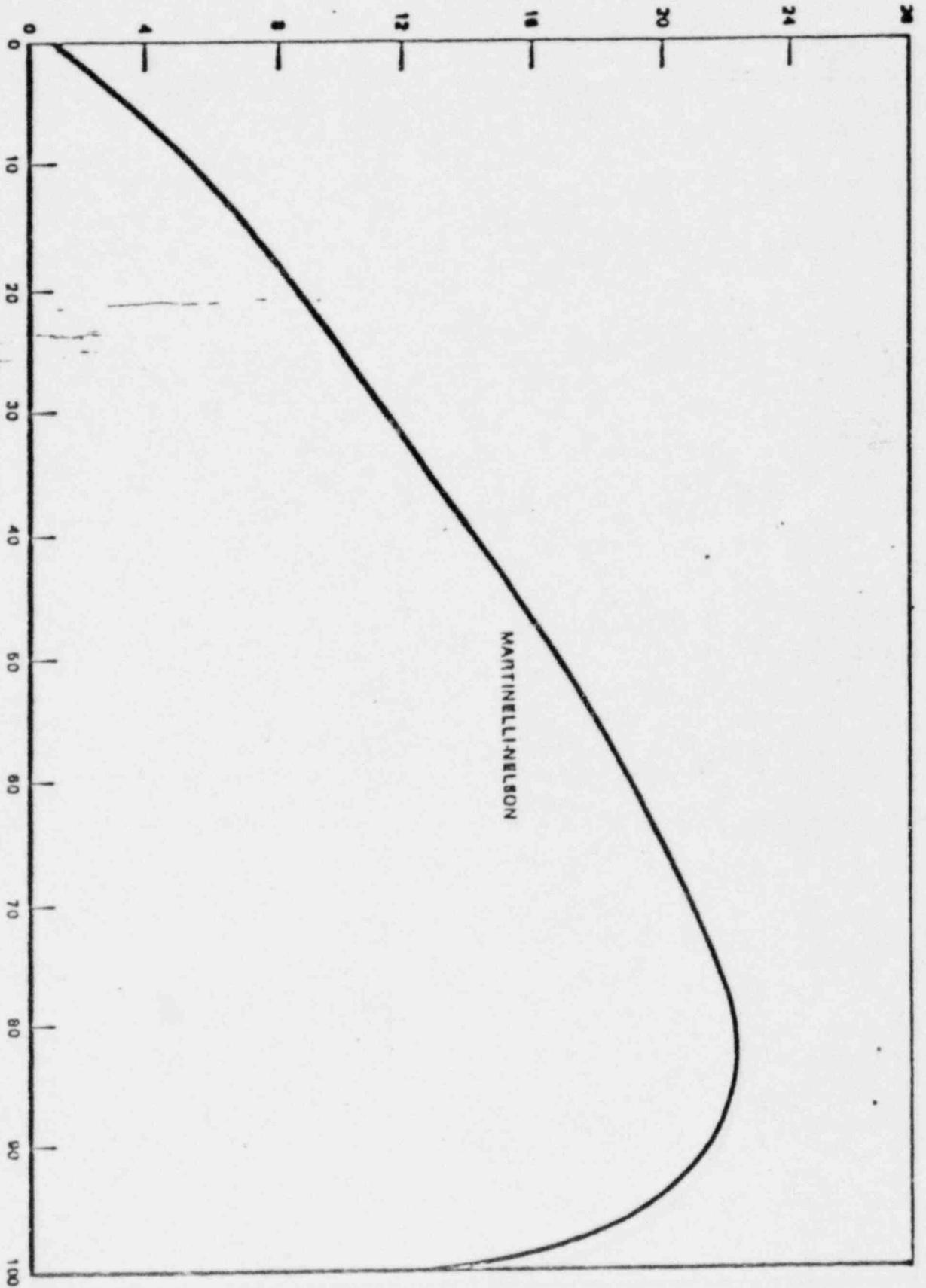
Core Orificing

At zero power, the velocity of water entering any fuel assembly is higher than the velocity leaving, assuming that there is a small amount of leakage from the channel. Since the core is a large series of parallel flow paths through the fuel assemblies and each path begins and ends in a constant pressure plenum chamber, all assemblies will have identical pressure drops at all times; the flow through all assemblies, without any difference in inlet orificing, would be the same, since the power is zero. The flow is single phase. Suppose that the power could be slowly increased in one and only one fuel assembly. What would happen to the flow in that assembly?

With minimum recirculation system flow, as power increases, the flow in the assembly increases. This is due to the natural circulation effect. The hotter water in the channel is less dense than the water in the downcomer area and gravity will force the cold water down, forcing the hot water up. As boiling begins, the steam bubbles will exert a buoyant force on the water above them. This also will cause the flow to increase. The volume change as the water is converted to steam will begin to cause the outlet velocity from the channel to increase above that of the inlet.

As power increases further and more of the water is converted to steam, the outlet velocity continues to rise and channel friction, being a function of

2



Ratio Two-Phase to Single Phase Pressure Loss Versus Quality



velocity, will increase. The increased friction results in an increased pressure drop, but it cannot result in an increased pressure drop because the inlet and outlet pressures are being controlled by all of the other zero-power assemblies in the core. So what happens is that the flow in the channel will decrease as power increases.

Obviously, it is not good to have the lowest flow in the highest power fuel, but that happens. More importantly, is the flow cutback caused by the power increase too high? The answer is yes. What can be done to improve this? Provide inlet orificing.

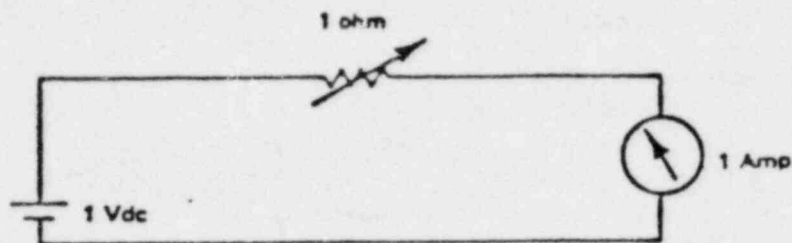
The inlet orifices work much as a biasing resistor in an electrical circuit works. Figure 4-2(a) depicts a simple circuit in which a 1-volt battery delivers a 1-ampere current through a variable resistor set at 1 ohm. In Figure 4-2(b), when the resistance is increased to 2 ohms, the current drops to $\frac{1}{2}$ amp. This represents a 50% reduction in flow. In Figure 4-2(c) a 9-ohm biasing resistor has been added and a larger battery so that with the variable resistor set at 1 ohm, the current flow is again 1 ampere. When the variable resistor is increased to 2 ohms [Figure 4-2(d)], the current drops to only 0.91 amp. In this circuit, for the same change in resistance, the flow or current reduction is only 9%.

Orifices are added to the core to reduce the magnitude of the flow cutback in the fuel channels as assembly power is increased.

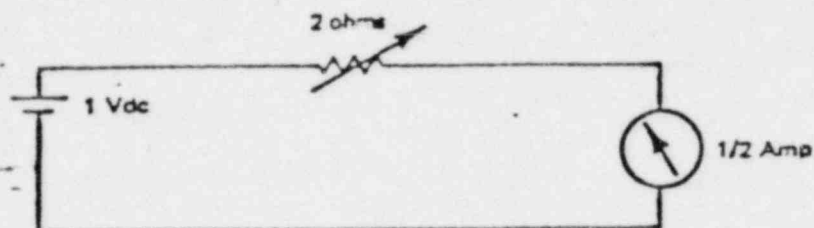
Two things are immediately apparent: First, the orifice will increase the pumping power required for a given flow; second, it should be possible to use smaller or tighter orifices on low-power bundles such as those with low enrichment or those found on the periphery of the core. Both of these are incorporated in the BWR design.

The effect of orifice sizing is shown in Figure 4-3. Note that the tighter the orificing, the flatter the lines as assembly power as indicated by an increasing radial power factor is increased. Without orificing, the slope would be almost vertical.

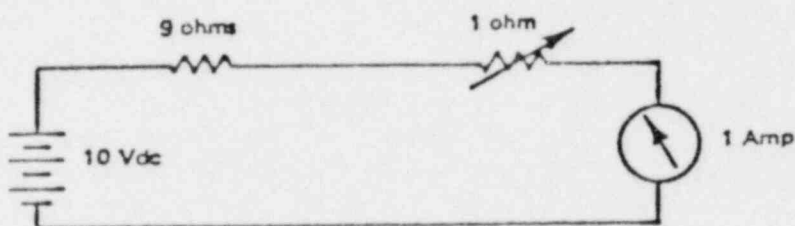
(a)



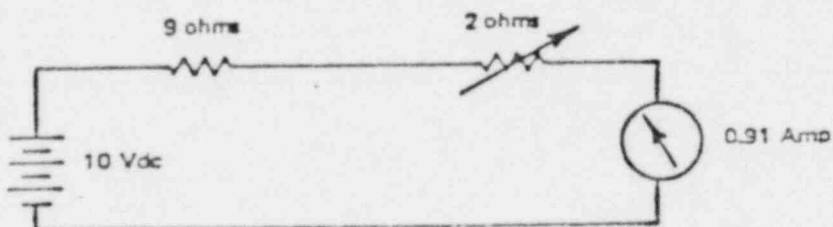
(b)



(c)



(d)



INLET ORIFICE ELECTRICAL ANALOGY

FIGURE 4-2

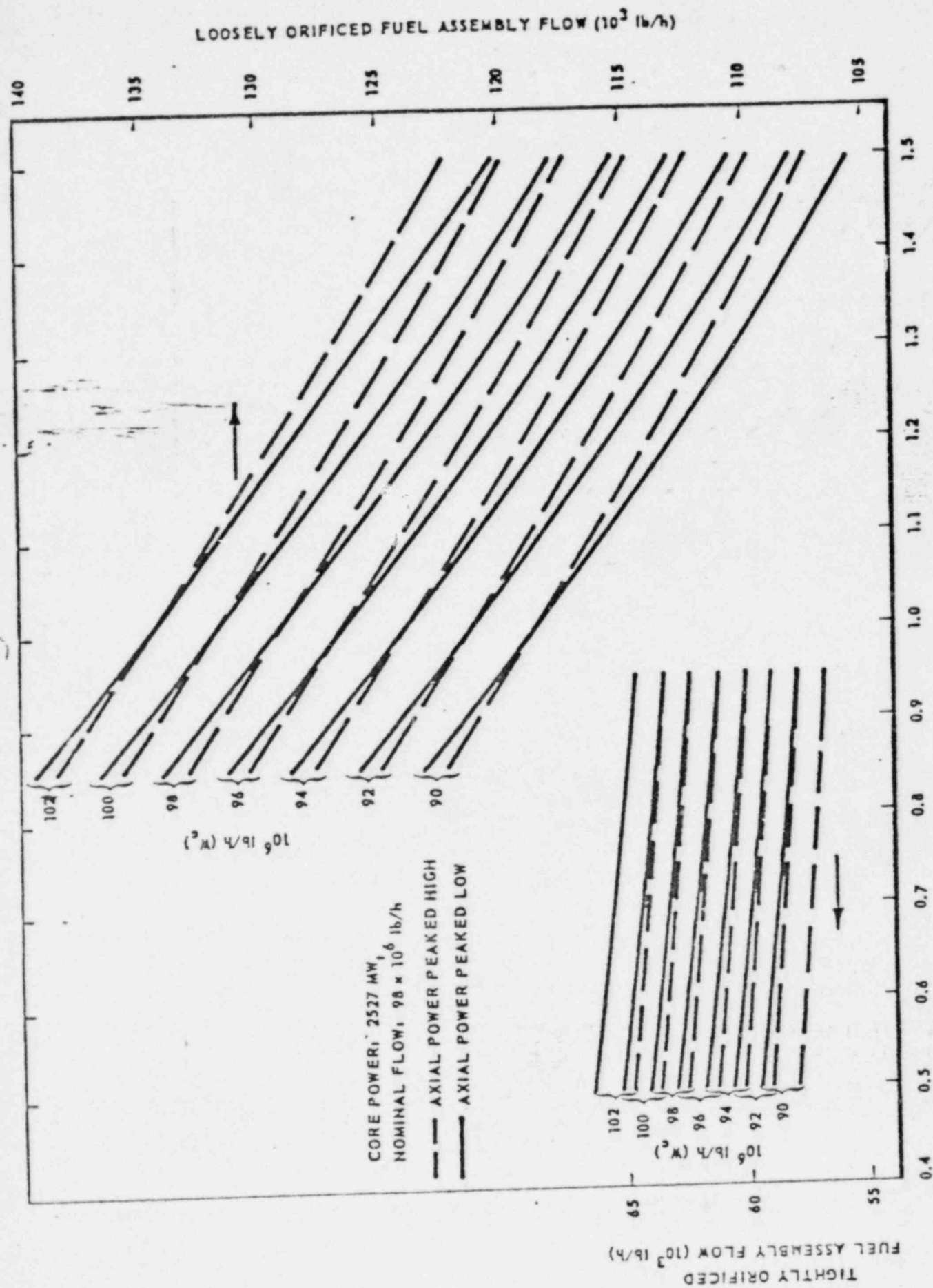


FIGURE 4-3 Fuel Assembly Coolant Flow as a Function of Radial Power Factor

FLOW THROUGH VESSEL PIPING

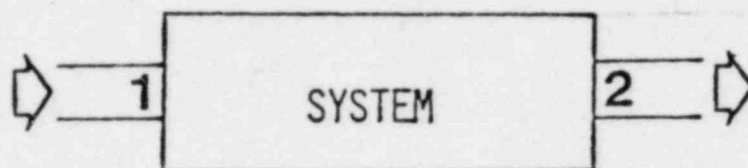
FLOW THROUGH PIPES

- FLUID FLOW THROUGH VESSEL PIPING IS GOVERNED BY THE FOLLOWING EQUATION:

$$W = \left(\frac{\Delta P 2 A^2 g_c \rho}{f} \right)^{\frac{1}{2}}$$

- IF A PIPE BREAK WERE TO OCCUR AT HIGH PRESSURE -
 - LARGE INCREASE IN ΔP
 - LARGE DECREASE IN f
 - THEREFORE THE FLOW THROUGH THE PIPE WOULD INCREASE DRAMATICALLY
- THERE IS A LIMIT ON THE MAXIMUM FLOW THROUGH A PIPE, CALLED CHOKE FLOW WHICH IS EQUAL TO THE SPEED OF SOUND.
- AS THE FLOW VELOCITY INCREASES IN A PIPE, THE PRESSURE IN THAT PIPE DECREASES (CONSERVATION OF ENERGY).
 - IF THE PRESSURE DECREASES BELOW THE SATURATION PRESSURE, THE FLUID WILL BOIL.
 - IF THE PRESSURE THEN INCREASES ABOVE THE SATURATION PRESSURE, THE BUBBLES WILL COLLAPSE, FORMING SHOCK WAVES WHICH STRIKE THE SIDES OF THE PIPE.
 - THIS PROCESS OF FORMATION AND COLLAPSING OF THE BUBBLES IS CAVITATION AND COULD CAUSE DAMAGE TO THE WALLS OF THE PIPE.

GENERAL ENERGY EQUATION FOR INCOMPRESSIBLE FLOW



THE TOTAL ENERGY CONTENT OF THE FLUID ENTERING THE SYSTEM AT POINT 1 PLUS ANY ENERGY CHANGE OF THE FLUID CAUSED BY HEAT TRANSFER OR WORK DONE BY OR ON THE FLUID WHILE IN THE SYSTEM WILL EQUAL THE TOTAL ENERGY CONTENT OF THE FLUID LEAVING THE SYSTEM AT POINT 2. THE TOTAL ENERGY CONTENT OF THE FLUID IS IN THE FORMS OF POTENTIAL, KINETIC, AND FLOW ENERGIES.

FIGURE 4-4

FORMS OF TOTAL ENERGY IN A SYSTEM

THE TOTAL ENERGY OF A FLUID PER LB_M IS THE SUM OF THREE TERMS:

$$\text{POTENTIAL ENERGY} = zg/g_c$$

z = elevation, ft

g = acceleration due to gravity = 32.2 ft/sec^2

$$g_c = 32.2 \text{ lb}_m \cdot \text{ft}/\text{lb}_f \text{sec}^2$$

$$\text{KINETIC ENERGY} = v^2/2g_c$$

v = velocity, ft/sec

$$\text{FLOW ENERGY} = P/\rho$$

P = absolute pressure, lb_f/ft^2

ρ = fluid density, lb_m/ft^3

THUS, WE CAN SAY THAT FOR ANY SINGLE POINT ALONG A
STREAM OF INCOMPRESSIBLE FLUID

$$Zg + \frac{v^2}{2g} + \frac{P}{\rho} = \text{CONSTANT}$$

IF THE FLOW IS ASSUMED TO BE HORIZONTAL, THE FIRST
TERM CAN BE DELETED. THUS

$$\frac{v^2}{2g} + \frac{P}{\rho} = \text{CONSTANT}$$

THUS, ALONG ANY HORIZONTAL STREAM OF INCOMPRESSIBLE
FLUID, AS THE STREAM VELOCITY GOES UP, THE PRESSURE
DOES DOWN PROPORTIONAL TO THE VELOCITY SQUARED.

CAVITATION IN A PIPE CONTRACTION

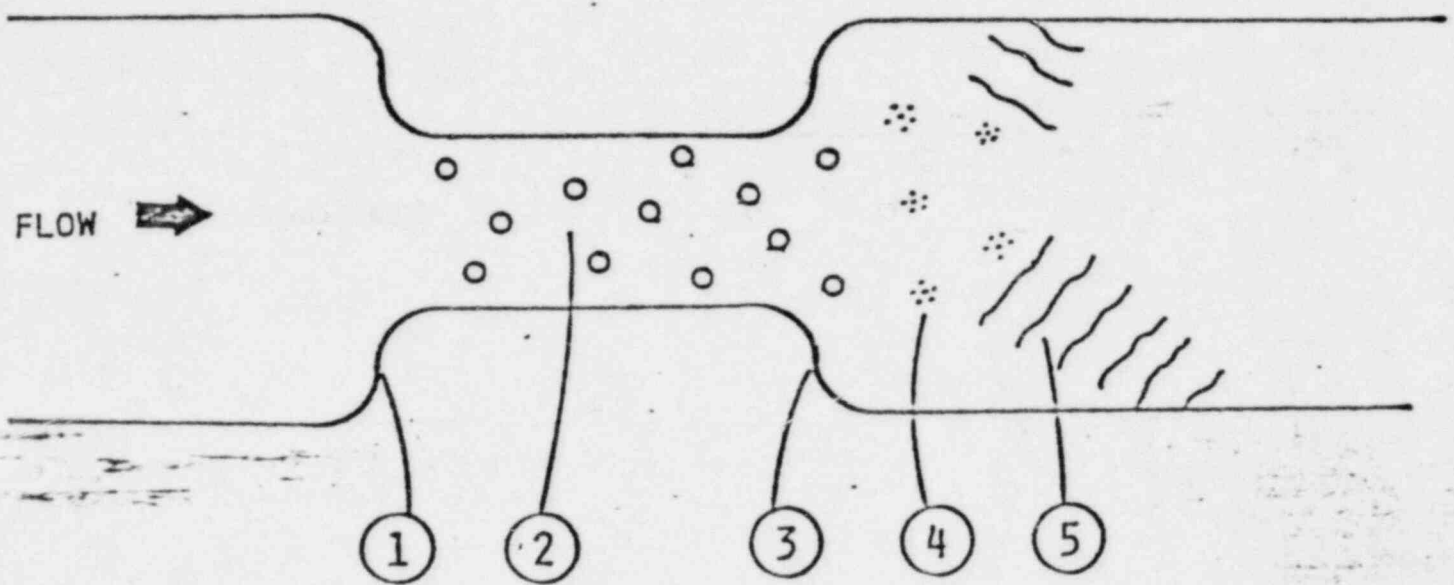


FIGURE 4-5

- ① CONTRACTION OF PIPE CAUSES FLOW VELOCITY INCREASE AND PRESSURE REDUCTION.
- ② BUBBLES FORM AS THE FLUID PRESSURE FALLS BELOW SATURATION PRESSURE.
- ③ EXPANSION OF PIPE CAUSES FLOW VELOCITY DECREASE AND PRESSURE INCREASE.
- ④ BUBBLES COLLAPSE.
- ⑤ SHOCK WAVES PRODUCED FROM THE COLLAPSING BUBBLES STRIKE THE PIPE WALL POSSIBLY CAUSING DAMAGE.

FLOW THROUGH PUMPS

- THE PURPOSE OF A PUMP IS TO TRANSFER FLUID FROM ONE POINT IN A SYSTEM TO ANOTHER BY INCREASING THE PRESSURE OF THE FLUID AS IT PASSES THROUGH THE PUMP.
- IN A CENTRIFUGAL PUMP, THE ROTATING IMPELLER DRAWS WATER INTO THE EYE OF THE IMPELLER AND INCREASES ITS PRESSURE BY CENTRIFUGAL FORCE AS THE WATER IS THROWN TO THE PERIPHERY.
- THIS CREATES A VERY LOW PRESSURE AT THE EYE OF THE PUMP.
- IF THE PRESSURE FALLS BELOW SATURATION PRESSURE, CAVITATION WILL OCCUR.
- NET POSITIVE SUCTION HEAD (NPSH) IS A MEASURE OF HOW CLOSE THE FLUID AT THE EYE OF THE PUMP IS TO SATURATION CONDITIONS.
- NPSH IS THE DIFFERENCE IN PRESSURE BETWEEN THE STATIC PRESSURE AT THE EYE OF THE PUMP AND SATURATION PRESSURE.
- NPSH IS GIVEN IN THE EQUIVALENT HEIGHT OF A COLUMN OF WATER AT 68°F AND 1 ATM.

$$NPSH = (P_i - P_s) \frac{g_c}{\rho g}$$

P_i = ABSOLUTE STATIC PRESSURE AT THE EYE OF THE IMPELLER, lb_f/ft^2

P_s = SATURATION PRESSURE OF THE FLUID, lb_f/ft^2

$\frac{g_c}{\rho g}$ = CONVERSION FROM lb_f/ft^2 TO ft OF H_2O

ρ = DENSITY OF THE FLUID, lb_m/ft^3 (AT 68°F)

$$NPSH = (P_1 - P_s) g_c / \rho g$$

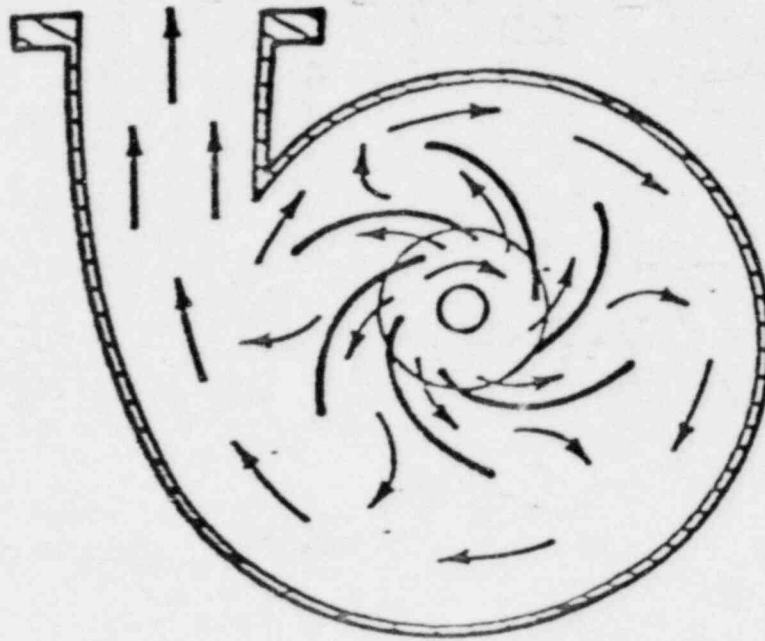


FIGURE 4-6

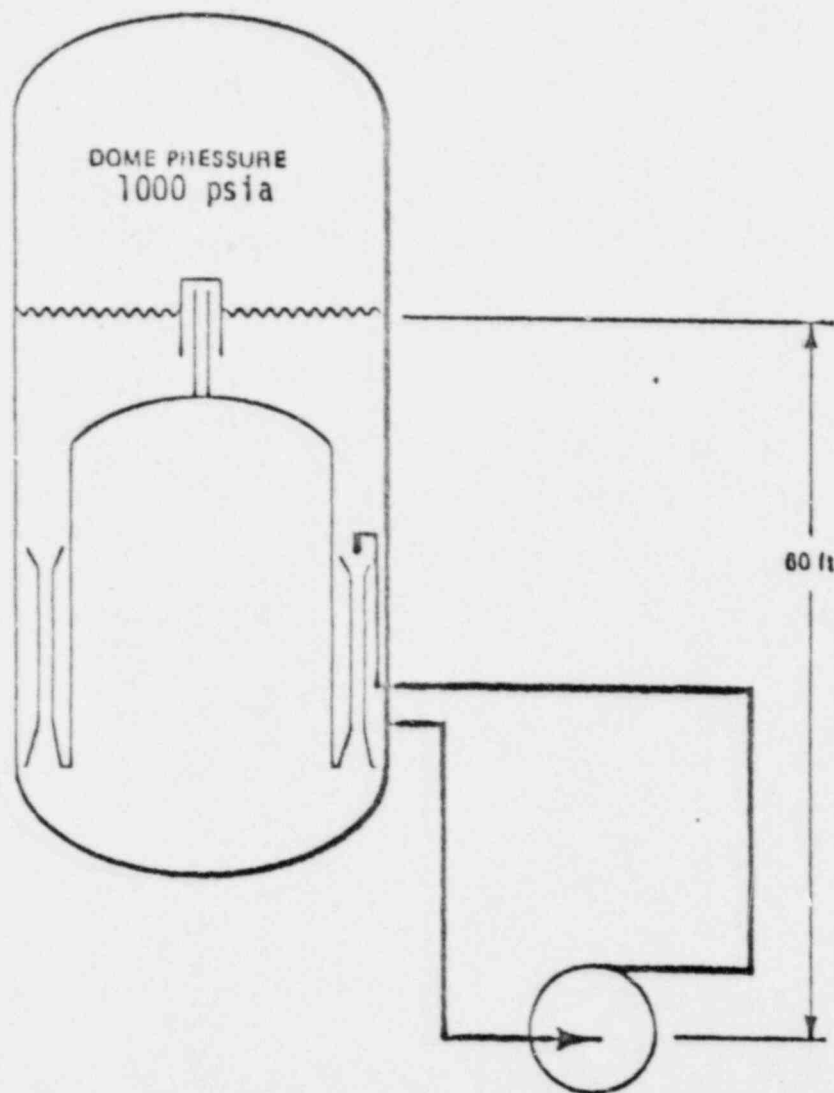
FACTORS AFFECTING AVAILABLE NPSH OF THE RECIRC PUMP

- PHYSICAL HEIGHT OF WATER ABOVE THE PUMP
- DENSITY OF THE WATER
- PRESSURE OF THE WATER
- TEMPERATURE OF THE WATER

FACTOR AFFECTING REQUIRED NPSH OF THE RECIRC PUMP

- SPEED OF THE PUMP
- *Design of the pump*

RECIRC PUMP NPSH



Given: Reactor at full power
 Pressure at Dome: 1000 psia
 Eye of Recirc Pump 60 ft. below H₂O level
 Subcooling: 20 BTU/lb

$$NPSH = \frac{P_1 - P_s}{\rho} = \frac{P_1}{\rho} - \frac{P_s}{\rho} \quad \left(\frac{g_c}{g} = 1\right)$$

$$\frac{P_1}{\rho} = \frac{P_{\text{dome}}}{\rho} + \frac{P_{H_2O}}{\rho}$$

density at 1000

$$= \frac{1000 \text{ psia} \cdot 144 \frac{\text{sq in}}{\text{sq ft}}}{62.4 \text{ lb/ft}^3} + \frac{60 \text{ ft} \cdot 463 \text{ lb/ft}^3}{62.4 \text{ lb/ft}^3}$$

$$= 2308 \text{ ft} + 44.5 \text{ ft}$$

$$\frac{P_s}{\rho} : P_s \text{ is the saturation of 20 BTU/lb}$$

subcooled reactor water at

$$1000 \text{ psia } h_f = 543 \text{ BTU/lb}$$

$$\text{subcooled } 20 \frac{\text{BTU}}{\text{lb}} : 543 - 20 = 523 \frac{\text{BTU}}{\text{lb}}$$

$$\text{thus } P_s = 878 \text{ psia}$$

$$\frac{P_s}{\rho} = \frac{878 \text{ psia} \cdot 144 \frac{\text{sq in}}{\text{sq ft}}}{62.4 \text{ lb/ft}^3} = 2026 \text{ ft}$$

$$NPSH = \frac{P_1}{\rho} - \frac{P_s}{\rho}$$

$$= 2308 \text{ ft} + 44.5 \text{ ft} - 2026 \text{ ft}$$

$$= 327 \text{ ft}$$

FIGURE 4-7

V CRITICAL POWER

The design and operation of a reactor must ensure that any accident initiated from an allowed operating condition will not result in a radio-activity release off-site. These allowed operating conditions are specified as temperatures, pressures, and power levels which must not be exceeded. As in heat transfer and thermal-hydraulics, the bases for a given Limiting Condition for Operation (LCO) are a bit more complicated for a BWR than for a PWR. Taken separately, however, the limits are not difficult to learn.

A. Transition Boiling

In the discussion of the pool-boiling curve, it was remarked that at the departure from nucleate boiling, a small increase in heat flux results in a large increase in the temperature required to drive the heat to the water. In a flowing system, when DNB is reached, patches of steam form on the cladding surface and detach, rewetting the cladding. As a result, the cladding surface temperature fluctuates as these steam patches form and detach. Increasing values of heat flux cause larger temperature fluctuations. This condition of steam patch formation and cladding surface temperature fluctuation is called transition boiling and is shown in Figure 14.

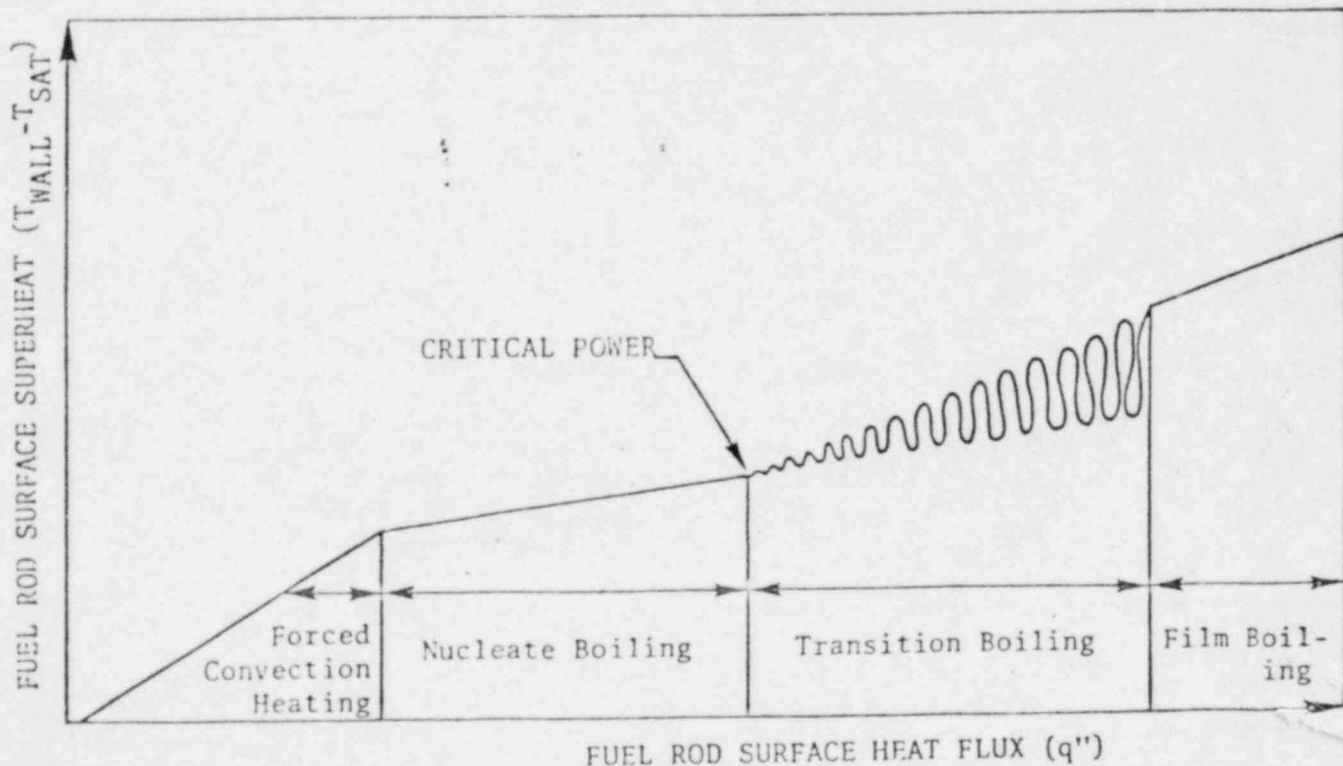
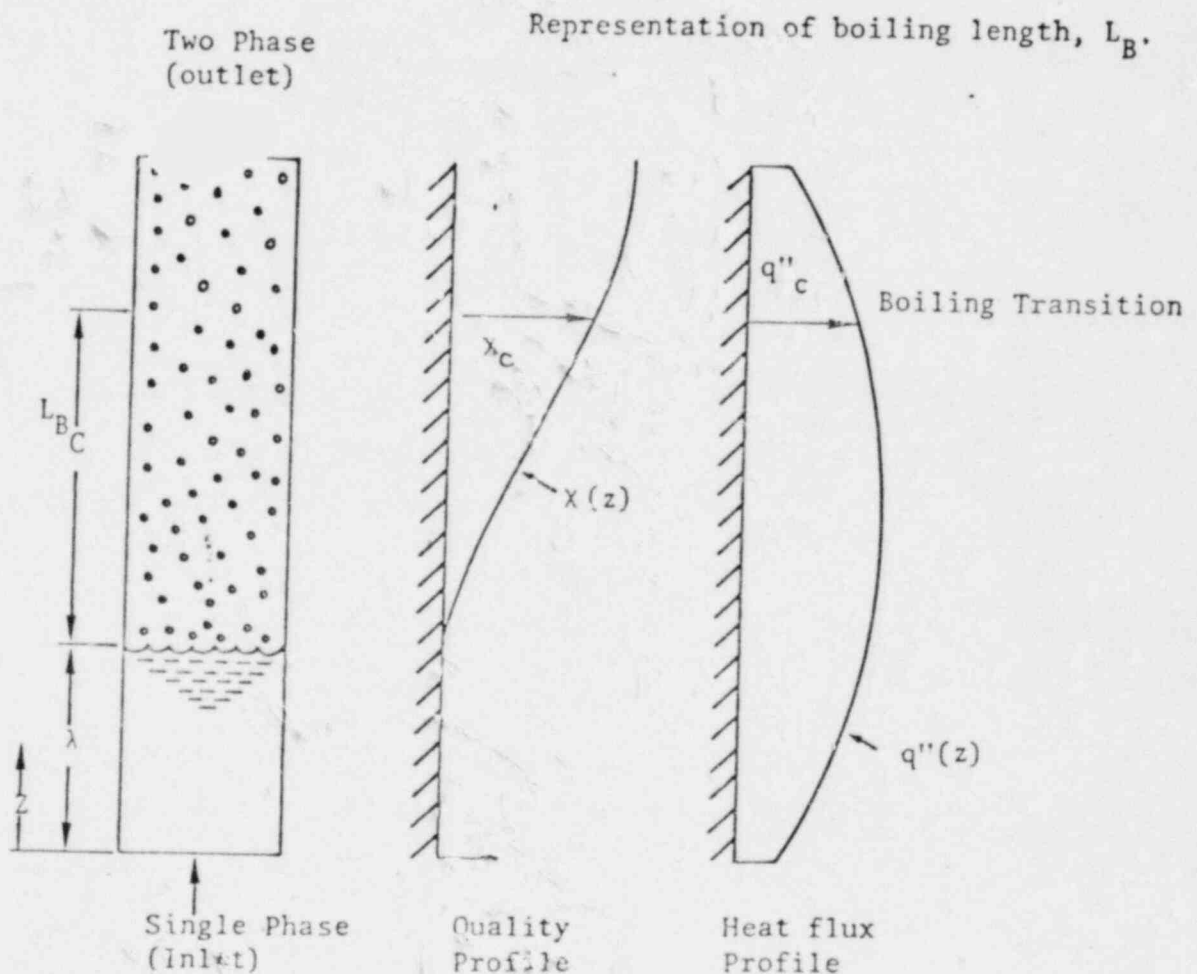


Figure 14

B. Critical Quality

As water flows up the coolant channel, the quality increases from zero to some exit quality. If the heat flux is too high, the quality increases rapidly and a boiling transition takes place at some point in the channel. The quality at the point of transition boiling is called the critical quality. Critical quality is a function of critical boiling length. The critical boiling length is defined as the distance from the onset of bulk boiling to the point of transition boiling. Figure 15 shows critical boiling length, quality, and the point of boiling transition for a typical BWR channel. At very high values of heat flux, transition boiling will occur a short distance from the point where bulk boiling begins. In this case, the



Figures 15

quality at the point of transition boiling is low. At lower values of heat flux, which are still high enough to cause a boiling transition, the boiling length is greater and the critical quality is higher. A sketch of critical quality versus critical boiling length is shown in Figure 16. A major advantage of plotting critical quality versus boiling length is that this curve is independent of the axial heat flux profile. Although all the experimental data on transition boiling plots very nicely on a graph of critical quality versus boiling length, neither of these parameters are measurable in an operating reactor. A correlation had to be made between the critical quality and the power required to produce critical quality in a fuel bundle.

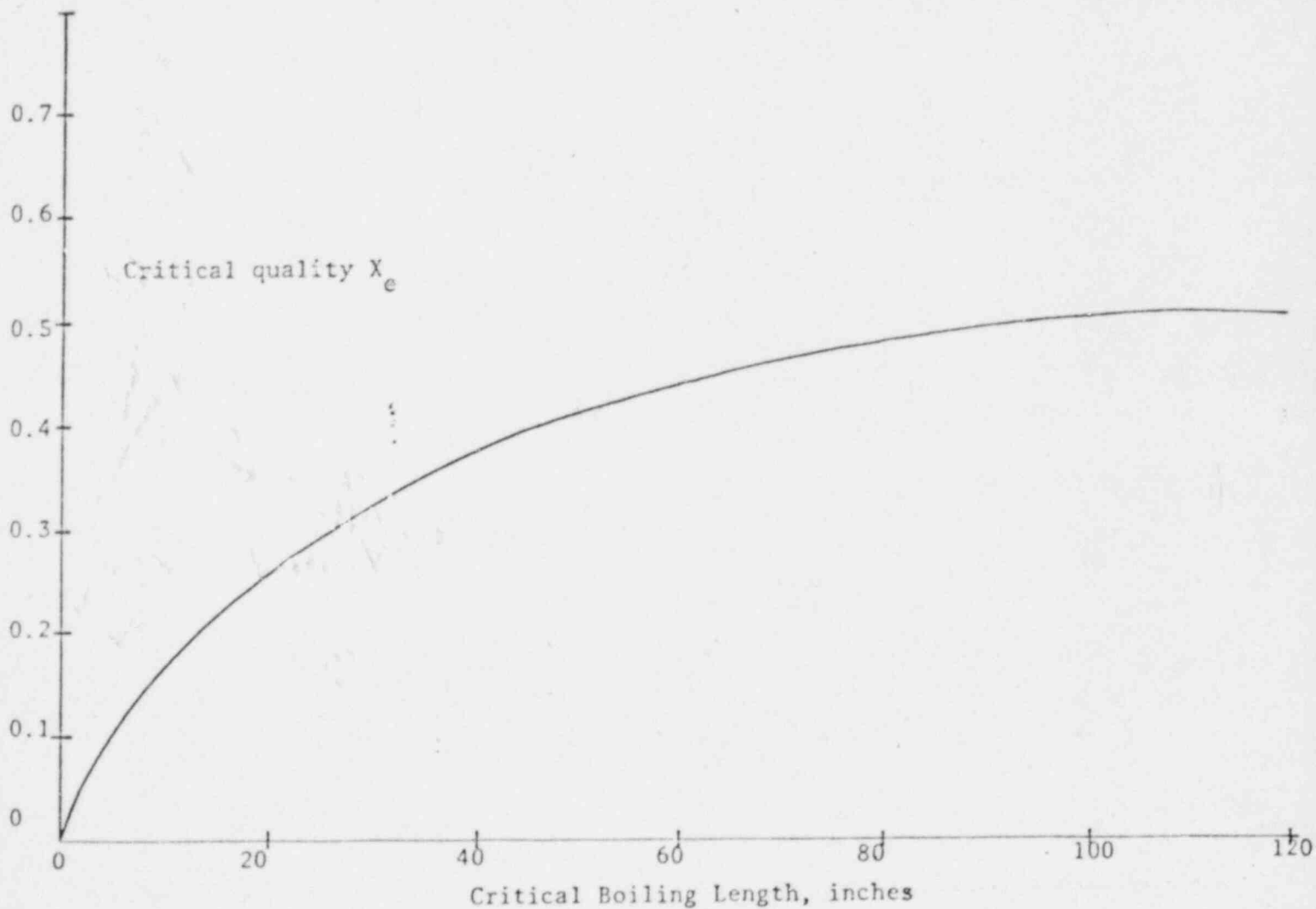


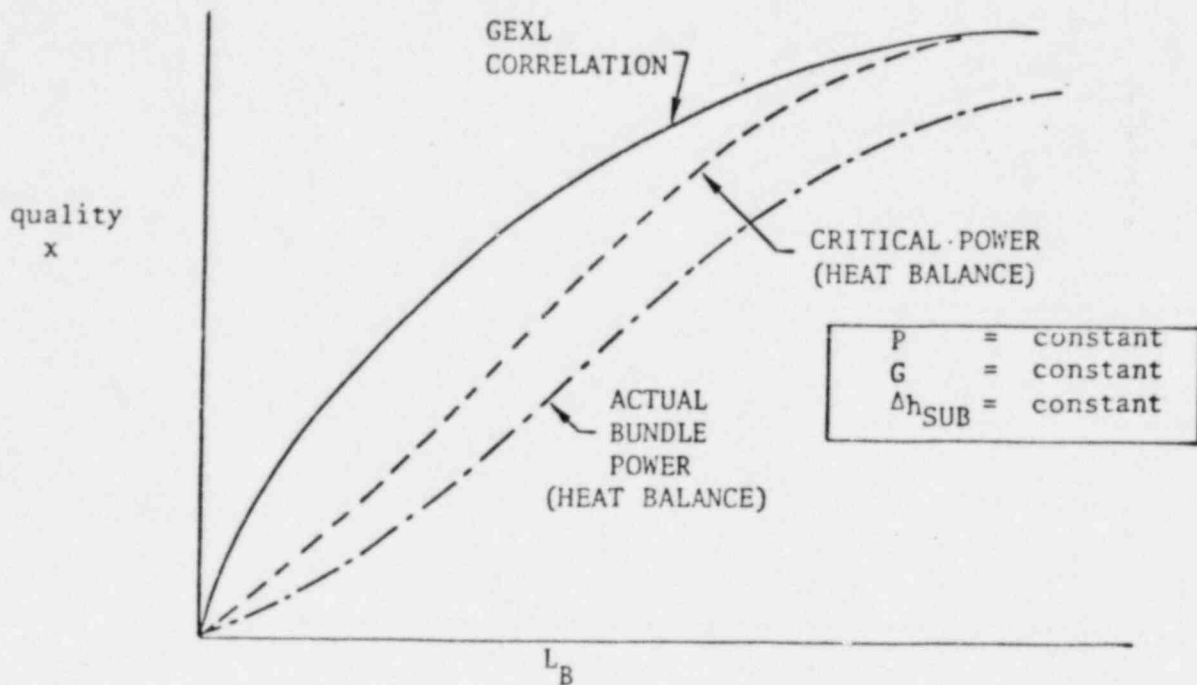
Figure 16

C. The GEXL Correlation

General Electric found an empirical correlation between critical quality and the bundle power required to produce a boiling transition in the bundle. This correlation is the GEXL correlation (General Electric Critical Quality, x_c , versus Boiling Length, L). GEXL is used to perform both steady state and transient analyses of the thermal behavior in a BWR. GEXL correlates the critical quality, x_c , to measurable quantities such as boiling length, coolant mass flow rate, reactor pressure, fuel element heated length, effective flow area, and local peaking factor. GEXL can accurately predict the critical quality as a function of each of these variables, while holding the other constant. The heat flux and bundle power required to produce critical quality can be obtained by performing a bundle heat balance. The bundle power required to produce critical quality (and, therefore, transition boiling) can then be found for any reactor condition.

D. Critical Power

The bundle power required to produce transition boiling in a channel is the critical power. On a graph of local quality versus boiling length, performing a stepwise heat balance of a channel will result in a curve of x versus L_B as in the curve on Figure 17, labeled "operating power (heat balance)". Using the same channel flow conditions, GEXL will produce a curve on this graph which is the critical quality versus boiling length. On Figure 17, the center curve is the "critical power (heat balance)" curve and it becomes tangent to (just touches) the "GEXL correlation" curve near the channel exit. This implies that for this condition, transition boiling occurs near the top of the fuel bundle. For other flow rates or axial power shapes, the critical bundle power may be higher or lower, and transition boiling may take place at a different boiling length. The important thing to remember is that any bundle power curve which touches the GEXL curve represents the critical power for that condition.



GEXL correlation and BWR heat balance curves

Figure 17

The ratio of the critical power to some operating power, like the one in Figure 17, is the critical power ratio, CPR. When the operating power equals the critical power, $CPR = 1.0$. For operating powers less than the critical power, CPR is greater than 1.0. The CPR must always be greater than 1.0. The minimum value that the CPR can have anywhere in the core (MCPR) is specified as a LCO in the Tech Specs. The OD-6 edit on the process computer gives MCPR data for the most limiting bundle in the core (Option 2) and lists the twelve most limiting bundles for each fuel type (Option 3). The LCO for MCPR is based on maintaining the MCPR greater than 1.07 for any analyzed operating transient. Remember that a MCPR greater than 1.0 ensures that transition boiling does not take place, and transition boiling causes rapid thermal cycling of the cladding and may lead to film boiling, which results in unacceptable cladding and fuel temperatures.

E. Summary

The bundle power required to produce a boiling transition can be accurately predicted by the GEXL correlation. This critical bundle power is the basis for the MCPR limit found in the Technical Specifications. Keeping the smallest value of CPR above the required Technical Specification limit ensures no transition boiling and, therefore, no fuel damage any analyzed operating transient.

VI LINEAR HEAT GENERATION RATE

The critical power ratio is used to prevent overheating of the fuel cladding due to deterioration of the heat transfer mechanism. To accomplish this, bundle power is never allowed to reach the value which would produce a boiling transition. Maintaining bundle power below the critical power does not always protect the fuel bundle from damage. To further protect the cladding from damage, other thermal limits, with different bases must be specified.

A. Linear Heat Generation Rate

The linear heat generation rate (LHGR) is the number of kilowatts of heat produced per foot of fuel rod length, kW/ft. The average LHGR (ALHGR) for an entire core can be computed, simply by dividing the core output by the total fuel rod length.

$$\text{ALHGR} = \frac{\text{Core Thermal Power}}{\text{Linear Ft. of Fuel Rod}}$$

For a BWR rated at 3293 MW_t with 764 bundles of 7x7 fuel, the total fuel rod length is:

$$764 \text{ bundles} \times \frac{49 \text{ pins}}{\text{bundle}} \times \frac{12 \text{ ft}}{\text{pin}} = 449\,232 \text{ ft.}$$

and the value of ALHGR is:

$$\frac{3\,293\,000 \text{ kW}}{449\,232 \text{ ft}} = 7.33 \frac{\text{kW}}{\text{ft}}$$

B. Peaking Factors

The average LHGR is the power that would be produced in every foot of fuel rod length if all fuel rods produced the same fraction of total power. The actual LHGR varies with radial position, axial position, and position

of a particular rod within a bundle. In addition, manufacturing tolerances can cause one rod to generate more heat than other rods in the same position. Peaking factors account for the non-uniformity of LHGR within the core and permit the determination of the maximum LHGR in the core. The fuel rod with the highest value of LHGR would be the hottest rod in the core.

The axial peaking factor, APF, accounts for the non-uniformity of the axial flux profile. The axial flux profile is determined for a given fuel cell (4 assemblies) by a TIP trace. The process computer uses TIP data to compute nodal powers for a given fuel assembly and obtain axial data for that fuel assembly. From the nodal powers for an assembly, the bundle power is found by summing the individual nodal powers. The axial peaking factor for a node is the nodal power divided by the average nodal power for the assembly. For example, if a node generates 0.30 MW_t , and the entire assembly is found to generate 4.73 MW_t , the average nodal power for the assembly is:

$$\text{Average Nodal Power} = \frac{4.73 \text{ MW}}{24 \text{ nodes}} = \frac{0.197 \text{ MW}}{\text{node}}$$

The axial peaking factor for the node under consideration is:

$$\text{Axial Peaking Factor} = \frac{\text{Nodal Power}}{\text{Ave. Nodal Power}} = \frac{0.30 \text{ MW}}{0.197 \text{ MW}} = 1.522$$

The axial peaking factors for every node in the core are stored for later use.

The radial peaking factor, RPF, also called the relative peaking factor, is computed for each fuel bundle, rather than every node. The bundle power, PBUN is computed for each fuel bundle using the TIP-based nodal powers. The average bundle power, PBUNAV is simply the core heat output divided by the number of bundles. Using the reference reactor, the average bundle power is:

$$\text{PBUNAV} = \frac{3293 \text{ MW}_t}{764 \text{ bundles}} = \frac{4.31 \text{ MW}_t}{\text{bundle}}$$

The radial peaking factor for this assembly is:

$$\text{RFP} = \frac{\text{Bundle power of Interest}}{\text{PBUNAV}} = \frac{4.73}{4.31} = 1.097$$

The radial peaking factor determines the highest power fuel assembly in the core, and the axial peaking factor determines the highest power node in any given assembly. The local peaking factor, LPF, determines the highest power rod at a node. The local peaking factor is a function of fuel type, exposure, void fraction, control rod configuration and surrounding fuel types. The local peaking factor can be found from graphs or read from an OD-6 edit. Assume for the node in the current example that the LPF is 1.118.

The total peaking factor, TPF, is the product of the individual peaking factors:

$$TPF = APF \times RPF \times LPF$$

The hottest node in the core is the node with the highest value of TPF:

$$TPF_{\max} = (APF \times RPF \times LPF)_{\max}$$

It should be noted that TPF_{\max} is not equal to $APF_{\max} \times RPF_{\max} \times LPF_{\max}$. This implies that the highest peaking factors all occur at the same node, which is not true. To find the maximum value of LHGR in the core, just multiply the average LHGR by TPF_{\max} :

$$\text{Max LHGR} = \text{Ave. LHGR} \times TPF_{\max}$$

For the current example, the LHGR at the node of interest is the node TPF times the average LHGR:

$$\text{Node LHGR} = \text{Node TPF} \times \text{Ave LHGR};$$

$$\text{Node LHGR} = 1.522 \times 1.097 \times 1.118 \times 7.33 \frac{\text{kW}}{\text{ft}} = 13.68 \frac{\text{kW}}{\text{ft}}$$

For comparison, look at the LHGR for this same node if the fuel assembly is 8 x 8 R. The value for PBUNAV is still $4.31 \frac{\text{MW}}{\text{t}}$, but the number of linear feet of fuel rod per bundle is:

$$\text{fuel length} = \frac{62 \text{ pins}}{\text{bundle}} \times \frac{12 \text{ ft}}{\text{pin}} = \frac{775 \text{ ft}}{\text{bundle}}$$

Then the average LHGR for this core with 8 x 8 R fuel is:

$$\text{Ave LHGR} = \frac{\text{PBUNAV}}{\text{fuel rod length}} = \frac{4.3 \text{ MW}_t/\text{bundle}}{775 \text{ ft}/\text{bundle}} = \frac{5.5 \text{ kW}}{\text{ft}}$$

For a given core size and total heat output, the LHGR for 8 x 8 R fuel is about 20% lower than for 7 x 7 fuel.

Linear heat generation rate tells how much energy is produced in a particular spot in a fuel rod. The lesson on heat transfer showed that the temperature gradient is proportional to the heat transfer rate. The reason for limiting the LHGR is that fuel and cladding temperatures will be too high if the LHGR limit is exceeded. The maximum LHGR limit is imposed to prevent cladding failure and leakage of fission products into the coolant.

At low power, the UO_2 fuel pellets look like those in Figure 18 (a). As reactor power and LHGR are increased, the pellets begin to distort as in Figure 18 (b). If the pellet is overpowered, that is the LHGR limit is exceeded, the edges of the pellet will push on the cladding hard enough to stretch the cladding as in Figure 18 (c). The LHGR limit is based on not exceeding 1% plastic strain on the cladding.

$$\text{At PBAPS the LCO for LHGR is: } \text{LHGR} \leq \text{LGHR}_d \left[1 - \left(\frac{\Delta P}{P} \right)_{\max} \left(\frac{L}{L_T} \right) \right]$$

where, LGHR_d = Design LHGR

= 18.5 kW/ft for 7 x 7 fuel

13.4 kW/ft for 8 x 8, 8 x 8 R, and 8 x 8 LTA fuel

$\left(\frac{\Delta P}{P} \right)_{\max}$ = Maximum power spiking penalty

= 0.026 for 7 x 7 fuel

= 0.022 for 8 x 8, 8 x 8 R, and 8 x 8 LTA fuel

L_T = Total core length

= 12 ft for 7 x 7 and 8 x 8 fuel

= 12.5 ft for 8 x 8 R and 8 x 8 LTA fuel

L = Axial position above bottom of core

For a 7 x 7 fuel bundle, the LHGR limit at node 4 (2 ft above core bottom) would be:

$$\frac{18.5 \text{ kW}}{\text{ft}} \left[1 - (0.026) \left(\frac{2}{12} \right) \right] = 18.5 [0.9957] = \frac{18.42 \text{ kW}}{\text{ft}}$$

For a 8 x 8 and 8 x 8 R fuel, the spiking penalty is included in LGHR_d , so that the LHGR limit is always 13.4 $\frac{\text{kW}}{\text{ft}}$.

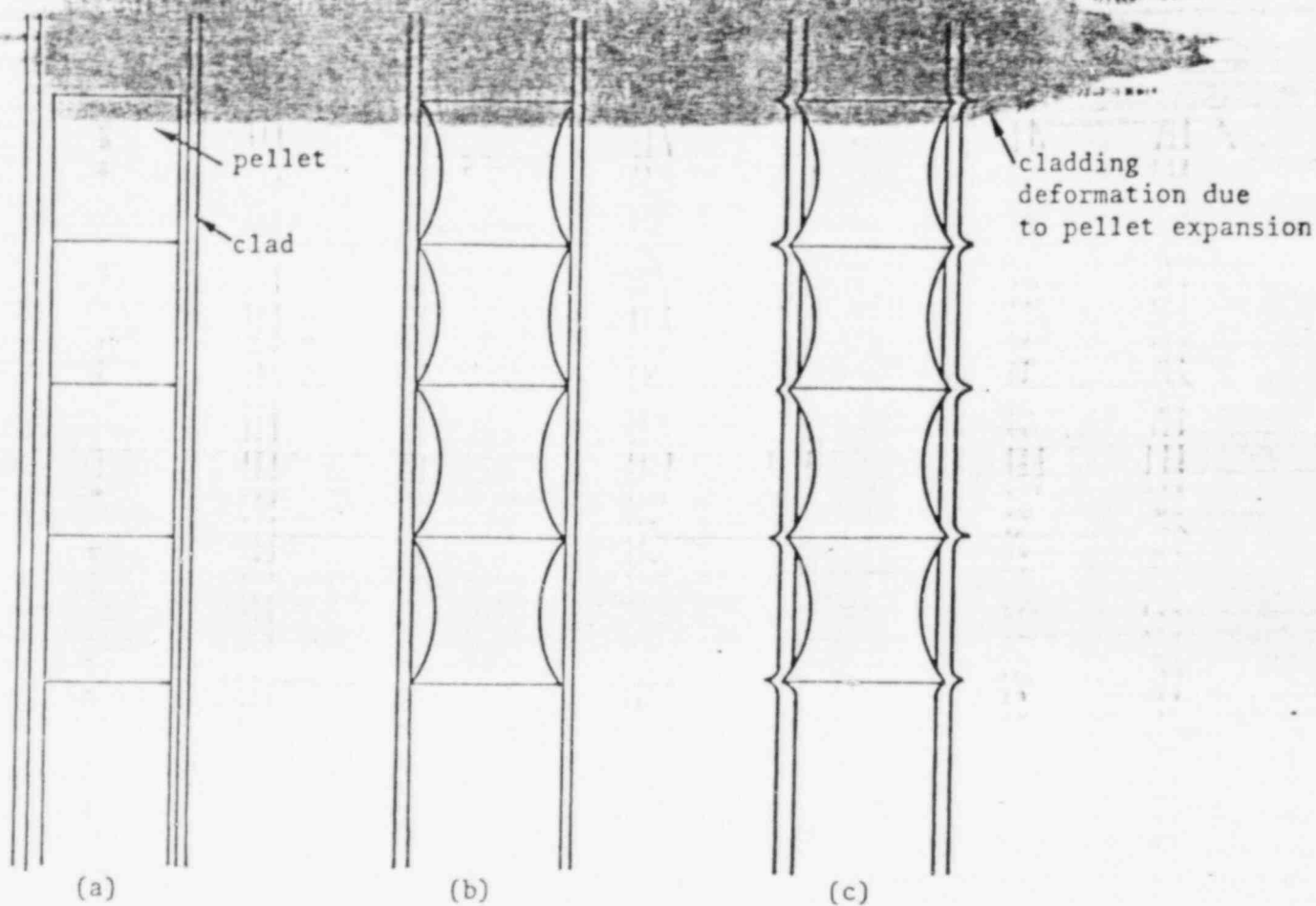


Figure 18

C. Summary

The LCO for LHGR insures that the plastic strain limit on the fuel cladding will not be exceeded. Notice that the basis for the LHGR limit is entirely different than the basis for the MCPR limit. The MCPR limit guards against deterioration of the heat transfer mechanism - nucleate bulk boiling. The LHGR limit guards against cladding damage due to excessive power output in any node of the core.

VII. AVERAGE PLANAR LHGR

In a BWR, heat removal rate, rather than fission rate, limits the maximum power of the reactor. There is no theoretical limit to the fission rate in a reactor. Instead, the ability of reactor materials to perform in high temperature environments limits the amount of heat the reactor can be allowed to produce. The two previous lessons discussed the CPR limit, which guards against a boiling transition, and the LHGR limit, which prevents damage to the fuel cladding. Operating the reactor at a power which stays within both of these limits insures that no fuel or cladding damage will occur if an analyzed transient occurs from an allowed reactor operating condition. The last thermal limit to be discussed guards against fuel and cladding damage during a design basis loss of coolant accident (LOCA).

A. APLHGR

The linear heat generation rate limits the power in a particular rod. The LHGR limit basically protects the hottest rod in the core from cladding damage during operating transients. The hottest rod in the core could be in either of the following general locations: It could be in a low-power bundle with a high local peaking factor; or it could be in a high-power bundle with a low local peaking factor. The case of a high-power bundle with a high local peaking factor is not a significantly different problem.

In the event of a LOCA, the fuel would dry out rather quickly, and the primary heat transfer mechanism prior to rewetting would be thermal radiation. Title 10CFR50.46 specifies that the peak cladding temperature during a design basis LOCA shall not exceed 2200°F. If a fuel bundle (or node) dries out during a LOCA, the edge rods could dissipate heat by radiation more easily than the central rods. The edge rods can radiate heat away from the fuel bundle, while the central rods radiate much of their heat to other central rods. Even though the edge rods, and the corner rods in particular, have higher local peaking factors, the central rods are more likely to exceed the 2200°F limit during a LOCA.

It is, therefore, the average heat generation rate in a node, rather than the peak heat generation rate in a rod, which more accurately reflects the heatup characteristics of the fuel during a LOCA. The average planar linear heat generation rate (APLHGR) is the average LHGR across a particular node. The value of APLHGR is just the heat deposited in the fuel divided by the number of feet of fuel pin in the node. The APLHGR is calculated by the process computer and printed as the maximum average planar linear heat generation rate (MAPLHGR) for the node, using the following formula:

$$\text{MAPLHGR} = \frac{P_n \times (1-f_c)}{N \times L}$$

where: P_n = nodal power, kW

f_c = fraction of power deposited directly in coolant, decimal

N = number of fuel pins per bundle

L = length of node, ft.

For a node generating 0.3177 MW in an 8 x 8 R fuel bundle, with 6% of the power deposited directly in the coolant,

$$\text{MAPLHGR} = \frac{317.7 \text{ kW} (1-0.06)}{62 \times 0.5 \text{ ft}} = 9.63 \frac{\text{kW}}{\text{ft}}$$

The limiting value of MAPLHGR is found in the Technical Specifications and is a function of fuel type and exposure. The ratio of the actual value of MAPLHGR to the LCO value of MAPLHGR is the maximum average planar ratio (MAPRAT). The value of MAPRAT must always be less than 1.0.

$$\text{MAPRAT} = \frac{[(\text{MAPLHGR})_{\text{actual}}]}{[(\text{MAPLHGR})_{\text{LCO}}]_{\text{max}}} \leq 1.0$$

Maintaining the MAPRAT less than 1.0 ensures that the peak cladding temperature will not exceed 2200°F during a design basis LOCA.

Summary

A BWR has three separate thermal limits, and each is designed to protect the reactor against a different heat-transfer-related problem. The MCFR limit maintains nucleate boiling and prevents a boiling transition. The LHGR limit minimizes cladding strain caused by excessive heat generation in the fuel. The MAPLHGR limit ensures that the peak fuel cladding temperature will not exceed 2200°F during a design basis LOCA.

Although these three limits are different in many ways, they are all approached at about the same rate during normal reactor operation. At full power, all three of these thermal limits are about 80-90% of the LCO.

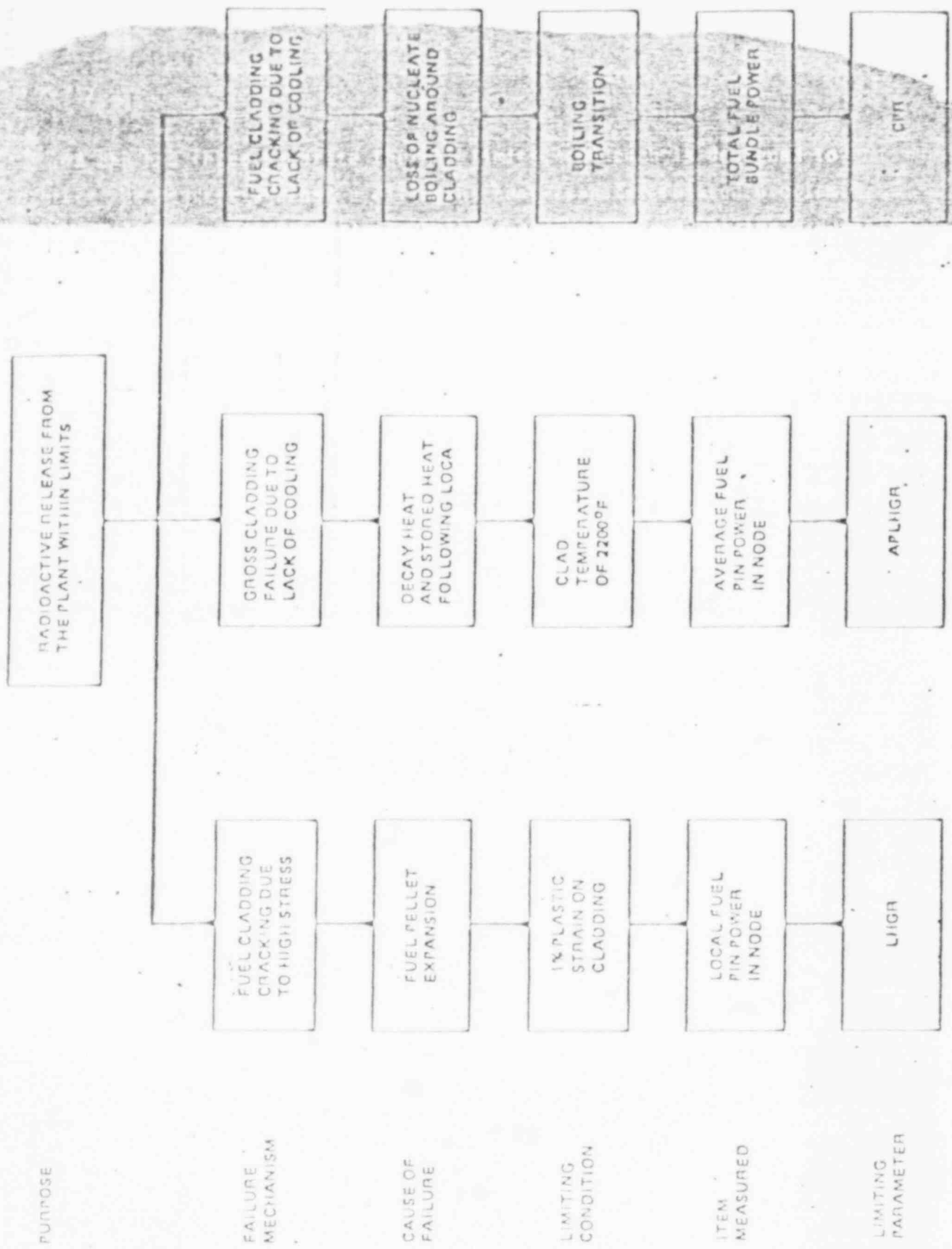


Figure 9-1. Thermal Limits

HEAT SOURCES

I. TWO HEAT SOURCES AFTER SCRAM

A. SENSIBLE HEAT

B. DECAY HEAT

CORE COOLING MECHANISMS

I. THREE HEAT TRANSFER MODES

II. BWR COOLING MECHANISMS

A. FLOOD

B. SPRAY

C. STEAM

D. BOIL

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

FUEL & CORE DAMAGE THRESHOLD

I. TECH SPEC SAFETY LIMITS

A. MCPR

B. PRESSURE

C. RVWL

II. FUEL DAMAGE THRESHOLD

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

RECOGNITION OF INADEQUATE CORE COOLING CONDITIONS

I Introduction

A. Causes of Inadequate core cooling

1. LOCA - Degraded ECCS

2. SORV - Degraded ECCS

3. Channel blockage

B. Attempt to recognize adequate cooling

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

II. Diverse methods of detecting adequate core cooling

A. Primary method to assess adequate cooling - Reactor Water level

1. Natural circulation is an inherent feature

2. Adequate Water Level = adequate cooling

a) Normal level or above

b) 2/3 coverage

c) Above core plate

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

B. Secondary assurance of adequate core cooling

1. One ECC system operating

III Water Level Instrumentation

A. Confirms adequate core cooling and detects an approach to inadequate core cooling.

1. All mode of normal operation

2. Anticipated transients

3. Credible accident conditions

B. Measures using differential pressure devices.

1. Senses mass of fluid above the sensing tap.

C. Ranges of Level Indication during accident conditions

1. Narrow range

2. Wide range

3. Fuel zone

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

a. At higher than calibrated pressure - reading is less than actual.

b. Natural circulation flow same as above.

c. LPCI flow in BWR's 3&4 gives high reading

1) confirms injection flow

D. Water level inside shroud versus outside the shroud

1 Higher than indicated

a) except when ECCS is injecting

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

IV Indication of Operational ECCS

A. If ECCS functions

1. No core uncover for small breaks
2. Little or no clad heatup for large breaks

B. ECCS Indications

1. Valve Positions
 - a) flow path
 - b) most valves outside primary containment

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

2 Positive Flow

- a) Pump(s) running

3. Discharge Pressure

- a) Piping integrity

IV Indication of Superheated Steam

- a) No superheat core covered

- b) Difficult to do

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

1) fast responding TC on SRV tail pipes

2) where to measure saturation temperature for vessel

Summary

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

INCORE INSTRUMENTATION

I. 4 BASIC SYSTEMS

II. SRM RESPONSE TO ICC

A. LOW LEVELS - UNCOVERED

B. LEVEL JUMP

III. LIMITATIONS

A. 5 FT BELOW TAF

B. FULL UNCOVERY TOO LOW

C. DRIVE MOTORS NOT 1E

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

D. DECREASING COUNT RATE

IV. DETERMINING EXTENT OF CORE DAMAGE

A. TIP MOVEMENT

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

REACTOR WATER LEVEL INSTRUMENTATION

INTRODUCTION

Objectives:

Simple DP cell (Fig. 1)

Pressure compensation (Fig. 2)

Variable & Reference leg theory, etc.

Design criteria: Wide range vs. accuracy

Range coverage (Fig. 3)

INSTRUMENT DESCRIPTION

YARWAY (Fig. 4)

Temperature compensation

DP cell outside drywell

Local & remote indication

Safety functions

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

COLD REF LEG (Fig. 5)

No heating of RL (ambient)

DP RPV water level

Accurate when $X = Y = Z \pm 1$ ft.

Location of instrument taps (Fig. 6)

LEVEL READOUT & TRIP FUNCTIONS

Review level assignments (Fig. 7)

INSTRUMENT RESPONSE

Drywell Temperature Effects due to

Normal ΔT of Startup

Error - f (DW to calib ΔT , Tap elevation, Type of RL)

COLD REF. LEG (Figures 5&3)

If $X = Y = Z \pm 1$ ft, no error

If $X = Y = Z$, 2.2% (5") error @ 60°F ΔT w

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

Effect of ΔT outside DW < 3% error

YARWAY

Error = 2.4% (<5") @ 60°F ΔT dw

Trips occur at $L(\text{act}) < L(\text{ind})$

Effect of ΔT outside DW is negligible

SUMMARY

Error < 3% for normal ΔT in/out of DW

Errors not of operational significance

Drywell Temperature Effects during Upset & LOCA

RX PRESSURIZED

Error $\propto \Delta T$ dw, RL thermal time constant, & length of RL.

Yarway & Cold RL where $X / Z \rightarrow L(\text{ind}) > L(\text{act})$

Measurement error in Fig. 9

No error for Cold RL where $X = Y = Z$

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

Lower Yarway tap can uncover (Fig. 10) and prevent trips and mislead operator.

REACTOR DEPRESSURIZED

Cold RL Thermal Response (Figures 11-18)

Transient boiling

Flashing

These require $T_{dw} > T_{sat}$ in RPV

$P_R > 200$ psig assures no flashing or boiloff

Worst flash \rightarrow 20% error

Flash + boiloff \rightarrow 30% error 1 hour from ADS

Refill RPV to steam lines restores accuracy

Yarway Thermal Response (Figures 19 & 20)

Flash \rightarrow 15% error

Boiloff \rightarrow 35% error 1 hour from ADS

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

Faster boiloff than Cold RL

No degrading of ECCS initiation due to errors

Drywell Temperature Effects With Loss of Drywell Coolers

Peak Tdw < Tsat of RPV

No flashing or boiling occurs

CONCLUSIONS

When pressurized, high Tdw may cause error (indicated > actual) in Yarway & Cold RL instruments (except X = Y = Z Cold RL).

When depressurized, high Tdw may cause error (indicated > actual) due to flashing and subsequent boiloff. This may cause a reduction in redundancy in safety system initiations and mislead the operator.

Lower initiation levels of safety systems caused by high Tdw are inconsequential from safety standpoint.

PRESSURE & JET PUMP FLOW EFFECTS OFF CALIBRATION

$P > P_{\text{calib.}} \rightarrow \text{Lind} < \text{Lact by } 80'' \text{ max @ TAF}$

Nat. Circ. $\rightarrow \text{Lind} < \text{Lact by "several inches"}$.

These inaccuracies are conservative WRT operator response.

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

LPCI flow → higher press in diffuser → Lind > Lact however, is indication of ECC flow

ALTERNATE MEANS OF LEVEL DETERMINATION

Superheated steam = core uncover, but determination depends on flow thru S/RV's ∴ limitations.

N flux anomalously low for uncovered core.

∠ flux lower but not significantly
∴ not useful.

Refilling steam lines & watch T of S/RV tailpipes.

Install (or use installed) pressure gauges above and below expected level and apply thumb rule.

PRESSURE INSTRUMENTATION

Several reliable sources of P indic

Normal control room pressure indicators and controllers

Annunciators associated w/press switches

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

Steam supply press to RCIC/ISO Cond'r

Recirc pump seal press

ECCS pump discharge press

TEMPERATURE INSTRUMENTATION

Little value in determination of internal R.P.V. temperature, but temperature is readily available by P VS T in saturated conditions.

RADIOCHEMICAL RESULTS FROM A DEGRADED CORE

I BWR Safety Features As Radioactivity Barriers

A. Barriers

B. Characteristics of Nuclides

That effect release

II Fission Product Source Terms

A. Definition

B. Sources during normal operation

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

C. Types of sources during normal operation

1. Recoil

2. Pin-Hole Defects

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

D. Analyzing patterns

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

D. Distribution during abnormal conditions

1. Split Cladding Defects

2. Larger Cladding Defects

3. Accident Conditions

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

E. Fission product behavior trends

F. Noble Gases

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

G. Reactor Water Solubles

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

H. Reactor Water Insolubles

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

III Radioactive Sources under degraded conditions

A. Not defined yet

B. Hypothetical bases

C. NUREG-0578

D. TMI-2

E. WASH-1400 Table 12

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

IV Transport and Deposition of Fission Products in BWR Containment System

A. Release dependent on the accident sequence

B. Degraded core accident

C. Removal Mechanisms

1. Noble gases

2. Elemental Iodine

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

3. Iodide

4. Aerosol Particles

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

V. Fission Product Analysis

A. Preparations are being made

B. High Activity measurement facility

C. Gas Sample Analysis

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

D. Liquid Sample Analysis

E. In line conductivity

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

DEGRADED CORE CORROSION EFFECTS

EXTENDED IMMERSION OF PRIMARY WATER ON BWR STRUCTURAL MATERIALS.

PARTIALLY FLOODED CONTAINMENT FLOOR WITH T1 $\approx 300^{\circ}\text{F}$

I. BACKGROUND

II. GENERAL CORROSION

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

III. GALVANIC CORROSION

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

IV. CREVICE CORROSION

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

V. PITTING CORROSION

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

VI. INTERGRANULAR ATTACK

VII. STRESS CORROSION CRACKING

VIII. CORROSION FATIGUE

IX. EROSION CORROSION

X. HYDROGEN DAMAGE

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

DETERMINING DOSE RATES INSIDE CONTAINMENT FROM DETECTORS
LOCATED OUTSIDE CONTAINMENT

I. METHODS

II. EXTERNAL MEASUREMENTS

A. $I = B(\mu x) I_0 e^{-\mu x}$

B. INTERPRETATION PROBLEMS

C. MITIGATION

III. NUREG 0578

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

GAS GENERATION

I. INTRODUCTION

II. SOURCES

A. H_2

1. $Zr-H_2O$

2. STEEL- H_2O

3. RADIOLYSIS

B. O_2 -RADIOLYSIS

C. FISSION PRODUCT GASES

III. TYPES OF COMBUSTION

A. BURN

B. RECOMBINATION

C. DEFLAGRATION

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

D. DETONATION

E. AUTO-IGNITION

IV. CHARACTERISTICS

A. MIXTURE (Table I)

B. COMPOSITION (Figure 1)

C. VESSEL

D. PRESSURE

V. MITIGATION OF H_2 COMBUSTION

A. $H_2 < 4$ VOL. %

B. $O_2 < 5$ VOL. %

C. HALON 1301

D. REMOVE H_2

E. BURN H_2

F. IGNITE H_2

G. VENT

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

VI. MITIGATION OF O_2 LIMITED COMBUSTION

A. RADIOLYSIS (INERT)

B. RECOMBINE

C. BURN

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

D. VENT

E. MITIGATION IN INERT

F. MITIGATION IN AIR

VII. MITIGATION OF NOBLE GAS

A. VOLUME PRODUCED

B. DILUTE ATMOSPHERE

Designed to provide a fundamental background in the principles of fluid flow and associated equipment needed for nuclear power plant operation and for pursuit of advanced or specialized training.

FLUID STATICS including units of measurement, pressure and pressure relationships, pressure measuring devices, density, specific volume and specific gravity;

FLUID FLOW including the continuity equation, the general energy equation, ideal (frictionless) fluid flow, laminar and turbulent flow, actual fluid flow, fluid friction, viscosity, system losses in head and pressure terms, fluid hammer, and flow measurement;

PUMPS including types and applications, pump work, efficiency, cavitation, Net Positive Suction Head, and pump operating characteristics;

TURBINES including basic principles of operation, nozzles and air enectors;

PIPING AND VALVES including types and applications, sizes and standard classification;

REVIEW AND EXAMINATION

Approximate duration - one week

June 22 to June 26

Designed to provide a fundamental background in the principles of heat transfer and thermodynamics needed for nuclear power plant operation and for pursuit of advanced or specialized training.

THERMODYNAMICS including properties, temperature, pressure, heat, fusion, evaporation, density, specific volume, internal energy, flow energy, First Law of Thermodynamics, General Energy Equation, enthalpy, work, system boundaries, Second Law of Thermodynamics, entropy and ideal gas law;

STEAM AND VAPOR PROPERTIES including steam tables, saturation temperature and pressure, quality, specific heat, superheated steam, TS diagram and Mollier diagram;

THERMODYNAMIC CYCLES including pressure-specific volume diagram, temperature-entropy diagram, nuclear boilers and steam generators, steam turbine, friction, efficiency, condenser, heat exchanger, pumps, cycle diagrams, work, condenser heat removal, reactor heat input, feedwater heating and cycle analysis;

HEAT TRANSFER including conduction, convection, radiation, heat exchangers, evaporative cooling and calorimetrics;

BOILING HEAT TRANSFER including convection, nucleate boiling, DNB, film boiling, boiling water curve and two phase flow;

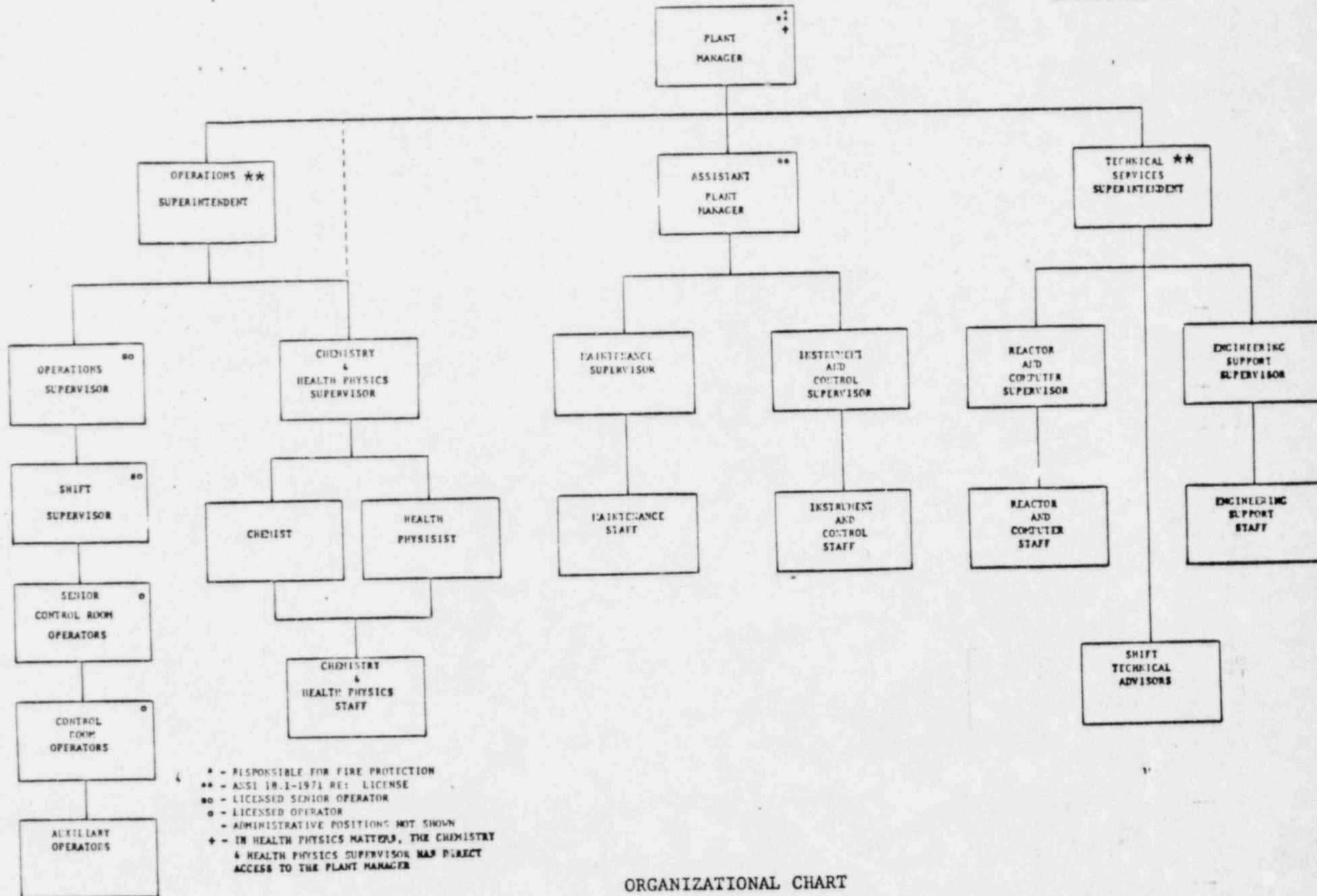
THERMAL STRESS including application to thickwalled vessels;

REVIEW AND EXAMINATION

Approximate duration - two weeks.

June 29 to July 3

July 6 to July 10



ORGANIZATIONAL CHART

FIGURE 6.1.2