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THERMAL HYDRAULICS OF ACCELERATOR BREEDER SYSTEMS FOR REGENERATION OF REACTOR FUEL ASSEMBLIES*

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ABSTRACT

Accelerator breeder systems have been designed [1,2] that can pre-enrich or regenerate fissile material in situ in fabricated fuel assemblies, with the assemblies subsequently used directly in existing reactor types (e.g., LWR and CANDU). Such systems have the potential to reduce uranium lifetime requirements for existing reactors by a factor of four or more, and to considerably increase the overall burnup of fuel assemblies, while operating in a non-reprocessing mode to minimize proliferation concerns. The thermal hydraulic characteristics of the LAFR [Linear Accelerator Fuel Regenerator] system are a very important determinant of overall performance. In the LAFR concept [1], the primary 1.5 GeV proton beam interacts with falling jets of liquid Pb-Bi, producing spallation and evaporation neutrons ($E_A \sim 5$MeV), which generate fissile material in a surrounding blanket of pressure tubes containing fuel assemblies. To maintain high fissile material production rate, as well as to minimize power peaking, the coolant for these fuel assemblies should not significantly moderate neutrons from the primary Pb-Bi target. In addition, fuel assemblies must be compatible with both the LAFR coolant and water, since the latter is the coolant in LWR and CANDU.

These restrictions appear to dictate that two-phase mixtures (steam and liquid) of either D_2O or H_2O be the primary coolant option for a LAFR. Thermal-hydraulic analyses have been carried out to determine desirable operating ranges for these coolants in a LAFR. The following limits are taken: 700°F maximum clad temperature on fuel elements (oxidation limits); ≤20% equivalent liquid density for two-phase mixtures; coolant temperature range corresponds to a thermal power efficiency ≥20%; and pressure drop of ≤10% of base pressure.

The following conclusions are obtained with regard to the thermal-hydraulic behavior of the LAFR for PWR and CANDU fuel.

1. Two-phase flow is a feasible coolant option for fuel element heat fluxes up to 1 x PWR (or CANDU) average value, which is the maximum design value for a LAFR.

2. Two-phase flow pressure drops are low (typically 10-30 psi) and film temperature drops very low (typically ~10°F) for PWR fuel, with inlet velocity

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range (50 to 75 ft/sec). A somewhat higher inlet velocity range (75 to 100 ft/sec) and pressure drop (50-100 psi) is necessary for CANDU fuel, however, to prevent dry out.

I. INTRODUCTION

The energy crisis seems to stay. The oil embargo made the American public conscious of the widening gap between energy consumption and domestic production, and of our unaccustomed but growing dependence on foreign supplies. It is our judgment that nuclear energy is an important element of our total energy system and will become even more so in the future.

However, nuclear energy as produced in Light Water Reactors (LWR) is wasteful in the long run, and at the rate of use presently projected, our economically recoverable uranium resources will be gone in a few decades. The development and commercialization of the Liquid Metal Fast Breeder Reactor (LMFBR) has been slower and costlier than first estimated. The reprocessing facilities required for the LMFBR system are now seen as presenting a risk of nuclear weapons proliferation, and the future commercialization of fast breeder reactors in this country is in doubt.

For these reasons, a program has been initiated by DOE to evaluate fuel cycles and alternative nuclear energy systems which could be made more proliferation resistant and which at the same time could help to stretch the uranium resources. This program is conducted under DOE's Non-Proliferation Alternative Systems Assessment Program (NASAP), organized for that purpose.

Under NASAP funding, the Department of Nuclear Energy at Brookhaven National Laboratory has investigated the use of particle accelerators for producing fissile material in conjunction with proliferation resistant systems. During this study, a linear accelerator fuel regenerator (LAFR) system was considered.

The concept of LAFR involves the use of a high energy proton or deuteron beam from a linear accelerator impinging on a primary Pb-Bi target that is surrounded by a lattice blanket of metal-clad rods of fertile-fissile material cooled by an appropriate fluid. The system is characterized by maximizing the total production rate of fissile material in the lattice as well as minimizing the radial variation of fissile fuel production. The fertile-fissile elements of interest are natural and depleted uranium oxides, and thorium metal. For best performance, the blanket coolant should moderate the source neutrons to the minimum possible extent, thus hardening the neutron energy spectrum within the lattice. The lattice consists of fabricated fertile fuel assemblies whose reactivity would be increased through irradiation to the desired level for subsequent burn up in a separate power reactor. After burn up in the reactor, the fuel elements could be recycled through the accelerator-regenerator for reenrichment, and then used again in the power reactor. This process would be repeated until fission product buildup and radiation damage to cladding was excessive. No reprocessing would be required.

In the course of this study, particular attention was given to engineering feasibility with regard to the thermal hydraulics of the system. The results of preliminary investigation including neutronics and system economics [1] indicate promising production of fissile fuel without reprocessing is technically feasible without major departures from existing technology. The existing LWR nuclear power economy can be retained with almost no modification; uranium resources can be extended substantially over that of the throw-away LWR cycle; there would be no need for reprocessing; even greater fuel-stretching benefits can be achieved in the U-233 cycle; in conjunction with HWR's and/or with reprocessing, the system can be developed to maturity in a reasonable length of time (∿20 years) with predictable performance and reasonable R&D.

II. DESCRIPTION OF LAFR

The conceptual design of a linear accelerator fissile fuel production facility consisting of a 1.5-GeV proton linear accelerator using a flowing liquid lead-bismuth primary target surrounded by a blanket or secondary target containing fertile uranium material was described in a Brookhaven Report [1]. The LAFR consists of two distinct parts: the 1.5-GeV linear accelerator or LINAC, and the target-blanket assembly with its power generation system. Figure 1 shows the design principle of the target. It consists of a primary lead-bismuth target stopping the entire proton beam, and a secondary target (blanket) containing the fertile uranium material.

The primary target consists of multiple falling liquid lead-bismuth columns or jets so arranged as to provide for an evenly distributed neutron source. This is accomplished by providing a variable target density with respect to target depth, and by stepping the target to intercept different parts of the proton beam at different points within the neutron source area. The liquid lead-bismuth jets operate in the vacuum of the containment vessel, which is directly connected to the accelerator via the beam transport system. No window material is required between the accelerator and the target.

The uranium fertile material in the blanket consists of PWR or HWR fuel assemblies located in individual pressure-tubes. These surround the primary target neutron source in a manner consistent with minimization of neutron leakage, parasitic absorption, and mechanical design constraints. The assemblies of fuel, coolant, and moderator, are designed to remain highly subcritical under any circumstance. During an irradiation cycle, fuel shuffling is required to achieve good control of reactivity buildup. For this reason, the blanket will need instrumentation to monitor the fuel being irradiated.

Neutron multiplication occurs in the blanket, generating thermal energy. This energy, as well as that generated by the proton beam in the lead-bismuth target, is used to generate electric power by a conventional steam-turbine system.

The safety considerations applicable to the target are the same as those for a PWR, with the notable exception that the target of the fuel regenerator is subcritical. However, the entire target-blanket and primary coolant systems are housed in a containment building similar to that of a conventional PWR, with all the safety features of a PWR.

The LAFR would produce fissile fuel in fabricated fuel rods, by means of an external source of neutrons. The LAFR could stretch the nuclear fuel resource by a factor of $\sim$4 or more without resorting to reprocessing, thus providing a long term supply of fuel to the LWR power system. Subsequent use of the LAFR with protected reprocessing could stretch the fuel supply by a factor of nearly 200 over that estimated from the U-235 availability alone. It would achieve the same purpose as the FBR but with the notable exception that the LAFR can also efficiently produce U-233 from thorium, thus opening up use of the thorium fuel resource. The LAFR is not a power reactor nor a breeder: it is a fuel generator. It would avoid the disadvantage of severely depleting the natural uranium resource which is the case with diffusion plants. One LAFR could support three or more LWR's. Further, the fuel regenerator would not impose new technology on the utility company, which would continue to operate LWR's. The characteristics of this system with respect to the neutron spallation-evaporation process, neutron multiplication and target blanket calculations are given in references [1] and [2]. A schematic chart for a LAFR system is shown in Figure 2.

III. BLANKET AND TARGET DESIGN

Figure 3 shows an overall view of the target-blanket configuration used for enriching or regenerating PWR fuel assemblies. The primary proton beam, entering from the beam transport region at the left of Figure 3 strikes a series of liquid metal Pb-Bi columns that fall from the jet spray nozzles at the top of the target-blanket assembly. The proton beam, initially circular in shape, is converted into a narrow ellipse by the final set of beam transport magnets. This elliptical beam enters a thin vertical slot in the target-blanket assembly and interacts with the liquid Pb-Bi columns. Instead of defocusing the beam in this fashion, we are also considering the possibility of sweeping the beam vertically to cover the entire target area.

A practical target-blanket assembly has to satisfy three fundamental constraints:

1. The power density in the blanket fuel assemblies must be compatible with available cooling methods.

2. The neutron source current at the target-blanket interface should be as uniform as possible. This minimizes power peaking and variations in fissile material build-up.

3. The neutron leakage from the target-blanket assembly should be small, so that leakage losses do not significantly reduce the fissile fuel production rate.

Because of the short range of protons in Pb-Bi (~90 cm at 1.5 GeV), a Pb-Bi target at normal density would generate excessively high blanket power densities in the first part of the target unless a very large target area were used. This would then cause a very large neutron leakage in the backward direction, as well as a very non-uniform neutron source distribution. The maximum neutron current from a Pb-Bi target consistent with good cooling capability in the blanket is on the order of 10^{14} n/cm^2 sec. For the example of 450 MW of beam power and a neutron yield of 45 per proton at 1.5 GeV, the surface area of the corresponding target would be 100 m^2. A single disk Pb-Bi target of 100 m^2 area and 90 cm thick would not be feasible, because of the high neutron leakage and non-uniformity of the source.

A practical Pb-Bi target-blanket assembly thus requires that the interaction distance of the primary protons in the target be much greater than the stopping distance in liquid Pb-Bi. This is possible if the effective Pb-Bi density along the target path is much less than that of the liquid. Figure 4 shows a cross sectional view of the target-blanket assembly with the Pb-Bi liquid colums. The effective density of the liquid columns increases with distance along the beam path. This spatial distribution compensates for the decrease in neutron production rate as proton energy is lost by interactions with the Pb-Bi columns. In addition to varying the spatial density of the Pb-Bi colums along the proton beam path, the Pb-Bi colums can be "stepped" forming a wedge or triangle partern to intercept successive portions of the primary beam. These two techniques, used either in conjunction or separately, should allow one to shape the neutron source function along the beam path to any desired level.

The PWR fuel assemblies are contained in 12 inch ID Zircaloy pressure-tubes as shown in Figure 5. The pressure-tubes are headered at the target-blanket assembly. Fuel assemblies are loaded and unloaded from the top of the pressure-tubes into fuel handling casks. The pressure-tubes project through the tube sheet at the top of the assembly, and are welded to the tube sheet to provide a vacuum tight seal. Plugs at the top of each pressure-tube can be removed for the insertion or withdrawal of fuel assemblies.

Coolant flows up through the crescent-shaped regions between the square PWR fuel assembly and the round pressure-tubes, and then down through the assembly, cooling the fuel elements. The pressure drop of the coolant acts to keep the fuel assembly centered on the guide structure in the bottom of the pressure-tube. A shroud structure is necessary to prevent short circulating of the coolant flow from the inlet flow regions to a PWR fuel assembly, since the assembly has no external box around it. Such a shroud is shown in the cross sectional view of Figure 5 with attached ribs that support it in the pressure-tube.

Since there is no window between the accelerator and the target-blanket assembly, the pressure-tubes must be vacuum tight and there must be a vacuum envelope surrounding the pressure-tube assembly. This envelope will in effect be a relatively large vacuum tank, on the order of 10 meters in length, $\sim$4 meters high and $\sim$3 meters wide, with the external pressure nominally at 1 atm. Because of the relatively thick structure required to carry external pressure over large flat areas, the reflector and thermal shield will probably have to be placed inside the vacuum enclosure, rather than outside as shown in Figure 4. Design of this portion of the target-blanket assembly will not be examined in detail until neutronic studies define the nature of the reflector and shielding required.

The vapor pressure of Pb-Bi inside the vacuum tank will be $\sim 10^{-5}$ Torr. A precise value is difficult to estimate since different portions of the Pb-Bi jets will be at different temperatures (e.g., the inlet temperature at the spray nozzle might be $\sim$300°C, and the outlet temperature $\sim$500°C), and the pressure-tubes will be at relatively lower temperatures, i.e., $\sim$300°C. The colder surfaces will tend to act as condensing "cold fingers" for Pb-Bi vapor, so that the actual vapor pressure will depend on local position within the target-blanket assembly. There will be some transport of Pb-Bi vapor up the accelerator tube, but this can be easily condensed by cooling panels and returned to the liquid Pb-Bi circuit.

Other configurations of the LAFR are possible, depending on the type of fuel assembly to be enriched and/or regenerated. Figure 6 shows a target-blanket assembly based on CANDU-type fuel assemblies. As in the CANDU reactor, the pressure-tubes are horizontally arranged. The proton beam enters through the narrow slot between the banks of pressure-tubes. Unlike the CANDU reactor, however, there is no moderator between pressure-tubes, so a calandria vessel is not required. Fuel assemblies would be loaded and unloaded at the ends of the pressure-tubes by a fuel handling mechanism similar to that in CANDU reactors. This mechanism is not shown in Figures 6 and 7. If it proves sensible to load and unload fuel on line, as in CANDU reactors, there would be substantial benefits in terms of operating plant factor.

As in the PWR version of the LAFR, described earlier, variable density Pb-Bi jets would be used to shape the neutron source distribution to the desired value (Figure 7).

IV. COOLANT OPTIONS

Coolant options for the blanket fuel assemblies are summarized in Table I together with their relative advantages and disadvantages. Only four options appear to be practical:

1. Liquid D_2O

2. Two-Phase D_2O

3. Two-Phase H_2O

4. Helium

Liquid H_2O does not appear to be acceptable because it will strongly moderate neutrons from the target, causing excessive fission power, excessive power peaking, and variation in fissile content in the blanket fuel assemblies. Sodium does not appear to be acceptable, although it is an attractive coolant, since it may not be possible to maintain a protective film on the Zircaloy cladding of the LWR fuel elements if they are subjected to a sequence of different coolant environments. That is, each cycle in the LAFR would involve exposure to sodium followed by exposure to water in a PWR.

Helium is an attractive coolant option. However, because of the potential for oxidative corrosion with oxygen and water impurities in the helium, the maximum operating clad temperature in He will probably be comparable to that in a PWR, that is, on the order of ∿700°F. Under these conditions, the thermal efficiency of the power cycle with the coolant will be considerably lower than that obtained with two-phase or liquid water coolants. Studies are being carried out to determine what cycle efficiency is achievable, and whether this would be acceptable for a LAFR.

If liquid D_2O is acceptable from the neutronics standpoint, it would be relatively easy to use as the blanket coolant. Pressure drops would be acceptable and thermal cycle efficiency would be acceptable and thermal cycle efficiency would be comparable to that in HWR's, i.e., ∿30%. The additional cost of D_2O coolant would not significantly affect overall system economics. The D_2O inventory in a LAFR is much less than in an existing CANDU reactor, first because it is not required as a moderator and second, because the thermal power of the target-blanket assembly is only ∿40% that of a CANDU reactor. In addition, since one LAFR regenerates fuel for three LWR reactors, the cost of D_2O for a LAFR, expressed as an equivalent additional fuel cycle cost for the LAFR/LWR system, will be negligible (i.e., much less than one mill/KWh).

If liquid D_2O proves not desirable for reasons of neutronics, two-phase (vapor plus liquid) mixtures are an attractive option. Either H_2O or D_2O two-phase mixtures could be used. The effective density of the two-phase mixture will be ∿10% that of the liquid, and both H_2O and D_2O two-phase mixtures should result in relatively "hard" neutron spectra.

In general, two-phase mixtures appear to have excellent characteristics for cooling the LAFR blanket. Two-phase mixtures are desirable because they do not moderate neutrons very well, compared to liquid coolants, and retain good heat transfer/transport capability.

The only reservation with regard to the use of two-phase coolants is the question of how possible is local "dryout" of fuel element surfaces. The excellent heat transfer properties of two-phase coolants depend on evaporation of thin liquid films on the heat transfer surfaces, which are continually replenished by transfer of liquid from the two-phase mixture. Experience with two-phase coolant mixtures indicates that this does not appear to be of serious concern unless very high exit qualities (>0.8) are employed. The exit quality of the two-phase mixtures in a LAFR would be kept well below 0.8, so that "dryout" should not occur.

The LAFR and its associated heat exchangers, piping, pumps, etc., would be located in a containment shell similar to that for LWR's. Figures 8 and 9 show one possible arrangement of a LAFR for the regenerating of PWR fuel assemblies. Penetrations through the containment vessel for the accelerator beam tube and steam lines could be closed by fast acting isolation valves, if necessary. Construction of the LAFR and the inside containment for its associated systems would be similar to that for LWR's.

Overall containment requirements will probably be comparable to those for LWR's though the lower thermal rating of the LAFR's, as well as the use of separate pressure-tubes for the fuel assemblies, may result in somewhat less demanding requirements.

The thermal energy from the LAFR reactor is taken equal to 1350 MWt and is obtained as a combined output from the blanket (900 MWt) and the Pb-Bi target (450 MWt). The hot Pb-Bi exiting the reactor is used to heat the feedwater (@2600 psia) which has been previously heated by the regenerative system up to near the saturation temperature (∿670°F) (Figure 10).

V. OBJECTIVES

For the LAFR system, in order to maintain high fissile material production rate, as well as minimize power peaking, the coolant for these fuel assemblies cannot significantly moderate neutrons from the primary Pb-Bi target. In addition, fuel assemblies must be compatible with both the LAFR coolant and water, since the latter is the coolant in LWR and CANDU.

These restrictions appear to require that a two-phase mixture (steam and liquid) of either D_2O or H_2O be the primary coolant option for a LAFR. The thermal-hydraulic limitations of these coolants in the LAFR fuel assemblies are important in the determination of economic and technical feasibility of the LAFR concept. Thermal-hydraulic analyses have been carried out to determine whether or not this coolant option is practical and what operating ranges are permissible in a LAFR. This presentation reports the findings of the thermal-hydraulic behavior of the two-phase mixture coolant in either PWR or CANDU fuel elements for a LAFR. The key questions examined in this study are:

1. Is adequate cooling of the blanket fuel assemblies achievable at low overall densities of two-phase mixtures (e.g., 10% of liquid density)?

2. Are two-phase pressure losses acceptable for the range of heat fluxes expected in the blanket?

2. Can hot spots occur due to local dryout?

VI. ANALYSES

Two-phase mixtures are examined as a function of heat flux and other thermal-hydraulic variables to determine whether or not they meet these limiting criteria. PWR and CANDU fuel assembly data used in the analyses are summarized in Table II.

The ranges of independent input variables used in the analysis are as follows:

P_o = Average heat flux from fuel pins in the blanket fuel assembly (expressed as a multiple of the average heat flux in either PWR or CANDU elements).

= 0.25, 0.5, 0.75, 1.0 x (average PWR or CANDU heat flux)

X_i = Inlet steam quality to channel

$$= \frac{\text{Mass of Steam}}{\text{Mass of steam and water}}$$

= 0.3, 0.5, 0.7

v_i = Inlet velocity of two-phase mixture to channel

= 25, 50, 75, 100 ft/sec

P_{2p} = Inlet pressure of two-phase mixture to channel

= 1000 psia

The following dependent variables are calculated for the above range of input variables:

X_o = Exit quality of two-phase mixture from channel. (calculated from overall enthalpy balance, assuming pressure drop is negligible)

v_o = Exit velocity of two-phase mixture from channel (calculated from overall mass balance)

ΔP_{2p} = Two-phase pressure drop across channel (based on average quality in the channel)

α_o = Exit steam volume fraction from channel (based on average quality in the channel)

$\bar{h}$ = Average heat transfer coefficient in channel (based on average quality)

ΔT_F = Film temperature drop (based on average heat transfer coefficient in channel)

The following correlations are used:

1. <u>Pressure Drop</u>

The Martinelli-Nelson two-phase friction factor [3,4] is used to determine the pressure drop. The two-phase pressure drop is expressed in terms of the single-phase liquid pressure drop as,

$$\left(\frac{\Delta P}{\Delta L}\right)_{TPF} = \left(\frac{\Delta P}{\Delta L}\right)_{LPF} \phi^2_{\ell tt} \tag{1}$$

where the correlation parameter $\phi_{\ell tt}$ is a function of

$$X_{tt} = \left(\frac{\rho g}{\rho \ell}\right)_{sat}^{1/(2-n)} \left(\frac{\mu\ell}{\mu g}\right)^{n/(2-n)} \left(\frac{1-x}{x}\right) \tag{2}$$

n is determined empirically from $f = C/Re^n$, and n=0.20-0.25 for turbulent flow.

2. <u>Void Fraction</u>

The steam volume fraction, α, is calculated from the Bankoff Correlation [5] of slip ratio,

$$S = \frac{1-\alpha}{K-\alpha} = \text{slip ratio} \tag{3}$$

$$K = K_j+(1-K_j)\alpha^\gamma \tag{4}$$

$$\gamma = 3.33 + 0.18 \times 10^{-3}\ P + 0.46 \times 10^{-6}\ P^2 \tag{5}$$

$$K_j = 0.71 + 10^{-4}\ P \tag{6}$$

P = pressure (psia)

The steam fraction is determined from the relationship;

$$\alpha = \frac{1}{1+S\frac{\nu_f}{\nu_g}\,\frac{1-x}{x}}$$

where, ν_f = specific volume of water

ν_g = specific volume of steam

x = flow quality

3. Heat-Transfer Coefficient

The heat-transfer correlation of Schrock and Grossman with Wright constants [6] is used to determine the two-phase heat-transfer coefficient;

$$h = 6700 \left[\frac{\phi s}{GH_{fg}} + 0.00035 \left\{ \left(\frac{x}{1-x}\right)^{0.9} \left(\frac{\rho f}{\rho g}\right)^{0.5} \left(\frac{\mu g}{\mu f}\right)^{0.1} \right\}^{0.66} \right] \left[\left(\frac{0.023\, k_f}{D}\right) \left(\frac{DG(1-x)}{\mu_f}\right)^{0.8} (Pr)^{0.4} \right] \tag{7}$$

The physical properties are evaluated at saturation conditions, where,

H_{fg} = heat of vaporization, B' /lb.

ρ = density, lb/ft^3

k = thermal conductivity, Btu/hr/ft/°F

VII. RESULTS

Thermal-hydraulic calculations have been carried out for the blankets of accelerator-reactor systems based on the geometry of PWR (Figure 3), and CANDU (Figure 6) fuel assemblies. The following results are obtained with regard to the thermal-hydraulic behavior of LAFR for PWR fuel as shown in Figures 11-15:

1. Two-phase flow is a feasible coolant option for fuel element heat fluxes up to 1 x PWR average value, which is the maximum design value for a LAFR. As shown in Figure 11, steam volume fractions are reasonable for an inlet quality of 0.3 and the range of heat fluxes and inlet velocities considered.

2. In Figure 12, the results show that the two-phase flow pressure drops in the PWR pressure tubes are low, in the range of 10-30 psi for all parameters studied. The film temperature drops as given in Figure 13, are also low (typically ~10°F). From Figure 14, the results show that the optimum inlet velocity should be in the range of 50 to 75ft/sec. A steam volume fr'ction of 0.80 corresponds to an effective density equivalent to ~20 percent that of the liquid. Results show effective density for the two-phase mixture is generally about 10 percent that of

the liquid, which is well within the limiting criterion. Dryout of the fuel assembly surface does not appear to be of concern, since exit quality as shown in Figure 14 can be kept well below 0.8, the point of maximum heat transfer coefficient (Figure 15), by suitable choice of coolant flow conditions. The coolant temperature operating range corresponds to a power cycle efficiency of ~30 percent, which considerably exceeds the allowable lower limit of 20 percent.

The following results are obtained for the LAFR with CANDU fuel elements as shown in Figure 16-20:

1. Two-phase flow appears adequate for cooling of CANDU fuel assemblies for heat fluxes up to 1 x the average CANDU value. Steam volume fractions are in a reasonable range at inlet quality of 0.3. However, a somewhat higher inlet velocity range of 75 to 100 ft/sec is necessary to prevent dryout.

2. Two-phase flow pressure drops, though higher for CANDU fuel elements, typically 50 to 100 psi, are still considered manageable. They are given in Figure 17. Film temperature drops are shown in Figure 18 and are again very low (typically ~10°F). Optimum inlet velocity is in the range of 75 to 100 ft/sec and is concluded from Figure 19. Effective density for the two-phase mixture is well within the limiting criterion. The favorable heat-transfer coefficients are shown in Figure 20 for CANDU fuel elements.

VIII. DISCUSSION AND CONCLUSIONS

Returning to the questions about the two-phase cooling of the blanket listed earlier, the following answers can be formulated:

1. Adequate cooling of the blanket fuel assemblies appears achievable at low overall two-phase coolant densities, i.e., ~10% of liquid density, at least up to fuel pin heat fluxes that are comparable to average PWR and CANDU heat fluxes (~200,000 BTU/hr/ft^2). In any case these heat fluxes would appear to be upper limits, regardless of coolant choice, so as to ensure good cooling and safe operation. Since the blanket area can be readily designed to keep maximum heat fluxes at below this limit, there appears to be no problem in achieving adequately low coolant densities with two-phase mixtures.

2. Two-phase flow pressure drops with PWR assemblies are quite low and have no significant impact on power generation efficiency or coolant properties. With CANDU assemblies, two-phase flow pressure drops are considerably higher, because of lower coolant/fuel volume ratio and longer pressure-tubes. The length considered in the CANDU cases examined, 19.5 feet, appears to be an upper limit to pressure tube length for maximum heat flux in the assemblies on the order of 200,000 BTU/hr/ft^2. If heat fluxes are lower, pressure tubes can be correspondingly longer. If longer pressure-tubes are not possible at high heat fluxes, it probably

would be necessary to use split flow headering, with the pressure-tube fed at its middle with inlet coolant.

The coolant would then exit at each end of the pressure tube. There appears to be no fundamental reason why this would not be feasible, but it would involve a more complex piping arrangement. With this flow, the length of the blanket pressure-tubes could be doubled, so that pressure-tubes could be ∿40 ft long at heat fluxes of ∿200,000 BTU/hr/ft^2.

3. Gross dryout of the fuel assembly is of no concern, since outlet qualities of the two-phase mixture can readily be in the range of 0.4 to 0.7, depending on coolant flow conditions. The question of localized dryout is a much harder question to answer, since it is affected by fuel element geometry, evenness of flow, etc. Ultimately, experiments on fuel assemblies would be required to assure that dryout could not occur. The margin of safety is measured by going to low inlet qualities and high inlet velocities. For PWR fuel assemblies, an inlet quality of as low as 0.2 and a flow velocity of 75 ft/sec yields a pressure drop of ∿36 psi, which is acceptable. The effective moderator density (Figure 11 and 16) would be higher, however, on the order of 20% of liquid density. This probably would be acceptable from the standpoint of neutronics. Outlet quality under these conditions would be ∿0.3, so that the margin of safety would be considerable. Conditions for CANDU fuel are similar, with an inlet quality of ∿0.2 corresponding to outlet quality of ∿0.4. It thus appears that a large margin of safety against local dryout can be provided. Experiments on electrically heated simulated fuel assemblies can be carried out to determine local heat transfer conditions, and these should demonstrate that local dryout is not a problem.

For the range of LAFR designs considered, two-phase coolant appears feasible. The choice between two-phase mixture coolant or other coolant options will probably be primarily determined by economics, and will depend strongly on the particular parameters of a given LAFR design.

REFERENCES

1. P. Grand and H. C. Kouts, "Conceptual Design and Economic Analysis of a Light Water Reactor Fuel Regenerator," BNL Report 50838, May 1978.

2. P. Grand, J. R. Powell, M. Steinberg, and H. Takahashi, "Linear Accelerator Driven and Regenerative Reactor Systems," Brookhaven National Laboratory, Submitted to 1978 Annual ANS Meeting, San Diego, CA, June 18-22, 1978.

3. L. S. Tong, Boiling Heat-Transfer and Two-Phase Flow, p. 79, John Wiley & Sons, Inc., New York, London, Sydney.

4. C. J. Hsu, Brookhaven National Laboratory, private communication.

5. S. G. Bankoff, "A Variable Density Single-Fluid Model for Two-Phase Flow", Trans, ASME, 82, 265 (1960).

6. V. E. Schrock and L. M. Grossman, Forced Convection Boiling Studies, Final Report on Forced Convection Vaporization Project, TID-14632 (1959).

7. Steam/Nuclear Installtions for Electric Utilities.

8. UE&G, ERDA, 770630, COO-2477-4.

TABLE I

BLANKET COOLANT OPTIONS

OPTION	ADVANTAGES	DISADVANTAGES
HELIUM	NON-REACTIVE NON-MODERATING	LOW THERMAL CYCLE EFFICIENCY WITH ZIRCALOY CLAD TEMPERATURE LIMITATIONS
D_2O LIQUID	INTERMEDIATE MODERATION NO DRYOUT CONCERN	ADDITIONAL COST -
2 PHASE	LOW MODERATION	POSSIBLE DRYOUT CONCERN
H_2O LIQUID	LOW COST, NO DRYOUT CONCERN	STRONGLY MODERATES NEUTRONS
2 PHASE	LOW MODERATION	POSSIBLE DRYOUT CONCERN
SODIUM	LOW PRESSURE HIGH THERMAL EFFICIENCY NON-MODERATING	REACTIVE NOT COMPATIBLE WITH ZIRCALOY - CAN ONLY BE USED IN FUEL PRODUCER

TABLE II

FUEL ASSEMBLY DATA

	PWR [7]	CANDU [8]
ASSEMBLY (OR CHANNEL)	205	380
FUELS/ASSEMBLY (OR/BUNDLE)	208	37
BUNDLE/CHANNEL	-	12
LENGTH/CHANNEL, FT	12	19.5
EQUIVALENT DIAMETER, FT OF FLOW CHANNELS	0.042	0.0290
FREE FLOW AREA/ASSEMBLY (OR/CHANNEL), FT^2	0.30	0.0360
HEAT-TRANSFER AREA/ASSEMBLY (OR/CHANNEL), FT^2	281	96.17
AVERAGE HEAT FLUX, $BTU/FT^2/HR$	$197{,}000 \times \begin{cases} 1.0 \\ 0.75 \\ 0.5 \\ 0.25 \end{cases}$	$193{,}000 \times \begin{cases} 1.0 \\ 0.75 \\ 0.5 \\ 0.25 \end{cases}$
VOLUME RATIO OF COOLANT/ FUEL IN FUEL ASSEMBLY	1.5	0.8

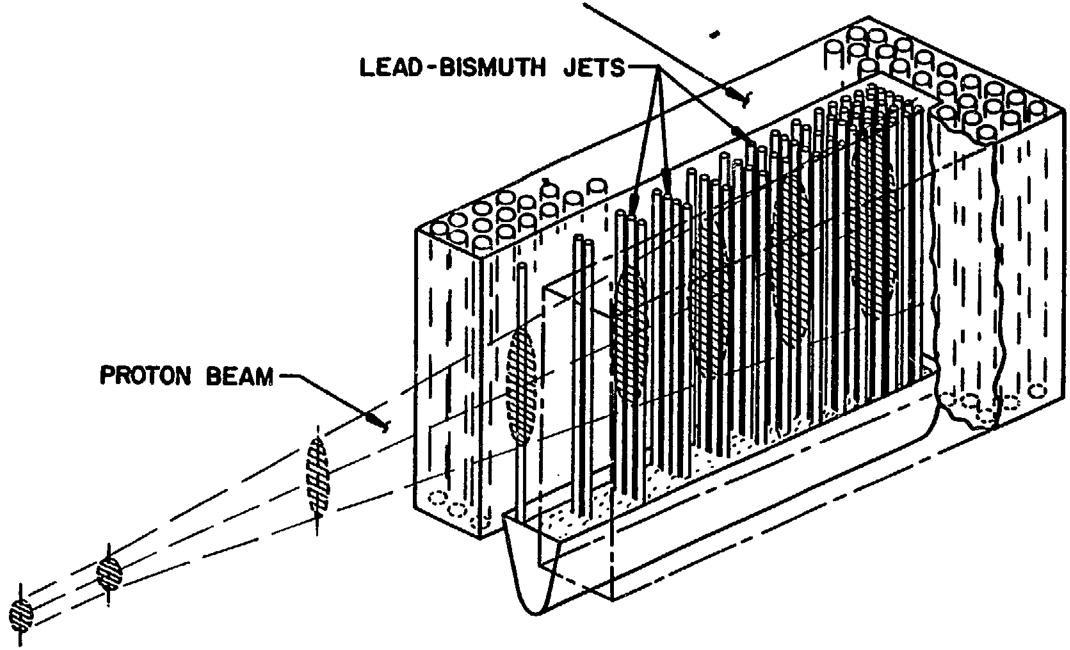


Fig. 1 Design Principle of the Target

2-922-79

LINEAR ACCELERATOR FUEL REGENERATOR (LAFR)

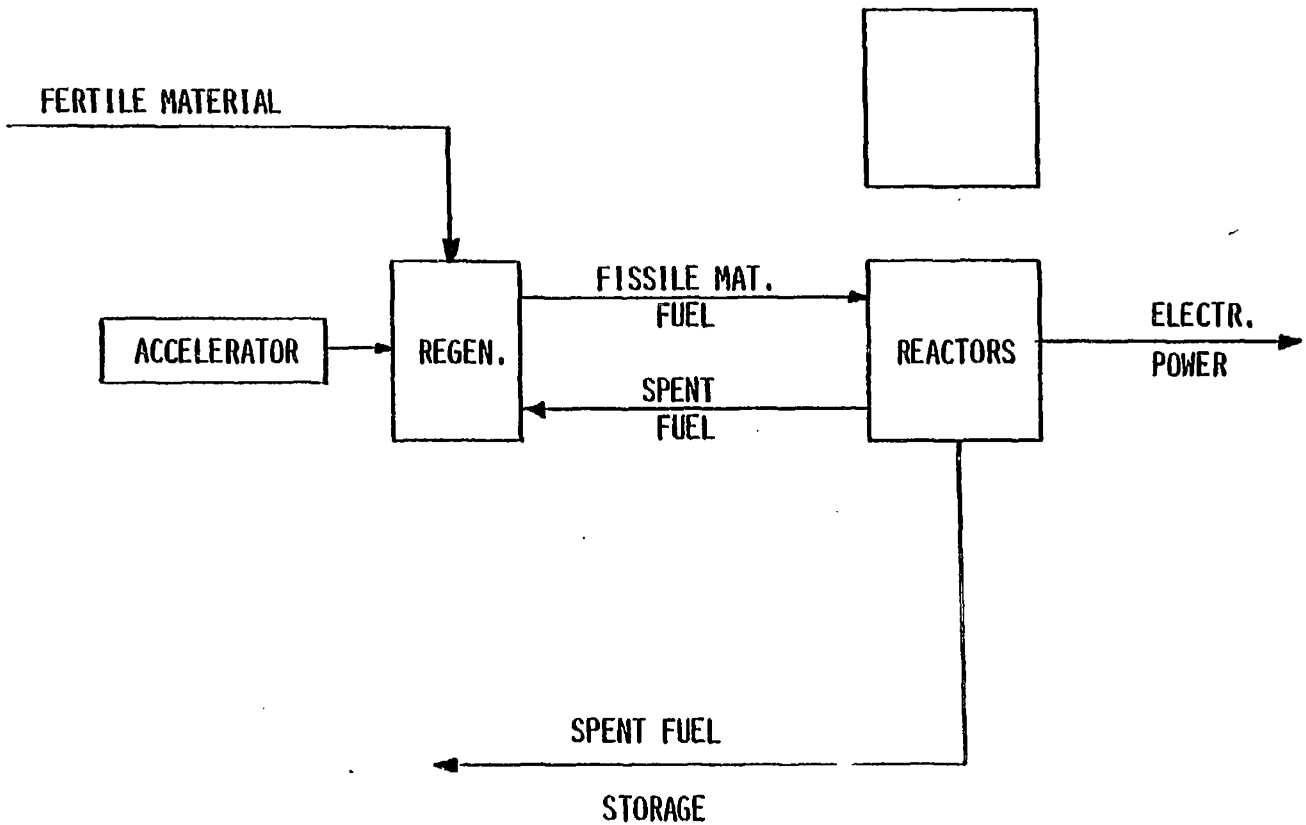


Fig. 2 Schematic Chart for a LAFR System

2-911-79

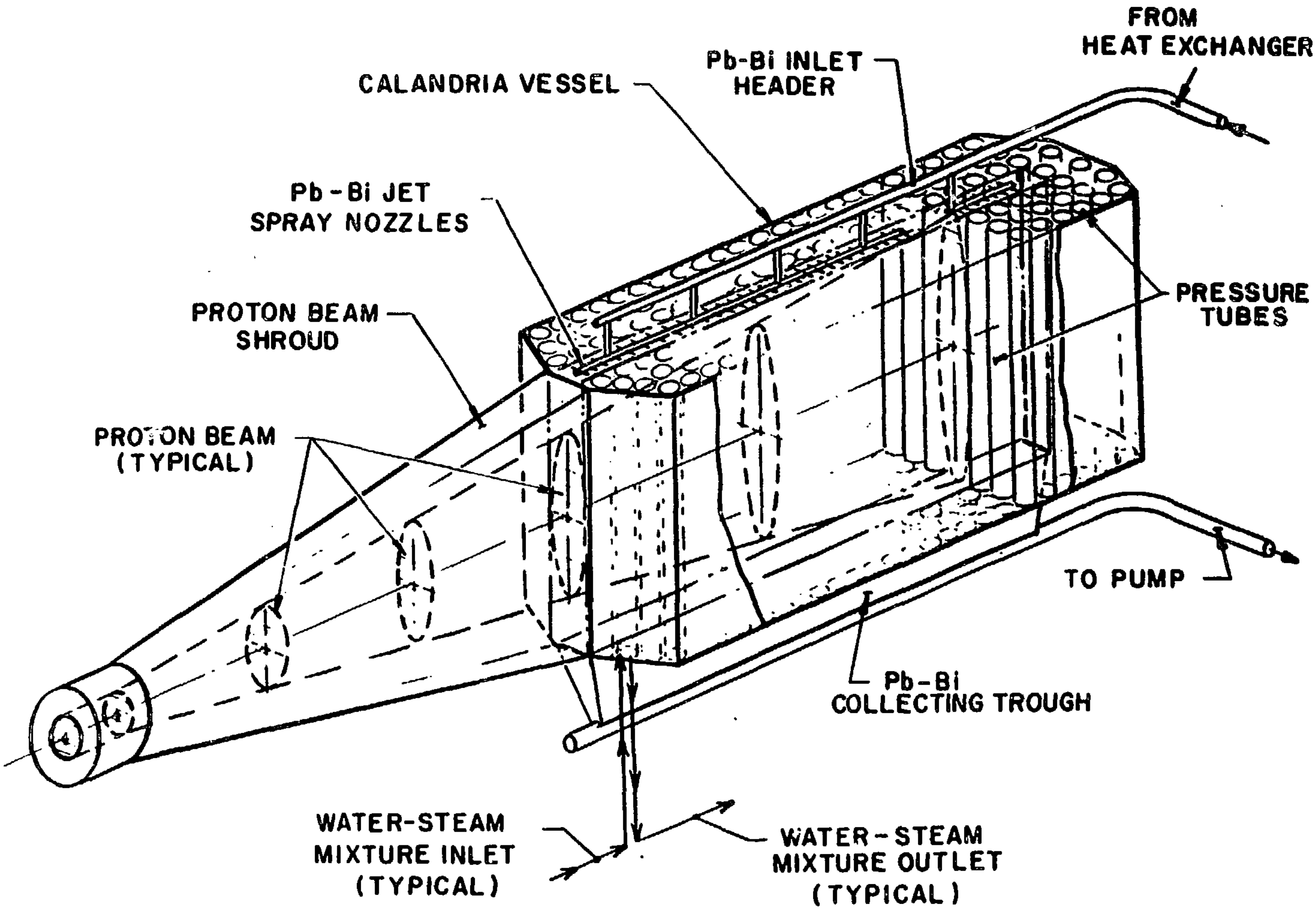


Fig. 3 PWR TYPE TARGET ASSEMBLY

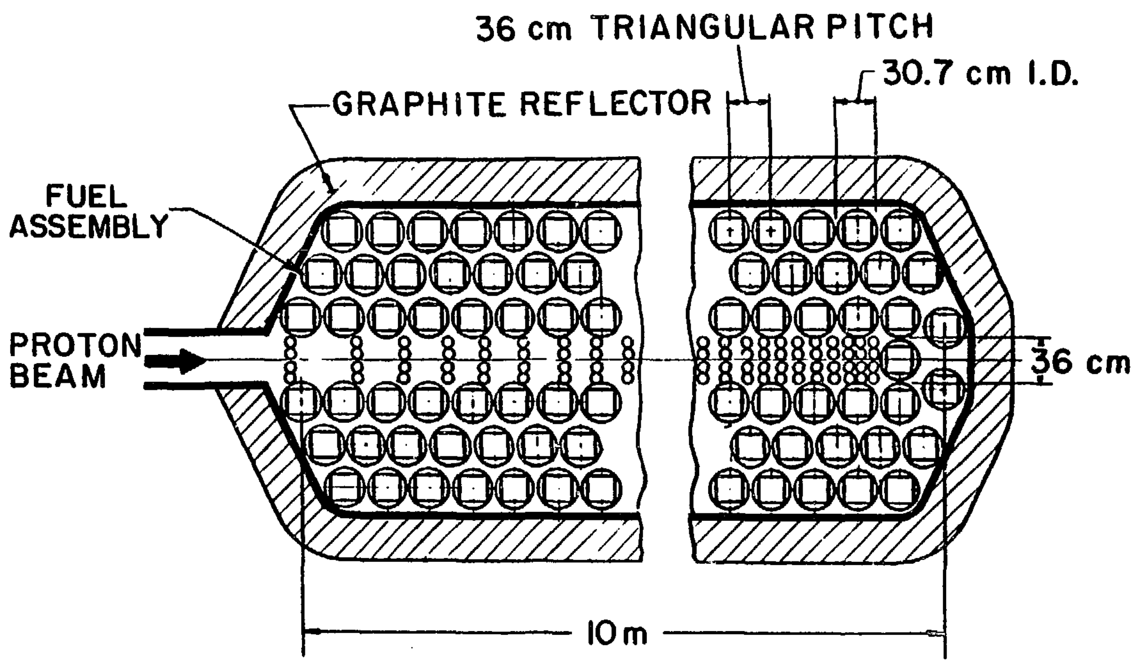


Fig. 4 PWR TYPE TARGET ASSEMBLY CROSS SECTION

2-920-79

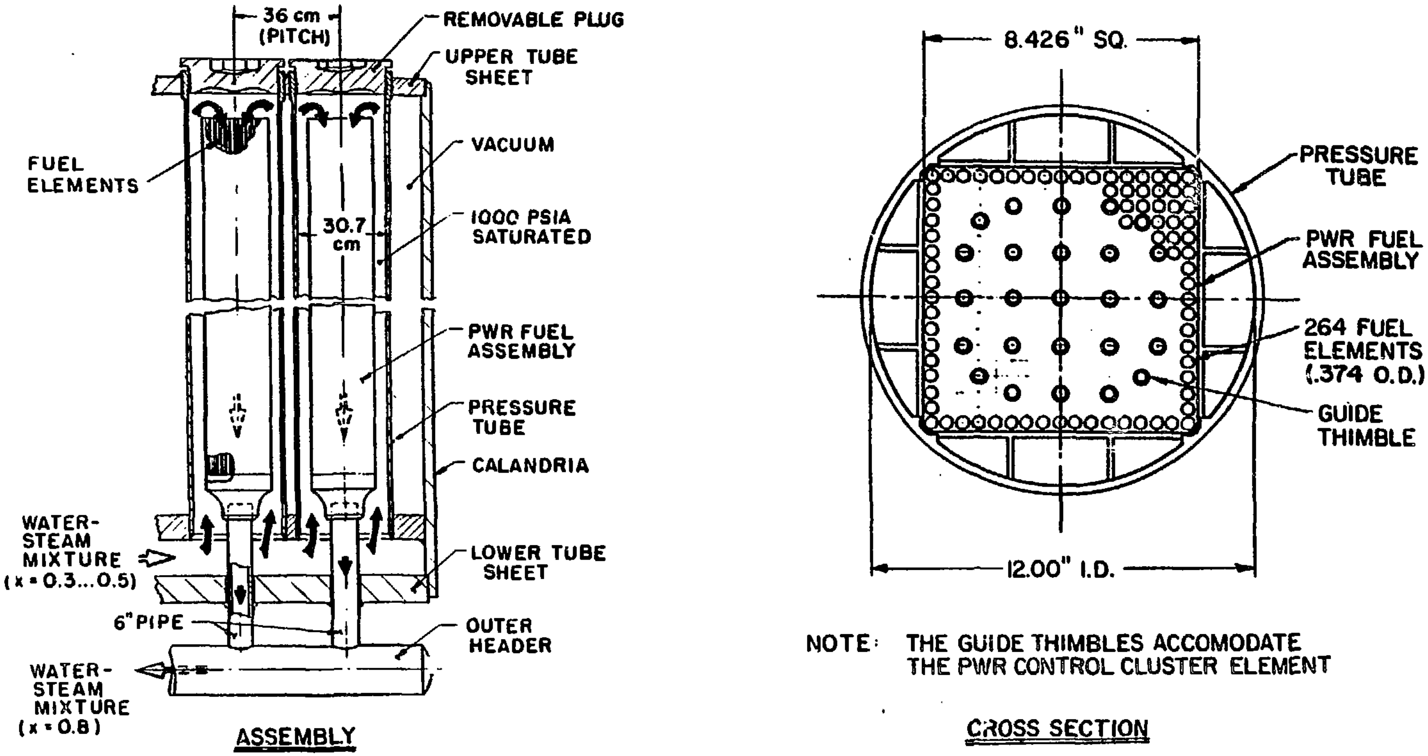
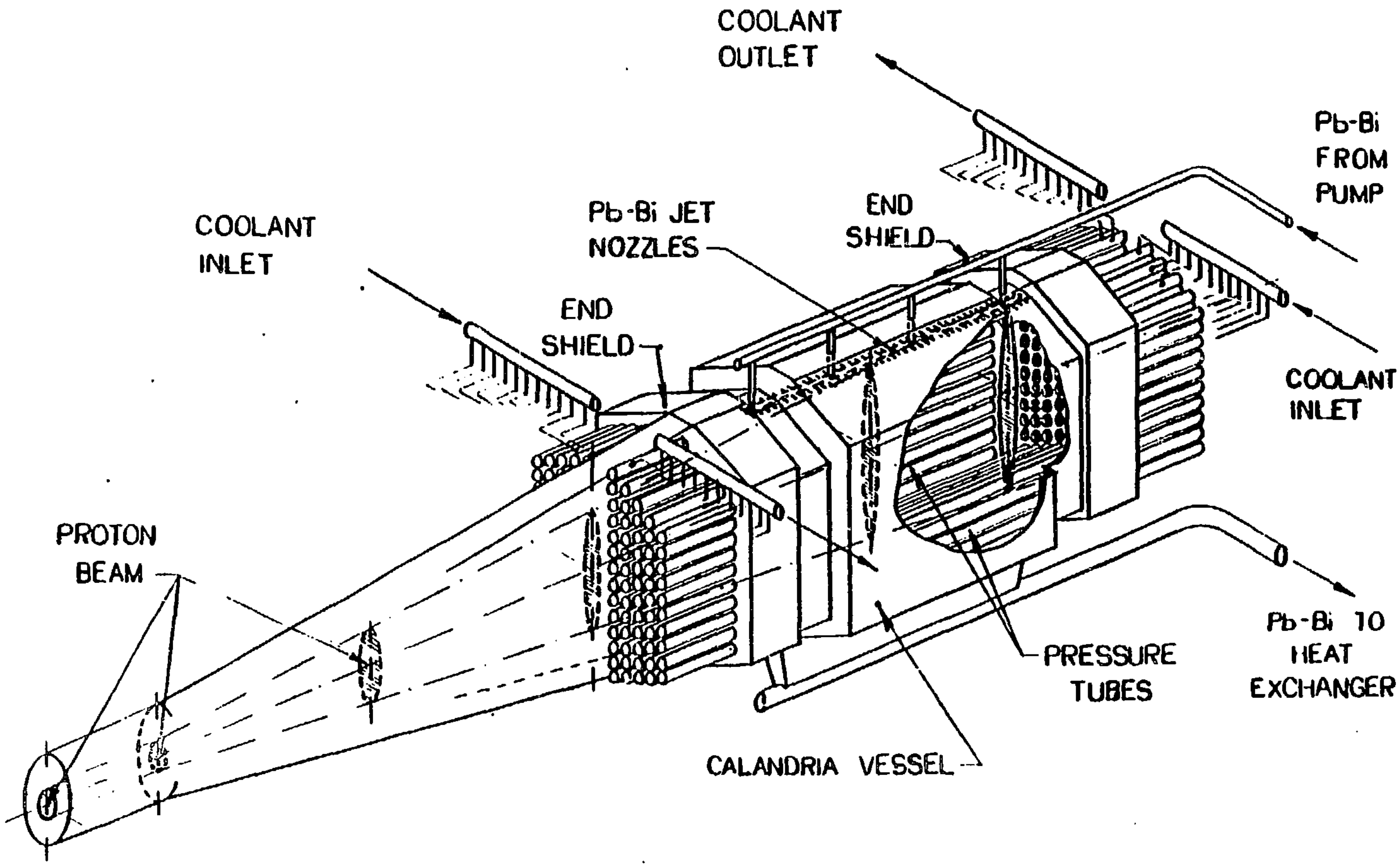


Fig. 5 PWR Type Pressure Tubes

2-916-79



2-914-79

Fig. 6 HWR TYPE TARGET ASSEMBLY

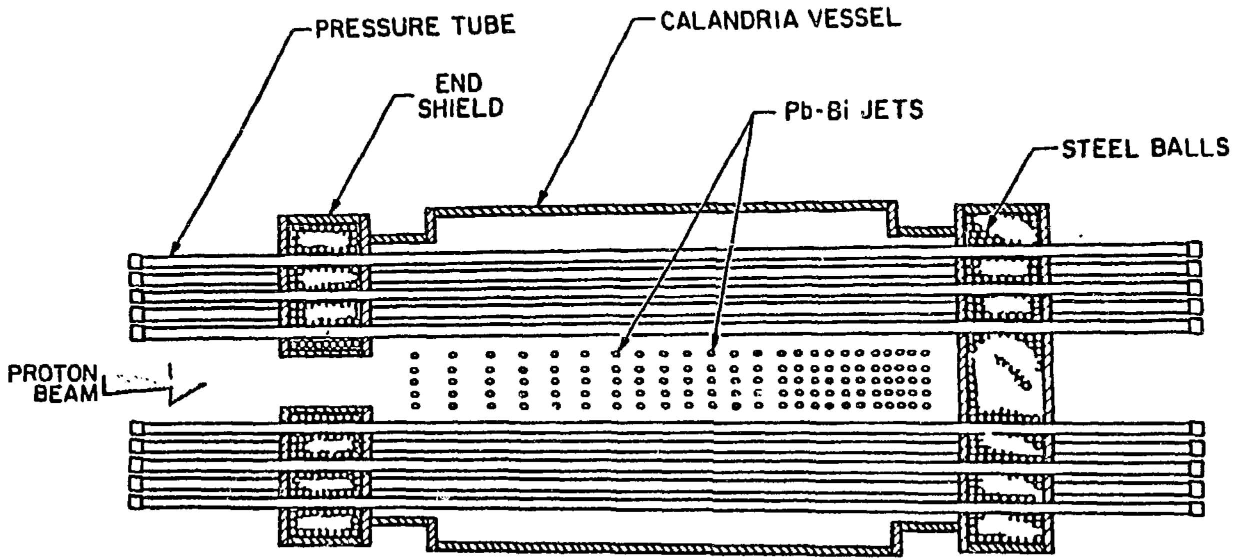
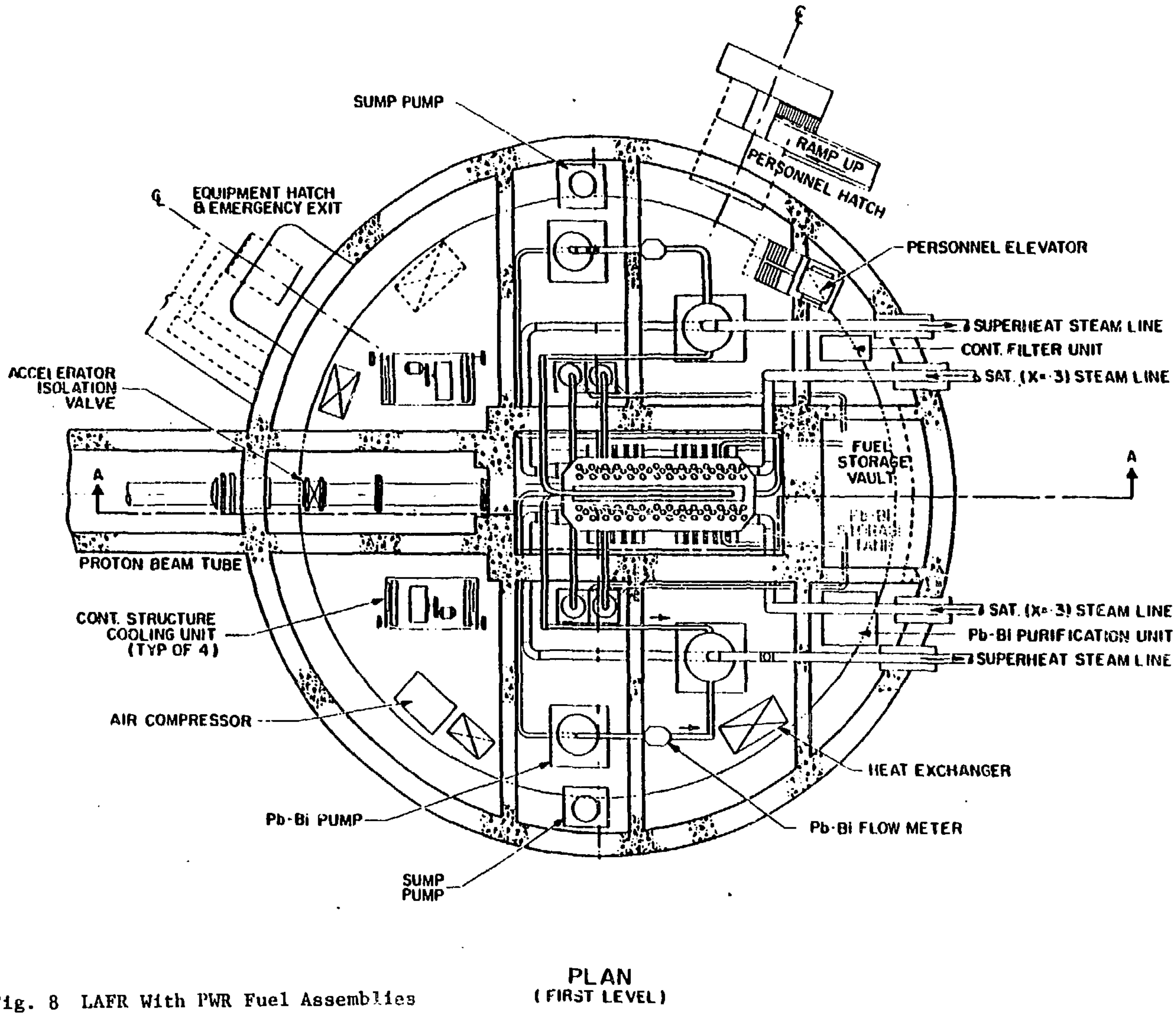


Fig. 7 HWR TYPE TARGET ASSEMBLY, CROSS SECTION

2-919-79



2-918-79

Fig. 8 LAFR With PWR Fuel Assemblies

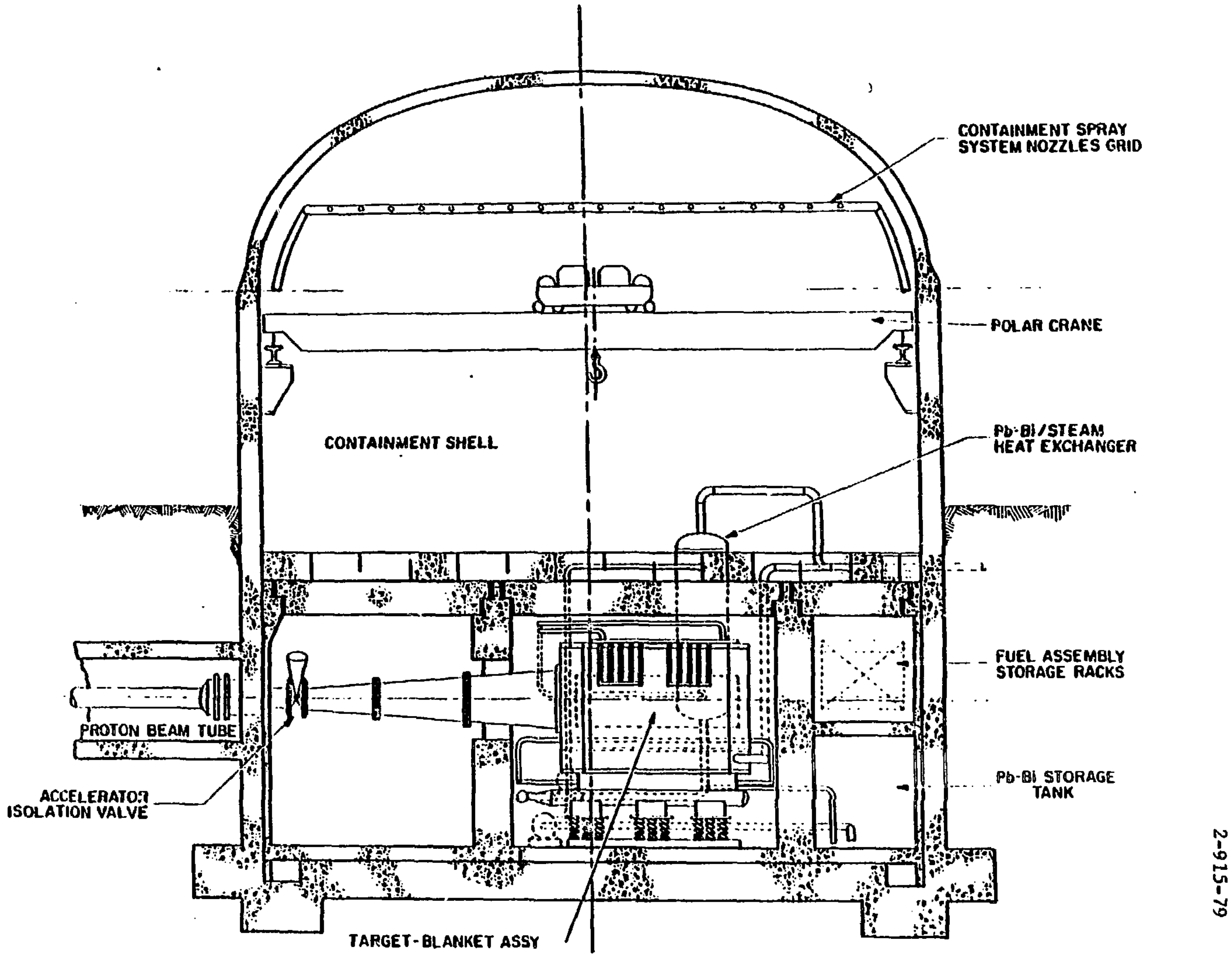


Fig. 9 LAFR With PWR Fuel Assemblies

ELEVATION A-A

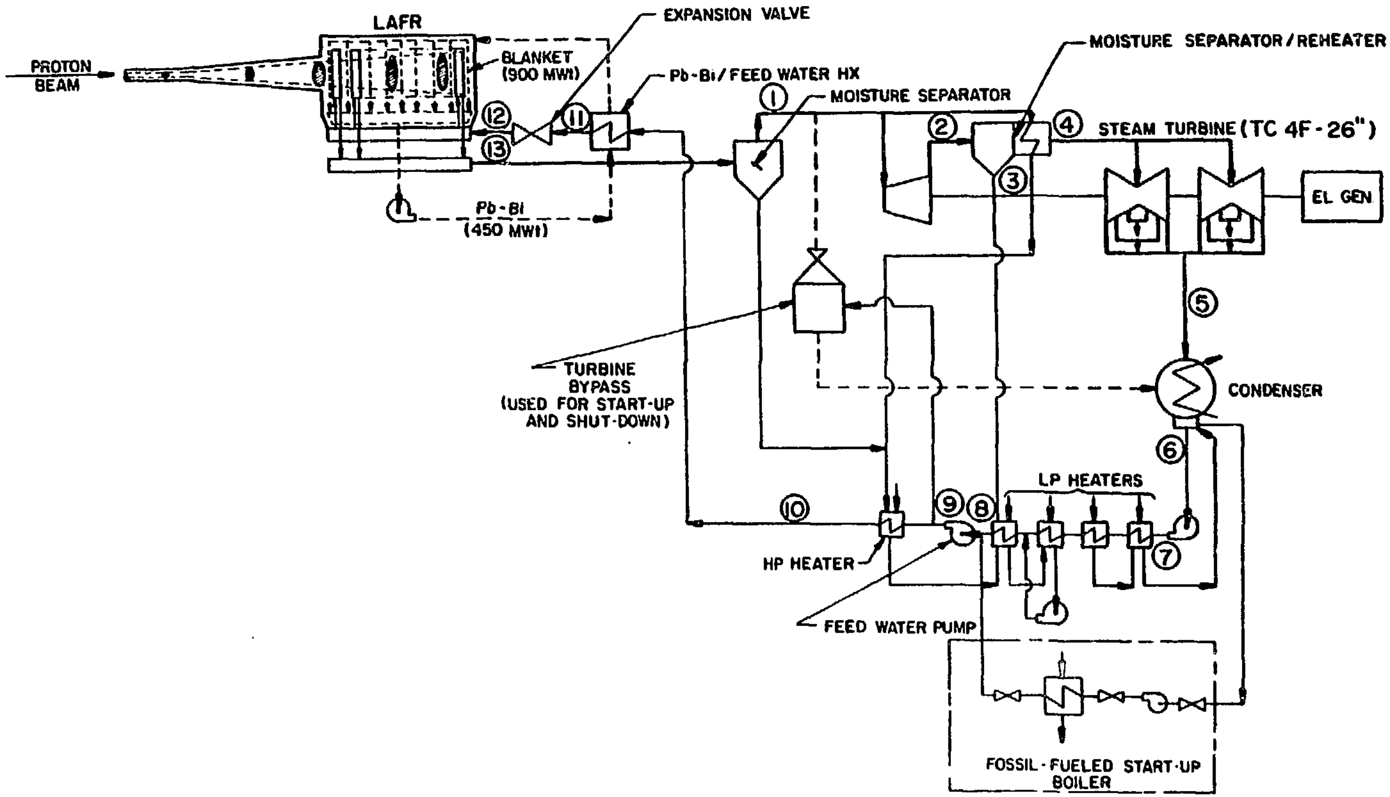


Fig. 10 LAFR Facility Steam Cycle

2-921-79

3-479-78

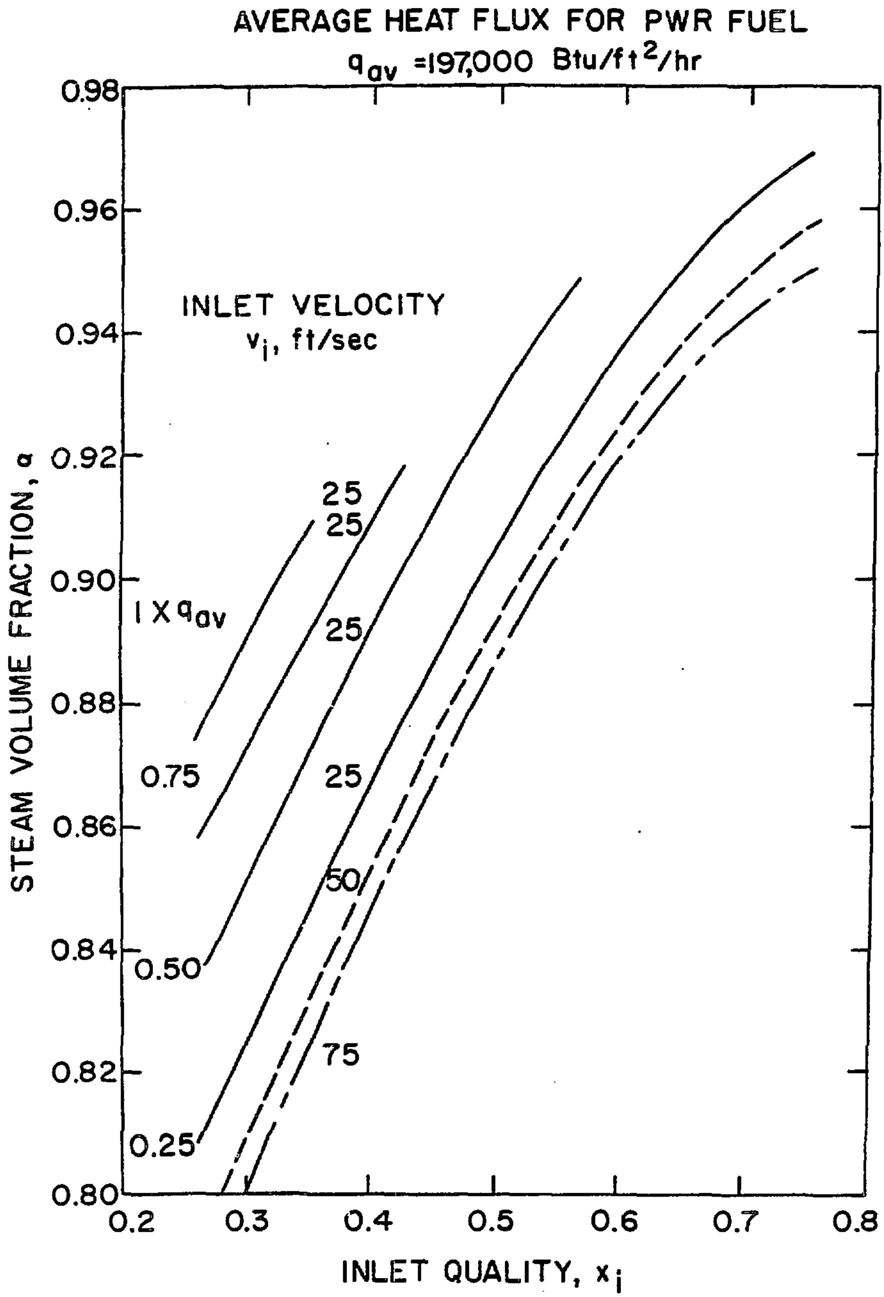


Fig. 11 Steam Volume Fraction vs. Inlet Quality at Three Inlet Velocities and Various Fuel Element Heat Fluxes for PWR Fuel

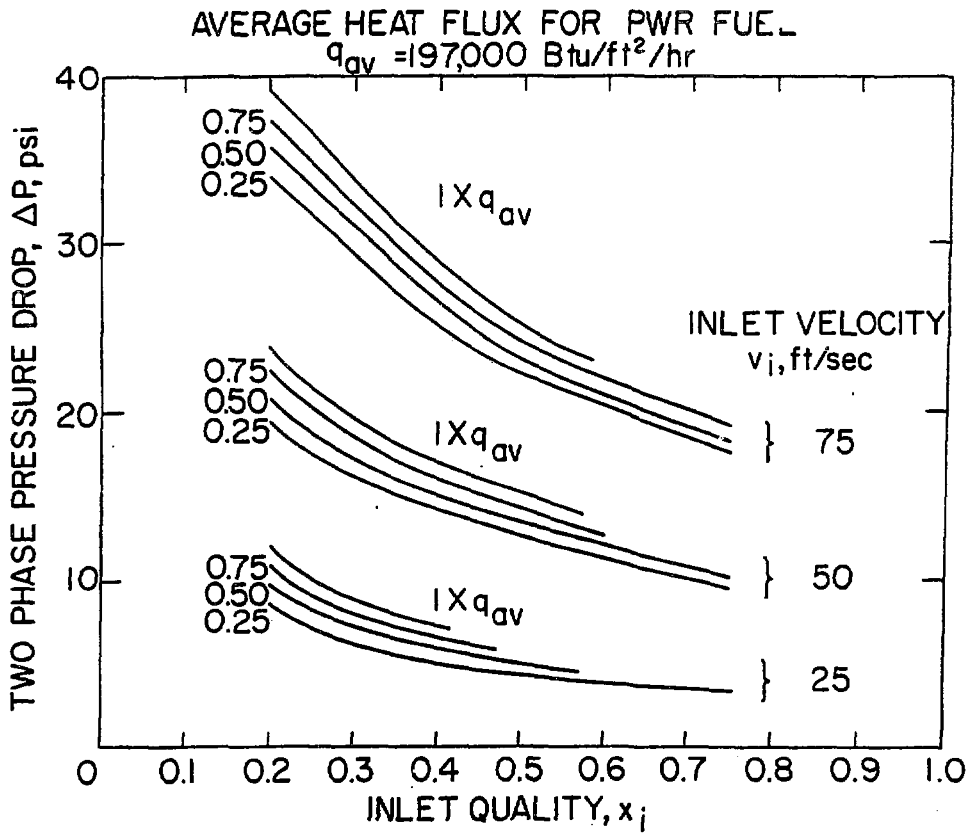


Fig. 12 Two-Phase Pressure Drop vs. Inlet Quality at Three Inlet Velocities and Various Fuel Element Heat Fluxes for PWR Fuel

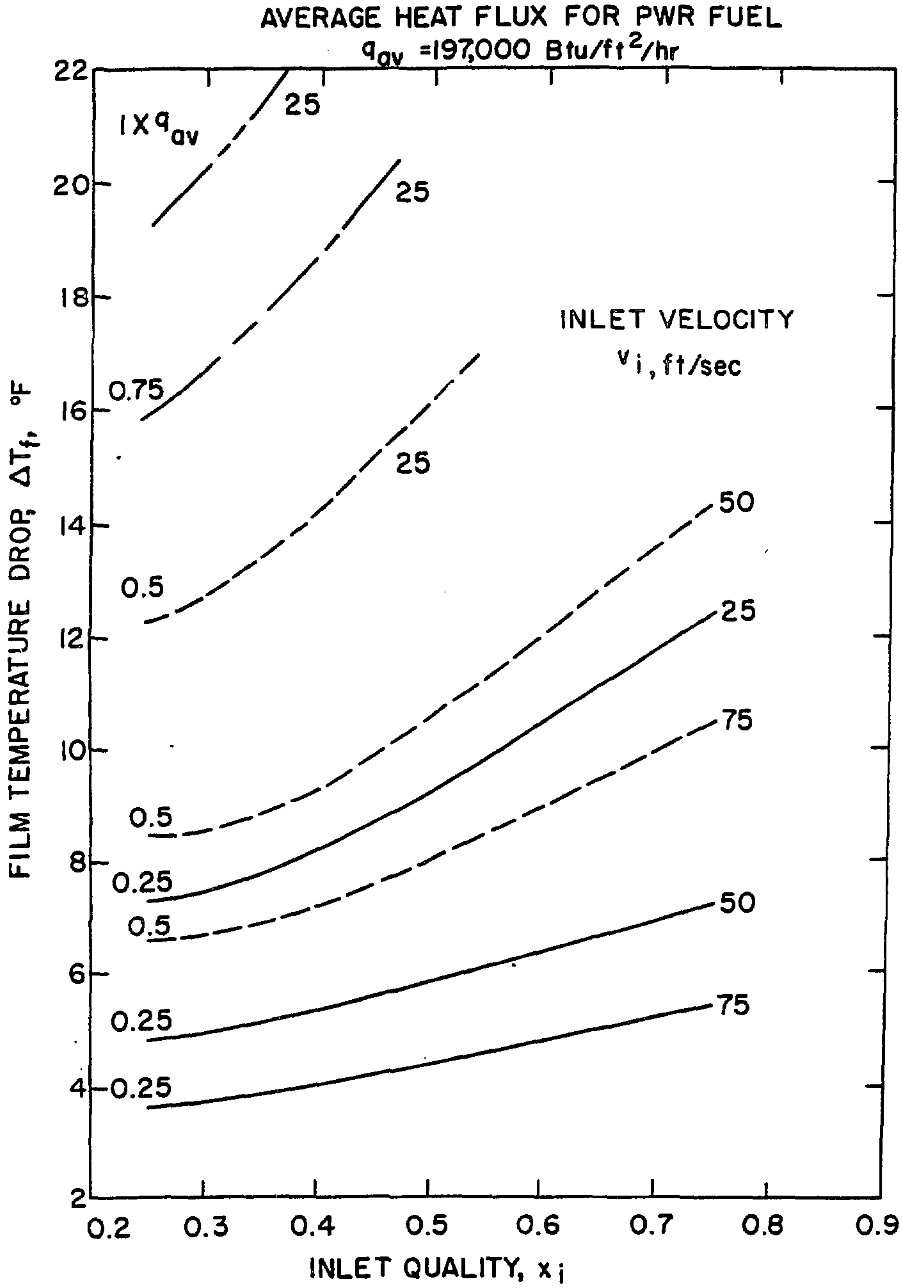


Fig. 13 Two-Phase Film Temperature Drop vs. Inlet Quality at Three Inlet Velocities and Various Fuel Element Heat Fluxes for PWR Fuel

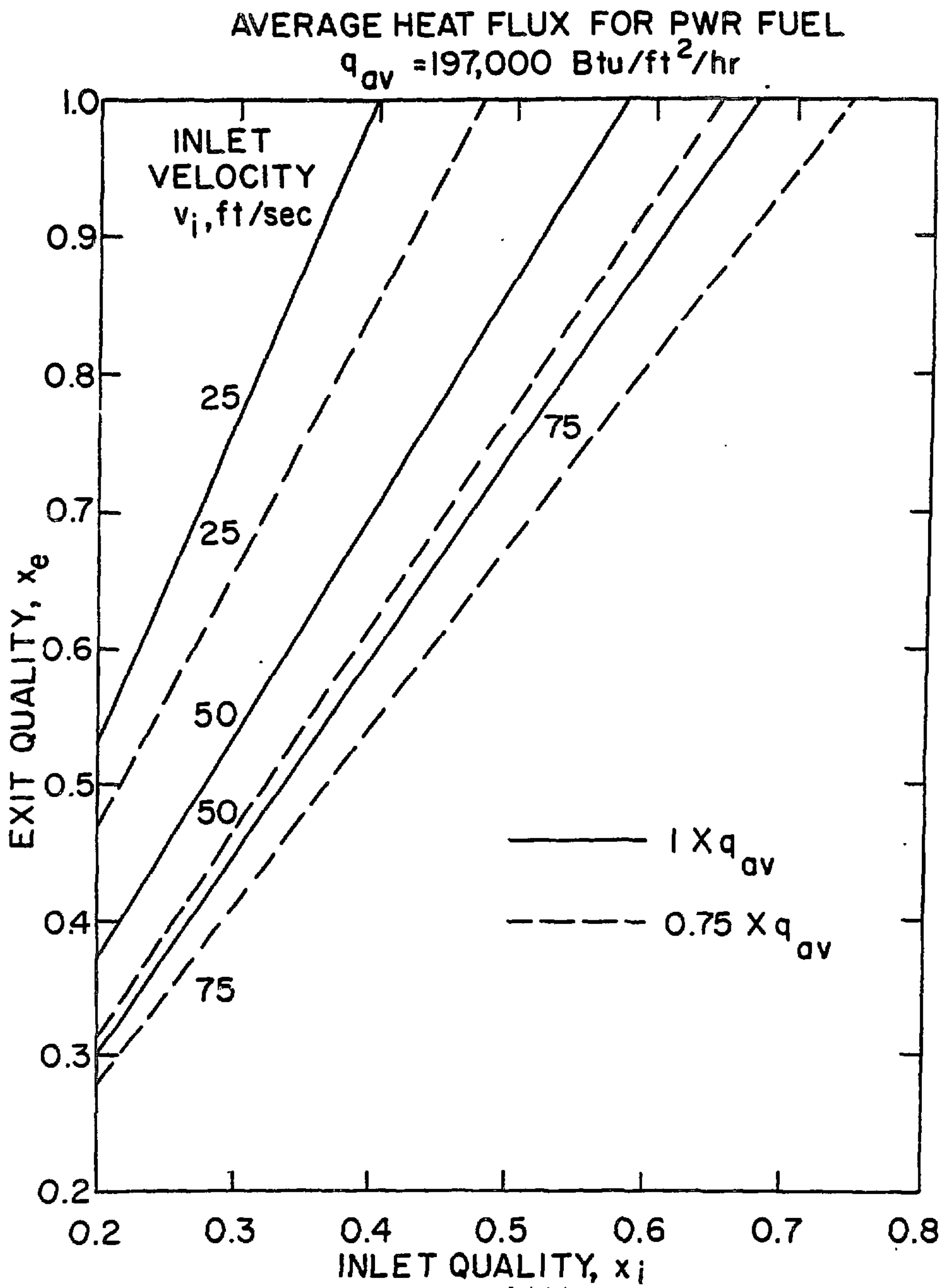


Fig. 14 Two-Phase Exit Quality vs. Inlet Quality at Two Different Fuel Element Heat Fluxes and Various Inlet Velocities for PWR Fuel

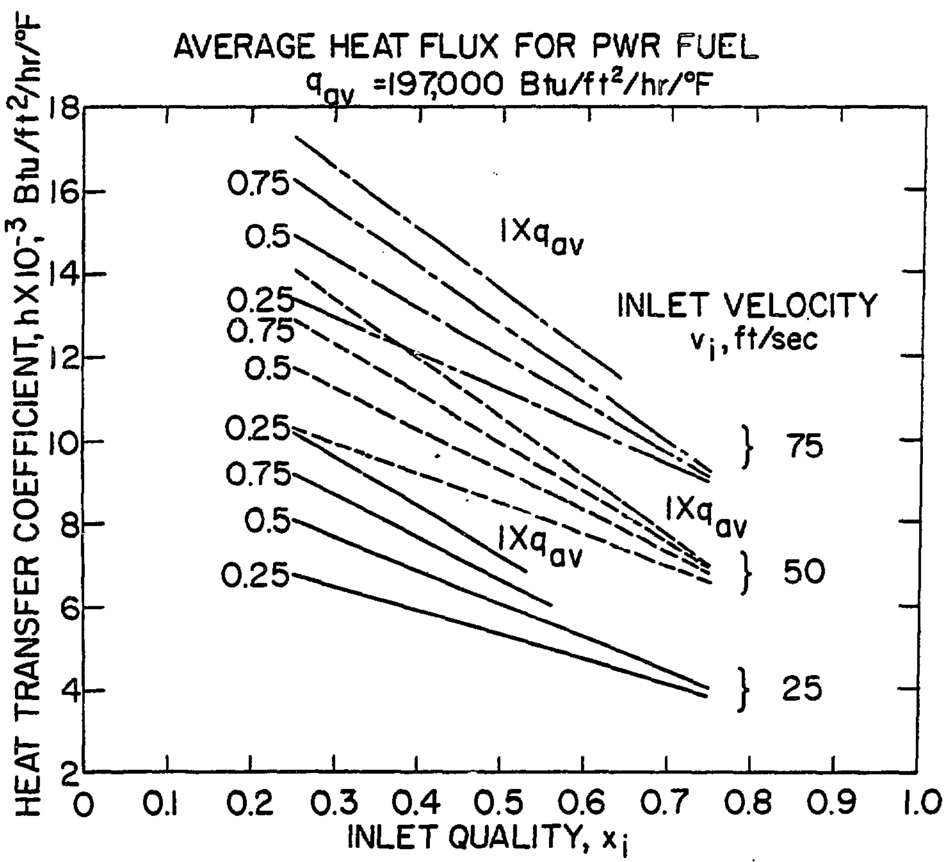


Fig. 15 Two-Phase Heat Transfer Coefficient vs. Inlet Quality of Three Inlet Velocities and Various Fuel Element Heat Fluxes for PWR Fuel

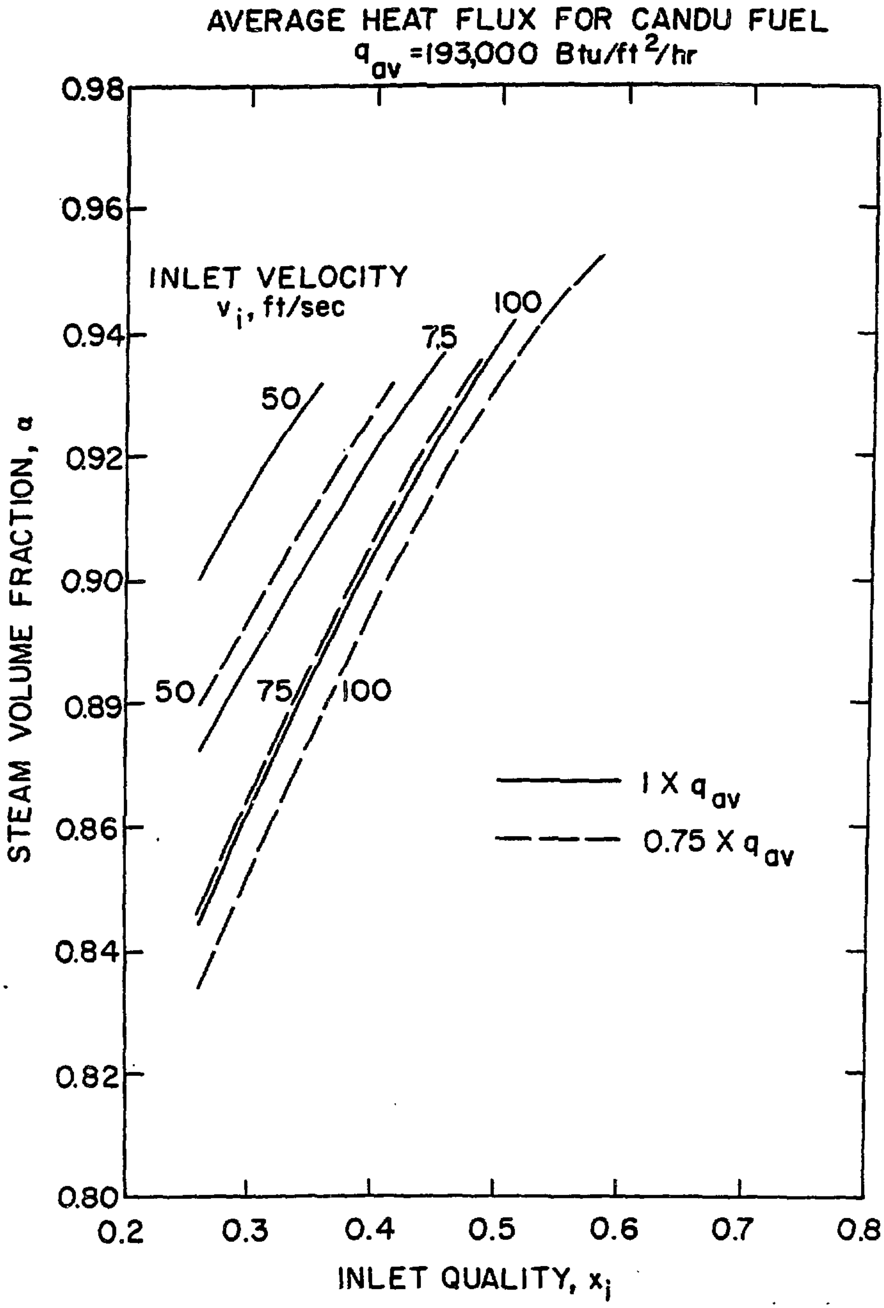


Fig. 16 Steam Volume Fraction vs. Inlet Quality at Three Inlet Velocities and Two Fuel Element Heat Fluxes for CANDU Fuel

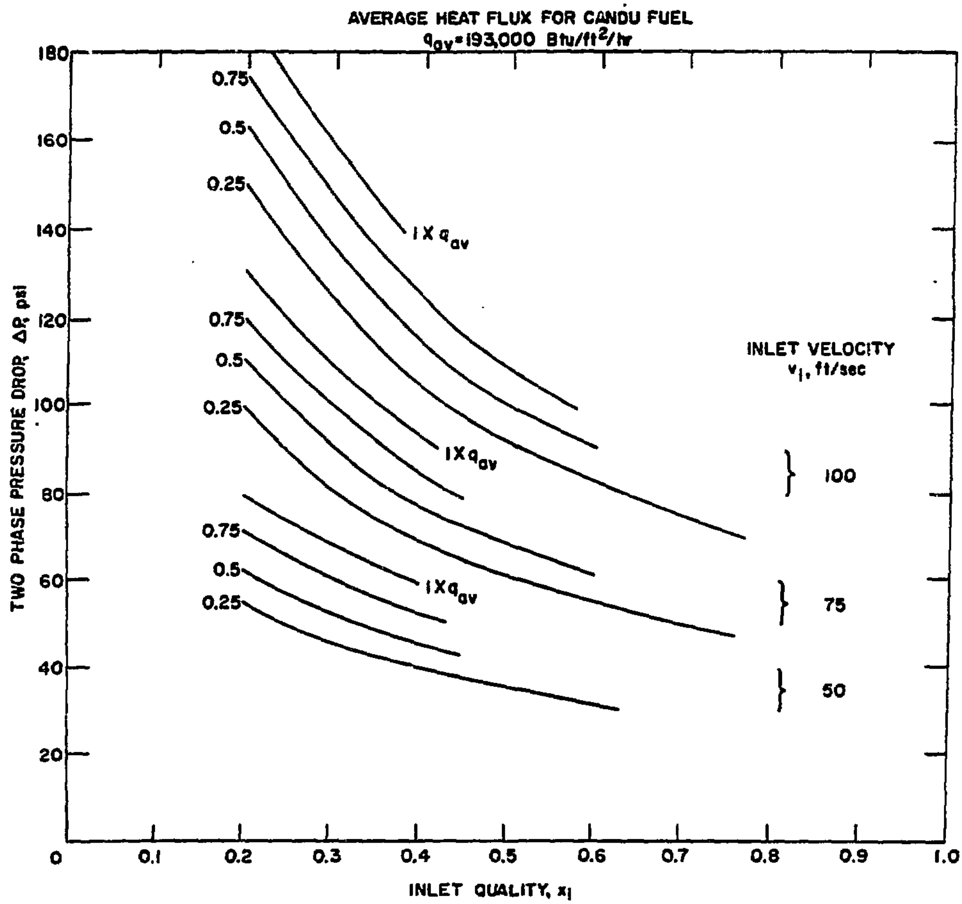


Fig. 17 Two-Phase Pressure Drop vs. Inlet Quality at Three Inlet Velocities and Various Fuel Element Heat Fluxes for CANDU Fuel

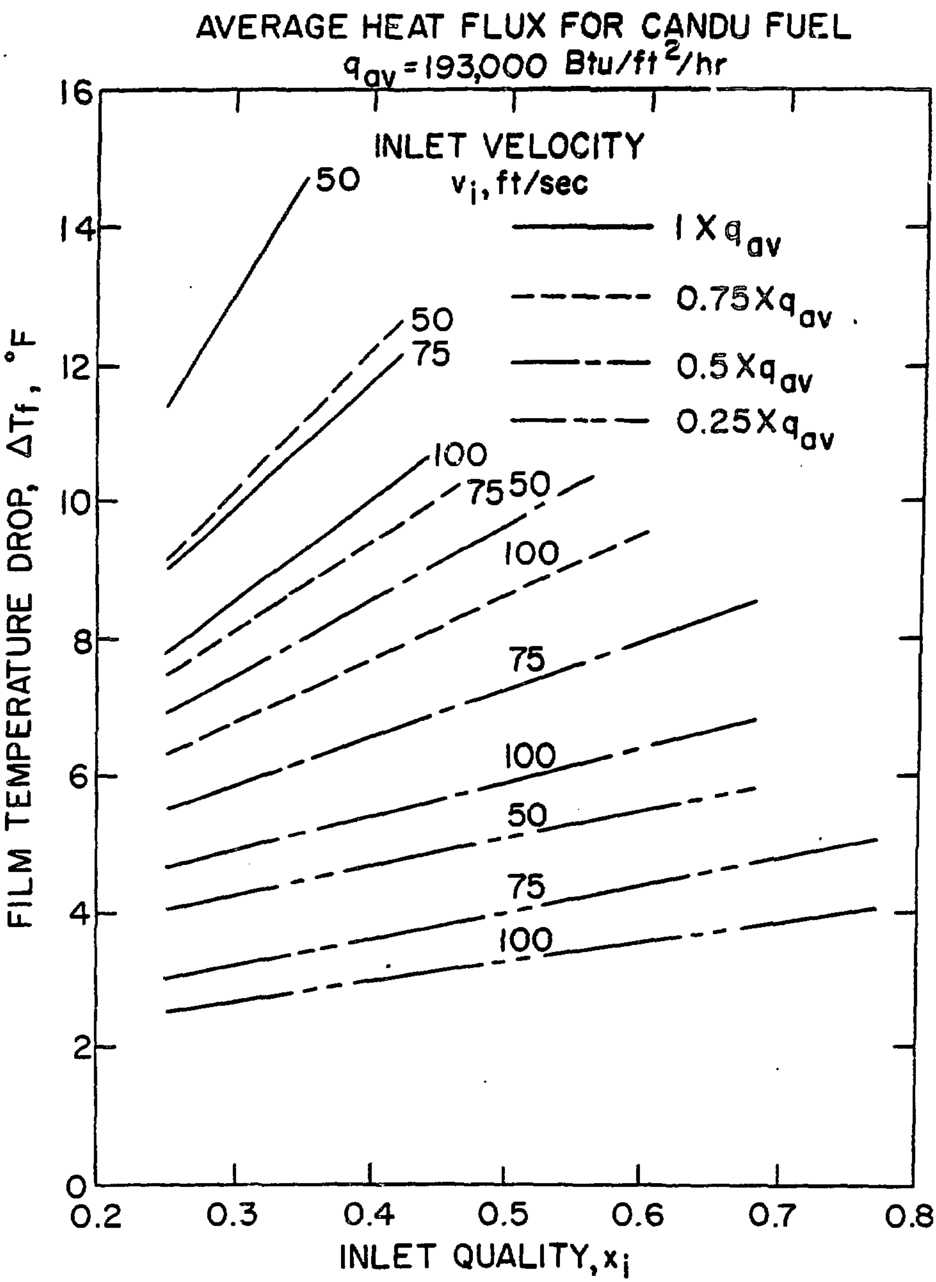


Fig. 18 Two-Phase Film Temperature Drop vs. Inlet Quality at Three Inlet Velocities and Various Fuel Element Heat Fluxes for CANDU Fuel

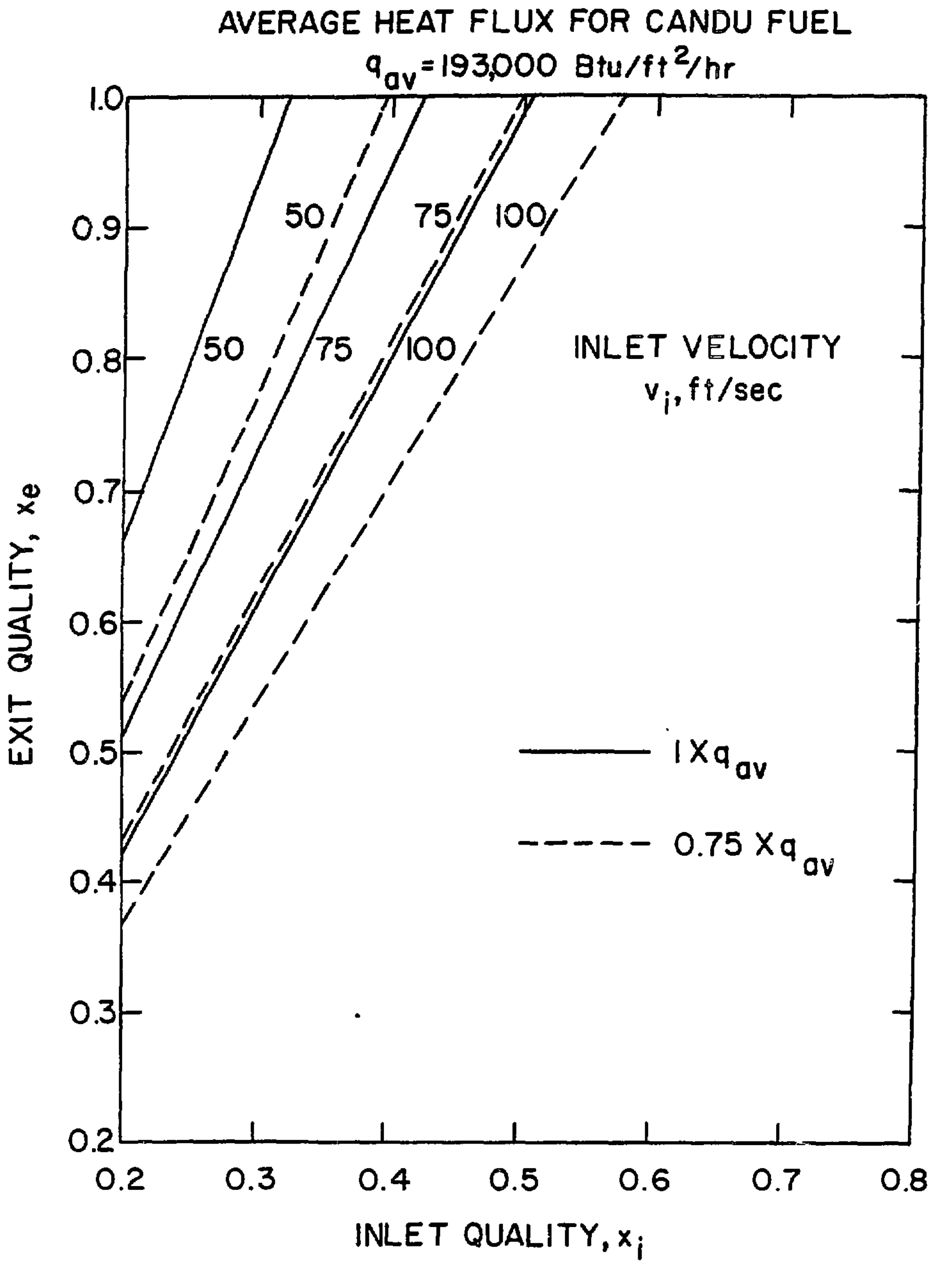


Fig. 19 Two-Phase Exit Quality vs. Inlet Quality at Two Different Fuel Element Heat Fluxes and Three Inlet Velocities for *CANDU* Fuel

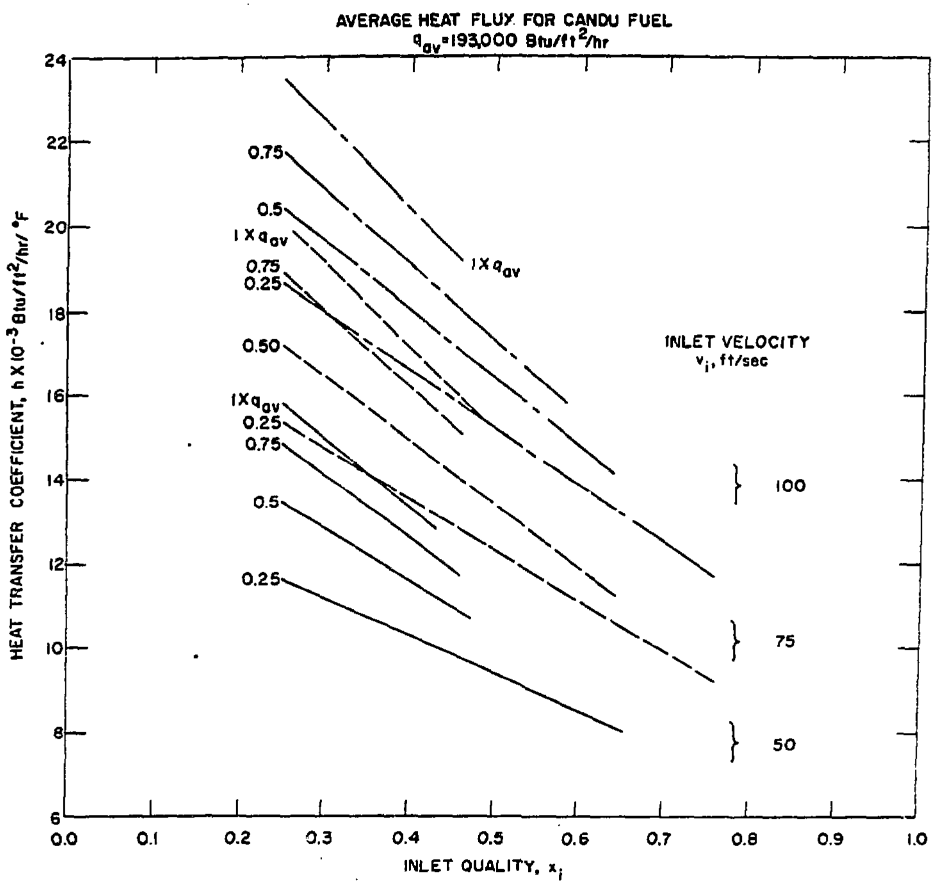


Fig. 20 Two-Phase Heat Transfer Coefficient vs. Inlet Quality at Three Inlet Velocities and Various Fuel Element Heat Fluxes for CANDU FUEL