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VOL. 4 ENGINEERING AND DESIGN CONSIDERATIONS FOR IN-PILE TESTS

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**PHILLIPS
PETROLEUM
COMPANY**

ATOMIC ENERGY DIVISION

**NATIONAL REACTOR TESTING STATION
US ATOMIC ENERGY COMMISSION**

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FUNDAMENTALS IN THE OPERATION OF NUCLEAR TEST REACTORS

Vol. 1 Reactor Science and Technology	IDO-16871-1
Vol. 2 Materials Testing Reactor Design and Operation	IDO-16871-2
Vol. 3 Engineering Test Reactor Design and Operation	IDO-16871-3
Vol. 4 Engineering and Design Considerations for In-pile Tests .	IDO-16871-4

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PHILLIPS
PETROLEUM
COMPANY

Atomic Energy Division

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SUMMARY

The purpose of this volume is to discuss the various aspects of the experimental programs at the Materials Testing Reactor, the Engineering Test Reactor, and the Advanced Test Reactor and to review some of the engineering and design fundamentals associated with the programs.

For the most part, the experiments inserted into the reactors are part of the AEC-sponsored activities which are intended to further the available knowledge about the effects of radiation fields on the materials and structural elements to be used in nuclear reactors. The programs are initiated by sponsoring laboratories or private industrial firms throughout the nation which are working under AEC contracts in development work for a certain phase of the nuclear field.

After a sponsor has received AEC approval for a specific program, negotiations for insertion of an experiment are made through the Project Engineering Branch. A project engineer is assigned to the project and acts as liaison between the sponsor and the reactor operators. It is the responsibility of the project engineer to work with the sponsor in designing the experiment and to aid in arranging the scheduling of the experiment into the reactor. The project engineer must inspect the experiment to be certain that all safety and size requirements are met. After an experiment is inserted into the reactor, its progress is noted by the project engineer, and in the absence of a sponsor's representative at the TRA site, data concerning the experiment is forwarded to the sponsor's home office. After an irradiation program is completed the project engineer makes arrangements for the ultimate disposal of the experimental apparatus.

Chapters I through IV review the principles directly involved when a project engineer makes calculations relating to an experimental program. Most of the fundamentals used in these chapters are covered in previous volumes; however, the material presented here relates the fundamentals more directly to the problems faced in an experimental program. The various terms and calculations which are peculiar to the nuclear field are presented in Chapter I, while the principles of fluid flow and heat transfer are related in Chapters II and III, respectively. Chapter IV discusses shielding from the experimental viewpoint and presents many of the specified maximum allowable limits for the various types of radiation and suggests methods and materials for shielding personnel from the radiation.

Chapters V and VI are concerned directly with the design of in-pile and out-of-pile equipment, respectively. Many of the conditions and requirements which must be met before an experiment can be irradiated at TRA are discussed, and procedures peculiar to these reactors are set forth. The many references included in Chapters V and VI are of particular interest to the design engineer.

Chapters VII through IX are concerned with sample handling, instrumentation and decontamination. The problems associated with the handling of experimental equipment are particularly acute after the experiment has been irradiated due to the possible high levels of induced radioactivity. Chapter VII discusses several of the methods of manipulating and transferring the radioactive components to assure minimum exposure of personnel to radiation. The instrumentation of the experiment, as discussed in Chapter VIII, must have provisions for reducing reactor power if conditions might arise which could have deleterious effects on the reactor or experiment systems. In addition, sufficient instrumentation to permit control

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IV	Shielding	F. R. Vandewiele
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VII	Loop Test Train and In-Pile Tube Handling	S. L. Friederichs
VIII	Instrumentation	L. H. Jones
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X	Capsule Irradiations	A. S. Richardson
XI	Water Loops	E. Hylsky
XII	Gas Loops and Gas Leads	C. V. Mullen
XIII	Liquid Metals and Fused Salts	L. H. Jones
XIV	Hazards Discussion	H. D. Gronbeck
XV	Shipping Radioactive Material	R. L. Ackaret
	Appendix A: Sample Capsule Experiment Calculations	L. H. Jones
	Appendix B: Sample Loop Experiment Calculations	L. H. Jones
	Appendix C: Gas System Heat Transfer Analysis	F. R. Vandewiele
	Appendix D: Hot-Spot, Hot-Channel Analysis for a Fueled Test Assembly	H. D. Gronbeck
	Appendix E: Engineering Part Numbering System	D. E. Purkey
	Appendix F: Calculation at Primary Loop Pressure Drop	D. E. Purkey
	Appendix G: Piping Stress Design Examples	D. E. Purkey
	Appendix H: Sample Calculations: Rupture Disc and Relief Valve Sizing	D. E. Purkey

of the experiment parameters, such as pressure, flow, and temperature, must be provided. Chapter IX discusses decontamination facilities as they apply to the experimental systems, particularly loop-type experiments. It is occasionally necessary to decontaminate loop equipment if the radiation levels throughout the system become excessive. A common cause of excessive contamination is a fission break occurring when a fueled sample ruptures or decomposes and fission products are spread through the system.

Chapters X through XII discuss the specific types of experimental programs: capsules, water loops, and gas loops. In general, capsule irradiations are the least complicated of the three since the sample, either with or without instrument leads, is placed directly in the reactor experimental facilities. For loop experiments, a specified environment is established for the sample by providing heating, cooling, pressure level, coolant type, etc. Loop experiments are designed to approximate the actual conditions to which the material under investigation will be subjected after the test phase of the program is completed. When considering the various instrumentation, auxiliaries, safeties, and data handling systems, loop experiments may approach the complexity of the reactors themselves.

Chapter XIII presents a brief discussion of the applications of liquid metals and fused salts to the nuclear field. Specific properties of various substances are noted as well as the precautions required in experimental systems involving them.

Chapter XIV discusses hazards, and ties much of the material in the volume together by describing the consideration given to the experimental program as to how it might affect the reactor system and the operating and maintenance personnel. In the hazards analysis of a particular program, the accuracy of the calculations and approximations are evaluated to ensure operation in the safest possible manner.

Regulations and requirements pertinent to shipping radioactive materials both on-site and off-site are discussed in Chapter XV.

Sample calculations and analyses based on actual experiments are presented in the first four Appendices. Appendix A considers the calculations required for a typical capsule experiment and includes such items as heat generation, heat transfer and shielding requirements. The calculations for a typical loop experiment are presented in Appendix B with much the same approach as is used in the capsule problem. A gas system heat transfer analysis is presented in Appendix C. In Appendix D, a sample "hot spot-hot channel" analysis is made to estimate the uncertainties to be expected in estimation of critical temperature conditions. Appendixes E, F, G, and H present calculations and information pertinent to the design considerations given in Chapter VI.

No attempt has been made in this volume to discuss the particular features of individual experiments, since this information is found in operating manuals for the programs. It is hoped that this volume will serve as a guide to those persons interested in the experimental irradiation programs at the TRA.

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CHAPTER I

EXPERIMENTAL NUCLEAR CONSIDERATIONS

A. INTRODUCTION

Experiments conducted at the MTR and ETR are designed to meet one or more of the following objectives: isotope production, radiation damage studies, and fuel element testing and development.

Total neutrons, neutron flux, and neutron energy requirements for various experimental programs vary widely. Radiation damage studies require high fast-neutron fluxes; isotope production requires high thermal-neutron fluxes. Fuel element development programs usually specify the fast and thermal fluxes which are desired in order to duplicate the fluxes expected in a proposed reactor.

Each facility in the MTR and the ETR has its own peculiar neutron flux and neutron energy spectrum. The experiment requirements can usually be met in some facility, although locations possessing extremely high fast- and thermal-neutron fluxes are oversubscribed. High fast fluxes and high ratios of fast to thermal neutrons are found in the reactor core. Highest thermal fluxes are found in the core; however, since these positions are usually spoken for, the available high thermal flux positions are in the beryllium reflector. Highest thermal to fast flux ratios are found in the aluminum reflector.

Each experiment assembly must be studied in great detail to be sure that the experimenter gets the neutrons requested, that the operation is safe, and that the experiment does not adversely affect the reactor.

Chapters I and II, Volume 1 of this text define the terms and basic equations used in this chapter. The basic equations generally used in evaluating the nuclear characteristics of each experiment will

be given (with examples) in this chapter. Basic features of various irradiation programs will be described.

B. ISOTOPE PRODUCTION

1. Introduction

An important group of experiments at the MTR and ETR is concerned with the production of isotopes. In this type of experiment the target material is placed in a neutron flux zone where it absorbs neutrons according to its absorption cross section. Each neutron absorption produces an isotope of the target material having a mass number which is one greater than that of the target. Many of these new isotopes are radioactive and, in turn, produce other isotopes by radioactive decay. An example of a typical isotope production reaction is the production of radioactive cobalt-60 for medical use:

Most radioisotopes are produced in capsule-type experiments where the target is encased in a suitable container and exposed directly to the process water. Since it is primarily thermal neutrons which contribute to the activity of the sample, gamma heat and fast neutron flux are usually not specified.

2. Activation Equations

The equations used for computing isotope production for any irradiation time or decay time are derived below.

Consider a target or parent material which produces other isotopes by either neutron capture or radioactive decay. The differential equations for a series of nuclear reactions are given by:

$$\frac{dN_1}{dt} = -N_1 \lambda_1 \quad (1-1)$$

$$\frac{dN_2}{dt} = N_1 \lambda'_1 - N_2 \lambda_2 \quad (1-2)$$

$$\frac{dN_i}{dt} = N_{i-1} \lambda'_{i-1} - N_i \lambda_i \quad (1-3)$$

where

$\frac{dN_i}{dt}$ = time rate of change in number of atoms of isotope i due to production and removal in atoms/sec

N_i = atoms of isotope i present at a given time

$$= \frac{(\text{wt of } i, \text{ grams}) (6.023 \times 10^{23})}{\text{at. wt of } i}$$

λ_i = removal constant for isotope i including removal by neutron capture, fission, and decay

$$\lambda_c + \lambda_f + \lambda_d$$

λ_c = removal constant for capture = $\sigma_c \phi$

λ_f = removal constant for fission = $\sigma_f \phi$

λ_d = removal constant for decay = $0.693/T_{1/2}$

σ = cross section in cm^2

ϕ = neutron flux in $\text{n/cm}^2 \text{ sec}$

λ'_i = the specific removal constant responsible for producing isotope i - 1, ie, either λ_c , λ_f or λ_d

Both σ and ϕ should pertain to the same neutron energy, usually thermal. The solutions for the differential equations

are obtained by assuming: (a) initial atoms of isotope 1 is N_0 and (b) no atoms of isotopes 2, 3, etc, are present initially:

$$N_1 = N_0 e^{-\lambda_1 t} \quad (1-4)$$

$$N_2 = \frac{\lambda'_1 N_0}{\lambda_2 - \lambda_1} \left[e^{-\lambda_1 t} - e^{-\lambda_2 t} \right] \quad (1-5)$$

$$N_3 = \frac{\lambda'_1 \lambda'_2 N_0}{\lambda_3 - \lambda_1} \left[\frac{e^{-\lambda_1 t} - e^{-\lambda_3 t}}{\lambda_3 - \lambda_1} - \frac{e^{-\lambda_2 t} - e^{-\lambda_3 t}}{\lambda_3 - \lambda_2} \right] \quad (1-6)$$

In these equations, t is the elapsed time. The equations can be used for materials either in or out of the reactor by using the appropriate removal constants. Additional equations can be derived for equations involving more isotopes.

In many cases the activity of the irradiated sample is desired. This is given by the activation equation:

$$A = \frac{N_{av} m \sigma \phi}{C \text{ at. wt}} (1 - e^{-\lambda t_1}) e^{-\lambda t_2} \quad (1-7)$$

where

A = specific activity, curies

N_{av} = Avogadro's number, 0.6023×10^{24} atoms/g atom

m = mass of target, grams

σ = cross section of target, cm^2

ϕ = neutron flux, $\text{n/cm}^2 \text{ sec}$

C = conversion from disintegrations/sec to curies, 3.7×10^{10}

at. wt = atomic weight of target, grams/g atom

- t_1 = irradiation time, seconds
 λ = decay constant = $0.693/T_{1/2}$,
sec⁻¹
 $T_{1/2}$ = half life of daughter
 t_2 = cooling or decay time (same units
as $T_{1/2}$).

For a simple irradiation at constant neutron flux, a saturation activity will be reached when t_1 becomes large. In order to increase the activity above this level, the sample must be moved to a higher flux region. The saturation activity is given by:

$$\begin{aligned}
 A_s &= \frac{N \sigma \phi}{3.7 \times 10^{10}} \quad (1-8) \\
 &= \frac{0.6023 \times 10^{24} \text{ m } \sigma \phi}{3.7 \times 10^{10} \times \text{at.wt}}
 \end{aligned}$$

where

A_s = saturation activity, curies

N = number of target atoms present

$$= \frac{0.6023 \times 10^{24} \text{ m}}{\text{at.wt}}$$

All other variables have the same units as in the activation equation (1-7).

C. MATERIALS TESTING

1. Introduction

A large number of the experiments run in the MTR and ETR are designed to test the effects of radiation on materials which are to be used in or near radiation fields. These experiments usually fall into one of two groups: radiation damage tests and corrosion tests.

2. Radiation Damage Tests

A discussion of the various types of

radiation damage to metals is presented in Chapter V, Volume 1. Structural metals are tested by irradiating hardness, impact, and tensile specimens. The metal testing programs require the highest available fast neutron (>1 MeV) fluxes in order to evaluate the results of the radiation damage studies in a reasonable time.

Two metals used in reactor structures undergo (n, α) reactions. These are boron, which is used as a poison in control rods and fuel elements, and beryllium, which is used as reflector material. These two metals suffer considerable damage from the helium gas formed from the alpha emission. Radiation damage studies on these two materials require high thermal neutron fluxes.

Most organic materials are seriously affected by neutron bombardment; they are not usually used in high neutron fields. However, materials for gaskets, seals, and lubricants are sometimes exposed to high gamma fields. These items are frequently tested in pure gamma fields furnished by irradiated fuel elements in the MTR Gamma Facility.

Organic materials which show promise as reactor coolants and moderators (eg, diphenyls) have been tested in capsules inserted in high thermal and fast neutron fluxes.

3. Corrosion Tests

Special tests may be designed to determine the effect of intense radiation on the corrosion resistance of a metal-to-reactor coolant. These tests may be run in capsules or in loop facilities.

The capsule tests usually consist of a piece of metal suspended in the coolant. A number of capsules will be irradiated to high integrated flux (nvt) and the amount of corrosion measured. Other capsules will be run in autoclaves at identical temperature conditions. The difference in corrosion rate can then be measured, evaluated, and attributed to the irradiation.

Loop tests usually consist of two identical assemblies located in a high-pressure water loop. The control section is in a region of zero neutron flux. The other section is placed in the high flux section of the loop. Differences in corrosion rate can be attributed to the neutron flux since all other conditions are identical.

The most important test parameters for corrosion study are water temperature, velocity, and specimen surface temperature.

D. FUEL ELEMENT DEVELOPMENT

1. Introduction

A major effort in the nuclear program is toward development of fuel elements that are stable under irradiation and clad with material that is resistant to corrosion at high temperatures.

Fuel elements can be fabricated from many materials including metallic uranium, uranium oxides, and carbides of uranium. Uranium enrichment runs from natural uranium to about 93 percent U-235.

The specific fuel enrichment, form of uranium used, matrix material, cladding material, and type of fabrication used for any given reactor depend largely upon the specific purpose of the reactor.

2. Fuel Element Fabrication

Fuel element configuration has generally been reduced to two types: plate-type elements and pin-type elements. Fuel element fabrication is essentially two processes: fuel matrix fabrication and cladding.

a. Plate-Type Elements

The fuel matrix material for plate-type elements is generally prepared by melting an ingot in an arc furnace and then rolling or machining it to the proper size.

Cladding can then be applied in one of three ways: roll bonding, pressure bonding, or diffusion bonding. Roll bonding and pressure bonding are essentially the same except for the method of applying the pressure. With roll bonding the preheated element is passed several times through rollers which apply high pressures until the final dimensions are met. Under pressure bonding, the bond is made by autoclaving at the temperature and pressure required for a specified time. In either case, a metallurgical bond is obtained between the clad and the matrix. With the diffusion bonding process, the clad is applied to the matrix by first applying a thin plating of a material by electrochemical plating and then plating the clad to that.

b. Pin-Type Elements

Fuel matrix for pin-type fuel elements may take several different forms, the most common being rods and pellets. Rod elements may be clad by extrusion through a die or by inserting the matrix rod into a tube of the cladding material and drawing the tubing down so that tight contact between clad and fuel matrix is obtained. If the fuel matrix is in the form of pellets, these can be stacked into the tubing, the heat transfer medium inserted, and the ends sealed. The heat transfer medium may be gas, metal, or NaK.

3. Nuclear Requirements

A large number of the fuel element development programs conduct preliminary studies of dimensional stability of fuel and matrix materials as a function of neutron flux and fuel burnup. These experiments are usually inserted in capsules in regions of high flux to ascertain the properties before more extensive controlled tests are undertaken. Neutron flux and fissioning rate must be predicted accurately so that the radiation damage as a function of nvt and power generation per unit volume can be determined. By comparing several samples with different exposures and power generation rates,

the materials can be evaluated and operating limits set.

Another type of fuel element development program incorporates the controlled parameters of a loop in order to investigate new cladding materials, new bonding techniques, effect of bonds on heat transfer, effect of failure, and many other parameters.

A proof test in a mock-up of an element proposed for a given reactor must be performed under conditions of controlled environment in a loop facility. Tests of this type have as their goal the proof of design, fabrication technique, inspection, specifications, and general reliability.

E. NEUTRON FLUX

1. Introduction

The neutron flux in a test reactor has a considerable range of energy values varying from fission neutrons in the MeV range to thermal neutrons having an energy of a fraction of an eV. Experiments conducted at MTR and ETR are usually designed so that neutron flux in one energy range may be of more interest than the remaining energy levels. The experimental position in the reactor (core or reflector) has an effect on the ratio of one neutron energy group to another. Other ways of affecting this ratio within the experiment are by the use of hafnium or boron shrouds and by varying the moderating ratio in the experiment.

2. Thermal Flux

Thermal neutrons are defined as neutrons that are in thermal equilibrium with the atoms or medium in which they are being scattered. Neutron flux is a measure of the neutron intensity in a specified energy range and, at MTR and ETR, this range of energies for thermal flux is from 0.0 up to 0.4 eV. The energies of the neutrons within the spectrum are

assumed to be on a Maxwell-Boltzmann distribution. Thermal fluxes available to experiments in the MTR and ETR range in value to $\approx 10 \times 10^{14}$ n/cm² sec (unperturbed). (See Figure 1.)

3. Fast Flux

Fast flux includes all neutrons with an energy of 1 MeV or greater, although sometimes fast flux is considered to include all neutrons with energies greater than 100 eV. This latter figure is used for radiation damage studies, while the 1 MeV base is used when considering fast fissioning in U-238. The fast flux (>1 MeV) available at MTR and ETR in the core experimental positions is as high as 3.8×10^{14} n/cm² sec (see Figure 2).

4. Epithermal Flux

Between the values of 0.4 eV and 1 MeV is a group of neutrons undergoing moderation that are called epithermal neutrons. As mentioned above, these become important during radiation damage studies. There are no known monitors to directly measure the epithermal flux as can be done for thermal and fast fluxes. The epithermal flux must be calculated; to do this, both thermal flux and fast flux must be known. The method of calculation is presented in IDO-16614[1]. The ratio of epithermal to fast flux is not constant throughout the reactor and must be calculated for each position.

5. Gamma Heat

One important source of heat in some test positions is gamma heat. Approximately five percent of the energy released in the fission process is released as gamma ray energy, both prompt and delayed. It is the absorption of this energy in the reactor in-pile components that gives rise to gamma heating. The gamma heat is usually expressed as watts per gram. To calculate the gamma heating in a position of the reactor, it is necessary to know the thermal flux and the composition of the area.

Fig. 1 ETR thermal flux, cycle 27 (10^{14} n/cm²-sec).

Fig. 2 ETR fast flux above 1 MeV, cycle 27 (10^{14} n/cm²-sec).

Gamma flux can be measured by use of gamma film and by radiochemical means (see Figure 3).

Fig. 3 ETR tank cross section gamma heat map, cycle 27 (w/g of graphite).

6. Flux Measurement

The flux available and its spectrum in the experimental positions are periodically measured and reported in one of two forms: a flux map for the entire reactor (Figures 1 and 2), or as a distribution in a facility (Figure 4). In either case, a careful check is necessary when using these values for design or safety analysis, since many factors affect them.

The thermal flux in a facility is measured by inserting both bare cobalt wires and cadmium covered cobalt wires into the facility and activating them by a low power reactor run for a specified length of time. Both the bare and covered monitors are then removed and their activity counted. The bare monitor will be activated by neutrons of all energies, while the cadmium covered monitor will be activated only by neutrons of an energy greater than 0.4 eV. By subtracting the

activation by epithermal and fast neutrons from the total activation and by use of the activation equation, the thermal flux is computed.

Fig. 4 Thermal neutron flux map, ETR position G-6-NW.

Fast flux in a facility is measured by the activation on a nickel wire. Here the activation is produced mainly by neutrons with an energy greater than 1 MeV. The monitor is counted and the flux is computed from the activation equation.

Both low power flux runs and integrated flux runs are made. Low power flux runs are generally made at N_L (1 percent of full power) for 30 minutes. The flux at N_F (full power) is then scaled up from the N_L flux as a direct proportion. An integrated flux run is one that is made over a cycle or a longer period of time. This method is generally used when the total nvt (integrated flux) that an experiment has acquired is desired. The flux monitors from an N_L run can be

handled much easier than those from the integrated run.

From the various N_L flux runs and integrated runs, a flux map of the reactor is made. The horizontal variation in flux across the core is plotted as midplane flux. From these plots, lines of constant flux can be drawn, and some estimate of the flux value in an unplotted position can be made. The vertical values of flux can be obtained by applying a cosine distribution to the midplane value.

Another type of flux plot that is reported is the vertical distribution in a facility (see Figure 4). A full length wire is activated in the facility and then it is cut and counted so that the vertical distribution of flux is obtained.

Both methods of flux reporting give values for the core loading, the rod positions, and the amount of poisoning that existed at the time the run was made. It may or may not be a maximum value for the cycle and may change considerably from cycle to cycle.

7. Flux Corrections

Before the values of fluxes as measured in the reactor can be used effectively by the experimenter, certain corrections must be made.

Since the value of the cross section for U-235 is dependent on the energy of the incident neutrons, it may be necessary to apply a $1/V$ correction to the cross section. In experiments where a low metal-to-water ratio exists and the water temperature is higher than the reactor system, the $1/V$ correction should be applied. In gas loops or NaK loops where Σ_g is relatively small, the correction is neglected.

Another correction to the measured value of reactor flux is that due to flux perturbation and attenuation. Perturbation is the decrease in flux value when a neutron absorbing material is placed in

the region. Attenuation is the decrease in flux through a sample from the outside edge toward the center. The Nucleonics article by W.B. Lewis[2] gives methods for calculating both perturbation and attenuation factors. This method is used in Appendix A for calculation of effective neutron flux in a typical capsule experiment.

Spectrum hardening, while basically a $1/V$ problem, becomes more complicated as the experiments get more complex. Spectrum hardening can be considered as the preferential absorption of the lower energy neutrons while traveling through a region. As an example, in some of the larger water loop experiments, the spectrum is first hardened as the flux passes through a fairly big pressure tube. It next enters a region of high temperature water where the scattering collisions raise the energy of the slower neutrons. After passing through the water gap, the flux passes through a hafnium shroud where again the lower energy neutrons are absorbed. When the flux finally reaches the fuel, its energy distribution is different from the energy distribution of the flux in the reactor.

For such experiments, it is no longer possible to compute "effective fluxes" or "effective cross sections" by homogenizing the region and applying perturbation theory. Here, machine calculations are used whereby the experimental components are represented explicitly. The PDQ code is a two-dimensional program written for the IBM 704 computer that gives effective flux values in the x and y directions as the experiment is conducted. The program is repeated at different elevations to get a complete distribution through the experiment. Also, by using machine calculations, the neutron flux can be considered in several groups. One method that is used on ETR experiments considers the neutron energy in two groups: one from thermal up to 0.625 eV, and the fast group above 0.625 eV.

8. Flux Variation

There are reasons other than reactor loading that cause the flux in a facility to vary. As fuel is burned up during the cycle, fission products with high cross sections accumulate; the flux must increase to maintain the required fissioning rate. Since the experiments are primarily dependent on the reactor neutrons, they must be designed to handle this flux increase. Changes in power generation in experiments of 50 percent have been attributed to this increase during the reactor cycle.

Another important cause of flux variation is that due to rod movement. Here again, as fuel is burned up new fuel is pulled into the core by rod movement. This causes a higher fissioning rate in the immediate area of the rod -- in essence, a shift in the power density of the core. Here again, the experimental power generation may be affected. Changes up to 15 percent have been attributed to rod movement.

9. Effective Flux Determination

As can be seen, the determinations of effective neutron flux and fissioning rate have many uncertainties. The errors associated with measurements, reactor power, and other parameters can be as much as 20 percent. Measuring the actual power generation of a fueled sample gives an accurate method of determining what the effective flux is, and a radiochemical analysis of the irradiated sample will determine what the accumulated nvt on the sample was.

Power generation of a sample can be determined in two ways: by measurement of the bulk coolant rise across the element, or by use of instrumented fuel plates. In order to use the first method, the bulk coolant temperature rise must be large enough to be measured accurately. This method gives the average power generation rate, and a correction factor must be applied to get the maximum rate.

The instrumented fuel plate is used for small assemblies where the total power generation is small but the heat flux on the plate is large. The thermocouple is located on the fuel assembly in such a position that there is no heat flow across it, although this may not be the hottest spot on the assembly. From the known heat conduction coefficients of the alloys and matrices and the coolant film rise, the power generation rate is calculated.

A radiochemical analysis of the sample may be performed to determine the quantity of the various fission products present. From this, the number of fissions can be determined.

By applying the equation (1-10) developed in the next section with a known power generation and fuel mass, the effective neutron flux can be obtained.

F. POWER GENERATION CALCULATIONS

The basis for calculating fission power generation is the reaction rate equation:

$$\begin{aligned} \frac{dN}{dt} &= N\lambda \\ &= \frac{mN_{av}\sigma\phi}{A} \end{aligned} \quad (1-9)$$

where

dN/dt is the reaction rate in events/sec and the other variables are as defined for Equation (1-7). By substituting the fission cross section and atomic weight of U-235, assuming that 200 MeV are released per fission, and applying conversion factors to obtain watts, the expression for finding the power generated from thermal neutron fissioning of U-235 is obtained:

$$P = 4.78 \times 10^{-11} m\phi \quad (1-10)$$

where

P = power generation, watts

m = mass of U-235 irradiated, grams

ϕ = effective thermal neutron flux, n/cm² sec.

G. SOURCE STRENGTH

Source strength formulas are important for activation calculations and for calculating fission product energy release in order to estimate shielding requirements and afterheat.

1. Activation

The basic activation equation (1-7) is used to determine the activity induced in samples, structural materials, and coolants by neutron flux. For samples and structural materials, the irradiation time, t_1 , is the actual time that the reactor is at power. For coolants which circulate through the experiment, the irradiation time is that portion of the time spent in the neutron flux. The remainder of the circulation time is considered as decay time, t_2 .

The source strength due to gamma rays from fission is determined by multiplying the fission rate by the gamma energy release per fission[3]. For U-235, 1 watt of power requires approximately 3.1×10^{10} fissions/sec, and each fission liberates approximately 20 MeV of gamma radiation. The source strength is, therefore, given by:

$$S_{\gamma} = (3.1 \times 10^{10}) (20)P \quad (1-11)$$
$$= 6.2 \times 10^{11}P$$

where

S_{γ} = gamma source strength from fissioning U-235, MeV/sec

P = power generation, watts.

2. Fission Products

In the case of fueled samples, the problem becomes complex after a short period of irradiation. This is due to the many different fission products present, each decaying with its own half life, decay scheme, and energy of particle or photon emission. The rate of energy release, due to the decay of fission products including both beta and gamma energy, is given approximately by the Way-Wigner equation [4]:

$$FP \approx 5.9 \times 10^{-3} P \left[(t - T_0)^{-0.2} - t^{-0.2} \right]$$

where

FP = rate of emission, watts

P = reactor or experiment power, watts

t = total time after reactor start-up (equivalent to $t_1 + t_2$), days

T_0 = irradiation time (equivalent to t_1), days.

This equation is used to determine the afterheat in a sample at any time after irradiation. It should be noted that Equation (1-12) does not give the energy of the emitted particles or photons. For shielding purposes, the energies of all fission products are divided into energy groups as discussed in Chapter IV.

Curves presented in IDO-16453 [5] and in Figure 16 greatly simplify source strength calculations.

H. REFERENCES FOR NUCLEAR CONSTANTS

The various constants for isotopes to be used in the equations are compiled in several places. Considerable information regarding cross section, half-life, decay scheme, decay energy, fission yield,

etc, is presented in the Trilinear Chart of Nuclides^[6]. A tabulation including density, cross sections, mean-free paths, diffusion length, and diffusion coefficient is found in the November 1960 issue of

Nucleonics^[7]. The same issue of Nucleonics includes a Reference Data Manual (pages 147 to 210) which contains considerable information for use in nuclear calculations.

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CHAPTER II

FLUID FLOW

A. INTRODUCTION

Fluid flow calculations have two basic objects: (a) to calculate fluid velocity and equivalent diameter for heat transfer calculations; and (b) to evaluate the possibility of buckling of structures due to unequal pressure drops in parallel channels. The first consideration is the only consideration in over 90 percent of the experimental fluid flow systems.

Three types of fluids are considered in this chapter: water, gases, and liquid metals. Basic fluid flow formulas cover all three fluids adequately although air, being a compressible fluid, has more complicated relationships at high velocities. The liquid metal properties are similar to water.

Fluid flow calculations are usually not accurate enough for the heat transfer problems in experimental fuel elements. Whenever possible, flow tests are run on fuel element assemblies in order to define the hydraulic variables involved more accurately.

This chapter will discuss the typical hydraulic problems involved in capsule experiments, high pressure water loops, gas loops, and liquid metal loops.

B. CAPSULES

1. Reactor Facilities Available

Capsule experiments in the MTR and ETR are usually inserted in metal baskets to aid in handling [1]. These are inserted into a position in one of the reactors where the neutron flux is satisfactory for the proposed test.

The basket may be an X-basket, L-basket, D-basket, S-basket, or N-

basket. The reactor position may be either an aluminum reflector piece or a beryllium core piece in the MTR; in the ETR, the reactor position may be a core filler piece, an aluminum reflector piece or a beryllium reflector piece. The critical dimensions and the gauge size for all capsule handling facilities are given in Table I. Table II gives the hydraulic parameters of the MTR and the ETR reactor cooling water, which cools the capsules.

For heat transfer considerations, the engineer must know the total flow of water past the capsule, the water velocity, and the equivalent diameter. Since the pressure drop across the reactor core is a known (relatively constant) value, flow can be calculated by basic hydraulic formulas given in Volume 1, Chapter III, paragraph B.

2. Basic Hydraulic Equations

The most important formulas are those dealing with entrance and exit losses and the friction drop formula [2] which are shown below:

- a. Entrance losses (sudden contraction)

$$h_L = K_c \frac{V_2^2}{2g} \quad (2-1)$$

where

h_L = head loss, ft of fluid

K_c = resistance coefficient

V_2 = mean velocity in smaller channel, ft/sec

g = acceleration of gravity, ft/sec² (32.2).

TABLE I

CRITICAL DIMENSIONS FOR FACILITIES AND CAPSULES

<u>MTR</u>	<u>Minimum Facility Diameter (inches)</u>	<u>Maximum Capsule Diameter (inches)</u>
LB one-hole L piece	1.375	1-5/16
LB 4X four-hole L piece	1.312	1-1/4
LB 4W four-hole L piece	0.873	13/16
AA one-hole A piece	1.375	1-5/16
AA 2W two-hole A piece	1.553	1-15/32
AA 4X four-hole A piece	1.312	1-1/4
DA8D eight-hole D piece	1.375	1-5/16
Hydraulic Rabbit VH-1 and VH-2	0.995	13/16
Hydraulic Rabbit VH-3 and VH-4	1.308	1-1/8
Pneumatic Rabbit HR-1, HR-2, T2H2	---	0.950
X-basket	1.184	1-1/8
L-basket	1.184	1-1/8
D-basket	1.184	1-1/8
S-basket	0.552	5/8
N-basket	0.652	5/8
<u>ETR</u>		
One-hole A piece	1.553	1-15/32
Four-hole A piece	1.312	1-1/4
Four-hole core filler piece	1.312	1-1/4
Beryllium reflector piece	1.312	1-13/64
X-basket	1.184	1-1/8
B-basket	1.060	1

TABLE II

MTR AND ETR COOLANT PARAMETERS

	MTR	ETR
Inlet water temperature (°F)	110 ± 5	110
Exit water temperature (°F)	122	130
Inlet water pressure (psia)	62.5	200
Exit water pressure (psia)	23.5	152

The resistance coefficient is calculated as follows:

$$\frac{A_2}{A_1} < 0.715 \quad K_c = 0.4 \left(1.25 - \frac{A_2}{A_1} \right)$$

when

$$\frac{A_2}{A_1} > 0.715 \quad K_c = 0.75 \left(1.00 - \frac{A_2}{A_1} \right)$$

where A_1 is the larger flow area, and A_2 is the smaller flow area. Under certain

conditions the ratio $\frac{V_1}{V_2}$ must be used rather than $\frac{A_2}{A_1}$ since this compensates

for variations in fluid properties and multiple channel flow.

b. Friction Pressure Drop

$$\Delta P = \frac{fL \rho V^2}{2g De (144)} \quad (2-2)$$

where

ΔP = pressure drop, psi

f = friction factor

L = length of channel, ft

ρ = density of fluid, lb/ft³

V = average fluid velocity, ft/sec

g = acceleration of gravity, ft/sec²
(32.2)

D_e = equivalent diameter, ft.

c. Exit Losses (sudden enlargement)

$$h_L = K_e \frac{V_1^2}{2g} = \frac{(V_1 - V_2)^2}{2g} \quad (2-3)$$

where

h_L = head loss, ft of fluid

K_e = loss coefficient = $\left(1 - \frac{A_1}{A_2} \right)^2$

V_1 = mean velocity in smaller channel, ft/sec

V_2 = mean velocity in larger channel, ft/sec

g = acceleration of gravity, ft/sec²
(32.2)

A_1 = smaller flow area

A_2 = larger flow area.

The head losses due to entrance and exit effects can be put in terms of pressure drop by the following relationship:

$$\Delta P = \frac{\rho h_L}{144} \quad (2-4)$$

where

ΔP = pressure drop, psi

ρ = density of fluid, lb/ft³

h_L = head loss, feet of fluid.

The parameter V , average fluid velocity, can be calculated from the flow rate in ft³/sec, and the cross sectional flow area by the following relationship:

$$V = \frac{F}{A} \quad (2-5)$$

where

V = velocity, ft/sec

F = flow rate, ft³/sec

A = cross sectional area, ft².

The more useful form of the above formula is:

$$V = 0.321 \frac{W}{A} \quad (2-5a)$$

where

V = velocity, ft/sec

W = flow rate, gpm

A = cross sectional area, in.².

Equivalent diameter, D_e , is a conversion factor used in hydraulic equations to enable the engineer to use rather simple formulas on complicated hydraulic shapes:

$$D_e = \frac{4A}{wp} \quad (2-6)$$

where

D_e = equivalent diameter, ft

A = cross section of flow area, ft²

wp = wetted perimeter of flow area, ft.

For circular flow areas,

D_e = diameter of flow area

For annuli between two cylinders,

$$D_e = D_1 - D_2$$

where

D_e = equivalent diameter

D_1 = outside diameter of flow area

D_2 = inside diameter of flow area.

For annuli between parallel plates where the length of the flow channel is large compared to the width, $D_e = 2$ (channel width).

The factor, f , in the friction loss formula [3] must be calculated from the curves shown in Figures 5 and 6. The curve in Figure 5 plots Reynolds number, $\frac{VD\rho}{\mu}$, versus VD_e at various temperatures

where VD_e is known. With the Reynolds number known, the friction factor is plotted against Reynolds number for various pipe roughnesses. For almost all experimental capsules, the curve for smooth pipes may be used.

Fig. 5 Reynolds number vs velocity times equivalent diameter.

Fig. 6 Friction factor vs Reynolds number.

3. Calculation of Flow and Pressure Drop

In the case where one sample capsule six inches long is located in an X-basket, the flow problem can be separated into the following components: (a) a pressure drop for entrance loss at the top of the X-basket; (b) a pressure drop for entrance loss at the top of the capsule; (c) an exit loss at the bottom of the capsule; (d) an exit loss at the bottom of the X-basket; (e) an exit loss at the bottom of the experiment tube; (f) friction pressure drops in the X-basket above and below the capsule; and (g) friction pressure drop in the annulus between the capsule and X-basket.

Since total pressure drop is known, the engineer must assume a flow rate through the system. With this flow rate and a cross section area of the flow path, the engineer must calculate a water velocity for each segment of flow channel mentioned above. Equivalent diameter of each flow segment must also be calculated.

Entrance, exit, and frictional pressure drops are calculated for each flow segment above using the basic formulas previously described. The sum of the pressure drops should be equal to the known total pressure drop. If it is not, the calculations must be repeated with another assumed flow. Trial and error calculations continue until the total pressure drop at the assumed flow equals the known pressure drop.

In the ETR, some experimental in-pile tubes pass through the core and grid plate. The outer surfaces of these tubes must be designed so that the flow channel between the adjacent reactor fuel element plate and the experiment in-pile tube assembly is similar to the flow channel between fuel plates in a fuel element. This is necessary to prevent buckling of the adjacent fuel plates by lateral pressure differentials.

4. Use of Experimental Curves

Calculation of flow by the trial and error method is a laborious process, and total accuracy is not better than plus or minus 30 percent. The hydraulic systems in most capsule experiments are similar, and the pressure drops across the facility piece (A- or L-piece) are constant. A series of hydraulic tests on various assemblies of capsule baskets and experimental pieces used at MTR and ETR were run with the results published in IDO-16516 [4]. With the aid of curves presented in this document, reactor coolant flow around capsules may be calculated with an accuracy of plus or minus 10 percent by a relatively simple method.

Figures 7, 8, and 9 are curves correlating hydraulic parameters obtained in hydraulic tests. Figure 10 indicates some of the special capsule configurations used in these tests. The procedure for solution of a capsule hydraulics problem follows: (a) calculate the total cross sectional flow area within the X-basket, (b) from Figure 7 or 8, depending on

Fig. 7 Flow through full X-basket vs cross sectional flow area.

Fig. 8 Total flow through X-basket vs percent of X-basket loaded.

Fig. 9 Coolant velocity in annulus between X-basket and 1.125 in. OD capsules vs total flow through X-basket.

Fig. 10 Dummy capsules used for tests in single hole aluminum reflector piece.

the percent of X-basket loaded, obtain the flow rate through the X-basket; (c) using the flow rate from step (b), obtain the coolant velocity in the annulus from Figure 9, and (d) calculate the coolant velocity in the hole in the capsule from the equation:

$$V_a A_a + V_h A_h = G \quad (2-7)$$

where

V_a = velocity in annulus

V_h = velocity in hole in capsule

G = measured flow rate

A_a = area of annulus

A_h = area of hole in capsule.

Results should be accurate to plus or minus 10 percent which is better than the usual results obtained by calculation.

C. WATER LOOP SAMPLES

1. Introduction

When fuel element configurations must be tested at bulk water and surface temperatures exceeding test reactor conditions, high pressure water loops are used for testing. In 2200 psig water loops, bulk water temperatures up to 600°F are possible. Fuel element surface temperatures may be as high as 650°F.

Most fuel element prototypes irradiated in high pressure water loops are of irregular shape and have several parallel flow channels. Total flow through the sample assembly is automatically controlled at a constant value during the test run, although it can be varied during the operation of the test if necessary.

Pressure drop through the sample and the flow through each flow channel can be calculated for the conditions set up by the test. However, the complex shape

of the flow channels makes calculation difficult and large errors can be introduced.

2. Hydraulic Testing

To minimize the errors inherent in calculation, the fuel element prototypes are tested hydraulically in order to determine the flow down each flow channel. This reduces the possible hydraulics error from plus or minus 30 percent to plus or minus 5 percent.

The general procedure is to measure total pressure drop across the experiment assembly at the flow to be used in the nuclear test and then to block off one flow channel and regulate flow until the same total pressure drop is obtained. The difference in the two flows corresponds to the flow through the channel which was blocked. The flow in each channel can be determined in this manner. Accuracy of the test can be estimated by comparing the sum of the individual channel flows with total flow. This method is accurate when many flow channels exist. The engineer must carefully analyze the hydraulic test system to be sure entrance and exit losses are approximately the same with one channel blocked as with all channels open. The sample problem in Appendix B uses hydraulic information obtained by this process.

When a test specimen is to be irradiated in a loop, neutron flux and, therefore, specimen power generation cannot be predicted with great accuracy. If specimen temperature is a critical test parameter, bulk water flow and temperature can be varied in order to compensate for variation of neutron flux from the prediction. For instance, if actual experiment power generation is 20 percent low, loop bulk water temperature may be raised or loop flow lowered in order to keep the fuel element at the requested temperature. For this reason, the hydraulics of the loop fuel element are always studied for a fairly wide range of flow rates and bulk water temperatures.

3. Flow Restrictors

Some high-pressure water loops have several individual parallel flow channels through the reactor core, with each channel filled with experimental fuel plates. Flow control on each channel may not be practical, but flow must be distributed very accurately. To do this, orifice-type flow restrictors are placed in each flow channel. These are sized so that approximately 90 percent of the total pressure drop in each channel is across the orifices. Since the pressure drop across the relatively complicated test specimens is only a small part of the total pressure drop across each channel, accurate balancing of flow can be accomplished.

The relationship of pressure drop and flow across the orifices is calculated and then carefully checked in hydraulic test facilities.

D. WATER LOOP EXTERNAL EQUIPMENT

1. Design Basis

In the preliminary design of loops, the location of the core facility, approximate access pipe routes, the area reserved for out-of-pile equipment, and the utilities available are specified. After a thorough review of the anticipated testing program, the range of pressures, flows, and bulk water temperatures needed for the tests can be determined. The equipment external to the reactor is then designed to operate under this range of conditions. After all major equipment and piping runs are determined, a pressure drop calculation for various anticipated flow rates must be made for the entire system to be sure the primary water pumps can furnish the flows at the anticipated pressure drops.

2. Pressure Pump Sizing

All pressure drops in the loop must be overcome by the primary pumps. The canned rotor centrifugal pumps used on the

high pressure loops have the ability to pump varying flow rates with a relatively constant pump head. If the pressure drop in the system is more than the rated pump head of approximately 150 psi, two or more pumps are installed in series. If the loop pressure drop is small compared to the pump head and large flows are required, the pumps can be installed in parallel. If maximum flexibility is requested, four pumps may be used in a series-parallel arrangement.

The pressure drop calculations are based on summing the pressure drops due to entrance and exit effects, piping lengths, ells, tees, and other piping fittings. After the loop is installed and before the loop is operated with nuclear power, tests are run to measure the pressure drops in each part of the loop at various flow rates and water temperatures. These tests serve two purposes: (a) to check the accuracy of previous engineering calculations in order to improve the accuracy of future calculations; and (b) to determine accurately the range of flows and temperatures at which the loop will operate.

On all experimental systems where the cooling fluid is circulated, extensive measurements are made correlating flow versus time after complete pump failure. These are needed in order to be sure that the sample will not burn out on a complete loss of pumping activity. With high-speed recording equipment, the actual flow drop and the flow drop as seen by the instruments are recorded. With this information the engineer can set the reactor power reduction setpoints so that the reactor will be scrammed before actual flow drops to the burnout point.

Water loops have a fast coastdown, generally reaching 10 percent of operating flow in 2 to 10 seconds (see Figure 13). Single pass air loops, with large capacity air receivers at the supply end, drop to 10 percent of operating flow in about 4 minutes (see Figure 51).

E. GAS FLOW

1. Basic Formulas

Compressible fluids such as air, gases, and steam have properties which complicate some fluid flow calculations. However, the formula given previously for water,

$$\Delta P = \frac{fL \rho V^2}{2g D_e (144)} \quad (2-8)$$

works equally well for compressible fluids if the pressure drop across the system is less than 40 percent of the inlet pressure. The friction factor, f , being an empirical function, can be used to compensate for differences between gas and liquid flow systems. Values for the friction factor for use in this formula can be obtained from nomographs in Crane Technical Paper No. 410, Flow of Fluids Through Valves, Fittings, and Pipe [5].

If the pressure drop is more than 40 percent of inlet pressure, special empirical formulas are presented in the Crane Paper mentioned above.

2. Mach Numbers

In compressible fluid flow, the density of air decreases as pressure decreases, tending to increase the velocity of the air as it flows through the flow channel. The limiting velocity of air is the velocity of sound, called the acoustic or critical velocity and given by:

$$V_s = \sqrt{144 K_g P V} \quad (2-9)$$

where

V_s = critical velocity of flow of a gas, ft/sec

K = ratio of specific heat at constant pressure to specific heat at constant volume

$$= \frac{C_p}{C_v}$$

g = acceleration of gravity, ft/sec² (32.2)

P = pressure, psia

V = specific volume of fluid, ft³/lb.

A more common term used to denote high velocity air flow rate is Mach number, the ratio between the velocity of the gas and the velocity of the sound:

$$M_a = \frac{V}{V_s} \quad (2-10)$$

where

M_a = Mach number

V = velocity of fluid

V_s = critical velocity of the gas.

In testing facilities at MTR and ETR, the average Mach number is restricted to 0.4; otherwise, local velocities in the test sample may reach sonic speeds. Resultant large local forces could damage the structural members of the test sample.

3. Dynamic Head

Dynamic forces on experimental samples in gas cooled loops are evaluated in terms of dynamic head:

$$q = \frac{kW^2 T}{P} \quad (2-11)$$

where

q = dynamic head, psi

W = mass flow, lb/sec

T = coolant absolute temperature, °R (°F + 459)

P = absolute pressure of coolant, psia

k = constant depending on sample geometry.

This force is the force which tends to deform the sample. Since it depends on gas flow, pressure, and temperature, control of this parameter is very difficult in the test facility. The three basic parameters are measured and dynamic head is calculated.

4. Flow Testing

The least dangerous nuclear test conditions are conditions of low dynamic head and low sample temperature. When a sample is to be put on test, the sample is inserted under conditions of high air pressure, low temperature, and rated flow. This minimizes the sample temperature, dynamic head, and Mach number. The conditions of temperature, pressure, and flow are then adjusted until dynamic head and sample temperatures are at the desired conditions, and the average air velocity is less than Mach 0.4.

It can be shown that the parameter, $\frac{\Delta P}{q}$, is a constant with small variations in air flow, pressure and temperature. $\frac{\Delta P}{q}$ may then be monitored to evaluate distortion of the sample as the test progresses.

A rise in $\frac{\Delta P}{q}$ with flow, temperature, and pressure held relatively constant can be caused only by distortion of the sample affecting the flow pattern of the coolant.

Since the parameters, Mach number, and dynamic head are critical in the op-

eration of test samples, it is necessary to use both high- and low-level alarms on air flow, pressure, and temperature. Mechanical stops on control valves limit pressure or flow fluctuations on failure of the controllers or valves.

Fuel elements in gas loops must have large surface-to-volume ratios because heat transfer coefficients are very low. Structural strength must be sacrificed to good heat transfer characteristics. In order to evaluate the structural strength of the fuel elements under the dynamic conditions found in the reactor test facility, all fuel element configurations are pre-tested extensively in nonnuclear facilities at conditions of air pressure, temperature, velocity, and dynamic head equal to or more severe than the proposed test conditions for the nuclear test.

F. LIQUID METAL FLOW

Most hydraulic formulas and laws can be applied to liquid sodium and NaK, since their properties are similar to water [6]. Other liquid metals are much more dense than water but are easily handled by standard formulas.

The viscosity of NaK is so near that of water that many NaK systems are flow tested with water because it is easier to handle. For a given NaK temperature, a water temperature can be chosen where the viscosity is identical to that of NaK at the operating temperature. Pressure drop and velocity data are taken by flowing water through the system. The NaK information is then obtained by applying a simple density correction to the water data.

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CHAPTER III

HEAT TRANSFER

A. INTRODUCTION

1. Applications of Heat Transfer to Project Engineering

Project Engineering heat transfer considerations fall into two general categories.

The first category involves the heat removal problems of test specimens under irradiation. Before a specimen is inserted into a test reactor, specimen heat removal and temperatures must be evaluated from a safety standpoint to ensure that the irradiation will not result in excessive temperature and possible failure of the sample. Excessive temperatures can result in melting of portions of a specimen or physical distortion through thermal expansion, phase changes, and undesirable changes in other properties. After an experiment has been analyzed from a safety standpoint, the possibility of loss of reactor operating time or loss of experimental data due to undesirable temperatures must be considered.

The second category of heat transfer considerations is functional and becomes important in irradiation programs that require auxiliary equipment. The equipment supporting these programs varies from very simple external components to pilot plants that approach the complexity of the test reactor itself. In evaluating this equipment, the project engineer must consider such factors as heating and cooling capacity, temperature of exposed equipment, heat losses to the environment, bearing temperatures, sample temperatures, and cooling of electrical components and moving parts.

2. Review of Previous Chapter on Heat Transfer

The fundamentals of heat removal from nuclear fuel are discussed in Chapter III, Elements of Reactor Engineering, in Volume 1. These fundamentals apply directly to nuclear test specimens. The application of heat transfer techniques to test specimens may be somewhat complicated, since many of the specimens have irregular and unusual configurations. Heat removal from nonfueled test specimens and specimen holders under irradiation must also be considered. It is not unusual to find that the surface heat flux of a specimen holder is greater than that of the specimen itself. Since nuclear test programs generally support future reactors, test specimens quite frequently operate under conditions more severe than those encountered in existing reactors. In some cases, a test program may be designed to explore the results of excessive temperatures or to define the conditions under which undesirable temperatures occur. Many tests require minimum temperatures in order to produce useful data. For these reasons, a standard approach to test specimen heat transfer analysis is not always desirable.

Because of the many uncertainties associated with heat transfer calculations, predicted specimen temperatures are used only as a starting point for some types of tests. Operating conditions for a test may be dependent upon data produced by special instrumentation. Thermocouples embedded in a test specimen, or coolant temperature differences, often are used as a basis for adjusting operating conditions.

3. Heat Transfer Techniques Peculiar to Sponsors

Engineering judgment must be exercised in selecting the proper heat transfer formulas. Certain types of formulas are based upon experimental data.

If the configuration of the specimen under consideration differs appreciably from that of the specimens used in obtaining data for the formula applied, the results of its application will be questionable. In general, text book formulas are derived from research on relatively simple heat transfer apparatus, while nuclear test specimens often are very irregular. For this reason, it is not unusual for the sponsoring agencies to do a considerable amount of heat transfer research specific to the geometries and environment of the specimens which they wish to irradiate. As a general rule, it is desirable to use a heat transfer correlation developed by a sponsor in considering his test specimens. As an example, the Bettis Atomic Power Laboratory has developed burnout correlations. In performing heat transfer analysis of Bettis test specimens, it is possible to apply these correlations with greater confidence than the more general correlations found in Jens and Lottes or McAdams [1].

4. Accuracy of Predicted Conditions

The "hot-spot, hot-channel" analysis discussed in Chapter III, Volume 1, is a good illustration of a method of evaluation of the possible deviations from predicted conditions. As part of any heat transfer analysis, there should be an evaluation and tabulation of the accuracy of all of the factors involved. In this evaluation, the possibility of deviations that tend to be systematic rather than random should be considered. An effort also should be made to establish probabilities.

If the test is designed to operate under very conservative conditions, a quick "hot-spot, hot-channel" analysis may show that the hazards associated with heat transfer are negligible as compared with other factors. When a quick uncertainty analysis of the heat transfer shows that undesirable temperatures are possible, probabilities should be assigned to the factors so that the uncertainty analysis can be refined to include probable temperature deviations.

5. Transient Analysis

After steady state conditions have been determined, the effect of transients upon the system should be considered. It is customary to make performance tests on loops to determine the maximum loss of flow rate and the difference between indicated and actual flow rates during a flow transient. In a recirculating water system, the loss of flow rate usually is a function of the loss of kinetic energy in flowing water and rotating parts of the pump and pump motor. In a single pass air system, the rate of loss of flow usually is a function of the loss of stored energy in high pressure air upstream of the test specimens.

B. HEAT TRANSFER CORRELATIONS

1. Conduction

Heat transfer by conduction through a volume of material, and the temperature drops as a result of the heat flow are related as shown by the following formula:

$$\Delta T = \frac{QZ}{kA} \quad (3-1)$$

where

Q = heat generation rate, Btu/hr

Z = thickness of material, ft

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

A = heat transfer area, ft².

McAdams [1], Jakob [2], and others have developed special formulas for various simple shapes such as cylinders, rods, and spheres. In special cases where formulas have not been developed for a specific shape, the basic equation given above can be adapted. Conduction of heat in complicated shapes can also be studied in analog computers.

Two special cases of heat conduction are especially important: (a) when heat is constantly generated in the conduction medium as in a fuel plate; and (b) transient heat transfer where rate of heat generation varies with time. Both problems can be solved by mathematical analysis or by analogs.

The formulas for ΔT through a plate and a rod with internal heat generation are given in formulas (3-7) and (3-8) in Volume 1.

In most cases, analysis of conductive heat transfer in experimental systems consists of: (a) determining whether internal heat generation in the various materials is significant (eg, in fuel cladding it is not, while in in-pile tubes, pressure tubes, or fueled or large samples, it is); (b) determining the proper thermal conductivity; and (c) determining the proper formula for the geometry of the sample.

The sample problems in Appendixes A and B illustrate the general approach; a great deal of effort is devoted to defining the heat transfer problem while the calculations are relatively simple.

2. Forced Convection

In order to formalize and systematize ways and means of performing heat transfer calculations, a document, IDO-16343 [3], was written in 1957 by the Heat Transfer Subcommittee of the Phillips Petroleum Company Reactor Safeguards Committee [4]. This committee made a comprehensive study of all correlations available at the time for calculating the temperature drops across the interface between a metal heat-generating body and a rapidly moving cooling medium. The result was the selection of one correlation for the calculation of film differential temperatures, and the selection of a method for presenting the uncertainties associated with the heat transfer calculations.

The modified Colburn equation was chosen by the committee for calculating forced convection heat transfer problems because of its relative simplicity, its tendency to be less conservative and at the same time fit experimental data at bulk water temperatures under 200°F, and its tendency to be quite conservative at bulk water temperatures over 400°F.

This equation has one definite disadvantage to the engineer; namely, all water properties must be determined in terms of the film temperature, which is the median temperature between the plate and bulk water. Therefore, the wall temperature must be estimated in order to begin the calculations and the calculations later refined by trial and error.

In order to simplify calculations, the modified Colburn equation has been rearranged so that the heat transfer coefficient, h , is a function of two groups of parameters b and K . b is a function of the film temperature and can be plotted on a graph (see Figure 11). K describes the water flow channel and is easily calculated.

$$\frac{hD}{k} = 0.023 \left(\frac{DV\rho}{\mu} \right)^{0.8} \left(\frac{c\mu}{k} \right)^{0.3} \quad (3-2)$$

$$h = bK$$

where

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

$$b = \frac{0.023(3600)^{0.8}(12)^{0.2}k^{0.7}\rho^{0.8}c^{0.3}}{\mu^{0.5}}$$

$$K = \frac{V^{0.8}}{D^{0.2}}, \text{ where } D \text{ is in inches, and } V \text{ is in ft/sec.}$$

A brief study of the equations and Figure 11, which plots the variation of b with film temperature, emphasizes the following points concerning the

Fig. 11 Base factor b vs film temperature (modified Colburn equation).

temperature drop across the film between a heated plate and flowing fluid.

- (1) The temperature drop decreases almost linearly as water velocity increases.
- (2) The temperature drop decreases slightly as the equivalent diameter decreases.
- (3) The temperature drop decreases slightly as bulk water temperature increases until the film temperature reaches $\approx 450^\circ\text{F}$.
- (4) The temperature drop increases slightly as bulk water increases from 450°F to saturation temperature.

In experimental systems where the sample center or clad temperatures are critical, the heat transfer under various conditions of heat flux, water temperature,

and water velocity must be studied. This is necessary for two reasons: (a) fission power generation, dependent on thermal neutron flux, can vary considerably from predicted values. To hold the same temperatures, bulk water temperatures and flows can be varied; (b) transient conditions caused by upsets or stoppage of bulk water coolant must be evaluated.

3. Natural Convection

Natural convection heat transfer is of interest to the nuclear engineer in assessing the hazards accompanying complete loss of flow to a system normally cooled by forced convection. Heat is transferred from the fuel and structural materials to the coolant under zero flow conditions by conduction and, since water density decreases as temperature rises, low velocity water currents are set up which remove heat from the fuel elements.

Forced convection cooling systems commonly used in the reactors and experiments handle heat fluxes on the order of 10^6 Btu/hr-ft^2 . Natural convection in an MTR or ETR core can handle $\approx 2 \times 10^4 \text{ Btu/hr-ft}^2$.

Several formulas have been developed to determine heat transfer coefficients for natural convection systems. One developed by Martinelli is explained in McAdams [1].

4. Nucleate Boiling

Nucleate boiling is explained in Chapter III, paragraph C 4a, Volume 1. Heat transfer during nucleate boiling is very efficient, and metal temperature is almost constant over a wide range of heat fluxes and water conditions. Some experiments operate in the nucleate boiling range in order to control fuel element clad temperature and center temperature accurately.

The difference between the saturation temperature for the liquid and the metal temperature, defined by Jens, and ΔT_{sat} ,

is not significantly dependent on fluid velocity or the temperature of the fluid but does depend on pressure [5]. For systems operating at atmospheric pressure, ΔT_{sat} is $\approx 40^\circ\text{F}$ and at 2000 psig, ΔT_{sat} is $\approx 6^\circ\text{F}$.

Since this is the case, the loop for experiments in the nucleate boiling range is operated at a pressure of which the saturation temperature is a few degrees lower than the desired cladding temperature. Loop water velocity and bulk water temperature are controlled so that nucleate boiling occurs but burnout heat flux is much higher than the heat flux expected in the experiment.

5. Burnout

The mechanism of burnout is explained in Chapter III, paragraph C 4b, Volume 1; and two formulas are given for calculating burnout heat flux for light water under conditions similar to those occurring in the MTR and ETR. Note that when the Mirshake equation [6] is given, the confidence level of the formula for the range of variables listed is also specified.

Westinghouse has developed a series of correlations on burnout specifically for heat transfer conditions found in high pressure water loops and reactors. They are described in the report WAPD-188 [7]. The correlations are available for both square channels and round tubes over a wide range of operating conditions. The design correlation for rectangular channels, which is commonly used for calculations on MTR and ETR high pressure water loops, is:

$$\frac{\phi_{\text{BO}}}{10^6} = 0.240 \left(\frac{H_{\text{BO}}}{10^3} \right)^{-2.5} \left(1 + \frac{G}{10^7} \right)^2 e^{-0.0012 L/S} \quad (3-3)$$

where

ϕ_{BO} = burnout heat flux, Btu/hr-ft²

H_{BO} = enthalpy at burnout, Btu/lb

G = mass velocity, lb/hr-ft²

L = distance from inlet to burnout point, ft

S = channel spacing, ft.

The range of variables over which these equations are applicable is as follows:

- (1) Pressure = 1850 to 2150 psia
- (2) Enthalpy = 500 to 1000 Btu/lb
- (3) Mass velocity = 0.2 to 5.0 x 10⁶ lb/hr-ft²
- (4) L/S = any values.

Ray Jacobs and Jim Merrill of Phillips Petroleum Company, AED, made a statistical study [8] of all available experimental data on burnout heat fluxes for round tubes and arrived at a correlation based on system inlet temperature, mass flow, tube length, tube diameter, and pressure which developed an empirical formula for calculating burnout heat flux for a wide range of operating conditions. A description of the study and a discussion of the formula and its limitations appears in the article [8].

The formula and the statistical fit appear in Figure 12. The range of variables applicable to the formula is:

- (1) D = 0.075 to 0.306 in.
- (2) G = 0.17 x 10⁶ to 7.79 x 10⁶ lb/ft²-hr
- (3) L = 6 to 27.4 in.
- (4) P = 500 to 2750 psia
- (5) T_i = 72 to 644 °F.

Although the form of the correlation appears difficult, solution on a computer

Fig. 12 Overall statistical "Ti" correlation.

or calculator is simple and straightforward.

Jacobs and Merrill completed a similar study of burnout heat flux in rectangular channels.

If the experiment maximum heat flux and burnout heat flux are calculated for various operating conditions, safety factors (ratio of burnout heat flux to maximum heat flux) can be determined for each set of conditions. These burnout safety factors, or DNBR's (departure from nuclear boiling ratios), give an indication of the relative safety of the operating conditions.

Three different safety factors are calculated for every experiment installed in a high pressure water loop: (a) normal loop flow and bulk water temperature; (b) conditions of loop temperature and flow where automatic reactor power reductions will be initiated; and (c) normal operating conditions, considering all uncertainties in operating parameters, sample configurations, and nuclear parameters.

With the Jacobs and Merrill correlation, there is an IBM program

which will calculate the associated uncertainty at the 99.5 percent confidence level for any predicted burnout heat flux. This calculated uncertainty is based on the experimental error associated with the experimental data from which the formula was derived. The uncertainties resulting from inaccurate or missing data in certain areas cannot be evaluated. This last statement is true no matter what correlation is used. Thus, the engineer must use judgment even in areas within the applicable range of variables of any formula where there was not any data taken for various parameter combinations.

6. Gas Heat Transfer

The heat transfer coefficients in gas systems are very low compared with water systems. Gas film coefficients as low as 50 Btu/hr-ft²-°F are not unusual. This compares with water film coefficients ranging from 2000 to 15,000 Btu/hr-ft²-°F. Also, there is a wide variation in heat transfer coefficients of gases. This principle is frequently used in controlling the effectiveness of thermal barriers. Gas system surface temperatures are frequently much higher than those encountered in water systems. Surface temperatures in gas cooled systems are frequently in excess of 1000°F, while water systems surface temperatures encountered are limited by the boiling point of water. The surface temperature of a water cooled specimen operating in the nucleate boiling range in a 2200 psig system will about 655°F, and the surface temperature of such a system may be controlled by varying the system pressure.

Because of the low film coefficients and the high surface temperatures associated with gas cooled specimens, radiant heat transfer must be considered as an important factor in dissipating specimen heat. Surface temperatures in gas cooled systems will exhibit a rather smooth variation with changes in coolant flow and heat flux. In water systems, transition from forced convection to nucleate boiling, then to burnout, can produce rather

dramatic effects on the rate of change of surface temperature. Appendix C presents a typical gas loop heat transfer problem where both forced convection and radiant heat transfer are considered.

7. Liquid Metals Heat Transfer

Liquid metals have several advantages over water or gases when used as a heat transfer medium. They have high boiling points, are very resistant to thermal decomposition, and have low viscosity which means low pressure loss in pumping [9].

Classical heat transfer theory and formulas can be used with fair accuracy for natural convection and possibly for boiling heat transfer in liquid metals; but forced convection theory must be modified to consider several factors [10, 11].

The very high thermal conductivity of the liquid metal complicates heat transfer in forced convection. Since heat may be transported throughout the liquid by conduction as well as the eddy currents which dominate heat transfer in water and air, forced convection heat transfer is more complicated. Other complicating factors are that heat transfer coefficients across the film between the heat source and moving fluid are so high that other factors ignored in water and air heat transfer become important. These include corrosion films, nonwetting of the heat source by the liquid, and the thermal resistance of the heat source. Formulas for calculating natural convection heat transfer rates under specific conditions are available from the literature.

Heat transfer for vertical plates and cylinders, vertical pipe, and thin vertical annuli have been studied and formulas are available. However, liquid metal experimental data for checking these formulas is not extensive, and heavy reliance is placed on extensions of theoretical work on water.

Several correlations have been developed for calculating the heat transfer coefficient for forced convection using liquid metals. These are similar to the formulas used for water, but include a correction factor which accounts for the effect of the high thermal conductivity of the liquid metal. The accuracy of these correlations appears to be as good as the modified Colburn equation for water, although the amount of verifying information is not as extensive. One form of the Martinelli equation^[11] for round tubes is:

$$h = \frac{k}{D} \left[7 + 0.025 \left(\frac{Du \rho c}{k} \right)^{0.8} \right] \quad (3-4)$$

where

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

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

D = diameter of tube, ft

u = mean velocity, ft/hr

ρ = density, lb/ft³

c = specific heat, Btu/lb-°F.

Heat transfer coefficients for liquid metals in natural convection and forced convection are considerably higher than water under similar conditions, so that nucleate boiling is not as attractive. For this reason, data on boiling heat transfer in liquid metals systems is not extensive and correlations are not available.

C. STRUCTURAL MATERIAL UNDER IRRADIATION

Structural material in a test reactor can generate as much as 40 watts of power per gram by absorbing gamma energy. For this reason, the engineer must

assure himself that all of this heat can be removed. Factors such as melting, thermal expansion, material phase changes, and loss of strength must be considered.

Pressure vessels, such as the in-pile tubes used in high pressure water systems, present a special problem. The intense gamma heating can induce thermal stresses which cause plastic deformation of the walls of the pressure vessel. The gamma heat induced thermal stress on one occasion was the limiting factor in the design of an in-pile tube for a high pressure water loop.

D. OUT-OF-PILE EQUIPMENT

Heat transfer considerations associated with out-of-pile equipment design and operation closely approach those used in ordinary industrial practice.

1. Dissipation of In-Pile Heat

Out-of-pile equipment must be designed so that all of the heat generated in the test specimens and auxiliary in-pile equipment can be removed at nominal coolant temperatures. It also must be designed so that operating temperatures can be maintained.

a. Controlled Heat Removal

The out-of-pile equipment for an engineering experiment is normally designed to remove the maximum amount of heat that can be generated by the associated in-pile facilities. In designing a test for this equipment, it is then necessary to make a heat balance calculation for the system to determine that the nuclear and gamma heating in the in-pile facilities does not exceed the design heat removal capacity of the system.

Water-water, water-air, and water-organic heat exchangers have been used in water loops. These are commercially available pieces of equipment, and the

capacities should be specified by the manufacturers.

Water injection systems have been used in conjunction with air loops to reduce the temperature of exit air. In such systems, a controlled amount of water is injected into the exit hot air stream. Vaporization of the water reduces the gas temperature to the desired level.

b. Heat Losses to the Surroundings

High temperature equipment will require some type of insulation to minimize heat losses to the surroundings. Frequently, in-pile equipment is insulated by a gas annulus between the high temperature piping and an outer jacket. Commercial insulation is used for out-of-pile equipment. Unavoidable heat losses through insulation can be estimated by conventional methods.

In areas where hot piping penetrates reactor biological shielding, it is necessary to provide some method of dissipating the small amount of heat conducted through the insulation. Cooling coils and air flow have been used for this purpose. A simple heat balance will determine if cooling is adequate.

It is desirable to control the ambient air temperature in cubicles that contain large pieces of hot out-of-pile equipment. Commercial space coolers or air flow have been used successfully. Again, a simple heat balance will determine the adequacy of the system.

2. Electrical Equipment

Proper cooling of electrical equipment must be considered. Circuit breakers, relays, and wiring will generate a small amount of heat. If provision is not made for removal of this heat, failures can be expected. There are codes in effect that cover normal situations; however, nonstandard electrical equipment must be examined for proper cooling.

Heaters for high pressure water systems are a common source of trouble. Clam shell heaters often are used on high pressure piping. Poor heat transfer between the heater surface and piping can result in heater failures. Improper cooling of the electrical leads to heaters is another common source of equipment failure.

3. Auxiliary Cooling

There are special heat transfer problems connected with certain pieces of auxiliary equipment. Cooling must be provided for the bearings of moving equipment. Motors of high pressure water loop canned rotor pumps must be cooled. In this case, failure of coolant flow for a few minutes can result in the loss of a \$10,000 pump.

Ion exchange resins and paraffin moderated, delayed neutron monitors used in high pressure water loops cannot tolerate high temperatures. Special cooling systems must also be provided for these systems.

E. TRANSIENT ANALYSIS

When analyzing the nuclear, hydraulic, and heat transfer characteristics of an experimental loop, problems other than expected steady state conditions must be considered. These include: (a) the variation of experiment heat flux and burnout heat flux with time on a complete flow stoppage accompanied by a reactor power reduction; (b) variation of experiment sample temperature with time under conditions of flow stoppage and reactor power reduction; (c) variation of experiment temperature on a flow failure caused by accidental valve closure; (d) variation of bulk water temperature under conditions in (c); and (e) variation of sample temperature or bulk water temperature with neutron flux.

Coastdown of loop flow versus time is carefully measured at operating con-

ditions for all possible failures, ie, one, two, or all pumps failing simultaneously. At the same time, instrument response speed is measured since this must be considered in setting reactor power reduction setpoints (see Figure 13).

The relationship of experiment sample temperature versus flow and bulk water temperature over a wide range of conditions is also calculated for transient analysis. For experiments using forced convection cooling, the modified Colburn equation [3] is used. For experiments operating in the nucleate boiling range, the burnout correlation in WAPD-188 [7] or the Jacobs-Merrill correlation [8] is used to correlate burnout heat flux as a function of water flow and temperature. When these variables are plotted on a graph of coolant flow versus sample temperature or burnout heat flux, curves similar to those in Appendix B are obtained. Study of these curves, the reactor power reduction rates, and instrument response characteristics allows the project engineer to set annunciators and reactor power reductions which allow reasonably good control of the loop and adequate safety for the sample. If this cannot be done, loop operating conditions will be lowered until it is possible.

On gas cooled and air cooled loops, similar heat transfer, coastdown, and instrument response analyses are made. For any given experiment, the project engineer must guarantee that sample meltdown will not occur on a power failure or compressor failure.

Liquid metal loops have been installed where instrument response was not fast enough to prevent burnout of the sample on flow failure, and auxiliary cooling suppliers would not prevent it. In these cases the loops were designed so that, if a flow failure occurs and the sample melts down, the defected assembly can be safely removed from the reactor.

Fig. 13 Water loop coastdown curve.

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CHAPTER IV

SHIELDING

A. INTRODUCTION

Basic concepts of source strength, activity, attenuation, and dose rates are covered in Chapter I, Volume 1. Chapter I, Volume 1, reviews basic nuclear physics with emphasis on the basic laws covering sources, source strength, and the mechanics of radiation attenuation.

This chapter will attempt to supplement the shielding information given in Chapter I, Volume 1. Fundamental concepts of source strength calculation will be reviewed, various source geometries will be studied, and basic shielding laws will be covered for neutrons and gammas.

Several simple shielding formulas will be presented with their nomenclature. Appendix A and Appendix B will have sample shielding problems integrated into a more complete engineering problem.

B. REVIEW OF FUNDAMENTALS

1. Cross Section

Microscopic and macroscopic cross sections of materials are defined in paragraph D, Chapter I, Volume 1. Both are used as basic material constants for use in shielding problems. It is extremely important that the engineer be very familiar with the units, definitions, and conversions used with nuclear cross-sections (shown in paragraph D, Chapter I, Volume 1).

2. Basic Shielding Formula

The basic shielding formula is used, with refinements, in all shielding calculations. This formula as given in Chapter I, Volume 1, is:

$$I = I_0 e^{-\sigma NX} \quad (4-1)$$

where

I = beam intensity after passing through target, $n/cm^2 \text{ sec}$

I_0 = initial beam intensity, $n/cm^2 \text{ sec}$

σ = microscopic cross section of target material, $cm^2/\text{nucleus}$

N = number of nuclei per unit volume, nuclei/cm^3

X = target thickness, cm .

The complicating factors are: (a) sources with several types and energies of radiation; (b) shielding material containing many elements of widely varying absorption cross sections; (c) the assumption, implicit in the equation, that the radiation is absorbed on the first collision (this ignores scattering, source buildup, and secondary radiation); and (d) source geometry and self-shielding.

3. Gamma Attenuation

Paragraph D, Chapter I, Volume 1, discusses the three types of interaction of gamma rays with matter. These are photoelectric effect, Compton effect, and pair production. Generally, the gamma absorption cross section of a given material includes the effect of all three interactions. The cross section also varies with gamma energy.

Tables or curves of cross section versus gamma energy are available for

most shielding materials. Typical curves are shown in Figure 14, taken from the Reactor Shielding Design Manual [1]. Note that the cross sections given here have the dimensions, square centimeters per gram (cm^2/g). This value is similar to the microscopic cross section and must be multiplied by density in grams per cubic centimeter (g/cm^3) in order to get the macroscopic cross section (Σ) with the dimension cm^{-1} .

Fig. 14 Mass absorption coefficients vs energy.

Study of these curves will indicate that the macroscopic cross section of materials for gamma radiation varies considerably with the gamma energy. For this reason all shielding problems where precision is needed should have source strength given in units of MeV/sec or photons per sec with energy specified. If several gamma energy levels are released, separate shielding calculations should be done in each increment of gamma energy.

4. Fast Neutron Attenuation

Fast neutron attenuation problems can be calculated by the formulas given in paragraph D, Chapter I, Volume 1. For a given neutron energy and shielding material, a microscopic cross section in barns can be determined from tables or curves. The recommended source is BNL-325[2]. However, the cross section given is a total cross section which includes scatter and activation. Neutrons which scatter going through the shield are not removed from the beam; they are merely degraded slightly in energy. Therefore, the effectiveness of the shield is only a fraction of that calculated.

Effective removal cross sections which are measured values for water and other hydrogenous materials can be used with good results in some shielding calculations. On shields where nonhydrogenous materials are used, no simple, accurate method has been developed. The engineer may be forced to mock up the problem and measure attenuation or go to elaborate machine calculations.

5. Thermal Neutron Attenuation

Thermal neutron attenuation can be calculated simply and accurately by the basic formula if the absorption cross section is used. BNL-325 has thermal neutron cross sections for all elements [2].

For shields containing several materials, it is necessary to calculate the macroscopic cross section for each element and use the volume-weighted sum of all macroscopic cross sections in the attenuation problem.

C. SOURCE STRENGTH CALCULATIONS

1. Introduction

Radiation sources usually can be grouped into four categories. These are: (a) fissioning U-235, (b) gross fission products of U-235, (c) selective fission products of U-235, and (d) activated

elements. Source strength calculations, though basically the same in all cases, can be simplified for each of the categories above.

2. Prompt Fission Gammas

The gamma energy spectrum for fissioning U-235 is well known. Table III shows the recommended source strength and energy spectrum of prompt gammas from fissioning U-235. This source can be used whenever it is necessary to consider shielding during reactor operation.

3. Prompt Fission Neutrons

The prompt neutron energy spectrum for fissioning U-235 is shown on Figure

TABLE III

GAMMAS PER FISSION
IN VARIOUS ENERGY RANGES

Energy (MeV)	Photons/Fission in the 1/2-MeV Range Centered on this Energy
1/2	3.1
1	1.9
1-1/2	0.84
2	0.55
2-1/2	0.29
3	0.15
3-1/2	0.062
4	0.065
4-1/2	0.024
5	0.019
5-1/2	0.017
6	0.007
6-1/2	0.004

Total 7.0 photons/fission
7.8 MeV/fission

15. Yield may be determined from the relationship:

$$N = 7.75 \times 10^{10} \times P \quad (4-2)$$

where

N = number of prompt neutrons/cm³

P = power density in watts/cm³.

Fig. 15 Approximate MTR neutron spectra.

4. Fission Product Gammas

Fission product activity from gammas only at various times after irradiation can be calculated from the formula:

$$P_G = 0.0315 \left[P t^{-0.2} - (t + t_0)^{-0.2} \right] \quad (4-3)$$

where

P_G = gamma power, w

P = operating power of the source, w

t = time after shutdown, sec

t_0 = operating time, sec.

This formula is valuable for calculating nuclear afterheat for decay times greater than 10 seconds.

For shielding purposes, it is better to use the curves shown in Figure 16,

Fig. 16 Gross fission product gamma decay energy, infinite irradiation.

which break the gamma radiation into energy increments [3]. This curve has plotted the gamma decay energy in MeV/sec/watt of operating power for various gamma energy increments against decay time, assuming infinite irradiation.

When irradiation time is long compared to decay time, the gamma energy may be read directly from the curve. If cooling time is long compared with irradiation time, the following procedure may be used: (a) find source strength per

watt of operating power for infinite irradiation time and the decay time postulated; (b) find source strength per watt for infinite irradiation time and the decay time corresponding to the sum of the postulated irradiation time and postulated decay time; (c) the difference between these two values corresponds to the source strength per watt under the conditions postulated; (d) multiply the source strength per watt by operating power in watts in order to determine total source strength.

5. Specific Isotopes

On some types of experiment failures, activity release is not from gross fission products. Only a few fission products may be released, with most fission products being retained in the fuel. For instance, uranium oxides used in fuel elements release gaseous fission products, retaining most of the metallic products. In these situations, the engineer must postulate which fission product elements will be released, calculate their abundance from fission product yields and fissioning rate, and use each fission product as an individual source. Yield and decay constants of all fission products may be determined from data presented in ORNL-2127 [4].

D. GAMMA ATTENUATION

1. Basic Formula

Rockwell [1] has derived formulas for various source geometries for the attenuation of gamma rays from the point source equation:

$$D = \frac{S_0 B e^{-\mu_0 X}}{K^4 \pi r^2} \quad (4-4)$$

where

D = dose rate, R/hr

S_0 = point source of gamma rays, photons/sec

B = buildup factor

μ_0 = linear absorption coefficient, cm^{-1}

X = thickness of shield, cm

r = distance from source to receptor, cm

K = conversion factor, gamma flux to dose rate.

A discussion on the use of the linear absorption coefficient and the source term has been covered in Chapter I, Volume 1, of the manual. However, since the buildup factor is peculiar to shielding calculations, a brief description follows.

2. Buildup Factor

Buildup factor is peculiar to shielding calculations and is defined as the ratio of the actual gamma flux to that which would be calculated using a simple exponential attenuation with the linear absorption coefficient. If the total macroscopic cross section is used in the calculations, the predicted radiation intensity will be too low because of the contribution of the scattered photons.

The simple exponential absorption equation is strictly applicable only to narrow collimated (parallel) beams of radiation since it is based on the postulate that scattered particles are completely removed from the radiation beam. In other words, the scattered radiation can be treated effectively as if it were absorbed. For a relatively thin layer of absorber, the exponential equation does represent a good approximation even if the beam is not well collimated. The increased path length and change in direction of motion due to scattering greatly reduce the probability that the radiation will reach the detector.

If the shield is relatively thick, some particles which have suffered two or more scattering collisions within the absorber may reach the detector. Since the scattered particles are not removed, the intensity

at the detector as calculated from the exponential equation will be too low.

If the energy absorption cross section is used, the radiation intensity predicted by the exponential absorption equation will be much too high because photons scattered through a large angle can be considered to be removed from the beam of radiation. In actual practice, the total cross section is used in shielding calculations. The effect of scattered radiation is taken into consideration by the use of a buildup factor in the exponential equation (4-1).

3. Source Geometry Considerations

Most shielding calculations deal with distributions of sources and attenuators. Shielding theories usually express attenuation from either a point source or an infinite plane isotropic source in an infinite medium. Most shielding measurements have been made with a uniform disc source in a "semi-infinite" medium. Reactors can be approximated by cubes, cylinders, or spheres, while some aspects of the experimental facilities can best be approximated by point, line, or plane sources. In order to convert from one shape to another, certain geometrical manipulations are used that are found in the literature. The basic attenuation equations, their derivation, and a discussion of their use can be found in Rockwell, pages 347 to 408^[1]. Diagrams for point, line, and cylindrical sources are shown in Figure 17. Basic formulas for the source geometries given in Figure 17 are:

For a point source

$$\phi = \frac{BS_o}{4\pi a^2} e^{-b_1} \quad (4-5)$$

For a line source

$$\phi = \frac{BS_L}{2\pi a} F(\theta, b_1) \quad (4-6)$$

For a cylindrical source

$$\phi = \frac{BS_v R_o^2}{2(a+z)} F(\theta, b_2) \quad (4-7)$$

Fig. 17 Effects of geometry of radiation source.

where

ϕ = radiation flux, MeV/cm² sec

B = buildup factor, dimensionless

S_0 = point source strength, MeV/sec

S_L = line source strength, MeV/sec-cm

S_V = volume source strength, MeV/sec-cm³

a = distance from source to receptor, cm

$$b_1 = \sum_{i=1}^n \mu_i \tau_i$$

$\mu_1, \mu_2, \dots, \mu_n$ = linear absorption coefficient of shield 1, 2, ..., n, cm⁻¹

τ_i = thickness of ith shield, cm

$$F(\theta, b) = \int_0^{\theta_1} e^{-b \sec \theta} d\theta$$

θ_1 = angle

R_0 = radius of cylinder, cm

Z = effective self attenuation distance, cm

$$b_2 = b_1 + \mu_s Z$$

μ_s = linear absorption coefficient of source material, cm⁻¹.

In order to optimize the shield for an experiment, the geometry of the source should be duplicated as closely as possible by one of the standard geometry shapes. Otherwise it is possible to end up with a shield that is overdesigned or underdesigned.

E. NEUTRON ATTENUATION

1. Introduction

In the reactor, fast neutrons are the most serious radiation hazard. The fast neutron is ten times more damaging to the body than a gamma ray of the same energy.

Fast neutrons are difficult to attenuate with shielding because neutron absorption cross sections for various shielding materials are very small for high energy neutrons. The usual approach is to slow down the neutron to below 1 MeV with heavy metals, to reduce the energy to the thermal range with hydrogenous materials, then capture the neutron with a material that will emit a very low energy gamma or charged particle.

The neutron can be slowed down by elastic scattering, inelastic scattering, or shadow scattering. Elastic scattering is a billiard-ball-type collision with a scattering nucleus where this nucleus is left in a ground state; the only energy absorbed

by it is in the form of kinetic energy. This reaction is most important for light nuclei which can absorb a large fraction of the neutron's energy in recoil. If the neutron collides with a hydrogen atom, energy will be reduced so low that it can be assumed to be removed from the beam.

Inelastic scattering is most important at high neutron energies. The high energy neutron is absorbed by the target nucleus, forming a compound nucleus. This new nucleus then has an energy above its ground state resulting from that kinetic energy of the incident neutron not absorbed as kinetic energy by the recoiling nucleus. This compound nucleus, if at a high enough energy level, can expel a low energy neutron followed by one or more gamma photons. As a general rule, inelastic scattering is more important for the heavy nuclei. Their allowable energy levels are more numerous, they contain more particles, and their energy increment from the ground state to the first energy level is smaller. By inelastic scattering, even high energy neutrons can be degraded in energy by a single collision to energies of 1 MeV or less. Elastic scattering with hydrogen atoms can then reduce their energy to the thermal range. Most materials exhibit a relatively high absorption cross section for thermal neutrons so when the neutrons are reduced to thermal energies they can be easily removed.

Shadow scattering becomes noticeable at neutron energies of about 5 MeV for heavy nuclei and 10 MeV for light nuclei. In this process, the neutron acts as a wave and is diffracted around the target nucleus. Usually the angle of scattering is small, resulting in small degradations. It is not an important phenomenon in shielding problems.

2. Effective Removal Cross Section

In order to make the neutron attenuation calculations identical with gamma shielding calculations, it must be assumed

that any of the nuclear interactions discussed above will remove the neutron from the beam. This assumption means that the neutron cross sections used must be experimentally determined, using pile neutrons as a source. By definition these effective removal cross sections are the values which give the correct answer, assuming exponential absorption. Experiments carried on at ORNL have shown that it is reasonable to lump the elastic scattering, inelastic scattering, and shadow scattering and to consider them as a single absorption mechanism for all shielding materials except hydrogen.

A good approximation for the effective removal cross section of hydrogen for neutrons having an energy in the range from 2 to 12 MeV is given by the formula:

$$\sigma_H = \frac{5.13}{E^{0.725}}$$

where

σ_H = effective removal cross section for hydrogen, barns

E = energy, MeV.

Effective removal cross sections for other materials commonly used for fast neutron attenuation are given in AECD-3978 [5] and Rockwell's Reactor Shielding Manual [1].

Effective removal cross section is valid only if large amounts of hydrogenous material are incorporated into the shield. Heavy metals will allow passage of neutrons below 1 MeV energy level because they are efficient only as inelastic scatterers. If thick slabs of heavy metal shielding are used for high energy neutron shielding, they must have hydrogenous materials behind or be mixed with the slabs in order to attenuate the neutrons whose energy is below 1 MeV.

3. Mixed Shield Calculations

In shield design problems associated

with some experimental programs, it is often impossible to use a hydrogenous shield because of temperature or other limitations. This often necessitates the consideration and use of nonhydrogenous materials for the neutron shield. There are several ways of calculating the non-hydrogenous shield problem; however, they are seldom both economical and accurate. Two methods will be mentioned here. The first method, which in many cases is the simplest and most economical approach, is to mock up the system and scale up the data obtained. In other instances, the proposed shield may closely approximate experimental mock-ups of facilities used for data determinations that have been published in the various trade journals. In the second method, mathematical treatises similar to the one Speilberg [6] has used for non-hydrogenous materials may fit the immediate problem best. This procedure applies the Fermiage-theory and Bethe-Tonks-Hurintz theory to shielding material containing a mixture of nuclides. Speilberg also includes a method of determining the dose rate of the gamma rays from neutron inelastic scattering and from thermal neutron capture.

4. Thermal Neutron Capture

After the fast neutrons have been slowed down, the shield designer must make provisions for thermal neutron capture. This is usually a relatively easy thing to do even though the shield does not contain elements with large thermal neutron capture cross sections. It is generally accepted that in a shield almost every thermal neutron that has suffered inelastic scattering may be regarded as removed since the probability of subsequent capture is very large.

Most materials have significant thermal neutron capture cross sections, but caution should be exercised in selecting the position of the absorber in the shield as well as the type of absorber. Iron, for example, has a cross section of 2.5 barns for thermal neutrons. However,

iron-capture gamma rays have an energy of over 7.5 MeV, essentially all of which is emitted as a single photon. Additional gamma-ray shielding might be necessary to attenuate these capture gammas.

Judicial choice of materials can partly overcome this difficulty. Boron-10 has a large cross section for the (n, α) reaction with thermal neutrons. Some of the liberated energy appears as 0.5 MeV gamma radiation, which is readily absorbed.

F. GENERAL SHIELDING CONSIDERATIONS

1. MTR In-Tank Facilities

Some reflector positions are available for engineered experiments in the MTR. Each of these experiments will require communication from outside the reactor tank to the experiment itself. The communicating lines from the reactor core to the reactor top cannot travel in a straight line because of the reactor design. The offsets can be designed to attenuate the fast neutrons and the gamma rays originating in the reactor core.

The experiment's cooling lines outside the reactor tank should be shielded to attenuate the induced activity in the coolant. The shield must also prevent undue exposures from failure of an experimental fuel element. This usually dictates the size of the shield. The radioactivity from failure of a fuel element must be considered, and a realistic maximum credible accident defined before a shield can be designed that will prevent undue exposure.

In almost all cases, the shields that are to cover the coolant lines can be prefabricated to fit existing equipment, keeping the physical safety hazards to a minimum.

2. MTR Beam Hole Facilities

a. Beam Holes

Several sizes and types of beam holes in the MTR are available for experimenters' use. The primary differences are size, angle of penetration, and depth of penetration into the reactor structure. The horizontal beam holes penetrate to the edge of the reactor core on the reactor horizontal centerline. Horizontal reflector holes penetrate to the edge of the reactor tank.

Beam hole shielding calculations should be performed in two ways. First, perform the calculation assuming a plane source at the inner end of the plug. The source strength may be obtained from the reactor flux maps and expected power level of the experiment. Second, repeat the calculation using plane sources at several other places in the reactor structure and calculate the attenuation due to the plug material between the source and the outside of the reactor.

The above procedure will ensure that the neighboring reactor structure will not cause a source "buildup" in the outer portions of a plug whose inner portion forms a "better" shield than the reactor structure. This effect should give little cause for concern outside the graphite region of the reactor.

b. Through Hole

A through hole is a square hole which extends from one reactor face to the opposite face. It passes through the reactor tank and is adjacent to one side of the active lattice. During the past few years, it has been used for gas cooled experiments. This has, in effect, left a void from one end of the reactor to the other, causing a serious shielding problem. The shielding used for this facility will serve to illustrate shielding methods where large voids are encountered. The following items must be considered for the shield design:

(1) The shield should be equivalent to the reactor shield since the shield extends from the reactor centerline outward (both neutrons and gamma should

be considered). A point isotropic source, when the source gives the reactor a positive reactivity, will usually give conservative results. Scattering effects should be considered, and the shield design should be compared to the reactor structure at several different distances from reactor centerline to ensure the absence of source buildups.

(2) Neutron and gamma streaming problems can be handled as outlined in TID-7004^[1], Chapter VIII.

(3) In some cases, it may be impossible to design an equivalent shield as you progress from reactor centerline outward. If so, the shield should start as close to the sample as possible; it should attenuate both neutrons and gamma better than the reactor shield to give attenuation lengths equivalent to the reactor structure at the graphite zone. Neutrons absorbing materials can be used as a coating on the facility walls to reduce scattering in the outer portions of the shield, which may cause source buildups.

(4) In order to minimize handling problems, the shield should be designed in sections. This will allow handling of the outer portions of the internal shield with little or no auxiliary shielding. The inner sections of the main shield will usually be radioactively hot and will require special handling procedures and casks to minimize exposures from the radioactivity of the experimental components.

(5) In general, the shield should be positioned in the facility to minimize induced activity in experimental component lines and parts that are not adjacent to the sample or reactor high flux zone.

3. ETR Tank Facilities

The ETR reactor shielding serves also as experimental shielding until the experimental piping leaves the subpile room wall. This shielding includes vertical sleeves from reactor top to bottom, plus the subpile room concrete structure

which shields the bottom head and experimental leaks until they penetrate the wall. All experimental loop equipment is located in an area adjacent to the subpile room wall. This includes piping, heat exchangers, pumps, motors, valves, etc. A concrete wall is then built around the loop equipment, and access to the equipment is limited to periods when the reactor is not operating. In addition to the cubicle wall, certain pieces of equipment such as resin beds and particulate filters are given additional shields to minimize overall shielding requirements. The primary material used for ETR experiment shielding is high density concrete. Some experiments use lead or polyethylene to supplement the concrete walls.

4. Loop Discharge Systems

a. MTR Beam Holes

The MTR beam hole experimental facilities were designed so that they could be serviced by one discharge cask. A universal cask of lead and steel has been used for experimental insertion and removals. The cask is mounted on wheels and is moved by battery powered trucks. It weighs ≈ 15 tons. Although the MTR in-pile assemblies have been designed for removal with the universal cask, some experimentors have built casks for handling and shipping the sample fuel elements.

b. MTR In-Tank Facilities

The in-tank experimental facilities are usually designed to use the discharge chute for removals. The discharge chute is designed to handle the 14-foot MTR shim rods, providing the necessary clearance needed to discharge the in-tank experiments and tubes.

c. ETR In-Tank Facilities

One basic difference between the ETR and MTR is the arrangement above and in the core. The MTR uses a removable top grid to hold the fuel elements. No experimental facilities are provided in the reactor core. ETR experimental

facilities pass through the core, and the area above the core facility is reserved for experimental use. With the great diversity of experimental requirements in the ETR, it has not been possible to design one removal cask that can accommodate all experiments. One experiment requires a cask weighing approximately 14 tons to remove a sample fuel element; this cask is in no way compatible with its neighbor's requirements.

All experiments in the ETR are designed so that samples are lifted upward out of the top of the in-pile tube. Some are then lowered into the discharge chute while others are pulled up into removal casks.

Since each major experimental facility must extend from the top of the reactor past the core and, in some cases, into the subpile room, the facility length (≈ 30 feet) becomes an extremely important consideration when designing removal casks for in-pile tube removals.

5. Criteria for Shielding of Equipment

a. General Criteria

Requirements on shields dictated by experimental programs can be divided into three groups: (a) sound element operation; (b) defected element operation; and (c) accidents.

An experimental system cannot be designed without considering the consequences of a major failure, so items (b) and (c) are studied very closely. In order to determine the shielding system required for defective elements, a combination approach is usually used. First, certain out-of-pile tests are performed on the elements to determine what areas of the experiment are weakest and to what extent they will fail or deform. This information is then used in making activity calculations for shield design. Many times the information available from out-of-pile testing and previous reactor testing of smaller nuclear mock-ups would indicate that a failure would be progressive and

require the consideration of a maximum credible accident. A maximum accident in any system would be complete failure or meltdown of the sample and distribution of it throughout the system. This, however, is not always credible failure. The shield is designed to protect personnel from the maximum credible accident.

In one of the examples noted previously, the use of secondary or double shields was mentioned. Certain parts of an experiment's out-of-pile equipment become more radioactive than the main part of the experiment. Depending on the type of experiment, these parts might be ion exchangers used for demineralized water cleanup, filters used for particulate removal, delay systems used for fission product holdup, etc.

These units are usually very heavily shielded inside the cubicle wall shield. This cuts down cubicle wall thickness, since these units serve as collecting points for radioactive material and keep the general field down. In most cases, they minimize the magnitude of cleanup or containment of an accident.

b. Radiation Level

On large engineered experiments, equipment must be designed and installed so that operating personnel are not exposed to unnecessary radiation levels and are not exposed to radiation levels above 2.5 mR/hr under any but emergency conditions. Shielding must be installed to protect the operator if, under normal or emergency conditions, the radiation level is expected to be too high.

The maximum allowable radiation level in the nearest occupied area during normal operation must be less than 2.5 mR/hr. Most cubicle walls are designed for 2.5 mR/hr maximum level. This radiation comes from activated coolants such as N-16 and N-17 in cooling water and A-41 in air. The experiment coolant also contains activated

impurities that plate out in various parts of the experimental equipment.

During reactor shutdowns, access to loop equipment is necessary. In areas where extensive maintenance is done, radiation levels must be kept down to a level where the maintenance can be done with a minimum of exposure to the mechanic.

If equipment in the cubicle gets too radioactive, the equipment must be shielded, decontaminated, or removed.

Some experimental loops are designed to operate with fission product evolution. If so, the cubicle walls must be adequate to reduce radiation levels outside the cubicle walls to below 2.5 mR/hr while the experiment is operating.

Most experiments operated in the MTR and ETR are subject to failure. In the Project Engineer's review of the acceptability of an experiment, he arrives at a condition compatible with the design of the experiment and its supporting facilities that would represent a maximum credible accident. The shield is then designed to protect personnel and prevent them from receiving more than the maximum allowable exposure.

During discharge of experimental fuel elements, two approaches have been used for design of removal equipment. Most discharge mechanisms are designed to reduce the radiation level during removal of the sample to below 2.5 mR/hr. This requires heavy shielding and complicated equipment, but allows close control of the discharge procedure.

Quite often, it is necessary to pull a sample from the reactor core up into a cask which provides adequate shielding. Radiation levels near the cask may be quite high as the experiment sample passes up from the water and approaches the bottom of the cask. If the radiation level is high for a short time only, if personnel

are not required to occupy the area, and if there is a negligible chance that the experiment will hang up in this unshielded position, this procedure is acceptable.

G. MATERIALS USED FOR SHIELDING

Following is a list of the shielding materials that are in general use at the MTR-ETR for experimental systems along with a brief comment on their adaptability to various needs.

1. Magnetite

The combination of light and heavy elements makes magnetite concrete a good gamma and neutron shield. In the OWR (Omega West Reactor), the fast neutron flux ($E \sim 2$ MeV) in the shield is attenuated with a decade length of 8.1 ± 0.4 inch. The gamma dose rate from the core of the reactor is attenuated in the shield with a decade length of 7.0 inches. The material is relatively abundant, cheap, and easily installed. All the shielded cubicle walls for ETR experiments were made of magnetite concrete blocks.

2. Barytes

Barytes concrete is a good combination shield; it is not quite as efficient a gamma shield as magnetite, but is a more efficient neutron shield. For example, gamma decade length for barytes has been reported (Rockwell [1]) as being 60 percent greater than for magnetite concretes (2 to 3 MeV range).

3. Polyethylene

Polyethylene is used primarily for epithermal neutron attenuation in relatively cool (thermal) areas where space and weight are critical.

4. Borated Polyethylene

Borated polyethylene is used to capture thermal and epithermal neutrons.

5. Boral

Boral is used to capture thermal neutrons in high temperature areas where capture gamma fields can be minimized.

6. Cadmium

Cadmium is used occasionally for capture of thermal neutrons in high temperature areas where resulting capture gamma energy is not critical.

7. Paraffin

Plain and borated paraffin have been used for neutron attenuation. A large percentage of the lines and equipment that must be shielded are thermally hot, and combustible materials must not be in close contact with them. Paraffin must not be used in these locations because of the fire hazard associated with low flash and ignition temperature hydrocarbons. As mentioned previously, concrete has been used as a substitute for paraffin whenever possible.

8. Lead

For the attenuation of high gamma fields, lead is used extensively at both MTR and ETR. High density, good working qualities and availability, coupled with low volume requirements, make it extremely useful.

Lead is particularly effective against low energy gammas because of high photo-electric absorption cross section, and against high energy gammas because of its high pair-production cross section. Its minimum cross section occurs at about 3 MeV. Because of its high density the volume required for a shield is low, resulting in a comparatively small, lightweight shield. This makes it attractive for shielding large removal casks where space and weight are important.

9. Steel

Steel usually finds its way into most

large shields as structural supports, braces, and in places that are heavily stressed. Some shields next to gas cooled experiments are so highly stressed due to thermal gradients that steels are required for the structural properties.

10. Water

Water is used in areas where delayed neutron production from fission product evolution is a problem. Space requirements are relatively low; however, handling is a serious problem. Solid shields are easier to build and maintain and generally do not require replacement, while water usually requires provisions for constant purging.

11. Uranium

Where weight and volume are extremely critical, depleted uranium is used. It is used in large transfer casks.

12. Metal Hydrides

Where neutron shields are required in high temperature areas, metal hydrides have been used.

13. Future Materials

a. Calcium Borate

This neutron shielding material is a combination of borated calcium minerals in a matrix of asbestos. The density is 70 lb/ft³ with a very low shrinkage at high temperatures (1/2 percent at 1000°F). Boron content is 12 weight percent.

b. Serpentine

Serpentine, a hydrous magnesium silicate, is used as a neutron shield. It can be obtained as a tamped powder or in concrete with a density of 130 lb/ft³. The calculated neutron and gamma attenuation properties are similar to barytes concrete.

H. TYPICAL MTR SHIELD

For maximum efficiency, a shield should attenuate fast neutrons and gamma rays at such a rate that the respective fluxes will be reduced to their maximum permissible levels at the same place, just at the exterior of the shield. This, of course, requires a combination of materials some having high mass number and others containing hydrogen.

The MTR has HT-1, a "through hole" that passes next to the reactor core and is open at both ends. It lends itself very well to gas cooled type experiments. The facility then becomes, in effect, an open tube through the reactor. This makes it necessary to shield the facility for gammas and neutrons from both the reactor and experiment with the shielding starting at the face of the reactor. Due to the temperature encountered and for handling ease, the following laminated shield has been designed and successfully used.

From the reactor face outward, the shield consists of a layer of high density concrete, a layer of lead brick, a layer of borated polyethylene, a layer of high density concrete, and a layer of borated paraffin.

In this experiment, the gamma and neutron fields combined determine the shield thickness and type. Since high density concrete has a balance of light and heavy elements, it seems very well suited as a combination shield material. Following the first section of magnetite, lead is used to further decrease the gamma field. Then borated polyethylene is used to decrease the thermal neutron flux and, to some extent, decrease secondary gamma radiation arising from neutron capture in the polyethylene. High density concrete is again used as a combination neutron and gamma shield followed by a layer of borated paraffin to thermalize and capture the remaining neutrons. The shield must be assembled without cracks.

Small openings can increase radiation fields to a level that prohibits operation of the reactor or experiment.

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CHAPTER V

IN-PILE TUBE DESIGN

A. INTRODUCTION

As a matter of convenience, reactor test loops are considered to consist of an in-pile section and an out-of-pile section. The design of the out-of-pile portion is presented in the following chapter. An attempt will be made in this chapter to establish only a few of the fundamentals in the extremely complex design of an in-reactor pressure tube and its accessory equipment.

Loops provide a useful means of duplicating a variety of environments for irradiating specimens of reactor construction materials in research-type test reactors such as the TRA (Test Reactor Area) reactors (MTR/ETR/ATR). The test work may include: (a) the development and prototype testing of fuel element designs to determine their radiation stability, integrity, and corrosion resistance; (b) the development and prototype testing of other structural and control materials of interest in the parent reactor; (c) studies of cooling medium chemistry and activity buildup; and (d) other studies of a basic or applied nature which require a particular combination of radiation level, pressure, temperature and coolant flow. The in-pile tube holds specimens in the desired flux zone of the test reactor and provides a channel for coolant flow. Factors in the design of an in-pile tube (sometimes termed a "test thimble") illustrate that it is actually a reactor vessel in miniature.

The basic types of in-pile tube designs presently installed in the TRA reactors are shown in Figure 18. The advantages of top reentrant systems over bottom reentrant systems have been outlined in a private communication [93]. The advantages of top reentrant over the through type include: (a) ease of changing experiments

and thimbles; (b) short lengths of connecting piping; (c) reactor pressure vessel penetrations are minimized; (d) small shielding requirements; (e) high fluid velocities; (f) problems of thermal expansion are minimized; (g) possibility of draining the in-pile tube with experiment in place is eliminated; (h) subpile room space conflicts are minimized; and (i) high radiation levels during maintenance are reduced. Disadvantages include: (a) the slightly greater design complexity; (b) relatively high pressure losses for a given flow; and (c) a larger test volume is required for a given size test specimen.

As the water loop is the most common type loop at TRA, only the more important problems connected with this group of in-pile tubes will be presented here.

B. DESIGN PARAMETERS

Prior to commencing any in-pile tube design calculations, it is necessary to establish design operating parameters of flow, pressure, and temperature; the position of the experiment in the reactor; and the gamma heat, thermal, and fast neutron flux values.

As may be observed in Table IV, the most common loop design operating pressures and temperatures at TRA are 2200 psig and 600°F, being established by the sponsors to simulate the design parameters of today's water-moderated power reactors.

To cover the range of present core fluid velocities, in-pile tubes must be designed so that 10 to 40 ft/sec can be attained past sample specimens. High fluid velocities are important for fuel element testing because the burnout heat flux is

WATER LOOPS

TYPE 1

TOP REENTRANT, VERTICAL, RECIRCULATING, SINGLE CHANNEL, FLOW REVERSAL AT EXPERIMENT OUTLET.

EXAMPLES: WAPD B-4; ALSO GEH B-3, EXCEPT BEING SINGLE PASS INSTEAD OF RECIRCULATING.

TYPE 2

THROUGH, VERTICAL, RECIRCULATING, SINGLE CHANNEL,

EXAMPLES: WAPD VH-3, WAPD C-7, GEH P-7, GEH G-7/M-3.

TYPE 3

TOP AND BOTTOM REENTRANT, VERTICAL, RECIRCULATING, SINGLE CHANNEL, FLOW REVERSAL AT EXPERIMENT INLET.

EXAMPLES: TOP REENTRANT, KAPL F/H-10, WAPD L-10, WAPD L-12/M-13, WAPD J-10/L-8; BOTTOM REENTRANT, KAPL L-42, ATR LOOPS.

TYPE 4

REENTRANT, HORIZONTAL, RECIRCULATING, SINGLE CHANNEL, FLOW REVERSAL AT EXPERIMENT OUTLET.

EXAMPLE: PAED HB-2.

TYPE 5

REENTRANT, HORIZONTAL, RECIRCULATING, MULTIPRESSURE TUBE, DOUBLE EXPERIMENT, FLOW REVERSAL AT EXPERIMENT OUTLET.

EXAMPLE: KAPL HB-1.

Fig. 18 Classification of in-pile pressure tubes at TRA.

TABLE IV

TRA IN-PILE TUBE DATA

Designation	PAED HB-2 (Formerly ANL-2)	WAPD-13	WAPD-25	WAPD VH-3 (See note 3)
References	ANL-5403		WAPD-PWR-CP-1783 WAPD-PWR-CP-1832	WAPD-PWR-CP-2248
Reactor and Position	MTR-HB-2	MTR-HB-5	MTR-L-42	MTR-VH-3
Present Status	Operating (Out-of-pile only)	Program Completed	Program Completed	Operating
Designed Operating Temperature	550°F	600°F	600°F	600°F
Designed Operating Pressure	1500 psig	2200 psig	2200 psig	2200 psig
Configuration	See Figure 19	1st: See Figure 20 2nd: See note 2	Cylindrical	Cylindrical
Dimensions	See Figure 19	1st: See Figure 20 2nd: See note 2	1 in. Schedule 40	1/2 in Schedule 40
Material	316 s/s	304 s/s	304 s/s	304 s/s
Assumed Heat Removal	Inside Surface	Inside Surface	Inside Surface	Inside Surface
Approximate Peak Thermal Neutron Flux	1.8×10^{14} n/cm ² /sec			1.35×10^{14} n/cm ² /sec
Maximum Gamma Heat	10 w/g at Tip	1 w/g at Tip	10 w/g	3 w/g
Maximum Calculated Stresses	See Figure 19		Pressure: 9,900 psi Thermal: 32,800 psi	Total 10,500 psi
Testing Done	Discussed in ANL-5403	Pressure Test	Pressure Test	Pressure Test
Number of Tubes Exposed	3	2	2	7
Maximum Fast Neutron Flux (>1 MeV neutrons)	6.0×10^{13} n/cm ² /sec at Tip	2×10^{12} n/cm ² /sec at Tip	5×10^{13} n/cm ² /sec	2.7×10^{13}
Length of Exposure (>1 MeV neutrons)	(See note 1)	1st: 9×10^{18} n/cm ² WAPD-20 Tube: 1.7 x 10 ¹⁸ n/cm ² 2nd: 2.8×10^{19} n/cm ²	1st: 6.5×10^{19} n/cm ² 2nd: 1.2×10^{20} n/cm ²	(See note 4)

TABLE IV (Cont.)
TRA IN-PILE TUBE DATA

Designation	WAPD-18	KAPL-37	KAPL HB-1 (Formerly KAPL-30)	WAPD J-10 (WAPD-34)	WAPD L-8
References	WAPD-P-544	KAPL-37 In-Pile Thimble	MDO-64-216 Transmittal of In-Pile Tube Data Package	1st: Vitro Report KLX 1807 2nd and 3rd: (See note 7)	(See note 7)
Reactor and Position	MTR-L-42	MTR-L-42	MTR HB-1	ETR-C-33-J-10	ETR-C-33-L-8
Present Status	Program Completed	Program Completed	Operating	Operating	Operating
Designed Operating Temperature	600°F	600°F	600°F	600°F	600°F
Designed Operating Pressure	2200 psig	2200 psig	2200 psig	2200 psig	2200 psig
Configuration	Cylindrical	U-Tube Cylindrical through Core	(See note 6)	Cylindrical	Cylindrical
Dimensions	1.949 in. OD x 1.278 in. ID	7/8 in. OD x 0.083 in. wall 3/4 in. OD x 0.062 in. wall	Each Tube Approximately 1.50 in. OD x 1.25 in. ID	3.000 in. OD x 2.124 in. ID	3.200 in. OD x 2.124 in. ID
Material	316 s/s	304 s/s	347 s/s	347 s/s	347 s/s
Assumed Heat Removal	Outside Surface	Inside Surface	Inside Surface	Inside Surface	Inside Surface
Approximate Peak Thermal Neutron Flux		1.5×10^{14} n/cm ² /sec	2×10^{14} n/cm ² /sec	4×10^{14} n/cm ² /sec	3.0×10^{14} n/cm ² /sec
Maximum Gamma Heat	10 w/g	10 w/g	8.0 w/g	20 w/g	13 w/g
Maximum Calculated Stresses	Pressure: 13,600 psi Thermal: 7,050 psi	(See note 5)	Pressure: 14,170 psi Thermal: 6,790 psi	Pressure: 14,500 psi Thermal: 91,500 psi	Pressure: 14,500 Thermal: 91,500
Testing Done	Pressure Test	Pressure Test	Pressure Test	Pressure Test	Pressure Test
Number of Tubes Exposed	2	3	1	3	1
Maximum Fast Neutron Flux (>1 MeV neu- trons)	5×10^{13} n/cm ² /sec	5×10^{13} n/cm ² /sec	5.5×10^{13} n/cm ² /sec at Tip	3.77×10^{14} n/cm ² /sec	3.0×10^{14} n/cm ² /sec
Length of Exposure (>1 MeV neutrons)	1st: 5.5×10^{19} n/cm ² 2nd: 1.5×10^{20} n/cm ²	1st: 5.5×10^{20} n/cm ² 2nd: 2.7×10^{20} n/cm ² 3rd: 4.8×10^{20} n/cm ²	1st: In Operation (Tentative- ly approved until 6/20/65)	1st: 7.2×10^{21} n/cm ² 2nd: 1.12×10^{22} n/cm ² 3rd: In Operation (Tentative- ly approved until approximately 7/1/66)	1st: In Operation (Tentative- ly approved until 9/13/65)

TABLE IV (Cont.)

TRA IN-PILE TUBE DATA

Designation	GEH M-3	KAPL H-10	WAPD M-13 (WAPD-31)	GEH G-7 (GEH-10)
References	HW-68672	(See note 7)	1st: Vitro Report KLX-1793 2nd: (See note 7)	ASME Technical Paper No. 59-Met-4 and Corrections Dated 5/12/59
Reactor and Position	ETR-R-66-M-3	ETR-C-33-H-10	ETR-C-66-M-13	ETR-C-69-G-7
Present Status	Operating	Operating	Operating	Operating
Designed Operating Temperature	600°F	600°F	600°F	600°F
Designed Operating Pressure	2200 psig	2200 psig	2200 psig	2200 psig
Configuration	Cylindrical	Cylindrical	Cylindrical	Cylindrical
Dimensions	3.441 in. ID x 0.475 in. wall	3.000 in. OD x 2.124 in. ID	(See note 8)	3-7/16 in. ID x 0.210 in. wall
Material	Zircaloy-2	347 s/s	347 s/s	348 s/s
Assumed Heat Removal	Inside Surface	Inside Surface	(See note 9)	Inside Surface
Approximate Peak Thermal Neutron Flux	0.65×10^{14} n/cm ² /sec	5.5×10^{14} n/cm ² /sec	5.5×10^{14} n/cm ² /sec	3.5×10^{14} n/cm ² /sec
Maximum Gamma Heat	2.5 w/g	22 w/g	20 w/g	20 w/g
Maximum Calculated Stresses	Thermal: 6,535 psi Pressure: 9,262 psi Total: 15,797 psi	Thermal: 91,500 psi Pressure: 14,500 psi	(See note 10)	Pressure: 15,250 psi Thermal: 57,700 psi Total: 72,950 psi
Testing Done	Pressure Test	Pressure Test	Pressure Test	Pressure Test
Number of Tubes Exposed	1	3	2	2
Maximum Fast Neutron Flux (>1 MeV neutrons)	0.15×10^{14} n/cm ² /sec	3.5×10^{14} n/cm ² /sec	1.9×10^{14} n/cm ² /sec	1.2×10^{14} n/cm ² /sec
Length of Exposure (>1 MeV neutrons)	In Operation (Tentatively approved until 1/1/66)	1st: 1.08×10^{22} n/cm ² 2nd: 1.5×10^{22} n/cm ² 3rd: In Operation (Tentatively approved until 9/10/67)	1st: 7×10^{21} n/cm ² 2nd: In Operation (Tentatively approved until 10/24/66)	1st: 5.35×10^{21} n/cm ² (Localized overheating failure) 2nd: In Operation (Tentatively approved until 6/1/66)

TABLE IV (Cont.)

TRA IN-FILE TUBE DATA

Designation	GEH P-7	WAPD C-7 (WAPD-32)	KAPL L-42 (Formerly KAPL 37)	WAPD L-10
References	HW-54797 (Classified Report)	Vitro Report KLX 1807	L-42 In-File Pressure Vessel Dated April 16, 1959	(See note 7)
Reactor and Position	ETR-R-33-F-7	ETR-R-66-C-7	MTR-L-42	ETR-C-33-L-10
Present Status	Operating	Operating	Operating	Operating
Designed Operating Temperature	600°F	600°F	600°F	600°F
Designed Operating Pressure	2200 psig	2200 psig	2200 psig	2200 psig
Configuration	Cylindrical	Cylindrical	Cylindrical	Cylindrical
Dimensions	3.25 in. OD x 0.275 in. wall	3.542 in. ID x 4.250 in. OD	2.375 in. OD x 1.960 in. ID	2.636 in. ID x 2.876 in. OD
Material	Zr-2	347 s/s	347 s/s	347 s/s
Assumed Heat Removal	Inside Surface	Inside Surface	Inside Surface	Inside Surface
Approximate Peak Thermal Neutron Flux	1.45×10^{14} n/cm ² /sec	0.9×10^{14} n/cm ² /sec	2.5×10^{14} n/cm ² /sec	3.0×10^{14} n/cm ² /sec
Maximum Gamma Heat	1.3 w/g	1.2 w/g	10 w/g	15 w/g
Maximum Calculated Stresses	(See reference above)	Pressure: 14,100 psi Thermal: 75,600 psi	Pressure: 13,200 psi Thermal: 23,600 psi	Pressure: 91,500 Thermal: 14,500
Testing Done	Pressure Test	Pressure Test	Pressure Test	Pressure Test
Number of Tubes Exposed	2-347 s/s, 1-Zr-2	1	1	1
Maximum Fast Neutron Flux (>1 MeV neutrons)	0.137×10^{14} n/cm ² /sec	0.07×10^{14} n/cm ² /sec	1.1×10^{14} n/cm ² /sec	3.4×10^{14} n/cm ² /sec
Length of Exposure (>1 MeV neutrons)	(See note 15)	Indefinite (Tentatively approved until 4/15/70)	In Operation (Tentatively approved until 7/26/65)	In Operation (Tentatively approved until 12/6/65)

TABLE IV (Cont.)

TRA IN-PILE TUBE DATA

Designation	GEH B-3 (Formerly GEH-4)	KAPL F-10 (KAPL-33)	WAPD-15
References		1st: KAPL-M-HEB-1 2nd: (See note 7)	WAPD-CF-636 B & WC No. 5425
Reactor and Position	MTR-B-3	ETR-C-33-F-10	MTR-BT-1
Present Status	Operating	Operating	Program Completed
Designed Operating Temperature	Assumed 200°F	600°F	600°F
Designed Operating Pressure	Assumed 1000 psig	2200 psig	2200 psig
Configuration	Cylindrical	Cylindrical	Cylindrical
Dimensions	2.247 in. OD x 0.309 in. wall	1st: 2.360 in. OD x 1.960 in. ID 2nd: 3.000 in. OD x 2.125 in. ID	3 in. Schedule 80 Pipe
Material	Aluminum	347 s/s	316 s/s
Assumed Heat Removal	Inside Surface	Inside Surface	Inside Surface
Approximate Peak Thermal Neutron Flux	1.5×10^{14} n/cm ² /sec	2.0×10^{14} n/cm ² /sec	
Maximum Gamma Heat	1 w/g	17 w/g	4.2 w/g
Maximum Calculated Stresses		Thermal: 55,000 psi Pressure: 13,500 psi	Pressure: 14,796 psi Thermal: 26,983 psi
Testing Done	Pressure Test	1st: Thermal Cycled 2nd: Pressure Test	1st: Thermal Cycled 2nd: Pressure Test
Number of Tubes Exposed	5	3	2
Maximum Fast Neutron Flux (>1 MeV neutrons)	1×10^{12} n/cm ² /sec	1.7×10^{14} n/cm ² /sec	2×10^{12} n/cm ² /sec
Length of Exposure (>1 MeV neutrons)	Indefinite	1st: 6.5×10^{21} n/cm ² 2nd: 1.6×10^{22} n/cm ² 3rd: In Operation (Tentatively approved until 3/10/69)	1st: 2.5×10^{19} n/cm ² 2nd: 4.5×10^{19} n/cm ²

TABLE IV (Cont.)

TRA IN-PILE TUBE DATA

Designation	WAPD-26	KAPL-30 (Formerly NRL-1)	WAPD B-4 (See note 13)	WAPD L-12
References	WAPD-LSR (CR)-7	NRL Report 4469	WAPD-A1W(M)-307	(See note 7)
Reactor and Position	MTR-HB-5	MTR-HB-1	MTR-A-31, A-33, B-4	ETR-C-66-M-13
Present Status	Program Completed	Program Completed	Operating	Operating
Designed Operating Temperature	600°F	600°F	600°F	600°F
Designed Operating Pressure	2200 psig	2200 psig	2200 psig	2200 psig
Configuration	(See note 11)	See Figure 21	Cylindrical	Cylindrical
Dimensions	(See note 11)	See Figure 21	3 in. Schedule 80 with OD reduced to 3.375 in.	3.000 in. OD x 2.124 in. ID
Material	316 s/s	ASTM-A-201-B	316 s/s (347 s/s crossarm)	347 s/s
Assumed Heat Removal	Inside Surface	Inside Surface	Inside Surface	Inside Surface
Approximate Peak Thermal Neutron Flux			1.75×10^{14} n/cm ² /sec	5.5×10^{14} n/cm ² /sec
Maximum Gamma Heat	1 w/g at Tip	8.3 w/g at Tip	3.5 w/g	15 w/g
Maximum Calculated Stresses		See Figure 21	Pressure: 16,500 psi Thermal: 9,400 psi Total: 25,900 psi	Thermal: 91,500 psi Pressure: 14,500 psi
Testing Done	Pressure Test	Pressure Test	Pressure Test WAPD-15 Test	Pressure Test
Number of tubes exposed	1	8	5	2
Maximum Fast Neutron Flux (>1 MeV neutrons)	2×10^{12} n/cm ² /sec at Tip	2×10^{13} n/cm ² /sec at Tip	1.4×10^{13} n/cm ² /sec	2.4×10^{14} n/cm ² /sec
Length of Exposure (>1 MeV neutrons)	3.3×10^{19} n/cm ²	(See note 12)	(See note 14)	1st: 1.5×10^{22} n/cm ² 2nd: In Operation (Tentatively approved until 7/1/68)

NOTES

1. 1st: 4×10^{20} n/cm²
 2nd: 5.4×10^{20} n/cm²
 3rd: 3.57×10^{21} n/cm².
2. 1-1/2 in. schedule 80 pipe with a 0.191 in. thick hemispherical tip.
3. WAPD-29; formerly WAPD-13 and WAPD-25.
4. 1st: 1.1×10^{20} n/cm² (Bent during tank work)
 2nd: 4.8×10^{19} n/cm² (Annulus leak)
 3rd: 3.9×10^{19} n/cm² (Annulus leak)
 4th: 3.2×10^{19} n/cm² (Bellows damage)
 5th: 5.25×10^{20} n/cm² (Annulus leak)
 6th: 7.57×10^{20} n/cm² (Redesigned)
 7th: In operation; tentatively approved until 2/1/66
5. 7/8 in. OD: Thermal 4450 psi: pressure 10,570 psi
 3/4 in. OD: Thermal 3450 psi: pressure 12,200 psi
6. Six in-pile tubes and closures mounted in a stainless steel bushing; tubes arranged in a circular-cone configuration with closures at the 2, 4, 6, 8, 10, and 12 o'clock positions and tube tips slightly closer together than closures. Tubes are also staggered axially in three sets of two tubes.
7. "Design Support Document for H-10 Standardized ETR 3x3 Pressure Vessel," December, 1959 (KAPL internal document). This standardized tube is now used in the F-10, H-10, L-12, M-13, J-10, L-8, L-10 positions.
8. 1st: Small tube: 4.01 in. OD x 3.40 in. ID
 Large tube: 5.5 in. OD x 4.628 in. ID
 2nd: 3.000 in. OD x 2.124 in. ID
9. 1st: Controlled to give equal wall temperature on both sides.
 2nd: Inside surface.
10. 1st: Large tube: Thermal 97,500 psi:
 Pressure 14,600 psi
 Small tube: Thermal 47,000 psi:
 Pressure 15,300 psi
 2nd: Thermal: 91,500 psi:
 Pressure: 14,500 psi
11. 3 in. schedule 80 pipe with a 0.300-in.-thick hemispherical tip.
12. 1st: 4×10^{19} n/cm²
 2nd: 2.4×10^{20} n/cm²
 3rd: 2.4×10^{20} n/cm²
 4th: 2.5×10^{20} n/cm²
 5th: 2.66×10^{20} n/cm²
 6th: 2.4×10^{20} n/cm²
 7th: 3.14×10^{20} n/cm²
 8th: 3.14×10^{20} n/cm²
13. WAPD-30; formerly WAPD-15 and WAPD-26
14. 1st: 1.1×10^{19} n/cm² (Acid corrosion failure)
 2nd: 1.7×10^{20} n/cm² (Stress riser fatigue failure)

NOTES -- Cont.

3rd: 4.820×10^{20} n/cm² (Unlocated crack)

4th: 2.7×10^{20} n/cm² (Design change in tube)

5th: 3.96×10^{19} n/cm² (Crack in annulus purge line)

6th: In operation; tentatively approved until 6/18/65

6th: In operation; tentatively approved until 6/18/65

15. 1st tube: 7.3×10^{19} n/cm² (Annulus failure)

2nd tube: 6.1×10^{20} n/cm² (Annulus failure)

3rd tube: In operation (tentatively approved until 1/27/66)

Fig. 19 PAED HB-2 tube tip.

Fig. 20 First WAPD-13 tip.

STRESS #/IN ²				
LOCATION	TYPE	THERMAL	PRESSURE	DISCONTINUITY
A	LONGITUDINAL	8200	8000	2200
B	TANGENTIAL	7560	15,850	-320

Fig. 21 KAPL-30 tube.

proportional to the velocity. The high velocities require small fluid passages in the vicinity of the test sample which produce high pressure losses. To compensate for the large pressure losses, flow channels leading away from the specimens should be relatively large. High velocities may also cause design problems in flow distribution, vibration, fretting, erosion, and possible corrosion and crud buildup.

The reactor in-tank facility is assigned by Phillips' Project Engineering. This establishes gamma heat, thermal, and fast neutron flux design values. Approximate, but typical, values for existing TRA water loops are given in Table IV. As the values are constantly changing with time, design numbers for these parameters must be obtained from Phillips Petroleum Company prior to commencing design and periodically reconfirmed as revisions in the figures are established

by Phillips' continuing flux monitoring program.

C. LAYOUT CONSIDERATIONS

Following the nuclear heat transfer and flow or hydraulics calculations for which methods are presented elsewhere in this volume, and after receipt from Phillips of the facility assignments, the in-pile tube assembly layout can be started. Drafting, utilities, size limitation, safety, and piping design considerations given in paragraphs B and C of the next chapter will also apply in general to this chapter.

Because the irradiation charges for a particular reactor facility are determined by the nvt (neutrons per square centimeter) accumulated by the entire volume of that facility and because the quantity and quality of research data are usually dependent on an experiment accumulating the maximum nvt in a minimum period of time, there exists a basic desire by a program sponsor to design an in-pile tube which will handle the largest possible fuel subassembly. However, increasing the diameter of the tube increases the wall thickness required to withstand the stresses produced by an internal pressure and temperature specified as the environmental test conditions. The tube outside diameter is fixed by the facility dimensions, less the flow and fit tolerances. Consequently, the increase in wall thickness reduces the internal diameter of the tube and limits the size of the experimental specimen which the in-pile tube can accommodate. The increase in wall thickness not only decreases the neutron flux inside the tube, but also results in more severe thermal gradients and resultant greater thermal stresses. Since high-pressure, high-temperature in-pile tubes usually are immersed in low-pressure, low-temperature reactor water, it is necessary to surround them with an insulating jacket to prevent excessive thermal stresses across the in-pile tube wall. This jacket also prevents

boiling of reactor water and excessive heat losses during reactor shutdown periods. The construction of the jacket may simply consist of an air gap, vacuum, or inert gas between the pressure tube and jacket. In some cases, it may be necessary, in addition to or in place of the air, vacuum, or gas jacket, to have a jacket for secondary water coolant flow. The necessity for a jacket further decreases actual sample irradiation space. The jacket should be designed to withstand the external pressure exerted by the reactor process water: 60 psig at 125°F for MTR, 250 psig at 140°F for ETR, and 390 psig at 285°F for ATR (design conditions).

The layout of an in-pile tube assembly is extremely complex, as may be observed from Figures 22 through 24. Kaiser Drawings ETR-5528-MTR-642-M606 and -607 can be of great assistance to the ETR loop designer, as can Babcock and Wilcox Drawings ATR-1075-MTR-670M-426 and -427 to the ATR loop designer; there are no similar MTR drawings. There are hundreds of reactor detail drawings which may be required in order to complete an in-pile tube layout; these may be obtained from the cognizant Project Engineer. The more important MTR reference drawings may be ORNL or Blaw-Knox drawings, the ETR drawings may be O. G. Kelley or Kaiser, and the ATR drawings may be Ebasco or Babcock and Wilcox. These drawings should be used with caution, as many modifications and as-built dimensions are now contained on Phillips drawings. In addition, drawings of the loops in adjacent facilities are especially important to avoid conflicts. A study should also be made to avoid possible conflicts with reactor equipment such as guide rod cover, top head, ion chamber drive mechanisms, etc. A number of the drawings are still not in an as-built condition and it is essential that field dimension checks be made.

Because MTR/ETR in-pile tubes have a cylindrical configuration and reactor facilities an angular configuration, filler

Fig. 22 Plan view of MTR tank.

pieces must be designed with appropriate tolerances for fit in the reactor internals. In the case of MTR/ETR core filler pieces, they are usually designed with fit and hydraulic flow dimensions and tolerances to be exactly interchangeable with a reactor fuel element [94, 95, 96]. Other general layout considerations are as follows:

1. In-Pile Tubes

For the MTR or the ETR, in-pile tubes should not extend above the reactor tank flange. In the ATR, they extend through the top head. The overall length may be determined by such things as: (a) crane headroom; (b) low level reactor tank level versus effectiveness of the thermal shield for the closure cap; and (c)

reactor tank water level necessary for shielding in removal of experiments, flow tubes, and in-pile tubes and jackets which have been irradiated to saturation. Although removals may be accomplished by the use of a cask, it may be preferable to utilize the discharge chute from a safety and speed standpoint because of the less complicated procedure. The minimum MTR reactor tank water level at shutdown is automatically controlled by a key operated air valve and weir arrangement to the seal tank and is ≈ 109 feet 8 inches elevation or at the spider guide ring. The maximum water level of the MTR at shutdown is 118 feet 8 inches elevation. The minimum water level used to date at ETR is to the bottom of the distribution header or about 89 feet 4 inches elevation, and the maximum

Fig. 23 Plan view of ETR tank.

Fig. 24 MTR core pictorial showing location of high pressure water loop in-pile tubes.

level is 99 feet 11-3/8 inches or, with existing NR spool pieces, about 103 feet 6 inches elevation. The minimum water level for ATR is 92 feet 4 inches elevation, and the maximum level is 100 feet 8 inches elevation. All the minimum water levels given are quite arbitrary figures and would have to be checked by Health Physics monitoring; at the levels specified, allowable working time would be of short duration. Permissible minimum water levels depend on such things as core loading and the length of time since the last reactor fuel or experimental fission break or since the last decontam-

ination of loops in the immediate vicinity. The MTR discharge chute will accommodate a piece 6 inches in diameter by 14 feet 6-3/8 inches long, or 4-3/4 inches in diameter by 15 feet 5-3/4 inches long, or 3-1/2 inches in diameter by 15 feet 6-7/8 inches long; the ETR, 14-3/4 inches in diameter by 16 feet 1-1/2 inches long; and the ATR, 16 inches in diameter by 24 feet 9 inches long.

2. Reactor Plan View "Skyrights"

Reactor plan view "skyrights" are not to be violated with respect to reactor

operation functions. This applies to all parts of the in-pile tube assembly, including closure plugs, cross-arms (in-tank inlet and outlet piping), support brackets, instrumentation and gas annulus lines. This is particularly important, for example in a core facility, where the closure plug should not be larger than the reactor fuel section, and the crossarm piping should be routed above reactor nonfuel areas to permit reactor fuel element insertions, removals and transfers.

3. Layout

a. Layout should avoid pockets or dead legs from the crud collection and decontamination standpoints.

b. Layout should avoid abrupt metal changes which might result in stress risers as foci for possible fatigue cracking.

D. MATERIAL PROPERTIES

After the preliminary layout of the in-pile section is made, and prior to commencing the stress analyses, the materials of construction must be selected. References for material selection are included at the end of Chapter VI. Metals or alloys such as copper, mercury, silver solder, or Monel are not permitted within the reactor^[9]. For the most part, filler pieces and jackets are fabricated of stainless steel, although, due to the pressures and temperatures involved, aluminum would serve the purpose ideally, not only because of its strength, but also because of its low neutron absorption cross section and its low cost. Choice of in-pile tube material is more difficult. Zirconium alloy, the martensitic 400-series, and the austenitic 300-series stainless steels appear to be the most promising materials, depending on the specific application although aluminum, carbon steel, and Inconel-X have been used occasionally at TRA.

Zircaloy is best from a nuclear standpoint and has excellent corrosion re-

sistance and fair mechanical properties; however, it has the disadvantages of high cost and the fact that the ASME codes have not recognized it as a material for pressure vessels.

The 400-series stainless steels have greater thermal conductivity and lower coefficients of thermal expansion than the 300-series, can be heat treated to increase yield and ultimate strength without too much loss of ductility, and are somewhat less subject to stress corrosion in a halide environment. On the other hand, the chief objections to the 400-series are (a) the ASME Code does not recognize increased strength produced through heat treatment; and (b) their rate of insoluble corrosion product formation in reactor environments appears to be greater than that of the 300-series stainless steels.

Obviously, aluminum is not suitable for most applications because of its low strength. Carbon steel has the main disadvantages of lower allowable stresses, a comparatively high rate of corrosion product formation, and a lifetime from a neutron damage standpoint from 1/10 to 1/40 as long as that obtainable from some of the 300-series stainless steels. Inconel-X is difficult to fabricate.

As may be seen by referring to Table IV, austenitic stainless steels types 304, 316, and 347 are the materials most commonly used for in-pile tubes installed in the MTR and ETR. Type 348 is being used at ATR. At the present time, it would appear that the selection of one type over the others can be made solely on the comparison of the need for chemical decontamination of a loop and the allowable in-pile tube lifetime based on neutron damage to the material. Past experience has shown some loops require decontamination periodically to reduce maintenance personnel radiation dosages to tolerable levels. To date there have been 14 decontaminations attempted on loops at TRA, one on VH-3, two on P-7, four on G-7/M-3, three on HB-2, one on F/H-10, one on L-12/M-13, and two on J-10/L-8^[1,2,105]. The need for loop decontamination is

determined by the extent of plating out of corrosion and fission products in the system. Crud deposition may be due to an observed phenomena of the preferential plating out of corrosion products in a high radiation zone^[3,4], on areas of high pressure drop such as on the seats of throttle valves, or on cooled surfaces such as in heat exchangers. Other contributing variables to the need for loop decontamination are: water quality conditions; presence and size of fuel cladding defects, either accidental or intentional; experimental loading size and type; velocity and turbulence of flow; pressure; temperature; reactor flux; number of dead legs in the system; shielding employed; materials used in the loop and their surface finish; quality of the cleanup or purification system; and possibly others. Because of the corrosive nature of most decontamination solutions, types 347 and 348 stainless steel are most commonly used in the newer loops. Quoting from the 1961 edition of the Metals Handbook, page 442:

"Sensitization involves the intergranular precipitation of chromium carbide in austenitic stainless steels --- The principal effect is depletion of chromium content in a thin envelope surrounding each grain, promoting susceptibility to corrosive attack --- Prevention of sensitization is achieved by --- using alloy additions of powerful carbide-formers such as titanium, columbium, or tantalum to 'stabilize' the alloy by pre-empting the carbon and thereby protecting the chromium."

The main concern in radiation damage to the materials used in in-pile pressure tubes is that of loss of ductility. A brittle failure could conceivably result in damage to the reactor core, reflector, or even the vessel itself. At the present state of the art, it is impossible to extend in-pile tube lifetimes by the extrapolation of known data. A definitive assessment of what constitutes allowable in-pile tube radiation damage is made infinitely complex because of such unknowns as the sig-

nificant neutron energy range, the effect of the discontinuity in having the neutron damage almost wholly in the comparatively short reactor core section of the pressure tube, the effects of the numberless metallurgical and quality control variations in the fabrication of the tube, and the effect of temperature on the rate of radiation damage and upon the post-irradiation test work. Conventional criteria for material property evaluation such as matching yield and ultimate strengths or specified values for hardness, Charpy impact, elongation, and reduction of area are of questionable worth. Consequently, permissible irradiation exposure for type 347 stainless steel in-pile tubes has been gradually extended to a total integrated dose of 1.6×10^{22} fast (above 1 MeV energy) neutrons per square centimeter based on the following:

- (1) The overall strength of the in-pile tube material is considerably greater than it was at the time of installation.
- (2) Existing data indicates that, although loss of uniform elongation has occurred, local ductility is present to a limited extent.
- (3) That section of the in-pile tube whose properties have been most severely altered are internal of the reactor core and well protected from forces normally considered troublesome in utilizing material with low ductility.
- (4) Burst tests at room temperature of 10-inch sections of the core portions of the F-10 and J-10 in-pile tubes irradiated to 0.61 and 1.12×10^{22} nvt, respectively, resulted in ductile fractures with no fragmentation.

It is expected that further hot cell test work will extend the permissible in-pile tube exposures to 2.0×10^{22} and beyond. Because of the possible temperature dependency of irradiation damage, some of this test work will be conducted at loop operating temperature. Also, to attempt

to establish pressure cycling fatigue limits, one 5-inch notched section of J-10 in-pile tube No. 2 irradiated to 0.85×10^{22} nvt peak has been pressure cycled at room temperature to 6000 psi, 5 cycles per minute, to a limit of 2971 cycles; another 5-inch notched section of the same tube irradiated to 0.6×10^{22} nvt peak, similarly cycled, failed at 3258 cycles. This data compares to that of an unirradiated control sample test of 1556 cycles. Corroborating results were obtained from tests conducted in March of 1965 on L-12 in-pile tube number one irradiated to 1.59×10^{22} nvt peak.

In-tube lifetime is calculated as follows:

$$LT = \frac{T}{F \times \frac{MWd}{MW} \times \frac{sec}{day}} \quad (5-1)$$

where

LT = the tube lifetime in calendar years

T = the limit of the material experimental data in nvt

F = fast flux (>1 MeV)

MWd = the reactor's expected yearly megawatt-day production. For estimating purposes, 11,000 MWd may be used for MTR and 35,000 MWd for ETR.

MW = The reactor power level in megawatts

sec/day = $60 \times 60 \times 24 = 8.64 \times 10^4$.

Some development work has been done on annealing out neutron damage by means of induction heating and, should a nvt limit be determined, additional development work along these lines could extend the allowable life of in-pile tubes. On the other hand, it is quite possible that other, and yet unassessed, factors than neutron damage may govern the ultimate life of in-pile tubes. These possibilities include the following:

1. Thermal and pressure cycling fatigue or ratcheting not only in the in-pile tube itself, but also in the branch piping; stress risers at discontinuities; vibration

fatigue; creep.

2. Acid corrosion by irradiation dissociation of the cooling fluid or fluids^[5]; stress corrosion; common corrosion, including both chemical and electrochemical attack on the metal. This last type of corrosion includes factors such as galvanic corrosion or that occurring at dissimilar metal contacts, excessive number of loop decontaminations, preferential weld attack, hydriding, and possible mistakes or omissions in water chemistry control. Materials containing chlorides should not be allowed to contact internal or external surfaces of stainless steel systems in order to avoid the possibility of chloride stress corrosion of the metal when subjected to high pressures and temperatures. Magnesia or similar pipe insulations allow chlorides to be leached out when soaked with water, and specifications for low chloride-high-silicate insulation were established late in 1964. Other possible sources of chlorides are cleaning solutions, leak checking materials, gaskets, valve packing, lubricants, and polyvinyl chloride containers.

3. Erosion, fretting, or wear in mating parts.

4. Specimen performance deviations from expected or specimen failures and loop or reactor excursions. One such example of this type tube failure is given in a report concerning the ETR GEH G-7 tube fracture on November 30, 1959^[6].

5. Metallurgical flaws undetectable by nondestructive testing methods or fabrication quality control omissions or misinterpretations^[7,97,98].

6. Field installation quality control omissions or misinterpretations or accidental and unobserved damage by operations personnel.

7. Lack of proper periodic post-installation inspections (boroscope, wall thickness, metallurgical analysis, etc) and stress and safety re-evaluation^[92].

In summary, the properties and factors to be considered in the selection of the material for an in-pile tube are as follows:

- (1) High allowable tensile strength.
- (2) High allowable 0.2 percent yield strength.
- (3) High allowable stress producing a creep rate of 0.01 percent per 1000 hours.
- (4) Minimum stress producing rupture in 100,000 hours.
- (5) Reasonably good ductility.
- (6) Low notch sensitivity.
- (7) Thermal conductivity greater than 10 Btu/hr-ft²-°F-ft.
- (8) Coefficient of thermal expansion less than 10⁻⁵ in./in.-°F.
- (9) Low neutron absorption cross section.
- (10) Corrosion resistant enough to possibly eliminate necessity for corrosion allowance.
- (11) Minimum adverse change to heat-treated properties by exposure to fast neutron irradiation.
- (12) Stability of the heat-treated properties at the operating temperature of the pressure tube.
- (13) Ease of fabrication, including weldability to itself and 18-8 stainless steels.
- (14) Low cost.

It should be noted with reference to the first four items above that the ASME Code permits the use of annealed properties of materials only; the allowances on heat-treated materials at operating temperatures are: (a) 25 per-

cent of the minimum tensile stress; (b) 62-1/2 percent of the 0.2 percent yield strength; (c) stress producing creep rate at 0.01 percent in 1000 hours; and (d) 80 percent of the minimum stress producing rupture in 100,000 hours.

It should also be noted with reference to item (10) above that most existing in-pile tubes at TRA have not been designed with a corrosion allowance. Use of high purity reactor and loop water, that is both cation and anion demineralized and specific resistivity and pH controlled, is responsible for observed low corrosion rates. Section 1 of the ASA Code for pressure piping calls for a 50-mil allowance for power piping based on a 10-year life, using carbon steel pipe. Section 8 of the ASME Unfired Pressure Vessel Code requires only an allowance based on the desired life of the vessel.

E. DESIGN CALCULATIONS

1. Stress Considerations

The stress calculation fundamentals in the design of in-pile tubes are presented in Chapter VI and will not be given here. These calculations may not only be required for the critical core section, but also may have to be performed for the out-of-core section, grid section, and branch arms of both the pressure tube and the jacket, as well as for the reactor facility filler piece and the reactor flange. Special investigations may also have to be made of: (a) axial thermal stresses at thermal barriers designed to protect the closure and at the point the gas jacket goes through the reactor flange; (b) the stresses and deflections of branch arms and the vertical pressure tube reactions at the branch arms; and (c) the differential thermal expansions developed between the various components of the in-pile tubes and branch arms. A glance through the glass ports in the TRA reactors will convince the designer also of the necessity for a thorough vibration analysis. As the principles for this last named analysis were not given in the succeeding chapter, the reader is referred

to Chapter 9 of Reference 8 which also contains an excellent bibliography of 24 additional references. Because of this vibration problem, all parts which might be loosed by the vibration during operation must have anti-rotation devices or be countersunk, peened, latched, or other provisions made to ensure against mis-orientation, loss, or damage.

2. Expansion Bellows

Upon completion of the above calculations, it will be possible to design the expansion bellows, if required. For the design of bellows, the reader is referred to Reference 8, Chapter 7. Many of the bellows installed on in-pile tube jackets at TRA have been of triple layer construction. A sleeve, or "can", should be provided around the bellows for protection against damage during installation or subsequent service. The sleeve must be designed so as not to interfere with the movement of the joint and should be removable for inspection of the bellows. Because of the likelihood of bellows leaks, the original type F-10 in-pile tube installed in the ETR successfully used a packing system. Complete instructions and material for bellows replacement should be provided at the time an in-pile tube is first installed.

3. Shielding

Pre-formed removable shielding (or at least "shadow" shielding) should be provided for protection of maintenance personnel during reactor shutdown periods when radiation emanating from piping and equipment could reduce working time by causing whole-body radiation tolerances to be approached. A block shielding layout should also be provided for cross-arms extending beyond the MTR spool piece flange and for the front of the MTR beam hole facilities.

4. In-Tank Supporting Members

The in-tank supporting members may be designed by taking into account: (a)

the water-filled weight of the in-pile tube assembly including crossarms; (b) the weight of the shielding required; (c) the estimated dead weight of the structural members; (d) expansion reaction stresses; and (e) vibration. Although drawings are available showing reactor support rings and bolt hole size and location, field dimension checks are recommended. In-tank supporting members should provide independent adjustments in three directions -- vertically, East-West, and North-South. These adjustments should be at the in-pile tube end of the support. In bracket bolts which are difficult to reach and which are used in tapped holes, a galling problem has been experienced in the tight makeup of austenitic stainless steel studs of the 300-series. It has, therefore, been recommended that 17-4 pH stainless steel in condition H-1025[99] (Rc 35-42) be specified in this application, provided the stresses (applied or residual) are maintained below the yield point of the material.

5. High Pressure Closure

The final major design problem to be considered in this chapter is that concerning the in-pile tube high pressure closure. Types of closures for high pressure vessels are given in Table 5-2, pages 156 and 157 of Reference 9. The two principal types used at TRA have been: (a) those similar to the one shown in Figure 25 using mainly Class 2 ACME treads and a Buna-N, Teflon or silicone "O" ring or rings [10]; and (b) a "quick-closing" plug held in place with a snap ring or rings and using the same type "O" rings as the threaded closure.

The chief difficulty experienced at TRA with threaded closures has been that of thread galling. This has been only partially solved through the flash chrome plating or electrolyzing of the male threads [11,12], a recent limitation on torque by one sponsor of 15 ft lb[100], and the use of silicon grease on the threads at the time of installation. Aside from occasional leaks and temperature damage, threaded closure "O" rings have

Fig. 25 MTR L-42 experiment closure and thermocouple leadout system.

produced few troubles. The design of "O" rings is comparatively straight-forward, and the designer is referred to References 13 through 23. Neoprene "O" rings have been more or less universally banned for use in in-pile tubes because of the possibility of chlorides leaching out and resulting in chloride stress corrosion.

If the closure must be installed beneath reactor tank water level, the threaded type has been proven easier to handle than the snap ring type. The handling tool for the thread type closure must have a means of positive latching or connection to the closure to prevent dropping into the reactor tank, should easily and quickly connect and release, and should be sturdy enough to withstand considerable torque in case of slight thread galling.

Spare plugs and taps or thread chasers should be furnished at the time of original installation. The design of the threads should have a liberal safety factor to permit several thread dressings.

F. MISCELLANEOUS DESIGN PROVISIONS

1. In-Pile Tube Changeouts

As noted in Section D of this chapter, periodic replacement of in-pile tubes is necessary. Therefore, considerable ingenuity may be employed to reduce the expense and personnel radiation exposure involved in such a changeout.

Flexibility in assembly of components would be highly desirable, ie, changeout of jacket only; changeout of core section of in-pile tube only; changeout of in-pile tube only, excluding crossarm connecting tees, etc.

Standardization and some quantity production of in-pile tubes, such as has been done for the NR facilities at the ETR, has proved to be an effective way of reducing replacement costs.

Another way of decreasing the labor and radiation exposure involved in changing out in-pile tubes is to keep the number of in-tank welds to an absolute minimum. One method of accomplishing this is by using high pressure, high temperature flanges or coupling devices, a variety of which are commercially available. These have had only limited acceptance to date. Obviously, flanges or couplings may not be suitable where there is a space problem. Also, any type of coupling or flange and gasket selected by the designer may in time be subject to leakage due to thermal cycling and radiation effects and, consequently, any final decision as to their use must take into account whether or not the connector can be seal welded or easily replaced if a leak does develop.

A number of publications for further study of flange and gasket applications are listed in References 24 through 44.

Extremely detailed installation and removal procedures have been developed for in-pile tube changeouts at TRA in order to reduce shutdown time, maintenance costs, and personnel radiation dose rates. The cutting method for tube removal should be specified. Acetylene torch cutting is prohibitively difficult on stainless steels and is unsatisfactory due to the high probability of metal particles falling into the reactor tank and the equally high probability of releasing air activity into the reactor building. Hydraulic cutters have occasionally been used on small diameter tubes. Development of new mechanical^[45], hydraulic^[46], or cartridge firing stud driver type^[104] cutters would appear to be highly advantageous from the standpoint of labor costs and safety. The two most common methods of cutting stainless steel in-pile tubes and jackets at the present time are to use an arc welder and electrodes with marking identification classification of 6010 or 6011, using reverse polarity during the process, or, if there is to be a subsequent close fitting problem, to use a portable, electric hacksaw which mounts on the work piece. A

carbon arc and air method is used for cutting stainless steel, but because of considerable sputtering, this method is not used for cuts in or over the reactor tank. Most in-pile tubes are removed with the reactor fuel in the immediate vicinity removed and the tube full of water in order to prevent radiation from streaming up the tube. In the case of through-type tubes, freeze plugs have been used with varying degrees of success to prevent the draining of the tube during the cutting and removal process, viz, WAPD M-13 and HRT experience^[47]. Lugs, clamps, eyebolts, slings, or other provisions for installing and removing the in-pile tube assembly must be designed and the material furnished. Where possible, in-pile tubes are removed through the reactor discharge chute to take advantage of the water shielding. When this cannot be done because of length or because it is a tube installed in a MTR beam hole, a removal cask must be employed. Discharge cans are provided for many in-pile tube removals at MTR, viz, the NAA, KAPL, and WAPD discharge containers. Throughout all insertion and removal processes, a crane or other similar positive lifting or holding device must at all times be connected to the cask, tube assembly, or other equipment to ensure that nothing will drop which might damage reactor components. Canal storage brackets must be designed and supplied if temporary storage of in-pile tube, flow tube, experiment, or other radioactive materials is desired. Most experimenters will have a spare tube, accessories, and specimen on hand for immediate replacement into the reactor; if not, a nuclear mock-up "dummy" tube must be available on site for insertion to avoid abrupt flux changes which would affect the operating parameters of other experiments in the reactor. Summarizing, changeouts must not allow: (a) damage to reactor lattice, pressure vessel, fuel elements, or neighboring facilities; (b) spread of contamination; (c) scattering of debris into the reactor; (d) exposing personnel to harmful doses of radiation; and (e) taking more than a reasonable length of time.

2. Tube Jacket Accessories

Because of the frequency with which TRA in-pile tube jackets experience leaks, a drain should be provided. Also a pressure relief, indication, and alarm system must be incorporated into the jacket pressurizing or purging system to comply with the ASME Code on a closed pressure vessel with a heat source and because of possible pressure buildup from a radiation-induced element transformation. Any drains or purge lines must be designed to dispose of radioactive wastes.

3. Pressure Tube Accessories

The pressure tube will also require a rupture disc-relief valve system to comply with the ASME Code because it can be isolated with valves with a heat source acting on it. Valves should be furnished at the high and low points of the in-pile section for purging and draining purposes, each valve being equipped with a safety plug to negate the consequences of opening the valve accidentally. In the case of reentrant type in-pile tubes, procedures or design, or both, should prevent any possible syphoning action which could occur from opening a drain with the high point vent closed.

4. Flux Wires and Critical Facility Mock-Up

Flux wire provisions in the in-pile tube are highly desirable. If the tube is destined to be installed in the ETR or ATR core, a mock-up tube and specimen must be furnished for the respective ETR or ATR critical facility, complete with detailed drawings, types and weights of materials, and $\Sigma\alpha$ (absorption cross section) per holder in square centimeters. Standard lifting bails must be designed so that the existing critical facility lifting tool hooks can be used^[106].

5. Control Specimen Facility

In some cases, it may be necessary to design a control specimen in-pile facility in conjunction with the in-pile tube. This is an external tube which will hold a control sample and is identical to the test sample in-pile tube except that it is shorter in length. The control sample is thus contained in the same environment as the test sample except for the neutron and gamma flux.

6. Specimen Fragment Catchers

In the case of through tubes, specimen fragment catchers will be required to prevent hot fragments from lodging in the piping in the subpile room and thus preventing or limiting entry.

7. Remote Positioning Devices and Constant Strain Test Mechanisms

The KAPL H-10 in-pile tube is provided with a hydraulic mechanism for remote manual positioning of a test train within the reactor core to compensate for flux changes or for cycling the train in and out of the core automatically at adjustable rates, strokes, and dwell periods. The WAPD L-10 in-pile tube is provided with a mechanism to load specimens in constant tensile strain while the reactor is at power.

8. Instrumentation

Some of the more ordinary instrumentation which may be desired for the in-pile section of a loop are for flow, pressure, and temperature parameters under various operating conditions. These parameters may be inlet and outlet temperatures for all flow channels, specimen temperatures, differential temperature of the coolant across the specimen, pressure differential pressure, and flow. Multi-channel tubes may have flow measuring devices, impulse lines, and thermocouple leads for each channel. Instrumentation philosophy is given in Chapter VIII of this volume. However, a few mechanics

of the provisions for instrumentation will be briefly explained here.

Thermocouple leads and impulse or drain line tubing should be properly secured to avoid damage from vibration. Sufficient slack should be provided, however, so that thermal expansion will not pull thermocouple junctions off, break the leads, or place undue stress on the tubing.

The method of making thermocouple junctions should be carefully studied due to possible adverse radiation effects to welds, ceramic beads, etc. Means should be provided to protect thermocouple leads from possible high-temperature damage during in-pile tube welding by the fabricator or during the field installation. Physical labeling of thermocouple leads should concur with the drawing terminology. Low pressure, threaded, "O" ring-type closures have proved successful in sealing top leadout thermocouple connectors from reactor tank water. The type of thermocouple wire, ie, iron-constantan, chromel-alumel, etc, should match the characteristics of the leadout to the instrument.

G. QUALITY CONTROL

1. Welding

To reduce reactor down time, installation, long-term operation, and replacement costs, and to ensure maximum reactor, experimental, and personnel safety, stringent welding procedures and quality control measures must be enforced, both during fabrication and installation of in-pile tube assemblies.

ASME Boiler and Pressure Vessel Code, Section 3, Articles 4 and 6, requires that all welds be designed to permit radiography to code standards, and that, when this is not possible, radiography be conducted, nevertheless, to best available practice. Some sponsors have insisted that in-pile tubes (as well as other pressure piping and equipment) be constructed, installed, and radiographed in accordance

with standards established by the Department of Navy, Bureau of Ships^[48]. On the other hand, Phillips Petroleum Company has required only that installation be in accordance with ASME Code and Phillips Petroleum Company AED Specifications W-1-AED through W-6-AED supplemented by a weld tagout procedure [101]. Where radiography to Code Standards is impossible, or to supplement radiography, other methods, such as liquid penetrant, magnetic particle, or ultrasonic, may be employed^[49,50,51].

Most pipe welding done at TRA employs EB or consumable insert rings. This requires additional time in fitup of machined ends (usually to within 1/64 inch maximum tolerance), but it is believed that this time is more than compensated for by Phillips' experience in the low volume of weld rejects requiring repairs or rework. It is also thought that consumable insert welding produces a smoother inside surface, minimizing pockets in which decontamination solutions can react and which could possibly result in stress corrosion cracks. Consumable insert welding may not be required on WED and HED (drain) or other low pressure lines.

As proper welding procedures and weld inspection methods are essential to the integrity of an in-pile tube, included in the list at the end of this chapter are 30 additional references for further study [52-81]. Additional miscellaneous welding comments are given below.

(a) No welds are permitted within 6 inches of the reactor core so that forging or swaging may be necessary.

(b) At the TRA, argon of welding purity has been used in preference to nitrogen for a purging atmosphere. The sponsor should furnish a flow diagram and procedure for the weld purge for all field welding.

(c) Care should be exercised in specifying socket welds due to the difficulty in interpreting radiographs.

(d) It is recommended that specifications calling for field welding of carbon steel pipe (such as for Dowtherm systems) by the oxyacetylene process also permit heliarc welding. This will allow more flexibility and will not be unduly restrictive, since not all Phillips' welders are certified for the oxyacetylene process. Heliarc welding, at least for the first pass, also gives a cleaner pipe interior.

(e) Copies of all in-pile pressure tube radiographs are required to be sent to Phillips for inspection prior to installation of the tube in the reactor [102].

(f) Proper allowances for weld shrinkage appear to be a problem, not only in the fabrication of an in-pile tube assembly, but also in its field fit contingencies.

(g) An "WXR" or wholly radiographed ASME Code stamp plate is to be placed on the in-pile tube assembly or sent to the cognizant project engineer or site sponsor representative for inclusion in his files.

2. Surface Finish

The surface finish of an in-pile tube, particularly in the core section, must be at least 32 microinches, although most of the assemblies received at TRA have had a finish of as good as 4 microinches. High quality surface finish is most desirable to avoid any possible points of stress concentration. With the approval of the sponsor, thimble core section minor scratches or rust received in shipment, storage, or during installation may be repolished using size 00 emery cloth. From past experience, it is recommended that the interior of in-pile tubes be inspected with a boroscope a number of times during assembly to check for roughness due to machining, fitting, welding, and gauging.

3. "As-Built" Dimensions

"As-built" dimensions should be obtained of all component parts and of the complete assembly. Especially important from Phillips' point of view will be those dimensions which are closely associated with the reactor's fixed dimensions. Straightness is one such vital determination and is customarily made through the use of an instrument similar to the Brunson Model 75 optical transit. All tools, gaugeplugs, flow tubes, and sample dummy specimens should be shop tested, placing the in-pile tube assembly vertically or horizontally, whichever may be its future orientation in the reactor. These tools and internal gauging devices should be shipped with the tube assembly to assist Phillips in pre-insertion check-outs of maintenance and operating procedures.

Certified copies of all the above quality control inspection reports should be sent to Phillips' project engineering to avoid duplicating the work after the shipment arrives at TRA.

4. Cleaning

All materials used in an in-pile tube assembly should be shop-cleaned prior to assembly. As assembly progresses, the heavy layers of scale formed by welding should be broken up by mechanical descaling and be swabbed out of interior surfaces. This may be followed by degreasing, pickling, or molten salt bath descaling and rinsing and drying. A few of the more important recommendations and precautions applicable for cleaning pipes and tubing are as follows:

(a) In the use of cleaning solutions and rinses, pipes and tubes should be inclined at an angle of approximately 15 degrees from the horizontal to preclude the possibility of trapping air, and thus to prevent thorough cleaning or rinsing.

(b) The pH of solutions and the possibility of corrosive attack on the metal

should be carefully studied. Any surfaces which are not to be treated or which may be severely attacked should be masked.

(c) After draining the cleaning solution, the parts must be rinsed with demineralized chloride-free water, drained, and dried immediately. To assist in drying quickly, hot rinse water may be used, followed by exposing parts to a blast of hot clean air or infrared lamps. All openings should be plugged after drying.

(d) As perspiration or fingerprint stains are corrosive, clean, white, lint-free gloves must be worn by personnel handling the cleaned part.

(e) Many cleaning compounds are health or fire hazards, and plant safety specialists should be consulted before using them[103].

(f) Four additional references for cleaning in-pile tube assemblies are given[82-85].

5. Pressure Tests

It is required that pressure tests be conducted on all in-pile tubes and jackets in accordance with ASME Code requirements. These may consist of hydrostatic pressure and/or helium or bubble leak tests [86-90]. A convenient correlation between hydrostatic test pressure ratio and normal operating temperature is given in Figure 26. Note that, although the curves are drawn for a stress-temperature ratio of 1.5 as specified by Section 8 of the ASME Code, Section 3, the newer Nuclear Vessel Code, specifies that this ratio can be 1.25. The sponsor will usually have the vendor conduct the test to guarantee the fulfillment of the specifications and contract. If Phillips must conduct the tests at TRA, blind flanges, fittings, and other necessary equipment should be furnished by the sponsor. A specially designed concrete block cubicle is available in the MTR

mockup area for conducting pressure tests. Only high purity water should be

Fig. 26 Hydrostatic pressure test ratio for ss piping.

used in any hydrostatic pressure tests, and the equipment should be drained, dried, and handled in a manner similar to that given in paragraph 4 above to maintain high cleanliness standards.

H. MISCELLANEOUS

Details considered necessary to complete the design of an in-pile tube assembly are: (a) identification; (b) box design for shipment; and (c) the transmittal of spare parts and information to Phillips.

1. Identification

All dummy parts or gauges, tools, and spare parts should be marked with the sponsor program number, drawing number, and/or common name of the item by etching, vibratory tool, or other permanent marking. Arrows should be marked on handles of tools, closure plugs,

etc, to assist in orientation problems. Scribe marks or match marks should be freely employed on tools and assembly components for operation-type reference or, for example, to indicate where post-irradiation hot cell or canal cuts may be made. A marking on any item which may become radioactive should be such that it will be legible under 10 to 12 feet of water. Pencil marks are not permitted on aluminum because of possible corrosion caused by the graphite[91].

2. Box Design

High quality boxing and shipping methods for in-pile tubes are essential in order to attempt to prevent in-transit damage which could invalidate successful pressure tests at the sponsors' or vendors' plants. In the past, some of the boxing and crating design has been elaborate, using angle iron and H-beam supports and with pneumatic rubber bag or commercial spring-type shock absorbers and felt padding at critical wear points. All boxing should be designed for outside storage. Unloading instructions including weights of all components should be included in each shipment. Eyebolts or lifting lugs and special slings should be designed and furnished with the capabilities of the cranes in mind.

3. Transmittal of Information and Parts

Summarizing, information to be transmitted to Phillips may include the following, which is not intended to be a complete check list.

(a) Certification that materials are in accordance with ASTM specifications, composition analysis, and material testing results.

(b) Design calculations.

(c) Sponsor and vendor "as-built" assembly and detail drawings of the reproducible type.

(d) Specifications, data sheets, and copies of purchase orders and shipping notices.

(e) Chrome plating or electroplating procedures.

(f) Welding, cleaning, and pressure testing procedures.

(g) Vibration test results.

(h) Thermocouple calibration curves.

(i) Radiograph and quality inspection reports.

(j) Packing and shipping procedures.

(k) Installation and removal procedures (including reactivity calculations) and field function specifications.

(l) Special maintenance tools such as hydraulic cutters, taps, dies, etc, and handling tools and gauges.

(m) Spare parts and spare parts list.

(n) Critical facility and spare in-pile tube or dummy mock-up assemblies.

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CHAPTER VI

OUT-OF-PILE EQUIPMENT DESIGN

A. INTRODUCTION

As was the case in the preceding chapter on in-pile tube design, it is beyond the scope and intent of this work to present a complete guide to the mechanical design of out-of-pile equipment and facilities for TRA experimental loops. It is hoped, however, that enough introductory information can be given to assist novitiates in understanding a few of the basic fundamentals and problems involved in such an endeavor.

It will be assumed for the purposes of simplification that the design is a stepwise procedure. In actual practice, because of construction schedule deadlines and the large number of interrelated design variables, designing stepwise is impractical. Coordination and liaison among a number of different design and engineering groups becomes of paramount importance and is undoubtedly the customary method of designing in projects of consequential size. In such a cooperative venture, the major design problems become largely a matter of conception, preliminary calculations and drawings, judgment and decisions, some trial and error, and continual refinement of calculations and drawings made necessary by the progressive design changes.

B. DESIGN PHASE ONE: PRELIMINARY LAYOUT DRAWING

1. Prior Engineering Required

It is presumed that prior to commencing work on the preliminary layout drawing:

(a) The sponsor has perused the document, Policy and Procedure in the Operation of the Materials Testing Reactor

and the Engineering Test Reactor[3], and necessary clearances have been obtained.

(b) A scoping or conceptual design report has been prepared and approved.

(c) The nuclear, heat transfer, and flow calculations have been completed. (See preceding Chapters I, II, III, and IV).

(d) The process design and heat balance calculations have been effected, using minimum and maximum samples at normal, low pressure, and emergency operating conditions.

(e) A preliminary operational analysis has been made, and a basic flow sheet has been prepared.

2. General Drafting Standards

The drafting work in making the flow sheet and layout drawings must be accomplished in accordance with established standards. Engineered type experiments may require the preparation of 500 or more drawings and have an installed cost of as much as three million dollars. Consequently, in view of the number and complexity of these experiments, it is obvious why some degree of standardization is highly desirable, especially as regards: (a) drawing dimensions to accommodate existing filing facilities; (b) form, symbol, and abbreviation uniformity for easy and consistent readability by Phillips' Project Engineering, Operations and Maintenance personnel; and (c) engineering part numbering system for material and equipment identification for on-site receiving, warehousing, construction, and maintenance replacement purposes.

(a) The drafting standards of the Phillips' AED design group are given in the

Design Engineering Branch Standards Index. Supplementing these standards, a partial list of the applicable ASA standards is given below.

ASA Y 14.1	- 1957, "Size and Format"
ASA Y 14.2	- 1957, "Line Conventions, Sectioning and Lettering"
ASA Y 14.3	- 1957, "Projections"
ASA Y 14.4	- 1957, "Pictorial Drawing"
ASA Y 14.6	- 1957, "Screw Threads"
ASA Z B1.1	- 1960, "Unified and American Screw Threads"
ASA B 46.1	- 1955, "Surface Roughness, Waviness, and Lay"
ASA Z 32.2.3	- 1959, "Graphical Symbols for Pipe Fitting, Valves, and Piping"
ASA Z 32.2.4	- 1959, "Graphical Symbols for Heating, Ventilating, and Air Conditioning"
ASA Y 32.2	- 1954, "Graphical Symbols for Electrical Diagrams ^[1] "
ISA RP 5.1	"Instrumentation Flow Plan Symbols ^[1] "
ASME Standard 101	"Operation and Flow Process Charts"

Some drawings are still maintained on stick files. However, the majority of the drawings are on microcards. Because of this, it is recommended that the drawings be prepared without crowding too much detail on one sheet and with sufficiently large lettering and dimension numbering that microcard readout prints will be legible.

(b) Also supplementing Phillips' AED standards, an example for an engineering part numbering system is presented in Appendix E.

(c) In the field, drawings may be used one at a time without ready access to other reference drawings. It is, therefore, desirable to have legends placed on every applicable drawing rather than to refer by note from one drawing to another for symbol and abbreviation meanings. The same philosophy prevails for other similar cross referencing between drawings. By using mat overlays where duplication is indicated, very little additional drafting time will be involved for the extra convenience afforded.

3. Flow Diagram Standardization

To aid normal loop operating procedures and to possibly reduce the time required during emergencies when comprehension time for existing orientation of loop conditions must be at a minimum, flow diagrams are standardized as follows.

(a) The shading and size of the main loop piping lines are to be most prominent, ie, $\approx 1/8$ inch wide and dark.

(b) The secondary piping lines are to be less prominent than the main piping lines, ie, $\approx 1/16$ inch wide.

(c) The main instrument piping, such as the impulse lines, are to be less prominent than the secondary piping lines. The electrical and thermocouple lines are to be the same prominence as the main instrument piping, except that a dashed line is to be used rather than a solid line.

(d) Instrument sensing devices are to be shown in their proper location on the flow sheet. Connections to these flow devices are to be shown to the properly identified pressure switch and instrument. Any instrument connected to the reactor control system is to be marked "RC".

(e) Every valve in every piping and tubing line is to be shown graphically and is to be identified by its own characteristic number. Valve numbering systems presently employed on loops at TRA are varied, including mixed numerical and alphabetical designations; and, because of possible mistakes in correlation between drawings and loop valve tags, altering these numbering systems is not contemplated unless extensive loop modifications make such a change practical. On new loops in the design stage, however, the valve designations should be strictly numerical so as to best adapt to an IBM listing. It has been found from experience that an IBM tabulation affords the most convenient means to maintain a running inventory of the valve additions and deletions made necessary by the many, and apparently inevitable, year-to-year loop modifications. If desired, the sequencing of the valve numbers may be systematic, with blocks of numbers, including spare, unassigned numbers, being set aside for the various functions; viz, CV-1 to CV-10, air operated control valves, V-11 to V-99, primary water system, V-100 to V-199, pressurizer, V-200 to V-249, primary heat exchangers, V-250 to V-299, primary pumps, V-300 to V-499, purification system, V-500 to V-699, utilities, etc, with principal classifications and subclassifications possibly matching main headings and subheadings of the table of contents of the loop operating manual^[2]. It is unnecessary to place the valve functions on the valve tags except for the more important valves such as primary and pressurizer isolation valves, in-pile tube vent valves, purification system isolation valves, and the like, perhaps ten percent of the total number of valves, to assist operations personnel in accuracy of manipulation during critical operating procedures.

(f) Secondary lines are to be shown complete to the supply header. The impulse and control air lines are to be shown, but the instrument air supply lines and the relays need not be shown as they will appear on the instrument drawing.

(g) Where two or more pieces of the same type equipment are shown, such as pumps, heat exchangers, ion exchangers, etc, they are to be identified by numerical sequence, viz, No. 1, No. 2, etc.

(h) Electrical equipment on the loop should be shown but not the power supply lines.

(i) The above features are illustrated in Figure 42. This drawing is to be followed solely for the purpose of standardization to facilitate readability and is not to be used to formulate any design ideas.

4. Reference Material and Plant Facility Allotments

After the drafting mechanics rules are established and prior to commencing any preliminary layout work, Phillips transmits to the sponsor certain reference materials and plant facility allotments. Reference drawings will be transmitted to the sponsor on an individual need basis by the cognizant project engineer. Unless otherwise specified, only one copy of each drawing will be mailed. Commercial sponsors will be billed for nominal material and labor costs for the prints supplied. Because many reference drawings have not been revised to "as-built" basis, some field dimension checks may be advisable.

The reactor reference material available includes the following:

(a) Reproducible drawings and drawing lists of the MTR, ETR, and ATR engineering-construction firm drawings, sponsors' drawings of the equipment in adjacent facilities, and Phillips' drawings of additions or revisions to the original installations, as well as a complete photograph listing, on file in the Phillips' AED Design Engineering group.

(b) The MTR plant drawings, known as the Blaw-Knox drawings.

(c) The ETR plant drawings, known as

the Kaiser drawings. Especially for ETR installations, although a similar problem might exist in MTR or ATR, drawings of adjacent experimental equipment should be transmitted because of the increased probability of in-tank bracing and subpile room piping conflicts.

(d) The ATR plant drawings, known as the Ebasco or Babcock-Wilcox drawings.

A partial list of MTR-ETR-ATR allocations^[3] (in addition to reactor in-tank facility) of plant facilities is as follows:

(1) Cubicle space, shielded and unshielded, preliminary assignments being on a maximum allowable limits basis. Design will be to optimum compactness to utilize as little of the allotted space as possible, within practical limits, keeping in mind that:

(a) The replacement cost of the space is high. This cost is higher the closer one gets to the reactor.

(b) The maximum use of cubicle space should be made; ie, stacking of equipment vertically conserves valuable floor space.

(c) Compactness of layout should not compromise such things as accessibility and working space for installation, operating, maintaining, and removing or replacing equipment and safety features. However, no more space should be used for these purposes than is absolutely necessary. Some equipment which should require a minimum of maintenance will be relatively inaccessible. Installation may not be convenient, but in most cases space is worth more than convenience. Of course, pumps, heat exchangers, line heaters, or similar equipment requiring replacement over the years should be accessible and designed for removal.

(d) Equipment should be specified which will take the minimum space.

(e) Equipment that does not require shielding should be located outside of the shielded cubicle.

(f) No unused areas should be left in the cubicle, except in special cases where definite future changes in process requirements will make installation of additional equipment necessary.

(2) Miscellaneous equipment space including space allocation for: (a) fill, makeup, and oil pumps; (b) water purification and decontamination equipment; (c) motor-generators; (d) space coolers; (e) ion exchangers of the self-shielded type; and (f) Dowtherm equipment.

(3) Console space for: (a) panelboard; (b) desk and chair; and (c) pinboard.

(4) Miscellaneous instrumentation space including that required for: (a) D/P cells or transmitter cubicle; (b) D-N and gamma monitors; (c) crud probes and conductivity cells; and (d) glove box.

(5) Electrical equipment space for: (a) motor control centers; and (b) main circuit breakers.

(6) Miscellaneous equipment storage spaces for: (a) tools; (b) casks; (c) spare parts; (d) table and cabinet for sampling equipment manipulation and storage; (e) Dowtherm fire extinguishing equipment; and (f) hydrogen system for water chemistry purposes and helium system for jacket thermal barrier.

(7) Shielded cubicle to in-pile tube access including: (a) MTR slurry cubicle access, north and south; (b) MTR graphite holes (the temperature in VG-2 is measured at 200°F, and the maximum possible graphite temperature permitted in this hole is 570°F); (c) ETR vertical sleeves (temperature limitation is 200°F); (d) ETR horizontal sleeves both for reactor top block shielding and subpile room; (e) MTR spool piece flanges; and (f) ETR nozzles.

(8) MTR bottom-head penetrations: (a) four hydraulic rabbits; and (b) four dummy shim rod positons which are not furnished and must be negotiated. One of

these positions is in use; the others require modification.

(9) ETR bottom-head penetrations which are automatically furnished with through facilities.

(10) MTR-ETR top-head penetrations which are not furnished and must be negotiated.

(11) Miscellaneous piping, tubing, duct, and conduit runs must be negotiated. If any plant equipment must be moved, approval is to be obtained from Phillips' Plant Engineering group, and moving will be done at the sponsor's expense. MTR runs are along the reactor or building walls, floors, and ceiling; ETR and ATR have existing conduit and instrument line access holes which will be assigned.

5. Plant Utilities

The utilities shortages at TRA appear to be a chronic problem. Sponsors will submit for Phillips Petroleum Company approval a list of requirements during conceptual design, revising continually as the design becomes more firm. A form similar to that in Figure 27 will be used. In addition to the principal water and electrical power quantity and physical connection or tie-in allocations, secondary assignments may also be necessary for such plant utilities as plant and instrument air, instrument power, exhaust air, hot and warm drain access and capacity allocations.

6. Prefabrication

Past experience has shown that the desirable way in which to design the out-of-pile components of a loop is in the form of packages or prefabricated sub-assemblies which are as large as possible; viz, primary pump packages, make-up and fill pump packages, heat exchanger packages, etc. Each package contains the large, principal pieces of process equipment and closely associated equipment, structural supports, connecting piping,

and possibly some instrumentation and electrical items and piping insulation. It has also been found extremely beneficial, whenever possible, to completely assemble and test the out-of-pile loop at the sponsor's laboratories prior to shipping to the site. The size of the package, aside from shipping limitations, is limited by the site receiving facilities, mobile equipment, crane and elevator facilities, the reactor building access, hatchway and corridor sizes, and the allowable floor loadings.

a. Receiving Space

The indoor receiving space at the site is limited, and arrangements for inside storage must either be made well in advance of shipment, or the package must be designed for outside storage.

b. Mobile Equipment, Crane, and Elevator Facilities

For purposes of determining such things as method of removing the incoming package from the truck, movement within the reactor buildings, maximum allowable widths, thicknesses and weights, headroom available for maximum lengths, criteria for shipping container, skid and lifting ring details, etc, the mechanical designer will require certain mobile equipment, crane, and freight elevator data.

The mobile equipment and elevator statistics are given in Table VI. Note that the mobile equipment listed is only that normally assigned to the Test Reactor Area. However, there are other units situated on the NRTS site which may be procured by special arrangement.

The six cranes in the Materials Testing Reactor building are: (a) 30-ton main high bay crane; (b) 5-ton auxiliary high bay crane; (c) 1-ton high level crane with a 1/2-ton electrical cutout to avoid reactor damage; (d) 10-ton deep well hoist; (e) 2-ton canal crane; and (f) 15-ton mock-up area crane.

ETR EXPERIMENT NO _____

Date _____

Type _____

Facility Location _____

Project Engineer _____

UTILITY REQUIREMENTS

Service		Flow gpm	Inlet Pressure psig	ΔP Required psi	ΔT °F	Power Requirements KW
LDW	Normal					
	Maximum					
	Shutdown					
HDW	Normal					
	Maximum					
	Shutdown					
UCW	Normal					
	Maximum					
	Shutdown					
ECL	Normal					
	Maximum					
	Shutdown					
Diesel Power Bus 440 V 3 ϕ	Connected					
	Maximum					
	Normal					
Commercial Power Bus 440 V 3 ϕ	Connected					
	Maximum					
	Normal					

Fig. 27 Utility requirements form.

The four cranes in the Engineering Test Reactor building are: (a) 30-ton main high bay crane; (b) 5-ton auxiliary high bay crane; (c) 2-ton high level crane; and (d) 1-1/2-ton GE crane. Data on these cranes are presented in the following Phillips' drawings: ETR-B-1362, -1363, -1364, and -1365; and ETR-D-1584.

The seven cranes in the Advanced Test Reactor building are: (a) 40-ton main reactor crane; (b) 10-ton auxiliary crane; (c) 2-ton reactor room handling crane; (d) 2-ton jib crane; (e) 5-ton utilities crane; (f) 30-ton storage canal crane; and (g) 15-ton critical facility crane. Data on these cranes are presented in Babcock-Wilcox drawing ATR-1075-MTR-670-MS-4 and Ebasco specification drawings A-9950-S13, -S14, -S15, -S16, -S17, and -S19.

c. Reactor Building Access, Hatchway, and Corridor Sizes

The sizes of the main reactor building receiving doors, the hatchways, and the freight elevator doors are critical in determining the maximum size of the loop equipment package. These dimensions are given in Table V. In addition to these dimensions, aiseways and existing piping and equipment installations impose limitations such that the maximum allowable cubic dimensions for equipment packages may be arbitrarily established as follows: MTR, 7 feet 10 inches; and ETR and ATR, 9 feet 0 inches. These figures are to be used only as a rough guide for making the preliminary layout. Due to a continuously evolving experimental and plant alteration program, the final permissible sub-assembly size may be slightly larger or smaller.

d. Allowable Floor Loadings

For most efficient use of reactor floor space, compact equipment package design could possibly be limited by the allowable floor loadings. The allowable floor loadings for MTR, ETR, and ATR are tabulated in Table VII. Below are given four

design comments concerning floor loadings which may be of interest.

(1) It would appear that excessive floor loads are nearly always associated with lead shielding, ie, lead shielding walls, casks, self-shielded ion exchange columns, etc. Baseplates can be designed for these items which will properly distribute the load. For checking purposes, sponsors are requested to place the estimated equipment weights on preliminary drawings and the actual weights on the final approved outline drawings.

TABLE V

MTR-ETR-ATR REACTOR BUILDING ACCESS DIMENSIONS

MTR	
Main Receiving Door	12 ft, 0 in. wide x 14 ft, 0 in. high
Hatches	8 ft, 0 in. square
Freight Elevator Door	9 ft, 8 in. wide x 7 ft, 6 in. high (doors both ends of elevator)
ETR	
Main Receiving Door	14 ft, 0 in. square
Hatches	First Floor, No. 1 (south) 11 ft, 0 in. square First Floor, No. 2 (north) 9 ft, 0 in. x 11 ft, 0 in. Console Floor, No. 1 (south) 10 ft, 0 in. square Console Floor, No. 2 (north) 8 ft, 0 in. x 12 ft, 6 in.
Freight Elevator Door	10 ft, 0 in. wide x 8 ft, 3 in. high (door one end elevator only)
ATR	
Main Receiving Door	14 ft, 0 in. wide x 15 ft, 0 in. high
Hatches	Main Reactor Floor, 12 ft, 0 in. square First Basement Floor, 12 ft, 0 in. square
Freight Elevator Door	9 ft, 6 in. wide x 10 ft, 0 in. high (door one end elevator only)

TABLE VI

TRA ELEVATOR AND MOBILE EQUIPMENT DATA

Item Number	Type of Equipment	Make and Model	Lifting or Carrying Capacity	Platform Size or Fork and Width Length	Distance from Grade to Top of Fork (In Down Position)	Maximum Fork Height or Overall Lift
1	MTR Freight Elevator	Rotary Lift Company	10 ton	10 ft, 0 in. x 14 ft, 0 in.	-----	16 ft, 8 in.
2	ETR Freight Elevator	Canton Elevator & Mfg. Company	10 ton	10 ft, 0 in. x 14 ft, 0 in.	-----	38 ft, 3 in.
3	ATR Freight Elevator	Montgomery Elevator Company	12 ton	10 ft, 0 in. x 14 ft, 6 in.	-----	38 ft, 0 in.
4	Electric Tractor	Mercury Model No. A 800-5	-----	-----	-----	-----
5	Electric Forklift Truck	Clark Model No. 7025	7,000 lbs at 25 in. load center	6 in. wide x 42 in. long	2 ft	110 in.
6	Gasoline Engine Forklift Truck	Ross Carrier Company Model 15 SH 7	Range from 10,500 lbs with width from heel of fork 96 in. and C. G. 48 in. to 15,000 lbs at 50 in. and 25 in., respectively	8 in. wide x 42 in. long	2-1/4 in.	24 ft, 2 in.
7	Electric Forklift Truck	Yale & Towne Model K 51 AT-100	10,000 lbs at 24 in. load center	8 in. wide x 48 in. long	2-1/4 in.	110 in.
8	Motorized Platform	Allis-Chalmers Model CB 20	2000 lbs	40 in. x 49-1/2 in.	-----	-----
9	Straddle Carrier	Gerlinger Model SRH Size 7857	15 ton	52 in. wide x 5 ft, 6 in. high	6 in. (unloaded)	16 in.

TABLE VI (Cont.)

TRA ELEVATOR AND MOBILE EQUIPMENT DATA

Item Number	Primary or Free Lift	Range of Fork Width	Outside Turn Radius	Type Tires	Towing Capacity	Coupler Description and Dimensions	Miscellaneous
1	-----	-----	-----	-----	-----	-----	Lift from basement to 1st floor only; 3 landings.
2	-----	-----	-----	-----	-----	-----	Lift from basement to console to main floors; 3 landings.
3	-----	-----	-----	-----	-----	-----	Lift from 1st basement to 2nd basement to main reactor floors; 3 landings.
4	-----	-----	72 in.	Solid Rubber	30 ton	3-stage; middle stage will accommodate upper drawbar height of 15 in. by a drawbar thickness of 2 in.	
5	13 in.	12-1/2 in. to 40 in.	88 in.	Solid Rubber	-----	-----	
6	6 in.	18 in., 52 in.	12 ft, 8 in.	Pneumatic	-----	-----	-----
7	26 in.	18 in. to 50-1/2 in.	93 in.	Cushion	30 ton	Same as items No. 3 and 4 above.	
8	-----	-----	79 in.	Pneumatic	20 ton	1-stage; 1 in. diameter pin; bottom of coupler 1 ft, 2 in. from grade	Drawbar pull 750 lbs.
9	-----	-----	13 ft, 6 in.	Pneumatic	-----	-----	Gross weight 15,400 lbs.

TABLE VII

MTR-ETR-ATR ALLOWABLE FLOOR LOADINGS

MTR:

1. Canal Floor

2000 psf, including water

2. Basement Floor

2500 psf uniform distributed live load.

3. First Floor

1250 psf uniform distributed live load, or 50,000 lb moving dolly load, whichever governs.

4. Second and Third Floors and Stairways

100 psf uniform distributed live load.

ETR:

1. Canal Floor

2500 psf allowable live load in addition to water at depth of 20 feet for working area; storage area also limited to 250 psf uniform distributed live load and a concentric concentrated load of a 30-ton carrier above either pier N27.2-E21.1 or N29.3-E21.1 (see Kaiser drawing ETR-5528-MTR-642-S-9) provided there are no other heavy loads on areas adjacent to pier in canal or on first or console floors.

2. Basement Floor

From Kaiser data: 1500 psf uniform distributed live load, or 3000 psf (equivalent to 240 psf concrete walls of 12-foot height) for a loaded area of 2 to 4 feet wide and any length; this latter loading also applies to the trench covers; the floor and slabs over the hot sump tank pit, 300 psf uniform distributed live load or a rolling load of a 7000 lb capacity fork lift truck with a loaded weight of 17,200 lb and a maximum wheel load of

7000 lb. Authorized by AEC ruling, L. M. Leppich, Ref. ECE; PLR, to L. J. Weber, Web-647-58A-M, dated 11/3/58 to F. L. McMillan: 5000 psf.

3. Main and Console Floor

350 psf uniform dist. live load, or a rolling load of a 15,000 lb capacity fork lift truck with a loaded weight of 32,000 lb and a maximum load for dual wheels of 14,400 lb.

4. Main Floor

Truck aisle 1000 psf uniform, or a rolling load of a 20-ton capacity straddle carrier with a loaded weight of 58,000 lb; an area immediately around the reactor approximately 56 feet square with the center of square at reactor center has allowable uniformly distributed live loading of 750 psf; area over pipe tunnel from reactor structure to east wall rated 1000 psf; concentrated load limit of 30-ton cask equivalent over columns E22-N30.1, E23-N30.1, E24-N30.1, E25-N30.1, and E23-N27.1, and 3500 psf single loads on roof of pipe tunnel; greater loads over side walls of pipe tunnel may be possible but must be referred to Phillips' AED design engineers for analysis on an individual case basis.

5. Stairs and Landings

100 psf uniform distributed live load.

6. ETR Reactor Tank Block Shielding

Uniform distributed live load equivalent to a 50-ton lead and steel cask, to be referred to Phillips' AED design engineers for analysis in every case.

ATR:

1. First Floor (See Ebasco drawing ATR 1075-MTR-670-M-19)

TABLE VII-cont.

<p>A. General Storage and Laydown Area</p> <p>1000 psf live load</p>	<p>has 2000 psf allowable live load. The working canal is 250 psf live load in addition to 20 foot water depth.</p>
<p>B. Adjacent North Side of Canal</p> <p>2000 psf live load</p>	<p>B. Auxilliary Equipment Areas</p> <p>500 psf live load</p>
<p>C. Reactor Area</p> <p>1000 psf live load except on hatch cover C-2 which is 500 psf live load.</p>	<p>C. Cubicle and Sample Areas</p> <p>3500 psf live load</p>
<p>2. First Basement Floor</p>	<p>3. Second Basement Floor</p>
<p>A. Canal Floor</p> <p>250 psf live load in addition to water at depth of 20 feet for storage area.</p> <p>Two sections of the storage canal, each full; canal width, 12 feet long,</p>	<p>A. Auxilliary Equipment Areas</p> <p>500 psf live load</p> <p>B. Cubicle and Sample Areas</p> <p>3500 psf live load</p>

(2) Chipping of concrete and the cutting and retying of floor reinforcing bars will require authorization from the Phillips' Plant Engineering group; concrete chipping includes that necessary for drain lines. Because of the difficult stress analysis involved, this authorization, in most instances, will be quite difficult to obtain.

(3) Floors other than the basement floor should not only be analyzed for the loading of equipment on the top side of the floor, but also for piping, monorails, and any other loads hung on the underneath side.

(4) All loads should be calculated with the pipe and equipment assumed to be full of water.

7. Final Considerations

The final steps in the preparation of the preliminary layout drawing should consider the following:

(a) The tentative selection of equipment and piping sizes may involve no more than engineering judgment coupled with experience gained by the project engineer or sponsor on previously installed loops of similar size and characteristics.

(b) The layout should take into consideration relative ease of operations and maintenance functions as compromised by compactness. At this point, the designer may find it advantageous to make liberal use of templates and models^[4,5,6].

(c) The layout must comply with all health physics and safety regulations. In general, references for intelligent loop safety design include references 7 through 19 and internal communications 223 through 227.

(d) After the completion of the drafting work, six prints and one reproducible copy of the preliminary layout drawing or drawings are transmitted to the cognizant project engineer for his comments and the comments of other interested Phillips' departments.

C. DESIGN PHASE TWO: PIPING DESIGN

1. Piping Layout

Approval of the preliminary layout drawing permits the designer to commence the preliminary layout of the main loop piping. This presumes certain prerequisites in the form of good design practices and Phillips' ground rules or piping standards.

(a) In general, piping should run along the walls or ceiling of the cubicle for greatest working clearances. At the same time, accessibility must be such that the original installation effort and the subsequent operations and maintenance functions can be accomplished with a minimal expenditure of time notwithstanding the necessarily crowded conditions. Low points in the loop should be at least 18 inches above floor level for house-keeping reasons and to permit drain manifold installation.

(b) Dead spaces or areas of entrapment are to be eliminated or, if this is not possible, at least held to an absolute minimum. If they cannot be entirely eliminated, all dead spaces must be provided with a drain. It has been found from experience that the design should avoid pockets where fission products, crud, or chemicals might accumulate and create local hot spots or foci for corrosion. Minimizing high and low points also simplifies the vent and drain system.

(c) The ETR piping standards are detailed in internal communications[228,229,230]. The MTR has no similar sets of written rules but they will be furnished by the cognizant project engineer as the need warrants.

2. Pipe Sizing and Pressure Drop Calculations

After the preliminary piping layout is completed, a check can be made of the pipe internal diameter and main pump size assumptions. It is assumed, as mentioned

previously, that the flow and heat transfer calculations are complete and that the process flow diagram is established. The flow schematic will probably include standby cooling and pumping capacity. The arrangement of these standby heat exchangers and pumps may be in series or parallel and, because of the different flow resistance characteristics of the two types of arrangements, will have a direct bearing on the pipe and pump size checks.

The primary loop pressure drop is calculated^[20] using the physical and thermodynamic properties of water as given in Chapter III of Volume 2 of The Reactor Handbook: Engineering^[21]. If the pipe material is unknown at this time, it will be permissible to assume a relative roughness factor for Type 347 or 348 stainless steel, approximately 0.0001 for 2-inch diameter pipe. These are the materials in most common usage at the present time due to their apparent superior stress-corrosion properties when under attack by decontamination solution. The calculation is made using maximum expected flow at minimum calculated operating temperature and pressure. A typical calculation example is given in Appendix F.

The pressure drop calculations thus confirm the pipe internal diameter selection. Pipe wall thickness and outside diameter and pipe support analysis are determined from a detailed piping design analysis. For the sake of brevity, and as the basic theory is the same, the information in the remainder of this second design phase will apply not only to out-of-pile piping design but also to in-pile pressure tubes and pressure vessels.

In the most simple instance, a piping failure occurs when a certain function of stress or strain components reaches a critical value. The piping designer must know, first, the critical combinations of stress and strain at which failure occurs; and, second, the method or methods of calculating the stresses and strains, knowing the applied load. The first is concerned with the mechanical properties

of solids which is a relatively new field of science. It deals with such things as plastic deformation, creep, ultimate yield, working and thermal stresses, notch brittleness, fatigue, etc. The second is concerned with elastic deformation, the main subject of books which are usually entitled "Strength of Materials". A few references for study of the mechanical properties of solids are given in the reference list at the end of the chapter, 23 through 30.

3. Piping and Pressure Vessel Codes

The design of piping and pressure vessels for use at MTR and ETR must be in accordance with established code rules and standards. These rules and standards fix certain dimensional values and achieve minimum requirements for safe construction.

A listing of all standards and specifications and their mandatory effective edition references appears in the Appendix to section 7 of ASA B 31.1-1955, Code for Pressure Piping [22]. This listing includes the important reference of the ASME Boiler and Pressure Vessel Code, the four most valuable sections to the loop designer being: Section 1, which contains rules for the pressure design of boiler piping; Section 3, which is the Nuclear Vessel Code; Section 8, which is the Unfired Pressure Vessel Code; and Section 9, which contains the qualifications of welding procedures and operators of all pressure equipment.

Each of these has addenda and case interpretations which should be used in conjunction with the sections proper. Where required, the Code stamp should include the statement that material or fabrication is in accordance with applicable Code Cases.

Whenever liability may be involved because of design noncompliance with established codes, it is common practice to request a ruling on the design from the correct code writing body. Relating to ASME Code, inquiries should be addressed

to J. D. Wilding, Secretary, ASME Boiler and Pressure Vessel Committee, 29 West 39th Street, New York 18, New York. Sometimes, there is difficulty in classifying borderline pressure equipment and established national codes are of little help; eg, pulsation dampers are classed as pressure vessels in some states. Where there are doubtful cases encountered, it is advisable for the designer to check with local authorities; as examples, in Pennsylvania, the Department of Labor and Industry, Division of Boilers of the Commonwealth of Pennsylvania has jurisdiction; in Idaho, Industrial Accident Board, Boise, Idaho.

4. Stress Analysis

For the purposes of simplicity, design analysis of a pressure vessel or a piping system can perhaps be best illustrated by considering the stress (or strain) analysis as a function of three broad types of variables.

$$\sigma(\text{or } \epsilon) = f(\phi_1, \phi_2, \phi_3) \quad (6-1)$$

where

ϕ_1 = material properties

ϕ_2 = code type mechanical loads, taking into account the structure's geometry

ϕ_3 = noncode type thermal loads.

(a) As stated previously, Type 347 or 348 stainless steel are the pipe materials in the most common usage in the out-of-pile sections of the newer TRA high pressure, high temperature water loops at the present time. The reason for this is that it is believed that weld sensitization will not be a problem following use of most decontamination solutions in loops fabricated of stabilized stainless steels. Including the older loops, all have been constructed of austenitic chromium stainless steels, AISI Type 304, 316, 347, or 348 with the exceptions of one partially constructed of carbon silicon steel ASTM A

201, Grade B and two constructed of Inconel[a]. However, any material can be used which can be demonstrated as having the properties given in the preceding chapter on in-pile tube design.

It may be noted that the report entitled "Military Specification, Tube and Pipe, Corrosion-Resisting Steel, Seamless and Welded, Radioactive System Service" by the Bureau of Ships[31] specifies only Type 304, 304L, 347, and 348. These same steels are specified in NP-7454, also by the Bureau of Ships[32], to which was added Inconel[a] and Inconel-X[33,34].

Additional references for the selection of material for the out-of-pile piping, in-pile pressure tubes, and pressure vessels for loops are given in References 35-97, 99-125, 210-215, and 217-219, and in internal communications, 231 and 232.

For safeguard reasons, the cognizant Phillips' project engineer must possess evidence in support of adequacy of materials. This may take the following forms:

(1) Purchase specifications.

(2) Notarized certificates of the materials' physical and metallurgical properties.

(3) Completed ASME Forms U-1, "Manufacturer's Data Report for Unfired Pressure Vessels as Required by the Provisions of the ASME Code Rules".

(4) Receiving and inspection reports.

(5) Performance specification record of Welder's Qualification Test. In addition, all piping and pressure vessels received at TRA must be stamped with the type of material.

(b) Mechanical loads include the following three main categories:

(1) Dead loads such as the weight of the pipe or pressure vessel, contents, [a] T.M., International Nickel Co., Inc.

fittings, and insulation. It should also include lead shielding if it is planned to test defected fuel samples with exposed fuel. This category may also be considered for the sake of simplification to include dynamic loads such as earthquake and shock loading as from water hammer or flow surge effects.

(2) Internal and external pressure loads.

(3) Loads due to thermal expansion.

In general, the code-type piping design is a long, complicated, and tedious process. Tools have been developed by industry which are extremely useful to the piping design engineer such as Design of Piping Systems by the M. W. Kellogg Company, Simplified Piping Design by Spielvogel, and others by such firms as the Grinnell Company and Crane Company. Many shortcuts and simplified methods of design have been formulated, but they should be employed with caution. A simplified example of code design is given in Appendix G.

(c) As mentioned previously, codes establish minimum requirements for safe construction. Many design problems fall outside the normally accepted codes. The Bureau of Ships has formulated their own requirements[32] supplementing established codes. Information to date has shown the principles to have considerable merit when designing in areas not covered by existing codes; they are now accepted, with reservations, as one of the engineering tools. Occasionally proof tests are required to check the validity of new design principles. (See References 126 to 133, 216 and 235.)

The principal problem which has arisen in designing outside the ASME and ASA codes is that, due to the large gamma heat generation in the ETR and ATR cores, the thermal stresses induced in in-pile tubes may exceed the allowable yield point of the material used in fabricating the tubes. The cyclic nature of

these thermal stresses results in not only a fatigue problem^[134] but also the possibility of growth in the in-pile tube diameter^[135].

In brief, the currently accepted theory in the above two referenced documents permits the application of a thermal load causing a stress which exceeds the elastic limit of the material because yielding produces a relaxation of the thermal stress. At the same time, when the material is cycled through a thermal stress range larger than its yield stress, combined with a sustained internal pressure in the pipe or vessel, the designer should check that the stresses are not sufficiently high that there is a progressive expansion or increment of growth produced by each cycle of heat flux ("ratcheting").

Piping and pressure vessels should be designed so that the peak calculated thermal stress as compared to the known endurance limit is obtained from tests using complete stress reversal for millions of cycles. Thermal transients seldom reverse the direction of stress. Large power changes may occur in a reactor thousands of times during the expected life of the equipment, but not millions of times. Therefore, it is more economical to design to compensate for the fewer thermal cycles and to take advantage of the fact that, even though the sum of the thermal and pressure stresses may exceed the yield stress of the material, the yielding of the material will tend to relieve that portion of the stress produced by the thermal gradient.

An abbreviated example of designing in accordance with these principles is given in the second section of Appendix G. A partial listing of some MTR-ETR-ATR piping, in-pile tube, and pressure vessel design reports is given in References 136 to 144.

D. DESIGN PHASE THREE: MAIN LOOP EQUIPMENT DESIGN

1. Introduction

Other than the piping and in-pile tube, the principal equipment in the main pressure system of a MTR-ETR-ATR water loop with which the designer must concern himself, usually consists of the primary pumps, heat exchangers and line heaters, pressurizers, strainers, flow measuring devices, and valves.

2. Primary Pumps

The most common type of primary pump in use at the MTR-ETR-ATR at the present time is the hermetically sealed, vertical, centrifugal pump, more commonly called the canned rotor pump. The procedure in the selection of the proper size and number of pumps is relatively simple.

In Appendix F, sample calculations are given for the pressure drops at maximum flow for: (a) the out-of-pile portion of the loop; (b) the cross-arms; (c) the in-pile tube; (d) the fuel element contemplated having the configuration producing the maximum pressure drop; and (e) a standby pump in series with the normally operating unit. All five individual pressure drops are added arithmetically giving a total pressure drop, customarily expressed in feet of liquid being pumped.

With this value, together with the capacity as obtained from the heat transfer calculations, and knowledge of the specific gravity, the pump size may be selected from characteristic curves such as may be found on page 3 of the Westinghouse Descriptive Bulletin No. 57-550^[145].

Although only one pump could conceivably be selected for the desired head and capacity, current safeguard practices require at least two pumps, powered from independent electrical sources, both to be energized during reactor operation. If the loop experiment is considered to be more valuable than the pump, sometimes one of the running pumps will be wired so as to be able to be run to destruction. In addition to the running pumps, there may, for maintenance and operating flexibility purposes,

be one or more standby pumps, each electrically connected to operate from two different power sources.

The pumps may be in series or parallel flow. In series flow, the manufacturer's recommendation should be obtained since the bearings of the idle standby pump may not be designed to withstand the windmilling induced by the running pump.

It may be desirable to provide space in the pump package for other pumping units if it is planned to increase the flow requirements at some future stage of the experimental program. Because the lifetime of pumps for this particular type service is still somewhat unpredictable, it is also recommended that a spare replacement unit be provided by the sponsor.

Certain pump design factors which experience has indicated may be of value are as follows:

(1) The pump should be constructed of corrosion-resistant materials. Some bearings have failed after a comparatively short time. One pump presently in operation at the MTR has now logged nearly 70,000 hours running time. Considerable research has gone into screening bearing materials [146-148]. Bearings should also be selected to withstand the effect of decontamination solutions or should have purge connections for this purpose. One final point on pump materials: sometimes it is advisable to have the pump body and supply and discharge piping of the same material as the primary piping to avoid possible difficulties in the field welding of dissimilar metals.

(2) In forced-convection experiments, pumps can easily become sources of fluid contamination or paths of escape for radioactive vapors, gases, or fluids. Canned rotor pumps have been developed as one solution to the problem. Gas-seal pumps, although none are in use at MTR-

ETR-ATR, are another solution. To obtain zero leakage, the canned rotor pump can be hermetically sealed by means of renewable seal welds. Flexitallic gaskets have also proved to be satisfactory in many applications. The manufacturer's recommendation should be obtained for torque settings on the bolts and for seal welding procedure.

(3) Vents at the top of the pump casing and a low-pressure operation warning should be provided to help prevent cavitation. These vents should be operable from outside the cubicle.

(4) In the canned rotor type pump, wrap-around heat exchangers maintain safe winding temperatures. The cooling medium may be water or oil. For the prevention of damage to the pump, motor, loop, and experiment and for routine maintenance reasons, the following items may be furnished as accessories to the pump: cooling medium and bearing thermocouples, ammeter, wattmeter, running time count meter, and a motor starter of the quick reset type with carefully sized heat elements. During the pre-startup checkout, the loop should be carefully checked for actual pressure drops as compared to the calculated and for adequate flow through the pump cooling circuit.

(5) The designer should specify that the pump have a permanent proper rotation sign and a danger tag warning of the consequences of wiring the motor so that the pump turns backwards. Pressure taps at the pump inlet and outlet will facilitate checking for proper rotation.

(6) High background radiation can shorten the life of the pump by causing an electrical breakdown of the insulation or by causing hydrogen evolution from the insulating material and failure of the stator liner can. Two stator can failures have been experienced which may have been due to this or to a phenomenon from an inert gas system used for fast

loop pressurizing or to repeated hydrostatic pressure tests to $1\frac{1}{2}$ times design pressure. The pump designer should investigate the proposed pump radiation environment.

(7) For ordering pump and motor, it will be necessary to know that the MTR-ETR elevation above sea level is 4828 feet, the ATR is 4930 feet, and the ambient operating temperature is as determined in paragraph F-2 of this chapter.

It is not to be inferred from the foregoing that canned rotor pumps are the only ones utilized in high-pressure, high-temperature water loops. A canned rotor pump is unnecessary in a one-pass system since any leakage on the inlet side of the pump would not be radioactive. Other pump applications are given in References 149 to 151, and 220.

3. Heat Exchangers

Cooling capacity design for the primary heat exchangers should be on the basis of test operating conditions which will yield the worst combination (from a heat transfer point of view) of low tube side coolant temperature and high thermal load. The lowest temperatures are present at low pressure operating conditions. The tube side inlet and outlet temperatures are taken from the loop heat balance calculations. The shell side inlet and outlet temperatures, if reactor plant utility water is used, will be established by Phillips; viz, the ETR HDW supply temperature is about 130°F, and the maximum permissible return is 170°F.

Although the cooling capacity is calculated for the worst presently contemplated heat transfer conditions, it may be desirable to provide additional spare capacity for future experimental programs with even higher thermal loads.

For flexibility in loop operation and for ease of maintenance, it may also be desirable to have a duplicate unit paralleled in the line and equipped with isolation valves, and a complete spare unit on hand for quick replacement purposes in case of an operating failure.

At the present time at TRA, the majority of the loops employ water-to-water primary heat exchangers. Most of these are of the spiral coil type. The reasons for this are that: (a) they are compact and require less floor space than the more conventional types; and (b) the inherent flexibility of the spiral coil type has resulted in no failures due to the extremely large expansion stresses existing in most loop applications. Other types of heat exchangers have been employed successfully to permit the use of fixed tube sheets and allow for the differential thermal expansion between tubes and shell, such as the "U" tube configuration and specially designed bellows. Nearly all the primary heat exchanger difficulties experienced at TRA to date can be attributed to poor design in allowing for expansion and thermal cycling. Other reasons have been tubing-packing leaks, which were repaired by removing the packing and welding tubes to tube sheets, and stress corrosion cracking from chlorides in the coolant water, which was rectified by changing to higher purity demineralized water. Caution should be used in designing heat exchangers for UCW coolant because of possible silt fouling.

Although only the primary heat exchangers are in the primary loop circuit, it should be remembered for cubicle layout reasons that these heat exchangers may be only part of the heat exchanger prefabricated package. The complete package may consist of other heat exchangers such as a purification system cooler, a regenerative heat exchanger, and possibly one to cool plant utility water used previously as shellside coolant in other exchangers before returning it to the plant system. Other essential parts of the package are the structural steel supports, interconnecting piping, valves, fittings, and insulation.

For the detailed design of heat exchangers, the reader is referred to the report WAPD-AD (T) - 1095^[152]. The manual is written so that the reader can use it with minimum recourse to other

references. The report presumes that the exchanger heat transfer and thermal design calculations are complete and gives sketches and advantages and disadvantages of several typical heat exchanger arrangements, typical tube layouts, tolerances, limitations and allowances, glossary of terms, allowable stress values in design of formed heads, thicknesses of tube-sheets, tube layouts, the design of baffles, support plates, tie rods and spacers, and a complete calculation example. The report is very well presented and no attempt will be made to amplify it, except to state that for installation at TRA it will be required that all heat exchangers have vents, drains, nameplates, and ASME code stamps. Overpressure protection devices are also required on all heat exchangers except on low temperature, otherwise protected ones similar to purification system regenerative heat exchangers. A few references of general interest include 153 to 160.

4. Line Heaters

Line heaters are usually designed to heat the primary water from ambient to maximum operating temperature in a minimum period of time. The larger loops may require up to 8 hours; the smaller loops may require 2 hours. Four hours would be an average rate for a 3-inch loop. Spare heaters should be provided in the line, preferably in an electrical bank, so that in case of heat failures, normal operating temperature can be maintained until shutdown when repairs can be made.

The most common type of line heater at TRA consists of clamp-on halves or "clam-shell" type for easy initial installation and replacement. The heating elements themselves are usually contained in a steel sheath, insulated internally with magnesium oxide, and then the element assembly cast in halves in an aluminum-bronze matrix. Although the immersion-type heater is sometimes employed in a Dowtherm system, its application as a primary heater is undesirable because of possible difficulties with contamination due to a leak in the heater sheath.

The heater ID and the primary pipe OD should be machined to close tolerances for intimate contact for good heat transfer characteristics. Sometimes a material such as aluminum foil or thermal cement is used between the surfaces to improve the contact area. The clamps should be carefully designed to exert sufficient pressure to maintain the heat transfer contact and to compensate for both vibration and the type of structural supports provided for the heater sections.

The manufacturer of the heaters should provide the loop designer with the maximum limiting sheath temperature and the heat flux of the heaters in w/sq inch; viz, 900°F and 40 w/sq inch. Heat transfer calculations are then made of the differential temperature from design primary water temperature across the inside film, pipe wall contact and matrix, to the sheath. If the maximum sheath temperature is not exceeded, the design is valid. Thermocouple wells in the heater halves should be carefully designed to give accurate temperature measurement and over-temperature, and low-flow cutout controlling devices should be provided to prevent accidental melting of heaters.

Close attention must be paid to the design of the heater terminal connections. Because of high temperatures at these points, the terminals should be heavy, employ special fasteners to prevent loosening by vibration and use finned ferrules and extra conductors to reduce current density, or rheostat and Nichrome wire. To avoid terminal damage, it is recommended that they be enclosed by a protective sheet metal or clear plastic box.

Block-type insulation around the heater sections is preferable to other types to facilitate heater maintenance.

Electrical resistances of heaters, resistance to ground, and normal operating current as determined with a clamp-on ammeter should be logged during equipment check-out for future reference in checking out possible heater failures.

5. Loop Pressurizer

The loop pressurizer, sometimes termed the surge tank, and its heaters are designed to meet the following requirements:

(a) They provide heating capacity for the initial fill water pumped into the vessel as it is heated from ambient to design operating temperature, and provide space for the expansion of the primary water during the heating cycle.

(b) They maintain a constant pressure in the loop during normal operation and during transient or pressure surge conditions.

(c) They provide for the degassification of loop water and the purging of the noncondensable gases.

The pressurizer is customarily a cylindrical pressure vessel with ellipsoidal heads and is usually connected in parallel with the primary loop, between the inlet piping to the primary pump or pumps and the outlet from the in-pile tube. Heating of the pressurizer is accomplished by natural convection through electrically heated branch arms at the bottom of the vessel. Experience has shown a very satisfactory pressurizer pressure control system to be, briefly, a pressure controller transmitting pressure deviations in the form of electrical impulses which are picked up by a magnetic amplifier, changing the dc amperes to the core in a saturable core reactor. This correspondingly changes the inductance and the current flow to the pressurizer heaters. Overpressure damage is prevented by a rupture disc-relief valve system.

The degassifier is usually a spray nozzle arrangement within the vapor space at the top of the vessel with a top vent leading to the drain collection system. In isolated instances, the degassifier may consist of degassifier trays which liberate the gases from the steam by a scrubbing action.

Liquid level in the pressurizer is ordinarily sensed by a high leg piping loop from a point near the top of the vessel to a point near the half-full level and a low leg loop from the top to near the bottom. Each leg incorporates a condensate pot. Steam from the top of the pressurizer condenses and keeps the vertical legs full of water to avoid false readings. The differential pressure between the top and bottom taps of both legs is transmitted to the high side of a d/P cell and retransmitted on the low side to a level recorder. Temperature compensated curves convert the recorder readings to water level in inches.

Customarily, at least on the newer loops, the low leg pressure switches initiate high and low level alarms and turn off the heaters at extremely low water level and high primary water pressure; the high leg pressure switches turn the makeup pumps off and on.

The mechanical design of the pressurizer vessel is in accordance with principles presented in the piping and heat exchanger sections of this chapter. The design of the pressurizer heaters is similar to that given in the preceding section on the line heaters. An additional reference for pressurizer heater design is given in Reference 161. The calculations for the pressurizer size may be accomplished by conventional, though tedious, arbitrary and even somewhat empirical methods by determining: (a) the cold water loop volume and the increase in volume due to heating the water; (b) the volume of the cylindrical portion and dished ends of the vessel and the expansion of the initial fill water; (c) the maximum water level; and (d) the minimum water level assuming the loop to be cold and at pressure. These calculations are valid where only the initial fill is heated from ambient to design operating temperature. Examples of these calculations are given in pages V-33 to V-36 of Reference 143 and in Reference 162. An example of the determination of quantity of degassing flow based on heater

capacity is given in Reference 143, pages V-41-42. A possible method of calculation refinement for these parameters is indicated in Reference 163 in which equations, capable of being solved on the general analog computer, are developed for the various pressurizer design parameters.

Two final miscellaneous comments on pressurizer design may be pertinent. First, several existing loops utilize an inert gas system connected into the pressurizer vapor space for emergency pressurizing. The pros and cons of the inert gas system are thoroughly discussed in Chapter XI. Second, the design of the pressurizer should take into account the closeness of the vessel and heater legs to the concrete floor of the cubicle to prevent concrete damage from high temperature. In one instance at ETR, it became necessary, at great expense, to install cooling coils in the floor to counteract the damaging effect of the excessive temperature after the pressurizers were installed. The high temperature of the entire vessel may become a problem, not only because of possible floor damage but also because of the potential damage to other nearby equipment and plant and cubicle structural materials.

6. Strainers

During the installation of a loop, there is an excellent chance that foreign substances will be introduced into the inside of the piping, viz, stainless steel filings and chips, welding rod and slag, paper, rags, insulation, etc. In addition, of course, during operation, particulate corrosion products are formed (see Reference 164 including the 16 references at the end of the report). As these substances could increase wear by abrasion of closely fitting moving parts, become highly radioactive upon passing through the reactor, and, more seriously, could impede heat transfer mechanisms or could block flow passages altogether, strainers are installed in the primary loop and sometimes in the in-pile tubes. These strainers could take a number of different forms such as the

sintered stainless steel type, centrifuges, cyclones, ion exchangers, the bucket trap or screen types, and charged plates and magnets. Ion exchangers are used extensively in continuous purification systems. Due to the danger of the screen plugging up and stopping flow, strainers are of the flexitallic gasketed, flanged screen type with 20-mesh or larger openings and fitted with pressure taps. During the initial running of the loop, the screen is removed periodically and cleaned as indicated by observing the differential pressure across the strainer. Finally, when a reasonable period of operation produces an insignificant quantity of material at the screen, and prior to the insertion of a fueled experiment, the screen is removed. Some strainers are designed to be left in place during reactor operation by having bypass ports to prevent flow starvation to the in-pile experiment should the screen become plugged.

7. Flow Measurement

Flow measuring devices in the primary system usually number two. The common types of flow measuring devices, the preferred type, and miscellaneous design information are given on page 21 of IDO-16395 [165] and Chapter 6, Section D of Reference 166. One additional reference is 167, pages 15 to 22.

The design of the flow measuring devices has become highly specialized and the loop designer will do well to seek the advice of the flow instrument manufacturers. Upon request, they will supply the calculations and, from actual test results, will furnish certified calibration and temperature correction factors or curves. For flow measuring device design information, the reader is referred to Reference 168, Sections 5-9 to 5-13. The theoretical formulas are presented, followed by the quasi-empirical formula adopted by the ASME Special Research Committee on Fluid Meters. Some of the values of the constants are also given. A more recent reference is 20, Chapter 3, pages 14 and 15; Chapter 4, page 15.

8. Valves

The three types of valves employed in the primary system of a water loop are the main isolation and pressurizer isolation valves, check valves, and the pneumatic temperature and flow control valves.

a. Isolation Valves

Most main isolation valves are of the globe or plug type but there has been some experience gained with a special type ball-seat valve. The MTR and ETR water loops have main isolation valves, but because the valves invariably leak at high pressures and temperatures, they are seldom used except for certain out-of-pile loop modification and testing work. Because of the poor experience with these valves, none are provided at the ATR; instead "jack-leg" loops or U-bends are situated in the primary piping immediately adjacent to the reactor vessel to prevent accidental draining of the in-pile tubes. Double block valves are used to isolate the pressurizer and are used to prevent the loss of the steam dome to conserve reactor shutdown time for commencing maintenance work and to permit quicker start-ups.

Some typical materials for main isolation valves are as follows:

(1) Body, bonnet, stuffing box, and pipe nipples if welded to body are the same material as the main loop piping, eg, Type 347 stainless steel.

(2) Stem, disc, internal washers, liners, guides, pins, springs, keys, and bushings are a chromium ($\approx 20\%$) -tungsten ($\approx 15\%$) -nickel ($\approx 10\%$) -cobalt (balance) alloy which will have a minimum Brinell hardness of 375.

(3) Integral seating faces are a chromium ($\approx 28\%$) -tungsten ($\approx 4\%$) -cobalt (balance) alloy.

(4) Yoke bushings are aluminum or manganese bronze.

(5) Packing is teflon impregnated pure asbestos with a maximum chloride content of 100 ppm.

Additional information for valve material selection is given in References 169 to 171. Particular mention should be made of the fact that to date there is no solution to the problem of removal of contamination from stellited parts where the stellite has been applied by spray methods.

Valves for high pressure service require careful design of the valve stem, of the packing arrangement for the valve stem, of the seat, and of safety features. Special valve designs are described in paragraph 5.3 of Reference 56 and in References 172 and 173. Reputable valve manufacturers are a principal source of design information. Miscellaneous design considerations are listed below.

(1) The commercially available valve body ratings are 1500 lb, Class A, in accordance with ASA B 16.5.

(2) Body-bonnet joints shall be designed with a seal weld that may be cut and rewelded at least four times. As all seats and discs will fail sooner or later, the valve should be designed and constructed to permit easy removal and replacement of these parts. The valves should have in-line ports and all internal flow passages should be designed to provide a minimum pressure drop. The manufacturer should provide suitable information to allow the loop designer to make his pressure drop calculation.

(3) Valves should be designed to be exceptionally tight at all differential pressures across the disc in both directions and to incorporate back seating features for zero to 1.5 times design pressure at any temperature between ambient and design temperature. Exceptional tightness is defined, for example, as the maximum leakage across the seat not exceeding 1 cc/hr/in. of mean seat diameter or 2 cc/hr of nominal pipe size after 250

valve opening and closing operations. The manufacturer should specify the maximum allowable stem torque and design the hand-wheel extensions to obtain exceptional tightness.

(4) The packing, if used, should be capable of eliminating visible leakage under the full range of design service conditions by a balanced load application to the packing^[173]. The manufacturer should specify the maximum gland loading.

(5) Valves should be designed to withstand thermal transient cycles of, eg, 1000 cycles for both a 400°F range in 2 seconds and a 500°F range in 2 hours, and 25,000 cycles for a 50°F range in 15 seconds and also should be designed for maintenance-free operation and equipment life of 2000 cycles without repairs.

(6) Although an interlock could be provided which would permit valve closure only with the reactor at zero power level, or a scram signal could be initiated from a valve position indicator, a more satisfactory and less expensive method would be to use a tag-out procedure or provide a bracket, chain, and lock for the hand-wheel.

(7) Because of the closing force required for the larger size isolation valves to eliminate or minimize leakage, motor operated valves may be preferred. The motor operators should be located outside the cubicle walls to facilitate on-stream electrical maintenance and to decrease motor electrical insulation damage from radioactivity and high ambient temperatures. In the case of both manual and motor operated valves, the extension rod routing from the valve stem through a cubicle wall (cubicle should be designed so at least two or possibly three walls are accessible for this purpose) to the motor or handwheel should be a direct, straight-line one, if physically possible. If not, universal joints or countershafts will be required. The design of these devices will require much care to avoid binding forces^[174]. Extension rods should

be equipped with shear pins to avoid undue forces on valve trim and body. The pins should be located outside of the cubicle wall to permit replacement while the loop is at pressure and temperature. Lead inserts should be designed for the holes in the cubicle walls for the extension rods to prevent gamma-ray streaming. These inserts should be so designed as to take into account the linear expansion of the piping.

The fabrication and inspection of isolation valves should be of exceptionally high quality. The specifications for the valves must be painstakingly written. They should include a good description of the intended service, and the tight shutoff requirement should be clearly stated. The term "zero leakage" is inadequate. Inspection and tests at the manufacturer's plant should be required. Seats should not be welded into the valve body as heat distorts the metal and will prevent a tight seal.

The proper operation and maintenance of isolation valves will extend the period of time that the valves will retain their all-important, low leakage requirement. Some operation and maintenance points which should be considered are as follows:

(1) Low velocity and high quality water (ie, noncorrosive and containing few particles which will result in erosion) and as infrequent as possible operation of the valves will reduce the maintenance.

(2) Small leaks will only develop into larger ones, so prompt routine maintenance of a minor nature may prevent large overhaul jobs.

(3) Spare plugs and seats should be ordered at the time the valves are purchased and should be in matched pairs which have been factory lapped, or, as is more common at TRA, a complete spare valve should be provided.

b. Check Valves

The foregoing statements on the design of isolation valves also apply, for the most part, to check valves. Check valves are used in the primary loop when there are two or more normally running pumps in parallel arrangement. Check valves located at the discharge of the pumps prevent the possibility of a bypass flow circuit on failure of one of the pumps. The valves are usually of the globe type with in-line ports and may be of the lift, ball, or swing design. There should be an arrow etched, engraved, or cast on the body of the valve showing the direction of flow. If no other provisions are made to drain the piping ahead of the check valve when closed, a bypass line should be provided around the valve. A study of check valve water hammer characteristics is presented in Reference 221.

c. Control Valves

In Appendix F, primary loop pressure drop calculations for maximum flow were presented for the purposes of determining pipe size and pump head characteristics. The first step in determining flow control valve size would be to extend the calculations for minimum flow conditions and subtract this value from the actual available head of the pump (Appendix F). This difference in head must be lost at the control valve in order that the system will be in balance. The next step is to determine the flow coefficient or C_v factor for both maximum and minimum flow conditions. This flow coefficient is defined as the number of U. S. gal of water/min at 60°F that flow through a valve at maximum opening or with the plug at maximum lift, and a pressure drop of 1 psi measured in the inlet and outlet pipes directly adjacent to the valve body. The basic equation for a liquid is:

$$C_v = \sqrt{Q \frac{G}{\Delta P}} \quad (6-2)$$

where

C_v = flow coefficient

Q = flow in gpm

G = specific gravity of the liquid

ΔP = pressure drop across the valve in psi.

The high value of C_v , obtained at maximum flow, should probably be revised upward to allow for the accumulation of crud which decreases the pressure differential available across the valve, requiring a larger valve to maintain the desired rate of flow. How far the figure is adjusted upward will depend on the plug characteristic selected. Usually an equal percentage plug is selected for main flow control valves because they are best suited for the variable conditions of flow and differential pressure encountered in a test loop. Three-way valves, such as a temperature control valve, require linear characteristics so that the total area of the ports is exposed at all times. Obtaining from manufacturer's data the C_v characteristic of the size plug chosen and dividing by the C_v at minimum flow, C_v (maximum)/ C_v (minimum), the rangeability is determined and compared with the vendor's data to see if it falls within the capability of the type of plug characteristic selected.

The foregoing is over-simplified, and the reader is referred to an excellent series of articles on flow control valve design in Chemical Engineering by Werner G. Holzbock. The titles and dates of the publication of the magazine are "How Control Valves Behave" (March 9, 1959), "Control Valve Construction" (April 6, 1959), "How to Select Control Valve Size" (April 20, 1959), "Select Special Control Valves to Handle Fluids with Unusual Properties most Effectively" (May 4, 1959), and "For High Thrust Forces on Valve Plugs Use Control Valve Positioners" (June 1, 1959). This series is available as a reprint. The April 20 issue contains calculation examples so none will be given in this volume's appendixes. Many of the comments in the previous sections on isolation and check valves will also apply to control valve design with the following additional design considerations:

(1) Control valves should be designed to fail-safe on power failure, mechanical failure, or controller failure. In special cases, bypass controls should be initiated on primary flow failure from any cause.

(2) A bypass valve manifold should be provided to facilitate control valve repair.

(3) The control valve should be mounted vertically.

(4) For possible future increases in flow and to rectify an incorrectly sized valve, the standard plug and seat rings may be substituted with a smaller set, retaining the full, nominal body size.

(5) In some cases, two valves in parallel may be needed to give control over a wide range of conditions.

(6) Stops and manual overrides are advisable for most TRA applications.

(7) Air piping to the valve should preferably be 3/8-inch-OD tubing, except where the air line is less than 25 feet long, in which case 1/4-inch tubing is permissible.

(8) Occasionally trouble has been experienced during shutdown cold leak tests by excessively tightening the packing nuts on control valve stuffing boxes, causing binding of the stem after reactor start-up. The nuts should be finger-tight only, and permissible pressure drop figures during leak or hydrostatic tests should take into consideration a normal amount of control valve stem leakage.

E. DESIGN PHASE FOUR: SHIELDED CUBICLE DESIGN

1. Introduction

Following the design of all major primary loop equipment and interconnecting piping, the preliminary layout drawing can

be revised and the design of the shielded enclosure started. Design should be in accordance with the Uniform Building Code of Pacific Coast Building Officials Conference.

2. Allowable Radiation Levels

The walls and doors of MTR and ETR cubicles are to be sized so that the external dosage rates do not exceed the following:

Thermal Neutron	720 n/cm ² -sec
Fast Neutron	3 x 10 ⁻⁴ rep/hr
Gamma	3 x 10 ⁻³ rem/hr
Alpha	3 x 10 ⁻³ rem/hr
Beta	3 x 10 ⁻³ rem/hr
Total Dose Rate	3 x 10 ⁻³ rem/hr

The above permissible gamma, alpha, beta, and total dose rates are specified as 2.5 x 10⁻³ rem/hr for the ATR. The methods for performing the wall shielding calculations are indicated in Chapter IV. A sample set of cubicle shielding calculations for a loop might include the following considerations:

(a) The primary piping inside the cubicle, in the absence of better information, may be assumed to follow around the interior surface of the walls at a constant distance of 9 in. from the wall. The pipe may be treated as an infinite line source with the principal contributors being N¹⁶ generated in the loop water by the O¹⁶ (n,p) N¹⁶ reaction and the neutrons produced in the decay of N¹⁷. This calculation should not be confused with the maximum credible accident calculation which utilizes a point source.

(b) Within the main section of the cubicle, the radiation sources of importance are the pumps, heat exchangers, and piping. These sources may be lumped into point sources.

(c) It may be necessary to check scattering through cubicle wall penetrations such as the ventilation duct.

3. Barricade Design

Under certain circumstances, especially when high gas pressure may be involved, cubicle walls should also be designed as barricades in case of failure of equipment which could project metal fragments at high velocity, endangering personnel and possibly causing damage to other facilities. This design can be in accordance with Reference 177.

4. Shielding Materials

Due to the high shipping costs for high density shielding materials, the sponsor usually makes arrangements to have Phillips furnish them in the field. As a result, Phillips has prepared specifications for barytes or magnetite 4- x 8- x 16-inch concrete blocks^[233] and aggregate for high density concrete^[234]. Shot bags (commercial grade number 11 shot), 2- x 4- x 8-inch lead brick, lead wool, and portland cement (Type 1 low alkali) are standard Phillips stock items, but prior arrangements should be made for more than nominal quantities. The cubicle shielding doors may also be filled with shielding materials by Phillips by prior arrangement. Technical Report TID-10094 suggests a high density concrete mix ^[175]. A comparison between this mix and one recently used by Phillips is given below (quantities given are for 1 yd³ of concrete).

<u>Material</u>	<u>TID-10094</u>	<u>Phillips</u>
Barytes Aggregate (BaSO ₄), coarse	2210 lb	2778 lb
Barytes Aggregate, fine	1050 lb	2360 lb
Limonite (2Fe ₂ O ₃ - 3H ₂ O)	1220 lb	---
Portland Cement	610 lb	470 lb
Water	46 gal	325 lb

The average density of this concrete is 3.2 gm/cc. Report APAE-18, Supplement 1^[176], states that pouring is sometimes delayed because of a specification requiring a minimum temperature of 50°F on poured concrete and suggests relaxing the specification to 35°F or using additives.

Because of the fire hazard, the use of paraffin for neutron shielding is prohibited at MTR, ETR, or ATR. In laying courses of concrete block, lead brick, or polyethylene sheets, seams should be staggered to avoid radioactivity streaming. Additional references for radiation shielding are given in References 98, 178 to 182, and 222. In addition, a wealth of information is available from the Portland Cement Association, 347 Madison Avenue, New York 17, New York.

5. Structural Considerations

Structural steel members will probably not be required for the cubicle walls because: (a) the walls will not have exceedingly long spans; (b) they will be relatively thick as required from a shielding standpoint; and (c) the high compressive strength of the high density concrete blocks will withstand the stresses without the need for steel columns and beams. However, structural steel will be required for the pump, heat exchanger, pressurizer, pipe, valves and miscellaneous equipment supports, a monorail system to service all equipment, door, window and other access opening frames, baseplates, lead walls and equipment, steel floor plate and balconies, walkways, handrails, stairs, ladders, toe plates and guard plates for protection against rolling equipment damage, etc, if required. Material for these item shall conform to ASTM Specification A-7, latest edition, except that Bessemer steel is not acceptable. Construction and workmanship shall be as specified in the latest edition of AISC (American Institute of Steel Construction) "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings". All welding shall be

done in accordance with AWS (American Welding Society) Code for Welding in Building Construction. Pipe support design fundamentals are given in Chapter VIII of Reference 27. The loop designer will also find hanger detail selection drawings (such as drawing No. A-11 by Grinnell Co., Inc, Providence, Rhode Island) and catalogues (such as by the Power Strut Division of Unistrut Company, 933 W. Washington Boulevard, Chicago 7, Illinois) available from most major pipe suppliers to be of immeasurable assistance in designing piping supports. The piping support calculations should take into account the dead weight of pipe and lead shielding used to shield possible localized "hot spots" which could develop after a period of operation. This, together with the shielding of pumps, will permit continuous incubicle maintenance without personnel exceeding yearly exposure limits.

a. Baseplates

Baseplates should be provided for all equipment, and their thickness and area can be designed in accordance with the principles and examples given in the "Steel Construction Manual" as published by the AISC, using the allowable floor loading figures given in Table VII. Past experience has shown the most critical pieces of equipment regarding baseplate design and floor loading may be lead shielding walls, self-shielded ion exchangers, and access doors. For example, lead walls 12 feet high will impose a uniform distributed live load of about 8500 psf. Where needed, baseplates should be recessed and flush with the floor to prevent personnel tripping on them and to present fewer obstacles to fork lifts, welding equipment carts, and other rolling equipment. Many specifications call for the use of aluminum powder or "Embeco Standard" for grouting material beneath baseplates. However, it is believed that these materials are seldom necessary for equipment service conditions at TRA. Very briefly, the good design of anchor bolts and grouting involves first the selection of the bolt size from allowable bolt load in pounds tension and

obtaining from any reputable expansion bolt manufacturer's catalogue such things as diameter and depth of hole in concrete, minimum edge distance, etc. Note that the minimum distance between bolt centers should be four diameters in concrete. Next, it is specified that concrete and baseplates are to be clean, and the equipment shimmed and leveled 3/4 to 1 inch depending upon the weight of the equipment above foundation. A one part cement-two part sand-water admixture is forced into the space between baseplate and foundation; puddled; and a dry one part cement - two part sand mixture added to compensate for excess moisture. After a week, to develop full strength of the grout, the shims are removed.

b. Floors

The cubicle floors should be sloped to a drain and lined for a foot above the floor either with a thermoplastic polyester-resin or with high polish stainless steel to facilitate cubicle decontamination. The same treatment, water glass, or a thick coat of high gloss paint may also be advisable for the interior cubicle walls for the same purpose. Door dams should be provided to prevent leakage outside of the cubicle.

c. Access to Equipment

For the original installation and maintenance of equipment, if floor space permits, it would be preferable to have the layout and access door sized so that a fork lift truck could be used to place and remove any piece of equipment in the cubicle without regard for sequence. Removable concrete ceiling slabs and a crane might be used for this purpose. In most cases, however, a monorail system will prove to be the best solution to the problem. It could have a removable section at the door, the overall length being such as to enable easy transfer of equipment from fork lift or hand cart and also to service outside cubicle equipment, such as self-shielded ion exchange columns. The monorail superstructure may be

selected using beam formulas or may be picked off a chart such as that shown in Figure 28. The monorail should probably be supported by members resting on baseplates set near the top of the cubicle walls because of the limited floor space and low allowable ceiling loading. The crane and hook may be selected from any reputable manufacturer's catalogue. The eyebolts and lugs on equipment will have the hole size determined by the crane hook cross section.

d. Doors

Doors should be provided in any shield wall where stacking and unstacking of individual blocks for maintenance purposes would be a time-consuming job. The access door may be suspended on ceiling guides and rollers, on floor rails, or may be hinged. Because of its shielded weight, it may require some kind of mechanical advantage to open and reseal easily and quickly, or sufficient maneuvering room should be available if a fork lift is to be used. Of course, in the latter case, it should be determined that the door weight is within the capacity of the fork lift truck.

e. Lighting

Industrial type lights are installed as part of the standard reactor plant equipment. However, the light fixtures will undoubtedly have to be moved to enable installation of equipment in the cubicle. Therefore, adequate illumination should be designed into the cubicle so that all work may be carried on safely without undue strain to eyes. Illumination shall conform to the provisions of Part XIV of Reference 7 and ASA Code A-11, "Industrial Lighting".

f. Stairs and Ladders

Stairs shall be designed in accordance with ASA Code A-12, "Floor and Wall Openings", and stairs and access designed to ASA Code A9.1, "Building Exits". Ladders shall be designed in accordance with ASA Code A-14, "Construction, Care,

and Use of Ladders". Ladder construction shall conform with the standard drawings by Phillips' AED, Engineering Branch Standards, MTR-B-5449, MTR-B-5450, and MTR-B-5451.

F. DESIGN PHASE FIVE: PLANT UTILITIES AND LOOP AUXILIARY SYSTEMS

1. Introduction

To complete the cubicle design, it will be necessary to layout and design all plant utility connections and the loop auxiliary systems and their connecting line penetrations into the cubicle.

2. Plant Utilities

Information concerning the plant utilities available at the MTR, ETR, and ATR is given in References 183 to 185 and Chapter XI in this volume.

Whenever plant utilities are used, the source must be protected from experimental failure by suitable circuit breakers, fuses, relief or check valves, or similar equipment. Even though some of the water utilities have a valve at loop connection points (such as the gate valves on the ETR HDW system), the sponsor must furnish a positive shut-off valve inside or near the cubicle. A bleed line may be required between the double block valves, except possibly for the lower pressure services such as the 300 psig HDW or the 125 psig UCW systems at ETR. Regarding the UCW connection, lines 2 inches and under may have screwed connections. At the discretion of the cognizant project engineer, it is suggested that a tee with a blind flange be furnished at utility connection points so that if another experimenter requires the services in the future, the connection can be made without disturbing operation of the loop. For supporting utility lines, inserts are available in the MTR, ETR, and ATR basement ceilings for the installation of hangers. These are the concrete type, slotted,

ESTIMATED SAFE LOADS FOR MONORAIL SUPERSTRUCTURES

REPRINTED FROM THE 1952-53 FLOW DIRECTORY

CONCENTRATED LOADING

LATERALLY FREE

LATERALLY FIXED

SPAN IN FEET →		4	6	8	10	12	14	16	18	20	22	24	4	6	8	10	12	14	16	18	20	22	24
SINGLE CHANNELS	3	4.1	1.1	1150	520								1650	1100	800	650							
	5	5.0	1.2	1300	620								1850	1200	900	700							
	4	5.4	1.9	2200	1050								2850	1900	1400	1150	950						
	6	6.25	2.1	2450	1150	650							3100	2100	1550	1250	1050						
	5	6.7	3.0	3600	1800	1000							4450	2950	2200	1800	1500	1250					
	7	9.0	3.5	4350	2250	1250							5300	3550	2650	2100	1750	1500					
	6	8.2	4.3	5400	2750	1550							6500	4350	3250	2600	2150	1850	1600				
	8	10.5	5.0	6500	3350	1950	1200						7550	5000	3750	3000	2500	2150	1900				
	7	9.8	6.0	7950	4200	2450	1550						9050	6000	4500	3600	3000	2600	2250	2000			
	9	12.25	6.9	9300	5000	3000	1900						10300	6850	5150	4100	3450	2950	2600	2300			
	8	11.5	8.1	11000	5900	3550	2250						12100	8100	6050	4850	4050	3450	3050	2700	2400		
	10	13.75	9.0	12500	6800	4100	2650						13450	8950	6700	5400	4500	3850	3350	3000	2700		
DOUBLE CHANNELS	3	4.1		3300	2000	1350	980						3300	2200	1600								
	5	5.0		3700	2200	1450	1000						3700	2400	1800								
	4	5.4		5700	3650	2450	1750	1280					5700	3800	2800	2300							
	6	6.25		6200	4000	2700	1980	1400					6200	4200	3100	2500							
	5	6.7		8900	5800	4000	2850	2100	1600	1250			8900	5900	4400	3600	3000	2500	2200				
	7	9.0		10600	6850	4720	3450	2550	1950	1520			10600	7100	5300	4200	3500	3000	2600	2300			
	6	8.2		13000	7900	5800	4250	3150	2400	1980	1480		13000	8700	6500	5200	4300	3700	3300	2900			
	8	10.5		15100	10000	6700	5000	3780	2900	2300	1800		15100	10000	7500	6000	5000	4300	3800	3400	3000		
	7	9.8		18100	12000	8450	6250	4650	3650	2850	2320	1850	18100	12000	9000	7200	6000	5200	4500	4000	3600		
	9	12.25		20600	13700	9750	7250	5550	4350	3480	2750	2050	20600	13700	10300	8200	6900	5900	5200	4600	4100	3800	
	8	11.5		24200	16200	11500	8500	6500	5050	4080	3320	2400	24200	16200	12100	9700	8100	6900	6100	5400	4800	4400	
	10	13.75		26900	17900	13000	9650	7400	5800	4650	3700	3060	26900	17900	13400	10800	9000	7700	6700	6000	5400	4900	
STANDARD I BEAMS	3	5.7	1.7	2350	1250	750							2500	1650	1250	1000							
	5	6.5	1.8	2500	1350	850							2650	1800	1350	1100							
	4	7.7	3.0	4300	2450	1500	1000						4400	3000	2250	1800	1500						
	6	9.5	3.3	4750	2750	1700	1150						5000	3350	2500	2000	1650						
	5	10.0	4.8	7100	4150	2650	1780	1250					7250	4850	3650	2900	2400	2050					
	7	12.25	5.4	8050	4750	3050	2100	1450					8100	5400	4050	3250	2700	2300					
	6	12.5	7.3	10900	6600	4300	2950	2100	1550				10900	7250	5450	4350	3650	3100	2750				
	8	14.75	7.9	11900	7250	4750	3300	2350	1750				11900	7950	5950	4750	3950	3400	3000				
	7	15.3	10.4	15500	9650	6400	4500	3200	2350	1800			15550	10350	7750	6200	5200	4450	3900	3450			
	9	17.5	11.1	16700	10350	6950	4900	3550	2600	2000			16700	11150	8350	6700	5550	4750	4150	3700			
	8	18.4	14.2	21300	13600	9200	6500	4800	3600	2750	2150		21350	14200	10650	8550	7100	6100	5350	4750	4250		
	10	20.5	15.1	22600	14500	9850	7100	5200	3900	3000	2350		22600	15050	11300	9050	7550	6450	5650	5000	4500		
C-B SECTIONS	3	25.4	24.4	36600	24200	16550	12100	9150	6950	5450	4300	3400	36650	24400	18300	14650	12200	10450	9150	8150	7350	6650	
	5	30.0	26.7	40000	26500	18550	13550	10250	7900	6150	4900	3900	40050	26700	20050	16000	13350	11450	10000	8900	8000	7300	
	7	31.8	36.0		36000	25350	18550	14050	11000	8500	6800	5600	35960	27000	21600	18000	15400	13500	12000	10800	9800	9000	
	8	35.0	37.8		38000	26750	19750	15000	11500	9150	7400	6000	37850	28400	22700	18900	16200	14200	12600	11350	10300	9450	
	10	5.25-17	14.1		14000	10100	7450	5700	4450	3520	2850			14100	10500	8450	7050	6050	5300	4850			
	8	5.25-21	18.0		18000	12900	9500	7250	5650	4500	3200			18000	13500	10800	9000	7700	6750	6000			
	6	5.50-24	20.8			15600	11850	9280	7400	6050	4900	4100	2750		20800	15600	12500	10400	8900	7800	6900		
	5	5.75-21	21.5			15650	11750	9100	7200	5750	4650	3800			21500	16150	12900	10750	9200	8050	7150	6450	5850
	4	5.75-26	27.6			20100	15100	11700	9150	7350	6000	4850			27600	20700	16550	13800	11850	10350	9200	8300	7500
	3	8'-33	35.0			21000	16750	13650	11350	9500	8050	6850	5850			26250	21000	17500	15000	13150	11650	10500	9550
	2	6.50-25	30.9			17600	13800	11050	9000	7400	6100	5150	4300			30900	23200	18550	15450	13250	11600	10300	9250
	1	6.50-32	40.7			30500	23000	18150	14400	11750	9750	8100	6750	5600			30550	24400	20350	17450	15250	13550	12200
	0	8'-40	51.9				24800	20250	16800	14100	11900	10200	8750			38950	31150	25950	22250	19450	17300	15550	14150
	0	6.75-30	41.8				18950	15200	12400	10250	8550	7150	6000			41800	31350	25100	20900	17900	15700	13950	12550
	0	6.75-32	54.6				41450	31300	24750	19750	16200	13350	11150	9350	8000		40950	32750	27300	23400	20500	18200	16400
	0	8'-43	62.7				47000	37600	29950	24500	20250	17000	14450	12300	10500		47050	37600	31350	26850	23500	20900	18800
	0	7'-36	56.3					25750	20850	17000	14150	11800	9950	8500			56300	42250	33800	28150	24150	21100	18750

NOTES: LOADINGS FOR SPANS TO THE RIGHT OF THE — LINE WILL EXCEED A RATIO OF SLENDERNESS OF 40.
LOADINGS TO THE RIGHT OF THE — LINE WILL PRODUCE DEFLECTION OF 1/360 OF THE SPAN OR MORE.
LOADINGS TO THE RIGHT OF THE — LINE WILL PRODUCE DEFLECTION OF 1/300 OF THE SPAN OR MORE.

Fig. 28 Estimated safe loads for monorail superstructures.

rated at 2000 lb/linear ft, and similar to that shown in Section 2-C Tr in Sweets Catalogue. The load rating of the hangers is meaningless without considering the allowable ceiling-floor ratings, taking into account any loads on the top side.

One of the most difficult of the utility design problems is the cubicle exhaust and cooling system. The cubicle ambient temperature is limited to a maximum of 130°F, primarily because there appear to be deleterious effects to common types of air-operated control valve diaphragms and electrical wiring insulation where this temperature is exceeded. Also, Section 14 of Reference 7 states that vessels, pipes, or flues storing or conducting fluids at temperatures in excess of 100°F must be covered with sufficient insulating material to reduce the exposed surface to 100°F or less. This stipulation is made in order to reduce the likelihood of contact burns to maintenance and operating personnel. The ventilation system should also prevent acute discomfort or physiological damage to personnel from abnormally hot working conditions inside the cubicle. Insulation may be required to prevent damage to adjacent concrete or concrete block surfaces which have a high temperature limit of 200°F or to motor bearings and other similar equipment. The required air face velocity in primary cubicles, as well as in sample houses, laboratory hoods, and the like, is 150 ft/min. This minimum air face velocity is specified in order to minimize the possible spreading of airborne contamination both inside and outside the cubicle, and to reduce the subsequent cleanup work that would be necessary from such an incident.

a. Insulation

The first step in the design is to make the pipe and equipment thermal insulation design calculations in accordance with the principles given in the "Heating, Ventilating, and Air Conditioning Guide" which is an annual publication by the American Society of Heating and

Ventilating Engineers (ASHVE). The most common type of insulation used at TRA is the 85 percent magnesia type; a listing of standard thicknesses and other valuable information may be obtained from the Magnesia Insulation Manufacturers Association, 1317 F Street, N.W., Washington 4, D.C. In selecting the manufacturer of the insulation, there is recent evidence that accidental wetting of some magnesia insulations results in the leaching out of chlorides and subsequent stress corrosion cracking of stainless steel, which should be kept in mind. It appears that specifying an excess of soluble silicates over the chlorides in an insulation will effectively prevent this source of stress corrosion cracking. Although none is in use at TRA at the present time, an insulating material of concentric cylinders of stainless steel is on the market which might prove valuable for application in loop cubicles. It is water resistant and, because halves easily clip together, it is an ideal material for use for loop piping, which experience indicates undergoes numerous modifications during its lifetime.

b. Cooling and Air Load Estimates

Following the design of the insulation, a cooling load estimate should be made. A form similar to that shown in Figure 29 is convenient for this purpose. From next-to-the-last line of the form, the total cooling load in Btu per hour is determined. A less accurate, although less time-consuming, method of determining heat losses from piping and equipment inside the cubicle is given in pages 275 to 279 of Reference 144. The cubic volume of the cubicle is then determined and multiplied by the required air changes per hour to give the required cfm rating of the space cooler fan. In the ATR calculations, page 338 of Reference 144, six air changes per hour were used; and as a conservative assumption, no credit was given for the plant ventilation system to absorb any of the heat generated in the cubicle. The cubicle plant ventilation system should be checked for its ability

FIGURED BY _____	COOLING LOAD	PROPOSAL NO _____
DATE _____	ESTIMATE SHEET	PLAN NO _____
		SHEET NO _____

JOB NAME _____ LOCATION _____

ROOM _____ WIDTH _____ LENGTH _____ HEIGHT _____ FLOOR _____

OUTSIDE DESIGN CONDITION DRY BULB _____ WET BULB _____ DEW POINT _____ R H _____

INSIDE DESIGN CONDITIONS DRY BULB _____ WET BULB _____ DEW POINT _____ R H _____

TIME _____ AM _____ PM LAT _____ WALL COLOR DARK _____ LIGHT _____ MEDIUM _____

WINDOWS SHADES _____ AWNINGS _____ BARE _____ ROOF COLOR DARK _____ LIGHT _____ MEDIUM _____

ITEM NO	ITEM	DIMENSIONS	AREA SQ FT	U	D	SENSIBLE HEAT BTU PER HR
1	EXTERIOR WALL GROSS					
2	EXTERIOR GLASS					
3	EXTERIOR WALL NET					
4	PARTITIONS, NET					
5	GLASS IN PARTITIONS					
6	FLOOR					
7	CEILING OR FLOOR					
8	MISC					
9	TOTAL CONDUCTION HEAT GAINS	*	*	*	*	*
	EXCESS SOLAR HEAT GAINS					
10	NORTHEAST					
11	EAST					
12	SOUTHEAST					
13	SOUTH					
14	SOUTHWEST					
15	WEST					
16	NORTHWEST					
17	ROOF					
	GLASS FACING					
18	NORTHEAST					
19	EAST					
20	SOUTHEAST					
21	SOUTH					
22	SOUTHWEST					
23	WEST					
24	NORTHWEST					
25	SKYLIGHTS					
26	MISC					
27	TOTAL EXCESS SOLAR HEAT GAINS	*	*	*	*	*
	DUCT HEAT GAINS					
28	INSULATED DUCTS					
29	NON INSULATED					
30	TOTAL DUCT HEAT GAINS	*	*	*	*	*

ITEM NO	BODY HEAT GAINS	HEAT LOAD BTU PER HOUR	
		SENSIBLE	LATENT
31	SENSIBLE NO OF PEOPLE X 225		
32	LATENT (QUIET) NO OF PEOPLE X 175		
33	LATENT (ACTIVE) NO OF PEOPLE X 450		
34	TOTAL BODY HEAT GAINS		
	EQUIPMENT HEAT GAINS		
35	ELECTRIC LIGHTS WATTS X 3.4		
36	SMALL ELECTRIC MOTORS TOTAL HP X		
37	LARGE ELECTRIC MOTORS TOTAL HP X		
	ELECTRIC EQUIPMENT		
38	NO X		
39	NO X		
40	NO X		
41	NO X		
	GAS EQUIPMENT		
42	NO X		
43	NO X		
44	NO X		
45	NO X		
	WARM PIPES		
46	FT X		
47	FT X		
48	TOTAL EQUIPMENT HEAT GAINS		
	INFILTRATION HEAT GAINS SENSIBLE FACTOR LATENT FACTOR	*	*
49	ROOM VOLUME CUBIC FEET		
	MISCELLANEOUS HEAT GAINS		
50			
51			
52	TOTAL MISCELLANEOUS HEAT GAINS		
	SUMMARY OF HEAT GAINS		
	ITEM NO ITEM		
	9 CONDUCTION LOAD		
	27 SUN LOAD		
	30 DUCT		
	34 PEOPLE LOAD		
	48 EQUIPMENT LOAD		
	49 INFILTRATION		
	52 MISCELLANEOUS		
53	TOTAL SENSIBLE HEAT GAINS		*
54	TOTAL LATENT HEAT GAINS		*
55	TOTAL HEAT GAINS = ITEM 53 + ITEM 54		*
56	SENSIBLE HEAT PERCENTAGE = ITEM 53 X 100 = %		
57	DRY BULB TEMPERATURE OF AIR SUPPLY = DEGREES		
58	WET BULB TEMPERATURE OF AIR SUPPLY = DEGREES		
59	RISE IN DRY BULB TEMPERATURE OF AIR SUPPLY = ROOM DB - ITEM 57 = DEGREES		
60	TOTAL AIR SUPPLY = ITEM 53 X ITEM 59 = CFM		
	HEAT LOAD OF VENTILATING AIR		
	NO OF PEOPLE X CFM PER PERSON = CFM OUTSIDE AIR		
	CFM OUTSIDE AIR X BTU PER CFM = BTU PER HOUR		
	TOTAL COOLING LOAD ON COILS AND REFRIGERATING UNIT		
	ITEM 55 + ITEM 61		
	TOTAL COOLING LOAD BTU PER HOUR		
	TONNAGE EQUIVALENT OF COOLING LOAD = ITEM 62 = 12,000 TONS		

Fig. 29 Cooling load estimate sheet.

to handle possible airborne activity in accordance with Reference 186. The space cooler is a recirculating system and, therefore, will not be of assistance in the removal of airborne radioactive particles. At this point, in-leakage around cubicle doors and penetrations into the cubicle, requirement for automatic louvers or vents at bottom of cubicle to maintain cubicle air changes per period of time and to maintain a negative pressure inside the cubicle at all times should be determined. Experience at TRA indicates that the design of a cubicle ventilation and cooling system should take into account proper air circulation and also preventive maintenance of the space cooler. With the cubicles being customarily congested, air openings should be of sufficient quantity, spacing, and distribution to eliminate dead air areas. The space coolers should be located so that cooling coils, fan belts, and motors can be replaced with a minimum of time and expense.

c. Duct Design

After the air flow calculations, the duct design can commence in accordance with ASHVE principles and the procedure given in paragraph 6 of Reference 187. The sponsor must furnish the butterfly valves for use in the ventilation system branch lines. Other general references are given as 188 and 189.

3. Auxiliary Systems

In the more complex loops, there may be a great many secondary, supporting, or auxiliary systems, as they may be termed, essential to the proper operation, control, or safety of the primary or main flow system. The auxiliary systems may include any or all of the following: (a) Dowtherm or secondary coolant system; (b) water preparation, makeup, and initial fill or charging system and decontamination system; (c) purification and water chemistry and sample system; (d) pressure relief, vent, and drain collection systems; and, finally (e) electrical and instrumentation systems including instrument panelboards or con-

soles, motor control center, motor-generator pump speed control systems, transmitter cubicles, data logging systems, and setpoint checking systems. The design principles for this last group of systems will not be given in this chapter as they are adequately covered elsewhere (see Reference 165 and Chapter VIII of this volume). It should be realized that the design of any one of the other four groups of systems may be nearly as complicated as has been presented thus far for the primary system. In an attempt to limit the length of this chapter, only a portion of the design bases for each of the four groups will be presented here.

a. Dowtherm System

A Dowtherm system usually consists of a surge tank, fill, makeup, and circulating pumps, heater, and interconnecting piping. The surge tank may be pressurized to 100 psig or less by means of inert gas. Because of the low pressures involved, it can be equipped with an armored sight glass and ball float-type level control although dirty, unreadable sight glasses become a problem in time with this type system. The fill and makeup pumps may be low capacity rotary gear type, and the circulating pumps may be centrifugal with mechanical shaft seals. A small Dowtherm cooler may be supplied with the circulating pumps to cool the Dowtherm which cools the seals. All pumps here and elsewhere in the loop should have drip pans with drain connections. The heaters may be of the immersion type. Material for most of the system may be carbon steel, although some special parts such as the heat exchanger tubes may be fabricated from a cupronickel alloy. The equipment must be enclosed, usually in a vapor-tight sheet metal cubicle which incorporates an automatic fire extinguishing system.

b. Water Makeup and Decontamination Systems

The equipment associated with a water preparation, makeup, and initial filling system comprises a storage tank, degassifier, vacuum system, ion exchangers,

a hydrogen addition vessel and the fill and makeup pumps. This system may also include equipment for decontamination, consisting of a chemical mixing tank complete with immersion heaters, pump, and agitator.

The most common water supply for a loop is demineralized plant water, further treated by each loop's own water preparation equipment. Plant demineralized water is usually the secondary emergency supply, except in the solitary case of one MTR loop which has used plant demineralized water as its primary supply for the last several years. In rare instances, raw plant water has been utilized as a secondary supply, being treated in a Barnstead still. The most common water preparation system flow is from the plant demineralized water supply through a degassing device or system, into a storage tank, through two resin beds, through instrumentation for water quality control, through either the fill or makeup pumps, and into the loop at the discharge of the main circulating pumps. Two MTR loops use only deoxygenating columns to treat the plant demineralized water before discharging into the loops.

The rate of degassification should probably be a compromise between the desirable rate of initial filling of the loop, ie, a loop volume including pressurizer per 1/2 hour, and the makeup rate of 0.1 loop volume (including pressurizer) per hour, or a loop volume in 4 to 6 hours. Degassification may be accomplished by a combination of steam or electric immersion heaters in the storage tank and ~4-inch pipe packed with Raschig rings or Stedman packing^[190] or a similar degassification column with the gases being evacuated to the loop vent system by a vacuum pump requiring no longer than ~15 minutes to pull a satisfactory vacuum. If steam heating of the tank water is employed, the steam could also be used to steam flush the loop prior to filling to eliminate as much air as possible. Rather than the degassing column and vacuum pump, which may be the preferable

method, the degassing operation might be accomplished by heating the tank contents to about 300°F and then venting off about 10 percent of the water. Once degassed, the water in the tank is kept degassed or oxygen absorption prevented by bubbling nitrogen gas through the water or protecting it with a helium blanket at a pressure slightly above atmospheric.

The capacity of the storage tank should be about one loop volume including the pressurizer volume. This tank, as well as all other atmospheric tanks in the loop, should have removable lids to prevent foreign materials from falling into it.

At least two resin columns are provided for the fill and makeup water; one containing a deoxygenating resin such as the sulphite type, and one containing a mixed bed demineralizing resin. To avoid too frequent attention by the Phillips' maintenance personnel, the resin capacity of the columns should be designed for 12 to 24 loop fillings. Quality control equipment may include a conductivity cell, flow integrator, and a sampling point for oxygen samples.

The initial fill pump takes a suction from the storage tank. It is ordinarily a centrifugal pump with a capacity of one loop volume including pressurizer in 1/2 hour at a rated pressure of 50 to 100 psig to overcome pressure generated in the loop due to residual heat in the pressurizer after shutdown. For the theory, design, and application of centrifugal pumps, see References 191 and 192. Means must be provided to valve this pump out during normal operation with the makeup pumps.

Usually, one makeup pump and a spare makeup pump are provided for each loop. The normal and spare pumps usually do not incorporate an automatic switching arrangement in case of pump failure. Both pumps are usually of the constant speed, variable stroke, positive displacement type, one with a pneumatic stroke

controller, while the spare may have a manual stroke adjustment. For the theory, design, and application of this type pump, see Reference 193. The makeup pump size is determined by the loop losses from sampling and leakage and is designed to provide higher pressures for hydrostatic testing at up to 2 times design pressure. The pump capacity in gph will usually range between 4 percent of the loop volume including pressurizer for the smaller loops (viz, up to 100 or 150 gal) to 1 percent for the larger loops (viz, 100 or 150 to 350 gal range). The pumps are usually constructed of stainless steel, with nickel-molybdenum-chromium-iron valves, high pressure chevron ring packings, and have no plunger guide bushing. Each pump has double ball check valves on the suction and discharge, backed up in case foreign particles lodge under one. A relief valve is installed in the discharge line to prevent pump damage in operating against an accidentally closed discharge. The loop is prevented from depressurizing by a check valve between the relief valve and the point of injection into the primary loop. Instrumentation should be provided to turn off the makeup pumps on high pressurizer pressure when makeup feeds into the pressurizer and on high loop pressure when the makeup feeds into the loop. Two small lines may also come off the makeup pump discharge, one to fill the high leg of the pressurizer liquid level system to avoid false readings, and one to the main circulating pump vents to backflush bearings when using decontamination chemicals.

Decontamination procedures are given in Chapter IX of this volume. The equipment for a complete decontamination system might include a storage and mixing tank, blending and proportioning equipment for various chemicals used in a decontamination solution, an agitator, a pump, and catch or holdup tanks. The system should be designed for both a feed-and-bleed and a batch-type process. The materials of construction should be carefully selected because of the extreme corrosiveness of some decontamination

solutions. The storage and mixing tank should be equipped with immersion type heaters capable of heating contents to about 220°F and a carefully calibrated liquid level indicating system to control bleed flow rates. The tank should also have an overflow line and a removable lid with a top vent which can be connected to cubicle exhaust to remove toxic fumes. The tank capacity may be one loop volume minimum although the size may be determined by the agitator selection [194, 195]. The charging pump size should be selected for a capacity and head similar to the loop initial fill pump. Close attention should be paid to the bearing material since most decontamination solutions are particularly destructive to the Graphitar type. Pumps used in the past for this service at TRA have usually been of the high pressure, air operated type, but the capacities and materials of construction have not been suitable. From past experience also, the capacity of the holdup or catch tank(s) should be about ten loop volumes minimum to reduce probable levels of activities to tolerable levels, including flushes, and the tanks should be adequately shielded. A sampling system for tank water is a necessity. Connections should be furnished at the bottom of the tanks for draining to one of the available plant drain systems.

c. Purification, Sampling, and Water Chemistry Systems

The purification or cleanup system and the sampling-water chemistry system flow is a small sidestream of primary system water taken from the main circulating pump discharge through a regenerative heat exchanger reducing temperature from design operating value to about half of the design operating temperature, then through a purification heat exchanger further reducing from half design operating temperature to about 100 to 110°F, and finally through the ion exchangers. The regenerative heat exchanger also reheats the purification loop sidestream before returning it to the primary loop. Miscellaneous equipment in

the system may include conductivity cells before and following the ion exchangers, a gamma monitor after the ion exchangers, a shielded dry box or sampling station with valving for taking water samples and adding chemical reagents to the loop water, delayed neutron (D/N) monitor, crud probes, flow elements, flow integrator, etc.

The principal design problems in connection with the purification system are those related to the ion exchangers. Ion exchangers must be designed for three functions: (a) removal of particulate corrosion products; (b) control of pH; and (c) control of the concentration of ionized or colloidal radioisotopes in the coolant to minimize radiation and radiological difficulties.

Appendix IV of Reference 196 presents a method and formula for predicting apparent performance of ion exchange columns for removal of isotopes of trace species of various half-lives and gives such things as linear velocity of the solution through the interstices of the resin bed, resin particle size, bed depth, and the area flow loading. The minimum purification rate should probably be about $1/3$ loop volume/hr with a max of $2/3$ to permit lowering loop water activity quickly after a specimen failure. Flow loadings may range between 1 and 30 gpm/ft² with 3 to 15 being the most probable range. Currently, 6 to 7.5 is the most common range.

The design of ion exchanger size should keep the period of time between changeouts to a practical minimum; if leak and makeup rates are low and draining and refilling for maintenance work kept to a minimum, resin changeouts will be less frequent. The depth of bed and resin particle size, because of experience with pressure drop, bed-packing, segregation, exposed surface area, etc, will most probably be about 3 feet deep with a mean particle size of 16 to 60 mesh. The columns are usually fabricated from 1-inch to 10-inch diameter pipe with pipe caps on

both ends, the schedule as determined from code design.

Assuming an allowable concentration of mixed fission products in the primary water system and a 3-foot line source, using the allowable dose rate given previously, the shielding for the ion exchange column may be calculated. To facilitate column repairs, consideration should be given to the use of lead shot or formed lead brick in preference to poured lead for the shielding, and a monorail and/or hoist arrangement provided for removing the lead cap for maintenance work. For flexibility of operation, multiple ion exchangers should be furnished, half of them to be used for normal operation and half to be on standby.

Careful attention must be given to the charging of new resin and the discharging of spent, radioactive resins. Charging should be able to be accomplished without removing the top shielding. Discharging may be by a basket-cask arrangement, by back flushing resin into shielded disposal casks, or possibly by a combination of both. If back-flushing is used, an air connection should be provided for agitation of resin. On the column inlet and discharge lines, 200-mesh removable cartridge-type screens should be provided to prevent the possibility of resin ever entering into the main loop flow stream. If the ion exchangers are self-shielded and are located outside the main cubicle, the piping to the columns should be shielded. Additional references for ion exchangers design are given in References 197 to 200.

Although gross gamma monitors are replacing delayed neutron monitors on the ATR loops and on an increasing number of ETR loops as primary fission break detection instruments, there are still sufficient D/N monitors employed at TRA to make it worthwhile to outline their principles of operation and design fundamentals.

The delayed neutron monitor is a detection instrument designed to pick up

unnatural sources of neutron emission in the primary coolant. The delayed neutrons accompanying fission of uranium-233, uranium-235, and plutonium-239 fall into five major groups, each of which is characterized by a definite rate of decay. Of the five groups, the 55.6-second half-life delayed neutron group has the longest half-life for disintegration. Therefore, at any time 55.6 seconds after fission, there is only a nominal neutron emission rate from fission products. The delayed neutron monitor design incorporates a minimum time of 60 seconds delay of primary coolant traveling from the exit of the fuel specimen section in the reactor to the entrance so that the D/N counting tubes normally pick up few neutrons, the associated instrumentation being calibrated for a neutron rate accompanying rated fission. Other factors to be considered in the design of the D/N monitor are as follows:

- (1) Use of structural materials having low neutron absorption capacities.

- (2) Maintaining a maximum surface temperature at the counting tubes of about 120°F, using a plant utility coolant.

- (3) Utilizing a minimum amount of material between the neutron source and the D/N counting tubes.

- (4) A net volume of ≈ 1500 ml for the D/N tank, dependent on the reactor, sample, and instrument response.

- (5) Ease of manufacture, assembly, and maintenance.

Briefly, the required calculations for the design of the D/N monitor are as follows:

- (1) Using the velocities of flow through the piping configuration from sample exit to the D/N flow, at a maximum and also minimum expected primary flows, the elapsed times of flow are calculated to ensure being over 60 seconds in each case. It may be found that a reduction in flow through the parallel D/N

monitor circuit will be necessary at the highest primary system flow rate.

- (2) From the tank volume requirement, dimensions may be calculated and code stress calculations made.

- (3) Heat transfer calculations are made using design operating temperature of primary coolant at the delay tank wall and the 120°F maximum imposed at the counting tubes; the cooling water flow can then be calculated and the necessity for an intermediate insulating tube and air gap determined.

The sampling station or dry box should incorporate the following design considerations for operations and personnel safety reasons:

- (1) Adequate shielding; this may include lead glass windows.

- (2) Adequate illumination.

- (3) Ventilation: the air flow should be a minimum of 150 ft/min face velocity, flowing from sampling area into sample box and from box into cubicle radioactive exhaust system.

- (4) Protection from spillage and spread of contaminants outside of box and on personnel; glove box type is preferable.

- (5) Although hydrazine may be added to loop during reactor shutdown periods by means of sample bomb taps in the dry box, it is ineffective when the reactor is at power; and hydrogen must be used. Hydrogen may be added by means of sample bombs; or for convenience, a high pressure hydrogen bottle could be permanently piped to the sample system.

Crud filters may be of the hot or cold crud filter types dependent on the temperature of the water flowing through them. The hot crud filter is more indicative of the flow through the specimen as there is less chance of crud depositing in the

heat exchangers. The filters are located in the purification stream before the gamma monitor to avoid transient readings from crud bursts. They are also located in the stream prior to the ion exchange columns to keep the resin from becoming exhausted prematurely, especially in the older loops where crud may be more of a problem than it is in the new loops. In many loops, the function of the filters is principally to collect crud samples for laboratory analysis. Some filters may employ synthetic filter paper cartridges which can be changed out with the loop at operating conditions; others use a graphite bed which can be changed out only during a reactor shutdown period. The container must be designed as a code pressure vessel (see Figure 30).

NOTE The millipore filter is placed on top of the micro-metallic backing filter, and is held in place by the O-ring.

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Fig. 30 Crud probe.

d. Pressure Relief System

Article 9 of Section III of the ASME Boiler and Pressure Vessel Code applying to the design of pressure relief devices is presently in preparation. Until such time as it is published, the design of loop pressure relief systems shall be governed by Reference 236. Two pressure relief valves or a pressure relief valve and rupture disc in series shall be used with

a pressure buildup alarm device between the two. The pressure relief valves should reseal at a pressure sufficiently high to prevent bulk boiling. Paragraph P-270 of Section I of the ASME Code defines the safety valve capacity as "all the steam that can be generated by the boiler without allowing the pressure to rise above the defined level".

Since the pressurizer vessel contains a heat source, relief must be provided. The basis for sizing the pressurizer relief valve is to relieve the pressure buildup of the primary system due to the addition of fission and gamma heat in the core section. Also, most loop heat exchangers should have protection from overpressure because they could have a heat source in the form of primary water at operating temperature on the tube side with sufficient or no cooling water on the shell side. Each section of the loop that can be completely isolated must have protection from excess pressure. Sample calculations are given in Appendix H, and pertinent references are 201 to 209.

A few miscellaneous relief system design and operating considerations are given below:

(1) Rupture discs usually throw out a piece of metal when they break. This can be dangerous because the piece could lodge in the relief valve port. Some manufacturers make integral units which combine the rupture disc assembly with the relief valve, but are so arranged as to minimize obstruction of flow due to stray fragments of metal from the disc.

(2) If rupture discs are pressurized somewhat over the normal operating point, but not to the yield point, they may fail at some later date at a pressure lower than rated pressure. Also Reference 206 indicates that static and dynamic work hardening may increase the ultimate stress at which the disc will rupture. Corrosion, improper rupture disc material specifications, and experience with rupture disc leaks are secondary factors

which have influenced current Phillips' policy to change out rupture discs at least twice a year. For this reason, the sponsor should furnish about a dozen spare discs for each assembly. To comply with the Code requirement for yearly relief valve tests, Phillips has authorized equipment purchase and installation for a field testing rig for this purpose. These proposed tests and the disc changeout requirement make it desirable that both the relief valve and the rupture disc assembly be installed using flanged joints with flexitallic type gaskets.

(3) Relief valves used in the primary system should be constructed of 300-series stainless steel to reduce corrosive effects. If installed on a Dowtherm system, the standards of the Factory Insurance Association should be followed as set forth in their manual entitled, "Recommended Good Practice for Heat Transfer Mediums in Closed Systems".

(4) During hydrostatic pressure tests, rupture discs should be backed up with duplicate discs; they should be pierced and the relief valve gagged; or they should have a bypass with double block valves and an automatic alarm to warn operating personnel that the bypass is open. All three methods are potentially dangerous; but the last method, with proper administrative controls, appears to offer the least danger and, at the same time, reduces radiation exposure to maintenance personnel during the many hydrostatic tests a loop undergoes annually.

(5) Following are four installation tips:

(a) Avoid sharp bends near installation point which might produce turbulence.

(b) Always install valves vertically.

(c) The discharge of the valves should be in the same direction as the flow to avoid torsional stress on the pipe from back thrust.

(d) Section 4.04 of "Standard Health and Safety Requirements" issued by the Idaho Operations Office of the Atomic Energy Commission^[7] specifically prohibits the painting of relief valves.

e. Loop Drains and Vents

Both the loop drains and the loop vents should be manifolded to headers, and all valves should be valved to flow either to the plant warm drain system or the hot drain system. All drain and venting effluent should have means of being visible to the operator. Drain collection troughs should have hinged, not flanged, plexiglass covers. Safety plugs or protective covers over handles should be provided on those vent valves such as the in-pile tube vents, where, for example, accidental dropping of a lead shot bag might open the valve, thus depressurizing the loop and possibly allowing radioactive material to escape into the reactor building causing an evacuation.

G. DESIGN PHASE SIX: FINAL DETAILS

Following the design of the services and the foregoing auxiliary systems, there may remain the design of such affiliate systems as hydraulic systems and pump oil coolant or inert gas systems. The detailed design of all these systems will necessitate the layout and design of small tubing and pipe, including valves, fittings, hangers and insulation, metal partition design, and the sealing of penetrations into the cubicles. The layout drawing can then be revised and completed, checking such things as door swing, door stops, aisleway widths, headroom, etc.

After this, those calculations which the final revision of the layouts might have appreciably affected can be rechecked. This might include reviewing the piping stress calculations due to local components. The detail drawings can then be completed and vendor specifications written. An integral part of this step is the determination of fabrication and quality

control procedures. The capabilities of pressure equipment, especially for nuclear service, are measured by the general and detailed design, the analysis, structure and soundness of materials, the accuracy and adequacy of fabrication, and the degree in which the foregoing are assured by the extent and competency of quality control and inspection. The fabrication and quality control item of perhaps greatest importance is that relating to welding and radiography; but, of course, straightness and accuracy of linear dimensions, surface finish, cleanliness, code stamp requirement, pressure tests, and photographic and inspection records of all procedures, among others, are most essential. Following the vendor loop equipment specifications, the boxing detail drawings and specifications must be prepared, the greatest consideration being to prevent shipping damage. All drawings, specifications, and calculations should be sent to Phillips for scrutiny; and after approval, the material can be

purchased. Still a part of quality design and engineering is the preparation of field installation and quality control specifications including welding, radiography, cleanliness, insulation, painting, labelling, arrangements to obtain "as-built" measurements, smoke testing cubicles, determining radiation protection effectiveness of shielding by means of a source, and, again, photographing and logging all steps of the installation and check-out. The last steps for the loop designer include the transmittal of the following information to the Phillips' project engineer; please note that this is not intended to be a complete listing of all that may be required: (a) drawing and specification lists; (b) component installation procedure; (c) loop start-up procedures; (d) loop operating manuals, including valve and instrument lists^[2]; (e) material index for all components in contact with loop primary coolant; (f) loop design analysis report; and (g) spare parts lists for all components.

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235. Internal Communication, D. E. Purkey, Phillips Petroleum Company, AED, to F. P. Cromi, Knolls Atomic Power Laboratory, 2/13/62, Machining Mistake Standard NR In-Pile Tube.
236. Internal Communication, L. V. Weber, Phillips Petroleum Company, AED, to H. M. Leppich, IDO, Atomic Energy Commission, 1/28/65, Vessel Design Pressure Versus Relief Valve and Rupture Disc Settings.

CHAPTER VII

LOOP TEST TRAIN AND IN-PILE TUBE HANDLING

A. INTRODUCTION

This chapter has been prepared to give the reader a basic understanding of the methods used to insert and discharge loop experiment test train assemblies and in-pile tubes from the reactors at the TRA. Basic design features of handling casks will also be discussed.

B. FABRICATION OF TEST ASSEMBLIES

A variety of fuel specimens are irradiated in in-pile tubes at the MTR and ETR. These irradiations are carried out under controlled conditions of temperature, flow, and pressure. The majority of these irradiations are conducted under various Naval test programs. These test specimens are designed with certain test objectives in mind. A series of typical nonfueled specimens might be designed, fabricated, and irradiated: (1) To determine the effect of irradiation at reactor temperature on its mechanical properties. To this end, tensile stress, rupture, notch impact, stress fatigue and strain fatigue specimens would be irradiated. (2) To determine the effect of long term irradiations at higher temperatures and neutron exposure on the corrosion resistance properties of the material. These specimens might be further investigated for the effect of initial surface condition and the effect of heat treatment, hydrogen content, and composition effects.

A typical fueled test specimen might be designed, fabricated, and irradiated: (1) To determine the behavior of the particular type of fuel with varying compositions. The fuel elements might be examined to determine the effects of such variables as composition, crystallographic structure, density, grain size and microstructure on dimensional

stability. (2) To determine the irradiated behavior of metallic fuel alloys possessing exceptional mechanical properties at elevated temperatures. Of primary interest might be dimensional stability and corrosion resistance of the alloys under irradiation. (3) to determine the irradiation behavior of a metal base binary fuel alloy containing varying amounts of fuel and the effect of small additions of burnable poisons. (4) To determine the irradiation behavior of several boron bearing materials which might be considered as boron carriers for burnable poison elements.

The test specimens to be irradiated in loop experiment in-pile tubes are usually in the shape of a rod or plate, and are mounted in a holder, which in turn is part of a train. This train may consist of several tiers of specimens depending upon the length of specimens. The general method of suspending the specimens in a train is similar for all tubes. The train is constructed of stainless steel parts, because of the high resistance of this material to corrosion at elevated temperatures and the component lengths are so designed that the specimens to be irradiated are mounted within a core length elevation. The specimens are rigidly fixed within the train to reduce movement; yet the assembly must be so constructed that specimen changeouts can be made remotely in a canal or hot cell with a minimum of difficulty. Usually the train is so constructed and assembled that specimens can be removed by the removal of several clips or pins on the holders. The train is usually constructed with a maximum of four tiers, but in some instances may have as many as seven.

The principal components of a specimen train consist of the specimens, thermocouples if required, specimen cover plate, specimen connector, hanger

assembly, hanger support, thermal barrier, and closure assembly. The closure assembly is so constructed that lifting tools can be attached to it to insert and withdraw the train from the tube and perform other handling functions.

The number of test specimens in a train may vary from a few fuel specimens to 25 or 30 nonfueled specimens. The Naval loop experiment test trains arrive at the TRA completely assembled and ready for insertion into the reactors. The test trains are also sent to the hot cells and water pits from the TRA reactors in an assembled condition. All changeout of test specimens in a train is performed at the ECF water pits and hot cells. A few non-NR experimenters who conduct loop experiments at the TRA perform the changeout functions in the MTR-ETR canals.

During fabrication and upon completion of fabrication of a test specimen, a number of pre-irradiation integrity tests are conducted. These tests are usually conducted at the sponsor's laboratories or at vendor shops.

C. IN-PILE TUBE AND TEST TRAIN INSERTIONS AND DISCHARGES (See Figure 31)

Experiment test trains changeouts or discharges are made for any of a number of reasons. The most common reasons are: (1) The predetermined irradiation life of a specimen has been reached. (2) A fission break has occurred on a fuel specimen. (3) A change in operating parameters has occurred due to a physical change in a specimen. An example of this might be swelling of a plate which could cause changes in train flow distribution, specimen temperatures, and in-pile tube pressure drop. (4) An interim exam of one or more specimens to observe irradiation progress is desired. (5) A reorientation of specimens within the train to better meet flux requirements is desired. (6) An exam of specimens for corrosion observations is desired. (7) A

loop equipment failure has occurred, and repairs must be made. (8) A transfer of specimens to another loop train to better meet test objectives is desired.

In-pile tubes are discharged or changed out when: (1) The tube has reached its approved irradiation damage limit. (2) A failure occurs on the tube which cannot be repaired, and it becomes unsafe to operate the tube. (3) A failure occurs which prevents operation at specified test conditions. (4) The loop experiment program has expired and no future programs are intended for that position in the reactor.

In-pile tubes and test trains are inserted and discharged at the TRA reactors in three different manners: intank through the discharge chute or discharge receiver, over the top of the reactor, or from the side. In some cases, the in-pile tubes are discharged by a combination of two methods. The insertions and discharges are made in accordance with written procedures prepared by the sponsor and the cognizant loop project engineer. The procedures are carefully prepared and written to accomplish the removal or insertion (1) without endangering personnel due to radiation overexposure or improper handling techniques, (2) without endangering or damaging the experiment, in-pile tube, or the reactor, and (3) in a minimum of reactor down time. The procedures for test train changeout are being included in loop operating manuals as these manuals are upgraded.

At the MTR, all three types of insertion and discharge of in-pile tubes are used. Only the in-tank and side methods are used in the discharge of experiments. At the ETR, all the in-pile tubes are inserted and discharged over the top. Loop experiments are inserted and discharged over the top and also in-tank. Since the ETR has no side irradiation facilities, there are no in-pile tube or experiment discharges by this method. At the ATR, all in-pile tube and experiment discharges and insertions will be made over the top of the reactor.

Fig. 31 NR loop experiment train removal at ETR.

A number of preparations must be made to the reactor and loop prior to beginning actual discharge of an in-pile tube. These include reactor shutdown, opening of the reactor vessel, depressurization of the loop, opening of the in-pile tube closure plug, discharge of the experiment, removal of fuel elements from the vicinity of the tube, and installation of a shield plug into the tube. The shield plug is not necessary for tubes that are to be discharged in-tank through the discharge receiver, since tube cuts are usually made under water or sufficiently low in the tank that no hazard is encountered with radiation beams from them.

There is one loop experiment that is not removed from its in-pile tube when discharging the tube. In this case, the tube and experiment are removed as an integral unit. Changeout of this experiment from its in-pile tube is done in the MTR canal. This is also the only in-pile tube which is routinely reinserted into the reactor after having been previously irradiated.

There has been an occasion whereby a beam hole in-pile tube has been discharged with its experiment in place. However, since in-pile tube replacement in these locations are quite infrequent, no such discharges have been made recently.

No special equipment is required for in-pile tube insertions, and available maintenance and reactor support equipment is adequate. The sponsor provides any special casks required for in-pile tube or experiment discharges. A universal cask, horizontally mounted, is provided as support equipment by the MTR for discharge of in-pile tubes and dummy tubes from horizontal beam hole facilities.

In-tank experiment and in-pile tube insertions and discharges are perhaps best described as those which are made by way of the discharge receiver or chute. At the MTR, the KAPL L-42,

WAPD VH-3, WAPD B-4, and GEH B-3 loop experiments are discharged to the canal and returned to the reactor tank in this manner. In the case of the GEH B-3 loop, the experiment and in-pile tube are removed and inserted as an integral unit. Swagelok fittings within the tank permit the disconnection of the in-pile tube from its inlet and outlet piping and consequently a rather simple changeout. The B-3 experiment and tube will occasionally, if radiation levels permit, be reinserted into the tank over the top. Unirradiated in-pile tubes and experiments are of course inserted from the top of the reactor.

Transfer of the WAPD B-4, VH-3, or KAPL L-42 experiments between the canal and the reactor tank is accomplished by means of a discharge can. The purpose of the can is to provide protection of the experiment during the handling operations. Each experiment has its own discharge can. Transfers of the experiments between the MTR canal and the ECF water pits are also made within these cans.

In removal of in-pile tubes by the in-tank method, the tubes are cut into at least two sections. After cutting of the crossarm piping, the tube is cut at some point near the top of the core area. This upper section is then discharged over the top of the reactor. The highly radioactive lower section containing the core region is then discharged to the canal via the discharge receiver. Cuts made on the vertical sections of the tubes usually are made under water and require special hydraulic cutting facilities.

At the ETR, the GEH G-7/M-3 and GEH P7 experiments are discharged and inserted by the in-tank method. No in-pile tubes are discharged in this manner. No experiments or in-pile tubes will be changed out in this manner at the ATR.

The length and diameter of those tube sections to be discharged through a dis-

charge receiver or chute are limited by the size of receiver or chute. At the MTR in-pile tube section, experiments or discharge cans are limited in size to 14 feet at 6-3/8-inch OD, 15 feet at 4-3/4-inch OD, or 15 feet 6-7/8 inches at 3-1/2-inch OD. At the ETR, the limit is 16 feet 1-1/2 inches at 14-3/4-inch OD, and at the ATR, 24 feet 9 inches at 6.0-inch OD.

The top method is the most common method of experiment and in-pile tube discharge at the TRA. There are no experiment changeouts by this method at the MTR. However, with the exception of a previously irradiated GEH B3 in-pile tube, all in-tank, in-pile tubes are inserted in this manner. All in-tank, in-pile tubes, again with the exception of GEH B3, are discharged in this manner. The WAPD VH-3 and WAPD B-4 in-pile tubes were formerly discharged in this manner. However, in the recent past, these tubes have been changed out by the top method, primarily due to the lack of necessary remote hydraulic cutting equipment. This method proved to be a considerably quicker method of removing the tubes without any increase in hazards. The GEH 5-inch split cask has been used to make these discharges.

The top method is the most common method for experiment and in-pile tube changeout at the ETR. All NR loop experiments and in-pile tubes are discharged by use of either the NR I or NR II casks. For a description of a typical experiment discharge using either of these casks, refer to the section describing these casks later in this chapter. Both NR casks are able, without modification, to discharge the present NR in-pile tubes, since these in-pile tube dimensions are within the dimensional limitations of the casks. In those cases where it is necessary to discharge in-pile tubes of larger diameter than the present 3-inch NR tubes, the upper extension and inner cooling coil must be removed. Special clamps and rigging are necessary to effect the

discharge. All loop experiments and in-pile tube discharges at the ATR will be top discharges.

The side method of discharge or insertion refers to experiment and in-pile tubes in the horizontal beam holes at the MTR. At the present time only one loop experiment, KAPL HB-1, is located in a beam hole facility. The experiment insertions and discharges are made by the use of a horizontally mounted cask provided by the sponsor. In-pile tube changeouts are made by the use of the MTR universal cask. The use of the universal cask is limited to tubes no greater than 9-1/2 inches in diameter.

D. TRA HANDLING CASKS

A variety of casks are used at the TRA to effect an in-pile tube and/or experiment changeout. These casks are designed by the sponsoring organization to meet individual requirements. The casks vary in effective shielding length from 4 feet to approximately 7-1/2 feet, in shielding thickness from 5-1/2 inches to 12 inches, and in weight from 3700 to 50,000 lbs. Operating procedures for the use and handling of these casks are prepared by project and sponsoring engineering personnel. See Table VIII for a tabulation of the TRA handling casks.

1. NR Casks I and II (See Figure 32)

The NR I and II casks are presently in use at the MTR-ETR. These casks were designed for inserting and removing experiments from the NR water loops at the ETR and to transport experiments between ECF water pits and ETR. The casks are also used to transport two WAPD and one KAPL loop experiments between the MTR canal and ECF water pits. The casks are not used for direct insertion or removal of experiments at the MTR. The casks are also used for removal of NR in-pile tubes at the ETR. They are used in the vertical position

TABLE VIII

TRA HANDLING CASK DATA

Cask	NR I	NR II	NR III
Weight Empty (lbs)	22,000	27,000	47,500
Length	28 ft, 1-9/32 in.	28 ft, 0-3/4 in.	29 ft, 1-3/4 in.
Outside Diameter	32 in.	32 in.	32 in.
Shielded Height	6 ft, 11-3/8 in.	7 ft, 6-3/4 in.	12 ft, 5-3/4 in.
Shielding Thickness	11 in.	12-1/8 in.	12.1 in.
Experiment Cavity Diameter	4.8 in.	4.9 in.	5.46 in.
Shell Cavity Diameter	10 in.	7-3/4 in.	7.8 in.
Length Above Door	25 ft, 7-1/16 in.	25 ft, 9-13/16 in.	28 ft
Rated Cask Load (lbs) (Winch and Cable)	1,000	1,000	460
Type of Door	Rotating	Rotating	Rotating
Door Thickness (min)	5/8 in. -- 1-1/2 in.	5/8 in. -- 1-1/2 in.	5/8 in. -- 1-1/2 in.
Type of Winch	Lug -- All Model 2000-10 Modified	Lug -- All Model 3000-10 Modified	Lug -- All Model 3000-10 Modified
Drawing No.	G. E. Drawing No. 152F281 NR I and II Cask Data	G. E. Drawing No. 152F281 NR I and II Cask Data	G. E. Drawing No. 262E332B NR III Removal Cask Outline
Used	MTR-ETR-ECF	MTR-ETR-ECF	MTR-ETR-ATR (In design and fabrication)
Cask	ATR Handling Cask	HB-1	GEH 5 in. Cask G7/M3, VH-3, B-4
Weight Empty (lbs)	58,000	8,900	7,200
Length	29 ft, 9 in.	21 ft, 10.3 in.	5 ft, 8 in.
Outside Diameter	33-1/2 in.	24 in.	20 in.
Shielded Height	14 ft	48.1 in.	5 ft, 8 in.
Shielding Thickness	12-1/8 in.	13 in.	5-7/8 in.
Experiment Cavity Diameter	6-3/4 in.	2.5 in.	8-3/4 in.
Shell Cavity Diameter	9 in.	25 in.	8-3/4 in.
Length Above Door	28 ft, 2 in.	35.44 in.	DA
Rated Cask Load (lbs) (Winch and Cable)	500	DA	DA
Type of Door	Sliding	Rotating	No Door
Door Thickness (min)	9 in. -- 12 in.	13.0	DA
Type of Winch	Lug -- All Model 2000-10 Modified	DA	DA
Drawing No.	O. G. Kelley Drawing No. 63-8955, NR Loop Handling Cask Assay	G. E. HB-1 Removal Cask 124F446	HAPO H-3-9373 5 in. Lead Cask for Discharge of 6 x 9 Facility
Used	ATR (In design and fabrication)	MTR-ECF	MTR-ETR
Cask	GEH 5 in. Cask P7	ATR Gas Cooled Loop Tube Removal Cask	ATR Gas Cooled Loop Specimen Handling Cask
Weight Empty (lbs)	3,700	50,000	60,000
Length	4 ft, 2-3/4 in.	22 ft, 10 in.	30 ft, 6 in.
Outside Diameter	15-1/2 in.	32.2 in.	2 ft, 10 in.
Shielded Length	4 ft, 2-3/4 in.	22 ft, 10 in.	23 ft, 6 in.
Shielding Thickness	5-1/2 in.	8.5 in.	Bottom: 5 ft, 3 in. -- 10 in. Center: 3 ft, 10 in., 10 in. -- 2-1/2 in. Top: 15 ft, 3 in. -- 2-1/2 in.

TABLE VIII (Cont.)

TRA HANDLING CASK DATA

Experiment Cavity Diameter	4-1/2 in.	10.5 ft, 13.5 in.	3 Tube Turret 5 in. diameter each turret
Shell Cavity Diameter	4-1/2 in.	10.5 ft, 13.5 in.	-----
Length Above Door	DA	22 ft, 10 in.	23 ft, 6 in.
Rated Cask Load (lbs) (Winch and Cable)	DA	DA	1,000
Type of Door	No Door	Sliding	Rolling
Door Thickness	DA	6 in.	9 in.
Type of Winch	DA	DA	Internal -- Electric Motor Driven
Drawing No.	H-3-9023 5 in. Lead Cask for Discharge of 3 x 3 Facility	E-4012041-438 ETR Tube Removal Cask-Body Details	ATR-1075-MTR-670-MS-3117 Gas Cooled Loop Specimen Handling Cask Sectional Elevation
Used	MTR-ETR	ATR	ATR (In design and fabrication)

when removing, inserting, or loading in-pile tube or experiments. The casks are placed in a nearly horizontal position when transporting experiments to ECF.

These casks are only used on the NRTS site and cannot be used for off-site shipments. A low-boy trailer is used to transport the casks between the TRA and the ECF water pits.

The casks are basically hollow lead cylinders equipped with rotating leak-tight doors at the lower end and light aluminum superstructures at the upper end. Hand operated winches are provided to effect tube or experiment changeout. A removable standing platform, interchangeable between the casks, is provided for operation of the winch. When not in use the casks are stored on a pallet or on the low-boy trailer. Water fill and vent lines are provided on each cask. To assure adequate cooling of the experiment during shipment, all fueled experiments are shipped wet. A removable cooling coil in each cask and a gasoline engine driven pump are provided for forced convection cooling of an experiment should it be required.

A third cask, NR III, is presently being designed for use at the ATR, although it can also be used at the MTR-ETR if

required. This cask is similar to the NR I and NR II casks. However, it will have increased shielding length and be considerably heavier.

A brief description of the use of these casks in removing an NR experiment at the ETR follows. The reactor is shut down, the loop depressurized and closure plug loosened. An experiment removal tool is then attached to the experiment. The reactor working platform is then removed, and the spool piece tank attached to the reactor flange. Water level is next raised to within a few inches of the top, and shielding blocks II, IV, and Vb are positioned. The removal sleeve is then put in place to provide a leak-tight water passage between the cask and the experiment. A large rubber gasket is next put in place on the removal sleeve flange to provide a seal between it and the cask. The cask is then transferred to the vicinity of the shielding blocks and lowered to within a few feet of block Vb. Sufficient cable is then payed out through the cask so that a connection can be made joining the bayonet lifting mechanism to the bayonet receiver in the experiment lifting tool within the removal sleeve. The slack is then taken on the cable as the cask is lowered into place. The cask is then filled with water, and the experiment raised

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Fig. 32 NR cask and shielding blo

into the cask. The door is then closed and the experiment transferred to the low-boy trailer or canal as desired.

2. ATR Casks

Three of the four casks to be used at the ATR are, at this writing, still in the design and initial fabrication stage. The casks to be used at the ATR are the NR III cask, the ATR Handling Cask, the ATR Gas Cooled Loop Specimen Handling Cask, and the ATR Gas Cooled Loop Tube Removal Cask. The later cask is available on the NRTS site. The NR III cask is similar in construction and operation to the NR I and II casks and, therefore, will not be discussed further. It will be used to transfer NR experiments from the ATR to ECF.

The ATR Gas Cooled Loop Tube Removal Cask was formerly used for GE-ANP in-pile tube removal at the ETR. The cask assembly consists of an open ended, hollow lead cylinder. The cask is provided with a one-ton hoist mounted on a drawbar. The ATR 30-ton crane hook attaches to the cask drawbar.

The ATR Gas Cooled Loop Specimen Cask will be used to transfer experiments in an atmosphere of helium from the in-pile tube to the loop transfer facility. The cask is designed to remove specimen afterheat for several days after reactor shutdown, before discharging the specimen into the transfer facility. To accomplish this cooling, the cask is equipped with a helium blower and a water-to-gas heat exchanger. The gas cooling conditions which are to be maintained while transferring an experiment are extremely rigid; therefore, the cask has several seal assemblies not found on the standard NR casks.

All operations of the cask will be done with the cask in an upright position. The cask is designed with a turret which is capable of positioning any of three tubes or magazines within the cask

over a single unloading port. The unloading port consists of a manually operated shielding door and gas-tight ball valves. All major operations of the cask are performed electrically. A master control panel will be mounted on the cask to control and monitor all cask functions.

The ATR Handling Cask is designed to transfer ATR NR loop experiments from the in-pile tubes to a canal transfer station. The cask will have the same general features as the NR casks. However, since the operation of the cask at the transfer station will not require submersion of the cask, the winch and door operating mechanism are located near the bottom of the cask. The cask is equipped with a considerably heavier door than the NR casks. Because of the increased weight, the door is opened by a sliding motion rather than a rotating motion. The door is manually operated. The cask shielded length will be considerably greater than that of the NR casks.

3. GEH Casks

The GEH casks are in use at TRA for removal of irradiated in-pile tubes. Both casks are similarly constructed and referred to as GEH 5-inch split casks, since they are constructed in vertical half-cylindrical sections which are bolted together. Both casks have between five and six inches of lead shielding.

Reference to Table VIII will indicate the pertinent dimensions of each cask. These casks are used for discharge of the GEH in-pile tube at the ETR and have been used for discharge of the in-tank NR in-pile tubes at the MTR.

4. KAPL HB-1 Cask

The KAPL HB-1 cask is used to remove and insert test trains into the HB-1 in-pile tube at the MTR and to transport test trains between the MTR and ECF. A tilted transport skid is

provided to serve as a support cradle when transporting the cask. Auxiliary equipment used with the cask to effect a removal or insertion consists of a shielding boot and dolly and a cask lifting frame equipped with an elevating and transverse mechanism to properly align the cask with the in-pile tube (see Figure 33). Special removal tools are required to effect the insertion or discharge. The removal cask and shielding boot are stored in the plug storage area when not in use.

5. Universal Coffin

The universal coffin is used for removal and insertion of in-pile and dummy flow tubes from horizontal beam holes at the MTR. It is made up of three sections and has a total length of 18 feet.

The first section is 33 inches in diameter and 4 feet 5 inches long. It is equipped with the coffin mounting flange and has a radiation door at its for-

ward end. The middle section has a conical shape and slopes from a diameter of 33 inches to a diameter of 20 inches at the back side. This section is 6 feet long. The third section has flanges at each end and is equipped with the driving mechanism for operating the radiation door in the beam hole liner. To remove an in-pile tube, the piping to the tube is cut, and a cable is attached to the tube. The universal coffin is then maneuvered into position flush with the cave wall and bolted to it. The cable is threaded through the cask and attached to a yoke, and the tube is manually withdrawn. The facility shield door and cask front door are then closed. The coffin is then transferred to the plug storage area where the tube is stored until final disposition.

Reference procedures for insertion and removal of major experiment samples and in-pile tubes are given in the list of recommended reading at the end of this chapter.

Fig. 33 HB-1 removal cask alignment and shielding facility.

RECOMMENDED READING

1. KTO-P-302 KAPL L-42 Test Assembly Insertion Procedure
Revision 5 September 9, 1964
2. KTO-P-224 HB-1 Discharge Cask Handling Procedure
Revision 4 January 14, 1963
3. NR Cask Operating Manual January 11, 1965
4. KTO-P-222 KAPL HB-1 Irradiated Test Assembly Removal Procedure
Revision 3 May 29, 1962
5. KTO-P-227 KAPL HB-1 Irradiated Test Assembly Insertion Procedure
May 23, 1962
6. KTO-P-330 KAPL HB-1 Irradiated Flow Tube Removal Procedure
February 13, 1962
7. KTO-P-387 KAPL HB-1 Irradiated Flow Tube Insertion Procedure
July 25, 1962
8. KTO-P-331 KAPL HB-1 Unirradiated Flow Tube Insertion Procedure
9. KTO-P-221 KAPL HB-1 Unirradiated Test Assembly Insertion Procedure
Revision 2 January 23, 1963
10. KTO-P-301 KAPL L-42 Test Assembly Removal Procedure
Revision 6 September 9, 1964
11. KTO-P-445 Installation of L-42 In-Pile Tube and Dummy Shim Liner
November 17, 1964
12. KTO-P-446 Removal of the L-42 In-Pile Tube
November 5, 1964
13. Operating Manual for the Division of Reactor Development Pressurized Water
In-Pile Irradiation Test Loop in the Advanced Test Reactor, Section 3.3, In-Pile
Handling Procedure
14. ATR Specification for NR Loop Handling Cask dated April 24, 1963
15. GEH P7 In-Pile Tube Removal Procedure dated March 23, 1962
16. BAPL VH-3 In-Pile Tube Insertion Procedure dated 1-22-64
17. VH-3 In-Pile Tube Removal and Insertion Procedure 4-16-64
18. WAPD B-4 In-Pile Tube Non-Destructive Removal Procedure dated 6-10-63
19. WAPD B-4 In-Pile Tube Removal and Insertion Procedure dated 5-23-63 Rev.1
20. GEH P7 Experiment Removal -- Insertion Procedure Revised January 13, 1964

21. G7/M3 Experiment Insertion-Removal Procedure dated September 20, 1963
22. BAPL B-4-12 Test Handling Procedure WAPD CM(II)-1562 dated 6-11-63
23. BAPL VH-3 Test Assembly Handling Procedures MTR WAPD-CM(II)-438 April 19, 1962

CHAPTER VIII

INSTRUMENTATION

A. INTRODUCTION

Volume 1, Chapter VI of Fundamentals in the Operation of Nuclear Test Reactors describes basic measuring and control circuits used in the instrumentation of nuclear test reactors. This chapter will describe the measurement and control instrumentation used on experimental facilities.

Generally speaking, experimental systems are designed to utilize the standard, commercially available types of instruments. Parameters which must be considered are temperature, flow, pressure, radiation level, liquid level, and such chemical factors as electrical conductivity and pH. Thermocouples are used for most temperature measurements; however, resistance bulbs are used in some locations where greater accuracy is required and where the temperature range and available space will permit. Pneumatic transmission systems are used for the transmission of differential pressure and pressure information and for the operation of related control elements in the system. Electronic and electrical systems can provide much faster response times than can pneumatic systems and are used for control of the more critical phases of experiment operation. Radiation instruments are of two types: area monitors and fission break monitors.

Most experimental systems are designed to hold coolant parameters constant during the test. All parameters except temperature have constant loads and are relatively easy to control. Temperature is affected by reactor neutron flux and is a slowly varying parameter.

Several control loops will be described in this chapter. The example controls chosen are in operation and represent "typical" systems. Control systems

are not uniform, however, since requirements of the individual systems vary widely.

Alarm and reactor control circuit philosophy is standardized for experiments in both reactors, though some variations in detail are incorporated to fit individual experiment needs. Since instrument setpoint setting is an important safety requirement on each experiment for each shutdown, setpoint setting systems have been developed. These are designed to minimize the time needed for setting setpoints without compromising maximum reliability and accuracy.

B. BASIC TYPES OF INSTRUMENTS

1. Temperature

For experimental process measurements and for fuel specimen temperatures, thermocouples are the standard measuring device. Resistance bulbs have been used in some cases where high precision is necessary.

(a) Thermocouples have a tremendous potential in nuclear work. In addition to small size and high reliability, they are relatively unaffected by radiation, have acceptable accuracy, offer a great deal of flexibility in temperature ranges, and are relatively unaffected by pressure, corrosion, and other environmental hazards. The accuracy of thermocouple systems varies from plus or minus 13°F for Chromel-Alumel at 1200°F to plus or minus 5°F for platinum-rhodium at 2200°F. This is not sufficient accuracy for use in differential temperature measurements of less than 30°F. Generally speaking, however, the thermocouple is the preferred temperature measuring device because of its simplicity and freedom from trouble.

Iron-Constantan thermocouples are widely used in water loops where they are not exposed to the water. Installed in protective thermocouple wells, they measure water temperatures throughout the process. Experience has shown that they do not stand up well under irradiation. They have a strong EMF (electromotive force) signal at relatively low temperatures (below 600°F) and are relatively inexpensive.

Chromal-Alumel thermocouples are highly resistant to corrosion, quite strong, and cover a wide range of temperatures (up to 1200°F) with fair accuracy. They are difficult to attach to any metal and are relatively inexpensive.

Platinum-rhodium thermocouples are accurate, reliable, and easily attached to other metals. Their cost is extremely high. They are used chiefly for high temperature work in highly corrosive environments.

Stainless steel clad, magnesium oxide insulated thermocouples are used extensively in nuclear experimental work. They are corrosion resistant, very strong, compact, and reliable even under vibration. They are easily led through high pressure seals.

One trouble area for this type of thermocouple is the connection of these thermocouples to commercial leadwire. These connections are notoriously unreliable because of the extremely fine thermocouple wire (usually 0.010 inch in diameter), its proximity to the stainless steel sheathing, and the tendency of the insulation to break up into a powder and flow out of the end of the tubing.

Glass insulated thermocouple wire is very useful for normal process work. In high radiation fields, however, glass insulation becomes highly radioactive and extremely brittle. Serious contamination problems can, therefore, result during post-irradiation handling and examination unless extreme care is exercised.

In most process work, thermocouple wells are used to protect the thermocouple from the process stream. The wells increase the thermal inertia, slowing down response time of the system. For most process use external to the reactor, the advantages of thermocouple wells outweigh the disadvantages.

Thermocouples are more accurate and have a faster response time if grounded at the hot junction. However, if two thermocouples are to be used to measure differential temperature, they must not be grounded. Ground loops make them completely unreliable. The use of grounded thermocouples for ΔT measurement is a common thermocouple application error.

Sample thermocouples are used to measure fuel element center temperature, cladding temperature, and coolant temperature. When the samples are enclosed in capsules (usually 1-1/8 inches in diameter and less than 3 feet long), the thermocouple leads are handled as described below.

The capsules usually use clad thermocouples. These pass from the capsule to the lead tube through a minimum of two pressure-tight seals. The thermocouple cladding is welded or silver soldered at each seal to prevent diffusion of fission products from capsule to lead.

Above the reactor core, the clad thermocouples are joined to standard leadwire. From the capsule in the reactor core to a sealed junction box outside the reactor tank, the thermocouple leads are enclosed in sealed stainless steel tubing. This tubing protects the thermocouple leads from vibration, corrosion and mechanical stresses, and simplifies the seal where the thermocouple leads pass through a reactor tank access flange.

Samples irradiated in loops are handled in a similar manner. Two seals are usually needed: one where the leads exit from the loop and one where the leads exit from the reactor. In some

cases, it is possible to eliminate one of the seals by keeping the leads inside the loop closure until it exits from the reactor. Seals of this type impose extremely tough design problems. High temperatures, high pressures, corrosion problems, and access problems exist, with the major problem being the extremely limited space allowed for such seals. Still, the success or failure of the experiment depends on these seals.

With all possible precautions, the chance of thermocouple failure is too great on primary process temperatures. These should have standby thermocouples run into the recorder so the standby thermocouple can be put in service with a minimum of delay without entering the area where the loop operating equipment is located.

Normally, the thermocouples are brought out to an accessible area and connected to standard thermocouple leadwires. These run to the instrument console in grounded conduit. No power wiring is allowed in the thermocouple conduit.

The connection from thermocouple to leadwire should be an easily disconnected unit. Plug-in panels, terminal boards, and various plug connectors are used.

Each thermocouple leadwire must have a distinctive number, which appears on all wire ends, terminals, and junctions. These numbers must appear on all blueprints which show the wiring.

(b) The preferred temperature sensing instrument is a null-balance, potentiometer-type recorder. This may be either single point or multipoint. Response speed and chart speed are dependent upon the specific application.

(c) Resistance bulbs are much more accurate than thermocouples. Accuracies of a fraction of a degree are possible. They are usually used for coolant temperature measurement, especially where differential temperature is critical. They

are ideal instruments within their range, which is constantly being raised. At present, resistance bulbs are available which will operate accurately to approximately 1200°F.

Standard resistance bulb assemblies are relatively bulky (1/2 -inch diameter) but can be obtained as small as 1/4-inch in diameter at the present time. Resistance bulbs are usually enclosed in wells for mechanical protection and for protection against corrosive liquids.

The sensing element of the resistance bulb consists of a length of very small diameter platinum wire. The resistance of the wire increases linearly with increase in temperature and is measured by a null-balance recorder using a Wheatstone bridge circuit.

2. Pressure, Flow, and Differential Pressure

Experimental loop process pressures, flows, and differential pressures are usually measured by pneumatic transmitters. These are located in an accessible area as close as possible to the sensing point. The transmitters send a 3 to 15 psig signal to the console, where recorders convert it to recorder parameters.

(a) Several firm rules cover the installation and use of these instruments.

(1) To prevent contamination of the experiment console and instrumentation, they must be used to sense pressures on any possibly contaminated fluid.

(2) They must be installed in a negative pressure area. This is usually a metal enclosure connected to cubicle exhaust. This minimizes air contamination of occupied areas if an instrument fails, leaks, or requires venting.

(3) They must be manifolded so they can be checked out, repaired, or recalibrated while the loop is operating,

with minimum hazard to the instrument man.

(4) They must be easily and quickly removed for major repairs.

(5) Sensing lines should be laid out so as to minimize the possibility of entrapped air in water lines or water in air lines.

(6) The greatest weakness of pneumatic systems is the time delay between the time a process variable changes and the recorder at the console senses the change. Loss-of-flow accidents may be much more serious than necessary just because of this time lag. Use of strategically placed low capacity instruments, with properly sized pneumatic and process tubing connections, can minimize this problem.

(b) Controllers at the console usually send 3 to 15 psig air signals to control valves and other control media in order to automatically control process variables. The controllers are usually miniature strip chart instruments because console space is at a premium. If the control valve is located a long distance from the controller, time lag again becomes a problem. Large valves usually need a valve positioner in order to speed up control valve response.

(c) Flow measuring systems should be designed so that the normal operating point is ≈ 80 percent on the recorder chart. The flow is recorded on a square root chart, which amplifies the upper part. Therefore, 70 percent of maximum flow reads out approximately midway on the recorder paper, and 50 percent of maximum flow reads ≈ 25 percent from the lower end of the chart paper. If normal flow is less than 50 percent of maximum, accurate setting of low flow setpoints is very difficult and unreliable.

(d) Differential pressure measurements for flow and liquid level are accurate and reliable only if fluid conditions

are known. Temperature and pressure must be considered on every reading in order to avoid gross errors in interpretation. Figures 34 and 35 show the temperature dependance of typical high temperature flow and liquid level systems.

The ATR water loops will have temperature compensated flow recorders. However, since the compensated flow system consists of three milliamp computer units in addition to the two units (transmitter and recorder) needed for uncompensated flow, it may prove to be an operating and maintenance problem rather than an advantage. On gas or air loops, both pressure and temperature corrections must be made to ΔP across an orifice in order to read out flow in standard values. On the ATR helium loop, gas temperature, pressure, and ΔP across a Venturi tube are fed into a computing device as milliamp signals, the output to the recorder being proportional to pounds of helium per hour.

(e) Pneumatic transmission systems have a very convenient feature for adding electrically operated alarms, automatic controls, and reactor controls. A pneumatically operated pressure switch is located at the console, sensing the pneumatic signal from the transmitter. As many switches as needed can be installed on any one line. Because of the simplicity, accuracy, reliability, and easy setpoint checking characteristics of these instruments, they are widely used for initiating various electrical controls, including reactor power reductions.

(f) All tubing must be adequately supported and protected from physical damage. Bundles of plastic or copper tubing can be used in many applications, minimizing installation costs. Plastic tubing can be run in a square duct, where temperature and pressures allow its use. Individual tubing lines should be run on Unistrut or similar support with bulkhead fittings used at entrances into transmitter cubicles and consoles. Each piece of tubing should be identified by number at each con-

Fig. 34 Flow calibration.

Fig. 35 Pressurizer liquid level.

necter for facilitating tracing and trouble-shooting. These numbers must appear on all blueprints which show the tubing.

3. Setpoint Panel

Setpoint setting and checking on all experiment annunciator, automatic control, and reactor power reduction circuits are necessary safety items. Setpoints are normally set on all experiments on each reactor shutdown, and all control circuits are checked for proper performance.

With the large number of experiments installed at TRA, aids to setpoint checking must be designed into the instrumentation. The recommendations listed below should be incorporated into the alarm and control circuitry on all major engineered experiments.

(a) All pressure, flow, and level parameters should be measured by a transmitter which transmits a signal proportional to the range of the measurement.

(b) All temperatures should be sensed by thermocouples or resistance bulbs with measurement on potentiometer type instruments.

(c) All contacts for power reduction or annunciation from potentiometer type instruments should be such that the actuation can occur at only one point regardless of where the trip point is set.

(d) When initiating reactor power reductions or local annunciators on any parameter which is measured by means of a pneumatic type transmitter, 0 to 15 lb adjustable range, adjustable differential pressure switches with visual setpoint indication must be used.

(e) Where two or more pressure switches sense the same signal, they can be manifolded on the pressure switch side of the solenoid valve.

(f) A pressure switch will be provided on the air test manifold so that alarms

will be sounded at the experiment panel and reactor control room if pressure builds up in the header during normal operation, indicating a leaking solenoid valve.

(g) A contact from the alarm testing "W" switch will be hooked up so as to alarm a separate annunciator in the reactor control room whenever the switch is in the "test" position. The control room annunciator will be common to all experiments.

Two setpoint setting systems have been developed at TRA. The first was designed specifically for use during periods of experiment and reactor shutdown. When setpoint checking and adjustment with the reactor and experiment at power became necessary, the second system was developed. This second system, used in conjunction with test blocks instead of "W" switches, is standard for ATR and is used on several recently remodeled MTR and ETR loops.

The standard pneumatic setpoint system consists of a multicontactor switch which when turned to the "test" position energizes all solenoid valves connecting all pressure switches to a test manifold system consisting of a pressure gauge and precision pressure regulator. Separate switches allow the operator to electrically invalidate all annunciator and control functions except the one to be tested, thus clearing the annunciator panel except for the one circuit being checked. This system is more convenient for setting setpoints during reactor and loop shutdown but does not permit setpoint adjustment during reactor operation. Figures 36, 37, and 38 illustrate the basic engineering features of this system.

In the second system, the electrical interlocks controlling the solenoid valves must be designed so that, at any one time, the set of pressure switches from only one transmitter may be disconnected from the transmitter and connected to the setpoint setting gauge and regulator. A test block interrupts the circuit from pressure

Fig. 36 Setpoint panel elevation.

CONTACT ACTUATION C-1,2,3, closes on abnormal condition C-4 & 5 opens on abnormal condition All "W" switch contacts shown in the operate condition

Fig. 38 Electrical schematic, setpoint panel.

switch to reactor power reduction relays, at the same time leaving the experiment control panel annunciators and controls connected to the pressure switch. This permits checking of the setpoints from one transmitter only without invalidating the remaining annunciator and control circuits.

4. Electrical Transmission

Electrical and electronic transmission systems have much faster response time than pneumatic transmitters. This is extremely important if reactor power reductions or other controls must be initiated when a parameter departs from safe operating conditions.

Great advances have recently been made in the accuracy, reliability, and reproducibility of these instruments. They are being used to a much greater extent on experiment instrumentation and are giving excellent service.

Most major instrument manufacturers now have a complete line of milliamp instruments available. Transducers for changing differential pressure, pressure,

NOTES

- 1 Pressure Switch on manifold to give alarm if manifold pressure builds up during normal operation, indicating solenoid valve leak.
- 2 Power to solenoids switched through common "W" switch.
3. In normal de-energized position, Solenoid valves pass air from process line to pressure switch.

Fig. 37 Tubing schematic, setpoint panel.

temperature, and many other variables to milliamp signals are now available. The current output is a stable signal, linearly proportional to the input. Alarm modules, recorders, and indicators are available for readout. The signal is easily run to several sensing devices in series, and is also ideal for input into data loggers.

Many commonly used computer modules are also available with milliamp inputs and outputs.

The several basic measuring systems are described in Chapter VI, "Basic Instrumentation" of Fundamentals in the Operation of Nuclear Test Reactors, Volume 1.

5. Radiation Instruments

Radiation instruments can be separated into two categories: area radiation monitors and fission break monitors. The area radiation monitors are used to survey points in the experimental system where high radiation fields occur on experiment failure. Specific points would be high pressure water loop ion exchange columns, pressurizers, delay tanks, and heat exchangers. Gas loops monitor exit valves, filters, and similar spots.

Many fuel plate tests now include plates with deliberate defects, and many tests are designed to continue operation after fission product evolution from plate failures has started. These tests do not require and, in some cases, are restricted by the high sensitivity fission break monitors described above.

Loop activity is checked by gross gamma monitors and analyzed by radiochemical means, as described in Chapter 11.

Fission break monitors may be either delayed neutron monitors or iodine monitors. Theory of operation, uses, and limitations of these two instruments are described in Volume 1, Chapter IV, paragraph C.3 of Fundamentals in the Operation of Nuclear Test Reactors.

C. EXAMPLES OF EXPERIMENT CONTROLS

1. Water Loop Surge Tank Level

The surge tank on a high pressure water loop has two primary functions: to compensate for changes in loop water density and to control loop pressure. In addition, it provides some reserve water to replace leakage.

The loop surge tank is described in more detail in Chapter XI, Volume 4, of Fundamentals in the Operations of Nuclear Test Reactors. Its relationship to the loop is shown schematically on Figure 42.

Surge tank liquid level is measured by a pneumatic transmitter monitoring the differential pressure between a pressure tap in the lower part of the surge tank and a pressure tap in the top of the tank. The transmitter must be located at a level slightly below the lower pressure tap.

The pressure tap at the top of the tank, called the reference leg, is designed to be full of cool (100°F) water at all times. It remains full because steam from the top of the pressurizer constantly condenses into this leg. Most loops also have a line which can be used to purge the reference leg from the makeup pumps as needed.

The pressure tap at the lower part of the tank, called the datum leg, has a head proportional to the liquid level in the tank. Differential pressure between these two legs is inversely proportional to both pressurizer liquid level and water density (proportional to water temperature).

The signal supplied to the receiver at the console is proportional to inches of water differential pressure and usually is recorded as percentage of maximum differential pressure. This reading must be converted to liquid level by nomographs which correct for water temperature and pressure. See Figure 38 for a calibration curve.

All surge tank liquid level systems must have a range which will indicate liquid level whether the surge tank water is cold or at 650°F. Therefore, the tank can be liquid full at 650°F before the recorder reaches full scale.

Usually, three pressure switches and a recorder are attached to the pneumatic transmitter at the console. The recorder has no control function. The three pressure switches perform the following jobs:

(a) The highest differential pressure represents lowest surge tank level. It actuates local and reactor control room annunciators and on some loops turns off the surge tank heaters.

(b) The second highest differential pressure turns on one makeup pump at the lowest safe surge tank level. It resets on a higher surge tank level, shutting off the pump.

(c) The third highest differential pressure turns on the other makeup pump. It resets at a higher surge tank level, shutting off the pump.

For example, assume the surge tank is operating at normal level. No annunciator is sounding and both makeup pumps are off. The surge tank liquid level drops until the differential pressure (c) is reached. One makeup pump turns on. If it can handle the leak, surge tank level will increase to a preset value which will shut off the pump. If one pump cannot handle the leakage, liquid level continues dropping until the differential pressure (b) is reached. The other makeup pump turns on. If both can handle the leakage, level will build up until both pumps turn off automatically. If both cannot handle the leakage, differential pressure (a) is reached, calling the situation to the attention of the reactor control room and the local operator.

A flow sheet diagram of the surge tank and the level controls is shown in Figure 42.

Two malfunctions can render this system inoperative: losing the water in the reference leg and leaving the instrument bypass valve open. In either case, usually at start-up, the instrument indicates a full pressurizer. If the surge tank drain is opened in order to establish normal liquid level after water has been heated above saturation, the surge tank can be drained with the liquid level instrument still indicating a full surge tank. This may result in overheating of the heater vessel, meltdown of the heater elements, and further damage to the operating system.

2. Water Loop Temperature Control

Bulk water temperature is controlled by two separate processes on most loops. Nuclear heat is removed from the reactor exit water by flowing a portion of the loop water through a heat exchanger. Electrical heat is added to the reactor inlet water to compensate for system losses. Both systems are needed because heat generation varies with test reactor power and heat losses vary with loop bulk water temperatures.

To avoid thermal cycling, the loops may operate at test temperatures even during reactor shutdown periods. Electrical heat must then be added to compensate for insulation losses. Increased nuclear heating in the loops as reactor power increases compensates for heat losses. If sufficient nuclear heat is generated, some must be dissipated through heat exchangers.

The typical loop temperature control system will have both line heaters and heat exchangers. Both will be operated from the same pneumatic controllers. The system flow sheet is shown on Figure 42.

The loop bulk water temperature to be controlled is usually located at the inlet to the reactor although it may be at any point in the system. Temperature is sensed and recorded by a console mounted null-balance recorder with pneumatic con-

trol units. The recorder will have three alarm settings: a low temperature alarm, a high temperature alarm, and a high-high temperature reactor power reduction. Control units will include both proportional band and reset. Rate is usually not needed because of long thermal lags but may be incorporated if desired.

The temperature control valve is a three-way valve with valve positioner. Zero pneumatic signal to the valve routes all water through the heat exchanger, the fail-safe condition. Nine pounds pneumatic signal to the valve routes all water around the heat exchanger. The line heaters are clamp-on type resistance heaters operated on three-phase, 440-volt, ac power. They are connected so that three individual banks are tied in parallel, with a starting relay for each bank. Each bank presents a balanced load on the three-phase line. The operator may select any one, two, or all three with a selector switch. The heaters so selected will turn on when the pneumatic controller output actuates a pressure switch set at ≈ 14 psig and turns off when the pressure switch resets at ≈ 13 psig.

The system will operate for all reactor powers and for all loop bulk water temperatures. Controller proportional band and reset must be set in the field to eliminate excessive cycling.

Assume the desired bulk water temperature is 350°F and that the loop is cold with the reactor at zero power. The setpoint on the controller is set at 350°F , and all three line heaters selected for operation. The pneumatic controller output will be 15 psig, and all three heaters will turn on. They will stay on until the bulk water temperature approaches 350°F . At this point, controller output will decrease, shutting off the heaters. Thermal lag will allow the temperature to drift upward slightly, then drop, decreasing controller output and turning on the heaters again.

All water bypasses the heat exchanger during this time, since the pneumatic controller output exceeds 9 psig.

If finer control is desired, one or two heaters may be turned off. This will lengthen the time cycle of heaters being turned off and on, smoothing out the temperature cycling.

When the reactor goes to power, adding nuclear heat to the loop water, the bulk water temperature will continue to rise, dropping controller output and turning off the line heaters. If the bulk water temperature rises sufficiently, controller output will drop to less than 9 psig. This will reset the three-way temperature control valve, allowing some water to pass through the heat exchanger. This will level out so that nuclear heat minus loop heat losses is dissipated through the heat exchangers.

This system is dependent on proper sizing of line heaters, control valve, and heat exchanger. If any component is oversized, loop bulk water temperature tends to cycle excessively. Attainment of specified operating conditions may be impossible if any component is undersized. Oversized control valves and heat exchangers needlessly complicate temperature control on many TRA loops.

3. Water Loop Pump Control

On high pressure water loops, from two to four primary pumps are usually arranged in series depending on flow and head requirements. One pump is considered an installed spare, but all pumps are usually operated, one on diesel power and the remaining pumps on commercial power.

As an absolute minimum, each pump must have a start button (CSM1), a stop button (CSM1), and an ammeter (AM-SW) on the control panel and a breaker with thermal overloads (42D and 42C) located

in the loop breaker panel. See Figure 39 for the identity of the items mentioned above.

Interlocks are usually incorporated to prevent the starting of a pump without cooling water to the motor stator (FS 15). Some loop interlocks will shut down an operating pump on loss of cooling water (FS 15).

High bearing temperature and high outlet motor stator cooling water temperature may actuate local annunciators (not shown in Figure 42).

Instantaneous shutdown of the pump may be actuated by current sensitive relays (50-A, 51-A, 50-C, and 51-C).

Low loop pressure (PS-26) may prevent starting a pump. On other loops, low loop pressure may shutdown the running pumps (PS-26). On all loops an alarm is initiated if low pressure conditions occur after the pump starts (63-PS).

If all features mentioned above are incorporated into a pump control circuit, the circuit is quite complicated; and it is possible to have an emergency condition where all pumps stop because of a relatively minor outage on a secondary system. In this case, one more feature is added to the pump circuits; a run-to-destruction feature (SSD) which will keep one pump running despite loss of cooling water flow, loss of loop pressure, instantaneous overcurrent, or thermal overloads opening.

D. ANNUNCIATORS AND REACTOR CONTROL SYSTEMS

1. Annunciators

The experiment annunciator system consists of three parts: a local alarm light identifying the parameter which has strayed from normal operating conditions, a local horn to draw attention to the

off-normal condition, and a control room annunciator to inform the reactor control room operator.

A separate local annunciator unit must be installed for each power reduction. It should be operated from a normally open contact on the reactor control circuit relay so that this local annunciator indicates the power reduction rate requested by the experiment controls. It should have the same operating features as the standard annunciator.

Any parameter which should be kept within specified bounds by the operator must have an annunciator. The annunciator will bring the attention of the operator to any off-normal conditions.

The following operation of the local annunciator system is recommended:

Variable	Red Light	White Light	Horn
Normal	Off	Off	Off
Alert	On	On	On
Acknowledge	On	Off	Off
Return to Normal	Off	On	Off
Reset	Off	Off	Off
Lamp Test	On	On	Off

Field contacts may open or close to sound the annunciator. A normally closed relay contact must be provided. This contact is to open when the annunciator horn sounds, actuating the reactor control room annunciator.

An elementary wiring diagram of this annunciator circuit is shown in Figure 40.

Another commonly used annunciator system has the following operation:

Contacts		Position						
Handle	Ends	C	Inter	B	Inter	A	Inter	Off
1	2	x	x				x	x
1	2		x	x	x	x	x	
3	4				x	x	x	x
3	4	x	x	x	x			

Maintained Contact

Maintained Contact

Contacts		Position		
Handle	Ends	Start	*	Stop
1	1C 2C	2	X	
3	3C 4C	4	X	X
5	5C 6C	6	X	X

Maintained Contact (make before break) * Contact Overlap Between Start & Stop

Contacts		Position	
Handle	Ends	Comm	Diesel
1		1	x
2		2	x
3		3	x
4		4	x
5		5	x
6		6	x
7		7	x
8		8	x
9		9	x
10		10	x

	(7)	(8)
Maintained Contact		

Contacts		Position			
Handle	Ends	Pump M-3	Pump M-2	Pump M-1	None
1	1C 2C			X	
2	1C 2C			X	
3	3C 4C			X	
4	3C 4C		X		
5	5C 6C		X		
6	5C 6C		X		
7	7C 8C	X			
8	7C 8C	X			
9	9C 10C	X			
10	9C 10C				X

_____ [15]	_____ [15]
Maintained Contact	

● — Not Used

Fig. 39 Pump control circuit for loop.

Variable	Nameplate	Horn
Normal	Off	Off
Alert	Flashing	On
Acknowledge	On	Off
Return to Normal	Off	Off
Lamp Test	On	Off

Dozens of lead experiments operated at TRA use standard control panels, usually consisting of one or two recorders, one controller, and necessary interlocks. Since only one annunciator circuit is needed for experiments, these are usually designed and assembled locally. The system shown in the following table is recommended for these lead experiment systems.

NOTE All relay contacts are shown with coils de-energized

The relays A,B and C are components of each individual annunciator unit

Relay D and the horn are common to all annunciators on the console

Relay D activates the reactor control room annunciator

Fig. 40 Elementary wiring, typical annunciator circuit.

Local Alarm Standard System (Panelarm or Scram System acceptable also)

<u>Condition</u>	<u>Acknowledge Switch Position</u>	<u>Horn</u>	<u>Red Light</u>	<u>White Light</u>	<u>Green Light</u> *(optional)
Over Temperature or other Abnormality	Alarm	ON	ON	OFF	OFF
Over Temperature or other Abnormality	Acknowledge	OFF	ON	ON	OFF
Normal Temperature with no other Abnormalities	Acknowledge	OFF	OFF	ON	OFF
Normal Temperature with no other Abnormalities	Alarm	OFF	OFF	OFF	ON

* If green light is used, it must sequence in this manner. The horn, red light, and white light must be used and must sequence as shown.

2. Reactor Control Systems

A multicontact, female socket, Amphenol Type No. AN-3102A-36-1S at MTR and Type AN-3102A-40-2S at ETR and at ATR, is located near each experimental facility around the reactor. This is connected to all the reactor control relays located in the instrument room and to the experimenter's control room annunciator. At MTR the Amphenols are also connected to the sound powered telephone and buzzer located in the control room.

Each experiment requiring more than an annunciator must provide some means of releasing the reactor control system from false signals for power reduction. The release can be in the form of a "W" switch (General Electric Type SB) or test blocks.

Two other aspects of this "W" switch should be mentioned. One is the key interlock feature. Whenever the switch is in the "OFF" position, or the Amphenol plug is disconnected, or there is loss of reactor control power to the experiment, that experiment's annunciator is alarmed in the control room. To silence this annunciator when the experiment is not in operation, a similar switch in the reactor's control circuitry is opened. This disconnects the experiment from all connection with reactor control including those circuits not rendered ineffective by the experiment's "W" switch. To ensure against unauthorized use of the latter switch and to be certain that the correct switch is thrown when so authorized, it is of the keylock type and can be opened only by using the key from the individual experiment's "W" switch.

The second aspect of the "W" switch is its use as a device, where needed, to change the character of the experiment's supervision of the reactor power under different operating conditions. For example, where the experiment run time does not correspond with the reactor charge cycle, there will be periods during which the ex-

periment will be withdrawn from the high flux zone or operated at reduced conditions.

All ATR loops and several MTR and ETR loops have eliminated "W" switches from the loop console and replaced them with a test block system. This system is similar to the test block system used on reactor control systems, with one additional feature. To eliminate the necessity for changing wires when a change in power reduction rates is needed, several test block plugs are furnished; one for each power reduction. Each power reduction rate has a distinctive test block cover color (red -- scram, blue -- fast recovery scram, brown -- reverse, and yellow -- setback). This feature allows change of reactor power reduction rate without rewiring, and allows the operator to determine at a glance what power reduction rate is connected to each loop parameter.

In addition, the test block system allows one parameter to be temporarily disconnected from reactor power reduction circuits. If two test blocks on related parameters are pulled, an automatic reactor power reduction is initiated. Annunciator contacts alert the local panel and the reactor control room whenever one or more test blocks have been pulled.

The reactor control relays will be Square D Type AO or equivalent. Two each will be required for scram, jr scram, reverse, and fast setback and one for slow setback: a total of nine. Two relays which will cause reactor power reductions on loss of control power must also be installed. The experiment reactor control relays will be actuated from a terminal board.

This system is designed for maximum reliability. The power reduction relays are normally de-energized to minimize false power reductions. All wiring is double-tracked from instrument sensing point to Amphenol for reliability and is color coded to avoid confusion.

In addition, the power supply to all instruments initiating automatic reactor power reductions is on one circuit which is looped in such a way that a break in the wire at any point will not disrupt power to any instrument. An automatic reactor power reduction is initiated if power to the instrument circuit is interrupted.

A full set of reactor control relays is required for each major experiment to allow maximum flexibility for later tests. The experimenter may lay out the system at one time and decide on the rate of power reduction for any given test run later. A jumper between two terminal

blocks or a change of test blocks at ATR is all that is needed to complete the circuit. This feature permits a convenient change of reactor controls as experiment conditions change.

Each console should include a manual power reduction pushbutton switch. The rate of power reduction should be equal to the fastest automatic power reduction used on the experiment. In the case of scram or jr scram, it should consist of a single pushbutton, momentary contact, protected or recessed to minimize accidental operation. If the power reduction is the driven type, the switch should be of a lock-in type.

CHAPTER IX

DECONTAMINATION

A. INTRODUCTION

Decontamination of experimental loop equipment is sometimes necessary in order to prevent radiation fields from becoming so large that maintenance work on the loop equipment is severely handicapped or impossible. Routine decontamination of loop equipment is, therefore, an operational necessity, especially in those loops used for conducting defect and rupture tests.

Several loops have been decontaminated in the past. Resulting decontamination factors (ratio of activity difference to activity after decontamination) have varied from one to greater than ten depending upon location of survey point, initial survey reading, and decontaminant velocity at the survey point.

1. Source of Contamination

Contamination of the internal surfaces of loop piping and equipment can result from several sources. The most common cause of fission product contamination is the natural result of running defected samples in the loop. Fuel element cladding failures that occur unintentionally also result in fission product contamination. Such failures, while rare, usually result in more serious contamination of the loop than do deliberate defect tests.

Corrosion products, given sufficient time, will accumulate in quantities large enough to create contamination problems in systems circulating water through neutron flux zones. Radioactive metallic oxides will be found throughout such systems and tend to concentrate as "hot spots" in low water velocity areas.

Impurities found in the circulating fluids cause contamination in much the

same fashion as corrosion products. Contamination problems from impurities are usually slight since special effort is made to produce high purity coolant for loop usage.

2. Cubicle Equipment

The loop out-of-pile equipment, which consists of circulating pumps, pressurizer, heater, heat exchanger, connecting piping, etc, is normally installed inside of a cubicle made of barytes or normal concrete block. These walls are usually 2 feet thick to contain radiation fields and/or hazards from high pressure. The area inside each cubicle is highly congested with the various loop components. The presence of fission product contamination and/or corrosion products contained in these loop components will contribute to activity throughout the cubicle.

3. Reasons for Decontamination

The primary reason for decontaminating a loop is to reduce the radiation field inside the cubicle, subpile room, and reactor tank so that personnel exposure is not excessive. It may be necessary to decontaminate a loop because corrosion or fission product contaminants may be harmful to the test specimen. Decontamination may be required so that test data will not be obscured by high activity.

B. WASTE DISPOSAL SYSTEM

1. Gaseous Wastes

The disposal of gaseous wastes is covered by the following references: Volume 3, Chapter XIV, paragraph Q, Cubicle Exhaust System (ETR); Volume 2, Chapter VI, paragraph C, Process Water Building Operators' Area; and IDO-16372-III, Air Effluent and Contaminated Water Disposal System^[1].

2. Liquid Wastes

a. General

Modifications and system expansions to the MTR-ETR Waste Effluent Systems during 1960 provide more efficient handling and control of experimental loop and laboratory wastes. The highly radioactive and corrosive aqueous solutions with pH ranging from 1.5 to 13.5 are stored in two 10,000 gallon capacity, underground, stainless steel storage tanks with ultimate transfer to ICPP for final disposal. The original hot waste storage tanks will continue to handle MTR effluents which, given sufficient time to decay, can be pumped to the retention basin. Drawing MTR-E-5303 is a flow schematic of the new system.

b. Experimental Hot Wastes -- MTR Facilities

The MTR HED system provides a 2-inch type 304L stainless steel collection header system located and shielded under the MTR reactor building basement floor. Flanged connections to the HED system are flush with the floor, positioned immediately outside each of the experimental cubicles with few exceptions.

The HED drain system drains to a 600 gallon capacity type 304L stainless steel accumulation tank located in a concrete vault under the MTR Reactor Building southeast basement stairwell floor. The accumulation tank is vented via a 2-inch stainless steel pipe to the reactor exhaust header in the yard southeast of the reactor building. A type 316 stainless steel ejector (30 gpm 130°F water, 10-foot lift and 20-foot discharge) is used to transfer the tank contents through a 2-inch stainless steel discharge line to the stainless steel 10,000 gallon capacity hot waste storage tanks.

The accumulator tank control panel, wall-mounted in the stairwell above the tank vault, includes liquid level indication by an air bubble type instrument, a tank

temperature indicator, and 2-inch low pressure steam valves for manually initiating the tank discharge operation. A tank high-level annunciator is located in the MTR basement monitor room.

c. Experimental Hot Wastes -- ETR Facilities

The ETR HED system in the basement and the console floors contains 25 double type connections strategically located around the reactor biological shield and adjacent to building columns. There are also three single connections on the reactor first floor and three located within the experimental nozzle trench immediately adjacent to the reactor vessel. These normally closed drains (blind flanged, flush with floor or wall) are connected to a 4-inch diameter stainless steel type 304L shielded drain system which drains to a 500 gallon capacity, shielded stainless steel hot catch tank under the ETR basement floor. Turbine type tank transfer pumps (150 gpm, 2 hp) discharge the tank contents via a 4-inch stainless steel underground line to the hot waste storage area or to the inlet of the retention basin. Tank and pump controls are located in the ETR reactor control room and consist of a tank level indicator, individual pump control switches, running lights, and a pump selector.

d. Hot Waste Storage System

The hot waste storage system is comprised of four 10,000 gallon capacity tanks, two of which are carbon steel glass-lined and two type 316 stainless steel tanks. Immediately north of the underground storage are two hot waste pump houses each serving two hot waste tanks according to tank material; ie, one pump house serves the glass-lined tanks while the other serves the stainless steel tanks. Normally, MTR-ETR experimental wastes and laboratory wastes, being extremely corrosive, are routed to the stainless steel storage tanks only, since high pH solutions would attack the glass lining.

Hot wastes also originate from the MTR Reactor Building, the Reactor Wing Building, the Hot Cell Building, and the floor drains in the Reactor Services Building. These wastes are accumulated in the four 1500 gallon capacity, glass-lined catch tanks. The pumping facilities for the catch tanks are installed in the Catch Tank Pump House and include one centrifugal pump, manually controlled, rated at 50 gal/min at 31-foot head and one 125 pound steam ejector with an estimated flow rate of 70 gal/min. Piping interconnections permit using either the pump or ejector for transferring the catch tank wastes via three parallel pipe lines, one 3-inch and two 4-inch, to the Hot Waste Storage Pump House, the Retention Basin, or Retention Basin Sump. If the catch tank contents exceed allowable radiation limits, the effluent is transferred directly through the Hot Waste Storage Pump House and into one of the two 10,000 gallon glass-lined storage tanks. With the storage tank overflow arrangement, maximum capacity is limited to 8700 gallons in each tank.

At one Hot Waste Storage Pump House, two transfer pumps, each rated at 100 gal/min at 110-foot head, transfer waste from either 10,000 gallon capacity glass-lined tank back to the other, or to the Retention Basin, or to the resin bed in the ETR Heat Exchanger Building. The 3-inch diameter line, which forms a closed recirculating loop to and from the resin bed for liquid waste cleanup, is no longer used since the four resin beds are used exclusively for primary water cleanup.

At the other Hot Waste Storage Pump House, one transfer pump rated at 64 gal/min at 63-foot head, transfers waste from either 10,000 gallon capacity stainless steel tank through a metered 1-1/2-inch stainless steel discharge line to the Tank Truck Loading Facility. Only extremely radioactive or chemically active experimental and laboratory wastes will ever be routed through this Hot Waste Storage Pump House to the hot storage tanks.

e. Tank Truck Loading Facility

The Tank Truck Loading Facility, consisting of a concrete shielding canopy and fill and vent piping, is located immediately east of the Hot Waste Storage Area. A low-boy truck trailer, with a concrete shielded 1000 gallon capacity transfer tank, is positioned under the canopy during the hot waste loading operation. A 1-1/2-inch vent line and the 1-1/2-inch fill line are connected manually to the portable tank through quick disconnect couplings prior to the loading operation when radiation fields are low. When the portable tank is full as indicated through the 1-1/2-inch fill line totalizer-indicator recorder located in the instrument cubicle above the Hot Waste Storage Pump House, the lines to the portable tank are disconnected and the tank is then transported to ICP for unloading.

C. DECONTAMINATION OF LOOPS

1. Summary

The reasons for internal decontamination of test loops are given in paragraph A. Loop contaminants may generally be classified either as crud, including corrosion products, or as fission products. Decontaminating procedures may be similarly classified, but are frequently modified for improved effectiveness for a specific situation. The types of chemicals used in water loops are described below.

Decontamination may never reach the "cook book" stage because of the variety of possible combinations of loop operating conditions, loop materials, contaminants, and new decontaminant solutions and procedures. Each case is individual and a new literature review is always desirable where time permits. A large number of reports are available on loop decontamination. However, many do not give data on possible damage to loop components. The use of commercial compounds, even though known to be effective, is not always desirable on high pressure

systems because of the unknown composition of the inhibitors used.

Two basically similar processes are commonly used for decontamination: the feed-and-bleed and the batch process. The feed-and-bleed process starts with the loop circulating demineralized water at specified conditions. Concentrated chemicals are added by a positive displacement pump with loop pressure controlled by a back-pressure regulator bleeding loop solution into the drain system. When the concentration of chemicals reaches the required value, feed is adjusted to hold this concentration constant for the required period. At the end of the run, demineralized water is added by the feed pump until the concentration of chemicals in the loop has reached a low value. The loop is then flushed, and the next step of the decontamination is started.

In the batch process, the chemicals are added to the loop at the proper concentration for decontamination. The loop is then pressurized and the solution circulated for the required time at specified conditions. It is then dumped to drain and the loop thoroughly flushed.

2. Special Equipment Needed

In addition to the loop components, certain specialized equipment is required for a decontamination run. These include one and preferably two tanks for mixing the chemicals. These should be insulated and must have provisions for heating and stirring the chemicals. A venting system for the tanks will protect the operator from noxious fumes.

A positive displacement variable feed rate pump for feeding chemicals into the loop is required. Most systems in use at TRA use air operated Aldrich pumps. A backpressure regulator leading from the loop to the hot drain system controls loop pressure. The surge tank is usually isolated, since it is usually not a serious source of radiation and because it is almost impossible to maintain the high water

velocities necessary for efficient decontamination.

The primary pump usually has Graphitic bearings, which can be damaged by the decontaminating solutions. These are protected from the solutions by a purge through the vent plug, past the bearings, and into the loop system. Purge water is furnished by loop makeup pumps (one makeup pump ties directly to each primary pump) or a large capacity pump feeding all primary pumps in parallel. Purge water flow through each pump must be monitored continuously.

Decontaminating solutions must not run through the resin columns, since they would quickly saturate the resins. Violent chemical reactions between resins and decontaminating solutions are also a distinct possibility.

3. General Description of Decontaminating Processes.

a. Removal of Radioactive Corrosion Films

The inside surface of all loop piping is coated with a thin film of magnetite (Fe_3O_4). This material has a fine crystalline structure which retards corrosion of the stainless steel. However, it migrates through the loop and, therefore, becomes radioactive after a long period of operations.

Magnetite is formed from long term operation of the loop at high temperature and low oxygen content. There is evidence that its formation is catalyzed by high radiation fields. Both radioactive corrosion products and fission products are mixed with the layers of magnetite as they are formed. When the radiation fields caused by the film approach intolerable levels, the film must be removed.

The magnetite (Fe_3O_4) can be changed by oxidation to Fe_2O_3 , a coarse, jelly-like material which hydrates easily, or to $\text{Fe}_2\text{O}_3 \cdot \text{H}_2\text{O}$, rust. Either material can easily

be flushed out of the system by high velocity water.

Many combinations of oxidizers and buffering agents such as hydrazine and sulfamic acid may be used. The most successful and widely used chemicals, however, are sodium hydroxide (NaOH) and potassium permanganate (KMnO_4). These constitute the first phase of decontamination used on all NR water loops at TRA.

The decontaminating solution is circulated through the loop at the optimum pressure, temperature, and velocity for approximately eight hours. The loop is then flushed thoroughly with demineralized water.

The loop is then filled with a weak solution of ammonium citrate or similar chelating solution. This solution selectively dissolves divalent metals, especially iron, and removes the last traces of NaOH and KMnO_4 and their compounds.

With all of the films removed from the metal surfaces, the loop radiation levels are at their lowest possible values and, after thorough flushing, the loop could be operated again. However, recent experience indicates that highly radioactive magnetite films would be redeposited within one or two reactor cycles and radiation levels would be as high as before decontamination.

It is necessary to redeposit a thin layer of nonradioactive magnetite on all exposed surfaces of the loop. This process, called passivation, can be accomplished by extended loop operation at over 400°F , high pH, and a very low dissolved oxygen content. One hundred hours of operation is the minimum time for proper conditioning. LiOH resin is used during this run since it promotes the deposition of iron oxide on the metal surfaces.

b. Removal of Fission Products

Most fission products are removed by the processes mentioned above. However,

if measurable amounts of uranium or uranium oxide have been released as well as fission products, a decontamination solution specifically designed for uranium removal may be used.

This process uses hydrogen peroxide, plus complexing agents and stabilizers. Hydrogen peroxide selectively oxidizes uranium and UO_2 , changing them to U_3O_8 . While uranium and UO_2 are relatively insoluble, U_3O_8 is soluble in dilute acid solutions such as sulfuric acid or ammonium citrate.

Operating conditions in the loop are similar to those used in removal of corrosion products; a peroxide run, a flush, a weak acid rinse, and another flush reduce radiation levels to acceptable levels.

Recent experience indicates that fission products from operation with defective elements can be handled more successfully by the decontamination methods used for corrosion products than for those methods specifically designed for uranium removal. The peroxide decontamination (CODS-4 on NR loops) is seldom used.

c. Loop Cleanup Processes

After a gross fission product release takes place in a loop, most fission products will be concentrated in the ion exchange columns. Thorough degassing will concentrate gaseous fission products in the surge tank. These can be vented to atmosphere at controlled rates during ideal weather conditions. If loop radiation levels are still too high, a corrosion product decontamination (APAC) will be accomplished, followed by a thorough conditioning run.

4. Limitations on Decontamination Efficiency

The efficiency of a given decontamination run varies widely throughout the loop. Piping can be thoroughly cleaned where decontamination solution velocity can be maintained at an optimum value.

However, dead spots in the lines such as the pressurizer and associated piping, dead legs in the piping, and similar areas are relatively unaffected.

These points of low velocity are also

the points where radioactive materials preferentially plate out, compounding the problem. In new loops, many of these problem areas can be avoided. On older loops, the high radiation areas should be removed or shielded.

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CHAPTER X

CAPSULE IRRADIATIONS

A. INTRODUCTION

Many experiments irradiated in the reactors at the TRA are of the capsule type. Capsule irradiations are used for the study of the effect of neutrons on practically all materials. Capsule programs are concerned with fuel element development, control rod development, study of radiation damage to materials, and radioisotope production. Capsule experiments can vary in size and complexity from a very small sample, contained in a small polyethylene capsule, to a 10- x 10- x 20- inch aluminum box containing complex electronic equipment.

The capsule experiment differs from other types of experiments in that a capsule experiment does not normally have its own cooling system. It is usually cooled by the reactor cooling system. Reactor water enters the core at constant flow, temperature, and pressure. Differential pressure across the core is also held constant. This being the case, the bulk water cooling the capsule will operate at conditions which vary only slightly with the capsule's position in the reactor. Most of the engineering studies conducted for the safety and success of capsule experiments concern nuclear problems, heat transfer, corrosion of materials, and rigidity of construction.

The capsule experiments can be broken down into two main groups: out-of-tank capsules and in-tank capsules.

B. OUT-OF-TANK CAPSULES

Out-of-tank capsules are inserted into irradiation facilities in the graphite section of the MTR and are cooled by reactor air. A description of these irradiation facilities can be found in Volume 2 on pages 25 to 42 and in Figures 5, 6, and 9. The

ETR and ATR have no out-of-tank facilities for experiments. The following three types of out-of-tank experiments are used:

1. VG Facility Experiments

VG facility experiments are those experiments that are suspended from the standard plug of a vertical graphite hole and inserted into the flux zone. Because most of the VG plugs cannot be removed while the reactor is at power, VG facility experiments are suspended above the active section of the VG hole by means of a piece of airplane control cable during shutdown. After the reactor is at power, the experiment is lowered into the flux zone by means of the cable, which extends up through a hole in the plug to the reactor top. This hole is in the shape of a spiral to prevent a radiation beam from leaving the hole. After the irradiation is completed, the experiment can be withdrawn from the flux zone by raising the cable. Discharge of the experiment is made during a reactor shutdown.

a. Purpose

VG facilities provide for the testing of materials in an air-cooled environment and in a thermal neutron flux of 5×10^{10} to 2.0×10^{13} neutrons/cm²-sec. VG facilities are often used for testing of complex electronic equipment under radiation environment.

b. Special Hazards

Contamination and radiation problems are important in considering VG facility irradiation. The capsule must be designed so that appreciable amounts of radioactive material will not be released while in the reactor and so that it can be handled without the spread of contamination after irradiation. The radiation problem is less

serious with VG facility irradiations since the induced activity in the experiment can be allowed to die off while the experiment is stored directly under the plug. For experiments that are expected to be very radioactive at removal, a special shielding cask can be constructed that will lower the radiation level during the transfer of the experiment to a shielded shipping container.

Since the VG facilities will accommodate experiments of considerable weight and experiments containing fissionable materials, a heat transfer analysis is often necessary.

As with all experiments that are inserted into the reactor, the irradiation stability of all materials to be irradiated must be considered. For example, when VG facility irradiations involve heater leads and thermocouple leads, wire insulation must be considered with regard to contamination problems. Fiber glass insulated thermocouple wire is often used for VG facility irradiations. Asbestos insulated thermocouple wire tends to flake off during and after irradiation, causing a contamination problem.

One problem is peculiar to VG facility experiments. If the experiment contains large quantities of high cross section materials, it can depress the flux in adjacent facilities when it is lowered into the flux zone. If the sensing device for a reactor instrument is located near a VG hole, the insertion of the VG facility experiment can affect the instrument reading. Examples of this problem are the large boral boxes which have been used to contain experiments in VG-9. The insertion of these boral boxes gives an indication of low flux on the reactor control instruments which are near the hole. To compensate for this, the sensing chamber is moved closer to the core.

2. VG Pneumatic Rabbits

The simplest type of out-of-tank experiment is the VG (Vertical Graphite)

pneumatic rabbit. Four special VG hole plugs have been designed to accommodate the insertion of small, light-weight capsules (pneumatic rabbits) into the flux zone of a VG hole. A "rabbit catcher" is available that permits the insertion and removal of a pneumatic rabbit while the reactor is at power. The four special plugs are currently located in VG-7, VG-23, VG-33, and VG-62.

Most pneumatic rabbits consist of a screw-cap type 0.6-inch OD x 1-7/8-inch long polyethylene capsule containing a small sample. Due to limited air pressure in the pneumatic tube, the capsule and sample can not exceed 15 g for successful pneumatic rabbit operation.

a. Purpose

The pneumatic rabbit facility provides a system whereby small samples can be exposed to a neutron flux for a short period of time. It is especially useful where studies are being conducted on specimens of short half-life, since pneumatic rabbit samples can be removed from the flux zone and transferred to the laboratory in a very short period of time.

b. Special Hazards

The advantages of the polyethylene capsules are that they hold up well under irradiation and that they do not become extremely radioactive. Although most of the pneumatic rabbits are run for a very short time, the polyethylene capsules will hold up satisfactorily for irradiations of at least four days in these facilities. Since the polyethylene rabbits are relatively cheap, it is standard practice to use a new rabbit for each irradiation. The polyethylene rabbits are stock items at the TRA storeroom.

Four engineering considerations important to analyzing the safety of pneumatic rabbits are: (a) stability of the sample under irradiation; (b) ability of the system to remove the heat generated; (c) proper containment to prevent the

escape of any material from the rabbit during or after irradiation; and (d) the radiation level of the rabbit and sample at the time of removal from the pneumatic rabbit facility.

3. Thermal Column Experiments

A number of experiments have been performed in the past in facilities contained in the thermal column on the east face of the MTR. The horizontal T2H6 hole has been used for the calibration of reactor instruments and for the irradiation of several experiments for which relatively pure thermal neutron fluxes of from 1×10^8 neutrons $\text{cm}^{-2} \text{sec}^{-1}$ to 1×10^{11} neutrons $\text{cm}^{-2} \text{sec}^{-1}$ are required. Special plugs can be used in T2H6 that allow an operator to remove an experiment while the reactor is running. The main engineering problems involved in thermal column irradiations are contamination and radiation level during removal of the experiment.

C. IN-TANK CAPSULES

The in-tank capsules are inserted into the reactor tank facilities. The capsules consist of the material to be irradiated, sealed against reactor water if necessary, in a container which will fit existing reactor facilities. A description of the reactor tank facilities is found in IDO-16428 [1]. This document also includes a section concerning the general design of in-tank capsules.

A complete analysis is conducted for each in-tank capsule experiment. It is similar to that followed for the example presented in Appendix A. The details of the analysis, as they apply to capsules, can be obtained as follows:

Nuclear Analysis Chapter I

Hydraulics Chapter II

Heat Transfer Chapter III

Hazards Analysis Chapter XIV

In addition to the details described in these chapters, capsules are unique in that they are frequently designed to perform a specific function in a specific manner. Two examples of this are temperature control and thermal shock.

The in-tank capsules can be broken down into four types: leaky capsules, metallurgical specimen bundles, sealed capsules, and instrumented capsules.

1. Leaky Capsules

The leaky capsule is a container which fits existing reactor facilities and has penetrations which allow reactor process water to pass through the capsule. A number of sizes of empty aluminum leaky capsules are available as stock items at TRA. The sample to be irradiated is inserted into the capsule. As the reactor process water is in direct contact with the material in the capsule, it is necessary to seal corrosive materials in a small aluminum or stainless steel container prior to placing it into the leaky capsule. The sample or container is packed in the leaky capsule with aluminum foil or stainless steel springs to minimize damage by vibration. After irradiation, the samples may be removed from the leaky capsule before shipment is made.

2. Metallurgical Specimen Bundles

Metallurgical specimen bundles may consist of several tensile or impact specimens with two end pieces. As the specimens are in contact with process water, only those metals or alloys that resist corrosion in process water can be irradiated in this manner. Because of space limitations, the tensile specimens irradiated in this manner are usually subsized specimens.

3. Sealed Capsules

Sealed capsules consist of a sample sealed in an outer container. Most of the in-tank experiments are of this type. The specimen temperature is determined by the heat transfer medium between the test

specimen and the outer container. Sodium or NaK is used when good heat transfer is required. Where high specimen temperatures are desired, the outer shell is constructed with a double wall having a gas annulus between. The temperature of the inner specimen is then determined by the width of the annulus, annulus gas composition, heat transfer area, and flux. By proper gas annulus design, extremely high core temperatures can be attained.

4. Instrumented Capsules

The instrumented capsule (lead experiment) consists of a capsule to which a lead tube has been attached in order to provide a method of bringing thermocouple wire, heater leads, gas tubes, etc, out of the reactor tank. Auxiliary equipment associated with instrumented capsules can include recorders, controllers, gas supplies, etc. When it is found to be necessary, circuits from lead tube instruments can be connected to reactor safety circuits to assure safe operation.

The lead tube of an instrumented capsule must be attached to the capsule so that a seal is made against process water entering the lead tube. In addition, the leads (thermocouples, gas lines, etc) must be sealed as they leave the capsule portion of the lead experiment in order to isolate the capsule from the lead tube. This is done to prevent materials in the capsule from getting into the reactor process water or into the reactor building atmosphere in the event of a lead tube failure.

Because of the extreme turbulence of the process water above the reflector, stainless steel lead tubes are generally more satisfactory than aluminum lead tubes. Between the capsule and the lead tube access flange, the lead is secured to the reactor tank at the upper and lower clamping rings. Prior to insertion into the ETR and ATR, bending schedules are prepared so that the lead tube can be bent to fit into the tank properly. Pre-bending is not normally necessary for MTR lead experiment insertion.

Before insertion into the reactors, all lead tubes are pressure tested with helium. All electrical and thermocouple leads are checked for continuity by obtaining resistance readings.

After the lead tube has been brought through a seal in the lead tube access flange, a sealed terminal box is attached to the upper end of the lead tube. The thermocouple leads from the experiment instrumentation are connected at the terminal box. Gas lines are brought through seals in the side or upper end of the terminal box. After the terminal box is attached, the lead tube, including the terminal box, is pressure tested and the thermocouples and heater leads are again checked for continuity.

During the irradiation of the lead experiments, any moisture which may have been in the lead tube prior to installation may decompose to form hydrogen and oxygen gases. In some lead tubes in the past, an explosive mixture of gas has been found. Therefore, it is required that the experimenters provide a purge line to the bottom of the lead tube so that any gas so formed can easily be removed.

Helium pressure slightly above reactor tank pressure is maintained in the lead tubes. If the pressure in a lead tube rises, the lead tube is purged with helium in order to remove accumulated gases.

There have been many complex instrumented capsules irradiated at the TRA. Three of these are briefly described below.

(1) Capsules containing heaters

Some instrumented capsules have electric heaters in the sodium or NaK bath for the purpose of temperature control. These heaters are not intended to increase the fuel temperature through heat radiation, but rather to increase the temperature of the heat transfer medium. Since the temperature difference, and therefore the film coefficient, between the fuel and the heat transfer medium is the

controlling factor, an increase in sodium or NaK temperature causes an increase in the fuel center temperature.

(2) Capsules containing a gas mixture for temperature control

Temperature control on some samples is accomplished by using a mixture of two gases, one an insulating medium and the other a heat conducting medium, in the annulus between capsule containers. Gases of different thermal conductivity are chosen so that a wide control range is possible by variation of the relative amount of each gas. A number of special problems are involved with this type of capsule. If the gases chosen become

quite radioactive (argon for example), provision must be made for shielding the outlet lines. The effect of filling the annulus with pure gas of either type must also be analyzed to determine whether adverse temperature conditions would arise due to the increase or decrease in thermal conductivity.

(3) Capsules designed to study thermal shock.

Capsules used to determine thermal shock effects are periodically inserted into and removed from the flux zone inside the reactor tank by remotely operated mechanisms.

REFERENCE

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CHAPTER XI

WATER LOOPS

A. INTRODUCTION

A loop is an assemblage of equipment which provides a specialized circulating environment in which the irradiation of experiments can be accomplished. A water loop, for example, uses water as the primary environmental medium. This water is at pressure, temperature, and chemistry conditions which are usually different from those found in the test reactor. Any attempt to expand the definition beyond this will inevitably encounter exceptions.

This chapter will describe various parts of the water loop and their functions. Basic loop operation and special hazards will also be described.

B. COMPONENTS OF THE LOOP

1. Introduction

A water loop can be separated into several parts, each having a specific function which is usually the control of some loop variable. These parts are the pressure control system, the flow control system, the temperature control system, the pressurizer liquid level control system, and the various side stream systems. Side streams include the water cleanup system, the fission break detection system, the makeup system, the sampling system, and the degassing system. In this section, each of these systems will be described.

2. Pressure Control

a. System Descriptions

The in-pile system pressure in a low pressure single pass loop is usually maintained by using the pump head and a back pressure regulating valve on the outlet of the in-pile tube.

All existing high pressure water loops, however, utilize a pressurizer or surge tank in which the water is maintained at the saturation temperature for the desired operating pressure.

In addition to maintaining loop pressure, the pressurizer should be large enough to compensate for changes in the loop water volume which result from changes in the water temperature. The pressurizer is usually operated approximately half full of water, with saturated steam occupying the remaining volume.

The pressurizer is connected to the loop by a single line running from the bottom of the pressurizer to the suction side of the loop pumps. The pressurizer is thus thermally isolated from the loop. This isolation is necessary so that the loop may be operated at a lower temperature than is required to maintain the desired operating pressure.

Heat is generally added to the pressurizer water by electrical heaters clamped on the outside of heater legs. Each heater leg consists of a small convection loop which comes off the bottom of the pressurizer and reenters a few feet above the bottom. Heaters clamped around the vessel proper and immersion heaters have also been used.

One system used to control the amount of electrical power supplied to the pressurizer heaters utilizes a proportional pneumatic signal to control the output of a saturable core reactor feeding the heaters. This system provides proportional control and can control pressure to plus or minus 1 percent. Load changes are compensated for automatically. A simpler system turns the pressurizer heater current on and off as demanded by pressure sensing elements and can control loop pressures to plus or minus 3 percent.

It is desirable to provide a method for removing dissolved gases from the loop water. To accomplish this, a small line is usually run from the discharge side of the loop pumps to the top of the pressurizer. By opening a valve in the line, the loop is purged by allowing the water to flow from the loop to the pressurizer and back to the loop through the line in the bottom of the pressurizer. An interchanger can be installed in the lines to prevent saturated water from entering the loop from the pressurizer and cold water from entering the top of the pressurizer from the loop. Degassing is usually not a continuous operation but is used for relatively short periods. The system may also be used to quickly depressurize the loop by cooling the surge tank water and to remove fission product gases from loop water after a fission break.

b. Emergency Pressurization Systems

Some loops at the MTR-ETR use an emergency pressurization system consisting of a nitrogen or helium supply to the top of the pressurizer. These loops were originally equipped with an automatic system for supplying the gas to the pressurizer should the loop pressure drop below a certain value. The automatic systems have been disconnected due to the feeling that if the system is losing pressure so fast that the gas blanket could not be established by hand, the gas would aggravate the situation by forcing or blowing the water out of the loop at an increased rate. As a general rule, it is safer to run the pumps at low suction pressures than to risk the possibility of having a hot experiment standing in a gas atmosphere.

c. Safeties

Two forms of pressure safeties are provided on the loops: (a) mechanical safeties, and (b) alarm and control safeties. The mechanical safeties consist of a rupture disc and relief valve in series. They are installed on the in-

pile tube and on the pressurizer. The relief valve would offer complete protection; but since they have a tendency to leak, rupture discs are installed to prevent leakage.

The alarm and control safeties consist of alarms, reactor power reductions, automatic pressurizer heater cutouts, and automatic shutoff of the makeup pumps. Low system pressure actuates an alarm; and if the pressure drops lower, an automatic reactor power reduction is initiated. High pressures actuate an alarm and higher pressures turn off the power to the pressurizer heaters and shut off the makeup pumps. A switch should be installed to bypass the makeup pump cutoff for pressure testing.

3. Flow Control

a. System Descriptions

The usual flow control system on water loops uses a flow element and a pneumatically operated diaphragm valve. The flow element pressure taps are connected to a pneumatic transmitter whose 3 to 15 psig air output goes to a recording flow controller. The controller output air signal activates the pilot unit on the diaphragm control valve. Pressure switches, which initiate reactor power reductions resulting from high or low loop flow, are also connected to the output of the transmitter. The flow control valves are of the air-to-close type. Loss of control air pressure thus results in the safe position of the valve.

Two flow sensing elements are usually installed in the primary loop system, one serving as a backup for safety reasons. The flow element used for control should always be located on the in-pile side of the bypass line.

Some loops have used a variable frequency MG set to supply power to synchronous motor driven primary pumps. Flow control was accomplished by varying the pump speed.

b. Emergency Flow Systems

The emergency flow system at the MTR is the ECL (Emergency Cooling Loop). This system circulates reactor process water by means of separate booster pumps. In the event flow cannot be maintained on an experimental loop during the reactor operating cycle, the in-pile tube is manually valved into the ECL system. The ECL flow cools the in-pile tube, allowing the reactor and the other experiments to operate for the remainder of the cycle. The loop experiment is usually removed before valving in ECL.

The ECL system at the ETR has been deactivated. The ETR system consisted of a separate loop, independent of the reactor, and the experimental loops. In the event flow cannot be maintained on an ETR experimental loop during the reactor operating cycle, the loop experiment is removed, and an inverse U-tube is installed. One leg of the U-tube is inserted into the in-pile tube and the other leg into a reflector position. Reactor process water flows into the in-pile tube and through the inverted U-tube. Driving force for this flow is the pressure drop across the core.

c. Pumps

Each loop must have a minimum of two primary circulating pumps installed. If, for example, X number of pumps are required to maintain safe flow rates at reduced temperature conditions, then $X + 1$ pumps are required. At least two pumps should be operating at all times. One of these operating pumps must be connected to diesel or failure-free power and the other to commercial power. The preferred system of multiple pump operation is to run all but one operating pump from the commercial power source. This gives maximum protection in the event of a failure in either power source without overtaxing the diesel or failure-free systems.

The most expensive and important items in a loop, except for the experiments, are the primary pumps. Proper operation of these circulating pumps is the major concern of everyone. With one exception, all the loops have primary pumps of the canned rotor type. Figure 41 is a cutaway drawing of a Westinghouse 150D canned rotor pump. Other makes of canned rotor pumps are different in construction, but the principle of operation and basic parts are the same. The figures are self-explanatory, and it should be noted that the auxiliary water is primary loop water which has passed through the thermal barriers into the upper regions of the pump.

A typical pump control circuit for a high pressure water loop is described in Chapter VIII, paragraph C-3.

d. Safeties

Reactor power reductions are installed on every loop. By utilizing proper setpoint values, the experiment can be protected from the possibility of continued reactor operation after loss of loop flow. A reactor power reduction does not provide positive protection against the effects of nuclear afterheat, since there is no guarantee of uninterrupted loop flow.

4. Temperature Control

a. System Descriptions

The desired operating temperature is maintained on most loops by balancing the heat removed by heat exchangers against that added by nuclear heat and by the line heaters shown schematically on the flow sheet (see Figure 42). Some loops use only a heat removal system and depend on nuclear heat supplied from the reactor and experiment to maintain operating temperature.

The secondary side of the heat exchangers is usually cooled with ECL

CANNED MOTOR PUMP

SCHEMATIC FLOW DIAGRAM

SECTIONAL VIEW

LEGEND:

1. PUMP CASING
2. FLANGE & STATOR ASSEMBLY
3. THRUST RUNNER
4. FLANGE CAP SCREWS
5. WEAR RING INSERT
6. CAP SCREWS
7. IMPELLER NUT
8. IMPELLER
9. CAP SCREWS
10. THERMAL BARRIER
11. CAP SCREWS
12. LOWER BEARING ASSEMBLY
13. IMPELLER KEY
14. ROTOR ASSEMBLY
15. CONDUIT BOX SCREWS
16. CONDUIT BOX
17. BOLTING PLATE CAP SCREWS
18. BOLTING PLATE
19. UPPER BEARING & PLUG ASSEMBLY
20. SNAP RING
21. TERMINAL ASSEMBLY
22. IMPELLER LOCKING CUP
23. DISCHARGE NOZZLE
24. SUCTION NOZZLE
25. STATOR BACKUP SLEEVE
26. BACKUP SLEEVE INSULATION
27. COOLING WATER OUTLET
28. COOLING WATER JACKET
29. STATOR SHELL
30. ROTOR CORE ASSEMBLY
31. COOLING COILS
32. ROTOR CAN
33. BACKUP SLEEVE INSULATION
34. ROTOR BALANCING PLUG
35. STATOR BACKUP SLEEVE
36. STATOR CAN
37. COOLING WATER INLET
38. VENT PLUG
39. SECONDARY IMPELLER
40. STATOR CORE ASSEMBLY

Fig. 41 Canned rotor pump.

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Fig. 42 MTR-ETR typical water loop flow diagram.

water in the MTR loops and HDW (high pressure demineralized water) in the ETR loops. Several MTR-ETR loops use Dowtherm heat exchangers, and one MTR loop uses an air-cooled heat exchanger.

Dowtherm systems are sometimes used for both heating and cooling. The Dowtherm temperature on the secondary side of the primary heat exchanger controls the direction of flow of thermal energy.

The water-to-water heat exchanger system uses a three-way valve to bypass primary water around the heat exchangers. The secondary water flow is maintained at some predetermined constant value, and temperature control is achieved through bypassing varying amounts of primary loop water around the heat exchanger. The three-way valve in the primary system is connected to the pneumatic output from a temperature controller. The valve is always installed so that loss of air will result in the entire primary water flow being routed through the heat exchangers. A solenoid valve may be placed in the air line from the controller to the valve, providing an automatic cool-down feature for the loop. The automatic cool-down may be actuated by low loop pressure or flow, high loop temperature, or fission break detection instruments.

The Dowtherm secondary systems have heat exchangers and heaters. The temperature of the Dowtherm is controlled by line heaters and by passing a portion of the Dowtherm through a Dowtherm-water heat exchanger. The loop temperature is maintained as a function of the Dowtherm temperature and flow rate.

The line heater systems utilize both the clamp-on and the immersion type heaters. In general, clamp-on heaters are preferred because they avoid the necessity for high pressure seals and alleviate the severe contamination problems which would occur during heater maintenance or repair. The control of the electric power to the heaters is similar to that for the

pressurizer heaters. It should be noted that the loop heaters usually do not require a saturable reactor control system for good, smooth temperature control if the three-way valve on the heat exchanger system is used as the prime control element. Experience indicates that the three-way valve system is suitable for "fine" temperature control.

A typical water loop temperature control system is described in Chapter VIII, paragraph C-2.

Several points need careful consideration in designing the temperature control system for the experimental loop:

(a) The loop heaters should be sized large enough to heat the entire loop to operating temperatures without nuclear heat. The heaters should be separated into several banks to minimize losses in case of heater failure during a cycle.

(b) It should be possible to isolate the secondary side of the heat exchangers. Vents and/or drains must be provided to prevent steam buildup on the secondary side. It is helpful if reach rods are provided on the secondary valves.

(c) The loop heater wiring to individual heaters should be connected to terminal boards outside the shielded cubicle so individual heater checks are possible during loop operation. This also applies to the pressurizer heaters.

(d) Extra long waterproof terminals should be provided on the heaters. Shut-down periods are brief, and time to properly dry fresh insulation is seldom available. The long waterproof terminals will eliminate short circuits caused by moisture in the insulation.

(e) All major loop thermocouples should be provided with spares connected to a patch panel at the control console. This ensures continued operation if a thermocouple should fail during reactor operation.

b. Emergency Cooling Systems

Emergency cooling systems are usually applied to the in-pile tube as part of the emergency flow systems described above. No loop is equipped with a full power emergency or backup cooling system.

c. Safeties

The overtemperature safeties on a loop consist of at least two separate sensing and alarm systems. Each is connected to a reactor power reduction circuit. A loop heater cutoff feature which will override the loop temperature controller is included with the high temperature alarm. Some loops have incorporated an automatic cooldown (ACD) feature in the high temperature safety system. An ACD usually applies maximum heat exchanger cooling capacity to the loop without regard to the loop temperature controller.

5. Liquid Level Control

a. System Descriptions

The liquid level systems used on the pressurizers at the MTR-ETR consist of a D-P (differential pressure) cell connected to a full reference leg and to the pressurizer itself. The output from the D-P cell will then be inversely proportional to the liquid level. Some of the loops use an inverter in the output of the D-P cell; the signal received at the panel boards will then be proportional to the liquid level. The system shown on the flow sheet incorporates a method for filling the reference leg without filling the entire pressurizer (see Figure 42). The system shown on the flow sheet is connected so that the liquid level range covers the entire height of the pressurizer. Many of the loops are built so that zero liquid level is just above the pressurizer heaters. The method shown on the flow sheet is preferred because it provides some indication of the liquid level over the entire length of the pressurizer. If

the liquid level indication includes level in the heater section and if the heaters are connected in banks from top to bottom instead of being grouped on the various heater legs, the operator can maintain pressure for longer periods in emergencies.

The liquid level controls on most loops consist of turning on one or more makeup pumps by means of pressure switches actuated from the output of the liquid level D-P cell. The automatic system should be equipped with an automatic-manual selector switch so that manual control can be used.

Pressurizer liquid level control is also described in Chapter VIII, paragraph C-1.

b. Safeties

Most systems include both high and low level alarms. These are usually not connected to reactor power reductions; however, they are interlocked with the makeup pumps and heaters as follows:

(a) High level should shut off the makeup pumps and the pressurizer heaters. However, a high level bypass switch or a manual operating switch must be installed to allow operation, under operator surveillance, during loop hydrostatic pressure tests.

(b) Low level should shut off the heaters. However, a bypass switch is usually installed to allow operation under operator surveillance as mentioned above.

6. Side Streams

a. Cleanup Systems

All loops should be equipped with a water cleanup system regardless of the necessity for water cleanup during normal operation. The complications of a major fission break can be largely reduced by the installation of an adequate loop cleanup system. The system should be designed

to provide a flow rate high enough to clean up the loop water after a fission break in a reasonably short time.

The usual loop cleanup system consists of two ion exchange resin columns and associated valves and piping. Since most resins are not able to withstand high temperature operation, the side stream must be provided with a cooling system. An interchanger is, therefore, a necessary part of the system, preventing the return of a cold water stream into the hot loop water.

One resin column should be of sufficient capacity to function for at least one reactor cycle. The resin column should be designed for remote removal and addition of resin. For those designs which use manual resin addition, each column should be of sufficient capacity to clean up the loop water in the event of a major fission break.

The columns should be located in a separate shielded area inside or adjacent to the cubicle. This is necessary to prevent high radiation levels which would develop inside the cubicle as a result of contamination buildup inside the resin columns. The entire cleanup system should be equipped with enough valves to make the system completely flexible. If the cleanup system is integrated with the sample and D/N (delayed neutron) system as shown in the flow sheet, the valves should be located inside a closed, ventilated sample house.

The preferred resin changeout system is of the remote controlled type. The old resin is flushed into a shielded, portable cask with a "throw-away" inner container which can be handled remotely if necessary. New resin is flushed into the columns from a remote location.

If the disposal container for the remote addition design is located inside the primary cubicle, the container must be capable of holding enough resin to clean

up the loop after a major fission break. It would be far better to locate the disposable container outside the primary cubicle where a volume twice that of the resin column would be sufficient. Locating the disposable container outside the primary cubicle involves other problems such as ventilation, leakage, and additional personnel shielding.

The cleanup systems are usually equipped with an automatic shutoff valve to stop the IX (ion exchange) system flow if the inlet water temperature to the resin exceeds a preset value.

b. Fission Break Detection System

The fission break detection system consists of one or more gamma counting heads and a D/N counting system.

Some loops have only the gamma heads. These detectors are located in the cubicle, usually mounted on or next to primary loop piping or other locations of fission product deposition (eg, heat exchangers, ion-exchange columns). A high cubicle radiation field will initiate an alarm and, in some loops, a reactor power reduction. It is important that the gamma system does not "saturate" and indicate a low level at increasing gamma fields.

The D/N system shown on the flow sheet (see Figure 42) is one of several possible approaches. Most delayed neutron counting heads cannot withstand high temperature operation; hence, it is necessary to cool the primary water sidestream before it reaches the counting head or cool the counting head directly.

It is possible to install the counting head inside the primary cubicle so that it directly scans the loop piping. The holdup chamber for N-16 decay is an excellent location. Locating the counting head inside the cubicle eliminates the possibility of repair to the system while the loop is operating.

c. Makeup Systems

The makeup system should consist of two high pressure makeup pumps and a large capacity fill pump. The fill pump is not necessary if the loop can be filled with the demineralized water provided as a plant utility.

Decontaminating chemicals are usually charged to the loop through the makeup line. The fill pump is used on some loops for decontaminating.

The makeup tank should hold at least one loop volume of water. It should be equipped with two access ports, one for the installation of a mixer and the other for adding chemicals which may be required for decontamination.

The water may be prepared for loop use by boiling or by circulating through a resin system. The system shown on the flow sheet shows both resin preparation and boiling preparation. If boiling is done in the makeup tank, cooling water can be circulated through the steam coils to provide faster cooling of the makeup water.

d. Sampling System

Loop sampling systems vary widely. The sampling system outlined below meets minimum requirements.

(1) A sample point should be installed in the system before and after the IX columns. These samples can be drawn from the loop at atmospheric pressure through a small throttling valve.

(2) A sample point should also be installed before and after the makeup system IX columns.

(3) A high pressure sample bomb should be installed in the cleanup system as shown on the flow sheet. This bomb will allow pressurized sampling for gas analysis and the addition of special chemicals should they become necessary.

All sampling points are located in a special sampling box which must be ventilated and provided with shielding to protect the operator from the possibility of an equipment failure while he is taking samples.

Loop water conductivity and pH measurements are usually made on samples. Inline instrumentation has not proved reliable in the past.

Crud filters are installed in some loops. These filters are used on either thermally hot or cold water streams.

7. Utilities

The ETR and the MTR utilities used by the water loops are listed below. A brief statement of description and purpose is given for each item.

a. Water

A summary of the water chemistry of the ETR primary cooling, HDW and LDW systems, and the MTR ECL and demineralized water systems is given below:

pH:	5.0-6.5
conductivity:	1-5 micromhos
total solids:	1-2 ppm
chloride ion conc:	< 0.1 ppm

Any sponsor with critical water chemistry problems should plan on supplying an adequate water preparation system of his own.

(1) ETR

(a) The HDW (high pressure demineralized water) system is a circulating demineralized water system intended for service where the water can be returned to the system. This system is not to be exposed to direct contamination. Its primary use is for heat exchanger secondary service and pump cooling. The water is at

200 psig and 80 to 90°F and flows in stainless steel piping.

(b) The LDW (low pressure demineralized water) system is the demineralized water supply system intended for nonreturning service such as filling the loops, flushing operations, etc. It is supplied through stainless steel piping at approximately 80 psig and 60°F.

(c) The UCW (utility cooling water) system is the cooling tower water system used as the secondary side of the ETR primary heat exchangers. This water is available to experimental facilities where the heat removal and volume capacity of the HDW system are not adequate, and where the high chloride content due to chlorination of the UCW system is permissible. This water has a hardness of approximately 1400 ppm CaCO_3 and contains dirt and foreign material. The water flows in carbon steel piping at 65 psig, 80°F maximum, and a pH of 6.0.

(2) MTR

(a) The ECL (Emergency Cooling Loop) is a portion of the reactor process water system which circulates through the ECL loop in the MTR basement. Pressure on the ECL loop is raised to 150 psig by booster pumps. This water is used for emergency coolant to the experiments, for heat exchanger secondary service, for pump coolant service, etc. Maximum temperature is 115°F.

(b) The demineralized water supply system is for nonreturning service such as filling the loops, etc. Maximum temperature is 70°F.

(c) The raw water system is essentially untreated well water. It is supplied as a nonreturnable utility water system in the MTR. Maximum temperature is 85°F.

(3) ATR

(a) Two HDW systems, similar in purpose to the ETR HDW systems, are

used at the ATR. Design conditions for these systems are 130°F/240 psig and 100°F/240 psig. The cooler system is used for the primary pump and purification cooler secondary flows. Both systems use LDW with sodium dichromate inhibitor added, and carbon steel pipes.

(b) An LDW system at design conditions of 70°F and 80 psig is similar in purpose to the ETR LDW system.

(c) A well water system at design conditions of 100°F and 125 psig is used in the cubicle space coolers and for cooling the pressurizer liquid level reference leg.

b. Electricity

(1) ETR and ATR

(a) Diesel power, 480 V, three-phase, is supplied at various locations in the reactor building. This system is supplied from continuously running diesel driven generators.

(b) Commercial power, 480 V, three-phase, is supplied at various locations in the reactor building. This power is obtained from the Northwest Power Pool.

(c) Instrument power, 480 V, three-phase, is supplied at various locations in the reactor building. This power is supplied through an automatic transfer switch from either the diesel or the commercial bus.

(2) MTR

(a) MG (motor generator) failure-free power, 480 V, three-phase, is supplied at various locations in the reactor building. This power is obtained from a battery supplied MG set which floats on the commercial power system.

(b) Diesel emergency power, 480 V, three-phase, is supplied at various locations in the reactor building. This power is obtained from a diesel driven

generator which starts automatically approximately 30 seconds after loss of commercial power.

(c) Commercial power, 480 V, three-phase, is supplied at various locations in the reactor building. This power is obtained from the Northwest Power Pool.

Experimenters are expected to supply their own transformers, circuit breakers, voltage regulators, and connections to the existing distribution points.

c. Air

Instrument and plant air at the MTR, ETR, and the ATR are all supplied from a common compressor system. The distribution headers normally run at 115 to 130 psig. The main difference between the two systems is that the instrument air is essentially failure free. If the pressure in the air storage tank, on the outlet of the compressors, drops below 115 psig, the plant air system is automatically shut off. The air remaining in the tank should then supply the required instrument air system for a number of hours.

All air is dried to a -30 to -40°F dew point.

d. Drains

The TRA reactors are equipped with a warm drain system (WD) and a hot drain system (HD).

Warm effluents contain radioactive contaminants which are within the allowable tolerance for leaching into the soil, or can be brought economically within this tolerance, by diluting or by holding for short-term decay. Waste discharged into the warm drain or the warm experiment drain (WED) systems goes directly to the retention basin. Since very little control can be exercised over this waste, all waste discharged to these systems, if not known to be uncontaminated, should

be sampled and approved by Health Physics prior to discharge.

Hot effluents contain radioactive contaminants which are above the allowable tolerance for leaching into the soil, or cannot be brought economically within this tolerance, by diluting or by holding for short term decay. The hot experimental drain system (HED) at the ETR and ATR is an integral part of the reactor hot drain (HD) system. Waste discharged into these hot systems is collected in catch tanks where it is sampled regularly by Health Physics. If the contamination level is below the permissible limits (50,000 dis/min/ml gross beta-gamma and 64 dis/min/ml alpha), it can be pumped directly to the retention basin. If the contamination level is above these limits, it is pumped to the 10,000 gallon hot waste storage tanks.

Hot effluents at the MTR can be discharged to either one of two separate systems, the hot drain (HD) or hot experimental drain (HED). Waste discharged to the MTR HD system is collected in 1500 gallon catch tanks where it is sampled and disposed of as mentioned above. Waste discharged to the MTR HED system is collected in a 600 gallon catch tank from which it can be discharged to the hot waste storage area only.

A tank truck loading facility is available for transferring waste from the hot waste storage area to ICPP.

e. Steam

Steam is supplied at the reactors at about 135 psig pressure.

f. Exhaust Air

The MTR cubicle exhaust system is estimated to provide about 1/4-inch H₂O vacuum. The ETR and ATR cubicle exhaust systems could provide more than 2 inches H₂O vacuum if the system were closed. Higher vacuum can be obtained at certain locations about the reactors.

The availability of these utilities varies. Requests for utilities should always be made as early as possible through the assigned Project Engineer.

C. DESCRIPTION OF A TYPICAL WATER LOOP

1. Discussion of the Flow Sheet

A flow sheet for a typical water loop is shown in Figure 42. The equipment and instruments shown on this drawing will provide a controlled circulating water environment for the irradiation of experiments and provide the necessary interlocks with the reactor to prevent a serious incident in the event of equipment or sample failure. Individual experiments may require additional instrumentation and equipment to achieve specific objectives.

Minimum instrumentation on a high pressure water loop includes at least two recorders with local alarm and automatic reactor power reductions on loop flow, loop pressure, and bulk water temperature. Pressurizer liquid level controls should be duplicated. Other requirements are brought out in this chapter and in other chapters dealing with experiment design.

Many details are not shown on the flow sheet. A partial list of these details would include: (a) the installation of double blocks and a bleed valve on all hydraulic tubing lines to equipment which require maintenance during operation; (b) the installation of "locally" mounted instruments inside closed, ventilated cabinets outside the main cubicle; (c) the installation of reach rods on all valves necessary for complete loop operation; and (d) the location of the sampling system in a closed ventilated cabinet. There are many other items, particularly instrumentation and interlocks, which can be covered only by detailed study of a given loop design. This study of the design con-

stitutes a major portion of the Project Engineer's effort on a loop.

The loop shown on the flow sheet is classed as a high-pressure, high-temperature, recirculating water loop with a through flow, single pass in-pile tube. There are several types of loop systems, some of which are described briefly below.

2. Types of Loop Systems

(a) The once through system, or the nonrecirculating system, is usually a low-pressure, low-temperature system using plant water. The instrumentation will be similar to that on the flow sheet. The major equipment differences would be the use of the primary pump head for pressurizing the system and a gas pressurized surge tank on the pump discharge for emergency flow of short duration. The outlet must be equipped with a diversion valve and a large catch tank for receiving contaminated water in the event of a fission break.

(b) The recirculating one-channel system is shown in the flow sheet.

(c) The recirculating multichannel system usually has two separate pressure tubes forming the in-pile system. A flow control valve can be installed in each channel. Another method is to control total loop flow and split the flow with a three-way valve or with flow restrictors. Each channel must have at least one flow element.

Several flow channels in one pressure tube have also been used. This arrangement presents several disadvantages, among them (a) difficulty in connecting flow elements which must be located inside the in-pile tube and (b) pressure tube wall thicknesses which present serious thermal stress and neutron loss problems.

A description of all types of in-pile pressure tubes is given in Chapter V and in Figure 18.

D. BASIC LOOP OPERATION

Operating manuals are issued for each loop prior to operator training on specific installations. This section will, therefore, deal with the general principles of control and operation applicable to high pressure water loops.

1. Filling and Venting

Prior to the operation of any water loop, it is necessary to fill the system and properly vent entrapped air. The venting operation consists of opening each vent valve and allowing water to flow until there is no discharge of air. After the filling and venting is completed, the primary pumps are run for a short period, and the loop is revented. This procedure is repeated until no air is obtained during venting.

Usually, demineralized water is used to fill the loop, and proper chemistry conditions are obtained before reactor start-up by chemical additions. Water from a chemically treated system can be used for loop makeup during reactor operation.

The steps in performing the filling and venting operation are listed below:

(a) Check position of all loop valves. This includes ascertaining that the loop drain valves are closed and the in-pile tube and pressurizer isolation valves are open. Valve in all loop flow systems and equipment even though these components may not be used during the next cycle.

(b) Open several main loop vents. Opening all vents is usually undesirable because so much water flows out that drain pans, etc., are overflowed before the vents can be closed.

(c) Valve in demineralized water to the loop; and if necessary, start the fill pump.

(d) After the loop has filled to overflowing, carefully vent the entire system. Water should be obtained from all loop vents.

(e) Run each pump for about 5 seconds (jogging) and then vent each pump. Some general rules can be set forth for the operation of canned rotor pumps. These rules cannot be considered to override instructions set forth in the operating manual for each loop or specific instructions from the Project Engineer. In the absence of more specific instructions, operation within these rules should prevent serious pump damage.

(1) The absolute minimum suction pressure for jogging is 50 psig.

(2) The pump must never be run without secondary cooling water flow. Some loops have interlocks to prevent running the pump without the cooling water flow. If the pump should ever be operated without cooling water flow, stop the pump and establish cooling water flow for about 5 min/min of pump operation without cooling water flow before re-starting the pump.

(3) Never vent a pump while it is running. This will upset the thrust balances and cause bearing damage.

(4) Avoid venting when the loop temperature is above 200°F. If venting is necessary at system temperatures above 200°F, vent less than 0.1 gallon over a 10 minute period. (This is equivalent to approximately 35 cc/min for 10 minutes.) Wait 30 minutes between ventings to allow bearing and motor temperatures to stabilize.

(5) At loop temperatures below 200°F, vent any quantity necessary, but keep the venting rate less than 35 cc/min. Higher venting rates can cause the rotor to "jump" and damage the bearings.

(6) At least 50 psig is required to perform a successful venting operation; however, never vent a pump with loop pressure greater than 500 psig. This would be an unnecessary safety hazard.

(f) Run one or more pumps to obtain normal flow circulation for 2 to 3 minutes. Then repeat the venting process on the entire loop. During this relatively long time of pump operation (compared to jogging), it may be necessary to raise the loop pressure (hydrostatically) above demineralized water pressure.

(g) Report all incidents of dirty vent water, higher than usual pump currents and fluctuating pump currents. These are the usual signs of bearing deterioration.

2. Pressure Testing

Two types of pressure tests are performed on the MTR-ETR water loops. These are the hydrostatic or proof test and the leak test.

a. Hydrostatic Test

The hydrostatic proof test is used to check the strength of new components added to the system. This test is usually conducted at 150 percent of design pressure. As a general rule, all new equipment is proof tested; however, small tubing additions, etc., on the low pressure side of root valves will not always be proof tested.

The steps in performing a proof test are listed below (assuming a liquid-full loop).

(1) Invalidate the mechanical pressure safeties on the loop. The preferred method is by puncturing or by-passing rupture discs and gagging relief valves. It is a major responsibility of every person associated with the test to see that the safeties have been invalidated before the proof test and reinstated after the test.

(2) All instrumentation which would be over-ranged by the test is

valved out. The specific loop operating manual should specify which instrumentation should be valved out.

(3) Hydrostatic pressure is raised to the desired level by means of the makeup pumps. If the desired pressure is not obtainable, a leak may be present or all the air may not have been removed during venting.

(4) An inspection is made of all new components and welds. This inspection may be made at the desired pressure or at a lower level.

(5) Leaks in welds or structural material must be repaired and the test repeated. Leaks in flanges, valve packing, etc., should be located because of a desire to maintain as tight a system as possible.

(6) All loop mechanical safeties and instrumentation are reinstated.

b. Leak Test

The leak test is used to check the tightness of the loop. The loop must be sufficiently tight to operate throughout the next reactor cycle without major leak problems or undue hazard. This test is usually conducted at 110 percent of operating pressure. The leak test should always be the last operation before pressurizer liquid level is established. All loop systems and instrumentation are usually valved in during the leak test.

The acceptable leak rate should be specified with the request for the test. The usual leak rates vary from 200 to 500 psig/hr.

3. Establishing Steam Dome and Liquid Level in the Pressurizer

There are three ways of establishing a steam dome and proper liquid level in a pressurizer:

(a) The usual method is to start with a liquid full pressurizer, turn on the heaters and bleed water out the

bottom of the pressurizer as the pressure builds up. This is a slow process and requires constant operator attention until an appreciable steam dome is established.

(b) This technique is the same as (a) except the bleeding off is done at the top of the pressurizer. This method is slower because so much heat is lost in the steam being vented off the top of the pressurizer.

(c) A nitrogen or helium blanket is applied to the top of the pressurizer to obtain the desired liquid level and pressure before starting the pressurizer heaters. This would allow almost immediate operation of the loop with the reactor at power. As the pressurizer heats up, it will be necessary to vent the gas blanket to avoid high pressures. This technique is not generally used because of fear of increased gas concentrations in the loop water and should not be used without specific approval.

Before turning on the pressurizer heaters or draining the pressurizer level near the heater level, the operator should personally fill the standing leg on the liquid level D-P cells. This is an operation which cannot be repeated too often.

4. Depressurizing

Depressurizing of loops causes a good deal of lost reactor operating time. Since circulation must be maintained on most loops for an appreciable time after reactor shutdown and since 100 to 500 psig system pressure must be maintained to continue pump operation, the time required

to depressurize the loop to atmospheric pressure is lost to the shutdown. Three methods are outlined below:

(a) One technique is to use the degasser line to reduce the loop pressure to the value where the main pumps must be shut off and then alternately filling the surge tank with the makeup pumps and draining until the residual heat in the pressurizer and its heaters has been removed.

(b) It is possible to greatly reduce the depressurization time by operating the makeup pumps and degassing, balancing the two so that a liquid full pressurizer is obtained before the loop pressure is reduced below the pump cutoff point. Then by maintaining the hydrostatic pressure with the makeup pump, circulation through the degasser can be continued until the entire pressurizer is cooled down to the loop temperature. This technique requires constant operator surveillance; but on most loops, it will cut the depressurization time in half.

(c) The time can also be cut by applying a helium or nitrogen blanket to the pressurizer and continuing to run the loop pumps and the degasser until the entire system is cooled down. This technique is not generally used because of the possibility of increased gas concentrations in the loop water.

An excellent innovation is the installation of a double block between the loop and the pressurizer which allows depressurizing the loop without losing the steam dome in the pressurizer.

CHAPTER XII

GAS LOOPS AND GAS LEADS

A. PURPOSE OF GAS LOOPS

(See Figure 43)

Information on radiation damage, cladding integrity, oxidation resistance, structural strength, fuel-matrix stability, and other properties may be obtained to a limited extent in small capsule-type tests. After this preliminary development work has been done, it becomes necessary to do pilot plant scale tests before the building and testing of a complete reactor is performed. The pilot plant tests for components of gas cooled reactors take the form of test loops in high neutron flux test reactors, such as the MTR and ETR.

In these test loops, samples may often be tested to failure to a degree which would not be possible in a full-scale reactor. With the information gained, the final reactor design will be more sophisticated than if the reactor were designed directly from nonnuclear tests and capsule irradiations.

1. Single Pass Air Loops

When air cooled reactor components are being tested, a single pass air loop should provide the proper test conditions. Air from the atmosphere is compressed, heated, passed through or by the test specimen, and then returned to the atmosphere through an exhaust stack.

a. Advantages

In a single pass air loop, air pump seals need not be leakproof since the air pumps do not handle radioactive gases. Very large quantities of air may be handled with standard design compressors. Blowers, axial compressors, or reciprocating piston compressors may be used.

A wide margin of control over pressures and flows may be obtained with single-pass air loops. Closely controlled inlet air temperatures may also be obtained.

A large quantity of air may be safely stored in receivers. Emergency air flow for afterheat removal may be obtained easily during power outages, compressor failures, or reactor shutdowns.

b. Disadvantages

In single pass gas loops, only air may be used since use of other gases would be too expensive.

Sample failures often make the exit air radioactivity quite high. When the loop gas is exhausted directly to the atmosphere, large delay tanks and special filters may be required to reduce the amount of radioactive material released to the atmosphere. Tests may be postponed until favorable weather conditions exist.

The energy consumed in compressing from atmospheric to operating pressure is wasted since the air is exhausted to the atmosphere after flowing through the loop.

A single pass air loop usually needs provisions to heat the inlet air.

The high temperature loop exhaust air must be cooled as soon as possible after passing specimen area to minimize temperature stresses in the exit piping, on the delay tanks and filters, and on the stack. Cooling may be accomplished either with heat exchangers or with water injection.

2. Closed Cycle Gas Loops

Fig. 43 Typical gas loop flow diagram.

Several gases have certain advantages over air for use as a loop coolant. Hydrogen and helium have very good thermal conductivities and have been used as loop coolants. Carbon dioxide has a very high heat capacity and fair thermal conductivity. The low oxidizing or corrosion effect of some of these gases on modern metals also suggests their use as a loop coolant.

The high cost of these gases calls for the design of a closed-cycle or recirculating-gas test loop. In a closed-cycle test facility, the gas is continuously recirculated through the loop and is not exhausted except to lower the loop pressure.

a. Advantages

A recirculated gas test loop may use any convenient gas or vapor, including high pressure water vapor. Design of nuclear super-heaters for boiling water reactors could use information obtained in closed-cycle test loops.

All gas passed through the loop need not be heated from ambient temperature to operating temperature by external heaters. Only loop piping heat losses must be replaced. With a well-insulated loop, fueled test specimens may be used to heat the gas to operating temperature. For ease of control and to compensate for high heat losses, a closed-cycle gas loop will usually contain a heat exchanger and heater, although they will be of smaller size than those required in a single pass gas loop.

A big advantage of the closed-cycle gas test loop is that any radioactive particles or fission products are contained in the loop. The piping is small and well shielded. If a test specimen fails it will not be necessary to shut down the reactor because of unfavorable weather conditions.

Another advantage of this type of test facility is that the smaller pumps require

less power to circulate the gas. The pressure increase across the pumps is smaller than in the air loop compressors because the closed-cycle pumps operate at loop pressure on the suction side. Only the loop pressure drop must be supplied by the closed-cycle pumps.

b. Disadvantages

Probably the greatest disadvantage of this type of loop is the compressor or blower requirements. No foreign materials such as oil may be introduced into the gas stream, as the concentration of the foreign material would continue to build up as the gas is recirculated.

Special cleanup systems are required in most closed gas loops containing inert gases. A typical system will include the following features:

- (1) A filter to remove particulate matter.
- (2) An oxidizer to convert hydrogen, hydrocarbons, carbon monoxide, or the like.
- (3) Molecular sieve to remove water, carbon dioxide, or any molecules which are any larger.
- (4) Charcoal traps to remove such elemental impurities as nitrogen, argon, or xenon.

Gas leakage must be held as low as possible.

Pressure fluctuations caused by compressors must be damped out.

As a rule, compressors for this service are so specialized that they must be built to suit the needs of the loop being designed.

Reserve flow in case of compressor failure must be provided.

The loop volume must be kept to a minimum for the following reasons: (a) minimize shielding requirements, (b) keep heat losses low, (c) achieve quick temperature response, and (d) minimize the volume of gas used.

3. Gas Cooled Leads

In recent years, special experiments have been designed which incorporate features of loops and of simple lead experiments. They are still called leads because the entire in-tank assembly is removed from the reactor when the test is complete.

Air-cooled leads are essentially the same as air-cooled loops. They have the following advantages over loops:

Capital investment is lower, since the air volume to be handled is much lower. Operating costs are also lower.

No external preheat is required, since preheat can be extracted out of the exhaust stream.

No external cooling for the air is required, since the total amount of heat to be removed is relatively low.

Disadvantages are as follows:

Lack of temperature control. Temperature can be varied only through relatively narrow limits. Neutron flux changes are much more significant in temperature control.

Dimensional tolerances must be very low. Small fabrication errors often result in poor experiments.

Air impurity tolerance is lower than it is for air-cooled loops. Small amounts of lubricating oil, dirt, and even fibers from thermocouple insulation will plug air passages.

Experiment power must be lower because the ability to remove heat is reduced.

Conduction-cooled leads are analogous to closed cycle loops. Gas is purged through conduction-cooled leads at a very low rate, in the range of 100 cubic centimeters per minute or one cubic foot per hour.

Heat is transferred by convection into the reactor water rather than being conducted away by the gas. Sample temperature is varied by varying gas composition, which changes heat transfer characteristics. Gas used in these experiments is lost; but unless a very expensive gas is used, gas costs are very low.

Normally, gases used in convection-cooled leads are helium, nitrogen, and air. Occasionally, argon is used instead of nitrogen or air when an inert atmosphere is required. Neon has also been used.

Helium has a high heat transfer coefficient and is nearly always used as one of the gases. Nitrogen, air, and argon all have about the same heat transfer coefficient; it is much lower than the heat transfer coefficient for helium. Conduction-cooled leads are designed to require a gas heat transfer coefficient lower than helium and higher than that of air, nitrogen, or argon. Helium and air, nitrogen, or argon are then mixed so that the desired heat transfer coefficient can be obtained. The heat transfer coefficient for neon is between the extremes represented by helium or air, argon, and nitrogen. It is used for a very slight temperature modification in conjunction with two other gases.

Specimens irradiated in this type of experiment can be clad or unclad. Factors which will affect decision to clad specimens are:

- (1) Normal specimen operating temperatures
- (2) Maximum possible specimen temperature

- (3) Chemical properties of specimens and gases used.

B. TYPES OF TESTS

Many different materials must be tested and developed before a final reactor design is completed. Fueled specimens, moderators, control rod materials, and structural materials are included in gas loop and lead tests.

1. Fueled Samples

Fueled samples come in many forms, shapes, and sizes. The fuel may be distributed evenly throughout the test material or be concentrated in a fuel matrix. Tests in nuclear reactors are made to determine special nuclear characteristics such as internal heat generation and transfer, fuel matrix swelling, cladding oxidation or separation and material changes due to neutron capture, scattering, and atomic transformation. Neutron flux perturbation and uneven power generation may also be studied.

a. Plate Designs

Many test specimens have been based upon a flat plate design, using parallel flat plates or concentric cylinder shapes. The materials usually involve a metallic cladding over either a metallic or ceramic fuel matrix. Since the use of gas as a coolant is usually considered for very high temperature reactors, greatest effort has been spent in the development of high temperature materials.

At high temperatures, fuel element oxidation, swelling, and warping are serious. Work has been done with fuel matrices using metallic uranium, UO_2 and U_3O_8 , in many different metals. U_3O_8 has the advantage of being more stable against oxidation. Plutonium fuels have also been made, but plutonium technology is not as highly developed as that of uranium.

A metallic cladding must be developed with high thermal conductivity, low oxidation and corrosion rates, high strength, and the ability to bond readily with the fuel matrix. Until recent years, promising materials in large quantities and of high purity were not available. Special materials are now being developed.

A large amount of nonnuclear work goes into the design and testing of structural materials. Large hot-gas facilities may be used to test the fuel element structure at design temperature and flow before a sample is tested in a nuclear reactor.

Due to poor heat transfer characteristics, most flat plate fuel element designs develop as large a heat transfer surface area as possible consistent with adequate structural strength.

Streamline flow along a flat plate allows low pressure drops through the reactor, but results in a high differential temperature between the fuel plate surface and the bulk gas temperature. To obtain a high efficiency reactor, the temperature difference between the fuel plate and the bulk gas must be kept low. The reactor gas outlet temperature is limited by the maximum allowable temperature at the fuel element hot spot.

b. Special Metallic Fueled Samples

Because of the poor heat transfer characteristics and structural strength of flat plate designs, many other forms have been designed. Some of these designs have been in the form of twisted ribbons, corrugated plates, and wire screens. All of these designs have three disadvantages: (a) they are hard to fabricate, (b) they offer a high pressure drop, and (c) they are hard to design into a reactor core.

c. Nonmetallic Fueled Samples

Fueled graphite samples have been operated at 4000°F . Graphite oxidizes

very rapidly at high temperatures in air, but has been used in carbon dioxide gas for high temperature reactors and in hydrogen for nuclear rocket work.

One disadvantage of ceramic and graphite material is their relatively low thermal shock resistance. Thermal conductivity is also lower than in metallic materials. With nonmetallic fuel elements, the moderator can be mixed with the fuel or used for the fuel structure. Graphite is a good moderator; it can be used for structural supports and can be machined. Beryllium oxide ceramics have been used in rocket propulsion reactors.

Since fission product release is high in ceramic fuels at elevated temperatures, a large amount of work has been done to develop cladding materials. The cladding must be stable, have good heat transfer characteristics, and bond readily to the fueled matrix.

2. Nonfueled Samples

The fuel element is only a small part of the finished reactor. Reactor structural parts, moderator sections, and control rods must be irradiated to determine material stability under reactor operating conditions. Sometimes combined tests are run with both moderator and fueled sections in one assembly. Complicated shapes offer difficult heat transfer analysis under transient conditions. These are measured in reactor tests.

C. SPECIAL TEST FEATURES

Gaseous coolants, which are compressible and usually operate at high temperatures, have many unusual properties. The design of test loops and samples must take these properties into account.

1. Thermodynamics

Thermodynamics is concerned with the transformation of energy from one form to another, with the availability

of energy for doing useful work, and with the stability of equilibrium among chemical substances. Of the many phases of thermodynamics, properties of flowing compressible fluids and heat transfer are of special interest in gas cooled loops.

a. Mach Number

An important thermodynamic relationship that must continually be monitored and controlled in gas loop tests is the Mach number. The sample Mach number should not be allowed to exceed a certain value, usually about 0.4 Mach; otherwise, local velocities may attain sonic speeds with resultant large local forces that may damage the structural members of the sample.

The Mach number for an ideal gas is given by

$$M_a^2 = \frac{v^2}{g_c P V} \quad \text{or} \quad M_a = \frac{v}{v_c} \quad (12-1)$$

where

M_a = Mach number

v = Fluid velocity

v_c = Acoustic or critical velocity

k = Ratio of heat capacities C_p/C_v

g_c = Dimensional conversion factor

P = Absolute pressure

V = Specific volume of gas.

Changing the gas temperature, pressure, or flow rate will change the Mach number.

b. Dynamic Pressures

On some fuel elements cooled by air, the quantity used to monitor integrity of the element is $\Delta P/q$ where ΔP is a differential pressure drop across the element or some part of it. Dynamic

head, q , is a function of mass flow, pressure, and coolant temperature given by

$$q = \frac{KW^2 T}{P} \quad (12-2)$$

where

q = Dynamic head, psi

W = Mass flow, lb/sec

T = Coolant absolute temperature, °R

P = Absolute pressure of coolant, psia

K = Constant depending upon element design.

$\Delta P/q$ is constant with small variations in coolant flow, temperature, and pressure; and varies only when the fuel element distorts enough to change the flow pattern.

Dynamic head is always an important parameter for measuring stress in the testing facility. Some element designs are such that sonic velocities or choking occur under all possible test conditions before dynamic head is large enough to damage the sample. Dynamic head is usually not a control function for ceramic tests, although very large forces may be built up on the test specimen at low pressures, high temperatures, and high flow rates.

A study of the dynamic head relationship will reveal the complex parameter interdependencies that must be evaluated before an experiment can be run in a reactor. It is imperative that the parameters be held within established limits throughout a test. Upper limit stops on flow valves and high and low alarms on pressure temperature and flow are designed to control dynamic head.

c. Heat Capacity

The heat capacity and heat conductivity of gases are much lower than for

liquids and solids, so it is necessary to use much larger quantities of gases than liquids to remove the same amount of heat. Some gases have as much as twice the heat capacity of other gases. Heat capacity for gases increases with increases in temperature and pressure. Neglecting the correction for pressure changes, the difference between the enthalpy at one temperature and a higher temperature will correspond to the total heat absorbed by the gas per unit mass of flowing gas.

d. Heat Transfer

Heat transfer analyses in gas systems are usually more involved than in liquid systems because radiation, convection, and conduction all play an important part.

In a gas-cooled reactor the specific power generation is often quite high. The temperature difference between the fuel element and the bulk stream is high due to the low thermal conductivity of the gas coolant. Very high fluid velocities are used to keep the heat transfer rate high.

Because of the low heat capacity of gases, a sample temperature may rise very rapidly in a short time upon loss of flow. Fortunately, because of the compressibility of gases, flow does not usually stop suddenly. As sample temperature and gas temperature increase, no burnout point is reached where heat transfer at constant ΔT falls off. In fact, heat transfer to gases improves as the temperature rises because of the lower viscosity and higher heat capacity.

Radiant heat transfer often plays a part in heat removal. Some gases, such as carbon dioxide, have a much higher emissivity than others and are better gas coolants for this reason. Radiant heat transfer is a function of the fourth power of the absolute temperature.

Fuel element design affects heat transfer to a great extent. To give good

heat transfer, the gas coolant must be continuously mixed so that the colder gas contacts the test specimen or mixes with the hot gas. A large surface area-volume ratio is required to give good heat transfer, although structural strength may suffer. In most fuel assembly designs, the structural load members are designed to run as cool as possible compared to the fueled section. This may require separate flow channels to cool the structural supports.

2. Fission Product Release

Large, gas-cooled test loops are designed to operate with fission product or sample radioactive product release within the loop.

a. Shielding

The equipment must not only be designed for high pressure and high temperature conditions, but must be shielded for protection of personnel against radiation. Protection from gamma, beta, and delayed neutron radiations must be provided. The high coolant velocities in gas test loops result in only a few seconds delay between the time the gas passes through the sample and the time it enters the out-of-pile equipment. This problem is not severe with gas leads.

Shielding materials are selective in their ability to stop radiation. Lead, uranium, and other heavy materials are good materials for the attenuation of gamma radiation. The heavy materials produce bremsstrahlung, a secondary gamma, when attenuating beta radiation.

b. Containment

The body tolerance for fission products is very low. Even a very small leak in the exit piping of a loop may be intolerable. Loop design must call for reliable valves and flanges. High temperature metal O-rings are usually used on exit piping.

c. Filters

To keep the radioactive solids content of the loop system or coolant discharge low, filters of special design are often used. The filters are designed for high temperature, low pressure drop, and very fine particle removal at high efficiency. Filters have been designed to remove 99.98 percent of all particles above three microns.

Charcoal filters have been used for fission product filtration, but have several drawbacks. Charcoal must be used at low temperatures in order to operate efficiently. Activated charcoal is selective and holds up gases in order of their molecular weight. Activated charcoal will ignite spontaneously due to many causes. A very serious fire was experienced at MTR in a charcoal filter when pressure, temperature, and particle size were such as to allow spontaneous combustion.

d. Release of Radioactive Gases

Single pass loops release the cooling air to atmosphere continuously. To keep the gas activity levels within tolerance, several methods are used. The gas may be delayed in large holdup tanks after the solids are filtered out. The inert gases, xenon and krypton, decay to their daughter products. These daughter solids are then filtered from the air stream. Iodine reacts with silver on the filters to form silver iodide. The remaining gas is then released to the atmosphere through the stack.

e. Weather Controlled Tests

Even after gas holdup and filtering, the radioactive release rate may be too high for release under some weather conditions. When tests are being run to test sample cladding failure or with some ceramic samples, continuous control of the test during unfavorable weather conditions is required. Fortunately, wind conditions may be forecast with great accuracy at the MTR-ETR area.

3. Coastdown

Pump and coolant failure studies are made on all engineered loop tests regardless of coolant type. Coolant failure may be caused by a power outage, pump failure, control valve failure, or instrument failure.

a. Loss of Pumps

In single pass air loops, several minutes' air supply is stored in large air receivers. Emergency blowers are usually installed to provide afterheat cooling for use after air in the receivers is exhausted. Cooling is often required for as long as eight hours after the reactor is shut down.

Fluid coastdown curves (see Figure 44) are calculated and, after instrumentation is installed, measured with high speed recorders. The required reactor power reduction is then determined from a study of heat generation, temperature response, and coolant flow curves.

Fig. 44 Air loop coastdown curve.

b. Control Valve Failure

The flow controller or flow control valve may fail. Main flow control valves are "air-to-close" type and bypass valves or bleed-off valves are "air-to-open" type. Mechanical stops are used on the control valves to limit the valve movement even though the valve stem might break. These stops are also used to keep the coolant flow within limits to control Mach number and dynamic head. In order to be useful, these stops must be set for every test.

c. Afterheat Transfer

Because of the high sample power generation, low sample mass, and low thermal conductivity of the gas coolant, afterheat removal becomes a problem. The afterheat removal is usually accomplished with reduced flow conditions. The emergency blowers may be used to supply the coolant. Even after the reactor is scrammed, some flow is usually required through the in-pile tube, although heat loss through the in-pile tube walls may be enough to keep the sample from melting. High temperatures may be experienced several hours after the reactor is shut down if coolant flow is not maintained.

d. Emergency Blowers

Both the sample and in-pile tube must be cooled after the reactor is shut down. Emergency blowers with failure-free or gasoline-driven power supplies are usually required in the loop installation. These blowers must be large enough to cool the sample and in-pile tube immediately after the reactor is shut down and for several hours thereafter.

D. LOOP EQUIPMENT DESIGN FEATURES

1. Blower Requirements

Pump capacity must be large enough to handle the maximum test requirements under both normal and emergency

conditions. Standby pumping capacity is needed to permit operation if one blower fails or if an emergency supply of air is needed to handle the heat if the pumping power is interrupted. On closed-cycle gas-cooled loops, the static pressure is supplied by bottled gas. The pump or blower is used only for supplying the required head to circulate the required gas. The open-cycle air loops air compressor must supply the static pressure needed for experimental operation as well as the pressure drop due to line resistance. Reciprocating-type compressors are usually used. They have high capacity and high pressure output and are quite reliable.

These compressors are not normally operated at full load during experimental testing. The output of the compressor is controlled by load steps and clearance pockets which cause fluctuations in output pressure. Pressure controlled bleed valves which release a portion of the air to atmosphere have been installed on the MTR and ETR Clark compressors to keep them fully loaded regardless of loop requirements. Pressure fluctuations and wear on the machine are minimized.

The reciprocating compressors are normally oil lubricated. This oil must be removed from the air stream before the air is heated in order to prevent fires and coking problems in the experimental system.

In some tests, humidity must be controlled. The usual procedure is to dry the air to a low dew point and inject just enough water to achieve the proper humidity.

2. Temperature Control

In closed loops, the temperature is controlled by three separate pieces of equipment: a regenerative heat exchanger, a waste heat exchanger, and line heater. The regenerative heat exchanger permits operation of the loop equipment at temperatures much lower than the sample

coolant temperature. Gas returning to the loop from the sample is cooled by the air going to the sample. A waste heat exchanger which removes excess heat from the stream and a line heater for adding heat to the stream, if necessary, are usually included in the closed loops.

The open cycle loops require large heaters capable of heating air at atmospheric conditions to 1200°F. The ETR oil-fired air preheaters are capable of preheating 24 lb/sec to 1200°F.

3. Out-of-Pile Piping

The high temperature and pressure requirements in the experiments dictate the use of stainless steel on most loop piping, and the expansion problems caused by the operating temperature range from ambient to 1200°F are very serious. To minimize these problems on the open cycle loops, air is cooled by injection of water as soon as it leaves the in-pile tube. In the high temperature areas, both expansion loops and expansion joints are used to take up pipe expansion.

If an inlet air line ruptured, the test sample would probably melt, releasing large quantities of fission products. The greatest danger to personnel, however, would be the sudden release of high temperature air into the occupied areas. In-pile tubes are usually designed to be cooled by reactor coolant in order to minimize the operating temperature.

4. Instrumentation

Commercial instrumentation is used wherever possible. Temperature measurement is usually accomplished with thermocouples and millivolt recorders. Some systems use data reduction facilities which log the test variables on data sheets at predetermined times.

Because of the rapid response of sample temperatures to variations in gas flow and pressure, quick instrument response is needed for control and safety

circuits. Electrical or electronic transmission of flow, temperature, and pressure signals is required on some experiment systems.

Reactor safety circuits and automatic control actions on gas loops usually are similar to other type testing facilities; however, special cases may call for special functions such as bypass flow around exit air filters and delay tanks. Because of the pressure expansion action of compressible gases, it may be possible

to get reverse flow if the system is not designed properly.

Compressible fluid control offers instrumentation some unique problems. The pressure shock of the flowing fluid is not felt in all parts of the system at once. A relatively long time may elapse before the flow and pressure stabilizes throughout the system when a control valve moves. Stability of control instrumentation is, therefore, very important.

CHAPTER XIII

LIQUID METALS AND FUSED SALTS

A. INTRODUCTION

Some experimental systems use molten metal, liquid metal, or fused salt as a coolant. These are predecessors of advanced reactors for power production, especially fast breeder reactors.

NaK (a sodium-potassium mixture which is a liquid at room temperature), molten sodium, molten bismuth, and molten lithium salts have been used in experimental systems in the MTR and ETR. Bismuth and lithium salts are mixed with fuel to form homogeneous mixtures. Sodium and NaK are normally used to cool clad fuel elements.

The primary advantages of these systems are high coolant temperature with low pressure, high thermal conductivity, high specific heat, and low density.

Major disadvantages of these coolants are a relatively high freezing point, high induced radioactivity, and high corrosion rates with most metals.

B. ENGINEERING CONSIDERATIONS

1. Introduction

In power reactor design, high coolant temperature and low coolant pressure are very desirable since the efficiency of converting heat to electricity is greater at high temperatures. The cost of pressure vessels, piping, and associated equipment is drastically reduced if pressures can be kept to a few hundred lb/in.² or less.

Liquid metals permit operation under these conditions. In addition, they have high thermal conductivity, high specific heat, and low density. These attractive features are balanced by a multitude of disadvantages and many unknown or poorly

defined properties which can be overcome only by a large amount of research.

2. Engineering Problems Peculiar to Liquid Metal Loops

Large experimental facilities designed for the testing of liquid metals and fused salts as reactor coolants have the following common engineering problems and design features.

(a) The coolant becomes high radioactive; and in many cases, the fuel is mixed homogeneously with the coolant. The resulting radiation fields prevent direct maintenance on loop components.

(b) The coolant mixtures are very expensive.

(c) One prime experiment variable is the rate of corrosion of the various loop containment materials so analysis of the coolant and piping prior to and following irradiation is necessary for performance evaluation. Sampling during the tests would be desirable, but other considerations usually prevent sampling.

(d) All liquid metal coolant react violently with water and air so they must be completely isolated from their surroundings by at least a double enclosure.

(e) Heat losses must be kept to a minimum. High operating temperatures (usually over 1000°F) aggravate the heat loss problem and dictate very compact systems.

(f) The experiment equipment inside the reactor is enclosed in a sealed container to isolate it from the reactor water. The space between the two enclosures is filled with an inert gas, which acts as insulation. The outer container also acts as an enclosure for the loop

coolant if any part of the primary equipment fails.

If the inert gas is purged through the annulus, the exit gas can be monitored to detect leaks. Other systems use electrode type leak detectors which will detect either water or liquid metal in the annulus. If either liquid metal or water is detected in the annulus, the test is terminated.

(g) Pumps may be centrifugal or electromagnetic. Centrifugal pumps handle the fluid efficiently, but the moving parts and seals are subject to failure. Electromagnetic pumps have no moving parts, no driving motor, and allow flow control by simply varying voltage to the pumps; however, they are less efficient than centrifugal pumps.

(h) Flow control, if incorporated, is usually accomplished by varying pump speed. Control valves, with their moving parts, large size, and high pressure drops, are usually avoided.

(i) All piping, pressure vessels, and in-pile tubes must withstand large thermal shocks. In some cases, the liquid metal expands on freezing, introducing extremely high pressure stresses.

(j) On all fluids except NaK, all piping, vessels, and pumps must be enclosed in electrical heaters. These are necessary in order to melt the coolant before nuclear power is generated. Electrical heater power requirements often exceed the nuclear power of the experiment if the system is large and heat losses are high.

(k) Handling of the system before, during, and after irradiation demands special consideration due to the induced radiation. Small physical size is a great advantage.

(l) No unusual problems are encountered in instrumentation. Both thermocouples and resistance bulbs are used for temperature measurement. Pressure

and differential pressure measurements are made by standard null-force balance transmitters. Plugging of sensing lines with impurities is a common problem.

(m) Electromagnetic flowmeters on liquid metal systems have many advantages over orifices or other differential pressure systems. Where flow measurement is necessary, the electromagnetic flowmeter is recommended. In many loops, variable pump output controls flow. Pump current, voltage, or speed can be calibrated in terms of flow rate, eliminating the need for conventional flow metering.

C. LIQUID METAL PROPERTIES

1. Sodium

(a) Melting point, 208°F.

(b) Boiling point at atmospheric pressure, 1618°F.

(c) Reacts vigorously with water.

(d) Burns vigorously in air, but this is inhibited by the oxide film.

(e) Sodium-23 has a 0.53 barn thermal neutron absorption cross section.

(f) Sodium-24 decays with a $T_{1/2}$ of 15 hours.

(g) Density and viscosity are similar to water, and specific heat is 1/3 that of water.

2. NaK

(a) The eutectic is composed of 77.2 percent potassium by weight.

(b) Melting point is 10°F.

(c) Boiling point is 1450°F.

(d) Reacts vigorously with water.

(e) Burns vigorously in air, but this is inhibited by the oxide film.

(f) Sodium-24 is the principal radioactive component.

(g) Physical properties are generally slightly less than sodium.

3. Bismuth

(a) Melting point is 520°F.

(b) Boiling point is 2691°F.

(c) As a liquid it will react with water to form the hydroxide.

(d) Oxidizes in air in the liquid state.

(e) Bismuth-209 has a 0.015 barn thermal neutron absorption cross section and a 9 barn scattering cross section.

(f) Bismuth-210 has a $T_{1/2}$ of 5.0 days, emitting a 1.17 MeV beta particle.

(g) Polonium-210 has a $T_{1/2}$ of 138 days, emitting a 5.9 MeV alpha particle.

(h) Specific gravity is 9.5, about the same as lead.

4. Other Materials

Liquid lithium, mercury, and magnesium have been considered as coolants for nuclear systems. If preliminary investigations indicate one or more of these materials might be satisfactory coolants, experimental systems may be installed at TRA.

A mixture of lithium, beryllium, and uranium fluoride salts has been successfully operated in a loop at MTR.

coefficient of NaK at 1600°F is 2-1/2 times as high for a given velocity as water at 600°F.

The thermal neutron absorption cross section of 0.53 barns for sodium-23 and 1.9 barns for potassium-39 is not excessive. The scattering cross section for fast neutrons is ≈ 3 barns.

Sodium-24, the prominent radioactive activation isotope, has a $T_{1/2}$ of 15 hours, decaying with emission of 1.39 MeV betas and 1.37 or 2.755 MeV gammas. Other isotopes of sodium and potassium are relatively unimportant.

The NaK system must be extremely clean. Oxygen contents above 100 ppm markedly accelerate corrosion. Water, hydrocarbons, and carbon dioxide also increase corrosion markedly. Nitrogen, argon, and helium can be used as cover gases on NaK. Helium is preferred in reactor systems.

For the reaction of water and NaK, reaction rate is almost instantaneous, total reaction time being dependent only on the rate of mixing. The reaction of sodium and water produces 31,000 to 34,000 cal/g-mole.

Austenitic stainless steels successfully resist corrosion by sodium or NaK, but are subject to corrosion by impurities. Pressure stresses are not a serious problem, but the vessels and piping are subjected to severe thermal stress cycling because of high operating temperatures. Expansion and insulation of process piping must be very carefully considered because of the high operating temperature.

D. NaK LOOPS

1. NaK Characteristics

Boiling point and melting point allow a range of operating conditions from room temperature to 1600°F at essentially atmospheric pressure. Heat transfer properties are excellent. The heat transfer

2. Typical NaK Loop

A package loop could occupy one of the outer test facilities in the ETR or ATR. The core section would be a re-entrant loop with the heat exchanger and pump located a few feet above the core. This design has many advantages: (a) NaK volume is kept to a minimum;

(b) out-of-pile equipment is mostly eliminated, avoiding very serious shielding, maintenance, and access problems in the surrounding areas; and (c) if the fuel ruptures, no special handling or decontamination is necessary, since the fission products are contained. Operation of high risk experiments would, therefore, be more feasible than with loops containing out-of-pile equipment.

The one NaK pump would be centrifugal. An NaK to air heat exchanger would remove the nuclear heat from the NaK. Most heat would be dumped from the air to reactor process water above the core; approximately 30 percent would be carried outside the reactor in the cooling air stream to be dumped up the stack.

The pump and heat exchanger section above the core would be less than 6 inches in diameter to avoid space conflicts with reactor fuel elements, control rods, and other equipment. The in-pile section of this experiment would be short enough to permit discharge through the reactor discharge chute. All vessels and piping containing NaK would be enclosed in another vessel with a helium barrier in the annulus. A double barrier would always exist between the NaK and reactor water or cooling air. The helium annulus would also serve as insulation, and the helium annulus would be monitored for leakage of NaK or water.

Fuel may be in any form: pins, rods, cylinders, or plates. The coolant can operate at $\approx 1600^\circ\text{F}$ with a pressure of several hundred lb/in.².

Coolant flow would be controlled by varying the frequency of power to the canned induction-type motor on the pump. Coolant temperature would be controlled by varying the flow of air to the secondary side of the heat exchanger. All critical temperatures would be monitored by thermocouples which would leave the reactor through a flange or top cap penetration. Air for the heat exchanger and power for the pump would be supplied through the same penetration.

Out-of-pile equipment includes filters for exit air, air flow control valves, and radiation monitors. An instrument console would control the experiment. Motor generator sets for furnishing variable frequency power to the NaK pump would be located nearby.

E. BISMUTH LOOPS

1. Bismuth Characteristics

Liquid bismuth will dissolve uranium metal to the extent of 0.5 percent at 932°F . This is sufficient to attain criticality in a homogeneous reactor. It also has excellent possibilities for removal of fission products by a salt extraction method.

These two features, in addition to good heat transfer characteristics and high boiling point (2691°F at atmospheric pressure), make the bismuth-uranium mixture attractive for homogeneous power reactors.

The best structural materials for liquid bismuth loops are low chrome steels. Croloy 2-1/4 has been used for most experimental systems.

Liquid bismuth has a freezing point of 520°F , a disadvantage since supplementary heat is required in order to melt the solution and start flow in the system.

Bismuth is not as toxic as many heavy metals. It has many of the properties of lead and is no more dangerous. On irradiation, polonium-210 is formed, a very dangerous alpha emitter. Tests at Brookhaven indicate that on spills of irradiated bismuth-uranium mixtures, fission products and polonium are contained in the metal, minimizing the contamination hazards.

Bismuth-209 has a thermal neutron activation cross section of 0.015 barns and a scattering cross section of 0.9 barns. Bismuth-210 has a $T_{1/2}$ of 5.0 days, decaying with emission of 1.17 MeV betas

to polonium-210. Polonium-210 decays by emission of 5.94 MeV alphas with a $T_{1/2}$ of 138 days.

Due to the high density of bismuth, the pumps in bismuth loops must be designed for heavy loadings.

2. Typical Bismuth Loop

A recirculating bismuth loop was proposed for an ETR reflector position and an MTR horizontal graphite position. The loop has heat exchangers, pumps, and sampling facilities in an enclosed cubicle.

Because fission products and polonium-210 would be present in the molten bismuth-uranium mixture, direct maintenance on loop components would be extremely difficult. Removal of equipment after irradiation would be accomplished by dumping the molten bismuth solution into a shielded dump tank. The loop would then be remotely cut up into removable assemblies, each having shielding adequate to protect personnel and to permit shipping the assemblies to the experimenting laboratory.

The experiment was designed to test the feasibility of the handling procedures outlined above, as well as to test the nuclear and corrosion properties of the mixture. The information desired from the experiment includes the effect of irradiation on corrosion rate, on corrosion inhibitors, and on mechanical properties of the structural and moderating materials. In addition, fission damage to and fission product adsorption on the structural and moderating materials would be studied. The effect of fission products on the solubility of corrosion products and additives in the uranium-bismuth mixture would be evaluated. An elaborate sampling system and analytical procedure were outlined in order to gain this information.

The loop would be prepared for operation by preheating all piping and equipment to above the melting point of bismuth, using electrical heaters. The solu-

tion is melted in a melt tank then transferred to the pump sump. When all piping is at the proper temperature, the pumps are started, circulating the molten bismuth. The reactor would then be brought to power. Any loop malfunction, a power reduction, or an electrical failure would cause the solution to be dumped to the pump sump.

Special hazards in this loop, in addition to the shielding and maintenance problems, are flow stoppage due to freezing of bismuth in some part of the system and release of fission products and polonium-210 through system leakage.

F. FUSED SALT LOOPS

Fused fluoride salts or hydroxide systems have been used as alternates for liquid metals in reactor concepts where high temperatures and low pressures are desirable. These solutions have low melting points and low vapor pressure. The solutions have a low thermal neutron capture cross section and high fast neutron scattering cross section. Uranium fluorides are soluble in the solution.

When fluoride salts are used as the coolant and moderator in a homogeneous system, uranium fluorides could possibly be separated from the other salts and from fission products by volatilization.

Some fluorides which have very desirable properties are beryllium fluorides and lithium fluorides. Their melting points are $\approx 800^\circ\text{F}$. All other properties are approximately the same as other liquid metals.

In-pile tests of a fused salt loop have been successfully operated at MTR. The loop is similar to a small NaK loop; pump, heat exchanger, and heaters are assembled into a very compact system in a horizontal graphite position.

The principal problems encountered are component reliability, melting of the salts, removal of fission gases, and leak detection.

REFERENCES

1. Liquid Metals Handbook, Sodium-NaK Supplement, TID-5277 (1955).
2. Liquid-Metals Handbook, NAVEXOS P-733 (Rev.) (1952).
3. Summary Report of Design Criteria for a Thermal Flux Liquid Metal Package Loop in the Advanced Test Reactor, IDO-24042 (March 1963).

CHAPTER XIV

HAZARDS DISCUSSION

A. INTRODUCTION

It is the responsibility of TRA Project Engineering to completely evaluate the hazards associated with any experiment and experimental system that is installed in the MTR, ETR, or ATR. It is the purpose of such an analysis to allow the experiment sponsor the maximum operating flexibility for realization of his experimental objectives, while adequately safeguarding the particular experiment in question and maintaining an acceptable plant risk level with regard to the effect of any experimental malfunction upon personnel, the reactor plant itself, and other experiments.

Standard industrial and nuclear safety aspects of any major experimental facility or systems modification are very important considerations from the design stage through fabrication, inspection, installation, operation, and ultimate disposal. In like manner, the insertion of each new experimental element or capsule is subjected to no less serious scrutiny from a hazards standpoint.

Upon completion of engineering calculations and analysis and prior to insertion of a new experiment or operation of a new or modified facility, a formal request and/or hazards analysis (see Reference 2, 3, or 4 for a typical hazards analysis of a major loop facility) is submitted to the Safeguard Committee (Reference IDO-16569, Nertney, R. J., The MTR-ETR Safeguard Committee) for approval.

Nearly every pre-insertion or pre-installation effort has some safety implication. No attempt will be made here to list and discuss the safety aspects of the many steps that are preliminary to an irradiation test. Rather, this section will offer a general review of Chapters I through III and VII from a hazards

standpoint and will briefly consider two types of experimental failures, ie: (a) fuel element burnout and (b) piping failure and, ultimately, cleanup after failure.

B. NUCLEAR HAZARDS

By the very nature of the fact that MTR-ETR-ATR experimental facilities are operated primarily for the nuclear testing of fuel element prototypes, many of the greatest uncertainties and potential hazards involved in the pursuit of the overall experimental program are dependent upon the nuclear calculations and assumptions that are ultimately accepted as justification for a particular experimental insertion.

Uncertainties and variations (from cycle to cycle due to reactor loading, and from beginning to end of a cycle due to rod movement and fuel burnup) in neutron and gamma fluxes, perturbation calculations, and hypotheses accepted as a realistic maximum credible accident upon failure of an experimental element will necessarily present a relatively great potential for error when it is realized that information concerning these very unknowns may well be an objective of the particular experimental irradiation.

Nuclear coupling of the experiment to the test reactor is also an important consideration, particularly where many of the major experimental facilities form an important part of the reactor active core. In this regard, it is necessary to consider not only the effect upon the nuclear characteristics of the reactor core (primarily core reactivity and flux distribution) due to initial insertion of the experiment, but also the variation in these effects due to operating conditions of the particular experiment, ie, coolant voids, temperature changes, inhomogeneities, and burnup.

Prior to each ETR and ATR operating cycle, the ETR Critical Facility (ETRC) and the ATR Critical Facility (ATRC) are loaded with reactor fuel elements and nuclear mock-ups of all in-core experiments. Very low power runs are then made in these critical facilities and flux measurements obtained. By this means, it is possible to determine the effect of the particular loading upon reactor core reactivity and also to extrapolate flux measurements to full power conditions.

The intentional displacement or the possibility of an accidental displacement of all or part of the experimental components in the reactor core must also be considered. An example of the former is the operation of GEANP experiments in both the MTR and ETR, where test samples have been intentionally moved into and out of the respective facility high flux zones during full power reactor operation. Various in-tank and loop tests at the MTR and the ETR have also been intentionally cycled with respect to the core during operation. Where it cannot be demonstrated that this type of testing will not significantly affect reactor operation, the experiment may contain a compensating section with the same nuclear characteristics as the prime sample that can be inserted as the sample is withdrawn and vice versa. Hydraulic facility capsule irradiations must also be scrutinized from this standpoint since they are often inserted and discharged during full power runs. The latter possibility, ie, the accidental displacement of an experiment, could occur upon structural failure of the experiment itself, the experiment positioning device, or the hanger rod.

It is also necessary to evaluate the change in reactor core reactivity should a gas loop in-pile tube fail and become filled with water or should a water loop in-pile tube be voided either as a result of tube failure or a combination of low coolant flow and high heat generation.

C. HYDRAULICS UNCERTAINTIES AND CONSIDERATIONS

Flow tests have been run at the TRA site on various capsule sizes and configurations in the types of irradiation facilities commonly available. For the types of capsule experiments not easily referenced to these flow test results, the experiment sponsor may provide the results of flow tests run in his own laboratory, or he may rely solely upon hydraulic calculations with applicable uncertainties.

Flow rates in the major loop-type experiments can normally be controlled over a wide range, the limits being determined by pump, compressor, or blower capacity. Although the relative percent of error in flow measuring devices must be considered in an uncertainties analysis, this error is practically negligible when compared with the uncertainties of hydraulic calculations using the available pressure drop over an experimental system. In the determination of total coolant flow through an experimental system, it is, of course, necessary to analyze the distribution of flow in a multichannel system.

Aside from total flow determination by flow tests, calculations, or direct measurement, it immediately becomes necessary to consider the uncertainties in this flow due to mechanical changes in the experiment or other system disturbances, such as boiling in a water system. Whereas mechanical changes in a single channel experiment, short of element failure, may tend to restrict flow through a constant pressure drop system, this same phenomenon in a loop with automatic flow measurement and control may tend to increase coolant velocities and very likely improve heat transfer characteristics, assuming that the necessary driving head is available to compensate for the increased system pressure drop.

In a multichannel system or in a system with bypass flow in the experimental test section, the interrelation of channel flows necessitates a more rigorous analysis of the effect of a mechanical change or boiling conditions in one or more channels.

Another hydraulic consideration in a high-flow, multichannel system is the possibility of element distortion or failure due to vibration or differential pressure between channels.

In like manner, experimental component or element failure in a gas system may result from localized critical velocities or a high dynamic head on the test section. For these reasons, high flows and high inlet pressures in a gas system usually necessitate reactor power reductions, and careful analysis of system flow control characteristics is necessary to prevent critical velocities upon control component failure. These problems, ie, high dynamic heads and critical velocities peculiar to a gas system, are compounded by the fact that gas loop test elements are relatively prone to hydraulic failure, since structural strength is often sacrificed in the interest of improved heat transfer.

In addition to the considerations of loop and experiment hydraulic characteristics, reactor core hydraulics must receive careful study to assure that: (a) the reactor process water flow is sufficient to cool the experiment filler block and (b) that the channel spacing between the experiment filler block and reactor fuel elements or shim rods will not result in excessive lateral pressure differentials across reactor components.

D. HEAT TRANSFER UNCERTAINTIES AND CONSIDERATIONS

The primary uncertainty involved in experiment heat transfer calculations is the application of a particular heat transfer equation or correlation to the experiment

in question. For the forced water convection, nominally nonboiling, situation that is common to the in-tank capsule experiments and a majority of the loop experiments, several correlations are available, of which the Modified Colburn Equation is recommended. Even with this correlation, an uncertainty factor of plus or minus 20 percent is recommended, a relatively large consideration when compared to the uncertainties in flow and temperature measurements in the heat transfer analysis.

Whereas a great deal of information is available on forced convection heat transfer, considerably less experience is available as the heat transfer mechanism proceeds into the regions of boiling and, finally, film blanketing and burnout. It follows that relatively greater uncertainty factors must be applied to the available burnout correlations.

Although little is known of metal-water reactions resulting from extremely high adiabatic temperature rises in a fuel element due to loss of coolant flow, it may be hypothesized that a heavily loaded element with a very high heat flux could burn out in such a manner as to cause a metal-water reaction with sufficient explosive force to rupture the experimental container and damage the reactor itself.

Although forced convection heat transfer is applicable to a majority of the in-tank and water loop experiments and the addition of the nucleate boiling tests effectively includes all of the normal water-cooled situations, natural convection heat transfer becomes an important consideration upon pump stoppage, even though the reactor has scrambled. The afterheat from fission product gamma emission, particularly in the case of some larger, in-core irradiations, may easily be sufficient to cause excessively high specimen temperatures and even melting without some natural convection cooling.

CLOFA (Complete Loss of Flow Accident) tests have been run on some water loops to determine the effectiveness of natural convection heat transfer. In this type of test, loop flow is stopped, and the reactor is scrammed either manually or by automatic loop power reduction instruments. In-pile specimen and coolant temperatures are monitored by special high-speed instrumentation, and loop coolant flow may be re-established either manually or automatically if in-pile temperatures reach predetermined high levels.

Where the natural convection available in a loop system without the aid of auxiliary equipment is not deemed sufficient to provide adequate cooling, a convection cooler (a water-to-water heat exchanger located in the primary loop system at a position above the in-pile tube to increase the available natural convection cooling head and automatically controlled to begin operation at a certain low flow setpoint) may be included in the system. As an example of a more obtuse safety consideration, take the case of a convection cooler with secondary cooling water supplied from an auxiliary reactor demineralized water system. Although loss of loop flow may automatically valve the convection cooler into the primary system, this action would be effectively useless if the same malfunction which caused the flow loss were also to result in loss of power to the reactor demineralized water pumps.

Thus a loop hazards analysis will involve reactor plant utilities which supply electrical power to the loop and cooling for such equipment as convection coolers and pump bearings.

Gas and liquid metal heat transfer apply to a relatively small number of experimental irradiations at the TRA. Although the purpose of the analyses and many of the considerations are the same as for water systems, the heat transfer mechanisms, particularly in the case of gases, are different and somewhat more complex. Since gas film coefficients

are extremely low as compared with those for water systems, the temperature of the test specimen surface is sufficiently high to make radiant heat transfer an important consideration; in liquid metal heat transfer, however, the film coefficients in forced convection are even higher than those for water systems, but the conductivity of the liquid metal is so high that conduction becomes a consideration in forced convection heat transfer.

In addition to the actual correlation uncertainties, the effect of various system upsets upon the heat transfer mechanism must be carefully considered. Mechanical distortion and tolerances, fuel element homogeneity, and flux variations, in addition to the uncertainties in parameter measurements and boiling upsets mentioned previously, necessitate hot-spot and hot-channel factors (Reference Appendix D, Volume 4, and Chapter III, Volume 1) which must be applied to the nominal heat transfer situation. The magnitude of these factors is dependent upon actual measurement accuracies available and the probability that deleterious effects will occur.

Since the greatest potential nuclear hazard or incident during experimental irradiation is normally considered to be a serious element burnout or meltdown with gross fission product release, a complete heat transfer analysis is extremely important.

When experimental heat transfer calculations have been completed, a series of "safe operating curves" may be provided to justify a range of operating conditions and/or to provide the operator with a ready reference to evaluate system safety during off-normal and transient conditions. In this manner, it is possible to graphically present information to maintain nonboiling conditions on a particular experiment or to prevent burnout. Depending upon the particular loop parameters and transient characteristics plotted, a ready reference can be made available for safe loop operating

conditions and parameter setpoints at various reactor power levels.

In addition to specimen heat removal, the heat transfer analysis on a major experiment facility must also include the safe operation of out-of-pile components. Adequate cooling must be provided for electrical components, loop resins, and reactor components in the vicinity of hot loop piping. These considerations should assure that heat removal capacity in critical areas is sufficient to protect temperature-sensitive equipment and to prevent fires.

E. TRANSIENT ANALYSIS CONSIDERATIONS

When nuclear, hydraulic, and heat transfer calculations for a particular experimental system have been completed, the region of safe experiment operation has been defined as clearly as possible with applicable uncertainties. At this point in the analysis, it is necessary to carefully consider experimental parameter, ie, flow, pressure and temperature, transients.

Whereas flow coastdown characteristics for a recirculating water or gas loop can be determined quite accurately by test, single pass systems normally have a reserve coolant supply with a greater potential for correction should the prime mover malfunction or cut out temporarily. It should be noted here that although pump or blower coastdown characteristics for a recirculating system are normally accepted as a reasonable loss-of-flow transient, some component failures or operational errors could cause flow transients of greater severity. It would, however, be difficult or impossible to simulate these transients.

A pressure transient condition is much less clearly defined; and particularly, as concerns a loss of system pressure, a reasonable equipment failure or rupture must be hypothesized.

Temperature transients can be calculated, usually on an adiabatic or complete flow loss basis.

Since the severity of loop flow and temperature operating conditions is normally limited by a transient analysis which includes instrument response times, it may be possible to justify more severe operating conditions by installing faster-response instrumentation.

When these system parameter transients have been developed, their careful consideration together with an analysis of available reactor power reductions will provide the basis for adequate system or experimental protection during abnormal operating conditions. It should be noted here that it is necessary not only to choose a reactor power reduction rate of sufficient speed to protect the experiment, but the necessity for this power reduction rate should be clearly demonstrated if the operation of the test reactor is not to be unnecessarily compromised.

F. SAMPLE HANDLING

The hazards associated with the handling of irradiated experiments are essentially two-fold, ie, the possibility of direct radiation exposure to personnel and, in the case of fueled samples, the more nebulous consideration of fission gas release with resultant direct radiation and area contamination. Whereas the direct radiation level from induced activity in any experiment can be fairly well defined by calculation, fission product evolution from a defected or damaged sample interjects an uncertainty that should always receive careful consideration.

Although the handling of standard capsule type experiments at the TRA is fairly well defined and should be subject to minimum error, two areas of uncertainty or hazard peculiar to capsule irradiations should be mentioned: (a) many capsule experiments contain an internal

coolant that will react violently with water should the capsule be damaged or become defective during irradiation or handling, (b) misorientation of a capsule during irradiation could also result in a burnout situation that could be propagated into a serious nuclear incident.

Whereas the handling of standard capsule irradiations is fairly well defined, very thorough procedures must be developed for the checkout, insertion, operation, discharge, and transfer of the more complex capsule experiments and of each major loop type experiment. Loop experiments are generally much more subject to damage and fission gas evolution than are the contained capsule experiments, and a greater quantity of hazardous material is normally present. It is very important to carefully consider the design of each loop sample holder and transfer cask to minimize the probability of a failure which could result in serious radiation exposure to personnel during irradiation or transfer operations, or which could seriously compromise the operation of the experimental facility or the test reactor should a component failure or an operational error occur during insertion, removal, or operation.

The handling of in-pile tubes also requires extreme caution and thorough procedures. Since in-pile tube insertions and removals occur much less frequently than experiment insertions, removals, and transfers and by nature of the fact that an in-pile tube is a relatively large piece of equipment, the procedures are generally more complex; the probability of error is increased; and the possibility of damage to the reactor is greater. The level of induced radiation in an in-pile tube core section could result in serious exposure to personnel if an error should occur during removal operations.

G. CONSEQUENCES OF EXPERIMENT BURNOUT

1. Capsules

The consequences of fuel element burnout in an enclosed capsule experiment would not normally present a serious nuclear hazard, assuming that the integrity of the capsule seal were maintained. However, should the burnout be of a severity that results in a capsule rupture, release of fission products to the reactor process water system could easily result in a serious contamination problem. Should a ruptured capsule contain a liquid metal coolant such as Na or NaK, a violent reaction with the reactor process water might conceivably damage a reactor facility or even result in the displacement of reactor fuel or jamming of control rods. The probability of an element burnout and the consequences of such an incident must, therefore, receive careful consideration.

The consequences of burnout in an open-type (process water cooled) capsule would again be the release of fission products to the reactor process water system. The probable quantity of such a fission product release must be calculated.

2. Water Loops

The consequences of burnout in a recirculating water loop would be contamination of the experimental facility and radiation exposure to personnel from loop piping and components. The probable severity of such an incident must be carefully analyzed to assure that loop shielding is sufficient and that the loop contamination problem can be adequately handled.

In a single pass water loop, the added probability of immediate fission gas release to the reactor area is an important consideration.

3. Gas Loops

In a recirculating gas loop system the consequences of a fuel element burnout would be analogous to the problems just discussed for a recirculating water loop. Assuming that loop filters have an efficiency that would allow an acceptable

coolant cleanup, the gaseous fission products released upon discharge of the experiment may be less severe than the problem that could easily be encountered with fission gas release from a recirculating water loop.

In the case of a single pass air loop, there is the added hazard of immediate fission product release to the atmosphere with a potentially serious area contamination and personnel exposure problem. The probable quantity of fission product release and local atmospheric conditions must be carefully evaluated.

H. CONSEQUENCES OF PIPING FAILURE

1. Capsules

As mentioned previously, the consequences of a capsule failure would be the probability of fission product release to the reactor process water system with the added consideration of mechanical damage to a reactor facility, fuel elements, and control rods, depending, of course, upon the seriousness of the hypothesized accident.

2. Water Loops

The rupture of a recirculating or single pass water loop system could easily propagate itself into a serious nuclear incident due to loss of system pressure and flow and ultimate fuel element burnout and fission product release. As mentioned previously, core reactivity effects due to voiding of the in-pile tube must also be considered.

An additional hazard associated with a serious piping failure in a high-pressure, high-temperature, recirculating water loop system would be the possible exposure of personnel to high-temperature steam. Even if such a break were to occur within the confines of an exhausted cubicle, the driving force of a hot loop

surge tank would likely result in the escape of some steam and fission products into the reactor building.

3. Gas Loops

As in the case of single pass and recirculating water loop systems, the rupture of gas loop piping could easily result in a serious nuclear incident.

Also, where the concern would be for escaping hot steam in a water loop system, the discharge of hot gas from a rupture in one of the large, high-temperature gas loops could create a very serious hazard to any personnel in the immediate area.

The justification for installation of such a potentially hazardous system is the extremely small probability that such an incident could occur in a well-designed, high-integrity loop.

I. CLEANUP AFTER FAILURE

1. Reactor Process Water

Continuous radiation monitoring of the MTR-ETR-ATR process water streams will give an immediate indication of increased fission product activity due to an experimental capsule failure. Depending upon the nature and quantity of fission products released to the reactor process water system, the reactor may experience an unscheduled shutdown for determination and discharge of the failed experiment.

Due to the large volume of the reactor process water systems, a minor capsule failure will not ordinarily present a serious cleanup problem. As mentioned above, however, it may be necessary to shut the reactor down prior to a scheduled shutdown for removal of one or more suspect experiments. It is often difficult to determine the particular experiment that has failed. This determination is accomplished by means of fission product analysis and observation

of bubbles in the reactor tank which could well be emitted from a failed experiment.

Cleanup of the reactor process water systems is continuous by means of a small sidestream flow of process water (several hundred gpm) through anion and cation resins.

2. Water Loop Coolant

The severe failure of a fuel element in a water loop system can result in serious contamination of the loop piping due to plating out of released fission products. Highly contaminated loop water can be cleaned up to some extent by continuous flow through the loop resin columns. If this cleanup is not sufficient, the contaminated water can be dumped to the loop holdup tank for further decay and ultimate draining to the reactor hot drain system, or it can be pumped into a shielded tank-truck for disposal at the Chemical Processing Plant.

Experimental hot drain systems at the TRA provide several connecting points for pumping contaminated loop water to an external header for disposal at ICPP. The hazard of a serious radioactive spill during disposal is also an important consideration.

Depending upon the type of element failure in a water loop system and also depending upon the material of construction of the particular loop, acidic or basic solutions or a combination of acids and bases may be alternately pumped through the loop piping for decontamination purposes. The problem of disposal of these solutions is analagous to the above discussion concerning disposal of highly contaminated loop water.

It should be mentioned here that a water loop system may become con-

taminated slowly over several years of operation with defected fuel elements. The decontamination procedures and considerations are essentially the same as in the case of a serious element failure.

When loop decontaminations are undertaken, it is also very important to consider the possible deleterious effect of decontaminating solutions on the loop piping and welded joints. It may be necessary to cut out a section of loop piping following a decontamination to determine that the integrity of the system has not been compromised.

3. Gas Loops

If it is hypothesized that an element failure in a gas loop system could be of such a magnitude as to make filtering and immediate exhausting to the atmosphere intolerable, it is necessary to provide relatively large holdup or decay tanks where the gaseous fission products can be stored until dilution to the atmosphere is permissible.

4. General

The disposal of solid wastes, ie, highly contaminated loop components, used resins, air filters, etc, is ordinarily handled by shipment to the burial grounds. If the particular solid waste is too highly radioactive for shipment in a standard waste container, a shielded container must be provided specifically for disposal purposes.

It should be evident from the foregoing discussion that a careful analysis of possible system contamination, waste disposal, and decontaminating procedures must be made prior to insertion of any fueled experiment in the MTR-ETR-ATR facilities.

REFERENCES

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3. R. L. Gump, ETR Hazards Analysis of the WAPD-M13 Experiment, IDO-16503 (1958).
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CHAPTER XV

SHIPPING RADIOACTIVE MATERIAL

A. GENERAL

Because of the nature of the work at the National Reactor Testing Station, a great number of radioactive shipments are originated here. Experiments are irradiated at the TRA for the AEC, AEC Contractors, and private and public organizations from all over the world. Most of these companies or organizations desire to examine the irradiated experiments in their own facilities. This results in the shipment of a broad spectrum of radioactive materials to many places.

Compliance with various regulations and safety practices is the responsibility of the person who initiates the radioactive shipment. For this reason and because regulations of this nature are constantly being revised and upgraded, references to the applicable regulations are given here with only a brief description of some of the more common ones. It is suggested that persons making radioactive shipments become familiar with the current applicable regulations.

B. REGULATORY REFERENCES

The following references to regulatory and directive publications are provided for the use of those who must be familiar with the details of such regulations. Care should be taken that the latest revision of each publication is obtained.

(1) Interstate Commerce Commission Regulations for Transportation of Explosives and other Dangerous Articles, Parts 71 through 78 as outlined in T. C. Georges Tariff.

(2) ICC Motor Carrier Safety Regulations.

(3) Civil Aeronautics Board Regulations, Part 49.

(4) U. S. Coast Guard Regulations, Part 146.

(5) U. S. Postal Guide, Article 37, Chapter IV.

(6) AEC Manual, Chapter AEC-0529, "Safety Standards for the Preparation of Radioactive and Fissile Material for Transportation".

(7) AEC Manual, Chapter AEC-2402, "Security of Matter in Transit".

(8) Federal Register, 10 CFR 72, "Protection against Radiation in the Shipment of Irradiated Fuel Elements (Proposed Rule Making)".

(9) ID Manual, ID-5201, "Basic Transportation Policy".

(10) Annex L, ID Manual, Appendix 0500-1, Report No. IDO-12028, "Standard Health and Safety Requirements".

(11) Annex Q, ID Manual, Appendix 0500-1, Report No. IDO-12028, "Standard Health and Safety Requirements".

(12) Phillips Petroleum Company Administrative Instruction Bulletin, AIB-8.10, "Shipping and Receiving Radioactive Materials".

C. OFF-SITE SHIPMENTS

1. Exemptions from ICC Requirements for Specified Packaging and Labeling

Shipments of radioactive materials are exempt from the ICC requirements for packaging, marking, and labeling provided they fulfill the conditions outlined below.

(a) The package must be such that there can be no leakage of radioactive

material under conditions normally incident to transportation.

(b) The package must contain not more than:

(1) 0.1 millicuries of radium or polonium, or

(2) 0.135 millicuries of strontium-89, strontium-90, or barium-140, or

(3) 1.35 millicuries of any other radioactive material.

(c) The package must be such that no significant alpha, beta, or neutron radiation is emitted from the exterior of the package; and the gamma radiation at any surface of the package must be less than 0.433 milliroentgens per hour.

Other exemptions cover manufactured articles which include radioactive materials as component parts, radioactive ores, and other items not normally dealt with at the TRA.

2. Exemptions from AEC Manual, Chapter 0529, "Requirements for Preparation of Radioactive and Fissile Material for Transportation".

Materials listed below are exempt from the requirements of Part I of AEC Manual, Chapter 0529. These materials may or may not be exempt from ICC requirements (see above).

(a) Natural or depleted uranium, regardless of the amount of contained U-235.

(b) Any single package which contains less than 15 grams of fissile material (uranium-233, uranium-235, plutonium-239, and plutonium-241).

3. Quantity Limitations

Except by special arrangements, the quantities of radioactive materials that can be shipped in one outside container are limited as follows:

(a) Not more than 2.0 curies of radium, polonium, or other members of the radium family of elements.

(b) Not more than 2.7 curies of any other radioactive substance.

(c) Not more than 300 curies of solid cesium-137, cobalt-60, gold-198, or iridium-192 when shipped in ICC-55 containers.

(d) Air shipments are limited to a maximum of 2.0 curies of any materials. Exceptions can be arranged through the Federal Aviation Agency.

Shipments of greater quantities of radioactive materials than shown above can be made in containers especially approved by the Bureau of Explosives for excess curie shipments.

Shipments of irradiated fuel elements where total amount of radioactivity in any one container exceeds 2000 curies must, in addition to the following, meet the requirements outlined in Part II of the AEC Manual, Chapter 0529. Shipments of irradiated material that is not fissile material must meet the requirements of Part III of the AEC Manual, Chapter 0529.

4. Packaging and Shielding Requirements

All radioactive materials must be packed and shielded so the degree of fogging of undeveloped film exposed for 24 hours at 15 feet from the source will not exceed that produced by 11.5 milliroentgens of penetrating gamma rays of radium filtered by 1/2 inch of lead. In addition, the gamma radiation must not exceed 200 milliroentgens per hour at the surface of the container or 10 milliroentgens per hour at one meter from the source. The container must be prepared for shipment in such a manner that no significant surface contamination exists on any part of the container (not more than 1000 d/m per 100 cm² beta-gamma and 20 d/m per 100 cm² alpha).

In addition to the normal packing, radioactive materials such as plutonium, strontium, etc, which present special hazards due to their tendency to remain in the human body for long periods of time, must be packed in inside metal containers which meet ICC specification 2R or have special Bureau of Explosives approval. Figure 45 shows a typical 2R container.

5. Approved Shipping Containers

Any container for radioactive materials must be at least four inches in its smallest dimension and meet one of the following specifications.

(a) Fiberboard Boxes -- must meet ICC Specification 12B.

(b) Wooden Boxes -- must meet ICC Specification 15A or 15B.

(c) Plywood Boxes -- must meet ICC Specification 19A or 19B.

(d) Fiber Drums -- must meet ICC Specification 21C.

(e) Metal Barrels or Drums -- must meet ICC Specification 6A, 6B, 6C, 17C, or 17H.

(f) Metal Encased Lead or Uranium Metal Shielded Containers -- must meet ICC Specification 55.

(g) Special containers may be approved by the Bureau of Explosives or the AEC even though they do not meet

Fig. 45 Typical 2R container.

any of the above specifications. In order for the shipper to be assured that a shipping container meets the special approval, a copy of the approval letter or AEC license and a set of the container design drawings should be in hand before using the container.

6. Marking and Labeling

For the purposes of marking and labeling, radioactive materials have been divided into three groups according to the types of radiation emitted.

GROUP I -- Radioactive materials that emit gamma rays only or a combination of both gamma and alpha/beta radiation.

GROUP II -- Radioactive materials that emit neutrons and either or both types of radiation found in Group I materials.

GROUP III -- Radioactive materials that emit alpha and/or beta radiation only.

Material in Groups I and II is to be labeled with a red label as described in ICC Regulation 73.414. Material in Group III is to be labeled with a blue label as described in ICC Regulation 73.414. A properly labeled ICC-12B-40 box ready for shipment is pictured in Figure 46.

Magnesium-Thorium alloys in formed shapes or uranium (normal or depleted)

Fig. 46 Properly labeled ICC-12B-40 box ready for shipment.

in solid metal form are to be labeled with a white label with red lettering as described in ICC Regulation 73.414.

The white "empty" label is to be used on empty containers and must completely cover any old red or blue labels which have not been completely removed.

D. ON-SITE SHIPMENTS

1. Packaging and Shielding

Packaging must be accomplished in such a manner that radioactive material will not leak during transportation. Sufficient shielding must be provided to prevent excessive exposure to personnel during transportation. Any shipment with radiation levels in excess of 500 mrem/hr at the extremities of the vehicle must be approved for shipment by Health Physics supervision. The surface contamination on any part of the container shall be less than 1000 d/m per 100 cm² beta-gamma and 20 d/m per 100 cm² alpha.

2. Transportation

Taxis or buses are not to be used for transporting radioactive materials. Vehicles used for this purpose shall be identified by a sign in front and rear worded "Caution -- Radioactive Material" in three inch high or larger letters. Passengers in these vehicles are limited to personnel required to handle the shipment during transport.

E. TIE-DOWN REQUIREMENTS

1. Minimum Requirements

The following are considered minimum standards; however, due to the variety of shipping containers, the approving safety representative may direct additional action or modification as the situation demands.

(a) Good rigging practice shall be observed at all times.

(b) The carrier must be in good operating condition, of adequate capacity to carry the load, and provided with adequate tie rings or other acceptable means of securing the load to the frame (not just the truck bed).

(c) The load shall be positioned on the carrier with consideration for distribution of concentrated load, good tie-down positioning, and wheel and axle loading within regulating limitations of roadways enroute.

(d) Palleting ties to the cask are normally insufficient for consideration in the tie-down requirements. In other than specially designed frames, the cask proper must be tied down to the carrier.

(e) Tie points on containers shall be such that the container is held securely and connections cannot work or vibrate loose. The normal tie point on the container should be at least 2/3 of the height of the container.

(f) Tie-down from cask to carrier shall approximate a 45 degree angle with the horizontal, shall be drawn taut, and shall be protected from sharp edges or corners. The container shall be restrained in all directions with any supplementary restraint in the forward direction.

(g) Opposing basket hitches are normally acceptable with four separate terminal connections; however, containers are provided greater restraint with independent four-way tie-down in the event of partial failure.

(h) Cable shall normally be used to secure all off-site radioactive shipments. This normally excludes the use of chain (above 500-pound container), but does not prohibit the use of other tie-down media having shock load handling ability comparable to cable. The tables in Annex Q of the ID Standard Health and Safety Requirements shall be used for guidance in cable size, connections, number of clips, and torque for developing tie-down strength within safe working load.

(i) Chains, where specifically approved by the safety representative, may be used to secure on-site shipments. The applicable tables in Annex Q of the ID Standard Health and Safety Requirements shall be used for guidance in selecting chain and attachments of adequate size and type to develop tie-down strength within safe working load.

(1) Hook connection to chain shall be with grab hooks of sufficient depth to hold the full width of chain link.

(2) Boomer handles, clevis pins, and hooks used in tie-down shall be wired or otherwise secured to prevent displacement.

(3) Knots in chain shall not be permitted, and sharp edges shall be blocked or padded.

A truck loaded with large shipping containers properly tied and ready for shipment is illustrated in Figure 47.

2. Special Considerations

(a) Rail shipment under normal transport conditions are subject to greater vibration and vertical shock loading than truck transport, hence:

(1) Base plates for containers or cribbing used to hold the container shall be strongly secured by bolting, welded retainers, or equal.

(2) When the safety representative considers the car floor to have sufficient strength, bolting through the floor with support plates underneath is acceptable.

(3) All bolted connections shall be double-nutted, spot welded, or lock washered to prevent vibrational loosening.

(b) Unusually heavy containers requiring movement between areas of the

NRTS and which cannot be physically tied-down to the same degree of restraint as lighter containers shall be under continuous supervision and safeguarded in movement by the exercise of appropriate speed control.

(c) Alloy chain should not be used for tie-down in sub-zero temperature conditions.

(d) The safe working load of a lifting assembly should be reduced in accordance with wear factors of any piece which could affect the capacity of the assembly.

(e) Load binders are generally stamped with the ultimate tensile (breaking) strength for straight line pull in a closed position. Generally, the safe working load should be from $1/5$ to $1/3$ of the ultimate breaking strength which allows for accidental stresses. If the load binder is deformed in service, it is being used beyond the safe working load and a stronger binder should replace it.

(f) Turnbuckles should not normally be used; however, where accepted by the safety representative in an assembly with cable, they shall be within the working load designed with a safety factor of 5 based on the breaking strength and shall be provided with jam lock nuts or number nine wire twist-ties to prevent vibrational loosening.

(g) Safety Factor

(1) Tie-down within the safe working load provides a built-in safety factor of five for cable and three plus for chain.

(2) The tie-down or combination of tie-downs in the forward direction of restraint (inclusive of all its parts) shall be gauged to hold the entire container weight within the safe working limit.

Fig. 47 Shipping containers properly loaded and tied in truck.

APPENDIX A SAMPLE CAPSULE EXPERIMENT CALCULATIONS

A. INTRODUCTION

This appendix presents a relatively complete set of calculations for a typical capsule experiment. The calculations are representative of those required before an experiment can be irradiated at ATR. The various data for the experiment are taken from the experiment data package.

The capsule consists of four flat plates positioned end to end. Each plate is approximately $1\frac{1}{2} \times \frac{11}{16} \times 0.045$ inches, with the fuel portion occupying a volume of $1 \times \frac{1}{2} \times 0.033$ inches. The plates are immersed in NaK, and the NaK chamber is enclosed in an outer can. Each plate has one thermocouple measuring the plate surface temperature. Heat transfer across the annulus between NaK vessel and outer capsule wall is mainly accomplished through heat transfer fins. Figure A-1 is a photograph of the capsule assembly.

Information needed includes total nuclear power generations, hydraulic calculations on reactor water flow, a temperature profile from center of capsule to outer capsule wall, and an uncertainty analysis on the nuclear heat transfer and hydraulics calculations.

In addition, nuclear afterheat, source strength in curies, and source strength in MeV per second are needed for shielding and shipping calculations.

B. NUCLEAR CALCULATIONS

The purpose of the nuclear calculations is to determine the unperturbed thermal neutron flux which will give the heat flux necessary for operation at the desired center temperature of 1688°F. The most difficult part of the calculations is to determine the ratio between unperturbed thermal neutron flux and effective thermal neutron flux needed.

1. Flux

For calculation of the ratio between effective flux and unperturbed flux, called the perturbation factor, the methods presented in Reference 2, Chapter I of this volume, were used.

The relation between unperturbed reactor thermal neutron flux and specimen flux is based on W. B. Lewis' correlation:

$$\phi_E = \phi_R f A$$

where ϕ_E is the effective flux within the fuel, ϕ_R is the unperturbed reactor flux, $f = f(\bar{b}) = f\left(\frac{4v}{S} \sum a\right)$, ie, a function of four times the fuel volume to surface area ratio times the macroscopic thermal neutron absorption cross section. A is the capsule attenuation factor. Calculation of factor f,

$$\frac{4v}{S} = \frac{4 (1 \times \frac{1}{2} \times 0.033)}{2 [(1 \times \frac{1}{2}) + (1 \times 0.033) + (\frac{1}{2} \times 0.033)]} = 0.0602 \text{ in.} = 0.1528 \text{ cm}$$

Fig. A-1 Capsule components, BMI 33-1 and -2.

$\Sigma_a = \sum \Sigma_a$ for U-238, U-235, and stainless steel

$$\frac{\sigma_a}{\text{wt}} \text{ for U-238} = 0.0069 \text{ cm}^2/\text{g}, \text{ for U-235} = 1.779 \text{ cm}^2/\text{g}, \text{ and for SS} \\ = 0.034 \text{ cm}^2/\text{g}$$

Fuel Composition = 30 wt % UO_2 with highly enriched U (93.5 wt % U-235)

$$\text{U-238 wt \%} = \frac{238(0.065)(30)}{270} = 1.72$$

$$\text{U-235 wt \%} = (235/267)(0.935)(30) = 24.7$$

$$\text{SS wt \%} = 70$$

$$\text{O}_2 \text{ wt \%} = 3.58$$

$$\Sigma_a \text{ U-238} = 0.0172 \times 0.0069 = 0.0001 \text{ cm}^2/\text{g}$$

$$\Sigma_a \text{ U-235} = 0.2470 \times 1.779 = 0.439 \text{ cm}^2/\text{g}$$

$$\Sigma_a \text{ SS} = 0.70 \times 0.034 = 0.024 \text{ cm}^2/\text{g}$$

$\Sigma_a \text{ O}_2$ is negligible,

$$\text{total} = 0.463 \text{ cm}^2/\text{g}$$

$$\text{Core Density} = 8.72 \text{ g/cc}$$

$$\Sigma_a = 0.463 (8.72) = 4.03 \text{ cm}^{-1}$$

$$\bar{b} = \frac{4v}{S} \Sigma_a = 0.1528 (4.03) = 0.615$$

With b_{av} calculated, the following factors are taken from the graph for W. B. Lewis' correlation: $f = 0.66$, $Y = 0.75$, $F = 0.91$.

Calculation of factor A:

The attenuation factor is given by:

$A = e^{-\sum (\Sigma_a X)}$ where Σ_a represents the macroscopic thermal neutron absorption cross section of each annulus material in the capsule, and X is the annulus thickness.

Cladding -- (0.006-inch thick stainless steel)

$$\Sigma_a X = 0.034 \frac{\text{cm}^2}{\text{g}} \left(8.03 \frac{\text{g}}{\text{cm}^3} \right) (0.006 \text{ in.}) \left(2.54 \frac{\text{cm}}{\text{in.}} \right) = 0.0042$$

NaK annulus -- approximately 0.92 cm thick:

$$\Sigma_a X = 0.028 (0.875) (0.92) = 0.0225$$

Inner shell--SS + finned portion:

$$\Sigma_a X = 0.034 (8.03) \left[\left(\frac{140}{360} \times \frac{0.038}{0.288} \times 0.0625 \right) + (0.0625) \right] (2.54) = 0.0456$$

Outer Shell--0.0625-inch-thick SS:

$$\Sigma_a X = 0.034 (8.03) (0.0625) (2.54) = 0.0433$$

$$\Sigma \Sigma_a X = 0.0042 + 0.0225 + 0.0456 + 0.0433 = 0.1156$$

$$\begin{aligned} \phi_E &= \phi_R f A = \phi_R (0.66) e^{-0.1156} \\ &= \phi_R (0.66) (0.891) = 0.586 \phi_R \end{aligned}$$

Therefore, the effective flux in the sample will be 0.586 of the measured unperturbed flux.

2. Heat Generation

Total heat generation in the sample is computed from the reaction rate equation = $N \sigma \phi$. It should be pointed out that the basic problem from the design point of view is to calculate what fission heat generation rate is necessary to keep the sample at the requested temperature and then calculate what reactor flux is required.

For this sample, the heat generation rate is 1134 watts = 1.075 Btu/sec.

$$\text{Rate} = N \times \phi \times \sigma_f \times (\text{conversion from fissions to Btu})$$

$$1.075 = (1.492 \times 10^{21}) \phi (5.8 \times 10^{-22}) (3.2 \times 10^{-14} \text{ Btu/fission-sec})$$

$$\phi = 0.388 \times 10^{14} \text{ (effective)}$$

$$= 0.388 / 0.586 = 0.662 \times 10^{14} \text{ (required)}$$

$$\text{Total heat generation} = 4 (1.075) = 4.30 \text{ Btu/sec} = 15,500 \text{ Btu/hr}$$

This neglects gamma heat. Since reactor gamma heat in MTR is on the order of 1.5 watts/g in the reflector and there is not much mass represented here, the gamma heat will be neglected.

The heat flux on the fuel plates is obtained by dividing the heat generation by the heat transfer area:

$$15,500 / 4 = 3,875 \text{ Btu/hr/plate}$$

$$Q/A = 3,875 / 2 (1 \text{ inch} \times 0.5 \text{ inch}) / 144 \text{ in.}^2 / \text{ft}^2 = 558,000 \text{ Btu/hr-ft}^2$$

3. Gamma Energy

Gamma energy given off by the sample will be determined from Figure 17, page 40, for periods of one hour, one day, and 30 days after irradiation time. Irradiation time is assumed to be 32 days, and operating power is assumed to be 1134 watts.

$$\text{Operating power} = P_o = 1134 \text{ watts}$$

$$\text{Irradiation time} = t_o = 32 \text{ days} = 2.76 \times 10^6 \text{ sec.}$$

$$\text{Decay time} = t_d = 1 \text{ hour} = 3.6 \times 10^3 \text{ sec.}$$

$$= 1 \text{ day} = 8.64 \times 10^4 \text{ sec.}$$

$$= 30 \text{ days} = 2.59 \times 10^6 \text{ sec.}$$

The methods outlined in Chapter IV, Section C.4 will be used to find the source strength of the sample in MeV/sec.

$$(1) P_o = 1134 \text{ watts}$$

$$t_o = 2.76 \times 10^6 \text{ sec.}$$

$$t_d = 3.6 \times 10^3 \text{ sec.}$$

$$P_{g1} = 3.6 \times 10^{10} \text{ MeV/sec-watt}$$

$$P_{g2} = 9.6 \times 10^9 \text{ MeV/sec-watt}$$

$$P_g = (3.6 \times 10^{10} - 9.5 \times 10^9) 1134 \text{ MeV/sec}$$

$$= (2.64 \times 10^{10}) 1.134 \times 10^3 \text{ MeV/sec}$$

$$= 3 \times 10^{13} \text{ MeV/sec after 1 hour decay}$$

$$(2) P_o = 1134 \text{ watts}$$

$$t_o = 2.76 \times 10^6 \text{ sec}$$

$$t_d = 8.64 \times 10^4 \text{ sec}$$

$$P_{g1} = 1.9 \times 10^{10} \text{ MeV/sec-watt}$$

$$P_{g2} = 9.5 \times 10^9 \text{ MeV/sec-watt}$$

$$P_g = (1.9 \times 10^{10} - 9.5 \times 10^9) 1134 \text{ MeV/sec}$$

$$= 1.08 \times 10^{13} \text{ MeV/sec after one day decay}$$

$$(3) P_o = 1134 \text{ watts}$$

$$t_o = 2.76 \times 10^6 \text{ sec}$$

$$t_d = 2.59 \times 10^6 \text{ sec}$$

$$P_{g1} = 9.8 \times 10^9 \text{ MeV/sec-watt}$$

$$P_{g2} = 8.6 \times 10^9 \text{ MeV/sec-watt}$$

$$\begin{aligned}
 P_g &= (9.8 \times 10^9 - 8.6 \times 10^9) 1134 \text{ MeV/sec} \\
 &= 1.36 \times 10^{12} \text{ MeV/sec}
 \end{aligned}$$

C. HYDRAULIC CALCULATIONS

The equivalent diameter for the water passing between the outside of the capsule and the inside of the X-basket is easily calculated. However, the velocity of water passing through is difficult to determine. Water velocity will be calculated by use of experimental curves.

Information given:

Capsule Length = 13 inches

Capsule Diameter = 1.125 inches

X-basket ID = 1.253 inches

X-basket Length = 37-1/4 inches

ΔP Across Core = 40 psi

$$\begin{aligned}
 D_e &= D_1 - D_2 \\
 &= 1.253 - 1.125 \\
 &= 0.128 \text{ inch}
 \end{aligned}$$

$$\begin{aligned}
 \text{Flow Area} &= \frac{\pi}{4} (1.253)^2 - \frac{\pi}{4} (1.125)^2 \\
 &= \frac{\pi}{4} (1.253^2 - 1.125^2) \\
 &= 0.240 \text{ in.}^2 \\
 &= 1.67 \times 10^{-3} \text{ ft}^2
 \end{aligned}$$

Capsule length is 13 inches, X-basket length is 37-1/4 inches, so the X-basket is 35 percent loaded. Figure 8, page 18, indicates that for a flow area of $1.67 \times 10^{-3} \text{ ft}^2$, a 35 percent loaded X-basket would have a total flow of 23 gpm. Coolant velocity, determined from Figure 9, page 18, is ≈ 20 feet per second for a thin walled X-basket.

D. HEAT TRANSFER CALCULATIONS

1. ΔT Film Calculations

The following heat transfer analysis was made to determine temperature at various locations in the capsules. The method involves calculating temperature differences across the several annuli from the water-contacting outside surface of the capsule to the specimen core.

Total heat generation per specimen (assuming capsule loading of 64 g/in.)
 $q = 1134 \text{ watts} + (64 \text{ g/in.}) (1.5 \text{ w/g}) = 1134 + 96 = 1230 \text{ watts/inch} = 4200 \text{ Btu/hr-in.}$

Maximum capsule surface heat flux: (hottest 1-inch length)

$$\left(\frac{Q}{A}\right)_{\max} = \frac{4200}{\pi (1.125) \left(\frac{1}{144}\right)} = 170,500 \text{ Btu/hr-ft}^2$$

Estimated capsule surface heat flux. (Assuming that the total heat transferred takes place over a length of 6 inches. Each of the four specimens has an overall length of 1-1/2 inches):

$$\left(\frac{Q}{A}\right)_{\text{avg}} = \frac{4200}{\pi (1.125) \left(\frac{1.5}{144}\right)} = 114,000 \text{ Btu/hr-ft}^2.$$

Velocity = 20 feet per second

$$D_e = 0.128 \text{ inch}$$

$$T_{\text{in.}} = 110^\circ\text{F}$$

$$\begin{aligned} K &= \frac{V^{0.8}}{D_e^{0.2}} \\ &= \frac{(20)^{0.8}}{(0.128)^{0.2}} = \frac{11.0}{0.663} \\ &= 16.6 \end{aligned}$$

From Figure 11, page 27, the base factor for a film temperature of 125°F is 320.

$$\begin{aligned} h &= bK = 320 \times 16.6 \\ &= 5310 \text{ Btu/hr-}^\circ\text{F-ft}^2 \end{aligned}$$

$$\Delta T_{\text{Film}} = \frac{Q/A}{h} = \frac{1.705 \times 10^5}{5.31 \times 10^3}$$

$$\Delta T_{\text{Film}} = 32^\circ\text{F (Max)}$$

From Figure 11, the base factor for a film temperature of 126°F is 325.

$$\begin{aligned} h &= bK = 325 \times 16.6 \\ &= 5400 \text{ Btu/hr-}^\circ\text{F-ft}^2 \end{aligned}$$

$$\Delta T_{\text{Film (Avg)}} = \frac{1.705 \times 10^5}{5400} = 32^\circ\text{F Avg}$$

For the remaining nuclear calculations, a metal temperature of 142°F is assumed.

2. Through Outer Shell Wall:

$$\Delta T = q \frac{\ln r_2/r_1}{2\pi k L} \quad (\text{radial conduction})$$

(From this point on, temperature differences are calculated on the basis of actual heat transfer areas, as closely as can be estimated. A separate set of calculations to determine maximum possible temperatures is made in the Hot-Spot Analysis.)

$$\begin{aligned} \Delta T &= \frac{4200 \ln \left(\frac{1.125}{1.000} \right)}{2\pi \times 9 \times \frac{1.5}{12}} \\ &= 70^\circ\text{F}. \end{aligned}$$

(where the heat-transfer length is assumed to be 1.5 inch per specimen)

Temperature at inner surface of outer shell = 142 + 70 = 212°F

3. Through Finned Area:

$$\Delta T = \frac{q \ln r_2/r_1}{2 k L \pi}$$

Where q = total heat generation minus gamma heat from outer shell
(assumed to be 143 Btu/hr)

$$= 4200 - 143 = 4057 \text{ Btu/hr}$$

k = thermal conductivity of stainless steel fins and helium between fins.

The fins are 0.038 inch wide and 1/4 inch apart. They are in two segments of 70° arc on opposite sides of the inner shell and touch the outer shell.

$$\begin{aligned} \therefore k &= \frac{2(70)}{360} \left[L_{\text{He}} k_{\text{He}} + L_{\text{ss}} k_{\text{ss}} \right] + \frac{220}{360} \left[L_{\text{He}} k_{\text{He}} \right] \\ &= \frac{140}{360} \left[(0.868) (0.115) + (0.132) (10) \right] + \frac{220}{360} \left[(1) (0.115) \right] \\ &= 0.622 \text{ Btu/hr ft}^\circ\text{F} \end{aligned}$$

$$\Delta T = \frac{4057 \ln \frac{1.000}{0.875}}{(2\pi) (0.622) \left(\frac{1.5}{12} \right)} = 1100^\circ\text{F}$$

∴ Temperature at outer surface of inner shell (at root of fins) is:

$$212 + 1100 = 1312^{\circ}\text{F}$$

4. Through inner shell wall:

$$\begin{aligned}\Delta T &= \frac{q \ln r_2/r_1}{2 \pi k L} \\ &= \frac{4057 \ln \frac{0.875}{0.750}}{2 \pi (13.5) (\frac{1.25}{12})} \\ &= 70^{\circ}\text{F}\end{aligned}$$

(where the heat-transfer length is assumed to be 1.25 inch per specimen).

Therefore, the temperature at inner surface of inner shell is:

$$1312 + 70 = 1382^{\circ}\text{F}$$

5. Through NaK:

Calculation of this temperature difference is complicated by the fact that the heat will be transferred from a flat plate to a cylindrical wall through NaK by both conduction and convection. Further, convection is interrupted by the presence of specimen holders, thermocouples, and dosimeters near the specimen. Therefore, the ΔT value across the NaK was estimated by observing the recorded temperatures of thermocouples touching the respective surfaces in previous capsule experiments in which specimen heat generations were of the same order of magnitude as those estimated for these capsules, and then calculating the approximate temperature difference between the thermocouple bead and surface it touches.

An examination of previous data suggests a ΔT value of 270°F as being representative of the temperature difference between thermocouples touching the specimen surface and the inner wall of the NaK containing inner shell.

The ΔT between thermocouple bead and specimen or wall surface it touches can be computed approximately, if it is assumed that the heat transfer is by linear conduction and the thermocouples beads are an average distance of $1/32$ inch from the surface. (The thermocouple bead OD is $1/16$ inch.)

6. ΔT Between Specimen Surface and Adjacent Thermocouple.

$$q = \frac{k A \Delta T}{x}$$

where $q = 3890$ Btu/hr ($1134 + 6$ watts; fission plus gamma from one specimen).

$$\begin{aligned}\Delta T &= \frac{q x}{k A} \\ &= \frac{(3890) \left(\frac{1}{(32)(12)} \right)}{(15.6) 2 (1 \times 1/2 \times 1/144)} = 94^{\circ}\text{F}\end{aligned}$$

7. ΔT Between Inner Surface of Inner Shell and Adjacent Thermocouple.

$$\Delta T = \frac{q x}{k A}$$

where $q = 3920$ Btu/hr (fission and gamma from specimen and gamma from NaK).

$$A = 2 (1 \times 1/2 \times 1/144)$$

$$= 0.00695 \text{ ft}^2 \text{ (This area is approximately that of a 1-inch length of the inner shell which has finned contact with the outer shell.)}$$

Theoretically, more than 80 percent of the heat is conducted through the stainless steel fins between the inner and outer shells.

$$\Delta T = \frac{(3920) \left(\frac{1}{(32) (12)} \right)}{(15.6) (0.00695)}$$

$$= 94^\circ\text{F}$$

Therefore, the specimen surface temperature is

$$1382 + 270 + 94 + 94 = 1840^\circ\text{F}$$

8. Through Specimen Clad

$$q = \frac{kA \Delta T}{x}$$

$$\Delta T = \frac{q x}{k A}$$

where $q = 3890$ Btu/hr per specimen

$$A = 2 (1 \times 1/2) 1/144 = 0.00695 \text{ ft}^2$$

$$x = 0.006 \text{ inch} = 0.0005 \text{ foot}$$

$$\Delta T = \frac{(3890) (0.0005)}{(15) (0.00695)}$$

$$= 19^\circ\text{F}$$

Therefore, core surface temperature is approximately

$$1840 + 19 = 1859^\circ\text{F}$$

9. Through Core to Centerline

$$\Delta T = \frac{-q}{2k} (x^2 - Lx) \text{ (Ref. ANL-4690)}$$

where $q = \text{Btu/hr-ft}^3$

$$= \frac{(3890) (1728)}{(1 \times 1/2 \times 0.033)} = 407 \times 10^6 \text{ Btu/hr-ft}^3$$

$k = 15 \text{ Btu/hr-ft}^\circ\text{F}$ (assumed value for the UO_2 -SS dispersion

at $x = \frac{L}{2}$ (Centerline)

$$\begin{aligned} \Delta T &= \frac{-q}{2k} \left(\frac{L^2}{4} - \frac{L^2}{2} \right) \\ &= \frac{q L^2}{8k} \\ &= \frac{(407 \times 10^6) (0.00275)^2}{(8) (15)} \\ &= 26^\circ\text{F} \end{aligned}$$

Therefore, core centerline temperature is:

$$1859 + 26 = 1885^\circ\text{F}$$

E. SHIELDING CALCULATIONS

Previous afterheat calculations indicate that the capsule has the following activities at given decay times:

$$\begin{aligned} 3 \times 10^{13} \text{ MeV/sec after 1 hour decay} \\ 1.08 \times 10^{13} \text{ MeV/sec after 1 day decay} \\ 1.36 \times 10^{12} \text{ MeV/sec after one month decay} \end{aligned}$$

For use in this example, the radiation level at the surface of the water when the capsule is brought to within one foot of the surface of the canal for inspection will be calculated.

1. Point Source

Assume a point source with an activity of $3 \times 10^{13} \text{ MeV/sec}$.

The formula for attenuation from a point source is:

$$\phi = B \frac{S_0}{4\pi a^2} e^{-b_1}$$

All symbols are defined in Section D.1, Chapter IV.

$$B = 1 \text{ (assumed)}$$

$$S_0 = 3 \times 10^{13} \text{ MeV/sec}$$

$$a = 12 \text{ inches} = 30.5 \text{ cm}$$

$$b = \mu t \text{ where } t = 30.5 \text{ cm and } \mu = 0.028 \text{ cm}^{-1}$$

$$\begin{aligned}\phi &= \frac{(1) (3 \times 10^{13}) e^{-30.5(0.028)}}{4 \pi (30.5)^2} \\ &= 2.56 \times 10^9 e^{-0.855} \\ &= 1.1 \times 10^9 \text{ MeV/sec-cm}^2\end{aligned}$$

Line Source

Assume a line source, 13 inches long,

$$S_1 = \frac{3 \times 10^{13}}{(13) (2.54)} = 9.1 \times 10^{11} \text{ MeV/cm-sec}$$

The formula for attenuation from a line source is:

$$\phi = B \frac{S_1}{2 \pi a} f(\theta, b_1)$$

$B = 2$ (from the curve on p 434 in Rockwell, Ref. 1, Chapter IV)

$$S_1 = 9.1 \times 10^{11} \text{ MeV/cm-sec}$$

$$a = 30.5 \text{ cm}$$

$$f(\theta, b_1) \text{ where } b_1 = \mu t = (0.028) (30.5) = 0.855$$

$$\tan \theta = \frac{\ell/2}{a} = \frac{6.5}{12}$$

$$\theta = \tan^{-1} 0.541 = 28.5^\circ$$

From page 385 in Rockwell, Ref. 1, Chapter IV,

$$f(\theta, b_1) = 0.2$$

$$\begin{aligned}\phi &= \frac{(2) (9.1 \times 10^{11}) (0.2)}{2 \pi (30.5)} \\ &= 1.9 \times 10^9 \text{ MeV/sec-cm}^2\end{aligned}$$

This line source calculation is considerably more accurate than the point source calculation because the effect of the buildup factor and the effect of the geometry of the source is taken into consideration. More accurate calculations could be made if the energy of the radiation given off is more carefully considered.

The value given above is correct for one hour decay time. For one day and one month decay, the following results are obtained:

$$\text{One hour decay} = 1.9 \times 10^9 \text{ MeV/sec-cm}^2$$

$$\text{One day decay} = 6.8 \times 10^8 \text{ MeV/sec-cm}^2$$

$$\text{One month decay} = 7.6 \times 10^7 \text{ MeV/sec-cm}^2$$

3. Cask Calculations

It is necessary to determine if a cask with five-inch lead walls is adequate for shipping the sample (mentioned above) after 30 days decay. This calculation will be done as accurately as possible by breaking the fission product radiation into several energies and considering buildup factors, source geometry, and other factors as thoroughly as possible. Four separate calculations will be done in converting source strength to mr per hour, then the total of the four sources will be the calculated radiation level at the outside of the cask.

Reference to Figure 15, page 37, indicates that four energy groups of gamma should be considered. These groups, with their energy level to be used in the calculations and with a source strength obtained by the method of Rockwell (Reference 1, Chapter IV), follow:

0.28 to 0.51 MeV	Use 0.5 MeV	1.25×10^{11} MeV/sec
0.51 to 1.12 MeV	Use 1 MeV	1.13×10^{12} MeV/sec
1.12 to 1.62 MeV	Use 1.5 MeV	7.94×10^{11} MeV/sec
2.3 to 3.5 MeV	Use 3.0 MeV	2.27×10^{10} MeV/sec

(a) 0.5 MeV source

$$S_1 = \frac{1.25 \times 10^{11}}{(13)(2.54)} = 3.78 \times 10^9 \text{ MeV/sec-cm}$$

$$a = 12 \text{ in.} = 30.5 \text{ cm}$$

$$\mu = 0.15 \text{ cm}^2/\text{g} \times 11.3 \text{ g/cm}^3 = 1.70 \text{ cm}^{-1}$$

$$t = 5 \text{ in.} = 12.7 \text{ cm}$$

$$b = \mu T = 12.7 \times 1.70 = 21.6$$

$$B = 2.8$$

$$\theta = \tan^{-1} \frac{6.5}{12} = \tan^{-1} 0.541 = 28.5^\circ$$

$$f(\theta, b) = 1 \times 10^{-10}$$

$$\phi = \frac{(2.8)(3.78 \times 10^9)(10^{-10})}{2\pi(30.5)} = 5.52 \times 10^{-3} \frac{\text{MeV}}{\text{cm}^2 \text{ sec}} = \frac{5.52 \times 10^{-3}}{4.8 \times 10^5}$$

$$= 1.15 \times 10^{-8} \frac{\text{R}}{\text{hr.}}$$

(b) 1.0 MeV source

$$S_1 = \frac{1.13 \times 10^{12}}{(13)(2.54)} = 3.42 \times 10^{10} \text{ MeV/sec-cm}$$

$$a = 30.5 \text{ cm}$$

$$\mu = (0.068)(11.3) = 0.768 \text{ cm}^{-1}$$

$$t = 12.7 \text{ cm}$$

$$b = \mu t = 9.75$$

$$B = 3.7$$

$$\theta = 28.5^\circ$$

$$f(\theta, b) = 2 \times 10^{-5}$$

$$\phi = \frac{(3.7) (3.42 \times 10^{10}) (2 \times 10^{-5})}{2 \pi (30.5)}$$

$$= 1.32 \times 10^4 \text{ MeV/sec-cm}^2$$

$$= \frac{1.32 \times 10^4}{5.2 \times 10^5} = 0.025 \text{ R per hour}$$

(c) 1.5 MeV source

$$S_1 = \frac{7.94 \times 10^{11}}{(13) (2.54)} = 2.40 \times 10^{10} \frac{\text{MeV}}{\text{cm sec}}$$

$$a = 30.5 \text{ cm}$$

$$\mu = (0.050) (11.3) = 0.565 \text{ cm}^{-1}$$

$$t = 12.7 \text{ cm}$$

$$b = \mu t = 7.17$$

$$B = 3.4$$

$$\theta = 28.5^\circ$$

$$f(\theta, b) = 2.8 \times 10^{-4}$$

$$\phi = \frac{(3.4) (2.40 \times 10^{10}) (2.8 \times 10^{-4})}{2 \pi (30.5)}$$

$$\phi = 1.19 \times 10^5 \text{ MeV/sec-cm}^2$$

$$= \frac{1.19 \times 10^5}{5.7 \times 10^5} = 0.209 \text{ R/hr}$$

(d) 3.0 MeV source

$$S_1 = \frac{2.27 \times 10^{10}}{(13) (2.54)} = 6.88 \times 10^8 \text{ MeV/sec-cm}$$

$$a = 30.5 \text{ cm}$$

$$\mu = (0.041) (11.3) = 0.463 \text{ cm}^{-1}$$

$$t = 12.7 \text{ cm}$$

$$b = \mu t = 5.88$$

$$B = 3.3$$

$$\theta = 28.5^\circ$$

$$f(\theta, b) = 1.25 \times 10^{-3}$$

$$\phi = \frac{(3.3) (6.88 \times 10^3) (1.25 \times 10^{-3})}{2 \pi (30.5)}$$

$$\phi = 1.48 \times 10^4 \text{ MeV/sec-cm}$$

$$\phi = \frac{1.48 \times 10^4}{7 \times 10^5} = 0.021 \text{ R/hr}$$

Total radiation level from the cask

$$0.5 \text{ MeV source} = 0.000$$

$$1.0 \text{ MeV source} = 0.025$$

$$1.5 \text{ MeV source} = 0.209$$

$$3.0 \text{ MeV source} = \underline{0.021}$$

$$\text{Total} = 0.255 \text{ R/hr}$$

Since this value is greater than the limit of 200 mR/hr allowable, the cask is not adequate. Therefore, either longer decay or thicker shielding is required.

APPENDIX B

SAMPLE LOOP EXPERIMENT CALCULATIONS

Appendix B consists of a sample problem which incorporates many of the engineering considerations mentioned in the preceding chapter. Information in the chapters on nuclear fluid flow, heat transfer, and shielding will be applied on this problem.

The problem sample consists of nineteen rods of natural uranium oxide clad with zirconium located in a high pressure water loop. The problem solution consists of picking maximum and average neutron fluxes, calculating perturbation factors, calculating heat generation and heat flux, then studying the problem of transferring the heat to the water.

Operating curves, consisting of plots of burnout heat flux and sample surface temperature versus flow at various water temperatures, will be used to choose setpoints. Additional information will consist of loop flow coastdown curves and hot spot-hot channel analysis of the system.

A. NUCLEAR CALCULATIONS

1. Fuel Content

Data given with the experiment indicate that each of the nineteen rods contains 31.0 grams of natural uranium per inch and that the rods have 34 inches of fuel. The amount of U-235 per inch of rod must be determined, and a perturbation factor calculated for the rods.

Natural uranium contains 0.7 percent U-235, the remainder being U-238. Previous tests have shown that plutonium production and burnup from irradiation of U-238 will not add an appreciable amount of power to that produced from U-235 fissioning and reactor gamma heat. U-235 per inch then becomes 0.007×31 or 0.217 g/in. This is a total of 7.38 grams per rod or a grand total in the sample of 140.2 grams.

2. Neutron Flux

Peak and average values of effective neutron flux in the position used by this experiment must be determined, as well as the proper values of fission cross section for the U-235.

Neutron flux measurements have been made for similar samples for a previous cycle. The flux wires were positioned around the outside of the loop pressure tube. Peak centerline thermal fluxes around the outside of the tube ranged from 5.25×10^{14} to 2.0×10^{14} and averaged $3.64 \times 10^{14} \frac{\text{neutron}}{\text{cm}^2 \text{ sec}}$. Neutron attenuation through the pressure tube has been measured at ETRC and, for this position, was 0.55, so the maximum effective neutron flux in the area of the sample is $2.0 \times 10^{14} \text{ n/cm}^2\text{-sec}$.

The ratio between the peak flux and average flux was determined to be 1.2. Average flux over the fuel element length would then be:

$$\phi_{\text{avg}} = \phi_{\text{max}} \left(\frac{1}{1.2} \right) = \phi_{\text{max}} (0.833)$$

The neutron attenuation through the pressure tube can be calculated provided the effective neutron absorption coefficient for the pressure tube is known. However, direct measurements are generally preferred in order to minimize uncertainties in the calculations.

3. Flux Perturbation

The flux perturbation due to the U-235 in the fuel rods must also be calculated. The methods used in Reference 2, Chapter I of this volume, are used, although very accurate determination requires computer calculations. The flux perturbation was calculated to be 0.10.

Previous computer calculations and measurements have shown that the effective thermal neutron fission cross section in the environment inside this pressure tube (stainless steel, zirconium, uranium, and water at more than 500°F) is 452 barns. This compares with a cross section of 582 barns used for thermal neutrons at 20°C.

4. Power Generation Calculations

Power generation can now be calculated. The short formula 60-10 is not used because of the corrected value of fission cross section used in this calculation.

$$P_f = (1-p) \phi \sigma_f N k$$

where P_f = fission power, kW/rod

p = perturbation factor in fuel = 0.10

ϕ = effective thermal neutron flux = 2.0×10^{14} n/cm²-sec.

σ_f = effective fission cross section in barns = 452
or 452×10^{-24} cm²

N = number of atoms of U-235

$$= \frac{\text{grams of material} \times \text{Avagadro's number}}{\text{atomic weight}}$$

$$= \frac{7.38 \times 6.023 \times 10^{23}}{235}$$

$$= 1.89 \times 10^{22} \text{ atoms U-235}$$

k = constant to convert fission/sec to kilowatts

$$= \left(\frac{1}{3.1 \times 10^{13}} \right) \left(\frac{\text{kW}}{\text{fission/sec}} \right)$$

$$P_f = (1-0.10) (2 \times 10^{14}) (4.52 \times 10^{-22}) (1.89 \times 10^{22}) \left(\frac{1}{3.1 \times 10^{13}} \right)$$

$$= 49.6 \frac{\text{kW}}{\text{rod}} = \left(\frac{49.6 \text{ kW}}{\text{rod}} \right) \left(\frac{\text{rod}}{34 \text{ in.}} \right) \left(\frac{12 \text{ in.}}{\text{ft}} \right)$$

$$= 17.5 \text{ kW/ft maximum power generation rate}$$

$$P_f \text{ average} = 17.5 \times 0.833 = 14.6 \text{ kW/ft}$$

$$P_f \text{ per rod} = 14.6 \times \frac{34}{12} = 41.3 \text{ kW/rod}$$

$$P_f \text{ per assembly} = 41.3 \times 19 = 786 \text{ kW}$$

5. Gamma Heating

Gamma heating is measured at 20 watts per gram at reactor vertical centerline. This value will also be used as the average rate over the length of the sample, a conservative assumption. According to the sponsor's information, total weight of a fuel rod is 437 grams per foot, or 1540 grams per rod.

$$\begin{aligned} P_e &= (\text{grams of material}) (\text{gamma heating rate, watts/gram}) \\ &= (437) (20) (10^{-3}) \\ &= 8.74 \text{ kW/foot} \end{aligned}$$

or

$$\begin{aligned} P_e &= 1540 \times 20 \times 10^{-3} \\ &= 30.8 \text{ kW/rod.} \end{aligned}$$

6. Conversion to Btu/hr

In order to use these values of power generation for heat transfer calculations, they must be converted to Btu/hour. The conversion factor is 1 kilowatt equals 3416 Btu per hour. Therefore,

$$\text{Average total power generation rate per foot} = (14.6 + 8.74) (3416)$$

$$= 7.96 \times 10^4 \text{ Btu/hr-ft}$$

$$\text{total power generation per rod} = (7.96 \times 10^4) \left(\frac{34}{12}\right)$$

$$= 2.26 \times 10^5 \text{ Btu/hr}$$

$$\text{total power generation} = (2.26 \times 10^5) (19)$$

$$= 4.28 \times 10^6 \text{ Btu/hr}$$

$$\text{peak power generation rate per rod} = (17.5 + 8.74) (3416)$$

$$= 8.96 \times 10^4 \text{ Btu/hr-ft.}$$

7. Nuclear Afterheat

Nuclear afterheat at various times after a reactor power reduction is needed in order to determine when the sample can be safely stored without forced convection cooling and for making shielding calculations to assess hazards in handling. For this purpose, the Wey-Wigner equation will be used to determine nuclear afterheat one minute, one hour, and one day after the reactor power reduction. It is assumed that the sample operated for 30 days at full reactor power. The Equation 63.3 for gamma power will be used.

$$P_G = 0.0315 P \left[t^{-0.2} - (t + t_o)^{-0.2} \right], \quad (63.3)$$

where P_G = gamma power in watts

$$P = \text{operating power of source in watts} = \frac{4.28 \times 10^6}{3416} = 1.25 \times 10^6 \text{ watts}$$

t = time after shutdown in seconds

$$t_o = \text{operating time in seconds} = (30) (24) (60) (60) = 2.592 \times 10^6 \text{ sec}$$

for one minute decay:

$$t = (1) (60) = 60 \text{ sec}$$

$$\begin{aligned} P_G &= (0.0315) (1.25 \times 10^6) \left[(60)^{-0.2} - (2.592 \times 10^6)^{-0.2} \right] \\ &= (3.94 \times 10^4) (0.441 - 0.052) = (3.94 \times 10^4) (0.389) = 1.53 \times 10^4 \text{ watts} \end{aligned}$$

for one hour decay:

$$t = (1) (60) (60) = 3600 \text{ sec}$$

$$\begin{aligned} P_G &= (0.0315) (1.25 \times 10^6) \left[(3600)^{-0.2} - (2.596 \times 10^6)^{-0.2} \right] \\ &= (3.94 \times 10^4) (0.194 - 0.052) = (3.94 \times 10^4) (0.142) \\ &= 0.560 \times 10^4 \text{ watts} \end{aligned}$$

for one day decay:

$$t = (1) (24) (60) (60) = 86,400 \text{ sec}$$

$$\begin{aligned} P_G &= (0.0315) (1.25 \times 10^6) \left[(86,400)^{-0.2} - (2.678 \times 10^6)^{-0.2} \right] \\ &= (3.94 \times 10^4) (0.103 - 0.052) = (3.94 \times 10^4) (0.051) \\ &= 0.201 \times 10^4 \text{ watts} \end{aligned}$$

B. HYDRAULIC CALCULATIONS

Flow tests were run on the experimental assembly to determine the flow distribution through the fuel samples in the fuel basket and through the process tube-fuel basket annulus (see Figure B-1). The flow split was determined to be 35 gpm through the annulus and 125 gpm through the sample. Total requested flow was 160 gpm. The differential pressure across the assembly was measured at 29.3 psi at 50°F, corresponding to 15.1 psi at 450°F.

The pressure drop and flow distribution for various total flows and temperatures can be calculated. However, because of the complicated shape of the flow channels, the length and various small structural members which partially block flow, actual measurement of all hydraulics parameters is simpler, quicker, and more accurate.

Two calculated parameters must be determined. These are equivalent diameter (D_e) and average water velocity (V).

Fig. B-1 Section of in-pile tube and sample.

Velocity calculations at 450°F can be made as follows:

$$\begin{aligned}\text{flow} &= 125 \text{ gpm} = \frac{125 \text{ gpm}}{7.48 \text{ gal/ft}^3} \\ &= 16.7 \text{ cu ft/min} \\ &= \frac{16.7}{60} = 0.279 \text{ cu ft/sec}\end{aligned}$$

Flow area inside basket = basket inside area -- cross sectional area of elements

$$\begin{aligned}&= (3.25)^2 \left(\frac{\pi}{4}\right) - (0.562)^2 \left(\frac{\pi}{4}\right) (19) \\ &= 8.3 \text{ in.}^2 - 4.72 \text{ in.}^2 \\ &= 3.58 \text{ in.}^2 = \frac{3.58 \text{ in.}^2}{144 \text{ in.}^2/\text{ft}^2} \\ &= 2.49 \times 10^{-2} \text{ ft}^2\end{aligned}$$

$$\begin{aligned}\text{Average velocity inside basket} &= \frac{\text{flow}}{\text{flow area}} \\ &= \frac{0.279 \text{ ft}^3/\text{sec}}{2.49 \times 10^{-2} \text{ ft}^2} \\ &= 11.2 \text{ ft/sec}\end{aligned}$$

wetted perimeter = basket inner perimeter + outer perimeter of nineteen rods

$$\begin{aligned}&= 3.25 \pi + (0.562 \pi) (19) \\ &= (10.2 + 33.5) \text{ in.} \\ &= 3.64 \text{ ft}\end{aligned}$$

$$\begin{aligned}\text{equivalent diameter} &= \frac{(4) (\text{flow area})}{\text{wetted perimeter}} \\ &= \frac{(4) (2.49 \times 10^{-2})}{3.64} \\ &= 2.74 \times 10^{-2} \text{ ft} \\ &= 0.329 \text{ inches.}\end{aligned}$$

C. HEAT TRANSFER CALCULATIONS

1. Summary of Information

In previous sections, the following pertinent facts which will be used in heat transfer analysis have been determined:

$$\text{total power generation} = 4.28 \times 10^6 \text{ Btu/hr}$$

$$\text{peak power generation rate} = 8.96 \times 10^4 \text{ Btu/hr-ft on one rod}$$

$$D_e = 0.329 \text{ inches}$$

$$V = 11.2 \text{ ft/sec.}$$

2. Bulk Water ΔT

Bulk water ΔT will be determined from total power generation.

$$Q = 4.28 \times 10^6 \text{ Btu/hr}$$

$$F = \text{flow rate} = 0.279 \text{ ft}^3/\text{sec}$$

$$\rho = \text{density of water at } 450^\circ\text{F}$$

$$= 51.5 \text{ pounds per cubic foot}$$

$$c_p = \text{specific heat at } 450^\circ\text{F}$$

$$= 1.110 \text{ Btu/lb-}^\circ\text{F}$$

$$\begin{aligned} \Delta T \text{ Bulk} &= \frac{\text{Power Generation}}{(\text{flow rate}) (\text{density}) (\text{specific heat})} \\ &= \frac{\text{Btu/hr}}{(\text{ft}^3/\text{hr}) (\text{lb/ft}^3) (\text{Btu/lb-}^\circ\text{F})} \\ &= \frac{4.28 \times 10^6}{(0.279) (3600) (51.5) (1.110)} \\ &= 74.5^\circ\text{F.} \end{aligned}$$

3. Peak Heat Flux

$$\text{Peak power generation} = 8.96 \times 10^4 \text{ Btu/hr-ft-rod}$$

$$\text{Heat transfer area per foot} = \frac{0.562 \pi}{12}$$

$$= 0.147 \text{ ft}^2/\text{ft}$$

$$\text{peak heat flux} = \frac{8.96 \times 10^4}{0.147}$$

$$Q/A \text{ Peak} = 6.09 \times 10^5 \text{ Btu/hr-ft}^2$$

4. ΔT Film

The Modified Colburn equation is used for calculating the maximum expected fuel element wall temperature

$$\frac{hD}{k} = 0.023 \left(\frac{DV\rho}{\mu} \right)^{0.8} \left(\frac{C_{\mu}}{k} \right)^{0.3}$$

$$b = 0.023 \frac{(3600)^{0.8} (12)^{0.2} k^{0.7} \rho^{0.8} C^{0.3}}{\mu^{0.5}}$$

$$K = V^{0.8}/D^{0.2}$$

$$h = bK \text{ Btu/hr-ft}^2\text{-}^\circ\text{F}$$

For a calculation of wall temperature at the lower part of the element, assume the heat flux at the point of maximum bulk water temperature is $0.72 \times$ peak heat flux, or $4.4 \times 10^5 \text{ Btu/hr-ft}^2$ and an inlet temperature of 400°F .

$$\begin{aligned} T_{\text{bulk}} &= T_{\text{in.}} + \Delta T_{\text{bulk}} \\ &= 400 + 74.5 = 474.5^\circ\text{F} \end{aligned}$$

Assume a ΔT_{film} of 100°F . T_{film} would then be $474.5 + \frac{100}{2}$ or 524.5°F from Figure 11, page 27, look up the base factor for a T_{film} of 524.5°F .

$$\begin{aligned} b &= 580 \\ K &= \frac{V^{0.8}}{D_e^{0.2}} \\ &= \frac{11.2^{0.8}}{0.329^{0.2}} \end{aligned}$$

Note: For the purposes of using this curve only, equivalent diameter is expressed in inches and velocity in feet per second. For the standard form of the correlation, velocity must be in feet per hour and equivalent diameter in feet.

$$\begin{aligned} K &= \frac{6.9}{0.801} = 8.61 \\ h &= bK = 580 \times 8.61 = 4990 \text{ Btu/hr-ft}^2\text{-}^\circ\text{F} \\ \Delta T_{\text{film}} &= \frac{Q/A}{h} = \frac{4.4 \times 10^5}{4.99 \times 10^3} = 88^\circ\text{F}. \end{aligned}$$

The value of ΔT_{film} used for determining b was 100°F . Assume a ΔT_{film} of 80°F , and look up b for a film temperature of 514.5°F .

$$\begin{aligned} b &= 585 \\ K &= 8.61 \\ h &= bK = 585 \times 8.61 = 5040 \\ \Delta T_{\text{film}} &= \frac{4.4 \times 10^5}{5.04 \times 10^3} = 87^\circ\text{F} \end{aligned}$$

The value of b assumed is close enough. Therefore,

$$T_{\text{metal}} = 400 + 74.5 + 87 = 561.5$$

For a calculation of wall temperature at the point of highest heat flux, assume this point is two-thirds of the distance from the top of the element and that the peak heat flux is that calculated above.

$$\Delta T_{\text{bulk}} = 2/3 \times 74.5 = 50^\circ\text{F}$$

$$Q/A_{\text{peak}} = 6.09 \times 10^5 \text{ Btu/hr-ft}^2$$

$$T_{\text{bulk}} = 400^\circ\text{F} + 50 = 450^\circ\text{F}$$

Assume a ΔT_{film} of 100°F . T_{film} would then be $450 + \frac{100}{2}$ or 500°F from Figure 11

$$b = 590$$

$$K = 8.75$$

$$h = bK = 590 \times 8.75 = 5160$$

$$\Delta T_{\text{film}} = \frac{Q/A}{h} = \frac{6.09 \times 10^5}{5160} = 118^\circ\text{F} .$$

Assume a ΔT_{film} of 120°F , and determine b for a film temperature of 510°F

$$b = 587$$

$$h = (587) (8.75) = 5140$$

$$\Delta T_{\text{film}} = \frac{6.09 \times 10^5}{5140} = 118^\circ\text{F}$$

$$T_{\text{metal}} = 450 + 118 = 568^\circ\text{F} .$$

Therefore, it is fairly certain that the region of highest metal temperature will be in the region of highest heat flux rather than in the region of highest bulk water temperature. This is normally the case.

5. Safe Operating Curves

A curve, Figure B-2, is now calculated and plotted showing metal temperature versus water flow rate for bulk water temperature ranging from 200 to 500°F for total flow rates from 80 to 160 gpm. From this curve, the engineer is able to determine setpoints which will give maximum operating flexibility and adequately protect the experiment from accidental loss of flow or heat exchanger capacity. Note that wall temperature for the experiment can be determined quickly and accurately for any combination of flow and bulk water temperature.

Note that uncertainty factors are not shown on this set of curves. These are determined by a hot spot-hot channel analysis similar to that given in Appendix D and are used in the calculation of the specimen surface temperature which could result if all the uncertainties reached their maximum values within the experimental system.

6. Burnout Calculations

In this particular experiment, wall temperature of the specimen is of no real concern. Therefore, the experiment can be considered adequately protected if it

Fig. B-2 Specific surface temperature versus total loop flow.

does not approach burnout conditions. It would, therefore, be desirable to calculate burnout heat flux for a large combination of flows and temperature in order to aid the engineer in choosing adequate setpoints.

This experiment assembly has a very odd geometry, which really is not covered by the Jacobs-Merrill or WAPD-188 burnout correlations. For this reason, no curve of burnout heat flux versus temperature was made. However, in order to get some idea of burnout heat fluxes, calculations made using the WAPD-188 correlation at bulk water temperature flow conditions picked for normal operation, low flow, and high bulk water temperature annunciator conditions, and low flow and high bulk water reactor power reduction conditions.

The WAPD-188 design correlation for burnout for round tubes is:

$$\frac{\phi_{BO}}{10^6} = 0.28 \left(\frac{H}{10^3} \right)^{-2.5} \left(1 + \frac{G}{10^7} \right)^2 e^{-0.0012 L/D}$$

where ϕ_{BO} is the burnout heat flux in units of Btu/hr-ft². This correlation is applicable over the following range of variables.

Enthalpy at the burnout point (H): $500 \leq H \leq 1000$ Btu/lb

Mass velocity (G): $G/10^6 \geq 0.2$ lb/hr-ft² for round tubes

Pressure: $1850 \leq P \leq 2150$ psia

Length to diameter (L/D) ratio, round tubes: (data available $21 \leq L/D \leq 365$).

The design equation is equivalent to the correlation multiplied by a factor of 0.65 and is always used. This factor (0.65) was chosen so that 98 percent of the experimental heat fluxes would be higher than the values predicted by the design equations for comparable fluid conditions.

Example calculations:

$$\frac{\phi_{BO}}{10^6} = (0.28) (0.65) \left(\frac{H_{BO}}{10^3} \right)^{-2.5} \left(1 + \frac{G}{10^7} \right)^2 e^{-0.0012 L/D}$$

H_{BO} = bulk enthalpy at burnout point

G = velocity in pounds per hour-ft²

$$= \left(\text{gpm} \right) \left(60 \frac{\text{min}}{\text{hr}} \right) \left(\frac{\text{ft}^3}{\text{gal}} \right) \left(\frac{\text{lbs}}{\text{ft}^3} \right) \left(\frac{144 \text{ in.}^2}{\text{ft}^2} \right) \left(\frac{1}{\text{Flow area (in.}^2\text{)}} \right)$$

$$= \left(\text{gpm} \right) \left(60 \right) \left(0.1337 \right) \left(\frac{\text{lbs}}{\text{ft}^3} \right) \left(144 \right) \left(\frac{1}{3.58} \right)$$

$$G = \left(\text{gpm} \right) \left(\frac{\text{lb}}{\text{ft}^3} \right) (322).$$

Burnout at Normal Operating Conditions:

Sample surface temperature at normal operating conditions = 568°F

H_{BO} at 568°F = 572.9

G at 125 gpm past fuel = $125 \times 51.5 \times 322 = 2.07 \times 10^6$ lbs/hr-ft²

$$\frac{\phi_{BO}}{10^6} = (0.182) \left(\frac{572.9}{10^3} \right)^{-2.5} \left(1 + \frac{2.07 \times 10^6}{10^7} \right)^2 \left[e^{-0.0012 \left(\frac{22.7}{0.563} \right)} \right]$$

$$= (0.182) (4.03) (1.46) (0.953) = 1.02$$

$$\phi_{BO} = 1.02 \times 10^6 \text{ Btu/hr-ft}^2$$

Burnout at Annunciator Conditions:

G at 135 gpm total flow (100 gpm past fuel) = (100) (51.5) (322)

$$= 1.66 \times 10^6 \text{ lbs/hr-ft}^2$$

H at surface temperature of 590°F (at 100 gpm past fuel) = 602.9

$$\frac{\phi_{BO}}{10^6} = (0.182) \left(\frac{602.9}{10^3} \right)^{-2.5} \left(1 + \frac{1.66 \times 10^6}{10^7} \right)^2 (0.953)$$

$$= (0.182) (3.54) (1.36) (0.953) = 0.836$$

$$\phi_{BO} = 0.836 \times 10^6 \text{ Btu/hr-ft}^2$$

Burnout at Jr. Scram Conditions:

G at 115 gpm total flow (80 gpm past fuel) = (80) (51.5) (322) = 1.33×10^6
lbs/hr-ft²

H at surface temperature of 618°F (at 80 gpm past fuel) = 643.7

$$\frac{\phi_{BO}}{10^6} = (0.182) \left(\frac{643.7}{10^3} \right)^{-2.5} \left(1 + \frac{1.33 \times 10^6}{10^7} \right)^2 (0.953)$$

$$= (0.182) (3.02) (1.28) (0.953) = 0.671$$

$$\phi_{BO} = (0.671) (10^6) = 0.671 \times 10^6 \text{ Btu/hr-ft}^2$$

Burnout at Normal Operating Conditions Including Hot-Spot Hot-Channel Factors:

$F_h (\Delta T \text{ film}) + (2.17) (118) = 256^\circ\text{F}$; $F_b (\Delta T \text{ bulk}) = (1.75) (74.5) = 130^\circ\text{F}$ (2/3) (130)
= 87 °F

G at 160 gpm total flow (125 gpm past fuel) (density at 487°F = $49.7 \frac{\text{lb}}{\text{ft}^3}$)

$$G = (125) (322) (49.7) = 2.00 \times 10^6 \text{ lbs/hr-ft}^2$$

H at 487 + 256 = H at 743°F surface temperature = 1285

$$\begin{aligned} \frac{\phi_{\text{BO}}}{10^6} &= (0.182) \left(\frac{1285}{10^3} \right)^{-2.5} \left(1 + \frac{2.00 \times 10^6}{10^7} \right)^2 (0.953) \\ &= (0.182) (0.534) (1.44) (0.953) = 0.133 \end{aligned}$$

$$\phi_{\text{BO}} = (0.133) (10^6) = 0.133 \times 10^6 \text{ Btu/hr-ft}^2$$

You will notice that when the hot spot and hot channel factors are applied, the specimen surface temperature exceeds the saturation temperature; and therefore, boiling will occur. The enthalpy of steam which corresponds to a surface temperature, such as is found at TRA (745°F), is approximately 1215 Btu/lb, which exceeds the limitations of the WAPD-188 burnout correlation. Therefore, the burnout heat flux calculated for the normal operating conditions with the hot spot-hot channel factors applied is not valid.

The other burnout equation which is often used to calculate burnout heat fluxes is the Jacobs-Merrill burnout heat flux equation. This method is an application of statistical methods applied to the subcooled burnout data published by UCLA, Purdue, BMI, and McAdams to obtain a single correlation relationship of the polynomial type. Tests of significance were used for each variable in the correlation polynomial to make sure they add to the ability to predict burnout.

The Jacobs-Merrill correlation is as follows:

$$\begin{aligned} y &= 0.03332 - 0.06988X_1 + 0.1638X_2 + 0.5319X_3 - 0.2311X_4 \\ &\quad - 0.006663X_2^2 - 0.03205X_3^2 + 0.01211X_2X_3 \end{aligned}$$

where $y = q''_{\text{BO}} \times 10^{-6}$

$$q''_{\text{BO}} = \frac{q}{A}, \frac{\text{Btu}}{\text{hr-ft}^2}$$

$$X_1 = \text{Pressure, } \frac{\text{psi}}{1000}$$

$$X_2 = \text{Temperature, } \frac{\Delta T_{\text{sub}}}{10}$$

$$X_3 = \text{Mass Velocity, } \frac{\text{lb} \times 10^{-6}}{\text{hr-ft}^2}$$

$$X_4 = \text{Length, } \frac{\text{inches}}{10}$$

Again it is pointed out that this polynomial type correlation was derived from existing subcooled burnout data. This correlation should not be used for geometries of flow cross sections other than circular.

This correlation is applicable over the following range of variables:

Pressure, (psi): 30-30,000

Heated Length (inches): 3.0 - 27

T Sub (°F): 6 - 163

Velocity ($G \times 10^{-6}$): 0-23 - 7.8

The results calculated by the Jacobs-Merrill correlation agree quite well with those calculated by the WAPD-188 correlation and seem to be quite reliable.

APPENDIX C

GAS SYSTEM HEAT TRANSFER ANALYSIS

A typical gas system heat transfer analysis is presented. In this system, a cylindrical fueled specimen (Figure C-1) is cooled by air flowing parallel to the specimen surface at a rate of 3170 lb/hr. The air is contained in the annular space between the test specimen and a stainless steel flow tube that is somewhat cooler than the specimen. The test specimen has an outside diameter of 2.0 inches and a wall temperature of 1300°F. The stainless tube has an outside diameter of 3.5 inches, a wall thickness of 0.083 inch, and a wall temperature of 968°F. The air stream temperature is approximately 900°F at the inlet. The flow tube is surrounded by helium contained in an aluminum pressure tube having an outside diameter of 5.563 inches, a wall thickness of 0.258 inch, and a wall temperature of 140°F. This is not intended to be a complete heat transfer analysis but only a demonstration of the methods for each type of transfer.

For convective heat transfer from specimen wall to gas use:

$$Q_{\text{conv.}} = h_c A_s \Delta T$$

where

$$\Delta T = T_1 - T_3 = 1300 - 900 = 400^\circ\text{F}$$

$$A_s = \text{surface area of tube} = \pi (2.0) (12) = 75.5 \text{ in.}^2/\text{ft} = 0.524 \text{ ft}^2/\text{ft}$$

$$A_f = \text{flow area} = \frac{\pi}{4} [(3.334)^2 - (2.0)^2] = \pi/4 (7.12) = 5.59 \text{ in.}^2 = 0.0388 \text{ ft}^2$$

$$h_c = 0.023 \left(\frac{k}{D_e} \right) \left(\frac{D_e V_\rho}{\mu} \right)^{0.8} \left(\frac{c_p \rho}{k} \right)^{0.4}$$

$$= 0.023 \left(\frac{1}{D_e^{0.2}} \right) \left(\frac{G}{A_f} \right)^{0.8} \left(\frac{k}{\mu^{0.8}} \right) \left(\frac{c_p \mu}{k} \right)^{0.4}$$

The constants are evaluated at the mean film temperature of $\frac{1300 + 900}{2}$ or 1100°F

$$k = 3.09 \times 10^{-2} \frac{\text{Btu}}{\text{hr ft } ^\circ\text{F}}$$

$$c_p = 0.270 \frac{\text{Btu}}{\text{lb } ^\circ\text{F}}$$

$$\mu = 8.21 \times 10^{-2} \frac{\text{lb}}{\text{hr ft}}$$

$$D_e = D_1 - D_2 = 3.334 - 2.00 = 1.33 \text{ in.} = 0.111 \text{ ft}$$

$$h_e = (0.023) \left(\frac{1}{0.111} \right)^{0.2} \left(\frac{3170}{0.0388} \right)^{0.8} \left[\frac{0.0309}{(0.0821)^{0.8}} \right] \left[\frac{(0.270)(0.0821)}{0.0309} \right]^{0.4}$$

$$= (0.023) (9.00)^{0.2} (81,700)^{0.8} \left[\frac{0.0309}{(0.0821)^{0.8}} \right] (0.717)^{0.4}$$

Fig. C-1 Cross section of a cylindrical fueled specimen.

$$= \frac{(0.023) (1.55) (8510) (0.0309) (0.875)}{0.134} = 61.2 \frac{\text{Btu}}{\text{hr-ft}^2-\text{°F}}$$

$$Q_{\text{conv.}} = (61.2) (0.524) (400) = 12,820 \frac{\text{Btu}}{\text{hr-ft}}$$

For radiative heat transfer between walls of concentric pipes use:

$$Q_R = \delta F_e A_1 (T_1^4 - T_2^4)$$

$$\text{where } \delta = \frac{\text{Stefan-Boltzman constant}}{\frac{\text{Btu}}{\text{ft}^2\text{-hr-deg}^4}} = 5.672 \times 10^{-5} \frac{\text{ergs}}{\text{cm}^2\text{-sec-deg}^4} = 1.8 \times 10^{-8}$$

$$F_e = \frac{1}{\frac{1}{P_1} + \frac{A_1}{A_2} \left(\frac{1}{P_2} - 1 \right)}$$

$$A_1 = \text{surface area of inner tube, ft}^2$$

$$A_2 = \text{surface area of outer tube, ft}^2$$

$$T_1 = \text{absolute temperature of inner tube, °R}$$

$$T_2 = \text{absolute temperature of outer tube, °R}$$

$$P_1 = \text{emissivity factor of inner surface}$$

$$P_2 = \text{emissivity factor of outer surface}$$

Assume that $P_1 = P_2$ for stainless steel surfaces and that $P = 0.75$. This value is conservative for 18-7 stainless steels [a].

$$\therefore F_e = \frac{1}{\frac{1}{0.75} + \frac{2.0}{3.334} \left(\frac{1}{0.75} - 1 \right)} = \frac{1}{1.333 + 0.8 - 0.6} = \frac{1}{1.533} = 0.652$$

$$\begin{aligned} Q_R &= (1.8 \times 10^{-8}) (0.652) (0.524) \left[(1.760)^4 - (1.428)^4 \right] 10^{12} \\ &= (1.8 \times 10^{-8}) (0.652) (0.524) (5.45 \times 10^{12}) = 33,500 \frac{\text{Btu}}{\text{hr-ft}} \end{aligned}$$

$$Q_{\text{total}} = Q_c + Q_R = 12,820 + 33,500 = 46,320 \frac{\text{Btu}}{\text{hr-ft}} = 556,000 \frac{\text{Btu}}{\text{hr}} \text{ per linear foot}$$

For conductive heat transfer between walls of concentric pipes use:

$$Q_{\text{cond.}} = \frac{kAm\Delta T}{\ell}$$

[a] Brown and Marco, "Introduction to Heat Transfer", Chapter IV, Articles 4-6, Table 4-1.

where: $k = 0.125 \frac{\text{Btu}}{\text{hr-ft-}^\circ\text{F}}$ at $T_{\text{mean}} = \frac{968 + 140}{2} = \frac{1108}{2} = 554^\circ\text{F}$

$$A_m = \text{mean area perpendicular to heat flow} = \left(\frac{5.047 + 3.5}{2} \right) (12)$$

$$= \pi (4.273) (12) = 161.2 \text{ in.}^2/\text{ft} = 1.12 \text{ ft}^2/\text{ft}$$

$$\Delta T = T_{\text{inside}} - T_{\text{outside}} = 968 - 140 = 828^\circ\text{F}$$

$$x = \text{thickness of gas} = 5.047 - 3.5 = 1.547 \text{ in.} = 0.129 \text{ ft}$$

$$Q_{\text{cond.}} = \frac{(0.125) (1.12) (828)}{0.129} = 899 \text{ Btu/hr-ft}$$

APPENDIX D

HOT-SPOT, HOT-CHANNEL ANALYSIS FOR A FUELED TEST ASSEMBLY

A typical hot-spot, hot-channel analysis for a fueled test assembly in a high pressure water loop is presented. This analysis presents the most extreme conditions and makes no attempt to predict probabilities. Explanation of this type of analysis is given in Chapter 29 of Volume 2 and in IDO-16343 [a].

The analysis considers a rectangular assembly which has a width of 1.075 ± 0.003 inch and is 8 inches long. The assembly contains five fuel plates suspended from the top of the assembly. The plates are each 0.048 inch thick, 1.045 inch wide, and have a fueled portion 0.800 inch wide. Five of the six channels formed by the plates and the assembly have a thickness of $0.085 \begin{smallmatrix} + 0.000 \\ - 0.007 \end{smallmatrix}$ inch while the remaining channel has a thickness of $0.115 \begin{smallmatrix} + 0.000 \\ - 0.007 \end{smallmatrix}$ inch. The average temperature of the system is assumed to be approximately 500°F. The plates have a cladding thickness of 0.015 inch with a local deviation of ± 0.005 inch and an average deviation of ± 0.002 inch.

A. MECHANICAL TOLERANCES (Hydraulic Effects)

(1) and (2) -- Effect on mass flow due to external, entrance, and exit effects. The flow is measured directly; therefore,

$$F_h = 1.0$$

$$F_b = 1.0$$

(3) and (4) -- Effect on mass flow due to channel effects and variation in equivalent diameter. The worst case situation for uniform flow distribution is the one in which four small channels remain at their regular value while one small channel changes to its minimum possible dimensions and the large channel exists at its maximum value. Since the length of each channel in the tier is short (8 inches), the entrance and exit losses are a significant proportion of the total tier pressure drop. However, the existing entrance and exit losses tend to diminish the flow redistribution between the nonuniform parallel channels described above. Thus, a calculation of the flow distribution in the six channels which does not consider the effect of entrance and exit losses will yield a conservative result because the calculated degree of flow nonuniformity will be greater.

$$F_b = \frac{W \text{ nominal}}{W \text{ min. avg.}}$$

where W = Mass Flow, lb/hr

nominal = The usual or regular value for the channel spacing if all channels in the system are equal in size. For a system of channels which are approximately equal in size, an effective nominal value may be used.

[a] R. J. Nertney, Calculated Surface Temperatures for Nuclear Systems and Analysis of their Uncertainties, IDO-16343 (July 1, 1957).

minimum average = The channel existing on the average at its minimum possible dimension based on mechanical tolerance deviations, specimen position eccentricity, and the effects of fuel swelling.

The extent of fuel swelling will be monitored by interim examination. Any severely swollen or possible waterlogged specimens will be removed from the test. It is assumed that extended irradiation may cause a moderate fuel thickness increase of five percent. In the worse case, $0.05 \times 0.48 \text{ inch} = 0.0024$ if swelling reduction occurs in a channel.

The minimum average channel dimension thus becomes $0.085 - 0.007$ mechanical tolerance and eccentricity - 0.0024 swelling effect = $0.0756 \approx 0.076 \text{ inch}$.

The effective nominal channel spacing for the tier becomes $(1/6 \times 0.115) + (4/6 \times 0.085) + (1/6 \times 0.076) = 0.0192 + 0.0567 + 0.0127 = 0.0886 \text{ inch}$. Assume $\rho_{\text{eff. nom.}} \approx \rho_{\text{min. avg.}}$

$$\begin{aligned} \text{Thus } F_b &= \frac{W_{\text{eff nom}}}{W_{\text{min avg}}} = \left(\frac{V_{\text{eff nom}}}{V_{\text{min avg}}} \right) \left(\frac{A_{\text{eff nom}}}{A_{\text{min avg}}} \right) \\ &= \left(\frac{D_{\text{eff nom}}}{D_{\text{min avg}}} \right)^{2/3} \left(\frac{A_{\text{eff nom}}}{A_{\text{min avg}}} \right) \\ &= \left(\frac{S_{\text{eff nom}}}{S_{\text{min avg}}} \right)^{2/3} \left(\frac{A_{\text{eff nom}}}{A_{\text{min avg}}} \right) \\ &= \left(\frac{0.0886}{0.076} \right)^{2/3} \left(\frac{1.075}{1.072} \right) \left(\frac{0.0886}{0.076} \right) \\ &= \left(\frac{0.0886}{0.076} \right)^{5/3} = (1.166)^{5/3} = 1.292 \end{aligned}$$

where D = equivalent diameter = $2S$ (where S = channel spacing). It should be noted that for a system of parallel channels markedly different in size, the above relation is no longer a very good approximation; and in this case, an actual mass balance using the actual channel spacings is required in order to obtain the proper flow distribution.

$$F_h = \left(\frac{D_{\text{eff nom}}}{D_{\text{min avg}}} \right)^{0.2} = \left(\frac{S_{\text{eff nom}}}{S_{\text{min avg}}} \right)^{0.2} = \left(\frac{0.0886}{0.076} \right)^{0.2} = 1.031$$

B. MECHANICAL DISTORTION

(1) and (2) -- Effect on mass flow due to external, entrance, and exit effects. The flow is measured directly; therefore,

$$F_h = 1.0 \qquad F_b = 1.0$$

(3) and (4) -- Channel effects and effect of variation in equivalent diameter. Fuel swelling effects on F_b and F_h have already been considered in A (3) and (4).

$$F_h = 1.00 \qquad F_b = 1.00$$

C. CORRELATION EQUATION ON h

$$F_b = 1.00$$

$$F_h = \frac{1}{0.8} = 1.25 \text{ since the uncertainty factor ("20 percent negative tolerance") of 0.8 was not previously used in the determination of h.}$$

D. DETERMINATION OF CRITICAL PARAMETERS

1. Flow Measurements

Gentile flow tube plus the recorder is assumed to have a 3 percent accuracy.

$$F_b = 1.03 \qquad F_h = (1.03)^{0.8} = 1.024$$

2. Temperature Measurements (Metal)

This effect is not applicable since there are no instrumented specimens in the test.

$$F_b = 1.00 \qquad F_h = 1.00$$

3. Pressure Measurements

Pressure has a negligible effect on physical properties and h

$$F_b = 1.00 \qquad F_h = 1.00$$

4. Temperature Measurements (Coolant)

The variation of temperature upon h and water physical properties was previously incorporated in the test calculations. For the purpose of direct measurement of the coolant water temperature, it is assumed that the test thermocouples have an accuracy of ± 1 percent. Therefore, $\Delta T_{\text{add}} = 0.01 (500) = 5^\circ\text{F}$.

E. FLOW DISTRIBUTION EFFECTS

1. Plenum Effects

There are no identifiable plenum effects in the test geometry.

2. Edge Effects

$$\text{The fuel edge is } \frac{(1.045 - 0.800)}{2} + \frac{(1.075 - 1.045)}{2} = 0.137 \text{ inch from the edge}$$

of the channel; or, in other words, the fuel width is only $\frac{0.800}{1.075} \times 100 = 74$ percent of the channel width. Since the specimen is centered in the channel, the radial variation in film coefficient, h, is expected to be small over the fuel heat transfer surface.

$$\text{Thus, } F_h = 1.00$$

$$F_b = 1.00$$

F. NEUTRON AND GAMMA FLUX DISTRIBUTION

The test calculations including DNBR have been based on maximum expected axial and radial neutron flux factors (used to calculate peak heat flux). Maximum values for gamma heating, based on loop measurements, have also been incorporated in the calculation of peak heat fluxes. The computer code considered the internal flux distribution in the fuel.

$$\text{Thus, } F_h = 1.00$$

$$F_b = 1.00$$

G. NEUTRON AND GAMMA FLUX UNCERTAINTY

1. Unperturbed Flux Source

The physics calculation did not make use of any empirical, unperturbed flux measurements made in the absence of an experiment in the loop. However, the overall uncertainty of the physics analysis must be expressed, and this is taken as ± 20 percent.

2. Average Depression

Although the nuclear analysis was performed on the basis of fuel specimen-water homogenization, the resulting uncertainty already has been included in the uncertainty expressed in G (1).

$$F_h = 1.00$$

$$F_b = 1.00$$

3. Internal Distribution

Again, the physics analysis, except for radial peaking factors, was not based on flux wire measurements. The uncertainty associated with homogenization has already been considered.

$$F_h = 1.00$$

$$F_b = 1.00$$

4. Power Variation with Time

As discussed in the Test Nuclear Description portion of the data package, the test was designed so that the values of fission and gamma heat generation used reflected time peaking over the reactor cycle. Maximum heat fluxes thus include time effects caused by control rod movement and reactor fuel depletion.

$$F_h = 1.00$$

$$F_b = 1.00$$

H. HEAT FLUX OVER ELEMENT SURFACE

1. Fuel Homogeneity

A factor of 1.05 for the ratio of maximum fuel concentration to nominal is suggested by Metallurgy.

$$F_h = 1.05$$

$$F_b = 1.00$$

2. Mechanical Effects

This effect considers only the effect of cladding variation on F_h and F_b . Non-uniformity of fuel meat thickness is considered in I (1). A local cladding deviation of ± 0.005 inch is taken and an average value of ± 0.002 inch. The first deviation affects F_h ; the second applies to F_b . The formula used is

$$F = 1 + \left[\frac{a - a_{\text{nom}}}{\frac{k_c b}{k_m} + a_{\text{nom}} + \frac{k_c}{h}} \right]$$

where $F = F_b$ or F_h

a_{nom} = specimen nominal clad thickness = 0.015 inch for a specimen in tier A

$a = a_{\text{max loc}} - a_{\text{nom}} + 0.005$ for F_h , inches

$a_{\text{max avg}} = a_{\text{nom}} + 0.002$ for F_b , inches

b = specimen fuel half thickness = 0.024 inch for all specimens in tier A

k_c = cladding conductivity = 9.0 Btu/hr-ft-°F

k_m = fuel conductivity = 7.0 Btu/hr-ft-°F

h = film coefficient = 11,510 Btu/hr-ft²-°F

Evaluation of F_h :

$$\begin{aligned} F_h &= 1 + \frac{0.020 - 0.015}{\frac{9.0}{7.0} (0.024) + 0.015 + \frac{9}{11,510} \times 12} \\ &= 1 + \frac{0.005}{0.031 + 0.015 + 0.009} \\ &= 1 + \frac{0.005}{0.055} = 1 + 0.091 \\ &= 1.09 \end{aligned}$$

Evaluation for F_b :

$$\begin{aligned} F_b &= 1 + \frac{0.017 - 0.015}{\frac{9.0}{7.0} (0.024) + 0.015 + \frac{9}{11,510} \times 12} \\ &= 1 + \frac{0.002}{0.055} = 1 + 0.036 \end{aligned}$$

$$F_b = 1.036$$

I. EFFECTS PECULIAR TO THE EXPERIMENT

1. Uranium-235 Loading Deviation

The specimen power generations used for heat fluxes and subsequent test calculations were based on fuel loadings derived from nominal design dimensions, compositions, and theoretical densities. This effect considers the difference between the nominal design uranium-235 weight in any specimen and its actual, as-fabricated, weight content in uranium-235. The actual weight of uranium-235 does not exceed the nominal by more than five percent. The average deviation is three percent.

$$F_h = 1.05$$

$$F_b = 1.03$$

2. Leakage Flow

The test calculations assumed that all the coolant flow in a channel was effective for specimen cooling. However, if no radial mixing occurs in the channel over the specimen length, all the flow which does not pass over the heat transfer surface of the specimens is ineffective and regarded as leakage flow.

$$\frac{0.613 \text{ in.}^2 - 5 (0.800 \times 0.085) + (0.115 \times 0.800)}{0.613} = \frac{0.613 - 0.432}{0.613}$$

$$= \frac{0.181}{0.613} = 0.295$$

where the total coolant channel = 0.613 in.^2

$$\text{Thus, } F_h = 1.00$$

$$F_b = \frac{1}{0.705} = 1.42$$

Summary: The combination of hot-spot and hot-channel factors is shown in Table D-1.

TABLE D-1
HOT-SPOT AND HOT-CHANNEL FACTORS

	Hot Channel Factor (F_b)	Hot Spot Factor (F_h)	Additive Factors (Δt add)
A. Mechanical Tolerances (Hydraulic Effects)			
(1) Effect on mass flow -- external effects (end boxes, etc)	1.000	1.000	
(2) Effect on mass flow -- entrance and exit effects			
(3) Effect on mass flow -- channel effects	1.291	1.031	
(4) Effect of D variation			
B. Mechanical Distortion			
(1) Effect on mass flow -- external effects (end boxes, etc)	1.000	1.000	
(2) Effect on mass flow -- entrance effects			
(3) Effect on mass flow -- channel effects	1.000	1.000	
(4) Effect of D variation			
C. Correlation Equation on h	1.000	1.250	
D. Determination of Critical Parameters			
(1) Flow measurement	1.030	1.024	5°F
(2) Temperature measurements (metal)	1.000	1.000	
(3) Pressure measurements	1.000	1.000	
(4) Temperature measurements (coolant)			
E. Flow Distribution Effects			
(1) Plenum effects	1.000	1.000	
(2) Edge effects	1.000	1.000	
F. Neutron and Gamma Flux Distribution			
(1) Vertical distribution	1.000	1.000	
(2) Horizontal distribution	1.000	1.000	
(3) Internal distribution	1.000	1.000	
G. Neutron and Gamma Flux Uncertainty			
(1) Unperturbed flux source	1.200	1.200	
(2) Average depression	1.000	1.000	
(3) Internal distribution	1.000	1.000	
(4) Experimental power generation variation timewise	1.000	1.000	
H. Heat Flux Over Element Surface			
(1) Fuel homogeneity	1.000	1.050	
(2) Mechanical effects (meat and clad thickness)	1.036	1.090	
I. Effects Peculiar to Experiment			
(1) U-235 loading	1.030	1.050	
(2) Leakage flow	<u>1.420</u>	<u>1.000</u>	
	2.42	1.90	5°F

APPENDIX E

ENGINEERING PART NUMBERING SYSTEM

A. DRAWING PARTS LISTS

Drawing parts lists or bills of materials shall indicate the material to be used for all parts plus the appropriate ASME, ASTM, or experiment sponsor's specifications. The drawing parts list will also contain an additional column for engineering part numbers.

B. TERMS OF THE ENGINEERING PART NUMBER

These part numbers, abbreviated "EP No.", will have four terms:

1. First Term

The first term will be the same as the experimental alphabetical prefix identifying the sponsor, ie, "KAPL", for Knolls Atomic Power Laboratory, "WAPD", for Westinghouse Atomic Power Division, and the like.

2. Second Term

The second term will be either "MTR", "ETR", or "ATR", dependent on whether the Materials Testing Reactor, the Engineering Test Reactor, or the Advanced Test Reactor is the applicable reactor in which the particular experiment is to be conducted.

3. Third Term

The third term will be a letter of the alphabet and will follow the schedule given in C below.

4. Fourth Term

The fourth term will be a sequential numbering system from "1" to as high a number as necessary to include all equipment in the experiment. Not all numbers in any block of numbers set aside for each alphabetical category need be used.

5. Example

Example: KAPL-MTR-C-5, where Knolls Atomic Power Laboratory is the sponsor, the equipment is planned to be installed at the Materials Testing Reactor; it is a pump; and it is the fifth piece of equipment assigned an engineering part number.

C. GROUP INDEX

Group Letter	Group	Description
A	Building, Cubicles and Shielding, Foundations	All building-type enclosures, including concrete foundations, structural steel, anchor bolts, concrete block, monorails, doors, windows, sheet plastic, lead bricks, and all other shielding forms.
B	Columns and Pressure Vessels	All unfired pressure vessels, ion exchange columns, delayed neutron monitor vessels, pressurizer or surge tanks, gas cylinders.
C	Pumps	Pumps -- centrifugal, reciprocating, positive displacement, and their drivers.
D	Tanks	All atmospheric tanks and their appurtenances.
E	Exchangers	All heat transfer equipment, except as noted under "Vacuum Equipment" in the next category.
F	Vacuum Equipment	Barometric condensers and mechanical vacuum pumps and drivers.
G	Filters	All filtering equipment.
H	Piping and Tubing	Piping, tubing, valves, and fittings except as noted under "Services" and "Instruments".
I	Structure	Stairs, ladders, platforms, pipe supports, etc.
J	Insulation	Thermal insulation on pipes, pumps, pressure vessels, etc.
K	Compressors	Compressors, centrifugal and disc fans, positive displacement blowers and their drivers, except for ventilating building spaces.
L	Instruments	All indicating, controlling, and recording instruments and panels, including supports.

M	Electric Power and Lighting	All electrical equipment, including switchgear, transformers, motor starters, generators and drivers, lighting and power panels, power and lighting, wiring and telephone, and signal systems.
N	Services	Space heating, ventilating, and air conditioning equipment, duct work, fire protection systems, handling tools, hoists, storage cabinets, testing equipment, and related piping.
O	Sampling and Laboratory Equipment	Sample bombs, bottles, laboratory equipment, chemicals, etc.
P	Special Reactor Loading and Discharging Equipment, Shipping Casks and Containers	Special reactor loading and unloading apparatus, not including carts to facilitate transporting, shielded shipping casks and shipping containers, crates and boxes.
Q	Miscellaneous	All equipment not otherwise described above.

D. PREPARATION OF MATERIAL AND EQUIPMENT LIST

Following completion of all drawings, a material and equipment list will be prepared, itemizing in sequence by engineering part number, giving a brief description of the equipment, quantities, pertinent drawing numbers, vendors, purchase order numbers, and their data of issue and scheduled delivery data. A typical form for this information is given in Table E-1.

TABLE E-1
MATERIAL AND EQUIPMENT LIST

Cost Code Equipment Piece No	Cost * Quantity	Description	Drawing No	Requisition		Item	Purchase Order No	Required Date	Vendor and/or Remarks
				Orig	Date Due			Promised Date	
				No	Issued		Date		
(1) XYZ (Sponsor Lab Code Initials)-ETR-1	9884 00 4	Pump, 1 gpm & 2000 psig (Positive Displacement)	H-3-8902	D72911	5-3-57	1	H7C 38877	Complete	ABC Pump Co Mercer Island, Washington
(2) XYZ (Sponsor Lab Code Initials)-ETR-2	901 12 2	Pump, 25 gpm & 30' head (Loop Filling and Flushing)	H-3-8903	D72883	4-15-57	1	H7C 37678	Complete	DEF Manufacturing Co Seattle, Washington
(3) XYZ (Sponsor Lab Code Initials)-ETR-3	1970 00 1	Tank (Make-Up) 250 gallon ss	H-3-8931	D72820	2-27-57	1	H7S 34650	Complete	GHI Copper Works Seattle, Washington
(4) XYZ (Sponsor Lab Code Initials)-ETR-4									
(5) XYZ (Sponsor Lab Code Initials)-ETR-5	452.00 1	Resin Addition Column	H-3-8934	D72814	2-27-57	1	H7T 35403	Complete	JKL Copper Works Portland, Oregon
(6) XYZ (Sponsor Lab Code Initials)-ETR-6	15 01 1	Steam Trap, 3/4 in	H-3-8902	D72878	4-15-57	1	H7H 36578	Complete	MNO Supply Co Portland, Oregon
(7) XYZ (Sponsor Lab Code Initials)-ETR-7	24 27 1	Steam Trap, 1 in	H-3-8902	D72878	4-15-57	2	H7H 36578	Complete	PQR Equipment Co San Francisco, Calif
(8) XYZ (Sponsor Lab Code Initials)-ETR-8	694 35 1	Manifold Regulator and Pigtail for 6 Helium Cylinders	H-3-8902	D72907	4-30-57	1	H7G 38973	Complete	
(8-1) XYZ(Sponsor Lab Code Init)-ETR-8-1	1	Cylinder Support		D74025	12-30-57	1			
(8-2) XYZ(Sponsor Lab Code Init)-ETR-8-2									

*NOTE: COST IS THE COST OF QUANTITY OF ITEM

APPENDIX F

CALCULATION OF PRIMARY LOOP PRESSURE DROP

A. MAXIMUM FLOW

The primary loop pressure drop is calculated for the following conditions:

Flow, lb/hr	37,200
Pressure, psig (low pressure operation)	300
Flow path	
Bypassed to heater, lb/hr	none
Flow through heat exchangers, lb/hr	37,200
Reactor Inlet Temperature, °F	267
Reactor Outlet Temperature, °F	325

1. Out-of-Pile Pressure Drop

a. ΔP from Reactor Outlet to Heat Exchanger Inlet

The 37,200 lb/hr flows through 71 feet of 2-inch Schedule 80 S piping and then divides itself in half and flows through 5.5 feet of 2-inch Schedule 80 S pipe to the two heat exchangers, in accordance with Figure F-1:

Two-inch Schedule 80 S pipe: Flow area = $2.95 \text{ in.}^2 = 0.025 \text{ ft}^2$;
ID = 1.94 in. = 0.162 ft.

(1) Total Length of Pipe for 37,200 lb/hr Flow

PW 2081 - 2 in. MIA Pipe	71 ft
14 - 90° Elbows (L/D = 20); Length = 14 (20) (0.162)	45 ft
PW - V1001 Valve (L/D = 13); Length = 1(13) (0.162)	2 ft
TCV - 2081 Valve (L/D = 340); Length = 1 (340) (0.162)	<u>55 ft</u>
Total Equivalent Length	173 ft

Pressure Drop for 37,200 lb/hr Flow

At 325°F and 300 psig:

$$\left. \begin{array}{l} \rho = 56.6 \text{ lb/ft}^3 \\ \mu = 0.42 \text{ lb/(hr) (ft)} \end{array} \right\} \text{ Approximate}$$

$$\text{Velocity (V)} = \frac{37,200 \text{ lb/hr}}{0.0205 \times 56.6} = 32,000 \text{ ft/hr} = 8.9 \text{ ft/sec}$$

$$\text{Reynolds Number (Re)} = \frac{\rho V D}{\mu} = \frac{56.6 (32,000) (0.162)}{0.42} = 699,000$$

$$\text{Friction Factor (f)} = 0.014$$

$$\text{Pressure Drop } (\Delta P) = \frac{f L V^2}{D (2g)} = \frac{0.014 (173) (8.9)^2}{0.162 (64.4)} = \frac{192}{10.4} = 18.5 \text{ ft}$$

Fig. F-1 Loop pressure drop diagram.

(2) Total Length of Pipe for 18,600 lb/hr Flow

PW 2082 - 2 in. MIA Pipe	5.5 ft
3 - 90° Elbows, Length = 3 (20) (0.162)	9.7 ft
PN-V1003 Valve, Length	<u>2.0 ft</u>
Total Equivalent Length	17.2 ft

Pressure Drop for 18,600 lb/hr Flow

$$V = 16,000 \text{ ft/hr} = 4.45 \text{ ft/sec}$$

$$Re = 349,000; f = 0.016$$

$$\Delta P = \frac{0.016(17.2) (4.45)^2}{0.162 (64.4)} = \frac{5.4}{10.4} = 0.52 \text{ ft}$$

b. ΔP through Heat Exchanger C-201 (pressure loss = 9.5 psi)

$$\rho \text{ at } 296^\circ\text{F} = 57.5 \text{ lb/ft}^3; T_{av} = \frac{325 + 267}{2} = 296^\circ\text{F}$$

$$\Delta P = \frac{9.5 \text{ psi}(144 \text{ sq in./ft}^2)}{57.5 \text{ lb/ft}^3} = 23.8 \text{ ft}$$

c. ΔP From Heat Exchanger Outlet to Reactor Inlet

(1) Total Length of Pipe for 18,600 lb/hr Flow

PW 2086 - 2 in. MIA	5.75 ft
3 - 90 Elbows, L = 3 (20) (0.162)	9.72 ft
PW-V1004 Control Valve, L = 1 (340) 0.162	<u>55.08 ft</u>
Total Equivalent Length	70.55 ft

Pressure Drop for 18,600 lb/hr Flow

$$(\text{Velocity} = 4.45 \text{ ft/sec}; f = 0.016)$$

$$\Delta P = \frac{fLV^2}{D2g} = \frac{0.016 (70.55) (4.45)^2}{0.162 (64.4)} = 2.14 \text{ ft}$$

(2) Total Length of Pipe for 37,200 lb/hr Flow

PW 2086 and 91 - 2 in. MIA	115.5 ft
25 - 90° Elbows (L/D = 30); L = 25(20) (0.162)	81.0 ft
FCV 2086 - Globe (L/D = 340); L = 1 (340) (0.162)	55.1 ft
PW V1000 - Gate (L/D = 13); L = 1 (13) (0.162)	<u>2.1 ft</u>
Total Equivalent Length	253.7 ft

Pressure Drop for 37,200 lb/hr Flow

$$\text{At } 267^\circ\text{F} \text{ and } 300 \text{ psig}$$

$$\rho = 58.5 \text{ lb/ft}^3$$

$$\mu = 0.51 \text{ lb/(hr) (ft)}$$

$$\text{Velocity} = \frac{37,200 \text{ lb/hr}}{0.0205 \text{ sq ft} \times 58.5 \text{ lb/ft}^3} = 31,000 \text{ ft/hr} = 8.6 \text{ ft/sec}$$

$$\text{Re} = \frac{\rho V D}{\mu} = \frac{58.5 (30,900) (0.162)}{0.51} = 574,000$$

$$f = 0.014$$

$$\Delta P = \frac{f L V^2}{D 2g} = \frac{0.014 (253.7) (8.6)^2}{0.162 (64.4)} = \frac{263}{10.4} = 25.3 \text{ ft}$$

d. Total Out-of-Pile Pressure Drop

	ΔP
Reactor Outlet to Exchanger Inlet	<u>19.0 ft</u>
Heat Exchanger	23.8 ft
Exchanger Outlet to Reactor Inlet	<u>27.4 ft</u>
Total (Out-of-Pile) Pressure Drop	70.2 ft

2. ΔP in Cross-Arms

a. Losses to be Considered

(1) Reactor Inlet Flow

Reducer loss -- negligible

1-90° L.R. Elbow loss

7.5 ft of 1-1/2 in. Schedule 80 pipe

2-90° S. R. Elbow loss

(2) Reactor Outlet Flow

Reducer loss -- negligible

1-90° L. R. Elbow loss

8 ft of 1-1/2 in. Schedule 80 pipe

2-90° S. R. Elbow loss

b. Total Equivalent Length

1-1/2 in. Schedule 80 pipe	15.5 ft
2-90° L. R. Elbows ($L/D = 20$); $L = (20) (0.162)2$	6.5 ft
4-90° S. R. Elbows ($L/D = 30$); $L = 4(30) (0.162)$	<u>19.4 ft</u>
Total Equivalent Length	41.4 ft

$$\text{At Avg Temp} = \frac{325 + 267}{2} = 296^\circ\text{F}$$

$$1\text{-}1/2 \text{ in. Schedule 80: Flow Area} = 1.5 \text{ in.}^2 = 0.0123 \text{ ft}^2$$

$$\text{ID} = 1.5 \text{ in.} = 0.125 \text{ ft}$$

$$\rho = 57.5 \text{ lb/ft}^3$$

$$\mu = 0.45 \text{ lb/(hr) (ft)}$$

$$\text{Velocity (V)} = \frac{37,200}{0.0123 \times 57.5} = 52,600 \text{ ft/hr} = 14.6 \text{ ft/sec}$$

$$\text{Re} = \frac{\rho V D}{\mu} = \frac{57.5 (52,600) (0.125)}{0.45} = 840,000$$

$$f = 0.014$$

c. Friction Loss

$$\Delta P = \frac{f L V^2}{D 2g} = \frac{0.014 (41.4) (14.6)^2}{0.125 (64.4)} = 15.3 \text{ ft}$$

3. In-Pile Pressure Drop, Vertical Section

a. Losses to be Considered

(1) Downflow

90° bend loss

Downflow loss (annular flow) $\approx$ 14 ft 9 in.

Downflow loss (outside of fuel section) $\approx$ 4 ft

180° bend loss

(2) Upflow

Upflow loss (1 in. Schedule 40 pipe) $\approx$ 16 ft

90° bend loss

(3) Element loss -- 3 ft

b. Downflow

$$\text{Annular Flow Area} = \frac{\pi}{4} (D_o^2 - D_i^2) = 1.59 \text{ sq in.} = 0.011 \text{ sq ft}$$

$$\text{Effective Diameter (D}_e\text{)} = \frac{4 (\text{Flow Area})}{\text{Wetted Perimeter}}$$

$$D_e = \frac{4 (1.59)}{\pi (1.32 + 1.94)} = \frac{6.36}{10.24} = 0.62 \text{ in.}$$

(1) Downflow Annular Flow

$$\Delta P \frac{fLV^2}{D_e 2g} = \frac{KV^2}{2g} \quad \text{where } K = fL/D_e$$

$$\text{Velocity} = \frac{37,200 \text{ lb/hr}}{0.011 \text{ sq ft} \times 57.5 \text{ lb ft}^3} = 58,800 \text{ ft/hr} = 16.3 \text{ ft/sec}$$

$$Re = \frac{\rho V D_e}{\mu} = \frac{57.5 (58,800) 0.62}{0.45 (12 \text{ in./ft})} = 388,000$$

$$f = 0.016$$

$$\therefore K = \frac{fL}{D_e} = \frac{0.016 (14.75)}{0.62/12} = 4.6$$

$$\therefore \Delta P = \frac{4.6 (16.3)^2}{64.4} = 19.0 \text{ ft}$$

(2) 90° Bend

Resistance factor $K = 1.3$

$$\Delta P = \frac{KV^2}{2g} = \frac{(1.3) (16.3)^2}{64.4} = 5.2 \text{ ft}$$

(3) Annular Flow Around Sample

$$\begin{aligned} \text{Flow Area} &= \pi/4 \left(\frac{1.82}{12} \right)^2 - 1.2 (1.12)/144 = 0.0181 - 0.0093 = 0.0088 \text{ ft}^2 \\ &= 1.27 \text{ in.}^2 \end{aligned}$$

$$D_e = \frac{4 (\text{Flow Area})}{\text{Wetted Perimeter}} = \frac{4 (1.27)}{\pi(1.82) + 2(1.2 + 1.12)}$$

$$D_e = \frac{5.08}{10.36} = 0.491 \text{ in.}$$

$$\text{Velocity} = \frac{37,200}{0.0088 \times 57.5} = 73,500 \text{ ft/hr} = 20.4 \text{ ft/sec}$$

$$Re = \frac{57.5 (73,500) 0.491}{0.45 \times 12} = 3.84 \times 10^5; f = 0.016$$

$$\Delta P = \frac{0.016 (4) (20.4)^2}{0.491/12 (64.4)} = 10.1 \text{ ft}$$

(4) 180° Bend

Resistance Factor K = 2.0

$$\Delta P = \frac{KV^2}{2g} = \frac{2(20.4)^2}{64.4} = 12.9 \text{ ft}$$

(5) Total Downflow

Annular flow	19.0 ft
90° bend	5.2 ft
Annular flow around sample	10.1 ft
180° bend	<u>12.9 ft</u>
	47.2 ft

c. Upflow

1 in. Schedule 40 pipe

$$\text{Flow Area} = 0.864 \text{ in.}^2 = 0.006 \text{ ft}^2$$

ID = 1.05 in.

$$\text{Velocity} = \frac{37,200}{0.006 (57.5)} = 108,000 \text{ ft/hr} = 30 \text{ ft/sec}$$

$$R_e = \frac{\rho VD}{\mu} = \frac{57.5 (108,000) (1.05)}{0.49 (12)} = 1.2 \times 10^6$$

$$f = 0.012$$

$$K = f \frac{L}{D} = 0.012 \frac{(16)}{1.05/12} = 2.2$$

90° Bend resistance factor, K, = 1.3

$$\text{Total } (K_T) = 1.3 + 2.2 = 3.5$$

$$\text{Pressure Drop } (\Delta P) = \frac{3.5 (30)^2}{64.4} = 48.9 \text{ ft}$$

d. Element Pressure Drop (Maximum 6-Plate Element)

Gap between plates = 0.1 in.

$$\text{Flow Area} = \frac{0.1 (1.2) (7)}{144} = 0.00583 \text{ ft}^2 \text{ (total)}$$

$$\text{Flow Area per channel} = \frac{0.00583}{7} = 0.000832 \text{ ft}^2$$

$$\text{Velocity} = 108,000 \text{ ft/hr} \times \frac{0.006}{0.00583} = 111,000 \text{ ft/hr} = 31 \text{ ft/sec}$$

$$D_e = 2 \times \text{spacing} = 2(0.1) = 0.2 \text{ in.}$$

$$R_e = \frac{57.5 \times 111,000 (0.2)}{0.45 (12)} = 234,000$$

$$f = 0.012$$

$$\Delta P = \frac{0.012 (3) (31)^2}{\frac{0.2}{12} (64.4)} = 32.2 \text{ ft.}$$

4. Total Primary Loop Pressure Drop (Maximum Flow)

	<u>ΔP (ft)</u>
Out-of-pile pressure drop	70.2 ft
Cross-arms pressure drop	15.3 ft
In-pile pressure drop	
Downflow	47.2 ft
Upflow	48.9 ft
Element (maximum 6 plates)	32.2 ft
Spare pump loss (assumed)	<u>30.0 ft</u>
Total Pressure Drop	243.8 ft

B. MINIMUM FLOW

Total flow = 8300 lb/hr. The 8300 lb/hr flows from reactor outlet to the control valve TVC 3081 at which point 3270 lb/hr is bypassed through the heater lines; the remaining 5030 lb/hr proceeds to the heat exchangers. The two flows then combine at valve FCV 2086.

1. Out-of-Pile Pressure Drop

a. ΔP from Reactor Outlet to Heat Exchanger Inlet

(1) Total Length of Pipe for 8300 lb/hr Flow

PW 2081 - 2 in. MIA	68 ft
14 - 90° Elbows (L/D = 20), length = 14(20) (0.162)	45 ft
PW V1001 valve (L/D = 13), length = 1(13) (0.162)	<u>2 ft</u>
Total length	115 ft

Pressure Drop for 8300 lb/hr Flow

At 432°F and 2200 psi

$$\rho = 52.6 \text{ lb/ft}^3$$

$$\mu = 0.30 \text{ lb/(hr)} \text{ (ft)}$$

$$\text{Velocity} = \frac{8300 \text{ lb/hr}}{(0.0205 \text{ ft}^2) (52.6 \text{ lb/ft}^3)} = 7700 \text{ ft/hr} = 2.14 \text{ ft/sec}$$

$$R_e = \frac{\rho V D}{\mu} = \frac{52.6 (7700) (0.162)}{0.30} = 219,000$$

$$f = 0.016$$

$$\Delta P = \frac{f L V^2}{D 2g} = \frac{0.016 (115) (2.14)^2}{0.162 (64.4)} = 0.81 \text{ ft}$$

(2) Total Length of Pipe for 5030 lb/hr Flow

PW 2082 - 2 in. MIA pipe	8.5 ft
3 - 90° Elbows, length - 3(20) (0.162)	9.7 ft
PW V1003 valve, length	<u>2.0 ft</u>
Total length	20.2 ft

Pressure Drop for 5030 lb/hr Flow

$$\text{Velocity} = \frac{5030}{8300} (7700 \text{ ft/hr}) = 4670 \text{ ft/hr} = 1.30 \text{ ft/sec}$$

$$R_e = \frac{4670}{7700} (219,000) = 133,000$$

$$f = 0.019$$

$$\Delta P = \frac{0.019 (20.2) (1.30)^2}{0.162 (64.4)} = 0.06 \text{ ft}$$

b. ΔP through Heat Exchanger C-201

<u>Operating Conditions</u>	<u>Minimum Condition</u>
Flow = 18,600 lb/hr	Flow = 5030 lb/hr
$\Delta P = 23.8 \text{ ft}$	$\Delta P = \text{--}$
Avg Temp = 296°F	Avg Temp = 365°F
$\rho = 57.5 \text{ lb/ft}^3$	$\rho = 55.4 \text{ lb/ft}^3$

The ΔP for minimum flow is calculated as follows:

$$\Delta P (\text{min}) = \Delta P (\text{op}) \frac{[Q (\text{min})]^2}{\rho (\text{min})} \times \frac{\rho (\text{op})}{[Q (\text{op})]^2}$$

$$\Delta P (\text{min}) = 23.8 \text{ ft} \times \frac{(5030)^2}{(55.4)} \times \frac{(57.5)}{(18,600)^2} = 1.81 \text{ ft}$$

c. ΔP Bypass Line (Valve TCV 2081--Through Heaters to Valve FCV 2086)

FCV 2086

Flow = 3270 lb/hr

(1) Total Equivalent Length

PW 2127 and 2128 - 2 in. MIA pipe	56.0 ft
≈ 11 - 90° Elbows, length = 11(20) (0.162)	<u>35.6 ft</u>
Total length	91.6 ft

(2) Pressure Drop

$$\text{Velocity (V)} = \frac{3,270}{0.0205 \times 52.6} = 3030 \text{ ft/hr} = 0.84 \text{ ft/sec}$$

$$R_e = \frac{52.6 (3030) (0.162)}{0.30} = 86,000$$

$$f = 0.021$$

$$\Delta P = \frac{0.021 (91.6) (0.84)^2}{0.162 (64.4)} = 0.13 \text{ ft} \quad \text{Use } 0.13 \text{ ft}$$

NOTE

This $\Delta P = 0.13$ foot is not added to overall ΔP since bypass line is in parallel with the heat exchanger lines.

d. ΔP from Heat Exchanger Outlet to Reactor Inlet

(1) ΔP from heat exchanger outlet to valve FCV 2086 is approximately equal to ΔP from TCV 2081 to Heat Exchanger Inlet; therefore, use: $\Delta P = 0.06$ ft.

(2) ΔP from FCV 2086 to Reactor Inlet Flow = 8300 lb/hr.

(a) Total Equivalent Length

$$254 \text{ ft} - 55 \text{ ft (Globe Valve)} = 199 \text{ ft}$$

(b) Pressure Drop

At 293°F and 200 psi,

$$\rho = 57.7 \text{ lb/ft}^3$$

$$\mu = 0.48 \text{ lb/(hr) (ft)}$$

$$\text{Velocity} = \frac{8300}{0.0205 \times 57.7} = 7020 \text{ ft/hr} = 1.95 \text{ ft/sec}$$

$$R_e = \frac{57.7 (7020) (0.162)}{0.48} = 137,000; f = 0.019$$

$$\Delta P = \frac{0.019 (199) (1.95)^2}{0.162 (64.4)} = 1.38 \text{ ft}$$

e. Total Out-of-Pile Pressure Drop

Reactor outlet to exchanger inlet	0.87 ft
Heat exchanger	1.81 ft
Bypass line (not added in)	---
Exchanger outlet to reactor inlet	<u>1.44 ft</u>
Total ΔP	4.12 ft

2. ΔP in Cross-Arms

a. Total Equivalent Length

41.4 ft

b. Pressure Drop

At Avg Temp = 365 °F

$$\rho = 55.4 \text{ lb/ft}^3$$

$$\mu = 0.37 \text{ lb/(hr) (ft)}$$

1-1/2 in. Schedule 80 pipe

$$\text{Flow Area} = 0.0123 \text{ ft}^2$$

$$\text{Velocity} = \frac{8,300}{0.0123 \times 55.4} = 12,200 \text{ ft/hr} = 3.38 \text{ ft/sec}$$

$$R_e = \frac{55.4 (12,200) 0.125}{0.37} = 228,000$$

$$f = 0.017$$

$$\Delta P = \frac{0.017 (41.5) (3.38)^2}{0.125 (64.4)} = 1.00 \text{ ft}$$

3. In-Pile Pressure Drop -- Vertical Section

Losses considered are similar to those used for the maximum flow condition.

a. Downflow

(1) 90° Bend

Resistance Factor, $K = 1.3$ (at cross-arm)

Downflow Resistance Factor

$$K = fL/D_e$$

$$\text{Velocity} = \frac{8300}{0.011 \times 55.4} = 13,600 \text{ ft/hr} = 3.78 \text{ ft/sec}$$

$$R_e = \frac{55.4 (13,600) (0.62)}{0.37 (12)} = 105,000; f = 0.020$$

$$\text{Equivalent Length} = 14.75 \text{ ft} \therefore K = \frac{14.75 (0.020)}{0.62/12} = 5.71$$

$$\text{Total } (K_T) = 1.3 + 5.71 = 7.01$$

$$\Delta P = \frac{K_t V^2}{2g} = \frac{7.01 (3.78)^2}{64.4} = 1.56 \text{ ft}$$

(2) Annular Flow Around Sample

$$\text{Length} = 4 \text{ ft}$$

$$r = \frac{1.82}{2} - \frac{0.125}{2} = 0.848 \text{ in.}$$

$$\text{Flow Area} = 2 \pi r t = 2(3.14) (0.848) (0.125) = 0.666 \text{ in.}^2 = 0.00462 \text{ ft}^2$$

$$D_e = 2(0.125) = 0.25 \text{ in.} = 0.0208 \text{ ft}$$

$$\text{Velocity} = \frac{8300}{0.00462 \times 55.4} = 32,400 \text{ ft/hr} = 9.0 \text{ ft/sec}$$

$$R_e = \frac{55.4 (32,400) (0.0208)}{0.37} = 101,000; f = 0.020$$

$$\Delta P = \frac{0.020 (4) (9.0)^2}{(0.0208) (64.4)} = 4.84 \text{ ft}$$

(3) 180° Bend (Bottom of Sample)

$$\text{Resistance Factor } (K = 2.0)$$

$$\Delta P = \frac{2.0 (9.0)^2}{64.4} = 2.52 \text{ ft}$$

(4) Total Downflow

90° Bend	1.56 ft
Annular Flow Around Sample	4.84 ft
180° Bend	<u>2.52 ft</u>
	8.92 ft

b. Upflow

90° Bend Resistance Factor, $K = 1.3$ (at cross-arm)

Upflow Resistance Factor:

$$\text{Velocity} = \frac{8300}{0.006 \times 55.4} = 25,000 \text{ ft/hr} = 6.94 \text{ ft/sec}$$

$$R_e = \frac{55.4 (25,000) (1.05)}{0.37 \times 12} = 328,000; f = 0.012$$

$$K = fL/De = \frac{0.012 (16)}{1.05/12} = 2.19$$

$$\text{Total } (K_T) = 1.3 + 2.19 = 3.49$$

$$\Delta P = \frac{3.49 (6.94)^2}{64.4} = 2.61 \text{ ft}$$

c. Element Pressure Drop (Minimum 2-Plate Element)

$$\text{Flow Area} = (0.05) (1.5) (3) = 0.225 \text{ in.}^2 = 0.00156 \text{ ft}^2$$

$$D_e = 2(0.05) = 0.10 \text{ in.}$$

(1) Flow Area around minimum element is orificed such that of the total flow, 7300 lb/hr flows through the element.

$$\text{Velocity} = \frac{7300}{0.00156 \times 55.4} = 84,500 \text{ ft/hr} = 23.5 \text{ ft/sec}$$

$$R_e = \frac{55.4 (84,500) (0.10)}{0.37 (12)} = 105,000; f = 0.020$$

$$\Delta P = \frac{0.020 (3) (23.5)^2}{\frac{0.10}{12} (64.4)} = 61.7 \text{ ft}$$

(2) Contraction Loss to Element

$$K = 0.45$$

$$\Delta P = \frac{0.45 (23.5)^2}{64.4} = 3.86 \text{ ft}$$

(3) Expansion Loss Leaving Element

$$K = 0.90$$

$$\Delta P = \frac{0.90 (23.5)^2}{64.4} = 7.72 \text{ ft}$$

(4) Total Element ΔP

Element	61.7 ft
Contraction Loss	3.86 ft
Expansion Loss	<u>7.72 ft</u>
Total	73.28 ft

4. Total Primary Loop Pressure Drop (Minimum Condition)

	<u>ΔP</u>
Out-of-pile pressure drop	4.12
Cross-arms pressure drop	1.00
In-pile pressure drop	
Downflow	8.92
Upflow	2.61
Element (2 plates)	73.28
Spare pump (assumed)	<u>8.00</u>
Total Pressure Drop	≈ 98 ft

The available designed pump head for this condition (8,300 lb/hr) is 340 feet. Therefore, the difference $340 - 98 = 242$ feet of head will have to be lost at the primary loop control valve (FCV 2086) in order that the system be in balance for this flow condition. However, if the pumps, as purchased, have a higher capacity-head curve, the FCV 2086 will have to take an increased loss.

APPENDIX G

PIPING STRESS DESIGN EXAMPLES

A. CODE STRESS DESIGN EXAMPLE

Assume the following three-dimensional piping system section:

As shown, linear expansion of member Z produces bending in member X and results in torsion of member Y.

Taking the X, Y, and Z planes, a centroid of the system is found, for example:

Very generally, the piping section material will respond to a temperature rise by nearly proportionate increase in linear dimensions; ie, ΔL is equal to the projected length of the section, L , multiplied by e , where e is the unit linear thermal expansion (dimensionless if both ΔL and L are in the same units). The values of e are selected from Table 22, on pages 93 and 94 of the ASA Pipe Code.

In the next step, the ΔL , or deflections, must be converted into reactions and stress. This is a time-consuming step and, dependent on the critical nature of the piping system, may be calculated by a simplified or a general analytical method such as presented in Chapters 4 and 5 of the M. W. Kellogg book, Design of Piping Systems. For this example, it will be assumed that these calculations have been completed and the maximum bending moment has been found to be 14,600 in. - lb and torsion at the point of maximum bending to be 19,000 in. - lb. The piping is assumed to be 2-inch Schedule 80, Type 347 stainless steel operating at 600°F and 2200 psig. The section modulus, Z , for this pipe schedule is taken to be 0.7309. The pipe dimensions are 1.939 inch ID and 2.375 inch OD. A piping tube life of 2000 thermal cycles is estimated from a three-week reactor refueling cycle, a maximum of 5 scrams per cycle, or 100 thermal cycles per year.

On page 9b of the ASA Pipe Code there is the following formula, based on the Maximum Shear Theory, for the combination of stresses due to thermal expansion:

$$S_E = \sqrt{S_b^2 + 4S_t^2}$$

where

S_E is the equivalent stress

S_b is the resultant longitudinal bending stress = $\frac{BM_b}{Z}$

where

B = stress intensification factor

M_b = resultant bending moment

Z = section modulus of pipe

S_t = the resultant torsional shear stress = $\frac{M_t}{2Z}$

where

M_t = resultant torsional moment.

The stress intensification factor is the ratio of maximum stress in local components of a piping system (flanges, bends, miters, branch connections, terminal connections, etc) to the maximum stress in a straight pipe subjected to the same moment. This would require detailed analysis of each component in accordance with information given in the 3rd chapter of the Kellogg book and ASME Technical Paper No. 58-A-207, "Stresses in Contoured Openings in Pressure Vessels," by Donald E. Hardenbergh. Assume a comparatively simple system with a stress intensification factor, $B = 1.0$.

$$\therefore S_b = 1.0 \times 14,600 / 0.7309$$

$$= 20,000 \text{ psi}$$

$$S_t = \frac{19,000}{2 \times 0.7309} = 13,000 \text{ psi}$$

$$S_E = \sqrt{(20,000)^2 + 4 (13,000)^2} = 32,800 \text{ psi}$$

This value is to be compared with the allowable stress as established in Section 622 of the Piping Code as:

$$S_A = f(1.25 S_C + 0.25 S_H)$$

where

S_C = allowable stress in cold condition

S_H = allowable stress in hot condition

f = stress-range reduction factor cyclic conditions from p. 96 of ASA code.

As there are less than 7000 cycles, 1.0 may be used for the reduction factor. On pages 22, 23, and 24 of the Piping Code, Tables 2 and 2a give the allowable stresses at various temperatures. Table 2 is for -20 to 650°F and Table 2a from 700 to 1200°F. Using Table II at 600°F for Type 347 S. S., seamless material:

$$S_H = 14,900 \text{ psi.}$$

Assuming ambient temperature as 100°F:

$$S_C = 18,750 \text{ psi}$$

$$\therefore S_A = 1.0 (1.25 \times 18,750 + 0.25 \times 14,900) = 27,165 \text{ psi}$$

In addition, the code specifically states that when the sum of the longitudinal stresses is less than S_H , the difference between S_H and this sum may be added to the 0.25 S_H term in above formula. The longitudinal stresses are those due to pressure, weight, and other sustained external loadings, and are determined simply by dividing the end force due to internal pressure by the cross sectional area of the pipe wall, or:

$$S_{lp} = \frac{F}{A} = \frac{\text{Internal pressure} \times \text{end area}}{\text{Pipe wall area}}$$

$$\begin{aligned} &= \frac{\frac{p\pi d^2}{4}}{\frac{\pi}{4}(D^2 - d^2)} = \frac{pd^2}{D^2 - d^2} \\ &= \frac{2200 \times (1.939)^2}{(2.375)^2 - (1.939)^2} = \frac{8270}{1.871} = 4420 \text{ psi.} \end{aligned}$$

This value is less than S_H of 14,900 psi, therefore,

$$14,900 - 4420 = 10,480$$

$$\begin{aligned} S_A &= 1.0 [1.25 (18,750) + 0.25 (14,900) + 10,480] \\ &= 37,665 \text{ psi.} \end{aligned}$$

The conclusion, S_E of 32,800 psi is less than allowable S_A of 37,665 psi and, therefore, meets code requirement.

For critical applications or where loadings are simultaneously analyzed, all stresses should be evaluated prior to combining them and compared to the total allowable stress range of 1.25 ($S_C + S_H$). The resultant principal stress at the outside fiber can be written as

$$S = 1/2 \left[S_L + S_p + \sqrt{4 S_t^2 + (S_L - S_p)^2} \right]$$

where

$$S_L = S_{lp} + S_b \text{ both terms of which have been previously defined.}$$

S_b , BENDING
DUE TO
EXPANSION

S_t = shear stress due to torsion as defined previously

S_p = circumferential pressure stress

In regard to this last term, the inside diameter and outside diameter (Barlow) formulas were first used for a thickness/diameter ratio below and above 0.1, respectively. These were later supplanted by the mean diameter formula and, more recently, by the universally adopted formula approximating the results of the Lamé' formula. All these formulas may be expressed in the common manner as follows

$$S = \left(\frac{pr_i}{t} \right) + K p$$

where

p = internal pressure

r_i = inside radius

t = wall thickness

K = constant having values between 0 and 1.

If K is given the value of 0, the inside diameter formula is obtained; for $K = 0.5$, the mean diameter; for $K = 1$, the outside diameter. When the value of 0.6 is used, stresses are obtained which correlate reasonably well for values up to about 0.5 r_i with the recognized inside circumferential stress formula of Lamé'. This approximation, discovered by H. C. Boardman, was rapidly adopted for moderate-temperature piping by both the Pressure Vessel and Piping Codes.

This formula for circumferential pressure stress, therefore, takes the form,

$$S_p = \frac{p}{Et} (r_i + 0.6 t)$$

t = wall thickness = 0.219 in.

r_i = inside radius = $\frac{1.939}{2} = 0.969$ in.

$\frac{t}{r_i} = \frac{0.219}{0.969} = 0.226$ which is less than 0.5; and therefore, 0.6 K value correlation is all right to use.

E = weld joint efficiency.

This term is a holdover from riveted construction; but with present examination procedures and pressure and mechanical testing, it is not difficult to provide design strength equal to the base material. Assuming that $E = 1.0$,

$$S_p = \frac{2200}{1 \times 0.219} (0.969 + 0.6 \times 0.219)$$

$$= 11,070 \text{ psi}$$

$$S_L = 4420 + 20,000 = 24,420 \text{ psi}$$

$$S_t = 13,000 \text{ psi}$$

$$S = 1/2 \left[24,420 + 11,070 + \sqrt{4 (13,000)^2 + (24,420 - 11,070)^2} \right]$$

$$= 32,345 \text{ psi}$$

$$S_A = 1.25 (S_C + S_H)$$

$$= 1.25 (18,750 + 14,900)$$

$$= 42,060 \text{ psi.}$$

Consequently, 32,345 psi is below allowable of 42,060 and meets code requirements.

If the stresses are all within the elastic limit and reversals of stress are strictly of the mechanical type, the fatigue endurance limit should be investigated [1]. Experimental fatigue tests have been conducted for nearly all common materials for these stress conditions and have been usually represented in the form of the Goodman diagram relating endurance limit to the ultimate stress. The theory is not now in accord with current theories, but can be used for the unlikely case of pure mechanical stress reversal by merely selecting values from existing graphs. The only defense of the Goodman diagram is that its errors, however large, usually lie in the safe direction.

In most piping systems and particularly in in-pile tube design, the simplified approach given in the foregoing simply will not suffice inasmuch as the temperature factor must be given greater weight. The two main temperature effects are the slow yield, commonly called creep, and that due to the stresses set up on the walls of the pipe or pressure vessel by virtue of a temperature gradient, commonly called thermal stresses. No methods of using creep data in design are covered by the Code. In past designs reviewed by the author, a figure of 1 percent creep on 100,000 hours of continuous load has been taken as a suitable limit. Refer to References 124 and 125 (Chapter VI of this volume) for further study of the creep phenomena.

The temperatures of the inside and outside walls of the pipe or pressure vessel are obtained from the heat transfer analysis. In many instances, it will be found advantageous to tabulate the principal pressure and thermal stresses as shown on page G-6.

Radius	Tangential Stress		Radial Stress		Longitudinal Stress	
	Pressure	Thermal	Pressure	Thermal	Pressure	Thermal
Inner radius	---	---	---	---	---	---
1. R. + convenient	---	---	---	---	---	---
increments, say .05 in.	---	---	---	---	---	---
"	---	---	---	---	---	---
"	---	---	---	---	---	---
"	---	---	---	---	---	---
"	---	---	---	---	---	---
"	---	---	---	---	---	---
"	---	---	---	---	---	---
Outer radius	---	---	---	---	---	---

The basic formula for the tangential pressure stress is derived from the Lamé' theory and can be given as:

$$S_T \text{ at point } r = \frac{p_1 - p_2 K^2 + (r_2^2 / r^2) (p_1 - p_2)}{K^2 - 1}$$

where

p_1 = internal pressure at internal radius, r_1

p_2 = external pressure at external radius, r_2

$K = r_2 / r_1$

when external pressure, p_2 , is zero, the equation becomes:

$$S_T = \frac{p_1 (1 + r_2^2 / r^2)}{K^2 - 1} .$$

The basic formula for the radial pressure stress is:

$$S_R \text{ at point } r = \frac{p_2 K^2 - p_1 + (r_2^2 / r^2) (p_1 - p_2)}{K^2 - 1} .$$

The basic formula for the longitudinal pressure stress is:

$$S_L = \frac{p_1}{K^2 - 1} .$$

Both the tangential and radial stresses are largest at the inner surface. All stresses are designated as positive when in tension and negative in compression. The radial stresses will ordinarily be in compression and are negative. The longitudinal stress is constant from inside to outside radius.

The basic formula for the corresponding tangential, radial, and longitudinal thermal stresses are given as follows:

$$S_T = \frac{E\alpha(t_2 - t_1)}{2(1 - \nu)} \left(\frac{K^2}{K^2 - 1} \frac{t_2 - t}{t_2 - t_1} + \frac{1}{K^2 - 1} \frac{t - t_1}{t_2 - t_1} \frac{r_2^2/r}{K^2 - 1} \right)$$

$$S_R = \frac{E\alpha(t_2 - t_1)}{2(1 - \nu)} \left(\frac{K^2}{K^2 - 1} \frac{t_2 - t}{t_2 - t_1} + \frac{1}{K^2 - 1} \frac{t - t_1}{t_2 - t_1} + \frac{r_2^2/r}{K^2 - 1} - \frac{1}{\ln K} \right)$$

$$S_L = \frac{E\alpha(t_2 - t_1)}{(1 - \nu)} \left(\frac{K^2}{K^2 - 1} \frac{t_2 - t}{t_2 - t_1} + \frac{1}{K^2 - 1} - \frac{t - t_1}{t_2 - t_1} - \frac{1}{2 \ln K} \right)$$

where

E = modulus of elasticity

α = coefficient of thermal expansion

ν = Poisson's ratio

t's = temperatures at points indicated

r's = radii at points indicated

$$K = r_2/r_1.$$

All formulas [2] are for thick-walled, hollow cylindrical tubes stressed not beyond the elastic limit of the material. The thermal stress formulas are for only one of a great many heat transfer systems [3].

Following the tabulation, the stresses can be added arithmetically and compared to the code allowable or the material yield point at 0.2 percent offset at operating temperature.

B. NONCODE THERMAL STRESS DESIGN EXAMPLE

For derivation of equations, refer to ASME Technical Paper No. 59-Met-4, K. R. Merckx, Hanford Atomic Products Operation, Richland, Washington, "Cyclic Operation of Pressure Piping with Gamma Heating", and corrections dated May 12, 1959. The calculations presented in the report were on the basis of 30 watts per gram. The calculations below are corrected to 20 watts per gram.

Assumed conditions:

$$p = 200 \text{ psig}$$

$$q = 20 \text{ w/g} = 2,630 \text{ w/in.}^3 \text{ (}\gamma\text{ - heating)}$$

$$a = 1.710 \text{ in. (inner radius)}$$

$$t = 0.242 \text{ in. (tube thickness)}$$

$$k = 11.7 \text{ Btu/hr} - ^\circ\text{F} - \text{ft} = 0.285 \text{ w/in. ("Stainless Steel Handbook," Allegheny Ludlum Steel Corp., at 700°F)}$$

$$\nu = 0.28 \text{ (Poisson's Ratio)}$$

$$E = 21 \times 10^6 \text{ psi (Elastic modulus)}$$

$$\alpha = 10.3 \times 10^{-6} \text{ in./in./F (Coefficient of thermal expansion, 32-932°F, "Metal Progress", July 15, 1954)}$$

$$U_o^2 = \left(\frac{a+t}{a} \right)^2 = 1.304 \quad \ln U_o^2 = 0.2654$$

Calculations:

$$\Delta T = \frac{q a^2}{4K} (U_o^2 \ln U_o^2 + 1) = \frac{2,630 \times 1.710^2}{4 \times 0.285} (1.304 \times 0.2654 + 0.304)$$

$$= 284^\circ\text{F (Temperature drop across the tube)}$$

$$\text{Flux} = \frac{a}{2} (U_o^2 - 1) q = \frac{1.710 \text{ in.}}{2} \times 0.304 \times 2.63 \text{ kW} \times 144 \frac{\text{in.}^2}{\text{ft}^2} \times 3413 \frac{\text{Btu/hr}}{\text{kW}}$$

$$= 336,000 \text{ Btu/hr} - \text{ft}^2 \text{ (heat flux on the inner surface of the tube)}$$

Circumferential and axial stress due to γ -heat at the inner surface.

$$S_R = S_T = \frac{E \alpha q a^2}{8K (1-\nu)} \left(\frac{2 U_o^4 \ln U_o^2}{U_o^2 - 1} - 3 U_o^2 + 1 \right)$$

$$= \frac{21 \times 10.3 \times 2,630 \times 1.710^2}{8 \times 0.285 \times 0.72} \left(\frac{2 \times 1.304^2 \times 0.2654}{0.304} - 3 \times 1.304 + 1 \right)$$

$$= 57,700 \text{ psi}$$

Observe that nearly the same answer can be obtained from an entirely different method as follows:

For a thin wall cylinder ($R/t \gtrsim 10$), within a parabolic temperature distribution, the maximum thermal stress is given by:

$$\sigma_{th} = 2/3 \times \frac{E \alpha}{(1-\nu)} \Delta T \text{ maximum}$$

$$= 2/3 \times \frac{21 \times 10.3}{0.72} \times 284$$

$$= 57,700 \text{ psi}$$

From the ASME Code, at 2000 psig, the allowable stress is:

$$S_A = p \left(\frac{a}{t} + 0.6 \right) = 2000 \frac{1.71}{0.242} + 0.6 = 15,300 \text{ psi}$$

$$\begin{aligned}\Delta\epsilon &= \frac{S_T + S_A - 2 SY}{E}, \text{ where } SY = \text{yield stress 347 stainless} = 30,000 \text{ psi} \\ &= \frac{57,700 + 15,300 - 60,000}{21 \times 10^6} \times 100 \\ &= 0.06 \text{ percent per cycle}\end{aligned}$$

From Mr. Merckx's corrections of 5/12/59 for Case I, $\Delta\epsilon = 0.182$ percent. Therefore, with the revised gamma heat figure, the plastic strain per cycle is only 1/3 the original estimate and is much lower than the range of strains causing low cycle fatigue in 347 stainless steel [5].

An alternate method, perhaps as satisfactory and certainly easier to apply, is that developed by B. F. Langer on low cycle fatigue (Reference 134, Chapter VI of this volume) and D. R. Miller on ratcheting (Reference 135, Chapter VI of this volume) as follows:

Assuming the maximum thermal stress to be 57,700 psi as found above, assume a yield stress of 30,000 psi at 2 percent offset and, say, 900°F temperature. The ratio of allowable stress to yield stress is:

$$\frac{S_A}{S_Y} = \frac{15,300}{30,000} = 0.51$$

From Figure G-1 the Miller ratchet limit is 2.7. The maximum allowable thermal stress to prevent ratcheting is therefore:

$$\sigma_{th} = 2.7 \times 30,000 = 81,000 \text{ psi.}$$

Since the actual $\sigma_{th} = 57,700$ psi, the tube should not ratchet. In order for the tube to have a 2 percent yield stress of 30,000 psi at 900°F, the material must be purchased to have a cold yield stress of:

$$30,000 \times \frac{18,750^{[6]}}{15,300} \approx 40,000 \text{ psi.}$$

In order to determine the cycle life of the tube, the maximum and minimum stress intensity during cycling must be known.

The minimum stress is taken as the noncyclic pressure stress:

$$\sigma_{\min} = \frac{P(a^2 + b^2)}{b^2 - a^2}$$

where

p = design pressure, psi

a = inside diameter, in.

b = outside diameter, in.

$$\sigma_{\min} = \frac{2000 (3.904)^2 + (3.420)^2}{(3.904)^2 - (3.420)^2} = \frac{53,860}{3.53} = 15,250 \text{ psi}.$$

The maximum stress is taken as the combined pressure and thermal stress:

$$\sigma_{\max} = 15,250 + 57,700 = 72,950 \text{ psi}$$

$$\sigma_{\text{mean}} = \frac{\sigma_{\max} + \sigma_{\min}}{2} = \frac{72,950 + 15,250}{2} = 44,100 \text{ psi}$$

$$\sigma_{\text{alt}} = \frac{\sigma_{\max} - \sigma_{\min}}{2} = \frac{72,950 - 15,250}{2} = 28,850 \text{ psi}.$$

Since σ_{mean} exceeds the yield stress, the real value of σ_{mean} is:

$$\sigma_{\text{mean}} = 1/2 (\sigma_{\max} + \sigma_{\min}) - (\sigma_{\max} - \sigma_{\text{yield}})$$

$$= 44,100 - (72,950 - 30,000)$$

$$= 44,100 - 42,950$$

$$= 1,150 \text{ psi, or essentially zero.}$$

$$\sigma = \sigma_{\text{alt}} = 1/2 (\text{thermal or membrane stress which is } 57,700 \text{ psi})$$

$$= 28,850 \text{ psi.}$$

The number of cycles allowed for these stresses is $\approx 60,000$ as taken from Figure G-2. This exceeds the 2000 thermal cycles expected during the lifetime of the tube by a factor of 30.

Fig. G-1 Allowable thermal stress if cyclic growth is to be avoided.

Fig. G-2 Stress versus fatigue life.

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1. Fatigue and Fracture of Metals, edited by W. M. Murray, John Wiley and Sons, 1952.
2. Chapter 6, High Pressure Technology, E. W. Comings, McGraw-Hill, 1956.
3. For more simple and complex systems, see Theory of Elasticity, Timoshenko and Goodier; Formulas for Stress and Strain, R. J. Roark, McGraw-Hill, 1954, 3rd Edition; Theory of Thermal Stresses, Bruno A. Boley, John Wiley and Sons, 1960; Thermal Stresses in Concentrically Heated Hollow Cylinders, ANL-6224; J. C. Heap, Thermal Stresses in Tubes, WAPD-P-543. Also refer to Chapters 5 and 6, The Reactor Handbook, Volume 4, Engineering, Stuart McLain and John H. Martens, Interscience Publishers, 1964.
4. Also see, "Pipe Thickness which Minimize Stresses Due to Internal Pressure and Gamma Heating", K. R. Merckx, Nuclear Science and Engineering, Vol. 3, No. 6, June 1958.
5. "An Investigation of Thermal Stress Fatigue as Related to High Temperature Piping Flexibility", L. F. Coffin, Sr., ASME Paper No. 56-A-178.
6. From the ASME Boiler Code, Section VIII.

APPENDIX H

SAMPLE CALCULATIONS: RUPTURE DISC AND RELIEF VALVE SIZING

A. PRESSURIZER

Fission and gamma heat (core section) = 3.63 MW

Design pressure of pressurizer = 2880 psig at 690°F

Latent heat of evaporation at 2880 psig = 260 Btu/lb

$$\begin{aligned}\text{Required capacity of relief valve} &= \frac{3415 \text{ Btu/kW-hr} \times 3630 \text{ kW}}{260 \text{ Btu/lb}} \\ &= 47,700 \text{ lb/hr}\end{aligned}$$

1. ∴ Condition to be relieved = 47,700 lb/hr steam
2. Set pressure of relief valve =
Set pressure of rupture disc (upstream) -- (tolerance of rupture disc setting + tolerance of relief valve setting + safety factor).
The above is required to guarantee that relief will pop whenever rupture disc bursts.

Tolerance of rupture disc = plus or minus 5 percent set pressure (ASME unfired pressure vessel code)

Tolerance of relief valve = plus or minus 3 percent set pressure (ASME unfired pressure vessel code)

Safety factor (assumed) = 5 psig.

Set pressure = 3000 - (150 + 90 + 5) = 2755 psig (Round off to 2700 psig)

3. Area required (Reference -- Manning, Maxwell, and Moore, Catalog 1900, 1900A, p 170 [a])

$$A = \text{orifice area, sq in.} = \frac{V_{ph}}{50 P}$$

V_{ph} = required capacity, lb/hr = 47,700 lb/hr

P = flowing pressure, psia = set pressure + overpressure + 14.7

Overpressure = 10 percent set pressure (ASME unfired pressure vessel code)

$$\therefore P = 2700 + 270 + 15 = 2985 \text{ psia}$$

$$\therefore A = \frac{47,700}{50 \times 2985} = 0.320 \text{ in.}^2$$

[a] For other sources of information cf. "References" at end of Appendix H.

4. Area provided: (Reference -- Manning, Maxwell, and Moore, Catalog 1900 A, page 13,

$$\text{"G" orifice area} = 0.503 \text{ in.}^2$$

As method of check for sizing of relief valve:

Required orifice capacity = 47,700 lb/hr steam

"G" orifice at set pressure of 2700 psig has an orifice capacity of 75,000 lb/hr steam

5. Valve Size

Valve size = 2 in. x 3 in.

B. HEAT EXCHANGER

The basis for sizing a heat exchanger relief valve is similar to "Marine Engineering Regulations and Material Specifications", whereby it is considered that one tube of the heat exchanger completely ruptures at its manifold connection.

1. Condition to be relieved:

Saturation temperature at set pressure is 399°F. This is less than the tube temperature (650°F). Therefore, flashing will occur (liquid + steam relief). (Set pressure is equal to design pressure of shell = 230 psig.) Flow of water from tube to shell side:

$$Q = K A_t \left[2g (H_t - H_s) \right]^{1/2} = \text{ft}^3/\text{sec}$$

K = coefficient = 0.62

$$2g = 64.4 \text{ ft/sec}^2$$

$$H_t = P_t \times 2.3 / \text{sp gr}_t = \text{head of liquid, tube side} = \text{ft}$$

P_t = pressure, tube side = psig

sp gr_t = specific gravity at tube side temp (650°F)

$$\therefore H_t = 2500 \times 2.3 / 0.558 = 10,000 \text{ ft of liquid}$$

$$H_s = P_s \times 2.3 / \text{sp gr}_s = \text{head of liquid, shell side} = \text{ft}$$

P_s = pressure, shell side = psig

sp gr_s = specific gravity at shell side temperature (130°F)

$$\therefore H_s = 230 \times 2.3 / 1.0 = 530 \text{ ft of liquid}$$

A_t = tube area based on OD tube (rupture at manifold worst condition)

$$A_t = 0.785 D^2 / 144 = \text{ft}^2$$

$$= 0.785 \times (1/2)^2 / 144 = 0.00136 \text{ ft}^2$$

$$\therefore Q = 0.62 \times 0.00136 \left[64.4 (10,000 - 530) \right]^{1/2}$$

$$Q = 0.602 \text{ ft}^3/\text{sec}$$

$$= 0.602 \text{ ft}^3/\text{sec} \times 62.4 \text{ lb/ft}^3 \times 0.558 \text{ sp gr} \times 3600 = 81,000 \text{ lb/hr liquid}$$

Heat balance calculations will show percentage of liquid flashing into steam:

Let Q_t = enthalpy of liquid, tube side

Let Q_s = enthalpy of shell side = percent liquid remaining x enthalpy of liquid (shell side) + percent liquid flashed x enthalpy of vapor.

Let (x) = percent liquid

Let (1-x) = percent liquid flashed

For 1 lb of steam + water

From the steam tables:

$$Q_t = 695 \text{ Btu/lb} \quad Q_s = (x) \times 374 \text{ Btu/lb} + (1-x) \times 1200 \text{ Btu/lb} \quad Q_t = Q_s$$

$$\therefore 695 = 374 x + 1200 (1-x)$$

$$x = 0.611 \text{ water}$$

$$1-x = 0.389 \text{ steam}$$

$$81,000 \text{ lb/hr liquid} = 0.389 \times 81,000 \text{ (steam)} + 0.611 \times 81,000 \text{ (water)} = 31,500 \text{ lb/hr steam} + 49,500 \text{ lb/hr water}$$

$$\frac{49,500 \text{ lb/hr}}{(8.34 \text{ lb/ft}^3) (0.859 \text{ ft}^3 \text{ gal}) (60 \text{ min/hr})} = 115 \text{ gpm}$$

Calculation for Area Required (Reference Consolidated Catalog 1900):

$$\text{a. For Water; } A = \text{C.F.} \times \text{Lgm} \times \sqrt{\text{sp gr}} / 27.2 \sqrt{P} = \text{in.}^2$$

Lgm = required liquid capacity, gpm = 115 gpm

sp gr = specific gravity at 400°F = 0.86

P = set pressure = design pressure of shell = 230 psig

C.F. (for 10 percent overpressure) = 0.6

$$\therefore A = 0.6 \times 115 \sqrt{0.86} / 27.2 \sqrt{230} \quad A = 0.155 \text{ in.}^2$$

b. For steam;

$$A = V_{ph}/50 P_1$$

$$V_{ph} = 31,500 \text{ lb/hr}$$

$$P_1 = 230 + 23 + 15 = 268 \text{ psia}$$

$$\therefore A = 31,500/50 \times 268 = 2.35 \text{ in.}^2$$

$$c. \text{ Combined areas} = 0.155 + 2.35 = 2.505 \text{ in.}^2$$

3. Area Provided

$$\text{"L" orifice area} = 2.853 \text{ in.}^2$$

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4. High Pressure Equipment Company, Inc, 1222 Linden Avenue, Erie, Pennsylvania.