**QUARTERLY TECHNICAL
PROGRESS REPORT
ON AEC-SPONSORED ACTIVITIES**

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2
1
2

2

2

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ABSTRACT

This report is a summary of the technical progress made by Atomic Power Development Associates, Inc., in the period January through March 1970 under AEC Contract No. AT (11-1)-865. The individual projects are covered under the following headings:

Sodium-Water Reaction Test Program
Project Agreement No. 10

Sodium Technology Project
Project Agreement No. 11

Comparison of Methods for Heterogeneity Analysis
Project Agreement No. 19

1

2

3

4

5

6

TABLE OF CONTENTS

LIST OF ILLUSTRATIONS.	<u>Page</u> vi
LIST OF TABLES	vi

Sodium-Water Reaction Test Program - Project Agreement No. 10- H. V. Chamberlain	10.1
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Data are presented on two material wastage tests with a 2-inch injection point-to-target spacing. Examination of the tube bundle following these tests revealed large deposits of reaction products in the bundle. Also presented is a detailed analysis of all wastage data accumulated in P.A. 10 tests.

Sodium-Technology Project - Project Agreement No. 11 - J. E. Meyers	11.1
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The status of various topical reports is given.

Comparison of Methods for Heterogeneity Analysis - Project Agreement No. 19 - J. B. Nims and E. M. Page	19.1
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The calculational approach for obtaining the effects of heterogeneity on reactivity and sodium voiding is described and an outline is presented of the method used in the IDIOT code for heavy-element resonance cross section self-shielding calculations.

LIST OF ILLUSTRATIONS

<u>Fig. No.</u>		<u>Page</u>
10.1	Effect of Leak Rate and Spacing on Wastage Rate of Target Tubes.	10.22
10.2	Elevation Cross Section of Tube Bundle Assembly . .	10.23
10.3	Correlation of Velocity-Modified Wastage with Leak Rate	10.24
10.4	Correlation of Velocity-Modified Wastage with Orifice Diameter	10.25
19.1	Specification of the Moderator Regions for Self-Shielding Calculations by IDIOT Code of Drawer Configurations of ZPR-III Assembly 48	19.11
19.2	Specification of the Moderator Regions for Self-Shielding Calculations by IDIOT Code of Drawer Configurations of ZPR-IV Assembly 6	19.12

LIST OF TABLES

<u>Table No.</u>		<u>Page</u>
10.1	Summary of Preliminary Data from Rig-10 Tests. . .	10.4
10.2	Summary of Rig-43 Steam Injection Tests	10.16

SODIUM-WATER REACTION TEST PROGRAM

PROJECT AGREEMENT NO. 10

PROJECT ENGINEER: H. V. CHAMBERLAIN

I. PREFACE

A. Background Information

Achievement of the LMFBR program objectives is highly dependent on the development of reliable system components. With regard to the current status of development, the steam generator is recognized to be the weakest link in the heat transport system. The question of what occurs in a steam generator during a sodium-water reaction and how to accommodate such a reaction in the design has been a problem since the inception of sodium- and NaK-cooled reactors. The damage resulting from a reaction may be caused by water leaks of two general categories: large leaks and small leaks. Until recently designers have been concerned with large leaks such as those resulting from the complete circumferential rupture of a water tube. However, in late 1962 a steam generator in the Enrico Fermi Atomic Power Plant (EFAPP) experienced a sodium-water reaction due to a tube failure produced by vibration. Subsequent examination of the unit indicated that a general corrosion had occurred in the area of the sodium-water reaction, producing wastage and failure of some of the adjacent tubes. It is not known at what stage in the propagation of the leaks the wastage occurred; that is, did the wastage occur as a result of the initiating leak, or did it occur as a result of the subsequent leaks? The questions raised by this experience clearly indicate that a sodium-water reaction program should consider both small and large leaks. Subsequent to the EFAPP experience, the British conducted some tests in which they found that significant tube wastage rates can be incurred with small leaks.

B. Scope

As part of the AEC program of steam generator development, a project was initiated at APDA for the purpose of investigating both small and large leaks. A series of tests is being run to

- Establish a means of estimating the extent of tube wastage during a small leak of water into sodium in a sodium-heated steam generator.

- Determine the dominant parameters of tube wastage during a small water leak so that these parameters may be taken into account in the design of a unit.

C. Relationship to Other Projects

This program is applicable to the development of sodium-heated steam generators and to the technology of sodium-water reactions in the LMFBR program. Data generated in the program will be used to prepare design criteria and design analysis procedures for steam generators from the standpoint of sodium-water reactions. All of the activities of this program are within the scope of the Components Element (Vol. 3) of the LMFBR Program Plan (Wash 1103). However, some of the activities interface with the Instruments and Control Element (Vol. 4) of the Program Plan. With regard to sodium-water reactions in steam generators, the emphasis in Program Element 3 is on developing information related to mechanisms producing wastage and to the effect of various system parameters and jet characteristics on material wastage. In Program Element 4, the emphasis is on evaluating and on demonstrating leak detection systems which would minimize damage to the steam generator and secondary heat transport system.

Information on the test program and the results to date are available in the proceedings of the Sodium Components Development Program Information Meetings held on April 14, 1964, in Hollywood, California, and on June 17, 1965 in Chicago, Illinois; APDA quarterly technical progress reports on AEC-sponsored activities; in the following papers: "An Investigation of Tube Wastage Caused by Small Leaks of Water in Sodium-Heated Steam Generators," presented at the IAEA Symposium on Alkali Metals - Corrosion Studies and Systems Operating Experience, held at Vienna, Austria, November 28 to December 2, 1966; "Tube Wastage in Sodium-Water Reactions" and "Surface Temperature Measurements in Sodium-Water Reactions," presented at the American Nuclear Society Winter Meeting in Chicago, Illinois, in November, 1967; "Sodium-Heated Steam Generators," presented at the American Nuclear Society National Topical Meeting on Fast Reactor Systems, Materials, and Components, held in Cincinnati, Ohio, in April, 1968; and "Wastage of Steam Generator Materials by Sodium-Water Reactions," presented at International Conference on Sodium Technology and Large Fast Reactor Design, held at Argonne National Laboratory, Argonne, Illinois in November 1968; and a report, "Evaluation of Materials Wastage Due to Reactions of Water in Sodium," APDA-227, June, 1969.

II. TECHNICAL PROGRESS

A. Rig-10 Tube Wastage (Task 20)

J. A. Ford
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The objective of this task is to establish a means of estimating the extent of tube wastage during a small leak of water into sodium in a sodium-heated steam generator.

1. Test Planning (Job 02)

The scope of the FY-70 program is to conduct 12 material wastage tests by the end of June 1970. The tests will be made using 2-1/4 Cr-1 Mo steel as the target material with nozzle-to-target spacings of 1/4 inch, 1 inch, and 2 inches. The water injection rate will range from approximately 0.005 to 1.5 lb/sec; the bulk sodium temperature will be approximately 600 F. The objectives of the tests are to (1) clearly confirm the phenomenon of a decreasing wastage rate that was observed at leak rates beyond approximately 0.01 lb/sec, (2) determine the effect of nozzle L/D on the wastage pattern, and (3) obtain preliminary wastage data for a 2-inch leak-to-target spacing.

2. Testing (Job 03)

Two additional wastage tests (Tests No. 52 and 53) were completed during this report period. A summary of the test data is given in Table 10.1, and the wastage results for the target tubes are shown in Figure 10.1. The purpose of these two tests was to obtain data needed to establish a wastage curve for a 2-inch injection point-to-target spacing using 2-1/4 Cr-1 Mo steel as the target tubes.

The tube bundle configuration was modified to conduct these tests with the 2-inch spacing by removing a 4-inch-long section of tubes C-3 and C-4 (normal target tube) in front of the nozzle and by converting tube B-4 into the target tube. This tube (B-4) had 1/8-inch stainless steel tubing connected to both ends for monitoring the pressure inside the target tube during the injection period.

Test No. 52 was conducted with a leak rate of 0.23 lb/sec. The examination of the temperature chart showed that a significant rise in temperature occurred when the test was triggered and none of the target thermocouples showed a decrease in temperature below that of the bulk sodium. Inspection of the tube bundle showed a very light coating of reaction products. Cleaning of the tube bundle by use of a steam lance was readily accomplished. Examination of the bundle after steam-cleaning showed the presence of general wastage over a wide area. The deepest penetration on the target tube

TABLE 10.1 - SUMMARY OF PRELIMINARY DATA FROM RIG-10 TESTS

	Test No.	
	52	53
<u>SODIUM SYSTEM</u>		
Flow rate, gpm	400	400
Velocity past target tube, ft/sec	2	2
Bulk temperature, F	600	610
Sodium level above injection point, ft	8.5	8.5
Plugging temperature, F		
Before injection	400	400
After injection	>620	>635
<u>INJECTION WATER SYSTEM</u>		
Water added, lb	10	20
Temperature, F	600	600
Pressure, psig	2650	2650
Orifice size, in.	0.043	0.085
Length of capillary, in.	0.31	0.31
Capillary length-to-diameter ratio, L/D	7.2	3.7
Injection point-to-target spacing, in.	2	2
Injection duration, sec	43.0	25.2
Injection rate, lb/sec		
Predicted	0.19	0.70
Actual	0.23	0.79
<u>RECIRCULATING WATER SYSTEM</u> (Target Tube Coolant)		
Coolant	N ₂	N ₂
Pressure, psig	100	100
<u>TARGET TUBE WASTAGE</u>		
Tube material	2-1/4 Cr-1 Mo	2-1/4 Cr-1 Mo
Wastage pattern	General	General
Depth of penetration, mils	29	11
Wastage rate, mils/sec	0.67	0.44
Wastage rate, mils/lb H ₂ O	2.9	0.55
Maximum measured tube temperature, F*	1860	1930
Time of maximum temperature, sec	26.3	26.5
<u>COVER GAS</u>		
Pressure, psig		
Before injection	5.7	5
During injection, peak	~ 20	~ 25
Hydrogen concentration		
Before injection, ppm	40,000	14,000
Peak during test, ppm	480,000	311,000
Change in concentration, ppm	440,000	297,000
Rate of change, ppm/sec	5,700	6,400
Elapsed time between leak initiation and initial H ₂ change, sec	40	74
Elapsed time between leak initiation and peak concentration, sec	176	160

* The maximum temperature of 2060 F occurred on TC No. 22 on tube A-4 at 41.0 sec (Test No. 52). The maximum temperature of 2090 F occurred on TC No. 22 on tube A-4 at 20.5 sec (Test No. 53).

was approximately 29 mils, resulting in a wastage rate of 0.67 mil/sec. The impression of some of the thermocouples and clips remained on the tube surface where they had protected the tube surface from being wasted.

Wastage was also observed on other tubes in the bundle as well as on the inner cylinder. Tubes A-2, A-3, A-4, A-5, and D-3 had wastage rates up to approximately 1 mil/sec, and the inner cylinder had a wastage rate of approximately 0.5 mil/sec. The relative locations of the tubes are shown in Figure 10.2. Observance of wastage on tube D-3 indicates that there was flashback of part of the jet from either the target tube or from the B row of tubes. The presence of wastage on the inner cylinder strongly suggests that wastage could have been obtained on other tubes farther removed from the leak if they had been present. This means that if a steam generator has undergone a leak of this magnitude, inspection of tubes over a wide area would be required instead of merely plugging a few tubes around the one that leaks.

Test No. 53 was conducted with a leak rate of 0.79 lb/sec. Analysis of the temperature chart showed a very unusual behavior in the target temperatures measured during the test. Immediately after the injection all of the thermocouples on the target tube indicated a rapid rise in temperature, and then all of the thermocouples indicated temperatures below 600 F, the bulk sodium temperature. Then for a period of approximately 10 seconds, temperatures remained below 600 F with only one or two of the thermocouples occasionally showing temperatures above 600 F. In the latter period of the injection all of the thermocouples generally indicated temperatures above 600 F. The low temperature readings were probably caused by the 2-inch spacing and the resulting larger diameter of the core of liquid in the jet that is cooled by flashing of a portion of the injected water.

Examination of the tube bundle after the test showed the presence of large deposits of reaction products in the bundle and on both sides of the upper flange of the outer shroud. Deposits were also noted among the tubes on the back side of the tube bundle. The target area was splattered with white deposits but had no large deposits. Examination of the bundle again showed that extensive wastage had occurred. The target tube had been wasted to a depth of 11 mils, resulting in a wastage rate of approximately 0.44 mil/sec. Tubes A-3, A-4, A-5, B-3, B-5, and C-4 had wastage rates up to approximately 1 mil/sec. As was observed in Test No. 52, there was also wastage on the surface of the inner cylinder (10 to 15 mils), resulting in a wastage rate of 0.4 to 0.6 mil/sec.

Examination of Figure 10.1 shows that the wastage rates obtained in these two tests are higher than those obtained with injection-to-target spacing of 1/4 and 1 inch at these same leak rates. Thus, the increase in wastage rates, and over a much wider area, with the increase in spacings at the larger leak rates is just opposite to that observed at lower leak rates.

This shift in the behavior can be explained on the basis of the core of liquid in the jet. For the lower leak rates, the argument was made that the maximum in the wastage curves was caused by the presence of a core of subcooled liquid that protects the surface from wastage and that this effect becomes more pronounced as the leak rate increases. This is evidenced by the change from a pit-type to a toroidal wastage pattern. At the closer spacing, the core of liquid at the higher leak rates is large enough to protect most of the surface from wastage, thus the occurrence of lower wastage rates as the leak rate increases. It has also been noted that the sodium-water reaction appears to occur over a very narrow band between the water-steam jet and the surrounding sodium. For large leaks and close target spacings, the injection fluid probably produces a spray of such magnitude that lower wastage rates are obtained. As the target is moved away from the leak it is conceivable that less spray is formed and that a part of the jet can be concentrated enough to cause appreciable wastage on the target. The appearance of wastage over a much wider area at these higher leak rates than those observed at closer spacings tends to support this theory. One additional observation that also supports this argument is that wastage began to occur on tubes in the rows behind the target row as the leak rate increases. This indicates that the primary jet from the capillary is broken up into smaller streams and in some cases a more damaging water-steam mixture is formed that causes wastage some distance away from the injection nozzle.

3. Test Analysis (Job 04)

The scope of this job is to perform additional detailed analyses of all Project-10 wastage data to (1) provide a better understanding of the material wastage phenomena, (2) determine the most meaningful correlation of the data, and (3) provide guidance for the experimental program to be conducted in FY-70.

a. Jet Characteristics

Dr. S. A. Weil*

Because of the complexity of the sodium-water reaction, development of a quantitative model from first principles that describe the wastage process is not possible because not enough is known about the mechanism producing the wastage. But an empirical correlation may point to the dominant mechanistic factors, and should allow estimation of rates under conditions not directly observed. Possible correlations of the wastage rate Z (mils/sec) with the parameters of orifice diameter d_o (mils), injection point-to-target distance X (mils), and leak rate W (lb/1000 sec) may be based on a mechanical, chemical, or combination wastage mechanism. To investigate the mechanical factors, the characteristics of ideal jets may be

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used as guides even though the real jets are far from ideal. Among the possible chemical factors are the amount and nature of the products of the sodium-water reaction and the temperature. The chemical complexity of the system makes it almost impossible to assign a quantitative value to anything but the temperature among the chemical rate-determining characteristics.

The important characteristics of ideal jets are expressed in terms of the orifice diameter d_o and the velocity at the orifice v_o . One exception is the jet diameter, which is approximately proportional to the distance from the orifice and independent of the velocity. In the case of the Rig 10 wastage tests with the injection point-to-target distance X at 250 mils and the axial wastage pattern, the hole diameters (i.e., the diameter of the wasted area) are relatively constant (140 to 190 mils) and not related to d_o (5.3 to 10.6) or W/d_o^2 (0.03 to 0.1), the latter being a measure of v_o . Note that this observation is for leak rates below approximately 10^{-2} lb/sec and also is for the pit-type wastage patterns. This divergence angle, if that is what the wastage pattern represents, is comparable to that reported for an air jet in water¹; however, the axial wastage patterns at $X = 1000$, while much more scattered in diameter, are not four times as big, thus emphasizing that the axial wastage area, at least, represents a region of either high temperature or velocity. The runs resulting in toroidal wastage have patterns of completely different magnitude and correspond to divergence angles very much greater than can be reasonably viewed in the same regime as the tests in which a pit-type pattern was obtained. The remainder of this discussion will, therefore, be limited to the wastage tests on Rig 10 that gave the pit-type wastage patterns.

The ideal jet velocity along the jet axis varies inversely as the distance from a point somewhat behind the orifice (except for the first few diameters). In the system of interest we can expect something like

$$v_{\text{axial}} \propto \frac{v_o}{(X_r + a)^n} \quad (1)$$

where

X_r = axial distance relative to d_o

a = constant

v_o = velocity at the orifice

In the cases of gas-in-gas and gas-in-liquid jets, n would be about unity.¹ With the vaporizing liquid, n is unknown. If it is assumed that the condition of the water at the exit of the orifice is independent of the orifice diameter, it can be assumed that $v_o \propto W/d_o^2$. Note that there may be several regimes

of water conditions at the orifice, as indicated by the poorly defined relationship between W and d_o .

With the view that the mechanical contributions to wastage are related to velocity, dependence of wastage rate on $W/d_o^2 (x_r + a)^n$ may be expected. The dependence of the wastage rate on temperature may be expected to be of the form using the absolute temperature $T \exp(-A/T)$. The pertinent temperature would be that of the target at the region of wastage and presumably heated by the hot products of the sodium-water reaction. As an estimate of a possible order of magnitude and of potential importance, the temperature effect was sought from the reported results of the caustic corrosion studies. From the results at 1500 and 1700 F, the rate of reaction appears to increase by a factor of at least 2 over the 200 F range. This is an effect of a magnitude we hope to detect in the wastage results.

To test some of the above proposed relationships, it is necessary to make additional assumptions: for one, that the mechanical and temperature effects are separable and multiplicative; for another, that the temperature of the target is primarily a function of the leak rate W , all other conditions being the same.

In Figure 10.3, one application of the above concepts is presented. On the basis that the wastage is proportional to the velocity and that $a \ll X_r$, a reasonable correlation is obtained.

The major deviations from the drawn curve are not significantly affected by changing the values of n and a . Of the tests with large deviations, Test No. 11 with a total wastage of 2 mils cannot be considered to have a significant deviation even when it is 100%; Tests No. 4 cannot, in any analysis, be rationalized with Test No. 6, and the latter appears in general to be more consistent with the other data; Tests No. 32, 37, and 39 have deviations that are troublesome. Tests No. 37 and 39 have, in this correlation, low wastage rates that may indicate a transition to the toroidal wastage runs, which have very low wastage at the jet axis. There is some appearance in the Test No. 39 pattern similar to toroidal wastage.

Up to leak rates of the order of 3×10^{-3} lb/sec, the correlation appears to be of acceptable precision. A line is drawn on the log-log plot of Figure 10.3 quite arbitrarily with a wastage dependence as W^2 . The choice of an integral power is based on convenience. The assumption that it represents a temperature-time dependence voids any rational reason for an integral power dependence. It is obvious that any extrapolation of the line to higher leak rates could not be justified.

The use of the leak rate to account for the temperature effect can be rationalized as follows: If the ideal jet picture is used for the axial wastage run, the jet geometry is independent of the orifice diameter. The

leak rate represents the amount of material that can be heated to the maximum temperature and this material, impinging on the target, spreads out from the target-jet axis. The main reason (in the pit-type wastage pattern regime) for a target temperature below the maximum is heat loss through the target metal to the surroundings. Higher leak rates mean hotter fluid covering greater areas of the target, thereby reducing the heat loss from the primary impact area. Attempts to describe such a system accurately proved too difficult. A simple view which assumes, at the target surface, the fluid and solid have the same temperature and the fluid streamline geometry along the wall is independent of leak rate gives

$$\ln \left(\frac{T_{\max} - T}{T_{\max} - T_{\text{sur}}} \right) \approx \beta W \ln (y/y_{\infty}) \quad (2)$$

where

- $T_{\max}$ = temperature of jet
- T_{sur} = temperature of the surroundings
- T = temperature of the target reaction zone
- y = radius of the reaction zone
- y_{∞} = radius where target temperature is T_{sur}
- β = constant
- W = leak rate

The observed dimensions of the wastage patterns are roughly constant. The value of y_{∞} may be proportional to $W^{1/2}$ or a constant depending on details that would be hard to know. But in any case, the primary dependence of the right-hand side would be proportional to the leak rate. If $T_{\max}$ is very much greater than T , a linear T -versus- W relationship may be expected; if not, some estimate of $T_{\max}$ is necessary. Thermodynamically, complete reaction of sodium and liquid water both at 600 F would result in a maximum temperature of 2540 F. If there exists a reaction zone due to sodium at 600 and water vapor at 212 F, a maximum temperature of 2880 F is possible. Attempts to empirically relate W and the observed target temperatures failed due to the large scatter in the data; however, using an average curve for the temperature data over the range of $W < 4$ and taking into account that the temperature dependence of the wastage as shown in Figure 10.3 is as W^2 , then

$$\sum X_r d_o^2 / W \propto \exp (-15000/T) \quad (3)$$

Chemically, this corresponds to an energy of activation of 30,000 Btu/lb-mole.

Other attempts were made at empirically correlating the wastage rate with d_o , X , and W . Of these, only that shown in Figure 10.4 is of interest in that it gives, in $(Z X_r d_o^2 / W)$ versus d_o , what appears to be a more satisfying fit to the high W data than Figure 10.3. If the leak rate were a single-valued function of d_o , either W or d_o would serve in any correlation. The choice lies between a dependence on W which can be rationalized (but never proven) and on d_o which appears to have a better fit but has not yet been rationalized.

With respect to the systems that result in toroidal wastage, the concepts given above are not applicable. The magnitude of the wastage rate is very much less than anticipated on the basis of the above correlations. If there is some similarity in mechanism, then the use of the leak rate in comparing the two regions may be misleading. Presumably, the stream velocity where the wastage occurs should be used instead of $(W/d_o^2 \cdot X_r)$, the axial velocity. However, the extensive wastage on tubes very far from the orifice and from the jet axis suggests that velocity as a mechanical factor has little to do with the wastage.

Since the wastage rates of those tests resulting in toroidal wastage patterns lie in a narrow range (Test No. 47 which appears to have a low wastage rate could easily be viewed as having a toroidal pattern so wide as to be on the adjacent tubes), it can be inferred that the wastage is due to a temperature or chemical regime created by the mixing process and that the shape of the mixing regime is determined by the tube shape and arrangement. Attempts at quantitative correlation between wastage rates and patterns have not yet been successful.

The above analytical approach points out the need for experiments to establish the mechanism producing wastage of materials during a sodium-water reaction.

b. Impact of Liquid Droplets on Solid Surfaces M. K. Deora

The results of the basic studies of corrosion and erosion have strongly suggested that the wastage of the target tubes during a sodium-water reaction is due to an erosion mechanism. Yet the very limited results of the hot erosion studies suggested that possibly erosion by itself could not account for the high rates of wastage observed. However, it has been suggested that erosion is the primary mechanism and that the high wastage rates are caused by liquid droplets moving at very high velocities. Calculations indicate that the flashing stream of water and steam issuing from the capillary can attain very high velocities. Thus, a review of the literature was made to determine if such a phenomenon could produce these high wastage rates.

Many investigators² have examined the erosion problems associated with the impact of liquid droplets on solid surfaces. They usually begin with the water hammer equation which gives the impact pressure for a cylindrical jet of liquid striking a solid surface; that is,

$$P = CV\rho \quad (4)$$

where

P = impact pressure
 ρ = density of the liquid
 V = impact velocity of the liquid
 C = the speed of compression waves in the liquid

For moderate impact velocities, the value of C is the velocity of sound in liquid; however, the probability of C , the sound speed, being replaced by a shock speed is very important for liquid impact velocities of a few hundred meters per second. Taking as an example an impact velocity of 500 m/sec and using the data of Cook, Keyes, and Ursenbach,² a shock wave of speed 2500 m/sec should be developed, as compared with the stress wave (sound) speed of 1500 m/sec. The water hammer pressure thus becomes about 125 kg/mm² rather than 75 kg/mm².

Even if a jet strikes the plane surface obliquely, the maximum pressure (stagnation) is the same for all angles of incidence, but the area over which the high pressure acts is much reduced as the angle of incidence θ becomes small. Though a jet cannot produce a pressure greater than the stagnation pressure as a steady state, it seems theoretically possible to attain a much higher pressure for a short time when a very oblique jet is moved sideways. Measurements showed the flow velocity of the liquid along the surface to be as much as two to three times the impact velocity.

The duration of these high pressures is shown to depend upon the shape and size of the impacting liquid mass. In the case of an impacting liquid cylinder, for example, the duration depends upon the time taken for the release waves from the sides of the cylinder traveling at velocity C to reach the center of the cylinder.

Jenkins and Booker³ deduced impact pressures from measurements of the radial flow velocity from under a water drop during impact. They found agreement with values calculated from Equation 4. However, Engel⁴ used a modified form of Equation 4 and found $P = \alpha/2 \rho CV$ where α approaches unity for high impact velocities and the factor of 1/2 is a consequence of the spherical shape of the drop. It should be noted however, that studies of damage to surfaces indicate that for droplets the kinetic energy and total momentum are factors, and these vary as the cube of the diameter,

whereas the area of impact varies as the square of the diameter. Thus, the pressure in the impact zone should increase with droplet diameter.

When the liquid drops impinge against the surface of the solid, high pressures and high fluid velocities are produced. An impinging drop has two damage-producing attributes. The drop exerts a localized pressure and flows out radially around the point of impact. At the periphery of the compressed area the principal stresses are tensile and compressive. The radial tensile stress around the compressed area constitutes one of the major damage-producing attributes of an impinging drop. The radial flow velocity is much higher than the impact velocity. This radially flowing liquid exerts a shear stress on the surface of the solid over which it is running. The shear stress is able to produce a small amount of erosion even when the surface is smooth and polished. If the radially flowing liquid runs over a sizable surface protrusion, such as the raised edges of cracks that have been produced by the impact pressure, the liquid exerts forces against the protruding material and tends to break the material away. Pressure exerted against the protrusion results in a shear stress at the base of the protrusion and in a turning moment that tends to bend the protrusion over.

The forms of failure of target material due to liquid drop impingement are

1. Failure in the region of the impact due to the rapid application of a large load over an area roughly equal to the cross-sectional area of the drop. The actual area depends on the shape of the drop. In metals and in materials capable of deforming plastically at very high rates of strain, a simple surface depression is produced.
2. Failure due to the shearing action of fluid flowing across the surface at high speed. Failure of this kind is true erosion, since it usually involves removal of material from the surface.
3. Failure caused by the reflection and interference of stress waves in the solids. This is found mainly in hard, brittle solids. When it occurs it is often catastrophic and more damaging than failure in the impact area.

Equation 4 has been found to be very useful for predicting the threshold velocity for damage to a surface, but it provides no clue as to the rate at which erosion will take place as the impact velocity is varied in the vicinity of the damaged thresholds. Effort to develop such correlations has generally depended on approaches in which the energy required to tear portions of the material out of the surface has been related to the kinetic energy in the projectile impinging on the target. One of the simplest relations of

this sort, and one that has given excellent correlation of experimental data,⁵ is

$$\text{Crater volume (cm}^3\text{)} = 4 \times 10^{-9} E/B \quad (5)$$

where

E = kinetic energy of projectile, g-cm
B = Brinell hardness number of target

Thus materials with the high Brinell hardness numbers should resist the impact of the projectile better than those having low numbers. This equation assumes a ductile material; brittle material will give similar relations, but spalling will yield higher erosion rates. The erosion rate will also depend on its fatigue strength characteristics. Other factors that will influence the erosion rate are the extent to which the impact pressure exceeds the fatigue strength of the material, the yield point, or the ultimate shear strength. The erosion rate in general will be quite sensitive to the particular characteristics of the target material.

The average Brinell hardness numbers for 2-1/4 Cr-1 Mo steel, Type 321 stainless steel, and Incoloy 800 materials are 200, 180, and 160, respectively. Based on Equation (5), the small variation in Brinell hardness numbers ($\pm 11\%$) would indicate that these three materials would have about the same resistance to metal loss by the impingement of a projectile; however, previous data⁶ show a very significant difference in wastage rate between the three materials. Therefore, if erosion is a significant factor in the wastage phenomena, then factors other than the surface hardness, as pointed out above, must be influencing the erosion rate.

Many different approaches have been employed in an effort to develop insight into the erosion problems, as well as procedures for evaluating the relative damage resistance of different materials. Some experiments have involved projecting high-velocity droplets or solid particles against the test surface. Some have made use of liquid jets on the premise that there is a close relationship between the damage imposed by a jet and that imposed by a droplet. Other tests have made use of high-velocity jets of vapor containing droplets. Still others have made use of a whirling arm which is passed through either a liquid jet or a spray of droplets. Still other tests have investigated the resistance of materials to cavitation damage using cavitating venturi pump impellers. In perhaps the simplest test of all, the vibratory cavitation test (CVT), a disk of the test material is vibrated in a pool of liquid at a frequency and amplitude such that cavitation in the liquid would occur and the surface would be damaged.

In summary, review of the literature pertaining to the metal erosion indicates that at an impact velocity in the range of 600-1000 m/sec

small droplets of water can cause extensive erosion of the metal from the target surface. These results in general pertain to conditions where either a whirling arm is passed through a liquid jet or spray droplets, or high-speed droplets are projected against the test surface (stationary). The conditions of high-speed droplets being projected against the test surface describes the condition quite comparable to the postulated hypothesis for sodium-water reaction wastage mechanism. Very little information is available, however, for the erosion mechanism and its rate for the steam jet type of experiments for direct comparison with the wastage data. Most of the information available is for the conditions of the impact of a single or discrete small volume of fluid and the conditions of a whirling arm which is passed through a liquid jet or spray droplets. The results of a single impact-type test apparatus and of a supersonic nozzle directing a steam and water jet against a stationary specimen are in general agreement, although certain inconsistencies manifest themselves. One must recognize, of course, that because of differences in material preparation, heat treatment, etc., direct comparisons are not completely valid.

If erosion describes the true mechanism of the material wastage phenomenon, then tests in which water at pressures and temperatures of interest (2650 psi and 600 F) is injected toward a heated target (600 F) in the absence of sodium should produce wastage comparable to the wastage obtained in the presence of sodium. To date, no such tests have been conducted.

B. Rig-43 Tube Wastage (Task 21)

J. A. Ford
G. H. Reicks

The objective of this task is to perform scoping wastage tests to establish the significance of various system parameters on material wastage during small sodium-water reactions. The three series of tests to be conducted in FY-70 are (1) tests to extend the range of leak rates for which correspondence checks are made between Rig 10 and Rig 43 and to determine the effects of injection water pressure on material wastage rates, (2) tests to determine the effect of steam injection on material wastage rates, and (3) tests to determine the effect of water leaks in the gas space on material wastage rates. The first two series of tests have been completed and modifications to Rig 43 are currently being made in order to accomplish the remaining series of tests.

1. Design, Procurement, and Fabrication
of Steam Injection System (Job 06)

Installation of the heaters and insulation on the steam injection system were completed, and shakedown tests were conducted to check the operation of the equipment.

In the first shakedown, 2.5 lb H₂O (the calculated volume needed to obtain 2800 psia at 900 F) was added to the steam generator and the system heated without any sodium in the test vessel. When the steam pressure reached 1100 psi and the temperature was 815 F, a leak developed at a pressure fitting on a thermocouple in the steam reservoir and the heat-up was discontinued. Because of the low pressure that was obtained, the volume of the steam system was measured and found to be larger than originally specified. Revised calculations indicated 3.1 lb of water would be required to reach 2800 psia at 900 F. After the fitting was repaired, the system was then refilled with 3.1 lb of water and again heated; but the pressure only reached 2100 psi at 900 F. The low pressure this time was attributed to small leaks in the stem packing on some of the valves. New packing had been installed prior to this heat-up; however, the repacked valves were evidently still leaking. Even though the pressure was low, the guillotine was triggered. Examination of the nozzle after this test showed that the capillary apparently had jammed between the nozzle and the cutter bar and prevented the cutter bar from clearing the orifice.

Another heat-up was tried after adding 3.5 lb of water to compensate for the leaks through the packing and the pressure went up to 2475 psig at 900 F. The capillary was sheared, and the pressure was allowed to decay to 0 psig in the steam reservoir by flowing all the steam through the injection orifice. For this shakedown test, a target tube had been installed to determine if loss of metal would occur from impinging steam in the absence of sodium. Inspection of the target tube after the test revealed a buildup of gray deposits, and there appeared to be some wastage in the target area. The depth of wastage was estimated to be a few mils but because of the very small diameter of the wasted area, no depth measurement could be made. The deposits were analyzed and found to contain 60% sodium hydroxide and 30% sodium carbonate. These deposits possibly could have formed from sodium that had flowed up into the injection water riser on the reaction vessel during a prior test. The vertical riser was replaced with a new riser.

2. Testing (Job 03)

Two steam injection wastage tests were completed during this quarter. A summary of the test data is given in Table 10.2 and the wastage results are shown in Figure 10.1.

For Test No. 4a, the steam generator was filled with 4.0 lb of water and the sodium system was filled to the bottom of the reaction products line, which is 16-3/4 inches below the top flange. During the heat-up to 900 F, it was noted that sodium flow could be increased to 45 gpm when the sodium temperature approached 900 F; at 600 F, the maximum flow is about 25 gpm. The steam pressure increased to 2805 psig at 905 F but suddenly started to decrease at a slow rate before the isolation valve was

TABLE 10.2 - SUMMARY OF RIG-43 STEAM INJECTION TESTS

	<u>Test Number</u>	
	<u>4a*</u>	<u>5a</u>
<u>Sodium System</u>		
Flow rate, gpm	25	25
Velocity past target tube, ft/sec	~1	~1
Estimated bulk Na temperature, F	900	900
<u>Steam Injection System</u>		
Pressure, psig $\pm$ 50		
Initial	2725	2150
Final	2650	2125
Change	75	25
Temperature, F	900	900
Orifice size, mils	8.20	8.07
Injection duration, sec	157	180
Average injection rate, lb/sec	0.0009	0.0008
<u>Target Tube</u>		
Material	$\leftarrow$ 2-1/4 Cr - 1 Mo $\rightarrow$	
Spacing, in.	1/4	1/4
<u>Wastage</u>		
Target tube		
Depth of penetration, mils	-	Complete (130)**
Wastage rate, mils/sec	-	>0.7
Maximum temperature, F	~900	1230
Guillotine (Type 304 stainless steel)		
Depth of penetration, mils	81	-
Wastage rate, mils/sec	0.52	-

* Guillotine bar jammed in front of capillary, spacing 5 to 10 mils; therefore, no water was injected on target tube.

** Inside surface opposite penetration was wasted a depth of 40 mils

opened to allow steam to be in contact with the orifice. Apparently, steam was leaking through the valve packing. In 20 minutes the pressure decreased to 2750 psi; the test was initiated when the pressure reached 2725 psi at 900 F. When the test was triggered, there was no rise in temperatures on the oscillograph. The guillotine was retriggered after the elapse of 9 seconds and again at 20 seconds without producing a temperature rise. The test was allowed to continue for 2 minutes and 37 seconds, when the isolation valve was closed and the steam system was vented off. At the end of the injection period the steam pressure was 2650 psi. This pressure decrease, corrected for the leakage rate, indicates that only 0.11 lb of steam was injected into the sodium. This gives an injection rate of approximately 0.0007 lb/sec. When the target tube assembly was removed from the test pot, there was a large build-up of reaction products around the target assembly and the gap between the nozzle and target was completely bridged over with reaction products. It was also noted that the cutter bar had not descended far enough to clear the injection orifice. After steam-cleaning, it was observed that a nut welded onto the cutter bar actuating rod had hung up on the target tube support bracket. Also, it was noted that there was considerable wastage on the cutter bar and nozzle. Both the cutter bar and nozzle are stainless steel and the measured depth of wastage was in excess of 81 mils on the cutter bar and 25 mils on the nozzle. The clearance of the injection spacing between the cutter bar and nozzle was only from 5-10 mils. The estimated wastage rate on the cutter bar was approximately 0.5 mil/sec. This is of significance in an actual steam generator, since it indicates that a leak at a close-fitting tube support could cause damage to the support itself.

In Test No. 5a difficulty was again encountered with leaking of steam through the stem packing of several valves. In spite of the leaks, however, the test was made because the leaks were quite small. In this test, complete penetration of the target tube (130-mil wall) occurred and approximately 40 mils of wastage occurred on the inside surface of the tube opposite the wasted hole. It is not known at what time during the injection period the tube was penetrated; therefore, it is not possible to determine the true wastage rate. The wastage rate is, however, in excess of 0.7 mil/sec.

Comparison of this wastage data with that reported previously for the Rig 10 wastage tests shows that these wastage rates are higher by a factor of approximately 3 than those obtained at similar leak rates with injection of water at 600 F. Review of the UKAEA data on the effect of temperature on wastage rate shows that the metal loss increases by a factor of approximately 150 in going from 600 to 900 F in temperature.⁷ Thus, the effect of temperature in these tests does not seem to be nearly as significant as would be indicated by the UKAEA data. The Rig 43 data does suggest, however, that increasing the temperature of the injection fluid does significantly increase the wastage rate. A pertinent observation on

the wastage rates observed in these two tests is that they are lower than the highest measured wastage rate observed with the water injections at 600 F — approximately 3.4 mils/sec. The data does indicate the need for more wastage data at higher leak rates with 900 F injections to determine if higher wastage rates than those from water injection are obtained. No additional steam injection tests are scheduled under this program. It is highly recommended that more steam injection tests be conducted over the total leak rate range of interest, to establish the wastage curve at temperatures up to 1000 F so that a comparison can be made with the water injections at 600 F to determine if more severe wastage can occur under steam conditions.

Two methods were available to estimate the injection rate. One is a theoretical calculation based on the capillary diameter and the other is based on a mass balance for water in the steam generator; however, the current installation has a 0-5000 psi gauge on the steam generator and reading the gauge at some distance from it is very difficult (operating procedures require personnel to be out of the area of the reaction vessel when conducting a test). Therefore, it is very difficult to obtain the true pressure change in the unit. It should be noted that a pressure transducer for installation in the steam generator with the readout to be on the optical visicorder had been ordered but had not yet been received by the time these two tests were conducted. Because of the short time available to conduct the tests, it was elected to proceed with testing without having the pressure transducer. A comparison was made between the calculated flow rate based on the capillary diameter with that calculated from the pressure drop. For Test No. 4a, the calculated leak rates were within 50 per cent of each other. For Test No. 5a, the calculated leak rates varied by a factor of 500%. In Test No. 4a, the difference between the calculated leak rates was equivalent to a final pressure difference of only 15 psia. In Test No. 5a, the difference between the calculated leak rates was equivalent to a final pressure difference of 125 psia. The readings of the pressure gauge are probably accurate only to within ± 50 to 100 psig; therefore, it was assumed that the leak rate based on the capillary diameter is the better value to use.

3. Water Injection into Gas Space Design (Job 11)

The detailed design of the modifications to Rig 43 needed to conduct the gas space leaks was completed, all of the mechanical components were received, and the two viewing ports were installed on the reaction vessel. The installation was completed and checkout of the system was initiated.

C. Leak Detection (Task 22)

The objective of this task is to determine the feasibility of various leak detection systems for detecting small sodium-water reactions in sodium-heated steam generators.

1. Transducer Mounting Techniques (Job 18)

C. C. Scott

S. P. Ying

The scope of the acoustic detection program is to establish the feasibility of this technique for detecting small leaks. Preliminary acoustic data have shown that a detectable noise level above the background noise existed during the injection of the water into sodium. However, essentially only sonic noise was identified with the transducer mounting techniques used. The scope of this job is to develop the proper mounting technique for the transducer so that frequencies greater than 20 kHz can be detected.

The work plan to carry out this part of the program was completed. Several bench-scale tests of adding chips of sodium to water were made to obtain acoustic spectra to assist in designing a suitable transducer mounting.

2. Functional Requirements
for Side Loop (Job 30)

J. A. Ford

J. H. Coleman

Several types of in-sodium leak detection devices are to be installed in a side loop on Rig 10 to (1) evaluate candidate leak detection methods and (2) obtain reaction product disposition information.

The scope of this job is to (1) prepare the functional requirements for this side loop, (2) perform a conceptual design of the side loop, (3) perform a conceptual design of tube bundle modifications, and (4) prepare a preliminary cost estimate to do the detailed design. Work was initiated on preparing the functional requirements for the side loop and the tube bundle modifications.

D. Material Wastage Topical
Report (Task 70)

H. V. Chamberlain

The objective of this task is to prepare a report covering the material wastage studies conducted by APDA. Included will be Rig-10 and Rig-43 material wastage tests, caustic corrosion studies, hot erosion studies, and jet characteristics.

1. Research and Planning (Job 01)

Review of the detailed outline of the report was completed and preparation of the report was initiated.

E. Leak Detection Topical Reports (Task 71)

The objective of this task is to prepare two reports covering the leak detection studies conducted by APDA. One report is to cover the leak

detection criteria and Rig 10 experience with impurity monitoring and the other is to cover the acoustic detection studies.

1. Preliminary Draft - Acoustic
Leak Detection (Job 02)

S. Barkhoudarian
C. C. Scott

Incorporation of internal comments is continuing.

2. Preliminary Draft - Leak
Detection Criteria (Job 03)

J. A. Ford

No activity pending receipt of the Commission's comment on draft of report.

- F. Project Summary Report (Task 72)

H. V. Chamberlain

The scope of this report is to summarize the significant results and conclusions developed under Project Agreement No. 10. This report will cover the studies on material wastage, leak detection, and gas-water modeling.

1. Planning (Job 01)

No activity pending receipt of comments on the detailed outline of the report from AEC-COO.

III. NEXT REPORT PERIOD ACTIVITIES

- A. Rig-10 Tube Wastage (Task 20)

No further work is to be done under this contract. Further testing in Rig 10 is to be performed under a subcontract to the Liquid Metal Engineering Center.

- B. Rig 43 Tube Wastage (Task 21)

Scoping tests with gas-space leaks will be completed.

- C. Material Wastage Topical Report (Task 70)

The report will be issued.

- D. Leak Detection Topical Reports (Task 71)

The two topical reports, one covering leak detection criteria and one covering acoustic detection of leaks will be issued.

- E. Project Summary Report (Task 72)

The report summarizing the activities of this project will be issued.

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FIG. 10.1 EFFECT OF LEAK RATE AND SPACING ON WASTAGE RATE OF TARGET TUBES

FIG. 10.2 ELEVATION CROSS SECTION OF TUBE BUNDLE ASSEMBLY

FIG. 10.3 CORRELATION OF VELOCITY-MODIFIED WASTAGE WITH LEAK RATE

FIG. 10.4 CORRELATION OF VELOCITY-MODIFIED WASTAGE WITH ORIFICE DIAMETER

SODIUM TECHNOLOGY PROJECT

PROJECT AGREEMENT NO. 11

PROJECT ENGINEER: J. E. MEYERS

I. PREFACE

A. Scope

The present scope of the sodium technology project encompasses the following elements:

1. Impurity monitoring devices, including the development and evaluation of the Rhometer, oxygen meter, hydrogen meter, and plugging meter.
2. Sodium purification processes, including the evaluation and/or development of cold trapping, hot trapping, centrifuging, movable bed gettering, and thermal decomposition.
3. Sodium sampling and analysis, including the reliability of through-flow samplers, oxygen analysis, and hydrogen analysis.
4. Physical and chemical behavior of sodium and impurities, including the investigation of the solubility of hydrogen and sodium carbonate in sodium, the equilibrium distribution of hydrogen between sodium and the cover gas, and interaction of hydrogen and oxygen in sodium.

B. Relationship to Other Projects

This project is oriented entirely to the Commission's research and development program on the LMFBR.

II. TECHNICAL PROGRESS

A. Impurity Monitoring Instruments

1. Rhometer (Task 20)

W. D. Huston
C. C. Scott

The objective of this program is to develop and test an automatic temperature compensator and resistivity meter system capable of operating

in the range of 400 F to 1000 F with sufficient sensitivity and stability to detect a change in sodium resistivity equivalent to 1 ppm of oxygen.

a. Topical Report (Job 07)

The objective of this job is the publication of a topical report on the development and testing of the Rhometer resistivity meter with automatic temperature compensation through FY-1970.

The final draft of this report is being prepared for submittal to the AEC for review and comment.

2. Oxygen Meter (Task 27)

J. E. Meyers

a. Topical Report (Job 06)

The objective of this job is the publication of a topical report on the oxygen meter evaluation program performed at APDA incorporating all of the results obtained through FY-70.

This report has been published and distributed as APDA-245.

3. Hydrogen Detector (Task 21)

J. H. Coleman
J. A. Ford

a. Topical Report (Job 10)

The objective of this job is the publication of a topical report covering the installation of the APDA hydrogen detector and its operation through FY-70.

The first draft of the report has been circulated internally for review and comment.

B. Physical and Chemical Behavior of Sodium Impurities

1. Hydrogen Solubility (Task 31)

A. A. Gordus*
E. F. Hill
S. A. Meacham

a. Topical Report (Job 06)

The first draft of this report has been reviewed internally and comments are being incorporated prior to submittal to the AEC for review and comment.

* University of Michigan

C. Sodium Sampling and Analysis

M. K. Deora
A. J. Friedland

1. Sampling Reliability (Task 70)

a. Topical Report - Oxygen (Job 03)

This report is being prepared for publication

2. ASTM Round Robin on Oxygen
in Sodium (Task 73)

E. F. Hill
S. A. Meacham

a. Reduction of Data and Report (Job 05)

This report has been published and distributed as APDA-238.

D. Project Summary Report

A draft of this report is being prepared for review and comment by the AEC.

Effort will continue on the preparation of the Rhometer, hydrogen detector, hydrogen solubility reports, and project summary report.

COMPARISON OF METHODS FOR HETEROGENEITY ANALYSIS

PROJECT AGREEMENT NO. 19

PROJECT ENGINEER: J. B. Nims

PRINCIPAL INVESTIGATOR: E. M. Page

I. PREFACE

A. Scope

The objective of this program is to provide a consistent comparison and evaluation among various methods of computing heterogeneity effects in fast reactor critical assemblies and power reactors, particularly with regard to its effect on sodium void analysis. In the case of plate-type criticals, specific voiding and heterogeneity experiments will be analyzed by various methods. In the case of the power reactor, a conceptual design will be chosen for analysis.

Four methods are to be applied in the heterogeneity analysis, as follows:

1. One-dimensional transport theory calculations by DTF-IV of the flux distribution for the plate or pin arrangement forming a symmetric cell followed by flux-weighting the cross sections in each region of the cell to define cell-homogenized cross sections. The homogenized cross sections are then used in diffusion theory calculations to incorporate the heterogeneity effects directly in the calculations.¹
2. The same as Method 1 above, but using the cell-homogenized cross sections in perturbation theory to calculate heterogeneity corrections to a homogeneous calculation.²
3. The same as Methods 1 and 2 above, but using bilinear averaging to define the homogenized cross sections.³
4. Use of collision probability-perturbation methods to obtain correction terms to homogeneous diffusion theory calculations.^{4, 5}

In addition, attempts will be made to furnish Oak Ridge National Laboratory with consistent data to allow Monte Carlo calculations of selected cases, thus furnishing a more nearly exact approach that can serve as a reference calculation.

B. Relationship to Other Projects

A consistent analysis of heterogeneity effects in fast critical assemblies and power reactors is of general interest to the LMFBR industry and complements the current ANL efforts in this area.

II. TECHNICAL PROGRESS

A. Resonance Self-Shielding

As discussed in an earlier report,⁶ the heterogeneity effect in fast reactors is comprised of two components: The first component results from the differences in heavy-element capture and fission resonant cross section energy self-shielding which occur due to system heterogeneity; the second component is the so called "fast effect" which results from perturbations in the spatial neutron flux distributions which occur at the locations of heterogeneities. In general, both components are more pronounced in the highly heterogeneous plate-type fast critical assemblies than in an actual fast reactor.

Clearly, it would be most convenient for heterogeneity investigations to first define a calculational approach that does not include either of the two heterogeneity effects so as to provide a reference against which the various analytical techniques that do account for heterogeneity can be compared. The most obvious reference case would be a completely homogeneous treatment of the system in question; however, as explained in Reference 6, due to the nature of the calculational techniques employed it becomes best suited for the purposes of the major portion of this study to define as a reference case a model in which the heterogeneity effects of cross section energy self-shielding of heavy element resonances have already been included and to investigate various techniques of calculating the effects of spatial flux distributions in a heterogeneous system. Therefore, the major portion of this study (Methods 1-4 in Section I. A.) and also the Monte Carlo calculations by ORNL will use for any given configuration, as a starting point, the same self-shielded cross section sets. The model employed to obtain such self-shielded cross sections is described in Section II. A. 1.

It is recognized that considerable uncertainty exists in the standard self-shielding techniques employed here; moreover, for the drawer configurations of interest in the critical assemblies studied here (particularly ZPR-VI Assembly 6), the effects of heavy-element resonance self-shielding are quite important and variations in their treatment could have large effects on the predicted behavior of the assemblies. Thus, it is planned to perform a limited study of the sensitivity of self-shielding techniques. Reference calculations will also be performed using a completely homogeneous approach (as outlined in Methods 1-4 above but with a homogeneous treatment of heavy-element self-shielding) to obtain a base to measure the heterogeneity effect of the plate array on energy self-shielding and to afford a basis for a consistent comparison with others who essentially start with a pure homogeneous model.

The methods being used to calculate the effects of the spatial flux distribution perturbations in heterogeneous systems have already been discussed in Reference 6. However, only brief mention has been previously made of the energy self-shielding calculations. A more detailed explanation of the self-shielding techniques being employed is given below.

1. Reference Techniques for Obtaining Resonance Self-Shielded Cross Sections

As outlined above, the reference model in this study for the Methods 1-4 and Monte Carlo heterogeneity analyses employs a consistent multigroup set of configuration-dependent, energy-self-shielded cross sections; i. e., a different set exists for each system being studied, but each set is obtained in the same manner. The method used for obtaining the cross section sets can be briefly outlined as follows: First, all multigroup cross sections, with the exception of those in the resonance regions of the heavy elements, are obtained through spectrum-weighting techniques consistent with a homogeneous composition such as that which would result from an MC² run for the system using the P1 mode and homogeneous self-shielding. Heavy element capture and fission resonance self-shielded cross section calculations are then performed with the IDIOT code in the resolved and unresolved resonance regions. The resonance parameters used are consistent with the data used in the MC² homogeneous spectrum and cross section calculations. These self-shielding calculations are done separately for each heavy-element plate in the drawer loading of the system utilizing an equivalence principle based on explicitly formulated Dancoff factors for asymmetrical plate arrays. This aspect of the calculation is treated in more detail below in Section II. A. 2. b.

2. IDIOT Model

A brief outline of the method used in the IDIOT code for heavy-element resonance cross section energy self-shielding calculations is described below.

The code first calculates the energy-dependent flux variation through a given resonance k of a particular isotope, as follows* (narrow resonance approximation shown):

$$\phi_k(E) = \frac{\sigma_n}{\sigma_n + \sigma_{sr}(E) + \sigma_{ar}(E)} \quad (1)$$

where,

σ_n = the total nonresonant scattering cross section per resonant absorber atom (Σ_n/N_r), which is assumed constant through a group

* Outside the resonance σ_{sr} and $\sigma_{ar} = 0$, and flat flux is assumed for $\phi(E)$; i. e., $\phi(E) = 1.0$.

$\sigma_{sr}(E)$ = the microscopic resonant isotopic scattering cross section as a function of energy

$\sigma_{ar}(E)$ = the microscopic resonant isotopic absorption (capture and fission) cross section as a function of energy

The values of $\sigma_{sr}(E)$ and $\sigma_{ar}(E)$ are calculated from the unresolved and resolved resonance parameter data furnished as library input to the code.

The resonant cross sections $\sigma_{sr}(E)$ and $\sigma_{ar}(E)$ are described by the single-level, Doppler-broadened Breit Wigner model, assuming an energy independent Γ_n . The resulting effective shielded cross section over a group width* ΔE due to the presence of N resonances in the group is given by

$$\sigma_x = \frac{\sum_{k=1}^N \int_{\Delta E} \sigma_{xk}(E) \phi_k(E) dE}{\int_{\Delta E} \phi(E) dE} \quad (2)$$

where,

$\sigma_{xk}(E)$ = energy-dependent cross section for reaction-type x of the k^{th} resonance

$\phi(E)$ = total energy-dependent group flux = ΔE

When the resonances are unresolved, the sum over resonances in Equation (2) is replaced by an average for the group. Provisions are allowed in the unresolved range for the case of isolated resonances where chi-squared distributions are used to develop the required averages and provisions for the case of strongly overlapping resonances are allowed using the methods of Nicholson.^{7, 8} For the isolated resonance approximation, it is also possible to provide an estimate of the effects of overlapping resonances of the same sequence or same value of total angular momentum.

Calculations in the resolved resonance region allow treatment by either narrow resonance approximation (NRA), the most common situation, or the intermediate resonance approximation (IRA).^{9, 10} The latter option allows for the average neutron energy losses associated with both heavy- and light-element scattering in the absorber plate to essentially carry the neutron part of the way through the resonance. In principle, the broader resonances sometimes found at the lower energies require such treatment.

* The group structure in IDIOT is typically of finer structure than the final multigroup structure used for subsequent diffusion and transport theory analysis. The finer group cross sections compiled in IDIOT are flux-weighted using MC² homogeneous spectra to obtain the desired broad group cross sections.

a. Intermediate Resonance Approximation

It was necessary to pay special attention to the application of the IRA as utilized by IDIOT in the lower energy groups. IDIOT makes the approximation that all resonances centered within an energy group lie completely within the group, an assumption that simplifies the numerical integration and is valid for the energy range of greatest importance in fast reactor core analysis. However, the reference multigroup energy structure utilized in the current study extends to 0.683 ev and encompasses some resonances that are quite broad relative to the group width used by IDIOT. Application of the IRA in such cases worsens the error created by the above approximation and in some instances even produces negative cross sections. While the energy region in question (<100 ev) is of little importance to the core neutronics, it is desirable to utilize reasonable cross sections in these groups. Thus, the NRA is sometimes applied to the broader resonances at the very low energies. It should be pointed out that the validity of the IDIOT formulation is highly dependent on the nonresonant cross section for the particular configuration so that the more appropriate model (NRA or IRA) for a given energy group can vary among configurations. The standard treatment by IDIOT applies the IRA to the last three fine groups (0.683 - 78.9 ev). Due to the considerations discussed above, this allocation of the IRA was modified for ZPR-III Assembly 48 by extending the NRA to IDIOT fine group 26 for U-238, Pu-239, and Pu-240 for the homogeneous case. In addition, an IRA option was used that applies only to heavy-element scattering for Pu-240 in IDIOT fine-group 28. For ZPR-VI Assembly 6, the IRA was omitted entirely for all except the normal drawer calculations.

b. Treatment of Heterogeneous Systems

The effect of a heterogeneous absorber configuration on the resonance escape probability of neutrons is taken into account through an equivalence principle where σ_n is adjusted to provide an equivalent nonresonant cross section that may be utilized in the normal homogeneous formulation. (see Equation (1)).

The value of σ_n used in Equation (1) for a fuel lump containing resonant material surrounded by moderator is found from the following expression:

$$\sigma_n = \frac{\Sigma_p + \Sigma_m + \Sigma_h}{N_f} \quad (3)$$

where:

Σ_p = the nonresonant cross section due to heavy-element scattering in the fuel lump

Σ_m = the nonresonant cross section due to light (moderator) element scattering in the fuel lump

Σ_h = the effective nonresonant cross section for the plate due to neutron leakage from the fuel lump in a heterogeneous system

N_r = number density of the resonance material in the fuel lump

(Σ_p and Σ_m are appropriately modified if the IRA is used in the resolved resonance calculations.)

To calculate Σ_h , the geometry of the fuel lump itself and the surrounding media must be taken into account. The discussion below and corresponding option in IDIOT will now deal specifically with the plate geometry as found in ZPR-III and ZPR-VI. For the calculation of a resonant material in a given plate, Σ_h is given by

$$\Sigma_h = \frac{(1-C)}{\bar{\ell}} \frac{A}{1 + (A-1)C} \quad (4)$$

where:

$\bar{\ell}$ = mean cord length of the fuel plate ($\sim 2 \times$ plate width)

A = correction factor to the Wigner rational resonance escape approximation for the plate (~ 1.25)

C = the Dancoff factor, the term which accounts for the effect of resonance capture in the same material in surrounding plates. For a symmetrical infinite plate array,

$$C = 2E_3(\Sigma_s v \bar{\ell}/2) \quad (5)$$

where

E_3 = exponential integral of the third order

v = ratio of the moderator volume to fuel plate volume

$\bar{\ell}$ = mean chord length of the fuel plate

Σ_s = total nonresonant cross section of moderator region

A simple extension can be made to an asymmetric case where a repetitive but different plate array is assumed on each side of the plate in question.* For this case

* The fuel plate geometry is assumed the same throughout the lattice, but the plate spacing and moderator cross section is allowed to be different on the left and right of the plate to be calculated

$$C = E_3 \left(\Sigma_s^{\ell} v \frac{\bar{\ell}}{2} \right) + E_3 \left(\Sigma_s^r v \frac{\bar{\ell}}{2} \right) \quad (6)$$

where:

Σ_s and Σ_s^r = the total macroscopic cross section of the homogenized moderator region on the left and right, respectively, of the fuel plate to be calculated

v_{ℓ} and v_r = ratio of the left and right moderator volume, respectively, to fuel plate volume

$\bar{\ell}$ = mean cord length of the fuel plate

It is this formulation of the Dancoff factor that will be applied to the various drawer configurations of interest in ZPR-III and ZPR-VI. However, it is evident that even within the context of the equivalence principle, the configurations of Assembly 48 of ZPR-III and of ZPR-VI do not exactly lend themselves to the Dancoff formulation expressed in Equation (6). For most cases, when a given fuel plate containing a resonance absorber is considered for calculation, there is no truly repetitive array of the same plate to the right or left within a drawer (see Figures 1 and 2 for the plate geometries encountered in ZPR-III Assembly 48 and ZPR-VI Assembly 6). For such cases, one of two approximations is utilized: First, the assumption can be made that a repetitive lattice actually does exist (to the left or right) and the repetition begins at the next plate containing significant amounts of the same resonance material; second, plates containing the same resonance material and physically located close to one another may be grouped to form an equivalent plate that is repetitive (or more nearly repetitive). By incorporating these assumptions, a reference Dancoff model was chosen for each heavy resonance isotope in each plate for the resonance self-shielding calculations for the drawer configurations shown in Figures 1 and 2. The model used is designated in the figures by using a labeled arrow to traverse the region to the left and right of a given plate assumed to be the moderator for the calculation of the specified isotope. Immediately adjacent to the point of the arrow is the position of the next fuel plate constituting the beginning of the assumed repetitive pattern. For example as shown in Figure 2a, the self-shielding calculation for U-238 in the depleted uranium metal plate of the normal drawer of ZPR-VI, Assembly 6 utilizes a Dancoff factor that assumes a nonresonant moderator region on the left that extends to the left-hand 1/4-inch U_3O_8 plate which is assumed identical to the metal plate to form a repetitive pattern. Similarly, on the right the moderator region is shown extending to the other 1/4 inch U_3O_8 plate, thus defining a different assumed repetitive pattern. Brackets shown in Figures 1a, 1b and 2c indicate that the enclosed plates are bunched to form a single equivalent plate employing a simple volume-weighting of the number densities of the plate constituents. The choice of the reference models defined in Figures 1 and 2 relied largely on judgement, taking note of the relative size, position, and density of the plates involved. The models

most open to question are the bunched drawer configuration of ZPR-III Assembly 48 (see Figure 1(b)) and the bunched drawer configuration of ZPR-VI Assembly 6 (see Figure 2(b)). In the first case, the shielding for the U-238 present in the Pu-U alloy and depleted U metal plates, and the Pu-239 and Pu-240 present in the Pu-U and Pu metal plates were both treated using the grouped equivalent plate concept. However, the actual uranium and plutonium concentrations in the individual grouped plates are different by factors of 2 and 5, respectively, which probably results in a situation where the plates having the smaller concentrations are calculated somewhat overshielded and the plates having the higher concentration are calculated somewhat undershielded. In the case of ZPR-V, each plate in the bunched drawer configuration was treated separately in the U-238 shielding calculations because of the widely varying differences in concentrations between plates (as much as 16 to 1). This approach can be seen implicitly to be inaccurate, since the contribution to the Dancoff factor of an immediately adjacent plate on one side is fixed at a value of 0.5 regardless of its density or thickness. Numerous nonexact situations also exist similar to that shown in Figure 2(a) (an extreme case) for the model used for the U-238 shielding calculations in the depleted uranium metal plate (Case 3). In this case it is obvious, if one considers the adjacent drawers, that the assumed left and right moderator regions are not actually portions of a repeating lattice, since the next U_3O_8 plate appears just beyond the matrix and drawer walls. However, the effects of most of these approximations are believed to be small. A simple sensitivity study is planned from which a determination can be made of the effects of using optional Dancoff model definitions for these cases.

B. Bilinear Averaging

A review of the fundamental concepts for using a bilinear weighting scheme involving the real and adjoint flux distributions to obtain spatially averaged cell cross sections (see I. A. 3) is almost complete. The approach chosen is essentially that of Nicholson³ and involves the use of variational techniques.

The DTF-IV code¹¹ (a FORTRAN-IV, a program for solving the one-dimensional multigroup transport equation with anisotropic scattering) will be modified in order to incorporate the bilinear scheme for obtaining homogenized core sections.

C. Collision Probability - Perturbation Methods

It has been decided to use the CALHET code rather than CELPERT for the collision probability perturbation method of calculating heterogeneity (Method 4 in Section I. A). This choice was made due largely to its more definitive documentation.

D. Monte Carlo Studies

Details of the normal and standard drawer configurations for ZPR-III Assembly 48 (see Figure 1) have been sent to Oak Ridge National Laboratory as a basis for reference calculations to be made by Monte Carlo techniques. A modified version of the ESP code is to be used for the Monte Carlo calculations.

III. NEXT REPORT PERIOD

During the fourth quarter of FY 1970, most of the analysis of ZPR-IV Assembly 6 by real flux-weighting (Method 1) should be completed. The CALHET code for use with Method 4 will be made operational. Review of the bilinear approach will be completed, and the appropriate modifications of the DTF-IV code will be initiated.

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FIG. 19.1 SPECIFICATION OF MODERATOR REGIONS FOR SELF-SHIELDING CALCULATIONS BY IDIOT CODE OF DRAWER CONFIGURATIONS OF ZPR-III ASSEMBLY 48

FIG. 19.2 SPECIFICATION OF MODERATOR REGIONS FOR SELF-SHIELDING CALCULATIONS BY IDIOT CODE OF DRAWER CONFIGURATIONS OF ZPR-VI ASSEMBLY 6

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