A CONCEPTUAL DESIGN OF A URANYL NITRATE FUELED REACTOR
FOR THE DESTRUCTIVE TESTING OF
LIQUID METAL FAST BREEDER REACTOR FUEL SUBASSEMBLIES

by

Steven Earl Ramsower

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SIGNED: Steve Ramsower

APPROVAL BY THESIS DIRECTOR

This thesis has been approved on the date shown below:

Robert L. Seale
R. L. SEALE
Professor of Nuclear Engineering

30 April 1979
Date

To Teri, Eric

and

Valerie

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ABSTRACT

A preliminary design of a uranyl nitrate test reactor is developed, with emphasis placed on the core neutronics and cross section development. ENDF/B-IV cross section data and the AMPX system were used to develop a 25 group neutron cross section library. A series of one-dimensional transport calculations were made in order to arrive at a reference design. Power densities of 16.5 Kw/l appear to be attainable in the 217 pin FFTF test subassembly, with a peak neutron flux in the test zone of 2.4×10^{14} n/cm²-sec.

Other engineering features pertinent to the overall system design are discussed, including: (1) corrosion, (2) treatment of radiolytic gas, (3) heat removal, and (4) reactor control.

CHAPTER 1

INTRODUCTIONS

1.1 Background

The series of Homogeneous Reactor Experiments (particularly HRE-1&2) performed at Oak Ridge National Laboratories (ORNL) during the 1950's, as well as the Water Boiler experiments at LASL during the 1940's, demonstrated the utility of aqueous solutions as a reactor fuel. These experiments indicated that high temperature, high power density operation (as required for efficient power production) of solution fuels was impractical due to severe corrosion problems. However, high solution temperatures are not required for a test reactor. This fact suggests a solution-fueled reactor as an attractive alternative to conventional test reactors for safety experiments. Solution-fueled reactors offer a great deal of flexibility, as the reactor's performance can be readily adjusted to fit the needs of the experiment. Additionally, the driver solution is pumped away after the experiment, reducing the residual radioactivity that must be dealt with in experiment removal and disassembly.

The experimental facility which was evaluated is a uranyl nitrate reactor for destructive testing of fast breeder reactor fuel. Such a facility could perform experiments pertaining to fuel pin failure and initial cladding motion, as well as fuel-steel motion and slumping in advanced accident scenarios. These experiments would also

serve as useful benchmarks for fast reactor safety codes, such as SIMMER-II, SAS, etc.

Major emphasis in this study was placed on the core neutronics. Cross sections were developed from ENDF/B-IV specifically for the reference core design. A series of transport calculations were made to determine accurately the relative power density that can be achieved in the test region. These calculations indicated that the power density in the test subassembly is 55% of the power density in the solution driver (for the reference design). The peak power density achievable in the solution is still uncertain, although it probably lies in the range of 30-100 Kw/l.

Preliminary design of the heat and radiolytic gas removal systems are discussed. Materials for the reactor core and heat removal system were selected, and approximate corrosion rates suggested. Finally, potential control systems for normal and emergency operation are developed.

1.2 Executive Summary

A reference experiment was selected for evaluation of the test facility to aid in determining the system design and performance characteristics. A 217 pin, 93% enriched UO₂ hexagonal subassembly made of FFTF type fast reactor fuel was chosen as the test specimen. The subassembly is 0.914 meters in active length and is contained in a stainless steel canister filled with non-circulating sodium. A buffer zone consisting of four rows of 30% enriched fast reactor fuel pins surrounds the test subassembly in order to reduce power peaking in the

central test assembly. Power peaking is further reduced by a 0.3 cm thick natural boron spectral filter located between the buffer zone and the driver solution. Figure 1.1 illustrates the experiment and driver reactor configuration. A summary of performance characteristics for the reference experiment is listed in Table 1.1. The experiment driver consists of an aqueous solution of 93% enriched uranyl nitrate. The solution density is 1.46 g/cm^3 , with a dissolved uranium concentration of 334 g/l. The uranyl nitrate solution operates at a power density of 30 Kw/l, with a total active volume of 53.6 liters. A 7 cm BeO reflector surrounds the core to reduce neutron leakage from the driver. Heat is removed from the solution by circulation through a heat exchanger at 14.9 l/sec. via a 10 cm I.D. safe geometry pipe. A 25°C ΔT is maintained across the core, with the core outlet temperature at 150°C . The entire system is pressurized to approximately 10 atmospheres. Radiolytic gas is removed at the rate of 7.6 l/sec. by Cu(II) induced catalytic recombination and a gas removal system. A summary of the characteristics of the uranyl nitrate test reactor are included as Table 1.2.

Fig. 1.1 Reference Core Configuration

Table 1.1 Performance Characteristics of Reference Experiment

test specimen	full 217 pin subassembly
power density	16.5 kw/l
specific power	4.12 kw/kg
peak flux	2.4×10^{14} n/cm ² -sec
peak/average flux	1.084
test enrichment	93%
total power in test	0.144 Mw
begin to melt (1120 J/gm)	114.4 sec
end of melt (1400 J/gm)	143.0 sec
start vaporization (1700 J/gm)	173.6 sec
end vaporization (3600 J/gm)	377.6 sec

Table 1.2 Summary of Uranyl-Nitrate Test Reactor Characteristics

power density	30 kw/l
average flux	4.7×10^{13} n/cm ² -sec
active solution volume	53.6 liters
ΔT across core	25°C
core outlet temperature	150°C
solution flow rate	14.7 l/sec
pressure	10 atmospheres
gas production	7.59 l/sec
solution density	1.46 g/cm ³
uranium concentration	334 g/l
uranium enrichment	93%
reflector	7 cm BeO
core tank	0.635 cm ZIRC-II

CHAPTER 2

CROSS SECTION DEVELOPMENT

The function of a cross section processing system is to process nuclear data into a form suitable for reactor design calculations. The data base utilized for these calculations was the Evaluated Nuclear Data File (ENDF/B-IV), which was processed into a fine group cross section library by the AMPX system developed at ORNL (Greene et al., 1976). Further processing and refinement of this fine group library introduced: (1) resonance self-shielding effects, (2) rod shadowing of the resonance neutron flux, and (3) problem-dependent weighting spectra. In summary, a fine group problem-independent library was processed into a broad group cross section library for subsequent transport calculations.

2.1 A Brief Description of ENDF

The Evaluated Nuclear Data File (ENDF) is a collection of data for all neutron induced reactions over a wide range of incident neutron energies, arranged in a standard format for easy retrieval (Drake, 1970). Data is arranged on magnetic tape into seven files, the contents of which are discussed in the following subsections.

2.1.1 General Information

File 1 provides a brief documentation of how the data for a particular material was evaluated. Also contained is a dictionary of data files and the cross section types that are available. Basic

nuclear data is also found in File 1, such as delayed neutron data (for fissile nuclei), fission product yield data, number of neutrons per fission, decay chains, etc.

2.1.2 Resonance Parameters

File 2 contains resolved and unresolved resonance parameters. Several representations are permissible for specifying the resolved resonance parameters. They include: (1) single-level Breit-Wigner, (2) multi-level Breit-Wigner, (3) Reich-Moor multi-level formalism, and (4) Adler-Adler multi-level formalism. In the unresolved energy region, only single-level Breit-Wigner resonance parameters are allowed.

2.1.3 Neutron Cross Sections

One-dimensional neutron cross sections, such as elastic scattering, radiative capture, and total cross sections are contained in File 3. These cross sections are tabulated as a function of the incident neutron energy. Interpolation schemes are provided to specify energy variation between tabulated values.

2.1.4 Angular Distribution of Secondary Neutrons

Angular distributions of secondary neutrons from scattering collisions are given in File 4. Both discrete level and continuum data are generally provided. The angular distributions are normally represented by energy-dependent Legendre polynomial expansion coefficients, although a normalized probability distribution function is allowed.

2.1.5 Energy Distributions of Secondary Neutrons

File 5 contains energy distribution data for secondary neutrons. Data is included in this file for all reactions that produce secondary neutrons. The energy distributions are normalized so that the integral of the distribution function over all possible secondary neutron energies is unity. These energy distributions can be represented by several allowed distribution laws, including: (1) discrete level, (2) evaporation spectrum, (3) fission spectrum, and (4) the Watt spectrum.

2.1.6 Energy-Angular Distributions of Secondary Neutrons

Two-dimensional (energy-angle) data for secondary neutrons is located in File 6. This file is used when an accurate representation is not possible using File 4 or File 5 data. The file is organized into subsections containing pertinent information for each reaction type. The distribution function is normally expressed as a Legendre expansion in both neutron energy and angle.

2.1.7 Thermal Scattering Law Data

Inelastic scattering data in the thermal energy range for moderating materials is given in File 7. The thermal neutron scattering law, $S(\alpha, \beta, T)$ is used to represent thermal inelastic scattering, where α is the momentum transfer, β is the energy transfer, and T is the moderator temperature.

2.2 The AMPX System

The AMPX system is a collection of interrelated computer codes (modules developed at ORNL for the processing of neutron cross section

data. The transfer of data between modules is via magnetic tape or disk in a standardized binary format. The four AMPX modules utilized in this study are discussed below.

2.2.1 XLACS

XLACS is the basic processor of the AMPX system. Its function is to process the ENDF/B data into a fine group neutron cross section library. This library is in a general format, containing resonance parameters and numerous reaction types, including n-2n, n-3n, n- α , fission spectrum, etc.

The calculational techniques utilized in XLACS were drawn from several earlier codes. Bound atom scattering matrices are calculated using techniques developed from the FLANGE-II code (Honeck and Finch, 1971). SUPERTOG (Wright et al., 1969) routines are used in the resonance and fast energy regions. The unresolved resonance treatment was taken from MC² (Toppel, Rago, and O'Shea, 1967), with modifications made for the overlap of sequences. Infinite dilution and background cross sections for the Nordheim Integral Treatment (Nordheim, 1961) are calculated in either the single or multi-level Breit-Wigner formalism. Doppler broadening of resonances is calculated using the familiar ψ and χ functions.

$$\psi(\xi, x) = \frac{\xi}{2\sqrt{\pi}} \int_{-\infty}^{\infty} \frac{\exp[-\frac{1}{4} \xi^2 (x-y)^2]}{1 + y^2} dy \quad (2.1)$$

$$\psi(\xi, x) = \frac{\xi}{\sqrt{\pi}} \int_{-\infty}^{\infty} \frac{y \cdot \exp[-\frac{1}{4} \xi^2 (x-y)^2]}{1 + y^2} dy \quad (2.2)$$

The self-shielding of unresolved resonances is approximated by a $(1/\sigma_T)$ weighting. An input weighting spectrum is used to calculate smooth cross sections. Elastic scattering matrices of any Legendre order are calculated from the angular distribution data, while inelastic scattering is calculated from either discrete level data or an evaporation model.

2.2.2 NITAWL

NITAWL utilizes the Nordheim Integral Treatment for the calculation of resonance self-shielding effects. A pseudo two-region transport calculation is made, assuming isolated resonances. Provisions are made for the treatment of two admixed moderators. Attenuation of the resonance flux due to fuel rod shadowing is accounted for by the Dancoff-Ginsburg factor, which is calculated by SUPERDAN. The self-shielded cross section library is then converted into a format suitable for transport calculations and written to magnetic tape or disk.

2.2.3 SUPERDAN

SUPERDAN (Knight, 1978) calculates the Dancoff correction factor for a square or triangular pitched lattice by a Gaussian integration scheme. The effect of fuel rod cladding is treated explicitly.

2.2.4 COMAND

COMAND collapses a multigroup transport library from an input weighting spectrum. Provisions are made in the code for truncation of upscatter or downscatter in the collapsed cross section set.

2.3 Preliminary Core Design

The accurate development of a multigroup cross section library requires a preliminary description of the system being designed. System parameters such as: (1) enrichment, (2) fuel rod pitch and diameter, (3) solution density, and (4) temperature dictate flux weighting spectra, self-shielding effects, Doppler broadening, etc. The preliminary core design utilized for cross section development was drawn from an unpublished report by Dr. R. L. Seale (1976). In this study, a series of scoping calculations were made to determine the feasibility of aqueous solutions as a safety experiment driver. A number of DTF-IV (Lathrop, 1965) calculations were made using the Hansen-Roach 16 group cross section set. The preliminary design which was developed from these calculations consisted of a uranyl nitrate driver zone and two test zones. These three zones were arranged in concentric right cylinders, with an active height of 91.44 cm (36 inches). The driver and outer test zone were neutronically isolated by a boron spectral filter. The outer test zone served as a buffer for reducing power peaking in the central zone.

The central test zone is a full 217 pin uranium dioxide hexagonal fuel subassembly. The Fast Flux Test Facility (FFTF) subassembly design was used for the test zones. The central test zone is 93% enriched in U-235. Six 30% enriched half subassemblies surround the central test zone. All fuel pins in both test zones are clad with 0.0381 cm thick #316 stainless steel. The 0.254 cm in diameter fuel

rods are arranged on a triangular pitch of 0.7624 cm. A summary of the preliminary design parameters is included as Table 2.1.

A boron spectral filter is used to isolate the driver and test regions. The initial design utilized a 0.5 cm thick boron filter. The driver region consisted of a homogeneous aqueous solution of uranyl nitrate. The solution density was 275 grams of uranium per liter, which corresponds to a hydrogen to uranium atom ratio of 100. The uranium in solution was 93% enriched in U-235. The solution tank was 5 cm thick and surrounded by a 3 cm BeO reflector.

The preliminary core design was utilized in the development of the 25 group cross section library, which is discussed in the following section.

2.4 Cross Section Development from ENDF/B to 25 Groups

The cross section data in ENDF/B-IV was condensed into a 25 group energy structure for subsequent one-dimensional transport calculations. Several steps were required before the 25 group library was developed, as illustrated in Fig. 2.1. First, ENDF/B-IV data was processed into a 218 group library with XLACS. A resonance self-shielding calculation was then made in NITAWL. The self-shielded 218 group library was collapsed by COMAND to a 70 group structure with a fission-1/E-Maxwellian weighting spectrum. A series of one-dimensional transport cell calculations were then made in ANISN (Söltez, 1969) to obtain appropriate weighting spectra for various regions of the preliminary core. These weighting spectra were then used to collapse the 70 group cross section set into a 25 group ANISN library tape.

Table 2.1 Preliminary Core Design Parameters *

fuel pin

diameter	0.5842 cm O.D.
pitch	0.7264 cm
active length	91.44 cm
cladding material	#316 s.s.
cladding thickness	0.0381 cm
enrichment	93%/30%

subassemblies

number of pins	217
spacing	11.9761 cm
measurement across flats	11.6205 cm
fuel duct thickness	0.3048 cm
fuel duct material	#316 s.s.

boron spectral filter

thickness	0.5 cm
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driver solution

type	uranyl nitrate
solution density	1.3818 g/l
uranium concentration	275.2 g/l
excess nitric acid	0.441 M
H/U ratio	100
thickness	5.0 cm
height	91.44 cm

* Seale (1976) - unpublished

Fig. 2.1 Flowchart for Cross Section Development

Fig. 2.1 (Continued)

2.4.1 ENDF/B-IV to 218 Groups

The 218 group master cross section library used in this study is a composite of non-resonance nuclei drawn from the 218 group ORNL library (Ford, Webster, and Westfall, 1976), and resonance nuclei processed from ENDF/B-IV by XLACS. The 218 group energy structure, which includes 140 epithermal and 78 thermal groups, is included in Appendix A. Energy boundaries were chosen to fit reaction thresholds and resonances of major nuclei in the library. The thermal cutoff is at 3.05 eV, which allows for the proper treatment of several broad thermal resonances with upscatter.

The non-resonance nuclei which were drawn from the ORNL library are listed as Table 2.2. Three different weighting spectra were used in the generation of cross sections for these nuclei: (1) fission- $1/E\sigma_T$ -Maxwellian, (2) $1/E\sigma_T$ #304 stainless steel-Maxwellian, and (3) fission- $1/E$ -Maxwellian. The first weighting spectrum (Fig. 2.2) was used for non-resonance nuclei, while the second weighting spectrum (Fig. 2.3) was used for structural material. The third weighting spectrum (Fig. 2.4) was used by XLACS for the processing of resonance nuclei. A summary of the three spectra is in Table 2.3. All nuclei were processed at 293°K (Doppler broadened and thermal scattering kernel evaluation) with a P-3 Legendre expansion.

The resonance nuclei which were processed from ENDF/B-IV are listed as Table 2.4. The corresponding σ_p (potential scattering cross section of the admixed moderator per absorber atom) used for the unresolved treatment is also included. The $\ell = 0$ J-state chosen for

Table 2.2 Nuclides from 218 Group ORNL Library Tape

<u>Nuclide</u>	<u>Mat No.</u>	<u>Weighting Function</u>
Be-9	1289	Fission-1/E-Maxwellian
B-10	1273	Fission-1/ $E\sigma_T$ -Maxwellian
B-11	1160	Fission-1/ $E\sigma_T$ -Maxwellian
C-12	1274	Fission-1/ $E\sigma_T$ -Maxwellian
N-14	1275	Fission-1/ $E\sigma_T$ -Maxwellian
O-16	1276	Fission-1/ $E\sigma_T$ -Maxwellian
Cr	1191	1/ $E\sigma_T$ #304 S.S.-Maxwellian
Fe	1192	1/ $E\sigma_T$ #304 S.S.-Maxwellian
Ni	1190	1/ $E\sigma_T$ #304 S.S.-Maxwellian
ZIRC-2	1284	Fission-1/E-Maxwellian
H*	1269	Fission-1/ $E\sigma_T$ -Maxwellian

* Mat 1002 water bound thermal
 See Table 2.3 for energy range of weighting functions.

Fig. 2.2 Fission- $1/E\sigma_T$ (Oxygen)-Maxwellian Weight Function.-- (Ford et al., 1976).

Fig. 2.3 $1/E\Sigma_T$ (Stainless Steel 304)-Maxwellian Weight Function.-- (Ford et al., 1976).

Fig. 2.4 Fission-1/E-Maxwellian Weight Function.-- (Ford et al., 1976).

Table 2.3 Energy Range of Weighting Functions

Weighting Function	Energy Range	Weighting
	20 - 1.4 MeV	Fission
Fission- $1/E\sigma_T$ - Maxwellian	1.4 MeV - 0.126 eV	$1/E\sigma_T$
	0.1264 - 1.0E-5 eV	Maxwellian
	20 - 0.1 MeV	Fission
Fission- $1/E$ - Maxwellian	0.1 MeV - 0.126 eV	$1/E$
	0.1264 - 1.0E-5 eV	Maxwellian
$(1/E\sigma_T)$ stainless steel #304-Maxwellian	20 MeV - .1264 eV	$1/E\sigma_T$ #304 S.S.
	0.1264 - 1.0 E-5 eV	Maxwellian

Table 2.4 Resonance Nuclei Processed in XLACS

Nuclide	Mat. No.	σ_p, b	AJIN*
U-235	1261	6.4E6	3
U-238	1262	2.7E5	1/2
Na-23	1156	3.1	2

*AJIN is the $\ell = 0$ J-state for which unresolved parameters are passed.

each nuclei was the sequence with the smallest mean level spacing, which was determined from ENDF/B-IV File 2 data. Resolved and unresolved resonances were Doppler broadened at 293°K by the ψ and χ functions for the NITAWL self-shielding treatment. The thermal scattering kernels for U-235 and U-238 were calculated from thermal scattering law data $[S(\alpha, \beta)]$ from ENDF/B-IV File 7. An analytic free gas thermal kernel was generated for Na-23 by XLACS, using routines drawn from the THERMOS code (Honeck, 1961). This step was necessary since no File 7 data was available for Na-23. All three nuclei were weighted with the fission-1/E-Maxwellian spectrum illustrated in Fig. 2.4. As a final step, resonance nuclei processed by XLACS and non-resonance nuclei from the ORNL library were merged into a 218 group master library and passed to NITAWL for self-shielding calculations.

2.4.2 Resonance Self-Shielding

A resonance self-shielding calculation was made for each of the three zones of the preliminary core discussed in Section 2.3. These three zones were self-shielded for the following geometries: (1) FFTF 93% enriched uranium dioxide fuel pin, (2) FFTF 30% enriched uranium dioxide fuel pin, and (3) homogeneous uranyl nitrate solution, 275 g/l. Dancoff correction factors were calculated in SUPERDAN for the first two geometries. (The Dancoff factor is zero for the homogeneous case.) The results of the self-shielding calculations for the three cases is included as Table 2.5. The expected trend of increasing resonance integral with increasing σ_p (decreasing absorber concentration) is apparent. Large differences in the resonance integral for identical

Table 2.5 Results from Self-Shielding Calculations

Isotope	Case	$\sigma_{p,b}$	Resonance Integral, b	
			Fis.	Abs.
U-235	93% pin	8.06	42.46	63.94
U-238	93% pin	107.10	.00005	44.01
U-235	30% pin	24.99	63.63	98.70
U-238	30% pin	10.71	.00004	14.00
Na	30/93%	0.00	0.00	0.08
U-235	solution	926.10	132.09	222.17
U-238	solution	1.28E4	.00005	262.31

nuclei in different geometries are observed. For instance, the U-235 absorption resonance integral is 63.94 barns for the 93% enriched fuel pin, while the 30% enriched fuel pin has an absorption resonance integral of 98.70 barns. These differences require the development of a separate cross section set for each of the three core zones. Each cross section set was weighted with a zone-dependent flux spectrum calculated in ANISN. These calculations are discussed in the following section.

2.4.3 70 Group Cell Calculations

Early in this study it was determined that cell calculations with the 218 group cross section set were impractical, requiring from 8-15 minutes of CPU time and 377 K_g on a CYBER-175 computer. Since a number of such runs were required, an alternative solution was sought. Investigation into current design practices revealed that such a detailed flux calculation was unnecessary. It was concluded that a fine group library of from 60-130 groups was sufficient for initial design calculations. With this in mind, a 70 (50 fast and 20 thermal) group energy structure was developed for cell calculations, and is included in Appendix A.

The 218 group cross section library was collapsed to the 70 group structure by COMAND with the fission-1/E-Maxwellian weighting spectrum depicted in Fig. 2.4. This 70 group library was then used in the following cell calculations: (1) 93% enriched fuel pin, (2) 30% enriched fuel pin, (3) uranyl nitrate solution, and (4) boron spectral

filter. These four cell calculations represent each of the major regions of the preliminary core. The flux spectra calculated for these regions are presented in Figs. 2.5-a, 2.5-b, and 2.6.

The 93% enriched fuel pin cell calculation was made with the materials and dimensions discussed in Section 2.3. ANISN was used for the one-dimensional transport calculation, with fuel, clad, and coolant zones being treated explicitly. The cross sections utilized were self-shielded for the 93% enriched fuel pin, as was previously discussed in Section 2.4.2. These results are illustrated in Figure 2.5-a, where the fraction of the total flux in each energy group is plotted as a function of group for the three fuel pin zones (fuel, clad, coolant). The spectrum in all three zones is seen to be the same, within the resolution of the plotter. Thus, one weighting spectrum was used to collapse the uranium dioxide, stainless steel, and sodium cross sections for each fuel pin. For completeness, a semi-logarithmic plot of the 93% enriched fuel pin flux spectrum is presented in Fig. 2.6. A slight softening of the neutron spectrum in the thermal energy range is observed in the sodium. However, as Fig. 2.6 illustrates, the thermal flux is approximately 10^{-14} of the fast flux, and therefore this difference in flux spectra is insignificant.

A similar cell calculation was made for the 30% enriched fuel pin. The physical dimensions and spatial mesh were unchanged. The cross section set (and naturally, the number densities) was different due to the differences in the resonance treatment. The neutron flux spectrum for this pin is illustrated in Fig. 2.5-b. As before, the

a. 95% Enriched

b. 30% Enriched

Fig. 2.5 Neutron Flux Spectrum in Test Fuel Pin

Fig. 2.6 Semi-Logarithmic Flux Spectrum for the 93% Enriched Fuel Pin

flux was normalized so that the area under each curve is unity. No differences in flux spectrum in the fuel, clad, and coolant zones were observed for the 30% enriched fuel pin. However, the flux spectrum in this fuel pin is noticeably softer than the spectrum in the 93% enriched fuel pin. The most significant differences are in groups 4 to 12 (4.3 to 0.44 MeV), where the 93% fuel pin flux is greater. The peak flux for both cases is in group 13 (0.44 to 0.27 MeV), and is of approximately the same magnitude.

An infinite medium spectrum calculation was made in ANISN for the uranyl nitrate solution described in Section 2.3. As before, specifically self-shielded cross sections for the resonance nuclei were used. The neutron flux spectrum for this region is illustrated in Fig. 2.7-a. The more thermal characteristics of the driver are apparent, although it is still predominantly a fast system. The peak flux in the solution (as compared to the 93% enriched fuel pin) is shifted from group 13 (0.44 to 0.27 MeV) to group 4 (4.3 to 3.0 MeV). The neutron flux is observed to be fairly constant in the epithermal region. A small thermal peak is observed in groups 62 to 65 (0.175 to 0.070 eV).

As one might expect, the neutron flux spectrum for the boron spectral filter is a composite of the flux in the 30% enriched fuel pin and the uranyl nitrate solution. The following one-dimensional model was used to determine a representative neutron flux spectrum in the boron spectral filter. A 0.5 cm thick natural boron cylinder was sandwiched between the reference uranyl nitrate solution and a

a. Uranyl Nitrate Driver Solution

b. Boron Spectral Filter

Fig. 2.7 Flux Spectrum in Driver System Regions

volume-weighted mixture of 30% enriched uranium dioxide, #316 stainless steel, and sodium. Both the uranyl nitrate solution and the uranium dioxide mixture were approximately ten mean free paths thick. The calculated neutron flux spectrum for the boron spectral filter is illustrated in Fig. 2.7-b. The spectrum is seen to be predominantly fast, with most of the neutrons in the 20 MeV to 30 KeV energy range (groups 1-20). Most of the thermal neutrons are effectively removed by the boron filter, as Fig. 2.7-b illustrates.

CHAPTER 3

REFERENCE DESIGN

This chapter deals with the assumptions, design criteria, and techniques which were employed in the development of the reference core design. Major emphasis was placed on core neutronics in order to accurately determine the power density and peak neutron flux in the test specimen. Other facets pertinent to the overall design were also studied, including heat removal, radiolytic gas production, corrosion, and reactor control.

3.1 Neutronic Design

Twenty-five group design calculations were made with the one-dimensional transport code, ANISN. A one-dimensional model of the reference core was developed and is discussed in Section 3.1.1. The validity of this model was verified by benchmark calculations. During the design process, the following parameters were varied: (1) reflector thickness, (2) boron thickness, (3) size of buffer assemblies, and (4) solution H/U ratio. These optimization studies are discussed in Section 3.1.2.

3.1.1 Design Summary

The final design of the homogeneous test reactor consists of two fast reactor subassembly zones driven by an aqueous uranyl nitrate solution. The active core height is 91.44 cm (36 inches). The central

test zone is a 217 pin, 93% enriched FFTF fuel subassembly. This central assembly is surrounded by a four pin thick zone of 30% enriched fuel pins, as illustrated in Fig. 3.1. This zone serves as a buffer for reducing flux peaking in the central test subassembly. Both zones contain non-circulating sodium and are separated by a #304 stainless steel can and sodium gap. Incoming thermal neutrons from the driver are effectively removed by a 0.3 cm thick natural boron spectral filter. The thickness of the boron filter determines the amount of flux peaking in the central assembly. For experiments amenable to more or less flux peaking, (for example, investigation of core-blanket wall interactions during melting of heterogeneous LMFBR cores) the filter effectiveness can be varied by adding or removing boron tubes of various thicknesses.

The uranyl nitrate driver solution contains 312 grams of 93% enriched uranium per liter, which corresponds to a H/U atom ratio of 68.5. An excess nitric acid concentration of 0.52 molar is present in the solution to prevent precipitation of solids. ZIRC-II was selected for the core solution tank due to its superior corrosion resistance. A 7 cm BeO reflector surrounds the core to reduce neutron leakage from the solution. The power density in the solution was modestly assumed to be 30 Kw/l, which was the average power density achieved in HRE-1. Greater power densities appear to be possible, as peak power densities of 58 Kw/l were attained in HRE-1. Even greater power densities (100 Kw/l) were achieved in experiments conducted at ORNL during the 1950's (TID-8518, pg. 64).

Fig. 3.1 Test and Buffer Subassembly Configuration

A summary of the neutronic design and core performance characteristics is included as Table 3.1.

Benchmark Calculations. In order to determine the accuracy of the 25 group cross section set and the one-dimensional model, a benchmark calculation was made for a highly-enriched uranyl nitrate solution. A series of critical experiments on highly-enriched uranyl nitrate solution systems were used as the benchmark (Rothe, 1978). The experiment which was modeled was a bare right-circular cylinder with a height to diameter ratio of 1.20. This provided a severe test of the one-dimensional model and the DB^2 approximation in ANISN to the transverse leakage. Thus, this benchmark calculation bounded any inherent error due to the DB^2 leakage in the one-dimensional design calculations, since the height to diameter ratio for the reference core is 3.6. A summary of the critical parameters of the benchmark solution is included as Table 3.2.

A 25 group cross section set was specifically developed for the experimental benchmark with the techniques discussed in Chapter 2. The first portion of the benchmark calculation entailed the selection of the proper order of anisotropic scattering. These results are presented in Table 3.3. From these calculations it was concluded that linearly anisotropic scattering was adequate. In order to determine if the 3.6% difference in k listed in Table 3.3 was due to the cross section set or the DB^2 leakage correction, an ANISN calculation was made for a 93% enriched aqueous uranyl nitrate solution with a height to diameter ratio of 6.0. The experiment used for this benchmark calculation was

Table 3.1 Core Performance and Neutronic Design Characteristics

<u>Core Performance</u>	
solution power density	30 kw/l
test assembly power density	16.5 kw/l
specific power in test assembly	4.12 kw/kg U
average neutron flux	2.4×10^{14} n/cm ² -sec
peak/average flux	1.084
begin to melt (1120 J/gm)	114.4 sec
end of melt (1400 J/gm)	143.0 sec
start vaporization (1700 J/gm)	173.6 sec
end vaporization (3600 J/gm)	307.6 sec
<u>Neutronic Design</u>	
test subassembly	217 pin UO ₂ FFTF
test enrichment	93%
active core height	0.9147 meters
boron filter thickness	0.3 cm
reflector thickness	7 cm BeO
driver	93% enriched uranyl nitrate
solution H/U atom ratio	68.5
solution density	1.4636 g/cm ³

Table 3.2 Summary of NRC Critical Benchmark Parameters

Uranyl Nitrate Solution

density	1.0825 g/cm ³
dissolved uranium concentration	59.65 g/l
excess HNO ₃	0.114 molar
H/U	427.7
temperature	20°C

Aluminum Solution Tank

critical solution height	36.67 cm
inside diameter	33.01 cm
outside diameter	33.65 cm
tank bottom thickness	0.64 cm

Uranium Isotopic Composition

U-234	1.024 wt-%
U-235	93.170 wt-%
U-236	0.434 wt-%
U-238	5.372 wt-%

Table 3.3 Effect of Anisotropic Scattering on the Benchmark Eigenvalue

P_ℓ	k
P_0	1.1131
P_1	0.9634
P_2	0.9641
P_3	0.9641

performed by the French in the "Consequences Radiologiques d'un Accident de Criticité" (CRAC) program (Lécorché and Seale, 1973). The experiment consisted of a 30 cm in diameter right-circular cylinder with a critical solution height of 182.1 cm. An eigenvalue of 1.0157 was calculated for the CRAC experiment, which is less than the 3.6% difference in k obtained for the NRC benchmark. This suggests that the error in the predicted eigenvalue is due to the one-dimensional model and not the 25 group cross section set. Since the height to diameter ratio for the reference core is much larger (3.6) than the benchmark solution (1.2), it was concluded that the one-dimensional model was adequate for the preliminary design, with the uncertainty in k being less than 3%.

Design Methods. A one dimensional model of the preliminary core described in Chapter 2 was developed for transport calculations in ANISN. The one-dimensional core model is illustrated in Fig. 3.2. The active core height (91.44 cm) was used in the DB^2 approximation to the transverse leakage. The benchmark calculation indicated the error associated with the one-dimensional model to be less than 3% in k . Detailed two-dimensional transport calculations should be made to refine this error during the final design process.

The first portion of this design study entailed choosing an adequate spatial mesh and proper order to anisotropic scattering. The initial calculation was with 76 mesh points, P-0 scattering and S-4 quadrature. An eigenvalue of 0.8596 was obtained, as listed in Table 3.4. (These calculations were for the preliminary design which was discussed in Chapter 2. The thickness of the BeO reflector was

All dimensions in cm.

Fig. 3.2 One Dimensional Core Model

Table 3.4 Effect of the Spatial Mesh and Anisotropic Scattering on the System Eigenvalue

Description	k
P-0 S-4 76 mesh points	0.8596
P-1 S-6 76 mesh points	0.7221
P-2 S-6 76 mesh points	0.7230
P-1 S-6 92 mesh points	0.7223

later increased from 3 cm to 7 cm, and the uranyl nitrate driver thickness was increased from 5 cm to 5.16 cm to achieve criticality.) The same problem was rerun with S-6 quadrature and P-1, P-2 scattering, yielding eigenvalues of 0.7221 and 0.7230, respectively. This indicated that P-1 scattering and S-6 quadrature was adequate for this study. Additionally, the large error associated with the isotropic case ruled out any diffusion calculations. As a final check, the spatial mesh was increased by 50%, which changed the eigenvalue by only 0.028%. This demonstrated the adequacy of the spatial mesh.

Design Criteria. Before delving into specifics of the core design, the criteria which served as design guidelines are discussed. The criteria (or figures of merit) which were used are the flux peaking factor, ψ , the power ratio, η , the power density ratio, ξ , and the solution volume, V . The power density ratio, ξ , is given by

$$\xi \equiv \frac{\text{average power density in test assembly}}{\text{average power density in driver solution}} \quad (3.1)$$

$$\xi = \frac{\frac{1}{V_T} \int_{V_T} \int_E \phi(r,E) \Sigma_f(r,E) dv dE}{\frac{1}{V_D} \int_{V_D} \int_E \phi(r,E) \Sigma_f(r,E) dv dE} \quad (3.2)$$

where V_T is the volume of the test zone and V_D is the volume of the driver zone. The power ratio, η , is simply

$$\eta = \xi \frac{V_T}{V_D} \quad (3.3)$$

The flux peaking factor, ψ , is given by

$$\psi = \frac{\phi_{\max}}{\bar{\phi}} \quad (3.4)$$

where $\phi_{\max}$ is the peak radial flux in the test subassembly and $\bar{\phi}$ is the average flux in the test subassembly.

A minimum solution volume is desired in order to reduce the total uranium inventory and the amount of solution to be cooled. Maximum values of η and ξ are desired in order to achieve power densities comparable to those present in post-shutdown LMFBR's. A flat radial flux profile is desired in the test subassembly, requiring the flux peaking factor to be near 1.0. These four figures of merit are conflicting, requiring trade-offs to be made in order to arrive at a final design.

3.1.2 Component Optimization

A number of calculations were made in order to determine a working core design within the guidelines set forth in the preceeding section. Four basic parameters were varied; the reflector thickness, the boron spectral filter thickness, the buffer zone thickness, and the composition of the driver solution.

Reflector. The optimum thickness of the BeO reflector was calculated in a straightforward manner. The reflector thickness was increased until the change in the eigenvalue was small per incremental thickness. As Fig. 3.3 illustrates, a 7 cm BeO reflector is effectively infinite, and was selected for this reason.

Fig. 3.3 Effect of the Reflector Thickness on the System Eigenvalue

The large swing in reactivity available by increasing the reflector thickness (approximately \$30 by increasing the reflector thickness from 2 to 7 cm) makes reflector movement an attractive means of reactor control. This possibility is discussed later in Section 3.5.

Boron Spectral Filter. The boron spectral filter, coupled with the buffer subassembly, serves to reduce the power peaking in the central test assembly due to incoming thermal neutrons from the driver solution. Various thicknesses of filters were investigated with different sized buffer subassemblies. It was determined that there is an inverse relationship between the boron filter and buffer subassembly thickness; i.e., a reduction in the size of the boron filter requires an increased buffer thickness in order to maintain the same flux peaking, ψ . The effect of the boron filter thickness on flux peaking in the central assembly is illustrated in Figs. 3.4-a and 3.4-b, where the power density as a function of core radius is plotted. Figure 3.4-a is for a 0.5 cm boron filter, while Fig. 3.4-b is for a core with a 0.3 cm boron filter. All other parameters are the same in both illustrations. The flux peaking in the core with the 0.3 cm boron filter is somewhat greater.

The interrelationship between the boron spectral filter thickness and the buffer subassembly size are discussed in the following section, as well as the selection of optimum filter and buffer sizes.

Buffer Subassembly. The optimum size of the buffer zone, as was discussed in the previous section, was seen to depend on the

a. 0.5 cm Boron Filter

b. 0.3 cm Boron Filter

Fig. 3.4 Power Density in Regions of Subassembly-Driver System

thickness of the boron spectral filter. The buffer subassembly should be as thin as possible in order to reduce the outer radius of the solution (and therefore the solution volume). With this in mind, the buffer zone thickness was reduced from a half subassembly (seven rings of fuel pins) to a four pin thick buffer. These four rows of pins were arranged around the central test assembly with the same diameter and triangular pitch. A series of calculations were then made for seven and four pin buffer assemblies with 0.1, 0.3, and 0.5 cm thick boron spectral filters. Figure 3.5-a shows the reduction in solution volume which was achieved with the four pin buffer. Additionally, the critical solution volume is seen to decrease with decreasing boron thickness. However, this reduction in the boron filter thickness increases flux peaking in the central assembly, as was shown in the previous section.

The effect of the buffer zone size and boron thickness on the power density ratio, ξ , and the power ratio, η , is illustrated in Fig. 3.5-b. A small improvement in η and ξ is achieved by reducing the buffer zone to four pins. Further improvement in η and ξ are realized by reducing the thickness of the boron spectral filter, but at the expense of increased flux peaking in the central test subassembly.

Figures 3.6-a and 3.6-b illustrate power peaking in the central test assembly with four and seven pin buffer zones. The thickness of the uranyl nitrate solution was varied until both cores were critical; all other parameters were held constant. The flux peaking in the four

a. Driver Solution Volume

b. Power and Power Density

Fig. 3.5 Effect of the Boron Filter Thickness on Subassembly-Driver Performance

a. Four Row Pin Buffer

b. Seven Row Pin Buffer

Fig. 3.6 Power Density Peaking in Regions of Subassembly-Driver System

pin buffer is seen to be somewhat greater. However, the peak to average flux in the assembly is 1.082, which is acceptable. Thus, the final design consists of a four pin buffer zone and a 0.3 cm boron spectral filter.

Uranyl Nitrate Solution. The most important neutronic characteristic of the uranyl nitrate solution is the H/U atom ratio, as this determines the amount of moderation in the system. The H/U ratio was varied from 68.5 to 167.5 in order to determine a design compatible with the figures of merit. It was determined that optimum moderation (as determined by the minimum critical solution volume) roughly occurs at a H/U ratio of 100, as Fig. 3.7-a illustrates. However, the H/U atom ratio which yields the minimum solution volume also produces the minimum power density ratio, ξ , as illustrated in Fig. 3.7-b. Thus, one concludes from Fig. 3.7-b the solution should be undermoderated or overmoderated. The rationale for choosing an undermoderated solution is illustrated in Fig. 3.8. The power ratio, η , is seen to increase monotonically with a decreasing H/U atom ratio. As the amount of moderation decreases, the neutron flux spectrum in the driver solution hardens. This reduces the number of neutrons absorbed by the boron filter, increasing the power density in the central test assembly. It was therefore concluded that the dissolved uranium concentration should be as high as practical in order to promote this effect.

3.2 Corrosion of Reactor Core and Externals

The uranyl nitrate fuel utilized in this design is strongly acidic, requiring careful selection of structural material in order to

a. Volume of Critical Solutions

b. Power Density Ratio

Fig. 3.7 Effect of the H/U Ratio on Subassembly-Driver Performance

Fig. 3.8 Effect of the H/U Ratio on the Power Ratio

safely contain and transport the uranyl nitrate solution. Additionally, corrosion is accelerated in a radiation environment at a rate roughly proportional to the power density. Two materials which have a demonstrated applicability in such an environment are ZIRC-II and the austenitic stainless steels (primarily ASA type 347). Titanium also has application in critical high-wear components such as pump impellers. ZIRC-II was chosen for use in the reactor core, due to the severe corrosion environment and the low zirconium neutron absorption cross section. Type 347 stainless steel is suitable for core externals, such as piping, heat exchangers, etc.

3.2.1 Reactor Core

Operation of HRE-1 and HRE-2 demonstrated that ZIRC-II was the definitive choice for core solution containment. The corrosion rate of ZIRC-II is a function of the radiation intensity, solution composition and velocity, temperature, surface preparation, and heat treatment. The rate of corrosion is most strongly a function of the solution temperature. By maintaining the solution temperature at or below 200°C, the corrosion rate should be less than 5 mills per year. The corrosion of ZIRC-II is uniform with no pitting, provided fluoride concentrations are kept below 45 ppm. No change in corrosion rates of ZIRC-II have been observed due to the presence of oxygen, hydrogen, or nitrogen in solution.

3.2.2 Reactor Externals

ASA type 347 stainless steel was successfully used for the solution circulation system in the Homogeneous Reactor Experiments. A strong temperature dependence in the corrosion rate was observed above 200°C. Below this temperature, the corrosion rate is within reasonable limits (less than 2-3 mills per year) and essentially independent of the solution velocity. In order to prevent stress corrosion cracking of type 347 stainless steel, the chloride concentration in the aqueous solution should be less than 10 ppm.

3.3 Treatment of Radiolytic Gas

The fissioning of U-235 in uranyl nitrate solution decomposes H_2O into H_2 and H_2O_2 according to the reaction,

yielding approximately 1.7 molecules of each per 100 eV of fission energy absorbed (TID-8518). The hydrogen peroxide further decomposes to

The total gas generation in the solution is approximately 4.7 l/Mw-sec. at STP. For the reference design, this corresponds to 7.6 l/sec. of radiolytic gas at STP. This amount of gas must be removed from the core safely and efficiently in order to prevent a reduction of power density in the solution driver. Two means of removal are available: (1) physical removal and combustion recombination, (2) chemical

recombination in situ by catalytic reaction. Both techniques were successfully demonstrated during the Homogeneous Reactor Experiments (HRE-1 and HRE-2).

3.3.1 Catalytic Recombination

The chemical recombination of hydrogen and oxygen occurs at a rate proportional to the dissolved hydrogen concentration (TID-8518, pg. 26). Cu(II) salts have been found to effectively catalyze this reaction. The rate determining step of this reaction is between the dissolved copper salt and hydrogen. This is expressed as (TID-8518)

$$\frac{-d[H_2]}{dt} = K_{cu} [Cu^{++}] [H_2] \quad (3.7)$$

The recombination constant, K_{cu} , is a strong function of the inverse solution temperature. Other important parameters required for evaluating K_{cu} are: the amount of excess nitric acid in solution, the uranium concentration, and the system pressure. No values of K_{cu} were found in the literature for the reference solution.

The complete recombination of radiolytic gas was successfully demonstrated in HRE-1 & 2. However, the inverse temperature dependence of K_{cu} could make this scheme unattractive for low temperature operations. In any event, chemical recombination has merit in reducing the amount of radiolytic gas to be handled by the gas removal system. Further studies are required in order to determine what fraction of radiolytic gas could be induced to recombine in a low temperature, low pressure environment.

3.3.2 Physical Removal

Physical removal and recombination of radiolytic gas was successfully employed in HRE-1 & 2. An axial gas separator was used in which stationary vanes centrifuged the radiolytic gas into a central void, where it was siphoned off for treatment. Steam was mixed with the gas for safety during transfer to a flame recombiner. Off-gas (mostly oxygen) was filtered prior to discharge in order to remove trace fission product gases. Treatment of up to 250 liters of solution was demonstrated, although HRE-2 only required a 25 l/sec capacity. Thus, it was concluded that the treatment of approximately 50 liters of solution generating 7.6 l/sec of radiolytic gas should not pose a serious engineering problem. Catalytic recombination induced by cupric salts should reduce the amount of gas to be physically separated, which would reduce the size of the gas separator.

3.4 Heat Removal

The uranyl nitrate test reactor will be cooled by circulating the driver solution through a heat exchanger. In order to promote sub-assembly melting, the sodium in the test region is not circulated. The energy deposition required for melting and vaporization of uranium dioxide is illustrated in Fig. 3.9 (Seale, 1976). Although a detailed approximation design parameters were established from scoping calculations. These calculations are listed below.

Volume of solution = 53.6 liters

Total power generated by solution = 1.61 Mw

Fig. 3.9 Energy Deposition Required for Phase Changes of UO_2

The solution specific heat is given by

$$= 1.0 - 0.0008154*U - 0.06*H \quad [\text{cal/g-}^{\circ}\text{C}]$$

where

$$U \quad [\text{g/l}]$$

$$H \quad [\text{moles/l}]$$

$$= 1.0 - (0.0008154)(312) - 0.06(0.52)$$

$$= 0.7144 \text{ cal/}^{\circ}\text{C-g}$$

The heat capacity is

$$(53.6 \times 10^3 \text{ cm}^3)(1.4636 \text{ g/cm}^3)$$

$$\times (0.7144 \text{ cal/}^{\circ}\text{C-g})(4.1819 \text{ cal/Joule})$$

$$= 0.2344 \times 10^6 \text{ J/}^{\circ}\text{C}$$

For a 25°C at ΔT across the core, the heat capacity of the active solution is

$$(0.2344 \times 10^6 \text{ J/}^{\circ}\text{C})(25^{\circ}\text{C}) = 5.859 \times 10^6 \text{ Joules}$$

The solution batch should be changed every

$$(5.859 \times 10^6 \text{ watt-sec.})/(1.61 \times 10^6 \text{ watts})$$

$$= 3.64 \text{ sec.}$$

The flow rate in the core is

$$(91.44 \text{ cm})/(3.64 \text{ sec}) = 25.1 \text{ cm/sec}$$

The mass flow of solution is

$$(1.4636 \text{ g/cm}^3)(25.1 \text{ cm/sec})(586.7 \text{ cm}^2) = 21.6 \text{ Kg/sec}$$

The mass flow rate of fuel is given by

$$(0.33477/1.4636)(21.6 \text{ Kg/sec}) = 4.93 \text{ Kg U/sec.}$$

The velocity of solution in a 10 cm safe geometry pipe is

$$V_{\text{pipe}} = V_{\text{core}} \times A_{\text{core}} / A_{\text{pipe}}$$

$$= 25.1(586.7/78.5) = 187.5 \text{ cm/sec}$$

The total U-235 inventory in solution is given by
(6 meters of 10 cm in diameter feed pipe, 15 liter heat exchanger,
and 160.8 l surge tank)

$$(\pi(5 \text{ cm})^2(609.6 \text{ cm})/(1000 \text{ cm}^3/\text{l}) + 15 \text{ l}$$

$$+ 53.6 \text{ l} + 160.8 \text{ l})(.33477 \text{ Kg U/l})(.93)$$

$$= 86.33 \text{ Kg U-235}$$

The surge tank provides sufficient volume for 3 core solution batches,
in order to induce catalytic recombination of radiolytic gas prior to
cooling. Elimination of the surge tank reduces the U-235 inventory to
36.3 Kg.

The corrosion considerations discussed in Section 3.2 indicated
that the solution temperature should be kept below 200°C. However, a
high solution temperature is desired in order to promote the rapid
recombination of radiolytic gases. In light of these considerations,
an outlet temperature of 150°C was chosen, with a 25°C ΔT across the
core. The system pressure was tentatively set at 10 atmospheres.
(Detailed calculations are required to quantify the pressure dependent
solubilities of radiolytic gas in the presence of recombination
catalysts. This coupled with the gas removal system design, will
establish the equilibrium radiolytic gas concentration and system
pressure.) The neutronics calculations indicated that the active
solution volume was 53.6 liters, operating at a power level of 1.61 Mw.
It was conservatively assumed that the equilibrium concentration of

radiolytic gas is determined by the residence time of the solution in the core. Since each batch is changed every 3.6 seconds, the equilibrium volume is approximately 27 liters of gas @STP. (Note that no credit was taken for recombination of the radiolytic gas in solution.) At 10 atmospheres pressure, this constitutes a solution void fraction of approximately 5%.

A summary of thermal-hydraulic design parameters (based on perfect gas law) is included as Table 3.5.

3.5 Reactor Control

A reactor control system must provide sufficient reactivity swing for normal operations, as well as sufficient shutdown margin for accident situations. Although detailed dynamics calculations were not performed, reactivity worths were calculated for the control system. The primary means of control during normal operations is by reflector movement. Approximately \$50 of reactivity is available by changing from full reflection to no reflection. In order to limit axial flux peaking, a rotating reflector similar to the Materials Test Reactor design should be considered. A complete design of such a system would require detailed two-dimensional transport calculations, which are beyond the scope of this study. However, as Fig. 3.10 illustrates, such a system could more than adequately handle the dynamic range of the test core. The amount of reactivity swing due to solution voiding depends on the equilibrium radiolytic gas concentration and the system pressure. With the design parameters discussed in the previous section, the solution void fraction is approximately 5%, which

Table 3.5 Summary of Thermal-Hydraulic Parameters

volume flow rate	14.7 l/sec
solution power	1.61 Mw
batch volume	53.6 l
solution heat capacity	0.234×10^6 J/°C
core flow velocity	25.1 cm/sec
feed pipe flow velocity ^A	187.5 cm/sec
pressure	10 atmospheres
core ΔT	25°C
core outlet temperature	150°C
equilibrium radiolytic gas volume at 10 atmospheres	2.71 l
U-235 mass flow rate	4.58 Kg/l
total U-235 inventory ^B	36.3 Kg

^A 10 cm I.D. safe geometry pipe

^B without surge tank

$K = 1.03646$	<u>\$ 0.0</u>	$\rho = \rho_0$ (full reflector)	+\$ 5.21
$K = 1.01145$	<u>-\$ 3.57</u>	$\rho = 0.95\rho_0$	+\$ 1.64
$K = 1.00000$	<u>-\$ 5.21</u>	$\rho = 0.9288\rho_0$	0.000
$K = 0.98448$	<u>-\$ 7.43</u>	$\rho = 0.90\rho_0$	-\$ 2.22
$K = 0.67826$	<u>-\$51.20</u>	$\rho = \rho_0$ (no reflector)	-\$45.99
$K = 0.43769$	<u>-\$85.54</u>	dumped solution (full reflector)	-\$80.33

Fig. 3.10 Dynamic Range of the Test Core

corresponds to a reactivity swing of \$3.57. Emergency shutdown (in the event of failure of the reflector control system) is accomplished by discharging the driver solution to surge tanks. The shut down margin for this system is \$85.54, as shown in Fig. 3.10. The emergency shutdown could be actuated by a pressure relief valve set to trigger when the system pressure exceeds design limits.

CHAPTER 4

SUMMARY AND RECOMMENDATIONS

The reference core design appears capable of performing meaningful safety experiments with LMFBR fuel elements. Power densities in the test subassembly in the range of 16.5 to 50 Kw/l are possible, which is sufficient energy deposition to melt the test specimen within 145 seconds, provided the sodium coolant is not circulated. Flux peaking in the test subassembly can effectively be controlled by varying the thickness of the boron spectral filter. A broad range of dynamic situations can be accommodated by reflector rotation; also, depressurization and driver solution removal provide excellent emergency shutdown control. Corrosion of core materials (which was a major problem in HRE-2) is controlled by maintaining the solution temperature below 150°C. Finally, heat removal and radiolytic gas control appear to pose no serious engineering problems.

In spite of these advantages, there are several questions which must be resolved before this concept is pursued to a final design. First and foremost is the question of the maximum power density attainable in the driver solution. From a heat transfer point of view, the proposed power densities are modest. The limiting factor in achieving high power densities is the production of radiolytic gas and corresponding reduction in the solution density. More information is needed on the rate of recombination in the presence of cupric ions, as well as

the effect of temperature, pressure, and uranium concentration on the recombination rate. A second point which must be addressed is the thermal insulation of the test zone and driver solution. The high temperature of the molten test fuel (approximately 3200°C) requires an efficient insulation and heat removal system to prevent melting of the test canister and a subsequent sodium-water reaction. With these aspects in mind, the following areas of additional study are recommended:

1. More research is needed to determine the power densities which are attainable for a low-pressure, low-temperature uranyl nitrate solution. The rate of recombination of radiolytic gas in the presence of Cu(II) catalysts should be determined, as well as the temperature and pressure dependence of the recombination constant, K_{cu} .
2. Detailed two-dimensional transport calculations should be made in order to refine the reflector control system design, as well as reducing the inherent error associated with a one-dimensional calculation.
3. A thorough thermal-hydraulic study should be initiated in order to determine pin by pin temperature profiles in the test specimen as a function of time. This will allow accurate prediction of the time and energy deposition requirements for fuel melting.
4. Dynamics studies should be initiated in order to predict temperature feedback coefficients, radial and axial sodium void coefficients, etc. The reactivity insertion associated with

fuel slumping must also be ascertained in order to insure safe operation of the test reactor.

APPENDIX A

ENERGY GROUP STRUCTURES

Table A.1 218 Energy Group Structure

Group	Upper Energy, eV	ΔU
1	2.0000E 07	1.1341
2	6.4340E 06	0.2930
3	4.8000E 06	0.1091
4	4.3040E 06	0.3609
5	3.0000E 06	0.1908
6	2.4790E 06	0.0517
7	2.3540E 06	0.2409
8	1.8500E 06	0.2097
9	1.5000E 06	0.0690
10	1.4000E 06	0.0364
11	1.3560E 06	0.0292
12	1.3170E 06	0.0522
13	1.2500E 06	0.0408
14	1.2000E 06	0.0870
15	1.1000E 06	0.0854
16	1.0100E 06	0.0933
17	9.2000E 05	0.0220
18	9.0000E 05	0.0282
19	8.7500E 05	0.0160
20	8.6100E 05	0.0489
21	8.2000E 05	0.0892
22	7.5000E 05	0.0995
23	6.7900E 05	0.0133
24	6.7000E 05	0.1103
25	6.0000E 05	0.0460
26	5.7300E 05	0.0410
27	5.5000E 05	0.0963
28	4.9952E 05	0.0609
29	4.7000E 05	0.0660
30	4.4000E 05	0.0465
31	4.2000E 05	0.0488
32	4.0000E 05	0.1924
33	3.3000E 05	0.2007
34	2.7000E 05	0.3001
35	2.0000E 05	0.2877
36	1.5000E 05	0.1563
37	1.2830E 05	0.2492
38	1.0000E 05	0.1625
39	8.5000E 04	0.0359
40	8.2000E 04	0.0892

Table A.1 (Continued) Energy Group Structure

Group	Upper Energy, eV	ΔU
41	7.5000E 04	0.0270
42	7.3000E 04	0.1961
43	6.0000E 04	0.1431
44	5.2000E 04	0.0392
45	5.0000E 04	0.1054
46	4.5000E 04	0.4055
47	3.0000E 04	0.1823
48	2.5000E 04	0.3857
49	1.7000E 04	0.2683
50	1.3000E 04	0.3137
51	9.5000E 03	0.1681
52	8.0300E 03	0.2914
53	6.0000E 03	0.4308
54	3.9000E 03	0.0419
55	3.7400E 03	0.2205
56	3.0000E 03	0.1508
57	2.5800E 03	0.1192
58	2.2900E 03	0.0401
59	2.2000E 03	0.2007
60	1.8000E 03	0.1495
61	1.5500E 03	0.0328
62	1.5000E 03	0.2657
63	1.1500E 03	0.2016
64	9.4000E 02	0.3194
65	6.8300E 02	0.0192
66	6.7000E 02	0.1974
67	5.5000E 02	0.5896
68	3.0500E 02	0.0678
69	2.8500E 02	0.1719
70	2.4000E 02	0.1335
71	2.1000E 02	0.0120
72	2.0750E 02	0.0750
73	1.9250E 02	0.0343
74	1.8600E 02	0.4217
75	1.2200E 02	0.0249
76	1.1900E 02	0.0342
77	1.1500E 02	0.0628
78	1.0800E 02	0.0770
79	1.0000E 02	0.1054
80	9.0000E 01	0.0931
81	8.2000E 01	0.0247
82	8.0000E 01	0.0513
83	7.6000E 01	0.0541
84	7.2000E 01	0.0645

Table A.1 (Continued) Energy Group Structure

Group	Upper Energy, eV	ΔU
85	6.7500E 01	0.0377
86	8.5000E 01	0.0635
87	6.1000E 01	0.0333
88	5.9000E 01	0.0997
89	5.3400E 01	0.0266
90	5.2000E 01	0.0273
91	5.0600E 01	0.0281
92	4.9200E 01	0.0185
93	4.8300E 01	0.0273
94	4.7000E 01	0.0391
95	4.5200E 01	0.0269
96	4.4000E 01	0.0370
97	4.2400E 01	0.0336
98	4.1000E 01	0.0347
99	3.9600E 01	0.0127
100	3.9100E 01	0.0285
101	3.8000E 01	0.0267
102	3.7000E 01	0.0414
103	3.5500E 01	0.0257
104	3.4600E 01	0.0249
105	3.3750E 01	0.0149
106	3.3250E 01	0.0462
107	3.1750E 01	0.0159
108	3.1250E 01	0.0408
109	3.0000E 01	0.0870
110	2.7500E 01	0.0953
111	2.5000E 01	0.1054
112	2.2500E 01	0.0690
113	2.1000E 01	0.0488
114	2.0000E 01	0.0513
115	1.9000E 01	0.0267
116	1.8500E 01	0.0846
117	1.7000E 01	0.0606
118	1.6000E 01	0.0579
119	1.5100E 01	0.0475
120	1.4400E 01	0.0462
121	1.3750E 01	0.0638
122	1.2900E 01	0.0807
123	1.1900E 01	0.0342
124	1.1500E 01	0.1398
125	1.0000E 01	0.0943
126	9.1000E 00	0.1164
127	8.1000E 00	0.1318

Table A.1 (Continued) Energy Group Structure

Group	Upper Energy, eV	ΔU
128	7.1500E 00	0.0212
129	7.0000E 00	0.0364
130	6.7500E 00	0.0377
131	6.5000E 00	0.0392
132	6.2500E 00	0.0408
133	6.0000E 00	0.1054
134	5.4000E 00	0.0770
135	5.0000E 00	0.0513
136	4.7500E 00	0.1719
137	4.0000E 00	0.0699
138	3.7300E 00	0.0636
139	3.5000E 00	0.1054
140	3.1500E 00	0.0323
141	3.0500E 00	0.0165
142	3.0000E 00	0.0101
143	2.9700E 00	0.0342
144	2.8700E 00	0.0355
145	2.7700E 00	0.0368
146	2.6700E 00	0.0382
147	2.5700E 00	0.0397
148	2.4700E 00	0.0371
149	2.3800E 00	0.0342
150	2.3000E 00	0.0399
151	2.2100E 00	0.0416
152	2.1200E 00	0.0583
153	2.0000E 00	0.0305
154	1.9400E 00	0.0421
155	1.8600E 00	0.0496
156	1.7700E 00	0.0522
157	1.6800E 00	0.0551
158	1.5900E 00	0.0583
159	1.5000E 00	0.0339
160	1.4500E 00	0.0351
161	1.4000E 00	0.0364
162	1.3500E 00	0.0377
163	1.3000E 00	0.0392
164	1.2500E 00	0.0202
165	1.2250E 00	0.0206
166	1.2000E 00	0.0211
167	1.1750E 00	0.0215
168	1.1500E 00	0.0087
169	1.1400E 00	0.0088
170	1.1300E 00	0.0089

Table A.1 (Continued) Energy Group Structure

Group	Upper Energy, eV	ΔU
171	1.1200E 00	0.0090
172	1.1100E 00	0.0090
173	1.1000E 00	0.0091
174	1.0900E 00	0.0092
175	1.0800E 00	0.0093
176	1.0700E 00	0.0094
177	1.0600E 00	0.0095
178	1.0500E 00	0.0096
179	1.0400E 00	0.0097
180	1.0300E 00	0.0098
181	1.0200E 00	0.0099
182	1.0100E 00	0.0100
183	1.0000E 00	0.0253
184	9.7500E-01	0.0260
185	9.5000E-01	0.0267
186	9.2500E-01	0.0274
187	9.0000E-01	0.0572
188	8.5000E-01	0.0606
189	8.0000E-01	0.0645
190	7.5000E-01	0.0690
191	7.0000E-01	0.0741
192	6.5000E-01	0.0800
193	6.0000E-01	0.0870
194	5.5000E-01	0.1054
195	5.0000E-01	0.1178
197	4.0000E-01	0.0645
198	3.7500E-01	0.0690
199	3.5000E-01	0.0741
200	3.2500E-01	0.0800
201	3.0000E-01	0.0870
202	2.7500E-01	0.0953
203	2.5000E-01	0.1054
204	2.2500E-01	0.1178
205	2.0000E-01	0.1335
206	1.7500E-01	0.1542
207	1.5000E-01	0.1823
208	1.2500E-01	0.2231
209	1.0000E-01	0.1054
210	9.0000E-02	0.1178
211	8.0000E-02	0.1335
212	7.0000E-02	0.1542

Table A.1 (Continued) Energy Group Structure

Group	Upper Energy, eV	ΔU
213	6.0000E-02	0.1823
214	5.0000E-02	0.2231
215	4.0000E-02	0.2877
216	3.0000E-02	0.1704
217	2.5300E-02	0.9282
218	1.0000E-02 *	6.9078

* Lower energy is 1.0000E-05 eV

Table A.2 70 Energy Group Structure

Group	Upper Energy, eV	ΔU
1	2.000E 07	1.1340
2	6.434E 06	0.2930
3	4.800E 06	0.1091
4	4.304E 06	0.3609
5	3.000E 06	0.2425
6	2.354E 06	0.2409
7	1.850E 06	0.2097
8	1.500E 06	0.1009
9	1.356E 06	0.2092
10	1.100E 06	0.2449
11	8.611E 05	0.3613
12	6.000E 05	0.3102
13	4.400E 05	0.4884
14	2.700E 05	0.3001
15	2.000E 05	0.4439
16	1.283E 05	0.4477
17	8.200E 04	0.3124
18	6.000E 04	0.2877
19	4.500E 04	0.4055
20	3.000E 04	0.1823
21	2.500E 04	0.3857
22	1.700E 04	0.2683
23	1.300E 04	0.3137
24	9.500E 03	0.4595
25	6.000E 03	0.4308
26	3.900E 03	0.2624
27	3.000E 03	0.2701
28	2.290E 03	0.2408
29	1.800E 03	0.4480
30	1.150E 03	0.1911
31	9.500E 02	0.3300
32	6.830E 02	0.2166
33	5.500E 02	0.5896
34	3.050E 02	0.2397
35	2.400E 02	0.2549
36	1.860E 02	0.4217
37	1.220E 02	0.3042
38	9.000E 01	0.2877
39	6.750E 01	0.2343
40	5.340E 01	0.2642

Table A.2 (Continued)

Group	Upper Energy, eV	ΔU
41	4.100E 01	0.3124
42	3.000E 01	0.2877
43	2.250E 01	0.1957
44	1.850E 01	0.2505
45	1.440E 01	0.2249
46	1.150E 01	0.3505
47	8.100E 00	0.3001
48	6.000E 00	0.2336
49	4.750E 00	0.2417
50	3.730E 00	0.2013
51	3.050E 00	0.2480
52	2.380E 00	0.2465
53	1.860E 00	0.2490
54	1.450E 00	0.2582
55	1.120E 00	0.2187
56	9.000E-01	0.2513
57	7.000E-01	0.3365
58	5.000E-01	0.2877
59	3.750E-01	0.2231
60	3.000E-01	0.2877
61	2.250E-01	0.2513
62	1.750E-01	0.3365
63	1.250E-01	0.3285
64	9.000E-02	0.2513
65	7.000E-02	0.3365
66	5.000E-02	0.2231
67	4.000E-02	0.2877
68	3.000E-02	0.1704
69	2.530E-02	0.9282
70	1.000E-02 *	6.9088

* Lower energy is 1.000E-05 eV

Table A.3 25 Energy Group Structure

Group	Upper Energy, eV	ΔU
1	2.000E 07	1.1340
2	6.434E 07	0.7630
3	3.000E 07	0.7941
4	1.356E 06	0.8154
5	6.000E 06	0.7985
6	2.700E 05	0.7441
7	1.283E 05	0.7600
8	6.000E 04	0.8755
9	2.500E 04	0.9676
10	9.500E 03	0.8903
11	3.900E 03	0.7732
12	1.800E 03	0.9690
13	6.830E 02	0.8062
14	3.050E 02	0.9163
15	1.220E 02	0.8262
16	5.340E 01	0.8643
17	2.250E 01	0.6712
18	1.150E 01	0.6506
19	6.000E 01	0.6766
20	3.050E 01	1.0018
21	1.120E 01	1.0942
22	3.750E-01	1.0986
23	1.250E-01	0.9163
24	5.000E-02	1.6094
25	1.000E-02*	6.9078

* Lower energy is 1.000E-05 eV

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