99
7-26-73
C-92 Dist

99
7-26-73

NC80

Dr: 247

AI-AEC-13075

REACTOR ASSEMBLY
5-kwe REACTOR THERMOELECTRIC SYSTEM

AEC Research and Development Report

Atomics International Division
Rockwell International

P.O. Box 309
Canoga Park, California 91304

MASTER

DISTRIBUTION OF THIS DOCUMENT IS **UNLIMITED**

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

NOTICE

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Atomic Energy Commission, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

REACTOR ASSEMBLY
5-kwe REACTOR THERMOELECTRIC SYSTEM

H. ROOD
Editor

Contributors

D. BOST
J. V. FACHA
L. D. FELTEN
P. H. HORTON
R. T. THEXTON
R. J. TUTTLE

NOTICE

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Atomic Energy Commission, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

Atomics International Division
Rockwell International

P.O. Box 309
Canoga Park, California 91304

CONTRACT: AT(04-3)-701
ISSUED: JUNE 30, 1973

MASTER

NOT FOR DISTRIBUTION
101

DISTRIBUTION

This report has been distributed according to the category "Systems for Nuclear Auxiliary Power (SNAP) Reactor — SNAP Program," as given in the Standard Distribution for Classified Scientific and Technical Reports, M-3679.

CONTENTS

	Page
Abstract	8
I. Introduction and Summary	9
A. Objectives	9
B. Reactor System Definition	9
C. Major Component List	11
D. Reactor System Design Summary	13
E. Reactor System Performance Summary	13
II. Reactor Design Description	17
A. Reactor Configuration	17
B. Core Assembly	17
1. Fuel Element	19
2. Grid Plates	23
3. Internal Reflectors	25
4. Core Liner	25
5. Core Assembly Sequence	27
C. Vessel Assembly	29
1. Vessel Shell	30
2. Reactor Support Assembly	31
3. Inlet Flow Distributor	31
4. Core Exit Instrumentation	33
5. NaK Piping	35
6. Assembly Sequence	35
D. Reflector Assembly	39
1. Design Requirements	39
2. Trade Studies Performed	39
3. Design Description	41
4. Expected Performance	47
III. Thermal and Hydraulic Design and Performance	53
A. Introduction	53
B. Plenum and Core Hydraulics	53
1. Inlet Plenum Hydraulic Tests	55
2. Mixing Tests	67

CONTENTS

	Page
3. Hydraulic Performance	75
C. Core Thermal Performance	77
D. Thermal Transient Behavior	83
1. Reactor Startup	83
2. Internal Reflectors	83
3. Reactor Inlet Nozzle	83
IV. Structural Design	85
A. Reactor Vessel Structural Design Status	85
1. Design Criteria	85
2. Analyses Performed	87
B. Fuel Element Operational Stress	93
C. Grid Plate Launch Stresses	97
V. Nuclear Design and Performance	99
A. Introduction	99
B. Parametric Studies	99
C. Nuclear Characteristics of the Selected Design	101
1. Reactivity and Reactivity Lifetime	101
2. Prepoison Selection and Loading	105
3. Control Worth	107
4. Temperature Coefficients	107
5. Power Distributions	109
6. Uncertainties and Shim Capabilities	113
D. Methods of Nuclear Analysis Used	114
1. Nuclear Codes	114
2. Cross Sections	118
3. Geometric Modeling	119
VI. Reactor Performance Margins	121
A. Introduction	121
B. Reactivity Lifetime Margins	121
C. Fuel Element Performance Margins	123
D. Automatic Control System Reliability	124
References	129
Appendix. Selected Layout Drawings	131

TABLES

	Page
1. Reactor Core Design Summary	12
2. Reflector Design Summary	14
3. Reactor Instrumentation Summary	15
4. Core Thermal Performance Summary	15
5. Core Nuclear Performance Summary	16
6. Control System Performance Summary	16
7. Fuel Element Dimensions	23
8. Operational Requirements of the Reflector System for the 5-kwe Reactor	40
9. Reflector Environmental Requirements	40
10. Nomenclature for Mixing Equations	70
11. Specific Test Geometry Characteristics	73
12. Coolant Channel Data	77
13. Nominal Operating Conditions	78
14. Hot Channel Factors, Center Element, Design Conditions	79
15. Hot Channel Factors, Center Element, BOL	80
16. Hot Channel Factors, Center Element, EOL	81
17. Reactor Vessel Structural Design Data	87
18. Reactor Parameters Varied	100
19. Number of Fuel Elements Trade Study	100
20. Fuel Length Trade Study	101
21. Summary of Parametric Studies, Weight Sensitive Coefficients ..	102
22. Reactor Design Summary	102
23. Reactivity Requirements	103
24. Nuclear Characteristics	103
25. Partial Temperature Coefficients	109
26. Comparison of Calculated and Experimental Reactivities of SNAP Reactors	112
27. Reactivity Lifetime Uncertainties	114

FIGURES

	Page
1. 5-kwe Reactor Conceptual Design	10
2. Core Section Details	18
3. Core Plan	20
4. Core Plan Details	21
5. Fuel Element Cross Section (Neutral Element Identical Except for Deletion of Fins)	22
6. Fuel Bundle Assembly Sequence	26
7. Core Assembly Completion	28
8. In-Channel Thermocouple Installation	28
9. Core Support Detail	29
10. Inlet Plenum	32
11. Core Exit Instrumentation Assembly	33
12. Reflector Control Verification Test	34
13. Install Channel Exit Thermocouple Assembly	36
14. Install Core Vessel	36
15. Criticality Tests	37
16. Reactor Vessel Closure	37
17. Sliding Reflector Control Drive	42
18. Backup Shutdown-Restart Mechanism	46
19. Alumina vs Carbon-Graphite Friction	48
20. Ground Test Torque Requirements	50
21. Flight Test Torque Requirements	51
22. Plenum Model	54
23. Potential Flow Lines at Top of Inlet Plenum — 1-1/2-in. Inlet Line Height	56
24. Horizontal Velocity Vectors at Top of Inlet Plenum — 1-1/2-in. Inlet Line Height	56
25. Potential Flow Lines at Top of Inlet Plenum — Horizontal Baffle in Plenum	58
26. Horizontal Velocity Vectors at Top of Inlet Plenum — Horizontal Baffle in Plenum	58
27. Diffuser Nomenclature	59

FIGURES

	Page
28. Flow Regimes in Straight-Wall, Two-Dimensional Diffusers	60
29. Hydraulic Test Data, Initial Tests	62
30. Quarter Section of Configuration I Baffle	64
31. Inlet Plenum Design	65
32. Hydraulic Test Data, Final Configuration	66
33. Hydraulic Test Grid Pressure Variation, Base Case and Final Configuration	68
34. 19-Element Mixing Test Bundle, RLN Fins, Triple Injection	69
35. Mixing Equation Nomenclature	72
36. Theoretical-Experimental Comparison, 30-mil Fin Data	74
37. Flow Channels at Edge of Core	76
38. Reference Startup Transient	82
39. Stress Categories and Limits of Stress Intensity, $T < 800^{\circ}\text{F}$	84
40. Stress Categories and Limits of Stress Intensity, $T > 800^{\circ}\text{F}$	86
41. Upper Limit Creep Prediction	88
42. Outlet Plenum Head Finite Element Analysis	90
43. Bending and 60 Stress	94
44. Beam Creep Analysis	96
45. Reactivity vs Time, 5-kwe Reactor, Design-To Conditions	104
46. BOL Power Contours, Segments in Normal Position	108
47. EOL Power Contours, Segments Fully Closed	110
48. Peak Power Per Element Relative to Core Average	111
49. 5-kwe System, Extended Life Characteristics	122
50. Reactor Lifetime Capability with One Stuck Control Segment	122
51. Functional Block Diagram, Automatic Controller	125

ABSTRACT

This report summarizes the reactor assembly engineering performed for the 5-kwe reactor. Areas pertaining to the reactor as a whole are emphasized such as the nuclear effects, plenum hydraulics and in-core coolant mixing, mechanical and structural design of the reactor, vessel design, and core assembly procedure. Also included is a discussion of reactor performance margins, and selected reactor layout drawings.

I. INTRODUCTION AND SUMMARY

A. OBJECTIVES

The overall objective of the ZrH reactor program was the development and demonstration of reliable long-life unmanned reactor thermoelectric space power systems in the 1- to 10-kwe range. To achieve this objective, the 5-kwe Reactor Thermoelectric System was designed. It was planned to test the system in an existing AEC facility at AI's Santa Susana Laboratory. The specific objectives of the 5-kwe system were to achieve:

- 1) Five-year life
- 2) Five-kwe net unconditioned power at the power system-spacecraft mating plane in a space environment at end of life
- 3) Compatibility with unmanned type spacecraft and launch vehicles such as Titan class boost vehicles
- 4) Radiation dose at the payload mating plane of 10^6 rads, gamma, and 10^{12} nvt, neutrons, average
- 5) Use of ZrH reactor and PbTe compact thermoelectric tubular module technology
- 6) Automatic startup and control capability
- 7) Compatibility with existing ground test facilities.

B. REACTOR SYSTEM DEFINITION

The ZrH Reactor System is designed to be the heat source for the 5-kwe thermoelectric system. The reactor system will be flight qualifiable as well as capable of operation in the ground test environment. As a flight system, minimum overall system weight will be obtained.

The reactor will be controlled by movement of the external beryllium reflector to regulate the neutron leakage from the core. The flight control system⁽¹⁾ will furnish the electrical power to the control motors. The reactor system shall be capable of fully automatic startup from room temperature to final operating conditions. The reactor operating conditions will be adjustable by means of discrete reflector steps.

Figure 1. 5-kwe Reactor Conceptual Design

AI-AEC-13075

The primary scram function will be accomplished by withdrawal of the reflector control elements to their least reactive position using the normal control actuators. This will be accomplished at a rate sufficient to safeguard the reactor from damage under conditions of a startup accident.

On the ground system, a backup shutdown system will be provided. This shall consist of allowing the control reflector to pivot away from the reactor vessel utilizing spring energy. In the flight system, this shutdown method will not be reversible. The ground test reactor system will include means to reset the backup shut down system.

The reactor system is shown in concept in Figure 1.

C. MAJOR COMPONENT LIST

The reactor system (Refer to Dwg. R-N652200001)* shall consist of the following major components:

- Core Assembly (Dwg. R-N652210001)

 - Grid Plates

 - Internal Reflectors (Dwg. R-N652210002)

 - Fuel Element Assembly (Dwg. R-N652240031)

 - Core Liner

- Vessel and Piping Assembly

 - Core Vessel

 - Inlet and Outlet Pipes

 - In-Core Instrumentation Assembly

 - Flow Control Baffles

- Reflector and Drive Assembly (Dwg. R-N652220032)

 - Fixed Reflector

 - Control Sector and Guide

 - Bearings

 - Drive Screw and Nut

 - Control Drum Actuator and Gear Assembly (Dwg. R-N652220095)

 - Reflector Support Plate

*All Layout drawings referenced here are included in the Appendix.

TABLE 1
REACTOR CORE DESIGN SUMMARY

Number of Elements	85
Lattice Pitch (in.)	1.025
Core Vessel OD (in.)	10.60
Active Fuel Length (in.)	16
Fuel Rod Diameter (in.)	0.938 in.
Cladding Material	Hastelloy X
Cladding Thickness, Average (mils)	25
Hydrogen Barrier Material	SCB
H ₂ Barrier Thickness (in.)	0.0025
Fuel-to-Barrier Gap, Diametral, Cold (in.)	0.012
Fin Type	Right-Left-Neutral
Number of Fins Per Element	3
Spiral Pitch (in.)	5.0
Fin Height (mils)	50 ± 0.10
Fuel Element Spacing (mils)	20
Element Backfill Gas	Helium
Backfill Pressure, Cold (psia)	3
Internal Reflectors	BeO, Type 316 Stainless Steel
Grid Plate Material	Type 316 Stainless Steel
Core Vessel Material	Type 316 Stainless Steel
Coolant Inlet and Outlet Lines	2 each
Line Size, Inlet (in.)	0.75 by 2.62, outside, Oblong round
Wall Thickness, Inlet (mils)	84
Line Size, Outlet (in.)	1.0 by 2.0, outside, Oblong round
Wall Thickness, Outlet (mils)	50

Flight Shutdown Device

Anti-Backlash and Reflector Pivot Springs

Position Switch

Backup Shutdown Reset Assembly (Ground System only,
Dwg. R-N652220033).

This report is oriented primarily toward the vessel and piping assembly and the reflector and drive assembly. Also included are all nuclear effects and the bulk thermal and hydraulic effects associated with the reactor as an assembly. The specific fuel element design details and effects of thermal-hydraulic-stress interaction of elements with their neighbors is covered in Reference 2.

D. REACTOR SYSTEM DESIGN SUMMARY

The reactor system design summary is divided into the subareas of core, reflector, and instrumentation. These are summarized in Tables 1, 2, and 3.

E. REACTOR SYSTEM PERFORMANCE SUMMARY

The reactor system performance characteristics are subdivided into core thermal performance, core nuclear performance, and control system performance. These are summarized in Tables 4, 5, and 6.

TABLE 2
REFLECTOR DESIGN SUMMARY

Reflector	
Control Type	Sliding Sector
Material	Anodized Be
Number of Sectors	Two
Axial Stroke (in.)	4
Sector Angle (°)	235°
Midplane Thickness (in.)	2.4
Actuator	
Motor Type	Redesigned S8DR
Diameter (in.)	5.125
Length (in.)	4.0
Bearings	
Ball Material	P5 Carbon Graphite
Shaft Material	Al ₂ O ₃ on Inconel 718
Socket Material	Al ₂ O ₃ on Inconel 750
Ball Screw	
Nut and Screw Material	Inconel 718
Ball Material	WC-6Co
Lubricant	MoS ₂
Pitch Diameter (in.)	0.75
Lead (in.)	0.75
Number of Ball Circuits	Two
Miscellaneous Components	
Position Switch	Modified S8DR
Cabling	Stainless-Steel-Sheathed Clad Copper
End of Life Shutdown	Modified S-10A EABRD

TABLE 3
REACTOR INSTRUMENTATION SUMMARY

Outlet Channel Temperature Thermocouples	15
Outlet NaK Line Thermocouples	2 each
Inlet NaK Lines Thermocouples	2 each
Control Reflector Thermocouples	6 maximum
Control Sector Full Out Position Switch	1 each
In-Channel Thermocouples	6

TABLE 4
CORE THERMAL PERFORMANCE SUMMARY

	EOL*	BOL†
Thermal Power to Coolant (kwt)	99.4	91.0
Outlet Temperature (° F)	1200	1135
Core Temperature Rise (° F)	95	87
Peak Fuel Temperature (° F)	1280	1225
Peak Cross Element ΔT (° F)		
Nominal Geometry	16.3	17.5
Cluster and Asymmetry	36.0	36.7
NaK Flow Rate (lb/hr)	17,000	17,000
NaK System Pressure (psi)		
Core Pressure Drop		
Fuel Bundle	0.16	
Nozzle to Nozzle	0.21	
Pressure Gradient at Inlet Grid (psi)	0.01	
Reactor Design Lifetime (yr)	5	
Coolant Velocity in Piping (fps)	5.7	
Core Average Velocity (fps)	1.64	
Reynolds Number — Core Average	6600	

*EOL = End of life

†BOL = Beginning of life

TABLE 5
CORE NUCLEAR PERFORMANCE SUMMARY

Fuel Uranium Content (wt %)	10.5
Burnable Poison Mix	100% Nat-Gd
Total Poison Loading (\$)	3.00
Cold, Clean Excess Reactivity (\$)	7.74
Short-Term Reactivity Losses (\$)	2.50
Long-Term Reactivity Losses (\$)	4.74
Total Reactivity Requirements	7.24
EOL Reactivity Margin (\$)	0.50
Available Control Worth (\$)	7.74
Minimum Shutdown Margin	3.00
Isothermal Temperature	
Coefficient	-0.18 ¢/° F
Prompt Temperature	
Coefficient	-0.07 ¢/° F

TABLE 6
CONTROL SYSTEM PERFORMANCE SUMMARY

Normal Operation	
Reactivity Step Size (¢)	0.5
Actuator Step Size (°)	7.2
Sector Step Size (mils)	5.4
Actuator Torque Requirement (oz. -in.)	78
Scram Performance	
Scram Mode	Actuator Drive
Scram Rate (¢/sec)	5
Actuator Torque Requirement (oz. -in.)	30
Backup Shutdown	Reflector Tilt-Out
Backup Shutdown Reset	Pneumatic

II. REACTOR DESIGN DESCRIPTION

A. REACTOR CONFIGURATION

The reactor designed for integration into a 5-kwe reactor thermoelectric power system is illustrated by Figure 1. It consists of: (1) a core 10.5 in. in diameter by 16 in. long (fuel-length) made up of 85 1-in.-diameter fuel elements packed in a triangular array, plus twelve internal reflector segments (stainless-steel-clad BeO) to fill the void spaces between the periphery of the fuel element array and the circular reactor vessel; (2) inlet and outlet grid plates plus a core liner to support the fuel elements within the vessel and properly control the flow through the fuel element array; (3) the reactor vessel which is the containment for the NaK coolant and serves as the basic structure of the reactor assembly; and (4) the reflector-control assembly, a tapered beryllium reflector assembled in two identical halves, with axially sliding control segments, and associated drives and actuators. The reflector mounting hardware allows the reflector halves to rotate 15° outward for end-of-life shutdown.

The NaK coolant is fed to the reactor through the two pipes which traverse the reflector through grooves in the outer face at the split line between the two halves. Flow is directed within the inlet plenum by vanes and baffling to produce a flat pressure profile over the inlet grid plate. After passing through the core, the flow exits from the outlet plenum through two nozzles oriented 15° off the reflector split-line axis.

A unique aspect of this reactor, relative to all previous SNAP reactors, is the tapered reflector with sliding control segments. All previous SNAP reactors used rotating drums for control. The primary advantage of this new design is the minimization of the diameter of the neutron shield required to shadow the payload.

B. CORE ASSEMBLY

The core is designed as a unitized assembly of the 85 fuel elements, inlet and outlet grid plates, fuel element locking rods, internal reflectors, and a

6531-5209

Figure 2. Core Section Details

AI-AEC-13075

core liner. The unit is bench assembled, with poison splines in the coolant channels to insure subcriticality, and then inserted into the reactor vessel. Figure 2 illustrates the component arrangement.

Of the 85 fuel elements in a core assembly; 27 are finned in a left-hand direction, 27 in a right-hand direction, and 31 are unfinned (neutral). The elements are arrayed in a triangular lattice as shown on Figure 3. Figure 4 shows the details of the array and the basic dimensional data for the core.

1. Fuel Element

Each fuel element consists of a segmented (5 piece) fuel rod of hydrided 10% uranium-zirconium alloy sealed within a Hastelloy-X cladding. A burnable poison is included as a vapor-deposited coating of gadolinium oxide applied to the radial surface of the fuel slugs. A ceramic coating on the inside surfaces of the cladding is the primary hydrogen retention barrier. Basic dimensional data for the fuel element are tabulated in Table 7.

The finned elements have three fins spaced 120° apart which spiral through three turns on a 5-in. pitch. The fins, 0.12 in. wide, are formed by hobbing a heavy wall starting tube. The hobbing leaves a fin height of 0.050 in. and then 6 flats are milled to a fin height of 0.010 in., giving a cross section as shown by Figure 5.

A vibration suppressor is installed between the fuel and the cold end cap as illustrated on Figure 2. This device, which resembles a collar button, prevents axial movement of the fuel rod under dynamic loadings incurred during launch. It is designed to yield by collapse of the pre-kinked column to accommodate axial fuel growth during reactor operation, without overstressing the cladding tube.

The hot end cap is basically a spherical head closure, butt welded to the cladding tube. The end pins have a 15° chamfer to provide a lead-in for the grid plate during core assembly. The pin diameter on all finned elements and on the twelve outer neutral elements is sized for a 0.020-in. (nominal) diametral clearance in the grid plate hole; and on the inner neutral elements, the pin is sized for a close fit (0.002 to 0.005 in diametral clearance). The basis for this feature of the design is discussed in Section V.

Figure 3. Core Plan

6531-51242

Figure 5. Fuel Element Cross Section (Neutral Element Identical Except for Deletion of Fins)

TABLE 7
FUEL ELEMENT DIMENSIONS

	Limit	Mean	Limit
Pitch* 1.025		1.025	
Spacing (in.)		0.020	
Cladding OD (in.)	1.002	1.005	1.008
Cladding Wall* 0.025 ± 0.001 (in.)	0.024	0.025	0.026
X 2	0.048	0.050	0.052
Cladding ID* 0.955 ± 0.001 in.	0.954	0.955	0.956
Glass (t)* 0.0025 ± 0.0005 (in.)	0.002	0.0025	0.003
X 2	0.004	0.005	0.006
Glass ID	0.948	0.950	0.952
Gap (Diameter)* 0.011 to 0.013 (in.)	0.011	0.012	0.013
Fuel OD† .9XX ± .0005 (in.)	0.935	0.938	0.941
Fuel Length, Nominal* (in.)		16.0	
Straightness, TIR*		0.012	

*Specified dimensions — all others are reference.

†Fuel OD ground and/or selected to give 0.011 to 0.013 in. -diameter gap.

The cold end cap is a flat head closure with an interference fit between the coated tube ID and the coated OD of the engaging socket portion of the cap. After the cap is inserted, the engaged section of the cap and tube is heated to the softening temperature of the ceramic to blend the coating into a continuous hydrogen barrier. The cold end pins are all sized for a close fit (0.002 to 0.005 in. diametral clearance) engagement in the grid plate holes. This close fit engagement is limited to a 0.125-in. length engagement adjacent to the end surface of the cap. The remainder is reduced in diameter to facilitate assembly of finned elements adjacent to finned elements as described in Section III-B-5.

2. Grid Plates

As illustrated on Figure 2 the inlet grid plate is the basic structural element of the core assembly. The fuel elements are secured to the grid plate

by the locking rods which are threaded through holes in the fuel element and pins. Each locking rod secures a complete row of elements across the core and thereby also performs a fuel element orientation function, to maintain the correct relationship between the flattened fins and the cladding OD of adjacent elements.

The inlet grid plate is designed as a grazed sandwich structure made up of two facing sheets 0.062 in. thick, an outer rim, and 85 spacer posts located at the fuel element axis positions. The spacer posts and the outer rim act as shear transfer elements and thereby cause the two facing sheets to function as though they are integral parts of a solid plate. This sandwich, with a 0.376-in. separation between the facing sheets has a strength equivalent to a 0.38-in.-thick solid plate and a stiffness equivalent to a 0.41-in.-thick solid plate. The sandwich plate approach thereby produces a significant weight saving.

The outlet grid plate provides only lateral spacing and support for the fuel elements. This plate is, therefore, of minimum thickness (0.062 in.). An integral rim at the outer edge (0.062 thick by 0.25 in. long) is included in the design to give the plate flatness stability and to provide a larger engagement with the core liner.

Both grid plates are fabricated from Type 316 stainless steel, and the upper grid plate assembly is furnace brazed using a nickel-base braze alloy. The fuel element positioning holes are precision located (within 0.001 in. of true position) and reamed to a tolerance of ± 0.0005 . The coolant passage holes are 0.25 in. in diameter except for twelve of the outer boundary channel holes are 0.190 in. in diameter due to space limitations. Each plate also has twelve 0.0170-in.-diameter holes near the periphery for engagement of internal reflector positioning pins.

The use of the brazed sandwich structural grid plate is based on experience on almost identical designs in the SNAP 10A and SNAP 2 DRM reactors. This covers fabrication of over twenty grid plate assemblies and experimental tests which verify the strength and deflection characteristics of the concept.⁽³⁾

3. Internal Reflectors

As shown on Figures 3 and 4, twelve internal reflector segments fill the spaces between the noncircular boundary of the fuel element array and the cylindrical core liner. These consist of BeO shapes contained in sealed stainless-steel enclosures. The surfaces which face the boundary fuel elements are configured to provide flow rates in individual boundary channels which minimizes the possible temperature difference across the boundary fuel elements.

Fast neutron irradiation of BeO causes material swelling and helium generation which produces a pressure buildup within the stainless-steel container. The design therefore includes a 1.5-in. axial gas space to limit the pressure at EOL to 18 psi, equal to the NaK coolant pressure if both NaK volume accumulator bellows fail. The internal reflectors, initially evacuated, therefore have a negative internal pressure throughout reactor lifetime.

Each internal reflector is held in position by a pin at each end which engages the grid plates. Because reflector location relative to the boundary elements is critical in establishing the boundary channel flow areas, these pins are custom located during the core assembly sequence as described in Section II-B-5.

4. Core Liner

The core liner is a 0.015-in. thick, stainless-steel cylinder closely fitted around the internal reflectors. Its primary function is to limit core bypass flow to a very low value (less than 0.5%), in order to minimize the bowing stresses induced in the boundary fuel elements by cross element temperature differentials, but still allowing a generous clearance for insertion of the unitized core assembly into the reactor vessel.

The liner has an external support flange at the core inlet end which fits between a step in the reactor vessel ID and the grid plate. The clamping pressure induced by the weld shrinkage, when the vessel closure weld is made, prevents coolant leakage from the core inlet area into the clearance between the liner and the reactor vessel. The outlet grid plate fits into the liner and is secured axially by a weld to the liner.

The clearance between the liner and the reactor vessel is open to the outlet plenum. The liner therefore operates under a low internal pressure, starting at zero at the outlet end and increasing almost linearly to the core pressure drop (0.19 psi) at the inlet end.

6531 51243

72 S20 30 39

Figure 6. Fuel Bundle Assembly Sequence

5. Core Assembly Sequence

Two basic approaches to the core assembly have been used on SNAP reactors. On the SNAP 2 and 10A reactors, the fuel elements were inserted into the core vessel one at a time, replacing dummy lucite elements, with multiplication measurements made at intervals until the assembly was completed. The SNAP 8ER and DR cores were assembled outside of the core vessel, "bench assembled," with poison splines inserted into the tri-cusp spaces between elements as the buildup progressed.

In the 5-kwe reactor core, the fins on adjacent fuel elements overlap, preventing axial insertion of finned elements. The bench assembly approach is, therefore, mandatory and the mechanical design of the core assembly is heavily influenced by this.

Prior to beginning the actual core assembly, a customized matching of the internal reflectors to the grid plates must be performed. This involves assembling the grid plates with a set of master boundary elements, which serve as high accuracy positioning fixtures for the internal reflectors, and then match drilling the holes for the reflector positioning pins. The internal reflectors are clamped into position against the master elements, and 0.12-in. -diameter holes for the locating pins are drilled and reamed into the internal reflectors, through drill bushings in the 0.188-in. -diameter holes in the grid plates.

The procedure described above positions the internal reflectors relative to the boundary elements such that the tolerance accumulations which cause deviations in the boundary flow channel areas are minimized. The only deviations in boundary flow area are those due to variations in the fuel elements (cladding OD and straightness, plus end pin eccentricity), end pin clearance, and any bowing of the reflectors or fuel elements that occurs during operation.

Figures 6, 7, and 8 illustrate the sequence proposed for assembly of the core. As shown on Figure 6, the sequence starts with the inlet grid plate mounted in a support stand. The fuel elements are inserted sequentially in rows, starting with the center row or an edge row. As each row is completed, the locking rod is inserted so that the orientation of the elements is maintained. Poison splines, 0.080-in. -diameter, stainless-steel tubes fitted with compacted B_4C , are inserted into each coolant channel as the build-up progresses. Although the bare core has a shutdown margin of approximately \$16, the poison splines are used to eliminate

Figure 7. Core Assembly Completion

Figure 8. In-Channel Thermocouple Installation

any possibility of accidental criticality during core assembly and subsequent handling. Completion of the core assembly is illustrated on Figures 7 and 8.

C. VESSEL ASSEMBLY

The reactor vessel provides containment for the NaK coolant and also acts as the basic structural element of the reactor, supporting the core internally and the reflector control assembly externally. All reactor mechanical loadings are carried through a support skirt extended from the vessel and into the neutron shield assembly by a bolted flange connection.

The core is supported within the vessel at the inlet grid plate as shown by Figure 9. The stack-up of the core liner flange, inlet grid plate, and flow distributor rests on a step in the vessel ID. The closure head has a step which engages the forward end of the flow distributor. Shrinkage of the weld between the closure head and the vessel exerts a clamping action on the stack-up and produces a rigid assembly.

Figure 9. Core Support Detail

Each of the reflector-control assembly halves is supported on pivot hinges which are part of the reactor support flange. All axial loads are reacted through these hinges. The forward ends of the reflector control assemblies are clamped against the reactor vessel by a spring-loaded tie between the halves.

The components that make up the vessel assembly are described as follows:

1. Vessel Shell

The vessel is a thin-wall cylindrical shell with an integrally formed bottom which is concave inward. The inside diameter is 10.500 in. and the overall height is 21 in. The wall thickness is 0.125 in. for the plenum areas and 0.050 in. at the core sidewall area. A 0.060-in. step in the inside radius just below the inlet grid plate provides a surface for supporting the core as described previously.

The closed end of the vessel, which forms the outlet plenum, has two oval nozzles for 1-in. high by 2-in. wide by 0.050-in. thick outlet piping elbows. These are located 180° apart on an axis that is 15° counterclockwise (looking aft) from the system Y axis.

The procedure proposed for fabricating the vessel shell is identical to that used for the SNAP 10A and SNAP 2DRM reactor vessels.⁽⁴⁾ A stainless-steel sheet is deep drawn into a shell form which has the complete inside shape and is slightly larger than the final inside diameter. By proper selection of the starting sheet thickness, the bottom plenum area will be very near the final 0.125 in. thickness and the side wall thickness will increase in a uniform taper. After annealing and cleaning, all surfaces are inspected for defects using a penetrant method.

The vessel blank is then put on a constant diameter steel spin chuck and spun to the final inside diameter using applied heat as required. After trimming the open end to length, the form is removed from the spin chuck and stress relieved. It is then installed on an expanding lathe mandrel and the outer surfaces are machined to arrive at the finished vessel wall thickness.

The inlet piping adapter ring is an extension to the open end of the shell. It includes the two balcony projections to which the inlet piping attaches. The adapter ring is machined from a ring forging and then welded to the vessel.

The completed vessel shell is given a 0.0004-in. thick nickel flash plating and baked to remove any volatiles from the plating. The nickel flash is applied to promote wetting in a subsequent brazing operation and also to protect the vessel from corrosion. Although the vessel material is very corrosion resistant at this time, it will later be subjected to temperatures of 1000 to 1200°F for extended periods during reflector assembly acceptance tests and NPU NaK loading and acceptance tests. This temperature will cause some carbide precipitation and sensitize the material to corrosion by a moist air environment.

2. Reactor Support Assembly

The reactor support is a welded assembly of the support cylinder and the base flange with integral reflector support fittings. The vessel end is configured for a furnace braze attachment at the vessel knuckle and approximately 0.38 in. up the sidewall of the vessel. This end has an internal flange with threaded holes for attachment of the center gamma shield. The support cylinder and base flange are cutouts as required for the control segment actuators which extend partially under the vessel. This assembly is attached to the vessel shell by furnace brazing (vacuum or hydrogen atmosphere) with a nickel base braze alloy.

The braze attachment of the reactor support base to the vessel shell was successively demonstrated on the S2DRM reactor.⁽⁴⁾

3. Inlet Flow Distributor

The purpose of the inlet flow distributor, shown on Figure 10, is to produce a uniform pressure over the inlet grid plate. The distributor assembly consists of a flat baffle, located 0.94 in. above the grid plate surface, with a 4-in. wide opening across the center of the plenum. The flow spreading vanes are welded to the upper surface of the baffle.

As shown on Figure 9, the flow distributor includes a circumferential rim, with cut-outs in front of the inlet pipes, which functions as a spacer in the core support arrangement.

The method proposed for fabrication is to machine the baffle plate, with its rim integral, from plate stock or a forged biscuit and then attach the flow spreading vanes, cut from sheet stock, by welding or furnace brazing.

72-017-48-216

Figure 10. Inlet Plenum

4. Core Exit Instrumentation

Fifteen thermocouples are provided for measurement of coolant temperatures at the exits of selected core channels. This is to be accomplished by centering the sensing ends of the thermocouples over the grid plate exit holes at an axial position of 0.06 to 0.12 in. away from the surface of the grid plate.

The instrument assembly, illustrated by Figure 11, is completely assembled prior to being installed in the vessel. The installation requires only one weld, that between the feedthrough and the vessel shell.

6531-5528

Figure 11. Core Exit Instrumentation Assembly

The thermocouples are chromel-alumel, magnesium oxide insulated in a 0.06-in. -diameter, stainless-steel sheath. The feedthrough for penetration of the NaK containment is a doubly-sealed, nickel-alloy-brazed design developed a number of years ago for in-pile experiments containing NaK. The design has been completely successful in this application and has more recently been used in other liquid metal (Na) systems. A key aspect of the design is the use of induction heating in a short brazing cycle to prevent the degradation of the thermocouple material physical properties caused by long times at the brazing temperature (above 1900°F) required for the nickel alloy brazes.

The sensing ends of the thermocouples are secured to the positioning posts projecting from the base (see Figure 11) by sleeves which are slipped over the thermocouple and the ends of the post, crimped, and then tack welded to the posts.

6531-5529

Figure 12. Reflector Control Verification Test

The index bracket (see Figure 11) engages a bracket extended down from the rim of the outlet grid plate to maintain the orientation of the instrumentation relative to the core.

5. NaK Piping

The NaK piping down to the reactor mating plane area is assembled as part of the reactor assembly. The reactor inlet pipes traverse the reflector at the split line. To keep the pipes within the shield shadow cone, which coincides with the outer surface of the beryllium reflector, grooves are provided in the outer surface of the reflector for the pipes. The pipes are made oval (0.75 in. by 2.52 in. outside), to minimize the depth of the grooves in the reflector, and have a wall thickness of 0.083 in. to withstand the 32-psi internal pressure.

The inlet pipe connections to the reactor vessel are slip-in socket weld joints which must be made from inside the pipe due to inadequate space between the pipe and the vessel shell. This weld will require a custom designed automatic weld head and some weld development. Just below the bottom of the reflector, the inlet pipes make a transition to 1.50 in. OD round for mating to the pipes that traverse the neutron shield.

A volume accumulator feed line, 0.37-in. OD by 0.020-in. wall, is routed through one of the reactor inlet pipes into the upper plenum area. The purpose of this line is to prevent the NaK level from dropping below the forward end of the fuel elements when the system is subjected to the accelerations of launch. NaK coverage of the fuel elements provides viscous damping of the vibration and shock loadings that occur during launch.

6. Assembly Sequence

The basic steps involved in the reactor assembly are illustrated by Figures 12 through 16. The first step in the sequence is to mate the reflector control assemblies to the reactor vessel. This involves fit-up adjustments to establish the correct clearances between the reflector and the vessel. This assembly is then put into the reflector-control verification test installation illustrated by Figure 12. A complete operational checkout of the reflector-control assembly, including the ground test backup shutdown mechanism, is then performed in vacuum with electrical heaters used to induce the component temperatures anticipated during nuclear operation. When this test is successfully completed, the

6531-5530

Figure 13. Install Channel Exit Thermocouple Assembly

6531-5212

Figure 14. Install Core Vessel

6531 5531

Figure 15. Criticality Tests

6531 5213

Figure 16. Reactor Vessel Closure

reflector is removed and the vessel is prepared for core installation. This involves the installation of the core channel exit instrumentation as illustrated by Figure 13.

Installation of the core assembly into the reactor vessel is accomplished by slipping the vessel over the core as illustrated by Figure 14A. This approach is proposed because it eliminates any handling of the core assembly. In addition, the possibility of any damage occurring during this mating is significantly less when the moving component is the 25-lb vessel instead of the 240-lb core assembly. After the vessel is secured, as illustrated by Figure 14B, it can be inverted and prepared for the criticality tests.

The installation for the criticality tests is illustrated by Figure 15. The system is located in the power test vault, and the controls and instrumentation are connected to the reactor control room through jumper cables to the vault terminal panels. All controls, including the ground test backup shutdown system, are active during the tests.

The initial approach to critical is accomplished by a stepwise removal of the poison splines from the core coolant passages. Each step involves manual removal of a group of splines, while the control segments are at full-out. The cell is vacated and sealed, and then the control sectors* are driven full-in while the neutron multiplication is measured.

After criticality is reached, and the initial reactivity and control worth measurements are completed, the reactivity adjustments are made. Reactivity is reduced by shimming up the reflector to vessel spacing pads which tilts the reflector halves outward from the vessel. Reactivity can be increased by adding beryllium shim blocks to the outer surfaces of the fixed reflector.

When all reactivity adjustments and measurements have been completed, the reflector-control assemblies are removed and a void sleeve is installed around the reactor to prevent accidental criticality during subsequent handling.

The operations involved in the reactor vessel closure are illustrated on Figure 16. This will include qualification of welding operators and procedures for each weld configuration, weld surface inspections, weld internal inspections by radiographic and/or ultrasonic methods, and helium leak test of the entire NaK containment boundary.

*The words "sector" and "segment" are used interchangeably in this report.

Sealing of the fill and drain tube extending from the closure head is accomplished as a final operation in the NPU system NaK loading sequence. This involves a double pinch of the tube, cutoff and weld at the outboard pinch, and then installation of a backup seal cap over the tube.

The void sleeve will be kept in place around the reactor vessel until the power system is assembled and ready for NaK loading and thermal acceptance tests. The void sleeve will then be removed and the reflector -control assemblies installed so that wet critical determinations and control segment calibrations can be performed as part of the thermal acceptance tests.

D. REFLECTOR ASSEMBLY

The reflector assembly for the 5-kwe reactor is tapered at 8.5° from ~ 1.2 in. thickness at the top of the reactor to ~ 3.6 in. thickness at the bottom of the reactor. Two segments of the reflector assembly, each subtending 117.5° of the total 360° of the reflector, slide vertically to provide reactivity control. Lowering the segments results in a 4-in. -high window at the core midplane, through which increased neutron leakage occurs.

1. Design Requirements

a. Operational

The operational requirements of the 5-kwe reflector system are listed in Table 8.

b. Environmental

The environmental design conditions are listed in Table 9.

2. Trade Studies Performed

a. Selection of Reflector Configuration

In selecting the reflector configuration, several designs were considered. These included tapered and untapered reflectors, using either rotating drum control or sliding segment control. As a result of trade studies comparing the four options, ⁽⁵⁾ the sliding segment, tapered reflector concept was selected, primarily on the basis of minimum weight. Other studies of control system reliability indicated that the optimum number of segments is two.

TABLE 8
OPERATIONAL REQUIREMENTS OF REFLECTOR SYSTEM
FOR THE 5-kwe REACTOR

Item	Function
Control Reflector Sector	0 to 4 in. vertical movement (2 sectors)
Control Motion	~0.005 in./step (1 electrical revolution of actuator phases)
Control Stepping Rate	1 step/sec or slower (alternate reflector sectors)
Drive Train Backlash	<0.005 in. axial and <4.0° rotational
Control Sector	
Counter Balance	Control sector weight counter balanced to prevent inward movement in the event of actuator malfunction
Flight Safety	
Launch	Control Section locked in shutdown (out) position
End of Life (EOL)	Reflector assemblies rotatable away from core vessel to achieve shutdown
Ground Test Safety	
Normal Scram	Simultaneous withdrawal of both reflector sectors at 5 steps/sec rate
Backup Shutdown	EOL reflector assembly rotation with stored energy. Reflector assemblies repositionable for restart

TABLE 9
REFLECTOR ENVIRONMENTAL REQUIREMENTS

Item	Function
Environmental-Ground Test	
Temperature (°F)	50 to 875
Pressure	Sea level ambient (low temperature operational checkout only) 10 ⁻⁵ torr or less (high temperature operation)
Acceleration (g)	1 gravity (toward the inlet end)
Shock and Vibration	Handling only
Irradiation (nvt)	1 x 10 ²⁰
Environmental-Launch (Non Operating)	
Temperature (°F)	50 to 130
Pressure	Sea level ambient to 10 ⁻⁸ torr or less
Acceleration (g)	Any combination of 7.5 axial and 2.0 lateral or 1.0 axial and 1.25 lateral, or 16.25 axial and 0.625 lateral
Shock and Vibration	To be determined (effects expected to be less than that of acceleration)
Environmental-Space Operation	
Temperature (°F)	50 to 875
Pressure	10 ⁻⁸ torr or less
Acceleration (g)	0.10 maximum in any direction
Irradiation	Same as for ground test

b. Reflector Control Drive Selection

Trade studies were also conducted to determine the method of reflector support, the type of drive mechanism, the gearing and bearing requirements, and the reflector anti-rotation devices. Studies were conducted to determine the preferred location of drive motors and other components on the basis of minimum envelope and weight constraints. A key restraint was the requirement that backup EOL shutdown for ground test be accomplished by swinging the reflector halves away from the core. A ball screw with a translating nut was selected as the final reflector drive design because it is efficient and simple. The major advantages are low drive torque, minimum self-welding potential, simple and reliable design, and availability of supporting test data at operational conditions.

The actuator was located near the shield instead of at the opposite (top) end of the reactor because of greater design flexibility, and because more space is available in which to locate the drive components. In addition, this layout shortens the screw drive and locates it near the center of gravity of the segment, locates the actuator adjacent to the load carrying structure, and shortens the electrical cables.

3. Design Description

The final conceptual design of the reactor control reflector system is shown in Figure 17. Two half reflector assemblies are individually hinged to the core vessel structure near the coolant outlet, and held together at the opposite end by a retaining device. The device incorporates a spring to take up the deflection between the two reflector halves caused by the thermal expansion of the core vessel.

The reactor control reflector is essentially a 16-in. -high, thick-walled beryllium cylinder, with an ID of 10.7 in. and a minimum wall of ~ 1.2 in. at one end, increasing uniformly at a half cone angle of 8.5° . Each fixed half reflector has a 10-in. -high window which subtends an angle of 117.5° , located at the thick end, to accommodate the movable control sector. The fixed reflector and the lower support bracket form the basic structure of each half reflector assembly. The reflector hinges, locating and retention devices, the movable control sector and its guide, and drive mechanisms are attached to this main

72-JU7-2-76

Figure 17. Sliding Reflector Control Drive

structure. The fixed reflector and the control sectors are of nuclear grade beryllium. The surfaces facing outward are black anodized, and surfaces facing the reactor core vessel are uncoated to minimize the beryllium temperatures.

a. Control Sector Guide and Drive Mechanism

The reactor operation, from the initial shutdown through startup and long-term power operation, is controlled by the axial positioning of the two control sectors which adjusts width of the neutron escape windows in the reflector system. A simple translating ball screw mechanism both guides and drives the control sector. The drive screw, in turn, is supported by bearings in the fixed reflector and the lower support bracket. All three of the guide bearings are of the self-aligning monoball design and are identical. The configuration and material combination of carbon graphite ball within the alumina coated spherical socket, and rotating on an alumina coated shaft, are the same as the S8DR control drum bearings. The bearing motion and loading (radial only or radial and thrust) on the drive screw shaft bearings are also similar to the S8DR. The control sector guide bearing undergoes simultaneous rotation and translation under very light radial loads. While this type of bearing action was not present on the S8DR, pure translational carbon graphite bearings were successfully operated on the S8DR scram snubbers. In addition, MoS_2 dry-film lubricant on the bearing surfaces will lower friction and enhance bearing operation.

The critical ball screw parameters (ball diameter, pitch diameter, and materials) of the conceptual design were selected to provide large design margin under the expected operational conditions. The calculated life is 10^6 in. of travel under 350-lb load, compared to the maximum operational requirement of 1.8×10^3 in. of travel under 26 lb. The double ball circuit feature was selected to resist the overturning moment induced by the center of gravity of the control sector being located off the drive screw center line.

A variable reluctance d-c stepping actuator generates the motive torque for the reflector drive. This actuator is a configurational and dimensional modification of the successfully tested S8DR actuator design. The materials, electromagnetic design, and the coil fabrication techniques are identical to the S8DR. As with the S8DR, the actuator produces discrete steps of 1.8° of rotation when the successive phases are energized. The 2.78:1 ratio of the integral gear box coupled with the 0.75-in. lead of the ball screw results in 0.0054 in. of control sector

motion (0.5¢ of reactivity change) for each 7.2° rotation of the actuator. The 7.2° actuator rotation per control step, which represents a complete electrical revolution of the electromagnetic phases, was selected to simplify the controller requirements.

A builtin mechanical brake locks the actuator rotor in place except when stepping motion is required. Face gears on the mating brake discs provide positive braking with minimal braking force. Rotational backlash of the brake is eliminated by utilizing a torsionally rigid bellows to give the fixed brake disc axial flexibility for the engagement and disengagement. The brake is mechanically designed to hold the control sector in place during the spacecraft launch.

Both the actuator stator and the brake are designed to accommodate sheathed cabling. The sheath of each single conductor cable is clamped at the terminal box and the conductor is welded to a cable junction which, in turn, is clamped at an insulated terminal strip. The termination design permits attachment of "jumper cables" to operate the actuator until the final installation and welding of the system cables.

The actuator assembly is attached to the lower reflector bracket and the output shaft of the gear box is attached to the shaft of the ball screw drive through a bellows-type flexible coupling. The axial and angular flexibility of the coupling separates the bearings and shaft of the actuator gear box and those of the ball screw drive to prevent any binding reaction between the two units.

Two tapered lock pins on the lower reflector bracket engage mating holes in the control sector when the sector is in the full-out position. These pins not only act as mechanical full-out stops, but also prevent rotation of the control sector during the launch. An S8DR type limit switch signals that the control sector is in the full-out position.

The hinge mounting feature of each reflector half assembly is utilized to achieve the end of life shutdown. Torsional springs located at each hinge pin are designed to overcome the bending resistance of the sheathed cable to rotate the reflector assembly away from the core vessel when the retaining device is separated. The reflector retaining device will be an electrically actuated fusible link similar to that of the SNAP 10A system. Upon receipt of the final shutdown

signal, electrical energy is applied to a small heater located within the device, heating the housing until its strength falls below the applied load, causing rupture and separation.

The normal shutdown requirement is achieved by the rapid withdrawal of both control sectors by the continuous operation of both actuators. In this operational mode, the actuator brake is disengaged by continuously energizing the brake solenoid, while the actuator is being stepped at 6-rpm rate until the full-out limit switch is actuated. At the end of the shutdown, the brake solenoid is de-energized and the mechanical brake engaged to lock the control sector prior to de-activating the actuator phases.

b. Ground Test Shutdown Mechanism

The end-of-life shutdown capability of the reflector system is also used to provide a backup shutdown feature for ground testing. As shown in Figure 18, the reflector system is modified slightly and reflector latching and reset mechanisms are added. The reflector retaining device and its thermal expansion spring are removed from the reflector, and a restart arm is attached to each reflector half assembly. This arm contains the armature of the scram magnet and the thermal expansion spring. A dual coil magnet acting on the armature, which, in turn, is spring mounted to the arm, holds each reflector half assembly against the core vessel during normal operation. When the backup shutdown signal is initiated, the power to both coils of the magnet is shut off, de-energizing the magnet, allowing the end-of-life shutdown torsion springs at the hinges to rotate each reflector half assembly away from the core. In the expected coolant-inlet-down orientation of the ground test, the center of gravity effect of each reflector half assists in the rotation away from the core. A snubber spring within the pneumatic reset cylinder cushions the reflector assembly at the end of the rotation. The thermal expansion spring, which mounts the magnet armature, acts as a secondary snubber.

To reset the reflector assembly for the reactor restart, each pneumatic reset cylinder is pressurized with inert gas to rotate the reflector half against the core vessel. A slight additional force compresses the expansion spring to permit contact between the magnet armature and the magnet face, establishing the magnetic latching. A limit switch is actuated when the reflector half is in

72-JU14-2-97

Figure 18. Backup Shutdown-Restart Mechanism

the reset position. Upon depressurization of the pneumatic cylinder, the return spring within the cylinder repositions the piston to the ready-to-snub location. An indicator switch on the cylinder indicates when this piston position is reestablished. This signal is necessary prior to reactor startup, to assure that each reflector assembly is capable of another backup shutdown operation.

In the coolant-inlet-down orientation of the reactor during the ground test, the gravitational effect on the movable control sector is toward the more reactive direction. Normally, the actuator brake holds the control sector at its position except during stepping. The torque generated at the actuator rotor during stepping is adequate to raise or lower the sector against the gravitational pull. If, however, one or more actuator phase is inactive, and the stepping is initiated, the gravitational pull on the sector may cause it to move downward until the stepping is completed and the brake is again mechanically applied. Such a failure of the actuator phase could be caused by the following malfunctions: open circuit in both coils of a phase, failure in the electrical cables between the controller and the actuator terminals, or malfunction of the controller. To preclude this possible incident, the weight of the control sector is counter-balanced. The normal flight anti-backlash springs between the fixed reflector and the sector are replaced with heavier anti-gravity springs. The counter-balancing action of these springs is sufficient to prevent inadvertent uncontrolled downward (inward) motion of the control sector in the event of actuator malfunction.

4. Expected Performance

The conceptual reflector system is designed to structurally withstand the launch environment and, subsequently operate for five years in space environment. Additionally, the system is capable of operation in the ground test environment of 10^{-5} Torr or less pressure and 1-g gravity.

During reactor startup the actuators are stepped to move alternate sectors at selected rate(s) not exceeding one 7.2° step/sec or $\sim 0.5\phi$ /sec. Once the power operation is established, an actuator is stepped periodically only to maintain the power level or to change power level. This periodic actuator

Figure 19. Alumina vs Carbon-Graphite Friction

motion will continue until both sectors are moved to the full-in position or the shutdown signal is given.

In addition to the startup and periodic stepping, reactor scram (5¢/sec) by the rapid rotation of the actuator may be required during the ground test. The system design is capable of 50 such scrams and subsequent restart during the 5-year lifetime. The number of shutdowns employing the backup shutdown system is expected to be very limited.

The torque required at the actuator rotor to position the control sector under the normal control mode is dependent on the net effective weight of the sector, bearing loads, the friction coefficient of the bearings, and the efficiencies of the ball screw and gear drives. Figure 19 shows a summary of friction data used for design of the reflector drive torque requirements.

An expected coefficient of static friction of 0.35 was selected for performance predictions with 0.10 and 0.60 as the possible lower and upper limits. The expected torque requirements under the ground test and flight conditions are shown in Figures 20 and 21. The torque values for friction coefficients of 0.35 and 0.60 were based on the lowest expected ball screw efficiency of 0.70, while the highest expected efficiency of 0.90 was used with the friction coefficient of 0.10. The gear efficiency was maintained at 90% for all cases.

Control sector scram, during ground test only, will require movement toward the full-out position. The maximum torque required to move the sector in this direction is approximately 66% of the maximum torque required to move the sector to full-in. The expected torque output of the S8DR-type actuator at the scram speed of 6 rpm is approximately 70% of slow speed stepping torque output. Thus, an actuator designed to meet the torque requirement to step the sector to full-in has adequate torque capability for the scram requirement. The actuator torque requirements of 100 oz-in. during stepping and 70 oz-in. during scram were selected to allow approximately 33% margin over the maximum expected drive requirements.

6531 5532

Figure 20. Ground Test Torque Requirements

Figure 21. Flight Test Torque Requirements

BLANK

III. THERMAL AND HYDRAULIC DESIGN AND PERFORMANCE

A. INTRODUCTION

The reactor core and fuel elements were designed to achieve improved thermal and hydraulic characteristics over the previous generation reactor, S8DR. The S8DR (SNAP 8 Developmental Reactor) operated at a power of 600 kw and a coolant outlet temperature of 1300°F. Fuel temperatures were well in excess of 1400°F (the design value) during power operation.

Several S8DR fuel elements were found with cracked cladding when the reactor was disassembled. The cladding ruptures were attributed to excessive cladding strain caused by fuel swelling, which had exceeded design limits because of overtemperature fuel conditions.⁽⁶⁾

The principal contributor to the high temperatures was the fuel element bow and cluster patterns, which formed clusters of hot coolant channels in the core. Very little coolant mixing between channels existed to moderate these hot channels. Other contributors to high fuel temperature included the greater than anticipated bypass coolant flow at the core periphery and the detrimental effects of entrained gas from the surge tank on the fuel to coolant heat transfer process. Post-operation hydraulic tests of the flow-through surge tank showed that gas was entrained into the coolant in the tank. The flow distribution in the coolant inlet plenum also contained two vortexes which would tend to concentrate the entrained gas at the vortex locations. Changes were made to alleviate all of the above problem areas for the current reactor design.

B. PLENUM AND CORE HYDRAULICS

The following changes were made to improve the hydraulic characteristics of the reactor system.

- 1) Fins were added to the fuel elements to prevent clustering and to promote interchannel mixing.
- 2) The core bypass flow was essentially eliminated by using a core liner.

AI-AEC-13075

Figure 22. Plenum Model

6531-4047

- 3) The flow-through surge tank was replaced by standoff volume accumulator units utilizing bellows to accommodate NaK volume expansion without a NaK-gas interface.
- 4) Spreading vanes and baffles at the core coolant inlet lines provide a uniform flow at the inlet grid plate.

Hydraulic tests were run with water to aid in the design of the fuel element fins and the coolant inlet plenum. Details of the test setups are given in the Hydraulic Test Summary Report.⁽⁷⁾

1. Inlet Plenum Hydraulic Tests

In keeping with the requirement of a lightweight reactor system, the inlet coolant plenum has been designed to be only about 2 in. high. The initial design had a convex, dished in, plenum head. The coolant enters the plenum through two opposing inlet nozzles. The inlet lines to the nozzles have a flattened cross section with cylindrical ends. The design dimensions at the time of the hydraulic tests were 2.62 in. by 0.75 in. outside by 0.070 in. wall. Currently the wall thickness has been increased to 0.083 in., but the inlet nozzles are essentially unchanged.

It is desired to spread the coolant in the plenum to provide a uniform flow to the inlet grid plate. Vortexes like those found in the S8DR plenum are to be eliminated. The purpose of the hydraulic tests was to find an acceptable baffle design to achieve the desired flow distribution. The reactor coolant flow is approximately 17,000 lb/hr of NaK, which at 1100°F is equivalent to 0.103 ft³/sec. This gives a Reynolds number of 1.85×10^5 in the plenum inlet lines. The acceptable pressure variation in the coolant at the grid plate is ± 0.008 psi, or $\pm 5\%$ of the reactor ΔP .

Hardware from a previous test was modified to provide a scale model of the plenum (see Figure 22). The model was tested at various flows in a water loop. Horizontal baffles and various shapes of vanes were tested along with different shaped plenum heads in the effort to achieve the desired uniform flow.

In preparation for the test, calculations were performed with a potential flow model to determine the ideal flow streamlines for two inlet plenum configurations. The calculations were performed for a total reactor coolant flow

6531-5458

Figure 23. Potential Flow Lines at Top of Inlet Plenum — 1-1/2-in. Inlet Line Height

6531-5459

Figure 24. Horizontal Velocity Vectors at Top of Inlet Plenum — 1-1/2-in. Inlet Line Height

AI-AEC-13075

of 16,300 lb/hr. For other flows, the resulting potential function and velocity vector magnitudes vary proportionally. The calculational results for the basic plenum with no vanes or baffles present are shown in Figures 23 and 24. Equipotential lines and velocity vectors are shown for a horizontal cross section just below (3/16 in.) the inlet grid plate. The flow velocity is determined by taking the negative derivative of the flow potential function.

Figures 25 and 26 show the results of similar calculations for the plenum, but with two horizontal baffles present. The baffles force the flow to spread laterally before passing through the slot midway between the two inlets. The baffles were assumed to block off about one-third of the inlet line openings because of their particular location in the plenum. The potential flow calculations give results for ideal, frictionless flow. The actual flow in the plenum is not frictionless, however, and it is necessary to force the flow to follow the ideal flow streamlines by using spreading vanes at the plenum inlets. The vane configuration was designed by a method similar to the method of Feil,⁽⁸⁾ with the flow being forced to spread slightly faster than ideal flow streamlines. Feil's paper gives the optimum amount to force the flow for a given diffuser configuration. Diffuser nomenclature is shown in Figure 27. The area ratio from the coolant inlet nozzle to the edge of the horizontal baffle which was considered is 3.36 (Figure 27). However, the nozzle and plenum do not form a simple straight diffuser. The area can be approximately separated into two sections, one coming out of the nozzle and the second on into the plenum. The vane design followed this approximation. The vanes were straight except for a single bend instead of having a slight curvature as the actual ideal streamlines would.

Figure 28 shows two charts from the work of Kline⁽⁹⁾ and his associates which indicate the type of flow which can be expected for various diffuser angles. From Figure 28b, for an area ratio of 3.36, it is essentially impossible to be in the region of "no appreciable stall" even with many closely spaced vanes. However, vanes can be used to minimize the extent of the stall and to direct the flow into the desired areas. Five vanes per inlet were chosen as a reasonable number. The initial divergence angle (2θ) between the vanes

6531-5460

Figure 25. Potential Flow Lines at Top of Inlet Plenum - Horizontal Baffle in Plenum

6531-5461

Figure 26. Horizontal Velocity Vectors at Top of Inlet Plenum - Horizontal Baffle in Plenum

AI-AEC-13075

6531-5462

Figure 27. Diffuser Nomenclature

6531-5463

Figure 28. Flow Regimes in Straight-Wall, Two-Dimensional Diffusers

was 11°, which is within the region of "no appreciable stall" (Figure 28a). The final divergence angle after the bend in the vanes was 21° and some stall might be expected.

The various plenum configurations were compared according to the following criteria:

- 1) Pressure variation at the grid plate surface
- 2) Visual observation of flow behavior in the plenum
- 3) Inlet line static pressure (related to ΔP through the plenum).

The pressure variation at the inlet gridplate was measured by 199 water manometers attached to the gridplate. These manometers provided quantitative data from which to make the final selection of the plenum configuration.

Flow in the basic plenum was dominated by the streams from the two inlet nozzles impacting in the center of the plenum to create a relatively high pressure in the central region with lower pressures at the periphery of the plenum. Two regions of relatively stagnant flow existed, one at each edge of the plenum midway between the two inlets. These regions were each characterized by two swirls or vortexes which irregularly formed and disappeared. The vane and baffle configurations tested were designed to spread the flow before the two inlet streams collided and, also, to minimize the swirls in the peripheral stagnant areas. A number of configurations were tested.⁽⁷⁾

The water flow in the test loop was sufficient to give a velocity almost twice the NaK coolant velocity. However, this provided a Reynolds number of only about 60% of the NaK Reynolds number. Pressure data taken at different flows were converted to equivalent NaK pressures and plotted as a function of inlet line Reynolds number. The data were then extrapolated to the design Reynolds number (1.85×10^5). The development of the equations for pressure conversion from water to NaK is shown as follows:

$$\Delta P = \frac{f \rho V^2 L}{2 g_c D}$$

6531-5464

Figure 29. Hydraulic Test Data, Initial Tests

$$N_{Re} = \frac{DV\rho}{\mu}$$

$$V = \frac{N_{Re}\mu}{D\rho} .$$

Also

$$f \propto N_{Re}^{-x}$$

Now

$$\frac{\Delta P_{NaK}}{\Delta P_{H_2O}} = \left(\frac{N_{Re}^{2-x} \mu^2 L}{D^3 \rho} \right)_{NaK} \left(\frac{D^3 \rho}{N_{Re}^{2-x} \mu^2 L} \right)_{H_2O} .$$

Since

$$D \propto L$$

$$\frac{\Delta P_{NaK}}{\Delta P_{H_2O}} = \left(\frac{N_{Re NaK}}{N_{Re H_2O}} \right)^{2-x} \left(\frac{\mu_{NaK}}{\mu_{H_2O}} \right)^2 \left(\frac{D_{H_2O}}{D_{NaK}} \right)^2 \left(\frac{\rho_{H_2O}}{\rho_{NaK}} \right)$$

at equivalent Reynolds Numbers;

$$\frac{\Delta P_{NaK}}{\Delta P_{H_2O}} = \left(\frac{\mu_{NaK}}{\mu_{H_2O}} \right)^2 \left(\frac{D_{H_2O}}{D_{NaK}} \right)^2 \left(\frac{\rho_{H_2O}}{\rho_{NaK}} \right) .$$

The converted pressure data for the basic plenum case is shown in Figure 29. The ΔP plotted for each run is the maximum to minimum variation at the grid plate as measured by water manometers (and converted to equivalent NaK pressure). The maximum allowable pressure goal is shown by an asterisk (*) in Figure 29. Also shown in the same figure are the case with the horizontal baffles and a case with spreading vanes at the inlet nozzles. Each case gave a slight improvement over the empty plenum, but not nearly as much as required to reach the goal. The lines shown through the experimental data points all have the same slope, which is the statistically averaged slope from the test data.

6531-5465

Figure 30. Quarter Section of Configuration I Baffle

An additional improvement was obtained by combining the modified vanes with a flat plenum head, in place of the convex head. A combination of the horizontal baffles, spreading vanes, and flat head resulted in the first configuration which gave an acceptable flow profile and was within the allowable pressure variation. A flat plenum head is structurally unacceptable. Therefore, the vane-baffle combination was tested with a concave (dished out) plenum head. These results showed a maximum pressure variation only half the allowable.

To improve the structural stability of the horizontal baffles, they were combined into a single baffle plate with a center slot (Figure 30). Also, the spreading vanes were re-designed to fit snugly between the horizontal baffle and the concave head. The previous vanes had to be cut down to clear the original convex plenum head. Pressure data from this configuration were greater than allowable. The baffle was further slotted (Figures 10 and 31) to decrease the ΔP . The spreading vanes, except for the center ones, were cut down so they no longer fitted snugly to the plenum head. The outside vanes were cut to a height of 1 in. and the adjacent ones to 1.2 in. These modifications brought the pressure variation at the grid plate back within the acceptable range.

Figure 31. Inlet Plenum Design

Figure 32. Hydraulic Test Data, Final Configuration

To further improve the flow profile, the plenum height was increased 0.15 in. to allow a full inch between the baffle plate and grid plate. This resulted in a noticeable improvement. Figure 32 shows the results of this configuration, which is the one incorporated in the reactor design.* A static pressure plot comparison of the accepted configuration with the base case is shown in Figure 33.

2. Mixing Tests

The finned elements provide much more coolant mixing between channels than non-finned elements. The amount of mixing caused by the fins is, however, difficult to predict theoretically. Empirical correlations found in the literature for mixing by finned elements are for fins with a much longer pitch and extrapolation is not recommended. A series of mixing tests⁽⁷⁾ were run with water to measure the interchannel mixing for finned elements similar to those to be used in the reactor.

A diagram (top view) of the mixing test bundle is shown in Figure 34. The RLN (right-left-neutral) fin configuration allows the fins to aid one another in moving fluid from one coolant channel to another. The mixing test bundle has 19 full elements with a neutral (no fins) element starting the RLN configuration at the bundle center. The mixing test was performed by injecting a salt solution (NaNO_3) into one or more coolant channels at the inlet grid plate. Mixing was then determined by measuring the salt concentration at different points in the test bundle and comparing the results to calculations with assumed mixing rates. Figure 34 shows three symmetrical injection points, which allow more simple calculations than for a single injection channel.

For the triple injection scheme, a $1/6$ section (60°) of the test bundle is the mirror image of its adjacent neighbor sections, and therefore, the calculations can be based on a 60° section. A symmetrical section is outlined in Figure 34. It was found that, within the axial length of the test section, no appreciable salt returned to the injection channel.

The detailed calculations divided each coolant half channel into 30 sub-channels. Fifteen of these subchannels (those near the neutral element) were

*Layout Drawing RN652200001, see Appendix.

6531-5468

Figure 33. Hydraulic Test Grid Pressure Variation,
Base Case and Final Configuration

Figure 34. 19-Element Mixing Test Bundle,
RLN Fins, Triple Injection

6531-5469

AI-AEC-13075

TABLE 10

NOMENCLATURE FOR MIXING EQUATIONS

A_{tricusp}	=	Area of a coolant channel (in. ²)
D_{elem}	=	Element outer diameter (in.)
D_R	=	Radial diffusion constant for salt (in. /sec)
H_{fin}	=	Average fin height (in.)
h_{mix}	=	Mixing factor (% in.)
ℓ_{gap}	=	Element length over which individual mixing measurement is made (in.)
P_{elem}	=	Element lattice pitch (in.)
P_{fin}	=	Axial pitch of fins (in.)
P_{flow}	=	Axial flow pitch (in.)
$P_{\text{flow}, i}$	=	Initial P_{flow} at bundle entrance (in.)
$P_{\text{flow}, f}$	=	Asymptotic P_{flow} (in.)
Q_{axial}	=	Axial flow in coolant channel (in. ³ /sec)
Q_{lateral}	=	Cross flow from one channel to another over ℓ_{gap} (in. ³ /sec)
t_{flow}	=	Height of cross flow stream (in.)
V_{axial}	=	Average axial velocity in coolant channel (in. /sec)
W_{fin}	=	Fin width (in.)
y	=	Axial distance into bundle (in.)
Z_{ent}	=	Effective entrance length of flow (in.)
θ	=	Flow angle of crossflow stream (°)

divided into rotational subchannels and diffusional subchannels. The remaining 15 subchannels are rotational only. It was assumed that the salt proceeded from one rotational subchannel to the next by mass convection. Salt movement between rotational and diffusional channels was assumed to occur by diffusion. It was found that correspondence between diffusional subchannels could be assumed zero.

The mixing factor is usually given in units of %/in., where the % is the ratio of lateral (mixing) flow to the axial channel flow and the length is the axial distance up the flow channel. Nomenclature for the mixing factor correlation is given in Table 10 and Figure 35. The lateral flow between two channels is given by the equation:

$$Q_{\text{lateral}} = 2 \times t_{\text{flow}} \times \ell_{\text{gap}} \times V_{\text{axial}} \times \tan \theta$$

where

$$\tan \theta = \frac{(1/6)\pi P_{\text{elem}}}{(1/6)P_{\text{flow}}}$$

The axial flow is

$$Q_{\text{Axial}} = A_{\text{tricusp}} \times V_{\text{Axial}}$$

The mixing ratio, $\frac{Q_{\text{Lateral}}}{Q_{\text{Axial}}}/\ell_{\text{gap}}$, then becomes

$$h_{\text{mix}} = \frac{2\pi t_{\text{flow}} P_{\text{elem}}}{A_{\text{tricusp}} P_{\text{flow}}} \times 100\%/\text{in.}$$

The flow pitch, P_{flow} , has an initial entrance value and a final asymptotic value, both of which are given in Figure 35. The value of P_{flow} varies according to the equation:

$$P_{\text{flow}} = P_{\text{flow, f}} + (P_{\text{flow, i}} - P_{\text{flow, f}})e^{-3y/Z_{\text{ent}}}$$

where

$$Z_{\text{ent}} = 0.0663/H_{\text{fin}}$$

6531-5470

Figure 35. Mixing Equation Nomenclature

The equation for coolant channel flow area is:

$$A_{\text{tricusp}} = \frac{\sqrt{3}}{4} P_{\text{elem}}^2 - \frac{\pi}{8} D_{\text{elem}}^2 - W_{\text{fin}} H_{\text{fin}} \quad .$$

We combine the above equations to get:

$$h_{\text{mix}} = \frac{236 P_{\text{elem}} H_{\text{fin}}}{P_{\text{fin}} (0.433 P_{\text{elem}}^2 - 0.393 D_{\text{elem}}^2 - W_{\text{fin}} H_{\text{fin}}) 1 + 0.91 e^{(-45.2 H_{\text{fin}} y)}}$$

where

h_{mix} is in units of % per unit length.

The above correlation was derived from experimental data for the three geometries listed in Table 11. The 20-mil-high fins were made from plastic tape, which was attached to the elements. The 15-mil and 30-mil fins were hobbled into the test element metal. The mixing data for each configuration was found to be independent of Reynolds number. A fit of the theoretical calculations to the experimental data for the 30-mil fin case is shown in Figure 36.

The results and correlations from the mixing tests have been incorporated in the design calculations of the reactor hydraulics.

TABLE 11
SPECIFIC TEST GEOMETRY CHARACTERISTICS

Characteristic	30-mil Fins	15-mil Fins	20-mil Fins
Fin Type	Hobbed	Hobbed	Plastic Tape
Number of Fins per Element	Three	Three	Three
Element Outer Diameter (in.)	0.642	0.657	0.560
Elem-Elem Separation (in.)	0.670	0.670	0.584
Actual Fin Height (in.)	0.0265	0.0115	0.0180
Fin Width (in.)	0.070	0.070	0.050
Axial Fin Pitch (in.)	5.50	5.50	4.00
Finned Length (in.)	16.0	16.0	16.0

Figure 36. Theoretical-Experimental Comparison, 30-mil Fin Data

3. Hydraulic Performance

The reactor coolant flow, supplied by the primary thermoelectric pump, is 17,500 lb/hr initially and 17,400 lb/hr after 5 years. The "design to" flow is 17,900 lb/hr, which corresponds to a 100° F core ΔT at the "design to" reactor power of 110 kw.

The 85 element core has 204 coolant channels. At the design flow the average coolant channel velocity is 1.71 ft/sec and the core bundle pressure drop (plenum to plenum) is 0.19 psi. The fuel element spacing in the core is 0.020 in. The finned elements have three fins each, which extend 0.050 in. into the coolant channel, but are cut down to 0.010 in. between fuel elements. An average fin height of 0.030 in. has been used for interchannel mixing calculations. The interchannel mixing factor for a typical coolant channel is 27%/in.

In addition to the regular coolant channels in the core, there are four types of different sized channels at the core periphery (Figure 37). Each regular (Type 1) coolant channel is bounded by two finned and one neutral fuel element. Each of the three fins on a finned element must be in one of the six channels surrounding that element at any given time, so that a finned element contributes 3/6 or 1/2 fin to each channel. Thus, a Type 1 coolant channel has an average of one fin in it throughout its length. Similarly, Type 2 edge channels (Figure 37) contain an average of one fin. Edge channel Types 3 and 4 average 1/2 fin. Channel Type 5 is between a neutral element and the core edge. It contains no fins. Hydraulic information for the various channel types is shown in Table 12. The edge channels purposely have less area than a normal channel since the edge channels service only one or two fuel elements instead of three as the inner channels do. Any variation in inlet plenum pressure at the grid plate is small enough in comparison with the fuel bundle pressure drop that it need not be considered.

Core bypass flow around the grid plates has been eliminated by the sealed liner around the internal reflectors. Leakage along any gap between the liner and the internal reflectors will be less than 0.5% of the flow by keeping the gap at 0.010 in. or less.

6531-5472

Figure 37. Flow Channels at Edge of Core

TABLE 12
COOLANT CHANNEL DATA

Channel Type	Number	Area* (in. ²)	Velocity (ft/sec)	N _{Re}	Fin Area (in. ²)	Flow (lb/hr)
1	138	0.0583	1.80	8300	0.0038	110.5
2	6	0.0402	1.09	3225	0.0038	44.7
3	24	0.0375	1.29	3790	0.0019	51.6
4	12	0.0249	1.08	2995	0.0019	27.9
5	24	0.0179	1.62	3580	0.0000	32.5
Average	204 total	0.0486	1.71	6940	0.0030	87.6

*Channel cross sectional area before subtracting the fin area.

C. CORE THERMAL PERFORMANCE

The core power requirements vary over the lifetime of the reactor because of the degradation of the thermoelectric components of the total system. In order to provide margins for uncertainties, the reactor is designed to operate at a level somewhat more severe than system requirements. The "design to" conditions are applied with the BOL (beginning-of-life) configuration when the fuel-cladding gas gap is largest, and thus, the thermal gradients are somewhat higher than the EOL (end-of-life) configuration. Because the exact manufactured configuration of each individual element is not known, other than that its dimensions are within a specified tolerance range, nominal operational conditions must be calculated for an "ideal" average element. The "ideal" is primarily the assumption that the element is perfectly straight and that the fuel segments are all centered in the cladding. The calculational methods used are described in Reference 2. Table 13 shows nominal operating conditions for "design to," BOL, and EOL conditions.

Hot channel factors have been calculated to take into account the expected range of uncertainties involved in the calculation of reactor power and temperatures. The uncertainties are divided into tolerance effects, property effects, and power flow effects. The parameter values assumed are the maximum

TABLE 13
NOMINAL OPERATING CONDITIONS

Parameter	Design*	BOL	EOL
Thermal Power (kwt)	110	95	101
Coolant Outlet Temperature (°F)	1200	1148	1200
Coolant Temperature Rise (°F)	100	88	95
Peak Fuel Centerline Temperature (°F)	1310	1247	1286
Peak Fuel Average Temperature (°F)	1292	1232	1270
Maximum Cross Element ΔT (°F)	21.4	18.8	17.2
Core Flow Rate (lb/hr)	17,900	17,500	17,400
Average Channel Velocity (fps)	1.71	1.67	1.66
Average Reynolds Number	6940	6760	6740
Minimum Reynolds Number	2990	2910	2900
Core Pressure Drop (psi)	0.19	0.18	0.18

*BOL Power Distribution

expected variations, or tolerances, and are thus considered 3σ limit values. The uncertainty factors are listed in Tables 14, 15, and 16 for "design to," BOL, and EOL conditions.

The tolerance effects have to do with the allowable range of parameters during the manufacture of the fuel element. The property effects are concerned with uncertainties in the material properties. The power and flow effects are concerned with the uncertainty in calculated power profile and with the possible variations in flow and heat flux distributions from fuel element clustering and fuel slug asymmetry in the cladding tube.

For the design conditions, Table 14, the coolant outlet temperature uncertainty is 18.5°F and the maximum fuel temperature uncertainty is 20.8°F. The maximum fuel temperature, after adding the 20.8°F uncertainty and a 5°F allowance for control band, is 1318°F.

TABLE 14
HOT CHANNEL FACTORS, CENTER ELEMENT, DESIGN CONDITIONS

	Uncertainty	$\Delta T_{\text{coolant}}$ (° F)	ΔT_{fuel} (° F)
Tolerance Effects			
Gas gap thickness (in.)	± 0.001		± 9.9
Hydrogen barrier thickness (in.)	± 0.0005		± 1.0
Cladding thickness (in.)	± 0.0015		± 0.2
Fuel uranium content (%)	± 0.3	± 2.5	± 3.8
Fuel hydrogen content (atoms/cc)	$\pm 0.15 \times 10^{22}$	± 1.7	± 2.8
Property Effects (%)			
NaK film heat transfer	± 30		± 2.9
Fuel thermal conductivity	± 12		± 1.9
Gas gap conductivity	± 14		± 8.8
Gas gap accommodation	± 30		± 1.5
Power and Flow Effects			
Channel flow distribution (%)	± 6.4	± 8.9	± 6.5
Element/element power distribution (%)	± 5	± 7.0	± 9.7
Fuel slug asymmetry	variable	± 14.0	± 6.0
Thermocouple error	1/2% per thermocouple (4 are averaged)	± 3.0	± 3.0
Statistical Sum		± 18.5	± 20.8 ± 31.2
		Nominal	Hot Channel
Nominal Operation-Maximum Values			
Channel outlet temperature (° F)		1229	1247.5
Fuel temperature (° F)		1292	1312.8
Top of Control Band Values (+5° F)			
Channel outlet temperature (° F)			1253.5
Fuel temperature (° F)			1317.8

9.4

AI-AEC-13075
79

TABLE 15
HOT CHANNEL FACTORS, CENTER ELEMENT, BOL

	Uncertainty	$\Delta T_{\text{coolant}}$ (° F)	ΔT_{fuel} (° F)
Tolerance Effects			
Gas gap thickness (in.)	±0.001		±8.6
Hydrogen barrier thickness (in.)	±0.0005		±0.8
Cladding thickness (in.)	±0.0015		±0.2
Fuel uranium content (%)	±0.3	±2.2	±3.0
Fuel hydrogen content (atoms/cc)	±0.15 x 10 ²²	±1.5	±2.4
Property Effects (%)			
NaK film heat transfer	±30		±2.5
Fuel thermal conductivity	±12		±1.7
Gas gap conductivity	±14		±7.6
Gas gap accommodation	±30		±1.3
Power and Flow Effects			
Channel flow distribution (%)	±6.4	±7.7	±5.6
Element/element power distribution (%)	±5	±6.1	±8.4
Fuel slug asymmetry	variable	±12.1	{ +5.7 -22.9
Thermocouple error	1/2% per thermocouple (4 are averaged)	±3.0	±3.0
Statistical Sum		±16.1	{ +18.1 -28.7
		Nominal	Hot Channel
Nominal Operation-Maximum Values			
Channel outlet temperature (° F)		1173	1189.1
Fuel temperature (° F)		1232	1250.1
Top of Control Band Values (+5° F)			
Channel outlet temperature (° F)			1195.1
Fuel temperature (° F)			1255.1

8.1

TABLE 16
HOT CHANNEL FACTORS, CENTER ELEMENT, EOL

	Uncertainty	$\Delta T_{\text{coolant}}$ (° F)	ΔT_{fuel} (° F)
Tolerance Effects			
Gas gap thickness (in.)	±0.001		±8.2
Hydrogen barrier thickness (in.)	±0.0005		±0.9
Cladding thickness (in.)	±0.0015		±0.2
Fuel uranium content (%)	±0.3	±2.4	±2.8
Fuel hydrogen content (atoms/cc)	±0.15 x 10 ²²	±1.6	±2.1
Property Effects			
Fuel swelling (in. ΔD)	0.0038		±16.0
NaK film heat transfer (%)	±30		±2.6
Fuel thermal conductivity (%)	±12		±1.6
Gas gap conductivity (%)	±14		±5.6
Gas gap accommodation (%)	±30		±0.8
Power and Flow Effects			
Channel flow distribution (%)	±6.4	±8.2	±5.9
Element/element power distribution (%)	±5	±6.5	±7.8
Fuel slug asymmetry	variable	±13.0	{ +5.1 -20.3
Thermocouple error	1/2% per thermocouple (4 are averaged)	±3.0	±3.0
Statistical Sum			
		±17.2	{ +23.2 -30.4
		Nominal	Hot Channel
Nominal Operation-Maximum Values			
Channel outlet temperature (° F)		1225	1242.2
Fuel temperature (° F)		1270	1293.2
Top of Control Band Values (+5° F)			
Channel outlet temperature (° F)			1248.2
Fuel temperature (° F)			1298.2

8.5

6531 51200

Figure 38. Reference Startup Transient

D. THERMAL TRANSIENT BEHAVIOR

1. Reactor Startup

The reference reactor startup transient is shown in Figure 38. The reactor first comes to sensible heat with the coolant flow still at 5%. As the reactor power comes up the reactor ΔT increases, but the startup is specified such that the maximum core ΔT does not exceed 150° F. The most severe temperature transients and core ΔT 's occur during the first two power peaks at 2800 sec and at 4200 sec. During the first power rise the reactor core ΔT reaches a peak of 148° F with the flow at 10% of full flow and the inlet temperature at 120° F. During the second rise, the core ΔT reaches 149° F with the flow at 25% and the inlet temperature at 240° F. The peak coolant temperature rise rate occurs during the second power hump. The outlet temperature peak rate is 0.9° F/sec and the inlet temperature peak rate is 1.5° F/sec.

2. Internal Reflectors

The thermal gradients imposed on the thicker of the two internal reflector pieces by the startup transient were determined. The results of an early startup study were used and the transient is more severe than for the reference startup. Thus we expect the results to be conservative. The reflector piece was studied for a vacuum in the annulus between the BeO and the cladding, and for 3 psi of helium in the annulus. The internal reflector piece will initially be evacuated, but helium will eventually build up from irradiation of the beryllium atoms.

The maximum ΔT across the cladding during the transient was 52° F for the helium-filled case, and 69° F for the vacuum case.

3. Reactor Inlet Nozzle

The response of the reactor inlet nozzle to various inlet temperature transients during reactor startup and for some accident conditions were reviewed. The highest rate found was 2.0° F/sec, but only for a short time. A 1.3° F/sec rate average for 4 min was found to be more realistic. Calculations were performed for both -1.3° F/sec and -2.0° F/sec ramp rates. As a limiting case, a collapse of the design ΔT , 100° F, was applied at the inlet as a 100° F step in NaK coolant temperature. For the most realistic transient, -1.3° F/sec, the maximum temperature difference in the nozzle was 30° F. For the -2.0° F/sec transient, the ΔT was 45° F.

NOTE 1 — This limitation applies to the range of stress intensity. When the secondary stress is due to a temperature excursion at the point at which the stresses are being analyzed, the value of S_m shall be taken as the average of the S_m values for the highest and the lowest temperature of the metal during the transient. When part or all of the secondary stress is due to mechanical load, the value of S_m shall be taken as the S_m value for the highest temperature of the metal during the transient.

NOTE 2 — The stresses in Category Q are those parts of the total stress which are produced by thermal gradients, structural discontinuities, etc., and do not include primary stresses which may also exist at the same point.

NOTE 3 — S_a is obtained from the fatigue curves. The allowable stress intensity for the full range of fluctuation is $2 S_a$.

NOTE 4 — The symbols P_m , P_L , P_b , Q , and F do not represent single quantities, but rather sets of six quantities representing the six stress components σ_t , σ_l , σ_r , τ_{tl} , τ_{lr} , and τ_{rt} .

6531 5334

Figure 39. Stress Categories and Limits of Stress Intensity, $T < 800^\circ\text{F}$

AI-AEC-13075

IV. STRUCTURAL DESIGN

A. REACTOR VESSEL STRUCTURAL DESIGN STATUS

Preliminary analyses were performed to scope the structural aspects of the reactor vessel assembly. The basic structural requirements of the vessel include support of the core within the vessel, coolant pressure containment, support of the reflector-control assemblies externally, and support of the entire reactor assembly through its attachment to the neutron shield. Other structural considerations involve stresses induced in vessel components by temperature gradients that occur during reactor startup and shutdown, external piping reactions, etc.

1. Design Criteria

The design basis structural criteria were formulated using the requirements of Section III of the ASME Boiler and Pressure Vessel Code (for Class 2 nuclear components) as guidelines. The ASME BPVC was not intended for application to space power systems, and it was considered unrealistic to adopt the Code structural requirements in total. The nature of the operating conditions to which each type of system is subjected differ sufficiently to warrant modification to the design basis. In particular, the 5-kwe Reactor Thermoelectric System is not subjected to the many cycles of severe thermal transients which are so critical in the design of high temperature stationary nuclear systems. Further, Code requirements do not consider the penalties associated with weight in a space system. The requirements for Section III, Class 2, nuclear components were, therefore, adopted with modifications based on consideration of specific requirements.

The criteria are presented on Figure 39 for temperatures below 800°F and on Figure 40 for temperatures of 800°F and above. The criteria are based on the use of elastic structural analysis methods, but do not preclude rigorous inelastic analysis of individual component areas as required to satisfy good engineering practice and/or specific functional limits. The values of S_m and S_a applicable to Figure 39 are those listed in Section III of the ASME BPVC, and the values of S_o , S_{mt} , and S_a applicable to Figure 40 are those presented in Code Case 1331-6 of Section III of the Code.

NOTE 1 — This limitation applies to the range of stress intensity. When the secondary stress is due to a temperature excursion at the point at which the stresses are being analyzed, the value of S_m shall be taken as the average of the S_m values for the highest and the lowest temperature of the metal during the transient. When part or all of the secondary stress is due to mechanical load, the value of S_m shall be taken as the S_m value for the highest temperature of the metal during the transient.

NOTE 2 — The stresses in Category Q are those parts of the total stress which are produced by thermal gradients, structural discontinuities, etc., and do not include primary stresses which may also exist at the same point.

NOTE 3 — S_a is obtained from the fatigue curves, fluctuation is $2 S_a$.

NOTE 4 — The symbols P_m , P_L , P_b , Q , and F do not represent single quantities, but rather sets of six quantities representing the six stress components σ_t , σ_l , σ_r , τ_{tl} , τ_{lr} , and τ_{rt} .

6531 5335

Figure 40. Stress Categories and Limits of Stress Intensity, $T > 800^\circ \text{F}$

AI-AEC-13075

Table 17 presents the reactor vessel structural design data applicable to defining the stress intensities and applying the design criteria.

TABLE 17
REACTOR VESSEL STRUCTURAL DESIGN DATA

Reactor Weight (lb)	497
Core	290
Vessel	38
Piping	9
Reflector	160
Launch Loads, Equivalent Static (g)	
Limit Axial	11
Limit Lateral	8
Ultimate Axial	16.5
Ultimate Lateral	12
Coolant Pressure (psi)	32
Design Temperature (° F)	1250
Design Lifetime (hr)	50,000
Material	Type 316 stainless steel
Startup Cycles	50
NaK 78 Coolant Flow (lb/hr)	16,800
Inlet Temperature, (BOL/EOL) (° F)	1057/1110
Radiation (nvt)	1.7×10^{20} (E > 1 Mev) 4.0×10^{20} (E > 0.1 Mev) (Maximum in vessel shell)

2. Analyses Performed

The structural analyses performed during the conceptual design were those required to define critical dimensions, such as where component-to-component interfaces were involved, and to verify the basic feasibility of the concept. In areas where adjustments in details of configuration and dimension could be made later without causing major changes in the design, the structural analyses were scheduled for subsequent phases of the design effort.

Figure 41. Upper Limit Creep Prediction

a. Vessel Shell Thickness

The inside diameter of the vessel was defined by the core design and internal clearance requirements. The thickness, and thereby the outside diameter, had to be established so that the design of the reflector assembly, which surrounds the vessel, could proceed on a firm basis.

The controlling criteria for the reactor vessel shell thickness are the 32-psi internal pressure at a temperature of 1250°F for 50,000 hours. The ASME code allowable membrane stress (S_{mt}) for these design conditions is 4900 psi and thereby a minimum shell thickness of 0.035 in. would normally be specified.

The ASME code does not limit creep strain as such but does specify that the designer verify that creep distortions do not destroy functional aspects of the components and modify the design if necessary. The vessel shell is surrounded by the reflector control assembly with the initial clearance set at 0.060 in. If the radial deformation of the vessel was to equal the clearance, reflector ejection for end-of-life shutdown of a flight system or backup scram of the ground test system would be impaired. A large margin of safety against reflector contact is, therefore, necessary.

The ASME creep data do not include the effect of radiation on creep. Although the code requires that radiation effects be considered, it does not provide any design criteria or guidelines, leaving it to the designer to determine the required allowances.

In AI work related to fast breeder reactor core design, irradiation creep prediction equations have been developed. These indicate that for the 5-kwe reactor conditions, the upper limit estimate of total creep be taken as the sum of ASME thermal creep data at 1250°F plus a radiation creep prediction using the AI equation at 1100°F.

Based on the above, the creep prediction curves of Figure 41 were developed. Figure 41 includes three curves of creep strain in 50,000 hours (in./in.) vs stress for (1) thermal creep at 1250°F based on data for Type 316 stainless steel from the ASME Code for Nuclear Vessels (Code Case 1331-6), (2) radiation induced creep based on the AI prediction equation applied at 1100°F at the dose rate of 10^{12} n/cm²-sec ($E > 1$ Mev), and (3) a total creep prediction which is the sum of thermal plus radiation creep.

6531-5337

Figure 42. Outlet Plenum Head Finite Element Analysis

From Figure 41, a limit of 3700 psi was selected for the primary general membrane stress in the vessel shell. The upper limit creep prediction (thermal plus radiation) is 0.0027 in. /in. in 50,000 hours giving a radial displacement for the 5.25-in. vessel radius of 0.0142 in. The elastic displacement is about 0.001 giving a total end of life displacement of 0.015 in.

b. Vessel Heads

The vessel plenum heads are spherically dished, both having a dish radius of 18 in. The inlet head is dished outward and has a knuckle radius of 1.00 in. The outlet head is dished inward and has a knuckle radius of 0.50 in.

The outlet head is completely axisymmetric and therefore amenable to analysis by a straightforward computerized finite element technique. The maximum calculated stress intensity is 6900 psi vs a design allowable of 8200 psi at 1250°F. Figure 42 shows the network used for the analysis and includes a tabulation of the principle stresses at selected locations.

The inlet head is not axisymmetric due to the two projections which extend outward to cover the projections from the vessel where the inlet pipes connect (see Figure 1 and 10). An initial scoping analysis was performed, in which the head was assumed to be axisymmetric, and then a detailed analysis of the non-symmetric areas of the head was performed.

The analysis of the nonsymmetric areas of the head, using the most advanced finite element techniques available, determined that the thickness in these areas should be increased to a minimum of 0.193 in., and that the other areas of the head are acceptable with a 0.125 in. thickness.

c. Reactor Support Skirt

The reactor is supported through a cylindrical skirt, approximately 3 in. long, which is brazed to the outlet end plenum of the vessel and has the other end flanged for a bolting attachment of the reactor to the top of the neutron shield. Cut-outs are required at two places in the skirt for the control segment drive motors.

A plain cylinder analysis indicates that a skirt thickness of 0.040 in. is required to accommodate the loading imposed by the axial and transverse accelerations of the reactor mass. The limiting condition is based on material

stress allowable rather than the elastic stability limit of the cylinder, not considering the cut-out edges. Local reinforcement is required at the edges of the cut-outs.

The neutron shield has a very large heat capacity, approximately 100,000 Btu for a 500°F temperature change. In the startup sequence, the reactor outlet rises from approximately 100 to 1140°F (BOL) over a period of about 2 hours. Even with this relatively slow reactor startup, the heat capacity of the shield is so large that when the reactor has reached its operating temperature, the shield temperature will have changed very little. The result is a large temperature gradient across the reactor support skirt, approximately 1000°F when the reactor first reaches power, which decreases slowly as the shield temperature rises. When the shield temperatures reach equilibrium, many hours after startup, the top of the shield where the reactor support skirt is attached will be in the range of 800 to 900°F, giving a temperature gradient of 300 to 400°F across the support skirt during operation.

The current design criteria require a capability of 50 startup-shutdown cycles. For a plain cylinder with fixed ends, having a linear axial temperature gradient, the maximum strain, E_m , is equal to $1.41 \frac{\alpha \Delta T}{L} \sqrt{Rt}$.

With the 0.40 in. thickness the design fatigue limit, based on Section III of the ASME Code (Figure 11 of Code Case 1331-5) is 100 cycles of application of the 1000°F temperature gradient to the cylinder. If tolerances allow the thickness to increase to 0.050 in., the design limit is reduced to 75 cycles.

Because the 50% margin on cycles was considered low, due to questions about the adequacy of the 50-cycle requirement, design modifications that would increase the cyclic capability of the support cylinder were studied. This included axial profiling of the thickness of the cylinder. A number of profiles were analyzed to find one which would locally decrease the temperature gradient at the vessel end of the cylinder. A beam-ended cylinder connection was used at the vessel end, thereby removing the circumferential restraint and reducing the stresses induced by axial flexure at this critical end.

A stepped thickness cylinder was selected as the reference concept for the conceptual design of the reactor support skirt. The thickness of the cylinder is increased over a 0.85 in. length at the vessel end. An optimization study showed

that with a 0.040 in. - thick cylinder, increasing the thickness of the vessel end of the cylinder to 0.150 in. raised the design cyclic limit from 100 to 500 cycles. If tolerances allow the 0.040 in. to increase to 0.050 in., the limit is reduced to 300 cycles.

d. Coolant Inlet Pipes

The reactor inlet pipes traverse the reflector through grooves at the split plane. In order to reduce the depth of the grooves, and to keep the pipes within the shield shadow cone, the pipes are flattened. Three basic options were considered: (1) flattened tube (flat sides with radiused ends), (2) flattened tube with internal support web, and (3) elliptical tube.

Although the internally supported flat tube offered a potential 2-lb weight advantage (neglecting meteoroid nonpuncture requirements), it was not considered worth the added difficulty and cost of installing the central support bar; i. e., obtaining and verifying a high integrity connection. The elliptical tube showed no advantage.

The basic flattened tube was selected for the reference conceptual design. Finite element analysis established that an 0.080-in. minimum thickness is required to keep the stresses, induced by the 32-psi internal pressure, within limits allowed by the ASME Code.

B. FUEL ELEMENT OPERATIONAL STRESS

Fuel elements, subject to a thermal gradient across the cladding, will assume a bowed shape dictated by the cross-element ΔT . If the thermal deflections are of sufficient magnitude and in the appropriate direction, element-to-element contact will occur resulting in forces being applied to the cladding. The RLN (right-left-neutral) fin system, used for positive spacing and inter-channel mixing, results in a fin on cladding contact which causes localized stress concentrations in addition to the bending stress caused by the force. A further stress is developed by the circumferential temperature variation (60 ripple) which results from a close-packed triangular array rod bundle. Metal subjected to a continuous stress tends to change shape in the elastic range. This creep causes strain hardening of the material and results in lowered ductility.

GIVEN Temperature Distribution at 11 Stations

120 Elements at Each Station
Temperature Defined at Hot-Cold Line Every 30 degrees

$$T = T_i + 1/2 \left[\frac{T_{i+2} + T_i}{2} \cdot T_{i+1} \right] \left[\cos \frac{2\theta}{60} \cdot 1 \right] + \left[T_{i+2} \cdot T_i \right] \frac{\theta}{60}$$

$$M_{xx} = \sum_{i=1}^{120} T_i Y_i \Delta s$$

$$M_{yy} = \sum_{i=1}^{120} T_i X_i \Delta s$$

$$\phi = \tan^{-1} (M_{yy}/M_{zz})$$

$$M_{R_{zz}} = \sqrt{M_{xx}^2 + M_{yy}^2}$$

$$T_{Bend\beta} = \frac{T_b}{2} \frac{d}{R} = \frac{T_b}{2} \cos \beta$$

$$M_{zz} = 4 \int_0^{\pi/2} \frac{T_b}{2} \cos \beta \cdot R \cos \beta ds$$

$$= R^2 \frac{T_b}{2} \int_0^{\pi/2} \cos^2 \beta d\beta = \frac{R^2 \pi}{2} T_{Bend}$$

$$M_{zz} = M_R$$

$$T_{Bend} = \frac{2}{R^2} M_R$$

$$T_{Bend} = \frac{T_{Bend}}{2} \cos(\beta)$$

$$T_{6\theta_i} = T_i - T_{Average} - T_{Bend}$$

$$\sigma_{6\theta_i} = E \alpha T_{6\theta_i}$$

$$\text{Thermal Moment} = \int_0^{\pi/2} (\text{Force/Unit Length})(\text{Distance X}) R d\theta$$

$$= 4 \int_0^{\pi/2} E \alpha (T_B/2) t R^2 \cos^2 \theta d\theta$$

$$= \frac{\pi}{2} E \alpha t R^2 T_{Bend} - \frac{E \alpha t T_{Bend}}{2R}$$

$$\text{Deflection} = \int \frac{M_{md} S}{EI}$$

$$= \frac{P a^2 b^2}{3 E I X} \quad (\text{for } \begin{array}{c} \text{---} a \text{---} b \text{---} \\ \text{---} \ell \text{---} \end{array})$$

Solve for Load (P) and Stresses Are Sum of Bending Due to P and 6θ

6531-51164

Figure 43. Bending and 6θ Stress

AI-AEC-13075

Calculation of the operational stresses was accomplished in the GEOM code simultaneously with the calculation of the temperature profiles in the reactor. The method has been fully described in Reference 10. The temperatures imposed in the fuel element were integrated at each of 11 axial stations to determine the amount of bowing due to a cross-element temperature gradient, and to find the θ stresses. The equations for determining the θ and bending stresses are shown in Figure 43. After the determination of the equivalent cross-element temperature gradient for each of 11 stations of all 85 elements, the contact forces were determined. Each element was balanced with respect to its neighbor, and an iteration technique was utilized to balance the whole core. The effects of manufacturing bow and outlet (lower) grid plate clearances were included in the analyses.

The creep analysis was a computerized extension of the bending and θ analysis. It calculated the stresses in the fuel element beam at each of 11 stations and determined the creep strain at 120 elements of each station for a given time increment. The resulting analysis was a step-by-step, strain-hardening solution leading to the stress redistribution and load relaxation as a function of time, assuming a constant displacement. Figure 44 shows the basic equations of the computer analysis.

An experiment was conducted to determine the minimum stress in the cladding which would cause gross hydrogen leakage through the SCB coating applied to the cladding. The test rig consisted of a permeation test system modified to allow uniaxial loads to be applied to fuel elements. Loads were applied in 20-lb increments until a significant increase in hydrogen permeation from the element was noted. From this test the allowable cladding stress at 1200°F was determined to be 34,000 psi. In general, barrier failure occurred at or above the 0.2% offset yield strength of the cladding material.

The GEOM code was used to perturbate various parameters to determine their effects on fuel element stress. The margin of safety was defined as:

$$M. S. = \frac{\sigma_{\text{allowable}}}{\sigma} - 1$$

with a minimum goal of a positive margin and a desirable lower limit of 1.

- USES SAME MODEL AS BENDING AND 6θ STRESS PROGRAM

- GIVEN STATIC INITIAL STRESSES AND KNOWN CREEP LAW

$$\text{Creep Moment}_i (M_{c_i}) = \int_0^{2\pi} \epsilon_{ii} \Delta t E R^2 \cos \theta d\theta$$

$$\text{Fictitious Deflection} = \int_{i=1}^{i=11} \frac{M_{c_i} m ds}{EI}$$

$$\text{Load Relaxation} = \text{Fictitious Deflection} \times \frac{3EI}{a^2 b^2}$$

- TO MAINTAIN MOMENT CONTINUITY THE CREEP MOMENT M_{c_i} IS REPLACED BY LINEAR MC/I MOMENT

- AXIAL LOAD CONTINUITY MUST BE MAINTAINED IN THE SAME MANNER

- TOTAL STRESSES

$$\sigma_i = \sigma_{i6\theta} + \sigma_{i\text{Bending}} + \sigma_i (\text{Creep Relaxation}) + \sigma_i (\text{Moment Continuity}) + \sigma_i (\text{Load Relaxation}) + \sigma_i (\text{Axial Load Continuity})$$

- RESIDUAL STRESSES ON REACTOR SHUTDOWN

- Creep Deflection Compared to Initial Bow to Determine if Contact Load Exists and Direction of Load
- Creep Moment Determined and Replaced with Equal and Opposite Linear Moment for Net Moment of Zero

$$M_c = 2 \int_0^{\pi} \epsilon_{ic} E_{rt} R^2 \cos \theta d\theta$$

- Do Same with Axial Load

- Add Moment Due to Contact with Adjacent Element

$$\sigma_{\text{Residual}} = \sigma_{\text{Creep Relaxation}} + \sigma_{\text{Linear Moment Continuity}} + \sigma_{\text{Axial Load Continuity}} + \sigma_{\text{Restrained Bending}}$$

6531 51165

Figure 44. Beam Creep Analysis

It was recognized that by studied manipulation of the variables (fuel asymmetry, manufacturing bow magnitude and direction, cladding tolerances, etc.), very low or very high safety margins could be calculated. However, unrealistic configurations must be modeled to result in the extreme conditions. The study made considered primarily plausible configurations.

The first five cases consider all variables to be completely random within tolerances with the exception that the manufacturing bow is at least 50% of the maximum allowable. This was done since a perfectly straight element is virtually impossible to manufacture. These conditions resulted in only three elements in contact out of the 425 elements modeled in the five runs. The lowest safety margin was 17.8 corresponding to a stress of only 1809 psi.

The remaining 15 runs considered all fuel slugs in an element to be oriented the same axially. This condition is quite probable since heat treatment of the final closure weld is done with the element horizontal. Asymmetry was limited to 90% of the gap to take the small fuel slug thermal bow into account. Details for all 20 runs are given in Reference 2.

C. GRID PLATE LAUNCH STRESSES

The weight of the 85 fuel elements (249 lb) is supported by the inlet grid plate. To account for the effects of acceleration, shock, and vibration imposed on the reactor during system launch, a 16.25-g equivalent static loading is specified for design.

For weight efficiency, the inlet grid plate is designed as a sandwich structure made up of 0.062-in. thick facing sheets, an outer rim, and 85 spacer bushings located at the fuel element axis positions; furnace brazed into a composite structure. This design concept was used for the grid plates of the SNAP 10A reactor and was extensively tested in developmental and qualification test programs.

Classical equations for calculation of stresses and deflections in circular plates loaded by pressure are readily available. These are for single thickness plates with no holes. The actual structure is a sandwich plate with many holes. A single plate thickness with a moment of inertia equivalent to that of the sandwich structure was calculated and used directly in the circular plate equations. The deflection equations were adjusted by the ratio of the total surface area

minus the area of unreinforced flow holes. The stress equations were adjusted to account for the greater distance to the extreme fiber of the sandwich structure plus the effect of unreinforced holes on the basis of a representative stress section.

The above methods of calculations were tested by applying them to the SNAP 10A grid plate design and comparing the results to experimental test data on those plates. The comparison showed the methods to be valid and they were then applied to the 5-kwe reactor grid plate design. Under the 16.25-g equivalent static design load, the calculated maximum stress is 18,200 psi, giving a margin of safety of 0.65 based on a minimum yield strength of 30,000 psi, and the peak deflection is 13 mils.

V. NUCLEAR DESIGN AND PERFORMANCE

A. INTRODUCTION

This section describes the nuclear analyses leading to the selection of the conceptual design of a 5-kwe reactor. Both the parametric analyses made to select a minimum weight design and the detailed analyses of the selected design are discussed.

B. PARAMETRIC STUDIES

Reactor parametric studies were performed to select a conceptual design which meets the objectives of minimum reactor plus shield weight and minimum reactor envelope. The studies were made using the timeshare computer program, ZIP.⁽¹¹⁾ The parameters which were varied are listed in Table 18.

The results obtained from varying the number of fuel elements are shown in Table 19. The active fuel length was held constant at 16 in. and the external reflector midplane thickness was held constant at 2.19 in. for these calculations. Reactivity was adjusted by varying the lattice pitch. In hexagonal geometry, there are certain total number of elements which pack into a circular core vessel better than others. This is illustrated by the results for a 73-element core in which the vessel diameter is greater than that of the 85-element core. For this reason, the 73-element core is 28 lb heavier than the 85-element core. By maintaining a constant fuel length and reactor power, the average linear rod power (kwt/ft) varies inversely with the number of fuel elements. The increase in kwt/ft with the decrease in number of elements results in higher fuel temperatures. The increased fuel temperature contributes to an increase in fuel growth and hydrogen loss. The 85-element core was selected over the lighter 55-element core because of the 40° F lower fuel temperature.

Variations in active fuel length were made over a range of 14 to 17 in. for an 85-element core. In these calculations, lattice pitch and external reflector thickness were varied to give the same end-of-life reactivity margin for all reactors. The results are shown in Table 20. As may be seen, the trade-off in reactor plus shield weight is small over the range. A 16-in. core was selected as a compromise between weight savings and reduced fuel temperature and fuel element stress.

TABLE 18
REACTOR PARAMETERS VARIED

Parameter	Range
Number of Fuel Elements	55 to 121
Active Fuel Length (in.)	14 to 17
Cladding Thickness (mils)	15 to 30
Interelement Spacing (mils)	15 to 25
Cone Half-Angle (degrees)	3 to 10
Minimum Reflector Thickness (in.)	0.75 to 5.25
Reactor Thermal Power (kwt)	70 to 100
Core ΔT ($^{\circ}F$)	60 to 90
Reactor Outlet Temperature ($^{\circ}F$)	1100 to 1250
EOL Reactivity Margin (\$)	0 to 2.50
Prepoison Loading (\$)	0 to 4.50

TABLE 19
NUMBER OF FUEL ELEMENTS TRADE STUDY

	55	73	85	121
Number of Elements	55	73	85	121
Lattice Pitch (in.)	1.233	1.095	1.026	0.893
Active Core Diameter (in.)	9.61	9.82	9.93	10.31
Internal Reflector Thickness (in.)	0.275	0.442	0.261	0.274
Core Vessel OD (in.)	10.28	10.82	10.57	10.98
Reactor Midplane Envelope (in.)	14.66	15.20	14.95	15.36
Relative Reactor Plus Shield Weight (lb)	-31	+28	*	+44
BOL H/Zr (δ -Fuel)	1.64	1.65	1.66	1.67
Cold BOL Radial Gap (mils)	7.0	6.1	5.8	5.3
EOL Nominal Peak-Average Fuel Temperature ($^{\circ}F$)	1326	1297	1286	1267

*Comparison Case

TABLE 20
FUEL LENGTH TRADE STUDY

Active Fuel Length (in.)	14	15	16	17
Lattice Pitch (in.)	1.094	1.057	1.026	0.999
Reflector Midplane Thickness (in.)	2.05	2.12	2.19	2.26
Core Vessel OD (in.)	11.26	10.88	10.57	10.30
Reactor Midplane Envelope (in.)	15.36	15.12	14.95	14.82
Relative Reactor Weight (lb)	-15	-9	*	+11
Relative Shield Weight (lb)	+14	+6	*	-4
Relative Reactor Plus Shield Weight (lb)	-1	-3	*	+7
EOL Nominal Peak-Average Fuel Temperature (° F)	1295	1290	1286	1283

*Comparison Case

The remainder of the parameters which were investigated proved to have a linear relationship in weight. For these parameters, weight coefficients were determined which are given in Table 21. These coefficients were used in the overall reactor thermoelectric system optimization studies.

C. NUCLEAR CHARACTERISTICS OF THE SELECTED DESIGN

From the results of the reactor parametric studies and the reactor-thermoelectric system optimization studies, the reactor design summarized in Table 22 was selected as the conceptual design. The reactor was designed for continuous operation for 5 years at 110 kw. The thermoelectric system power requirements to produce 5 kwe do not exceed 102 kw. This has the effect of providing additional end-of-life reactivity margin.

1. Reactivity and Reactivity Lifetime

The reactivity requirements for the design power level and the operating power level are shown in Table 23. The reactor nuclear characteristics are shown in Table 24. As may be seen, a cold, unpoisoned reactivity of \$7.74 is needed to provide the reactivity requirements for 5 years of operation at 110 kw

TABLE 21
SUMMARY OF PARAMETRIC STUDIES
WEIGHT SENSITIVE COEFFICIENTS

Parameter	Weight Coefficients		
	Reactor	Shield	Reactor + Shield
Cladding Thickness (lb/mil)	1.95	1.05	3.00
Interelement Spacing (lb/mil)	0.55	0.45	1.00
Cone Half Angle (lb/degree)	2.4	29.0	31.4
Minimum Reflector Thickness (lb/in.)	7.0	21.8	28.8
Reactor Thermal Power (lb/kwt)	0.41	1.38	1.79
Core ΔT (lb/ $^{\circ}$ F)	0.22	0.14	0.36
Reactor Outlet Temperature (lb/ $^{\circ}$ F)	0.096	0.066	0.162
EOL Reactivity Margin (lb/\$)	9.0	5.4	14.4
Prepoison Loading (lb/\$)	2.4	1.4	3.8

TABLE 22
REACTOR DESIGN SUMMARY

Thermal Power (kwt)	110
Number of Fuel Elements	85
Active Fuel Length (in.)	16
Lattice Pitch (in.)	1.025
Interelement Spacing (mils)	20
Cladding Thickness (mils)	25
Glass Thickness (mils)	2.5
Gap Thickness (mils)	6.0
Fuel Diameter (in.)	0.938
Active Core Diameter (in.)	9.920
Active Core Volume (liters)	20.3
Core Vessel OD (in.)	10.600
Vessel-Reflector Clearance (in.)	0.060
Reflector Midplane Thickness (in.)	2.386
Reactor Midplane Envelope (in.)	15.492
Taper Angle (degrees)	8.5
Segment Stroke (in.)	4.0
Movable Angle (degrees)	235
Fissile Mass (kg U^{235})	9.0
N_H in Fuel (10^{22} at./cc)	5.72

TABLE 23
REACTIVITY REQUIREMENTS
(\$)

	Design	Operating
Temperature Defect	2.18	2.09
Power Defect	0.04	0.03
Equilibrium Xe ¹³⁵	0.18	0.15
Hydrogen Redistribution	<u>0.10</u>	<u>0.09</u>
Total Recoverable Losses	2.50	2.36
U ²³⁵ Burnout	0.82	0.69
U ²³⁶ Buildup	0.31	0.26
U ²³⁸ Burnout	-0.02	-0.02
Gross Fission Products	0.92	0.78
Equilibrium Sm ¹⁴⁹	0.94	0.90
Hydrogen Loss	1.16	0.75
Residual Prepoison	<u>0.61</u>	<u>0.76</u>
Total Nonrecoverable Losses	4.74	4.12
Total Requirements	7.24	6.48

TABLE 24
NUCLEAR CHARACTERISTICS
(\$)

	Design	Operating
Cold, Unpoisoned Reactivity	7.74	7.74
Hot Prepoison Loading	3.00	3.00
Recoverable Losses	2.50	2.36
Hot, Poisoned Reactivity	2.24	2.38
Nonrecoverable Losses	4.74	4.12
Prepoison Burnout	2.39	2.24
EOL Reactivity Margin	0.50	1.26
Control Worth	7.74	7.74
BOL Shutdown Margin	3.00	3.00

6531-4615

Figure 45. Reactivity vs Time, 5-kwe Reactor, Design-To Conditions

with a \$0.50 end-of-life reactivity margin. At the operating power level, the EOL reactivity margin increases to \$1.26. The reactivity as a function of time at the design power level is shown in Figure 45. Also shown is the cold, shut-down reactivity with the control segments fully open.

2. Prepoison Selection and Loading

The nuclear parameters which influence the selection of a prepoison material are the burnout rate, the effect on the prompt temperature coefficient, and the specific worth (\$/gm). The nonnuclear parameters are the compatibility of the poison with other core materials, the availability and cost of the poison, and the method of depositing the poison in the core.

The reason for including a prepoison in the reactor is to reduce the control requirements by reducing the reactivity loss over life. The burnout of the prepoison helps to offset the losses from fuel depletion, fission product buildup, and hydrogen loss.

The poisons which have both nuclear and physical properties which make them suitable for use in SNAP reactors are samarium, gadolinium, gadolinium enriched in Gd^{155} , europium enriched in Eu^{151} , and erbium. The properties of these poisons are as follows:

a. Samarium

This is the only poison which has been used in operating SNAP reactors. Its burnout rate is suitable for 5-kwe reactor application. It has a positive effect on the prompt temperature coefficient but not as great as gadolinium. Natural samarium may be used because only one high cross section isotope, Sm^{149} , occurs naturally; therefore, the cost and availability are favorable. In previous SNAP reactor use, samarium oxide was mixed with the ceramic hydrogen barrier. Post-irradiation examination of fuel elements indicated some deterioration of the ceramic.

b. Gadolinium

The burnout rate of natural gadolinium is very similar to that of samarium. The naturally occurring material contains two high cross section isotopes, Gd^{155} and Gd^{157} . The effect on the prompt temperature coefficient of natural gadolinium

has not been measured, but calculations indicate that the effect is about the same as that of Gd^{155} which has been measured. Nonnuclear testing of gadolinium oxide vapor deposited on the fuel surface indicated that it is a very stable material.⁽¹²⁾ The availability and cost of natural gadolinium are favorable.

c. Gadolinium-155

The burnout rate of Gd^{155} is too slow for 5-kwe reactor application. It is more suitable for higher power level reactors. The availability and cost are not favorable. The other properties are similar to those of natural gadolinium.

d. Europium-151

The burnout rate of Eu^{151} is slow due to its low cross section and the buildup of the daughter product Eu^{152} . Eu^{151} has a strong negative effect on the prompt temperature coefficient which is the only reason for considering its use. It may be used to offset the positive temperature coefficient effect of samarium and gadolinium. Natural europium must be enriched in Eu^{151} because of the presence of Eu^{153} which is unsuitable as a burnable poison. Its availability and cost, therefore, are not favorable. Nonnuclear testing of europium oxide vapor deposited on the fuel surface indicated it to be an unstable material.

e. Erbium

The burnout rate of erbium is slow. The main function of erbium is to alter the temperature coefficient; therefore, its use is in conjunction with samarium or gadolinium. Natural erbium may be used since all isotopes except Er^{167} have negligible cross sections. The availability and cost of natural erbium are favorable. Erbium oxide is compatible with ZrH fuel.

Natural gadolinium oxide was selected for the 5-kwe reactor because of its chemical stability and because the required loading of \$3 did not reduce the temperature coefficient below an acceptable value.

A total loading of 13.25 gm of gadolinium oxide is required to give a hot worth of \$3. Assuming that the poison will be deposited on 15 in. out of the total fuel length of 16 in., the surface density is 0.547 mg/cm^2 . The calculated room temperature worth is \$3.73.

3. Control Worth

Reactivity control is provided by two sliding reflector segments. The movable segments subtend 235° of the reactor circumference and they open 4-in. windows about the vertical center of the core. The sliding segments move along the 8.5° taper angle rather than parallel to the core vessel surface. This also opens gaps between the reflector and core vessel, and between the sliding segment and fixed reflector which enhance the control worth.

A reliability analysis indicated that more than two segments did not yield a significant increase in reliability. Two segments were selected from the standpoint of simplicity and cost.

A total control worth of \$7.74 was determined from two-dimensional DOT⁽¹³⁾ calculations in R, Z geometry. In the DOT calculations, it was necessary to assume that the segments subtend the full 360° F. The worth obtained from the DOT calculations was multiplied by $235/360$ to obtain the worth of the 235° segments. Calculations were made with the segments closed, 2 in. open, and 4 in. open. The worth of the first 2-in. opening was \$3.95, and the worth from 2 to 4 in. open was \$3.79. This indicates that the control worth is very nearly linear over the 4-in. stroke.

4. Temperature Coefficients

Partial temperature coefficients of reactivity are shown in Table 25. The fuel coefficient is made up of the spectral, Doppler, and axial fuel expansion coefficients. The isothermal coefficient is the sum of the fuel, grid, and reflector coefficients. The isothermal coefficient is used to determine the reactivity loss in going from ambient to operating temperatures. The partial temperature coefficients are used to predict the kinetic behavior of the reactor.

The coefficients were determined using the k-difference method. The various effects which cause the reactivity to change with temperature, such as Doppler broadening of cross sections and thermal expansion of the different reactor components, were evaluated individually using the APC code.⁽¹⁴⁾ The calculations were performed at beginning-of-life conditions. As mentioned in Section V-C-2, the gadolinium prepoison loading has a positive effect on the spectral component

6531-4616

Figure 46. BOL Power Contours, Segments in Normal Position

TABLE 25
PARTIAL TEMPERATURE COEFFICIENTS
($^{\circ}$ F)

Spectral	+0.007
Doppler	-0.022
Fuel Axial Expansion	-0.051
Fuel Coefficient	-0.066
Inlet Grid	-0.060
Outlet Grid	-0.020
Reflector Heating	-0.034
Isothermal	-0.18

of the fuel coefficient. As the gadolinium burns out over life, the fuel coefficient becomes more negative. The effect of the burnout of gadolinium is partially offset by the buildup of Sm^{149} fission product. The fuel coefficient at end-of-life was evaluated to be -0.108°F .

5. Power Distributions

Two-dimensional power profiles were generated from the DOT calculations in R, Z geometry from which the control worth was obtained. The power profiles were generated for segments closed, 2 in. open, and 4 in. open. Figure 46 shows the profiles with segments 2 in. open which is representative of beginning-of-life conditions. Figure 47 shows the profiles with segments fully closed which is representative of end-of-life conditions.

In the R, Z geometry calculations, it was not possible to represent the fixed reflector between the two sliding segments. The segment opening in the calculations encompassed the full 360° of the reactor circumference. In order to account for the azimuthal variation in the power, a power distribution was synthesized from two R, Z calculations. The power distributions from the calculations with segments closed and 2 in. open were combined using a sinusoidal azimuthal variation. A timeshare computer program was written for this purpose. Figure 48 shows the resultant peak power relative to the core average for each fuel element in a quadrant. Due to symmetry, the power distribution is the same in all four quadrants.

6531-4517

Figure 47. EOL Power Contours, Segments Fully Closed

6531-4603

Figure 48. Peak Power Per Element Relative to Core Average

TABLE 26
COMPARISON OF CALCULATED AND EXPERIMENTAL
REACTIVITIES OF SNAP REACTORS

	Reactivity (\$)		Error
	Measured	Calculated	
Near Critical Reactivities			
S8DR-165 Elements Full Shims	0.16	1.35	1.19
SCA-4A-37 Elements 1.25-in. Reflector	-0.25	-0.73	-0.48
S8ER-211 Elements No Shims	-0.28	-0.88	-0.60
S8ER-186 Elements A Shims	0.14	0.40	0.26
S8ER-173 Elements A + B Shims	0.10	0.60	0.50
Average Error and Standard Deviation			0.17 ± 0.74
Supercritical Reactivities			
S8DR-169 Elements Full Shims	1.40	2.37	0.97
S8DR-211 Elements Full Shims	13.62	12.14	-1.48
SCA-4A-37 Elements 2.25-in. Reflector	6.55	5.33	-1.22
SCA-4A-37 Elements 3.25-in. Reflector	10.45	8.68	-1.77
S8ER-211 Elements Full Shims	10.09	7.36	-2.73
Average Error and Standard Deviation			-1.25 ± 1.36
Overall Average Error and Standard Deviation			-0.54 ± 1.28

6. Uncertainties and Shim Capabilities

The 28-group cross section library⁽¹⁵⁾ and transport theory method used in the analysis of the 5-kwe reactor were tested against several SNAP reactor experiments. The reactivity of a variety of core loadings and shim configurations was measured on SCA-4A, S8ER, and S8DR. Reactivities near critical and significantly supercritical reactivities were measured. Table 26 contains a comparison between calculation and experiment of ten of these configurations. In general, critical cores were calculated accurately and the large excess reactivities were underestimated. Part of the error in the supercritical reactivities may be due to the difficulty involved in their measurement. All of the supercritical reactivities were obtained from control drum calibrations and large variations in these calibrations were obtained due to interaction with neighboring drums. When these individual drum calibrations are added to obtain large excess reactivities, considerable uncertainty may result. Most of the near critical reactivities were obtained from inverse multiplication with all control drums full in.

The estimated uncertainty in the reactivity requirements for 5 years of full power operation are shown in Table 27. The overall uncertainty was obtained by adding the burnup, fission product, and prepoison burnout uncertainties and combining their total by root sum square with the remaining uncertainties. This method was used because it was felt that the long-term burnup effects are not independent of each other but are independent of the remaining uncertainties.

The uncertainty in control worth is based on calculations and measurements of sliding reflector worth on SCA-4A.⁽¹⁶⁾ The R, Z geometry calculational method underpredicted by 18% the worth of two 90° segments open 3.5 in. on SCA-4A. This means that the 5-kwe reactor control worth is underpredicted by about \$1.4.

Because of the uncertainty in reactivity predictions, the 5-kwe reactor will contain provisions for reactivity shimming. To increase reactivity, provision is made for adding beryllium shims to the fixed portion of the reflector. To decrease reactivity, provision is made for increasing the gap between the reflector and core vessel by adding shims beneath reflector bearing pads.

TABLE 27
REACTIVITY LIFETIME UNCERTAINTIES

Item	Value (\$)	Uncertainty	
		(%)	(\$)
Temperature Defect	2.09	20	0.42
Power Defect	0.03	20	0.01
Xe ¹³⁵	0.15	10	0.02
Hydrogen Redistribution	<u>0.09</u>	50	0.05
Total Recoverable	2.36		
Burnup and Fission Products	1.71	15	0.26
Sm ¹⁴⁹	0.90	15	0.14
Hydrogen Loss	0.75	50	0.38
Residual Prepoison	<u>0.76</u>		
Prepoison Burnout		20	0.45
Total Nonrecoverable	<u>4.12</u>		
Total Requirements	6.48		0.92

D. METHODS OF NUCLEAR ANALYSIS USED

1. Nuclear Codes

SNAP reactors are quite small from a nuclear standpoint. In order to accurately predict the nuclear performance of these reactors, transport theory (not diffusion theory) must be used. The one- and two-dimensional codes that have been developed for this purpose are as follows:

a. APC and APC II

APC and APC II are both buckling (B^2) or leakage (DB^2) iteration codes for reactor calculations based on the one-dimensional transport theory code, ANISN.^(17,18) In APC,⁽¹⁴⁾ a uniform composition (single material) must be specified for all core zones. In APC II, the composition in core zones may be different and options for flux and importance weighting of the different compositions are provided.

In both codes, two complete ANISN problems are specified. These two problems make up a case. The two problems are processed alternately transferring B^2 or DB^2 from each to the other until the results are mutually consistent. Termination of a case is controlled by convergence criterion on k_{eff} and leakage or by limiting the number of times the problems are processed (number of passes). Maximum execution time (CPU time) may also be used to terminate each problem or the case.

Although a B^2 or DB^2 iteration case must be a k calculation, APC and APC II may both be used to run single ANISN type problems. These codes are the main production tools for SNAP nuclear analysis, since explicit two-dimensional calculations are too expensive and time consuming to use for parametric studies.

b. DOT

DOT⁽¹³⁾ is a Fortran IV program which solves the linear, energy-dependent, Boltzmann neutron and/or photon transport equation for two-dimensional R-Z, X-Y, and R- θ geometries. The gradient or convection term in the Boltzmann equation is approximated by a finite difference technique known as discrete ordinates or Carlson's S_n method. The inscatter integral is approximated by expanding the differential cross-section in a Legendre series which allows the integral to be computed by quadrature. DOT will solve forward or adjoint, homogeneous or inhomogeneous problems. The inhomogeneous problems may have a volume-distributed source or a specified angular flux at the right or top boundaries; fissions may be included for a subcritical system. Homogeneous (eigenvalue) problems will determine the following: (1) static multiplication factor, " k "; (2) time adsorption, " $\text{Rossi } \alpha$ "; (3) concentration for a specified k , and (4) zone thickness for a specified k .

DOT is a successor to the Los Alamos codes TDC, DDK, and DDF. The latest version of this code (from ORNL) is DOT III. Cross sections can be input on cards from an ANISN-type, nuclide-organized data set, or from a special group-organized data set which is valuable for very large cross section sets.

c. COMPOZ

COMPOZ⁽¹⁹⁾ is a transport theory perturbation code designed to be used in conjunction with the multigroup transport code, APC. The code calculates the reactivity worth of cross section perturbations of materials which are included in the APC case or for materials from a library tape. The perturbation which is made to the cross section is equal to the value of the cross section. This makes it unnecessary to input any specific perturbation of a cross section or number density.

The majority of the input to a COMPOZ case is obtained from data sets created by APC; therefore, most of the COMPOZ input is control parameters. To run a COMPOZ case, it is necessary to have previously run both direct and adjoint APC cases.* The data set created by the direct APC case contains fluxes, cross sections, number densities, and geometry data. The data set from the adjoint APC case contains only the adjoint fluxes.

The COMPOZ perturbation worths in units of $\Delta K/K$ per atom/cm · barn are given for each type of reaction (capture, fission, leakage, etc.) by interval and by zone. Integrals of the worth over a given range of intervals may be obtained by group as well as the total over all groups. The worth in dollars of the materials appearing in the APC case is given for all zones in which they appear. In this instance, the worth is calculated for the total amount of the material appearing in the zone.

COMPOZ calculates the transverse leakage worth of a material which the effect a perturbation in the total cross section of that material has on the DB^2 term. Group-dependent DB^2 terms are only given for core zones by APC; therefore, the transverse leakage worth for noncore zones is inaccurate.

In addition to material worths, COMPOZ also calculates the effective delayed neutron fraction and neutron lifetime. The option is also available for performing a group-collapse to obtain importance weighted few-group cross sections.

*APC is designed so that two dimensions (R and Z) can be calculated in one run.

d. BUFRN

This code solves the nuclear reactor burnup problem for SNAP reactors with a quasi-two-dimensional approach. By coupling the APC-II buckling iteration code and the NUCY burnup equations,⁽²⁰⁾ axial and radial nuclide burnup effects are both taken into account. Therefore, a single nuclear case may be run to simultaneously obtain nuclear results for the beginning-of-life (BOL) and end-of-life (EOL) compositions of a SNAP reactor.

This code is useful for determining the impact of burnup on any reactor. However, it is a one-shot burnup code; that is, it does not contain fuel management provisions.

e. ZIP

ZIP⁽¹¹⁾ is a timeshare program for performing SNAP reactor parametric studies at nominal cost. ZIP calculates reactivity, reactivity requirements, and control worth for a specified reactor design. At the user's option, it searches on lattice pitch to determine a reactor which has a specified end-of-life (EOL) reactivity margin. ZIP also calculates hydrogen loss, fuel swelling, shadow shield requirements, and reactor and shield weights.

The program is written in FORTRAN for use on the Honeywell 440 computer. Input data are contained in "Z" files numbered from Z1 through Z40. A maximum of four cases may be run simultaneously. The output data are written in a file named "OUTPUT" for later listing. Some of the pertinent information is printed on line so that a rapid evaluation of the cases being run can be made.

f. APCTC

This code is written to facilitate the analysis of nonisothermal temperature reactivity coefficient measurements such as those that were made on the SNAP-TRAN reactor.⁽⁸⁾ Earlier studies which utilized isothermal calculations did not agree very well with the experimental measurements. Since the temperature distribution of the SNAPTRAN reactor was quite nonuniform during many of the temperature coefficient measurements, a nuclear code was needed which would handle these nonuniformities.

APCTC calculates the spatial distribution of the fuel's temperature. This information is combined with temperature-dependent cell calculations from APC-II and adjoint fluxes from COMPOZ to determine temperature coefficients for the reactor. This calculational procedure produced temperature coefficients which were about 40% higher than temperature coefficients that were calculated with flat temperature distributions. Comparisons with experimental measurements were considerably improved with this procedure.

2. Cross Sections

A multigroup library of 28 groups is presently used for the nuclear analysis of SNAP reactors. The energy breakpoints used in this library were designed to provide detail both at high energies, in order to accurately calculate the significant fast leakage from SNAP cores, and around the thermal peak, in order to accurately calculate multiplication factors, prepoison reactivities, and temperature effects. This library provides for downscattering to 17 groups and upscattering to 12 groups. Half of the energy groups have been placed below 3 ev since much of the flux in these reactors is near thermal energies and the median fission energy is about 0.2 ev.

While the flux spectrum in the SNAP reactors is essentially thermal, it cannot be accurately characterized by a typical Maxwellian thermal-neutron energy spectrum. This is a consequence of the zirconium-bound hydrogen in the uranium zirconium-hydride fuel matrix. Using the neutron-scattering kernel based on the SUMMIT II code, a 28-group set of cross sections was prepared for use in calculations of the SNAP reactor cores. A calculated neutron spectrum for the S8DR reactor core operating at 68° F had a peak in its thermal flux spectrum at 0.1 ev.

The SNAP library has been periodically updated in order to utilize the best data possible.

The 28-group library contains about 100 files of cross sections. These include all of the elements necessary for SNAP reactor analysis. The library also contains cross sections for various reactor operating temperatures and as a function of Legendre coefficient expansions (modified P_0 , P_1 , P_2 , and P_3). A complete description of this library is given in Reference 15.

3. Geometric Modeling

SNAP reactors are generally quite small. The active core is a right circular cylinder about 1 ft high, with a diameter of about 9 to 11 in. Since these reactors have a substantial leakage of neutrons, they can be adequately controlled by movable radial reflectors. The fuel elements for these reactors are clad rods of hydrided uranium-zirconium with an OD of 1/2 to 1 in. The coolant is a sodium-potassium alloy.

Although all of the geometric detail of SNAP reactors could be represented in a three-dimensional Monte Carlo calculation, this is not generally desirable from an engineering standpoint. In order to rapidly (and inexpensively) assess the impact of various design modifications, one-dimensional calculations are used extensively. Some two-dimensional calculations are used, but only for the calculation of nuclear parameters that cannot be adequately studied with 1-D methods (such as the worth of radial reflector drums).

The calculation of these reactors starts with an equivalent cylindrical (Wigner-Seitz) fuel cell. This cell typically contains a central fuel rod surrounded by annuli of cladding and coolant. Then cross sections are flux-weighted to spatially averaged cross sections for the active core.

These reactors are most commonly analyzed in a quasi-two-dimensional manner by iterating radial and axial one-dimensional, multigroup transport calculations (APC code). The radial geometrical model consists of the active core surrounded by annuli of the internal reflector region (≈ 1 cm thick), the reactor vessel (0.1 to 0.2 cm thick), and the outer reflectors which contain the control drums (5 to 14 cm thick depending upon the amount of reflection desired). The axial geometrical model contains the active core (30 to 35 cm high), upper and lower grid plates (each about 0.5 cm thick), and upper and lower coolant plena that contain 6 to 8 cm of NaK.

The iterative APC technique is generally adequate for producing material sensitivity coefficients and studying the performance of the active core.^(24,25) However, the accurate calculation of quantities such as reflector control drum worth frequently require two-dimensional transport calculations (using the X-Y version of DOT).

BLANK

VI. REACTOR PERFORMANCE MARGINS

A. INTRODUCTION

In the design of any mechanical, electrical, or nuclear system, it is necessary to build into the system sufficient margin to ensure the design objectives will be achieved with a high degree of probability. On the other hand, it is usually desirable to limit the amount of margin in order to avoid excessive cost, weight, or other undesirable characteristics. The overall 5-kwe Reactor Thermoelectric System margins and their allocation to each subsystem, of which the reactor is one, are described in Reference 21.

From past experience as well as the results of the studies of Reference 21, it is particularly important to examine in detail the reactor performance margins, since the reactor is the most complex component in the system, and has a large number of potential modes of performance degradation.

The primary way the reactor could cause the system to fail to achieve its design goal of 5 kwe for 5 years is by becoming subcritical before the mission is completed. There are a number of ways that this could happen; these may be grouped into three broad categories. First, the nuclear calculations of the required BOL reactivity, or the reactivity requirements over life could be in error, and reactivity could become zero before the 5-year mission is completed. Second, the stress level in the fuel element cladding or glass hydrogen barrier could exceed the yield point and hydrogen leakage could, as a consequence, be significantly larger than predicted. This also could result in insufficient reactivity to operate for 5 years at 5 kwe. Third, the reflector control system could fail, thereby possibly causing the reactivity addition capability of the control system to be inadequate to achieve the mission objectives. In order for the reactor to have a high probability of achieving its design goals, sufficient margin must exist in each of these three areas. This was a design objective in the conceptual design of the reactor presented in this report.

B. REACTIVITY LIFETIME MARGINS

The reactor is designed for continuous 5-year operation at 110 kwt, 1200° F outlet temperature, and 100° F core temperature rise. The reactivity margin under these conditions is \$0.50 which is sufficient for 1 year additional operation.

Figure 49. 5-kwe System, Extended Life Characteristics

Figure 50. Reactor Lifetime Capability with One Stuck Control Segment

In actual operation, the thermoelectric system demands are less severe than the design conditions. The reactor power level from beginning of life (BOL) to end of life (EOL) varies from 91 to 99 kw. The outlet temperature varies from 1135°F at BOL to 1200°F at EOL and the core temperature rise varies from 87°F at BOL to 95°F at EOL. There is sufficient reactivity at the actual (most probable) operating conditions to provide an additional 22 months operation.

After approximately 8 years at full power, it is possible to use temperature drift-down for continued operation at reduced power. This is illustrated in Figure 49.

An estimate of uncertainties in reactivity lifetime was given in Table 27. A total uncertainty of \$0.92 in the 5-year reactivity requirement is shown. The reactivity margin at operating conditions is \$1.26.

In the event of failure of one control segment, the reactor may still attain a significant fraction of its design life. Figure 50 shows the reactor life attainable as a function of the time at which one control segment sticks at its operating position. If one segment sticks in the full-out position at startup, it is still possible to attain about 4.3 years operation with the remaining segment. The minimum time at which one segment can stick and the reactor still achieve its design life is 1.6 years.

C. FUEL ELEMENT PERFORMANCE MARGINS

The reactor fuel element is manufactured with a 12-mil (diametral) gap inside the cladding to allow for fuel swelling, bowing, manufacturing tolerances, etc. Based upon the design operating conditions and the current fuel growth data, the predicted fuel growth in the peak rated element⁽²²⁾ is 6.3 mils (diametral), 4.7 mils at offset with an additional 1.6 mils occurring during the 5-year lifetime. This allows a 5.7-mil margin for fuel segment bowing and swelling uncertainty.

The hydrogen retention capabilities of the SCB (ceramic) hydrogen barrier, attached to the internal surface of the cladding, are described by the dimensionless parameter, A_D . This parameter defines the fractional internal surface

area of the cladding which is effectively bare metal. The leak rate is, typically, made up of about 75% due to leakage through defects in the glass (essentially leakage through bare metal) and 25% due to leakage through the glass and the metal, so the magnitude and timewise variation of A_D pretty much determines the hydrogen retention characteristics of a fuel element. Evaluation of available test data⁽²²⁾ indicates that A_D for the 5-kwe system will be of the order of 0.15% and will remain constant throughout the life of the core. The integral hydrogen loss over the design life of the reactor core results in a reactivity loss of \$0.75.

The fuel element stress study is discussed in Section IV-B. The margin of safety for the cladding stresses is always greater than one, except for cases where the element manufacturing bow was assumed to be much greater than the expected 6 mils. Even for an assumed 10-mil manufacturing bow the margin is based on that stress at which hydrogen leakage from the fuel element increases significantly.

D. AUTOMATIC CONTROL SYSTEM RELIABILITY

The purpose of the automatic control system for the flight reactor is to start up the reactor in orbit and maintain reactor power so as to generate electrical power at 5 kw within narrow limits for 5 years.

Estimates of the reliability of the automatic control system can be made from the preliminary and conceptual designs of this system. More precise calculation of reliability must wait for the final design; reliability analysis would serve as a tool in developing the final design. The estimates derived here are approximations of what may be achieved.

The criterion of failure taken for this analysis is that if any component failure results in the inability of the control system to start up the reactor and maintain the thermoelectric power at 5 kw for 5 years, the system has failed. This criterion is applied only to the automatic controller subsystem and does not include the segment drive system and actuators or the power supplies for the electronic components and actuators.

The automatic controller subsystem is shown in Figure 51, divided into functional blocks. During startup, pulses from the oscillators are passed by

6531 5534

Figure 51. Functional Block
Diagram Automatic
Controller

BLANK

the startup blocks, subject to conditions established by the position, temperature, and current indicators, to the delay block and then to the alternator, where the pulses are alternately passed to one or the other of the phase sequencers. The active phase sequencer turns on five pairs of power transistors in sequence to release the actuator brake and step the actuator through a cycle, resetting the brake at the end of the cycle.

Preliminary and conceptual designs have established approximate active element part counts as indicated on the figure. Only integrated circuits and power transistors have been identified as necessary active elements at this stage.

Representative component failure rates at 25°C of 0.05 failures per million hours for the integrated circuits and 1.32 failures per million hours for the power transistors have been used. These rates do not include adjustments that should be made to reflect differences in temperature, humidity, and pressure from the nominal, the effects of launch and orbit insertion, derating in the design, and degradation due to radiation damage.

The failure probabilities for the startup phase and the operating phase were calculated separately from the individual functional block failure probabilities. These, in turn, were derived from the quantity of components and the components failure rate.

For startup, the subsystem failure probability is the sum of probabilities for disabling failures:

$$Q(\text{startup}) = Q_{\text{osc}}^2 + Q_{\text{start}}^2 + Q_{\text{pos}} + \frac{1}{2} Q_{\text{temp}} + \frac{1}{2} Q_{\text{cur}} \\ + \frac{1}{2} Q_{\text{delay}} + \frac{1}{2} Q_{\text{alt}} + Q_{\text{ph}}^2 + Q_{\text{ps}}^2 + 2Q_{\text{ph}}Q_{\text{ps}}$$

where obvious abbreviations of the block names have been used to identify the failure probabilities.

Since both the oscillators and the startup blocks are in parallel and fully interconnected, both oscillators must fail or both startup blocks must fail in order to produce a disabling failure. The delay and alternator blocks can fail

in such a manner that the startup can still proceed. This is reflected by the factor of one-half for these probabilities. The phase sequencer and power switch blocks form parallel action paths but are not fully interconnected. It is possible to complete the startup phase using only one segment drive system. This is represented by the final three terms.

For the operating phase, the sum of failure probabilities is:

$$Q(\text{operating}) = Q_{\text{osc}}^2 + Q_{\text{op}}^2 + Q_{\text{cur}} + \frac{1}{2} Q_{\text{delay}} + \frac{1}{2} Q_{\text{alt}} + 2Q_{\text{ph}} + 2Q_{\text{ps}}$$

Changes from the startup failure probability are due to elimination of some of the functional blocks and the requirement that (except very near end of life) both segments must be driven correctly to satisfy the power requirement.

The block failure rates (per hour) are:

$$\begin{aligned} Q_{\text{osc}} &= 5 \times 0.05 \times 10^{-6} = 0.25 \times 10^{-6} \\ Q_{\text{start}} &= 8 \times 0.05 \times 10^{-6} = 0.40 \times 10^{-6} \\ Q_{\text{op}} &= 6 \times 0.05 \times 10^{-6} = 0.30 \times 10^{-6} \\ Q_{\text{pos}} &= 5 \times 0.05 \times 10^{-6} = 0.25 \times 10^{-6} \\ Q_{\text{temp}} &= 2 \times 0.05 \times 10^{-6} = 0.10 \times 10^{-6} \\ Q_{\text{cur}} &= 4 \times 0.05 \times 10^{-6} = 0.20 \times 10^{-6} \\ Q_{\text{delay}} &= 2 \times 0.05 \times 10^{-6} = 0.10 \times 10^{-6} \\ Q_{\text{alt}} &= 3 \times 0.05 \times 10^{-6} = 0.15 \times 10^{-6} \\ Q_{\text{ph}} &= 6 \times 0.05 \times 10^{-6} = 0.30 \times 10^{-6} \\ Q_{\text{ps}} &= 5 \times (1.32 \times 10^{-6})^2 = 8.7 \times 10^{-12} \end{aligned}$$

The last failure rate involves a square since two power transistors are used in parallel for each of the current switches. Using these failure rates, the failure probability for the startup phase (3 hours) is 1.6×10^{-6} while for the operating phase (44,000 hours) it is 41×10^{-3} . The corresponding success probabilities are 0.999998 for startup and 0.959 for full life operation. The phase sequencer blocks contribute over half of the failure probability while the current and alternator blocks account for most of the remainder. Improvements in design should concentrate on these functions with the aim to increase the full life reliability.

REFERENCES

1. R. J. Tuttle, "Scram Controller and Automatic Controller, 5-kwe Reactor Thermoelectric System," AI-AEC-13073, May 25, 1973
2. L. D. Felten and E. Moody, "Fuel Element and Core Bundle, 5-kwe Reactor Thermoelectric System," AI-AEC-13077, June 22, 1973
3. J. D. Sutherland, "SNAP 10A Reactor Structure," NAA-SR-9899, August 1, 1964
4. J. R. Licht, "Design of 2 DRM Reactor Assembly," NAA-SR-TDR-11137, March 9, 1965
5. P. H. Horton, "ZrH Reactor Control Reflector Systems Summary Report," AI-AEC-13078, June 1973
6. A. H. Lillie and V. L. Rooney, Jr., "Findings of the SNAP 8 Developmental Reactor (S8DR) Post-Test Examination," AI-AEC-13003, June 30, 1971
7. B. J. Ostermier, "ZrH Reactor Core Hydraulic Studies Summary Report," AI-AEC-13087, June 1973
8. O. G. Feil, "Vane Systems for Very Wide Angle Diffusers," J. Basic Engineering, V. 86, p. 759, December 1964
9. L. R. Reneau, J. P. Johnston, and S. J. Kline, "Performance and Design of Straight, Two-Dimensional Diffusers," J. Basic Engineering, V. 89, p. 141, March 1967
10. E. Moody, "GEOM, A Thermal-Hydraulic-Stress Code," AI-AEC-13065, May 10, 1973
11. H. Rood, "Selected Computer Codes and Libraries, Volume I. ZIP, A Timeshare Program for SNAP Reactor Parametric Studies," AI-AEC-13076, Vol. I, June 21, 1973
12. R. VanHouten, "Burnable Poison Deposition on ZrH Fuel - Process Development and Evaluation," AI-AEC-13085, June 1973
13. F. R. Mynatt, "A User's Manual for DOT - A Two-Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering," K-1694, January 1967
14. D. S. Bost, "Selected Computer Codes and Libraries, Volume II. APC - A Multigroup Transport Buckling Iteration Code," AI-AEC-13076, Vol. II, May 2, 1973

15. J. M. Otter, "Selected Computer Codes and Libraries, Volume IV. A 28-Group Cross Section Library for SNAP Reactor Analysis," AI-AEC-13076, Vol. IV, May 31, 1973
16. D. S. Bost, "Control Worth of Sliding Reflectors for ZrH Reactors," AI-AEC-13086, June 1973
17. Ward W. Engle, Jr., "A User's Manual for ANISN - A One-Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering," K-1693, March 30, 1967
18. M. A. Boling and W. A. Rhoades, "ANISN/DTF II," AI-66-Memo-171, 1966
19. V. S. Lekach, "Selected Computer Codes and Libraries, Volume III. COMPOZ - A Multigroup Transport Perturbation Code," AI-AEC-13076, Vol. III, May 22, 1973
20. D. R. Vondy, "Development of a General Method of Explicit Solution of the Nuclide Chain Equations for Digital Machine Calculations," ORNL-TM-361, October 17, 1962
21. V. S. Lekach and H. Rood, "System Statistical Reliability Model and Analysis," AI-AEC-13098, June 8, 1973
22. A. H. Lillie, "ZrH Fuel Element Performance Characteristics," AI-AEC-13084, June 19, 1973
23. J. Bell, "Summit, An IBM-7090 Program for the Computations of Crystalline Scattering Kernals," GA-2492, February 1, 1962
24. L. D. Swenson, "SNAP 8 Development Reactor Nuclear Analysis," AI-AEC-12864, October 31, 1969
25. B. J. Dray, et al, "Reactivity Effects of Gadolinium-155 on a SNAP Reactor," AI-AEC-12909, April 15, 1970

APPENDIX. SELECTED LAYOUT DRAWINGS

R-N652200001	Reactor Layout - Conceptual, Rev. A, March 26, 1973
R-N652210001	Core Layout - Conceptual, Rev. A, March 26, 1973
R-N652240031	Fuel Element Layout - Conceptual Rev. NC, October 9, 1972
R-N652210002	Internal Reflector Layout - Conceptual, Rev. A, March 26, 1973
R-N652220032	Control Reflector and Drive Layout - Conceptual, Rev. B, February 26, 1973
R-N652220095	Reflector Drive Actuator - Conceptual, Rev. NC, October 12, 1972
R-N652220033	Ground Test Backup Shutdown Relatch Mechanism - Conceptual, Rev. A, February 26, 1973

BLANK

BLANK

BLANK

REVISIONS			
REV	DESCRIPTION	DATE	APPROVED
1	DESCRIPTION R0 52702		

NOTES: UNLESS OTHERWISE SPECIFIED ON THIS DRAWING

- FUEL O.D. TO BE GROUND TO A DIAMETER WHICH RESULTS IN A DIAMETRICAL GAP OF .011 TO .013 BETWEEN FUEL AND COATING.
- NEUTRAL ELEMENTS ARE THE SAME AS FINNED ELEMENTS EXCEPT FOR DELETION OF FINS.
- CORE TO HAVE A TOTAL OF 85 ELEMENTS: 31-NEUTRAL, 27-LEFT FIN, 27-RIGHT FIN. CORE PITCH 1.025±.0005 OR ELEMENT SPACING OF .020 (NOM.).
- FUEL ELEMENT TO MEET REQ'MENTS OF 55652N240001.
- REFERENCE DOCUMENTS:
 - RTI-652-240-014 FUEL ELEM STRESS
 - RTI-653-240-027 FUEL SWELLING THEORETICAL MODEL
 - RTI-652-240-011 FUEL ELEM. GAP SELECT.

COLD END

HOT END

IN-CHANNEL T/C INST'L N
(6 ELEMENTS, SEE R-N652210001)

AI-AEC-13075
137

DESIGN REVIEW		DATE
PREPARED BY	APPROVED	
DESIGNED BY	DATE	
CHECKED BY	DATE	
REVIEWED BY	DATE	
FINAL & NEW	DATE	
STRESS	DATE	
WELDING	DATE	
MECHANICAL	DATE	
FINAL SIGN.	DATE	

DESIGNED BY	J. RACHA	P. 72
CHECKED BY		
REVIEWED BY		
DATE		
SCALE		
SHEET		
FUEL ELEMENT L/O 5 KWE T.E. SYSTEM - CONCEPTUAL -		
CODE IDENT NO.	09974	R-N652240031

BLANK

There was no Page 139 located in original document.

There was no Page 140 located in original document.

There was no Page 141 located in original document.

There was no Page 142 located in original document.

BLANK