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SNAP SYSTEMS CAPABILITIES
VOLUME 2
STUDY INTRODUCTION, REACTORS, SHIELDING
(Title Unclassified)

AEC Research and Development Report

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M-3679 (41st Ed.)

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SNAP SYSTEMS CAPABILITIES

VOLUME 2

STUDY INTRODUCTION, REACTORS, SHIELDING

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II. STUDY INTRODUCTION

The U.S. Atomic Energy Commission undertook the first program to develop nuclear reactor electrical power supplies for use in spacecraft in January 1957. Three weeks following the USSR launch of Sputnik I in October 1957, the first reactor critical assembly in this space power program began operation. In the next seven years, the SNAP Program produced a new technology of uranium-zirconium hydride fueled compact reactors, thermoelectric and mercury Rankine power conversion systems, and the associated minimum-weight shields, pumps, controllers, etc., for spacecraft nuclear electric power. Test flight of a low power SNAP 10A unit in early 1965 has demonstrated major portions of this technology.

Since October 1957, the United States space program has been strongly oriented toward manned and unmanned experiments in space characterized by low level electrical power requirements. Wherever possible, reliability for such spacecraft has been gained through the use of conservative technological methods. New goals for the space program are currently developing; the unmanned operational satellites designed for the scientific utilization of space, and the post-Apollo manned or multi-manned spacecraft for longer-term exploration of man's role in space. These new goals require electrical power in the multi-kilowatt range. Initial investigations by government and industry show that the timely availability of nuclear reactor electrical power technology greatly enhances spacecraft utility and capability for such missions. This technology makes possible manned and unmanned space activities not otherwise achievable.

Developers of nuclear reactor space power technology must, therefore, acquaint space program planners with the qualities and capabilities of nuclear systems. Such information must be in sufficient detail to provide a firm basis for R & D planning by development agencies. The data must establish performance and reliability specifications.

This report provides a basic document which compiles nuclear reactor space power system capabilities. The report details the development status of each component (Section III – Reactors; Section IV – Shielding; and Section V – Power Conversion Systems) and examines the performance of complete systems and their variations (Section VI).

The SNAP reactors currently in development and their future variations are described in a fashion which emphasizes a consistent view of the temperature and lifetime capabilities of uranium-zirconium hydride fuel. It will be seen that only foreseeable performance is used in proposed reactors and no process or device requiring technical "breakthroughs" is defined. This emphasis on consistent performance specifications and available technological status is a fundamental aim in the entire report. For example, weights of components and structures, pump performance, emissivity values, thermoelectric and dynamic power conversion efficiency, etc., are extrapolated from test data where possible and common values used for all systems described.

Beginning with SNAP Reactors (Section III), component modifications for manned vs unmanned spacecraft are identified. In Shielding (Section IV), uniform shielding analysis criteria are established, and the influence of system variations on shield design is analyzed, both in parametric form and through specific design examples. In SNAP Power Conversion Systems (Section V), the separation of manned vs unmanned systems is continued with inclusion of reliability, maintainability, restart, redundancy, etc., specifications. These specifications are then uniformly applied in the remaining sections on complete power systems.

III. SNAP REACTORS

Zirconium-uranium alloy hydride reactor development during the past seven years has been highlighted by the successful operation of three experimental reactors. The SNAP Experimental Reactor (SER) achieved criticality in September 1959, the SNAP 2 Developmental Reactor (S2DR) in April 1961, and the SNAP 8 Experimental Reactor (S8ER) in May 1963. Figure III-1 highlights the experience with these reactors, and with the SNAP 10A flight system. The operation of these reactors at design power levels and temperatures for extended periods of time was preceded by a major zirconium-hydride fuel and barrier development program.

This base of technology has lead to a series of reactors which are currently in varying stages of design, fabrication, development, and qualification. For purposes of determining the development status of these reactors, they can be categorized as follows:

Present Capability (ZrH reactors currently being developed)

1) SNAP 10A (flight tested)
2) Interim SNAP 10A/2 ("shelf" design, available for ground system testing and/or qualification)
3) SNAP 8 (being made available for ground testing and qualification)

Interim Development (ZrH reactors which could be assembled from available technology and used in development and subsequent qualification of mission-oriented systems with minimum development cost and time)

1) SNAP 10A/2 Upgraded
2) SNAP 10B Basic
3) SNAP 8 Manrated
4) SNAP 8 Upgraded

Future Development (ZrH reactors which push existing technology toward its limits and which may require substantial new development programs)

1) SNAP 10B Advanced
2) Advanced ZrH.

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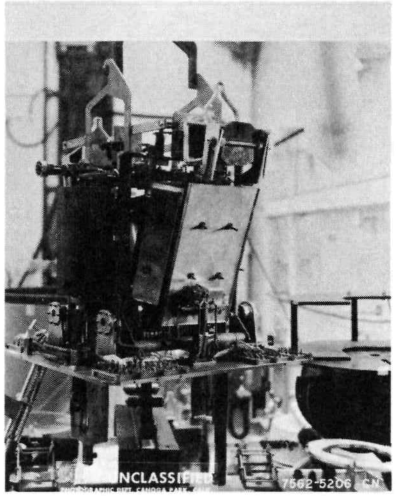

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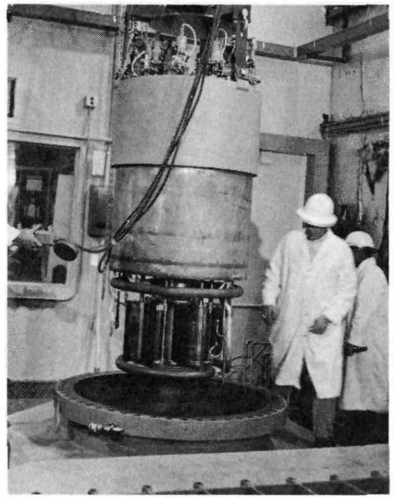

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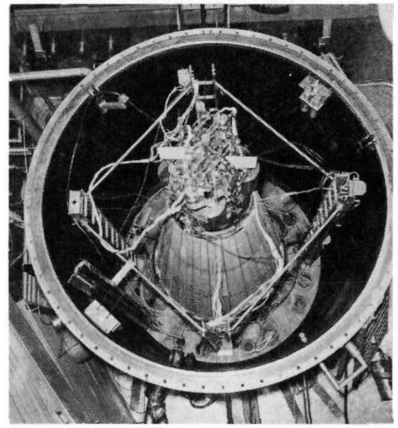

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	SNAP EXPERIMENTAL REACTOR (SER)	SNAP DEVELOPMENTAL REACTOR (SDR)	SNAP 8 EXPERIMENTAL REACTOR (S8ER)	SNAP 10A FLIGHT SYSTEM	
				GROUND TEST (FS-3)	FLIGHT TEST (FS-4)
CRITICAL	SEPTEMBER 1959	APRIL 1961	MAY 1963	JANUARY 1965	APRIL 1965
SHUTDOWN	DECEMBER 1960	DECEMBER 1962	APRIL 1965	OPERATING	MAY 1965
THERMAL POWER (kw)	50	65	600	38	43
THERMAL ENERGY (kw-hr)	225,000	273,000	5,100,000	230,000*	41,000
ELECTRIC POWER (watts)	–	–	–	400	560
ELECTRIC ENERGY (kw-hr)	–	–	–	2534*	574
TIME AT POWER AND TEMPERATURE	1800 hr AT 1200°F 3500 hr ABOVE 900°F	2800 hr AT 1200°F 7700 hr ABOVE 900°F	1 yr AT 1300°F 400 TO 600 kwt	257 days*	43 days

*AS OF OCTOBER 6, 1965

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Figure III-1. Operating Highlights

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While only the first three items listed above actually constitute firm reactor designs, all nine may be embodied in design examples for purposes of comparison and evaluation. This has been done in the following sections. In most cases, the designs presented were dictated by logical progression requirements based on existing hardware. In the more advanced cases, however, notably SNAP 10B Advanced and Advanced ZrH, the design points presented should be considered tentative rather than optimum. While based largely on logical progression requirements, the selected design points contain an element of engineering judgment since they are based on calculations of a preliminary nature.

The principal design features of these nine reactors, including four variations of the SNAP 10B Advanced, are summarized for easy comparison in Table III-1. A breakdown of reactor weights is given in Table III-2. In the case of advanced reactors (i.e., 10B Advanced, Upgraded 8, and Advanced Zirconium Hydride) no firm physical design presently exists. Weights given are estimates based on dimensional scaling of weights for presently established cores, components, etc.

Arranged in order of ascending power capabilities, the purpose of the following sections is to describe the range of operating parameters appropriate to each reactor, and to indicate the rationale and purpose of each concept in providing a continuous spectrum of reactor capabilities to power levels of several thermal megawatts. Insofar as is practical, discussion in this section will be restricted to the reactor proper, including reflector assemblies, control drive mechanisms, and other elements directly associated with the reactor.

A. SNAP 10A

1. Description

The SNAP 10A reactor is designed to produce 39.5 kwt for 1 yr in orbit. Figure III-2 shows the important components of SNAP 10A. The reactor core is fueled by 37 uranium-zirconium alloy fuel elements which are clad in Hastelloy N. The zirconium is hydrided to an N_H of 6.35 (6.35×10^{22} atoms H per cm^3) and thus the fuel elements also contain the moderator. The uranium is fully enriched and comprises 10% by weight of the fuel. The fuel elements are 1.25 in. in diameter (OD of clad) with an active fuel length of 12.25 in. The cladding is 15 mils thick and is coated on the inside with a 3-mil ceramic barrier to reduce hydrogen leakage. The coating contains a small quantity (8 mg per in.

TABLE III-1

SNAP REACTORS – NOMINAL DESIGN CONDITIONS

	SNAP 10A	Interim 10A/2	Upgraded 10A/2	SNAP 10B Basic	SNAP 10B Advanced				SNAP 8 Reference Design	SNAP 8 Manrated	SNAP 8 Upgraded	Advanced Zirconium Hydride Reactor
					Getter		TCA					
					High-P	Low-P	High-P	Low-P				
Power Level (kwt)*	39.5	100	100	100	325	100	325	100	600	600	1,200	3,000
Outlet Temperature (°F)*	980	1,200	1,300	1,300	1,300	1,300	1,300	1,300	1,300	1,300	1,300	1,300
Coolant Temperature Rise (°F)	125	200	100	100	200	100	200	100	200	200	200	200
Number of Elements	37	37	37	37	85	55	85	55	211	211	241	583
Element OD (in.)	1.25	1.25	1.25	1.25	0.855	1.06	0.855	1.06	0.560	0.560	0.560	0.428
Maximum Fuel Temperature (°F)	1,080	1,340	1,490	1,490	1,590	1,440	1,590	1,440	1,520	1,520	1,485	1,565
Prepoison Loading ($)	1.60	2.10	4.25	4.25	2.30	1.25	4.20	1.90	3.00	3.00	3.00	3.00
Barrier Material	Solaramic	Solaramic	SCB	SCB	SCB	SCB	SCB	SCB	SCB	SCB	SCB	SCB
Core Length (in.)	12.25	12.25	12.25	12.25	13.8	13.6	13.8	13.6	16.825	16.825	24.0	27.0
Core ID (in.)	8.875	8.875	8.875	8.875	8.875	8.875	8.875	8.875	9.214	9.214	9.7	12
Reflector Thickness, Nominal (in.)	2.0	2.625	3.25	3.25	3.0	2.0	3.0	2.0	3.0	3.0	3.0	3.0
Shielded Diameter, Shoulder (in.)	18.0	18.4†	19.3§	14.6	14.6	12.6	14.6	12.6	27.6	19.0	21.0	24.0
Shoulder Height, Above Shield (in.)	14.75	14.75	14.75	17.75	18.9	18.0	18.9	18.0	22.0	23.5	30.0	33.0
Control Method	Static	Active	Static	Static	Static	Static	Static	Static	Active	Active	Active	Active
Controlled Reflector Elements	2	2	2 or 4	1	1	1	1	1	3	7	7	8
Reactor-Reflector Weight (lb)**	282	299	334	339	388	337	413	362	600	600	975	2,000
Nominal Lifetime	1 yr	1 yr	1 yr	1 yr	1 yr	1 yr	1 yr	1 yr	10^4 hr	10^4 hr	10^4 hr	10^4 hr
NaK Flowrate (lb/hr)	4,920	8,125	16,250	16,250	26,400	16,250	26,400	16,250	48,800	48,800	97,600	224,000
NaK ΔP (psi)	0.17	0.67	2.7	2.7	0.95	0.4	0.95	0.4	4.8	4.8	7	10
Fuel Power Density (kwt/in.^{3} fuel)	0.08	0.19	0.19	0.19	0.54	0.17	0.52	0.16	0.765	0.765	0.940	1.51
N_H (10^{22} atoms H/cm^3)	6.35	6.3	6.3	6.3	6.3	6.3	6.3	6.3	6.05	6.05	6.10	6.10
Burnup, Peak (metal at.%)	0.02	0.06	0.06	0.06	0.17	0.05	0.17	0.05	0.28	0.28	0.35	0.66
Reliability Goal (nuclear system)	0.897	0.946	>0.95	>0.95	>0.98	>0.98	>0.98	>0.98	0.97	>0.99	>0.97	>0.97

*Nominal, end-of-life
†21.5 in. maximum for manned systems; assumes drum position at first sensible heat
§23.0 in. maximum for manned systems; assumes drum position at first sensible heat
**Includes weight of piping and NaK to base of shield

TABLE III-2

SNAP REACTOR WEIGHT BREAKDOWNS

	Fuel Element	Core Vessel and Structure	Reflector and Structure	Drive Mechanism	Piping*	NaK*	Wiring and Sens	Misc and Cont	Total (lb)
10A	125	38	73	15	6	10	10	5	282
I – 10A/2	125	38	88	15	8	10	10	5	299
U – 10A/2	125	38	123	15	8	10	10	5	334
10B Basic	125	38	128	15	8	10	10	5	339
<u>10B Advanced</u>									
Hi-P getter	142	52	115	20	18	25	11	5	388
Lo-P getter	140	41	78	20	18	25	10	5	337
Hi-P TCA	142	52	115	20	18	25	11	30	413
Lo-P TCA	140	41	78	20	18	25	10	30	362
8 RD	194	77	172	62	10	29	20	36	600
8 M	194	80	165	90	10	28	20	13	600
8 U	315	130	270	100	15	40	25	80	975
AZH	505	225	420	125	20	50	30	525	2000

*Includes weight of reactor loop piping and NaK to base of shield. Does not include PCS components (e.g., heat exchangers, expansion compensator) which may be mounted above base of shield.

of element length) of samarium oxide which serves as a burnable poison to provide reactivity control. The fuel material itself is 1.21 in. in diameter, which leaves a nominal 2-mil radial gap between the barrier and the cladding in the cold condition. In operation, this gap is filled with hydrogen gas which promotes heat transfer.

The reactor coolant is NaK, a liquid metal alloy of sodium and potassium. The alloy employed is the eutectic mixture of 22% sodium and 78% potassium by weight. The NaK is pumped through the system by an electromagnetic pump which contains no moving parts.

The fuel elements are positioned in the core on a 1.26-in. center-to-center triangular spacing by an upper and lower grid plate fabricated from 316 stainless

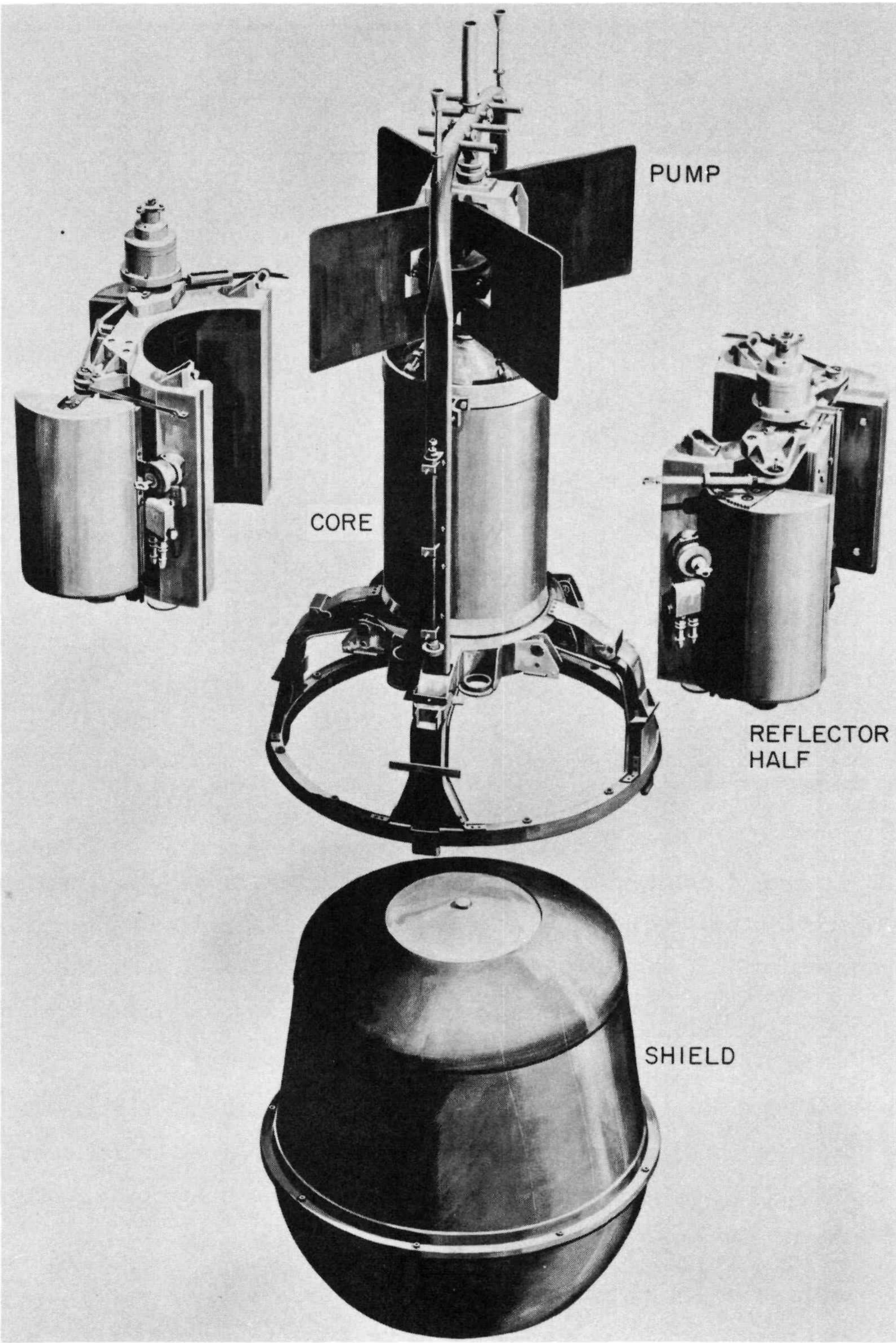

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Figure III-2. Exploded View of SNAP 10A Nuclear System steel. The upper plate is a single piece of material 0.125 in. thick. The lower plate is 0.500 in. thick and consists of a brazed assembly of an 0.060-in.-thick baffle plate and an 0.060-in.-thick orifice plate with spacers between them.

The reactor vessel houses the grid plates and fuel bundle and provides containment for the NaK. The vessel is essentially a right circular cylinder with an inside diameter of 8.875 in. and an overall length of approximately 16 in. It is closed on the bottom by an inverted dished head and on the top by a conical head. The vessel is fabricated from 316 stainless steel. Wall thicknesses vary from 0.032 in. in the core region to 0.125 in. in areas to which attachments are welded. Two diametrically opposed inlet nozzles are provided in the side of the bottom of the vessel and one outlet through the center of the top head. Total weight of the reactor core and vessel assembly is 163 lb.

The reactor is reflected by a beryllium sleeve, approximately 2 in. thick, which surrounds the reactor vessel. The inside of this sleeve is cylindrical, while the outside is bounded by plane surfaces approximating a cylinder. In addition, six beryllium internal reflector pieces are positioned inside the reactor vessel around its circumference and between the grid plates. The pieces are held in place by pins in the grid plates which mate with holes in the reflectors. The reflector pieces are 12.455 in. in length and are shaped to fit fuel element and reactor vessel wall contours. The internal reflectors have no major structural function.

There are four semicylindrical cavities, spaced at 90° intervals in the external surface of the reflector sleeve, which are utilized for control purposes. Reactor control is provided by four cylindrical segment beryllium drums. The drums are rotated into the semicylindrical cavities in the reflector block to vary neutron leakage from the core. Drum movement is actuated by control motors located on top of the reflector block. In the full-in position, the drums are rotated into the reflector block cavities and complete the reflecting sleeve. In the full-out position they are rotated 135° from the full-in position, leaving a cavity in the sleeve. With the control drums in the full-out position, the reactor is adjusted to be $5.60 subcritical at room temperature. The drums are cut from a cylinder of 3-1/2-in. radius. The bare (unshimmed) drum segment has a maximum thickness of 1.875 in. Each drum may be increased in thickness with as many as five 1/8-in.-thick shims which are attached to the back or flat side of the drum. The length of the drums is 10.875 in.

The control and reflector assembly is mounted on the reactor in two halves. Each half is supported on two hinges located at the lower end of the assembly. The halves are rotated into position on the hinges until stops located near the top of the reactor are contacted. The complete assembly is then held in position by a retaining band around the upper end of the reflector halves. Total weight of the reflector-control assembly is 103 lb.

The reactor is started up in orbit and controlled automatically during the initial power stabilization period. Two startup drums are spring-loaded. This adds $4.30 of reactivity. Fifty seconds later, the two fine control drums take their first step. They then move a half-degree step each 150 sec until reactor operating temperature is reached. Approximately 7 hr is required after startup before criticality is reached, and approximately 2 hr more before operating temperature is reached. Active power control is then maintained by moving the control drums whenever the NaK outlet temperature drops below 1010°F. Two temperature sensors are located in the NaK outlet line, and by means of a switch and controller in the instrument compartment, control the operation of the drums. When the outlet temperature drops below the set point, the fine control drums will take another step inward. This continues until reactivity transients are completed (approximately 72 hr after the startup command), at which time the control system may be commanded "off."

For the remainder of reactor operation, power and temperature are controlled by means of a balance between reactivity addition by burnout of the samarium oxide prepoison and reactivity reduction due to fission product buildup, fuel depletion, and hydrogen loss. During the startup transient, power and temperatures increase rapidly over a period of a few minutes. The resulting peak temperatures and stresses, however, are well within the system material strength capabilities.

2. Nominal Design Conditions

Table III-3 summarizes the nominal design conditions.

TABLE III-3

SNAP 10A DESIGN CHARACTERISTICS

Power Level, End-of-Life (kwt)	39.5
Outlet Temperature, End-of-Life (°F)	980
Coolant Temperature Rise (°F)	125
Number of Elements	37
Cladding Material	Hastelloy N
Element OD (in.)	1.25
Maximum Fuel Temperature (°F)	1080
Prepoison Loading ($)	1.60
N_H (10^{22} atoms H/cc fuel)	6.35
Hydrogen Leakage (%/yr)	0.05
Barrier Material	Solaramic
Core Length (in.)	12.25
Core Diameter (in.)	8.875
Reflector Thickness (in.)	2.0
Shielded Diameter at Top of Core (in.)	18.0
Distance From Shield Top to Core Top (in.)	14-3/4
Control Method	Static
Active Control Drums	2
Reactor-Reflector Weight (lb)	282
Lifetime (yr)	1
Power Density, Reactor (kwt/lb)	0.14
Power Density, Fuel (kwt/in.^{3})	0.08
Maximum Burnup (metal at. %)	0.02
Reactivity Contingency ($)	0.30

3. Limits of Operation

The following potential limitations on reactor operation were considered in determining the performance capabilities of the SNAP 10A reactor under static control.

a. Reactivity Limit

With only minor redesign (some relocation of cables and wiring harness), three additional sets of 1/8-in.-thick beryllium shims may be added to the nominal SNAP 10A flight reflector. The additional initial reactivity provided in this manner allows an increase in the prepoison concentration loaded in the core. The increased reactivity input rate due to the higher poison loading in turn allows for compensation of the larger reactivity losses associated with higher performance operation. In general, an increase in the prepoison loading leads to an increase in reactor capabilities for a given static control temperature-time profile. If a reactor operates, for example, at a given initial power and temperature level with a given prepoison loading, the temperature will degrade by that number by degrees defined as allowable (in this study, 50°F) in a certain time period. This length of time is then defined as the reactor lifetime under these conditions. After this time has elapsed, the reactor will continue to operate but at a steadily decreasing power and temperature. If the initial prepoison loading had been increased, however, the reactor would, in general, operate for a longer time period before degrading in temperature by the same amount. It should be noted that the nominal reactor temperature is defined as the end-of-life temperature, which is 50°F below the initial temperature.

This study used the maximum prepoison loading that could be incorporated in a given reactor commensurate with its cold clean reactivity, temperature and power defect, and initial reactivity transients. A reactivity contingency of $0.50 is provided to allow for loading tolerances and calculational uncertainties. The reactivity limits shown are, therefore, conservative and provide a reasonable design margin.

b. Maximum Fuel Temperature

A maximum fuel temperature of 1700°F, including all hot channel factors, has been imposed upon reactor operation. Selection of this upper limit is based upon the degradation of fuel alloy strength at this temperature. The conclusions of the study are not greatly affected by the choice of this temperature level, since this limit is reached in relatively few cases.

c. Fuel Swelling

Diametral fuel expansion is limited to 6 mils to avoid interference between fuel and cladding in the "hot" condition. Fuel swelling increases with reactor power, temperature, and lifetime. End-of-life at a particular power and temperature is defined as that time when the fuel touches the cladding in any part of the core. This approach is conservative for two reasons.

1) The reactor can operate for some period of time with some degree of interference between fuel and cladding.

2) Fuel swelling is a function of both fractional fuel burnup and maximum fuel temperature. However, these two maxima are not simultaneously present at the same location in the core. The analysis assumes that both occur in the same location and is, therefore, conservative.

d. Hastelloy-N Creep Limit

The internal hydrogen pressure in the cladding tubes is a function of reactor power and temperature. The resultant creep of the cladding is a function of this hydrogen pressure, cladding thickness, external NaK system pressure, cladding temperature, and a weak function of reactor lifetime. A cladding creep of 0.2% has been selected as the design criterion for these reactors. This value was based upon the experimental observation that the integrity of the hydrogen barrier was not compromised at this creep level. It is quite probable, however, both that the barrier will remain intact at higher creep levels and that the reactor will operate with one or more failed barriers.

e. Maximum Barrier Temperature Limitations

Some degradation of the SNAP 10A hydrogen barrier takes place at a barrier temperature of about 1300°F. A design criterion on barrier temperature of 1300°F, with hot channel factors imposed, was selected. This criterion is conservative since the reactor can continue to operate with some degradation of the hydrogen barrier.

f. End-of-Life Temperature Uncertainty Limitations

Although each reactor is designed to be at its nominal outlet temperature at the end of lifetime, unavoidable errors enter due to tolerances in loadings and dimensions and uncertainties in calculations of reactivity effects and temperature coefficients. These uncertainties may penalize the system operation either by reducing reliability if they are uncompensated or by increasing system weight to provide a system which will produce the design electrical output even with the lowest credible end-of-life temperature. End-of-life temperature uncertainties increase with reactor temperature, power, and lifetime. These uncertainty limits are, strictly speaking, not limits as far as the reactor is concerned but are limits only insofar as they affect confidence in the prediction of system operation.

g. Other Design Limitations

Other limits which were not considered for SNAP 10A are fuel phase change and beryllium swelling. These limits occur far above the appropriate performance limits for the SNAP 10A reactor, but are considered for some other reactors.

Performance limitations for the SNAP 10A reactor are shown in Figures III-3 through III-5 for reactor lifetimes of 1 yr, 3 yr, and 5 yr, respectively, as a function of nominal (end-of-life) coolant outlet temperature.

(1) One-Year Operation

The area of operation for SNAP 10A for 1 yr is limited by reactivity considerations below end-of-life outlet temperatures of 1200°F. Operation above this temperature is possible, but is strongly limited by barrier temperature considerations. Temperature uncertainties for 1 yr are small in all cases and should not impose a limit on reactor performance. Operation at 185 kwt at 1000°F outlet, 155 kwt at 1100°F, or 75 kwt at 1200°F is possible. It should be noted that although only minor reactor changes are required to achieve these conditions, major redesign of other components of the present SNAP 10A APU might be necessary. In particular, a NaK coolant pump of higher capacity and greater endurance must be available, components and wiring qualified for higher temperatures must be provided, and the radiation shield would have to be redesigned to shield against the higher power level and slightly greater shielded envelope size. Many of the auxiliaries to the reactor proper (e.g., bearings, actuators, etc.) are already being qualified at elevated temperatures as part of the SNAP 10A/2 program.

(2) Three-Year Operation

Operation for 3 yr is completely reactivity-limited, with end-of-life uncertainties well below 50°F. The reactor can produce 110 kw for 3 yr at 1000°F outlet or 80 kw at 1100°F outlet.

(3) Five-Year Operation

The performance limitations are again associated with reactivity considerations. Operation at 70 kw at 1000°F outlet or 45 kw at 1100°F outlet is possible.

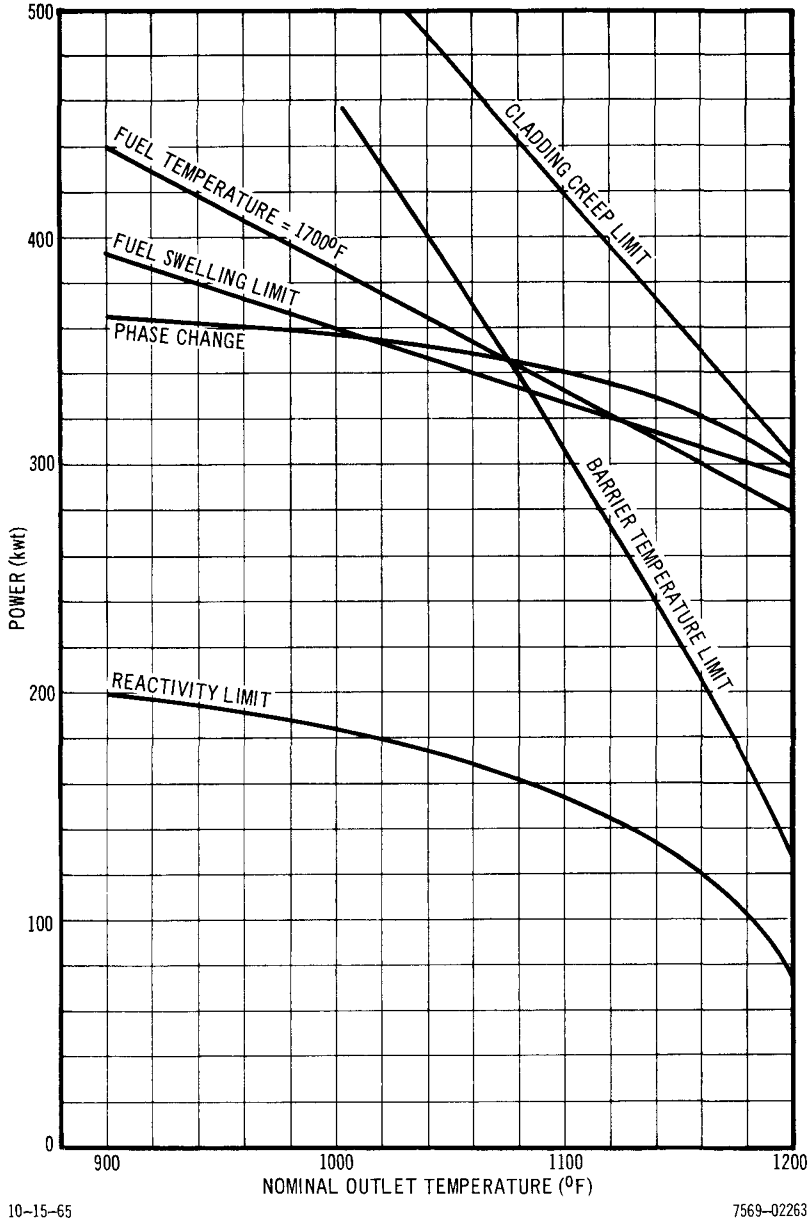

Figure III-3. SNAP 10A Parameters, 1-yr Life

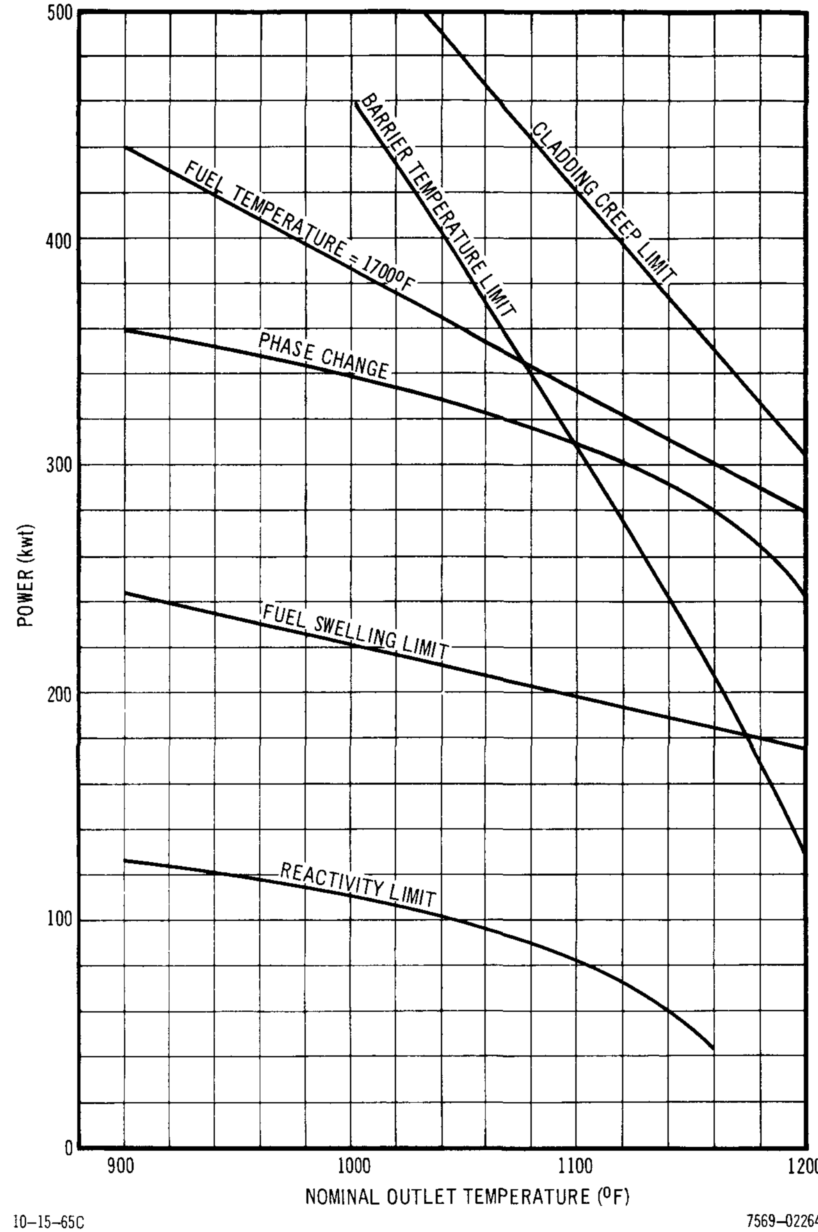

Figure III-4. SNAP 10A Parameters, 3-yr Life

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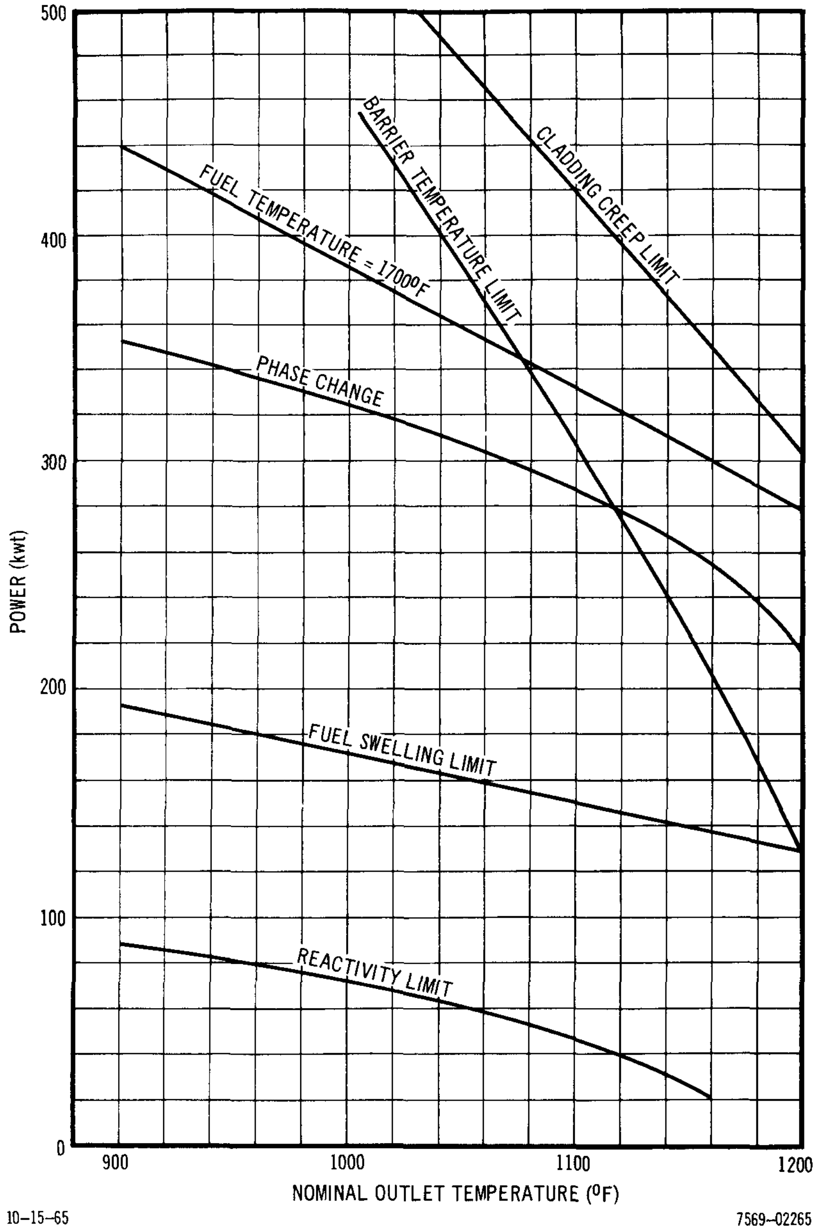

Figure III-5. SNAP 10A Parameters, 5-yr Life

4. Reliability*

The SNAP 10A reactor achieves a relatively high degree of reliability through simplicity and overdesign. It employs a combination of proven reactor components, inherent negative temperature and power coefficients, and low temperature operation to achieve high reliability. The reactor is statically controlled following the first few days of actively-controlled operation. The low temperatures in the core (~1080°F maximum fuel temperature) reduce the reactivity loss associated with hydrogen leakage to a negligible value. Potential failures caused by radiation effects are also minimized by low temperature, low power operation.

The SNAP 10A control drums must operate over a period of about 3 days. Two of the four drums are snapped-in at startup. The reliability goal for normal operation of the two fine control drums is 0.983. Reliability allocations for the remainder of the SNAP 10A reactor subsystem are shown in Table III-4.

TABLE III-4

SNAP 10A REACTOR SUBSYSTEM RELIABILITY ALLOCATION

Assembly	Cumulative Reliability		
	Launch and Ascent	Startup	1 yr
Reactor Structure	0.99987	0.99971	0.99874
Reactor Core	0.99937	0.99778	0.92833
Reflector	0.99918	0.99580	0.99001
Control Equipment	0.99932	0.98697	0.98690
Radiation Shield	0.99983	0.99952	0.99036
Complete Reactor Subsystem	0.99755	0.97989	0.89718

B. INTERIM SNAP 10A/2

1. Description

The Interim SNAP 10A/2 reactor consists of the 37-element SNAP 10A core surrounded by an augmented reflector to allow operation at higher power (100 kwt) and temperature conditions (1200°F). The only in-core change is an increase in

*Reliability as used herein is defined as the probability that the system (or component) will perform its required function under design conditions for the specified operating time.

the prepoison concentration so that the burnout rate will correspond more closely to the higher reactivity losses associated with higher performance.

The control drums operate for the entire lifetime of the Interim SNAP 10A/2 reactor as opposed to their short initial operation with the SNAP 10A reactor. The core outlet temperature is thereby maintained within the deadband of the controller throughout lifetime. This mode of control is referred to as "active control" as opposed to "static control" of the SNAP 10A reactor.

Figure III-6 shows the important features of the Interim 10A/2.

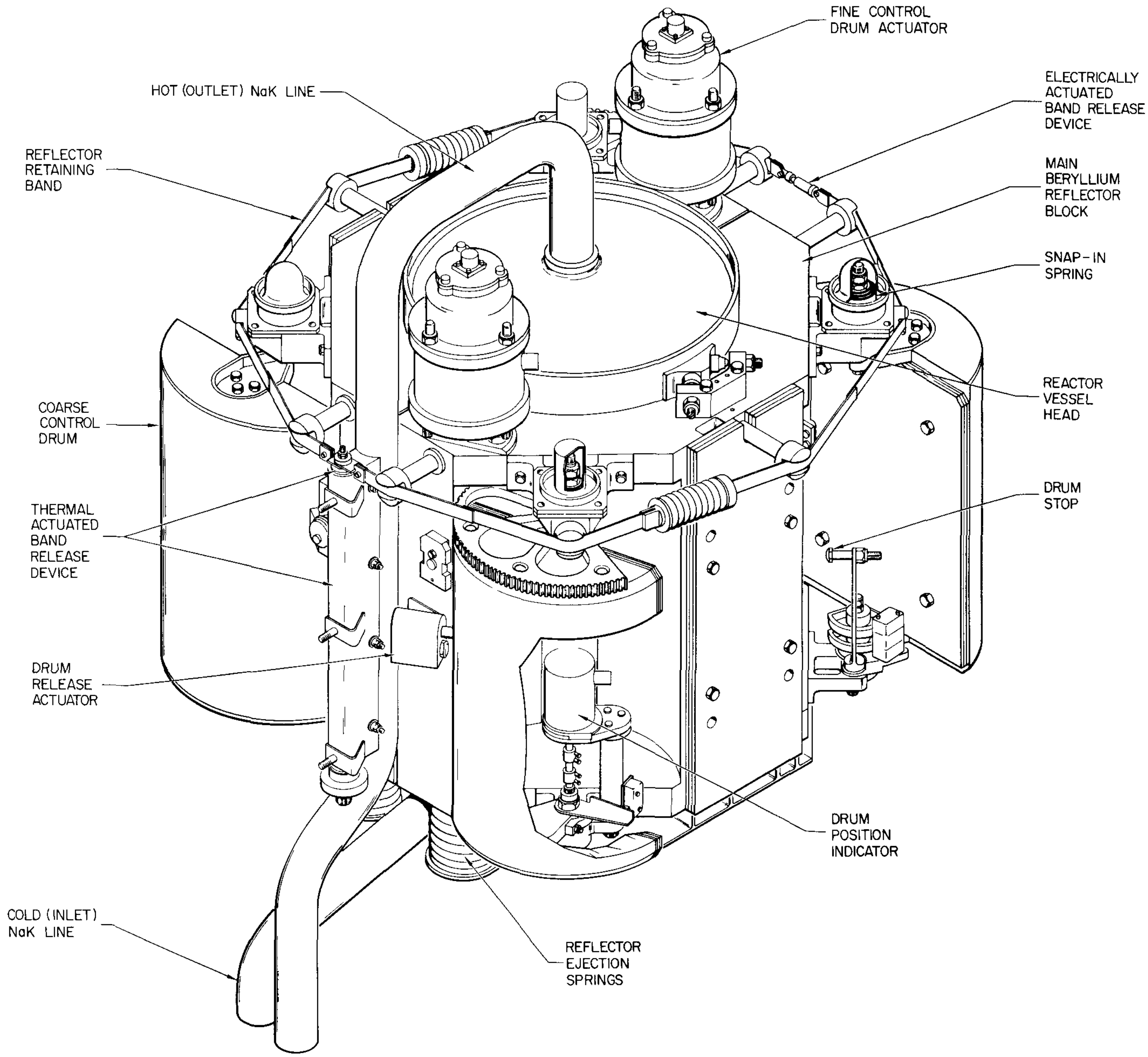

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Figure III-6. Interim 10A/2 Reactor

2. Nominal Design Conditions

Table III-5 summarizes the nominal design conditions.

TABLE III-5
NOMINAL DESIGN CONDITIONS

Power Level (kwt)	100
Outlet Temperature (°F)	1200
Coolant Temperature Rise (°F)	200
Number of Elements	37
Cladding Material	Hastelloy N
Element OD (in.)	1.25
Maximum Fuel Temperature (°F)	1340
Prepoison Loading ($)	2.10
N_H (10^{22} atoms H/cc fuel)	6.3
Hydrogen Leakage (%/yr)	0.3
Barrier Material	Solaramic
Core Length (in.)	12.25
Core Diameter (in.)	8.875
Reflector Thickness (in.)	2.4
Shielded Diameter at Top of Core (in.)	18.4
Distance From Shield Top to Core Top (in.)	14-3/4
Control Method	Active
Active Control Drums	2
Reactor-Reflector Weight (lb)	299
Lifetime (yr)	1
Power Density, Reactor (kwt/lb)	0.33
Power Density, Fuel (kwt/in.^{3})	0.19
Maximum Burnup (metal at.%)	0.06
Reactivity Contingency ($)	0.75

3. Performance Limitations

Five of the limitations mentioned for the SNAP 10A reactor (reactivity, barrier temperature, fuel swelling, cladding creep, and fuel temperature) also apply to the Interim 10A/2 reactor. The nature of the reactivity limit is slightly different because of the use of active rather than static control. End-of-lifetime in this case is defined as that time when the drums will have rotated inward to their full-in position. A $0.75 reactivity contingency is provided here. Because active control maintains a constant reactor temperature, end-of-life temperature uncertainty is not a limitation.

Because of the higher power levels considered, possible phase change limitations were investigated. There is a physical phase change from the δ- to the β-phase in the uranium-zirconium hydride as the hydrogen concentration is reduced. The phase change boundary location is a function of the fuel temperature. To avoid the volume changes associated with phase change, the reactor lifetime must be such as to maintain the hydrogen concentration in every part of the fuel above the phase boundary concentration at the local temperature. Hydrogen losses due to both leakage and redistribution were considered. Performance limitations for 1-, 3-, and 5-yr lifetimes are shown in Figures III-7 through III-9.

a. One-Year Operation

If the reactor is not shimmed, performance is limited below about 1265°F outlet temperature by reactivity considerations and above that temperature by barrier temperature considerations. Operation at 125 kwt at 1150°F outlet, 100 kwt at 1200°F, or 70 kwt at 1250°F is possible. The first approach to advanced performance would be to change the barrier material to the SNAP 8 coating to allow higher barrier temperature operation. This change would allow operation at the reactivity-limited case of 25 kwt at 1300°F. The full potential of the advanced barrier material cannot be realized, however, unless about 0.5 in. of beryllium is added to the reflector surface; i.e., the reflector must be "shimmed." Operation at 265 kwt at 1150°F, 240 kwt at 1200°F, or 175 kwt at 1300°F could then be attained.

b. Three-Year Operation

The unshimmed reactor is not as severely limited by the use of the present SNAP 10A barrier material as in the 1-yr case. Over almost its entire range of operation, the limiting criterion is reactivity. For this case, the limits are 90 kwt/1100°F, 70 kwt/1200°F, and 50 kwt/1250°F.

An examination of Figure III-8 shows that use of an improved barrier provides only minor operational improvements unless the reactor is shimmed.

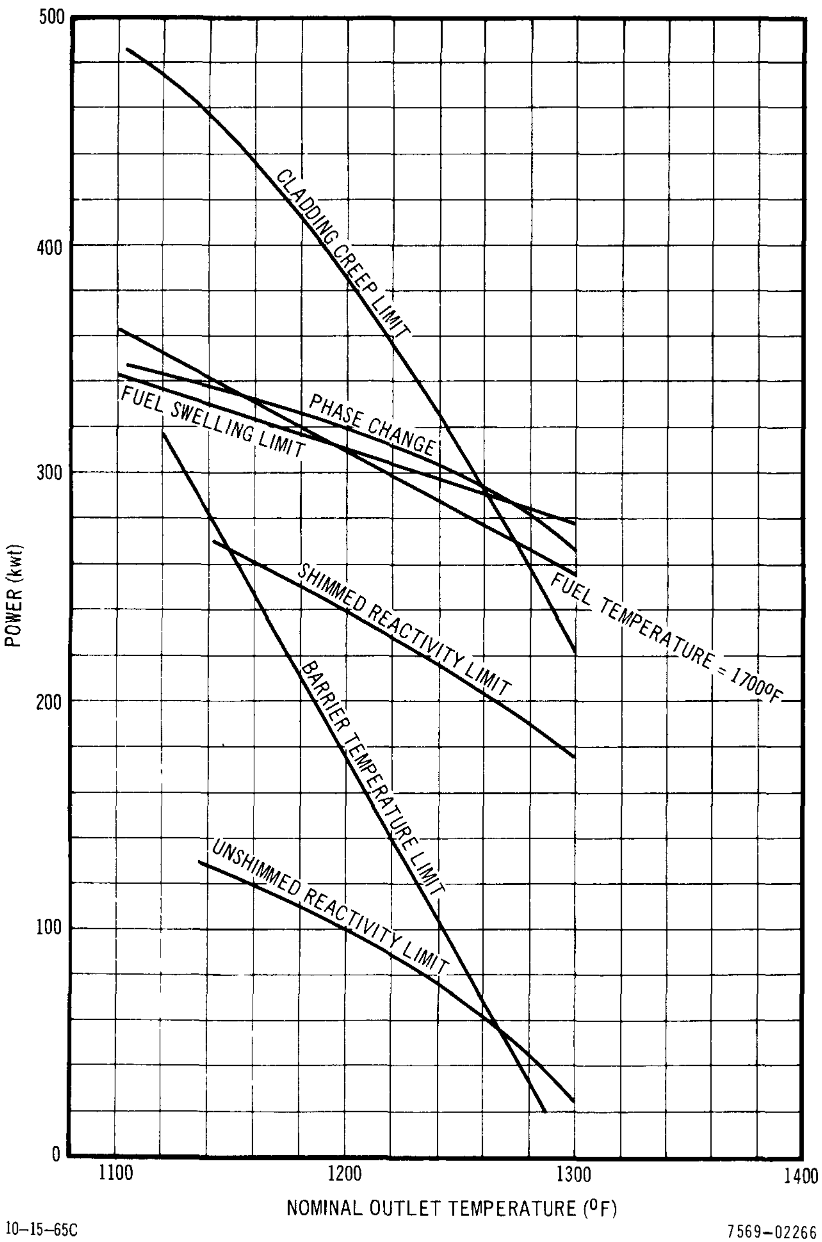

Figure III-7. Interim SNAP 10A/2 Parameters, 1-yr Life

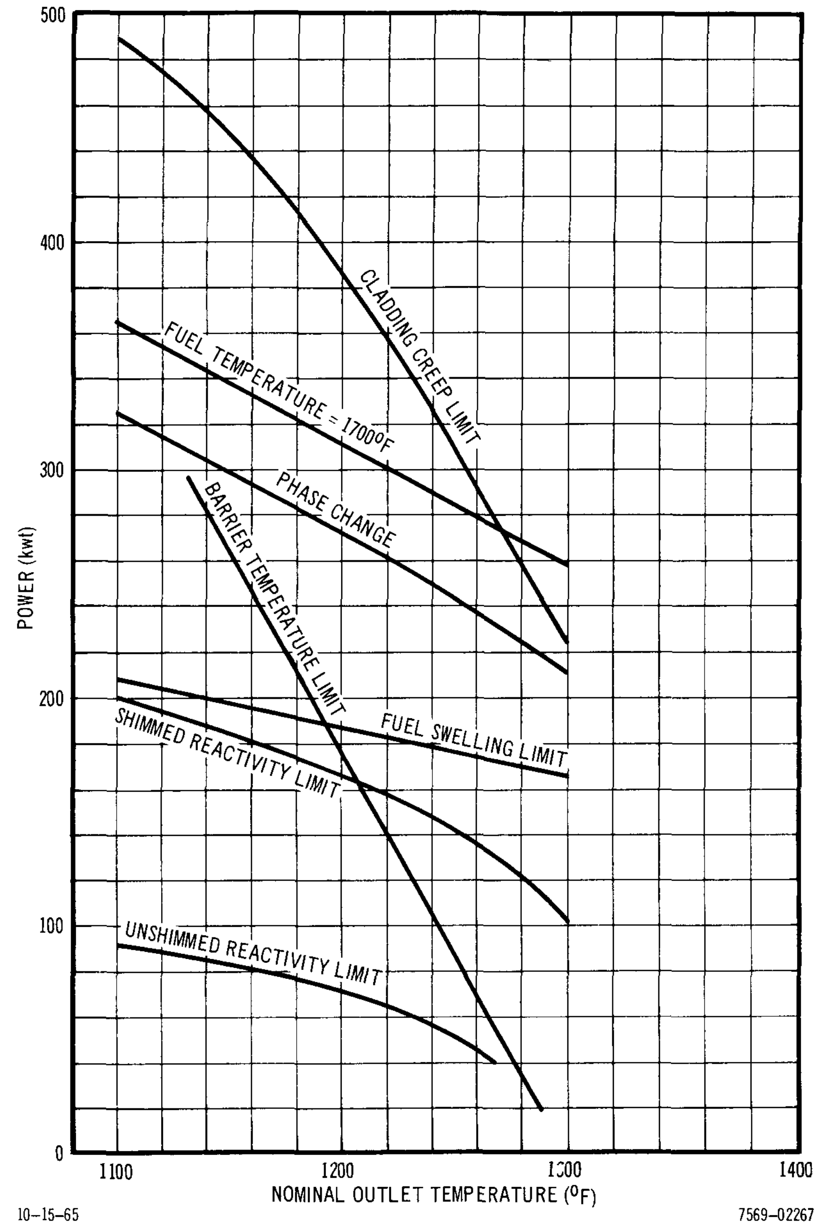

Figure III-8. Interim SNAP 10A/2 Parameters, 3-yr Life

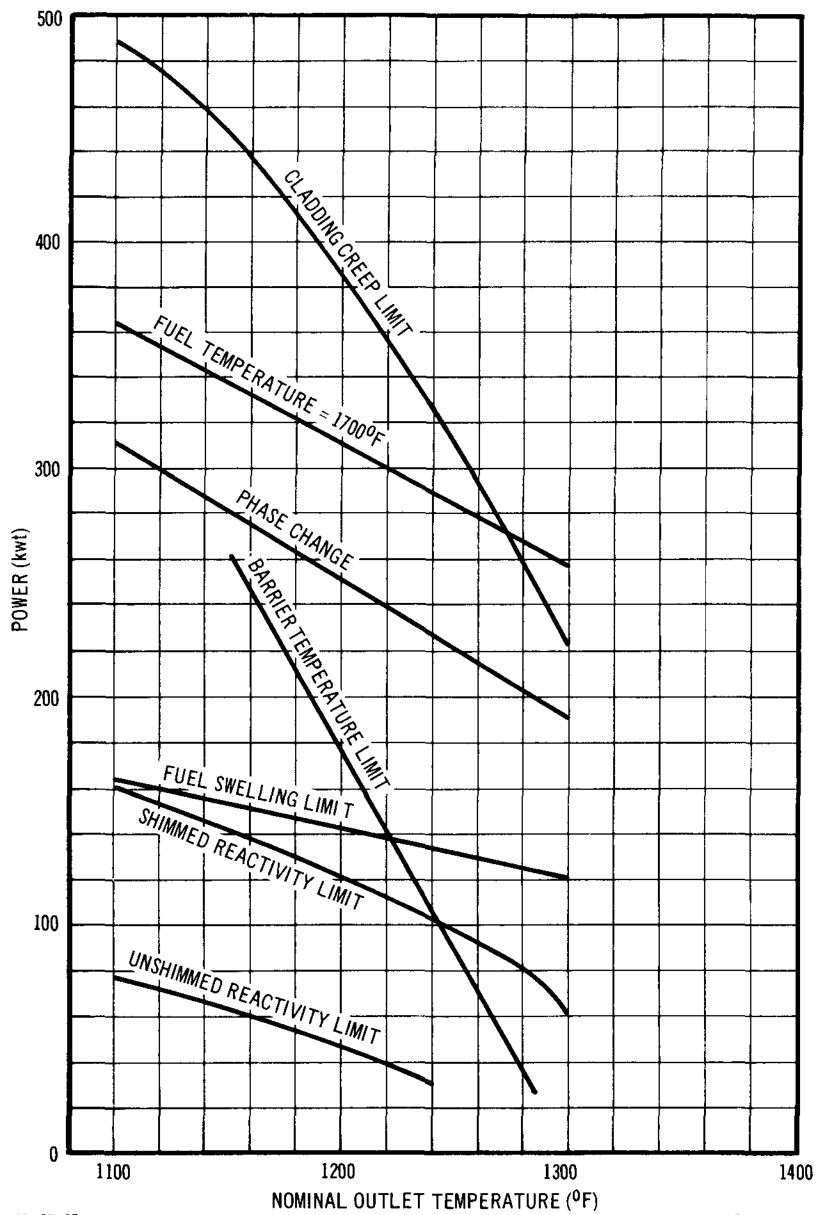

Figure III-9. Interim SNAP 10A/2 Parameters, 5-yr Life

Conversely, shimming provides little improvement unless the barrier is upgraded to allow higher temperature operation. If both of these improvements are made, the reactor becomes limited in all cases to operation described by the "Shimmed Reactivity Limit" line. Operation at 200 kwt/1100°F, 165 kwt/1200°F, or 100 kwt/1300°F is possible.

c. Five-Year Operation (Figure III-9)

The present SNAP 10A barrier material is satisfactory for unshimmed reactor operation, since the reactor is reactivity limited over its entire range. Operation at 75 kwt/1100°F or 45 kwt/1200°F is possible. Shimming the reactor without changing the barrier achieves improvements in lower-temperature operation and allows operation at 160 kwt/1100°F and 120 kwt/1200°F. Finally, substitution of a high temperature barrier material allows operation at 100 kwt/1250°F or 60 kwt/1300°F.

4. Reliability

The Interim SNAP 10A/2 reactor employs the same design as SNAP 10A, combined with partial redundancy, and operates at a higher temperature and power level. The resulting increase in system temperatures, radiation levels, and hydrogen leakage would normally lead to reduced system reliability. However, advancements in the state-of-the art combined with other recent developments, have alleviated these effects. As a result, the complete reactor subsystem reliability allocation is higher than for SNAP 10A. The reactor is actively controlled throughout lifetime. The power level is 100 kwt at a coolant outlet temperature of 1200°F. It is possible to design an actively controlled reactor which is also capable of operating under static control with minimal temperature drift. The Interim SNAP 10A/2 reactor is designed so that the control drums remove reactivity during the first half of the design life and add reactivity during the last half. The nuclear design is such that even if drum movement ceases following three days of active control, e.g., through actuator failure, the design objectives (life, power, and temperature) would still be satisfied in the resulting "static control" mode.

Reliability allocations for the Interim SNAP 10A/2 reactor subsystem are shown in Table III-6.

TABLE III-6

INTERIM SNAP 10A/2 REACTOR SUBSYSTEM RELIABILITY ALLOCATION

Assembly	Cumulative Reliability		
	Launch and Ascent	Startup	1 yr
Reactor Structure	0.99993	0.99965	0.99774
Reactor Core	0.99946	0.99888	0.99329
Reflector	0.99957	0.99790	0.99041
Control Equipment	0.99912	0.99291	0.96544
Radiation Shield	0.99993	0.99967	0.99779
Complete Reactor Subsystem	0.99801	0.98907	0.94556

C. UPGRADED SNAP 10A/2

1. Description

The Interim SNAP 10A/2 reactor is limited in performance by the time-dependent degradation of the hydrogen barrier. The Upgraded SNAP 10A/2 employs the high temperature SNAP 8 barrier to allow operation at higher outlet temperatures. In addition, the static control concept of SNAP 10A is employed. Outlet temperatures are maintained relatively constant by a balance between reactivity losses and reactivity insertion due to prepoison burnout. The temperature drift, whose magnitude is governed by the magnitude of the inherent negative temperature coefficient, is tailored by prepoison loading to drift from a temperature 50°F above the nominal outlet temperature at the beginning of life to the nominal temperature by the end-of-life.

The Upgraded 10A/2 corresponds closely to a previous reactor concept which was briefly examined under the designation SNAP 2A.

2. Nominal Design Conditions

Table III-7 summarizes the nominal design conditions.

TABLE III-7

UPGRADED SNAP 10A/2 DESIGN CHARACTERISTICS

Power level, end-of-life (kwt)	100
Outlet temperature, end-of-life (°F)	1300
Coolant temperature rise (°F)	100
Number of elements	37
Cladding material	Hastelloy N
Element OD (in.)	1.25
Maximum fuel temperature (°F)	1490
Prepoison loading ($)	4.25
N_H (10^{22} atoms H/cc fuel)	6.3
Hydrogen leakage (%/yr)	2.3
Barrier material	SCB
Core length (in.)	12.25
Core diameter (in.)	8.875
Reflector thickness (in.)	3.25
Shielded diameter at top of core (in.)	19.3
Distance from shield top to core top (in.)	14.75
Control method	static
Active control drums	2
Reactor-reflector weight (lb)	334
Lifetime (yr)	1
NaK ΔP (psi*)	2.7
Power density (kwt/lb reactor)	0.31
Power density (kwt in.^{3} fuel)	0.19
Maximum burnup (metal at. %)	0.06
Reactivity contingency ($)	0.50

*May be reduced to less than 0.7 psi by flow reduction and nozzle relief

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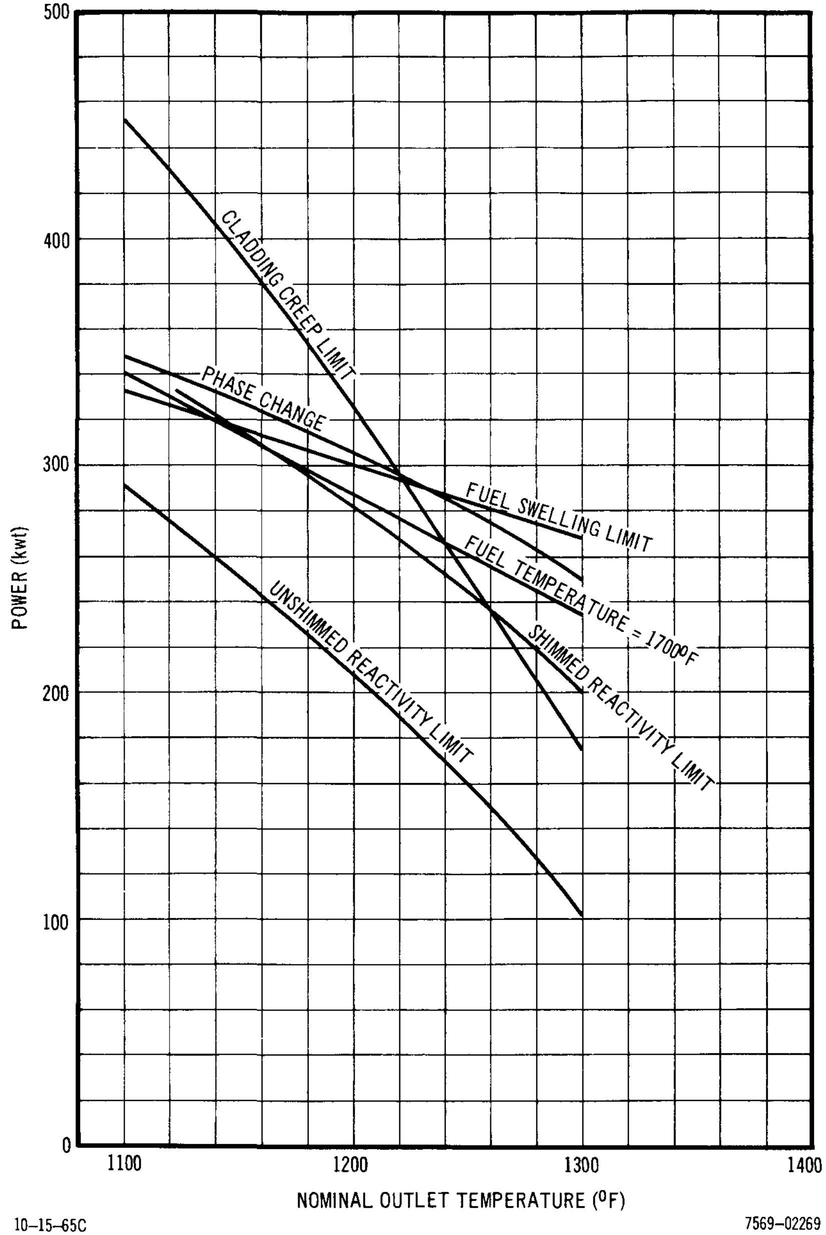

Figure III-10. Upgraded SNAP 10A/2 and (10B Basic) Parameters, 1-yr Life

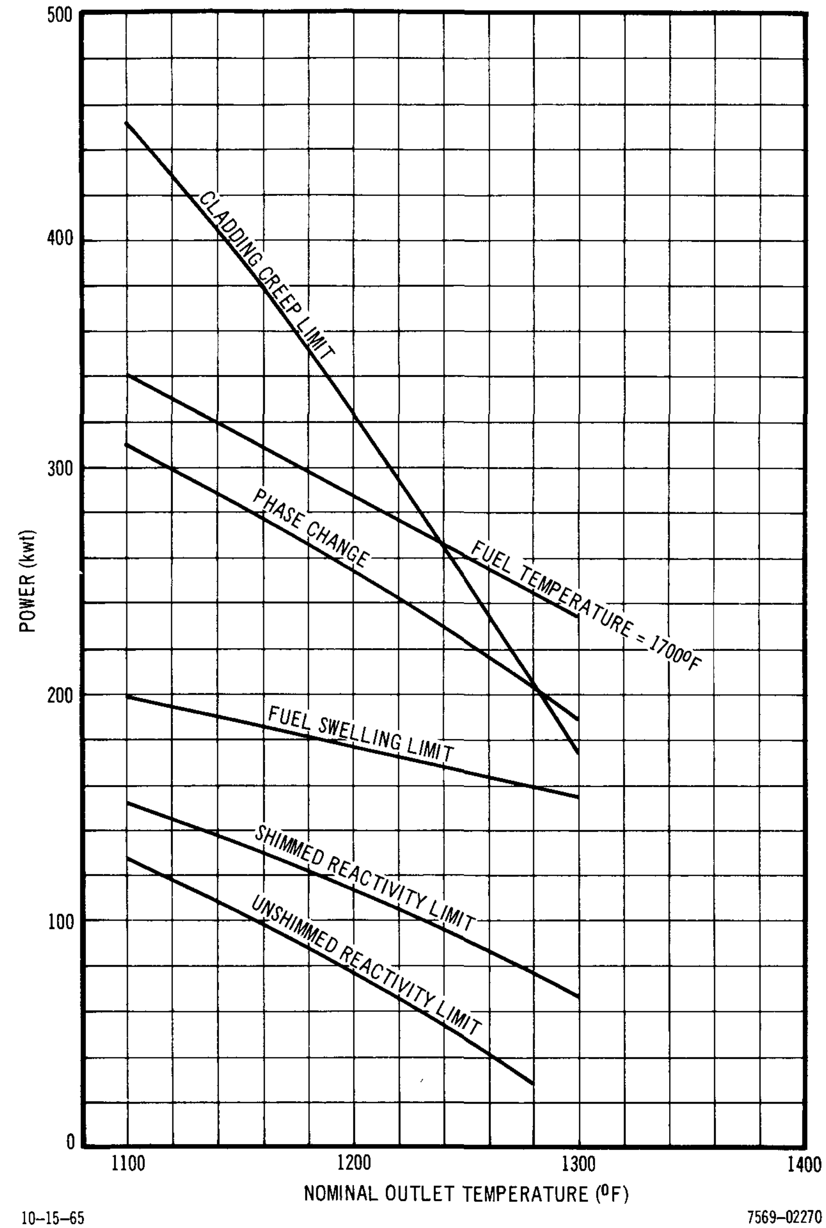

Figure III-11. Upgraded SNAP 10A/2 and (10B Basic) Parameters, 3-yr Life

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3. Performance Limitations

All of the potential limitations of the Interim SNAP 10A/2 exist for the Upgraded SNAP 10A/2 except for the maximum barrier temperature limitation. Use of the SCB coating effectively removes this limitation.

Limitations for the Upgraded SNAP 10A/2 reactor for 1-, 3-, and 5-yr operation are shown in Figures III-10 through III-12. The limitations are also applicable to the SNAP 10B Basic reactor discussed in the next section.

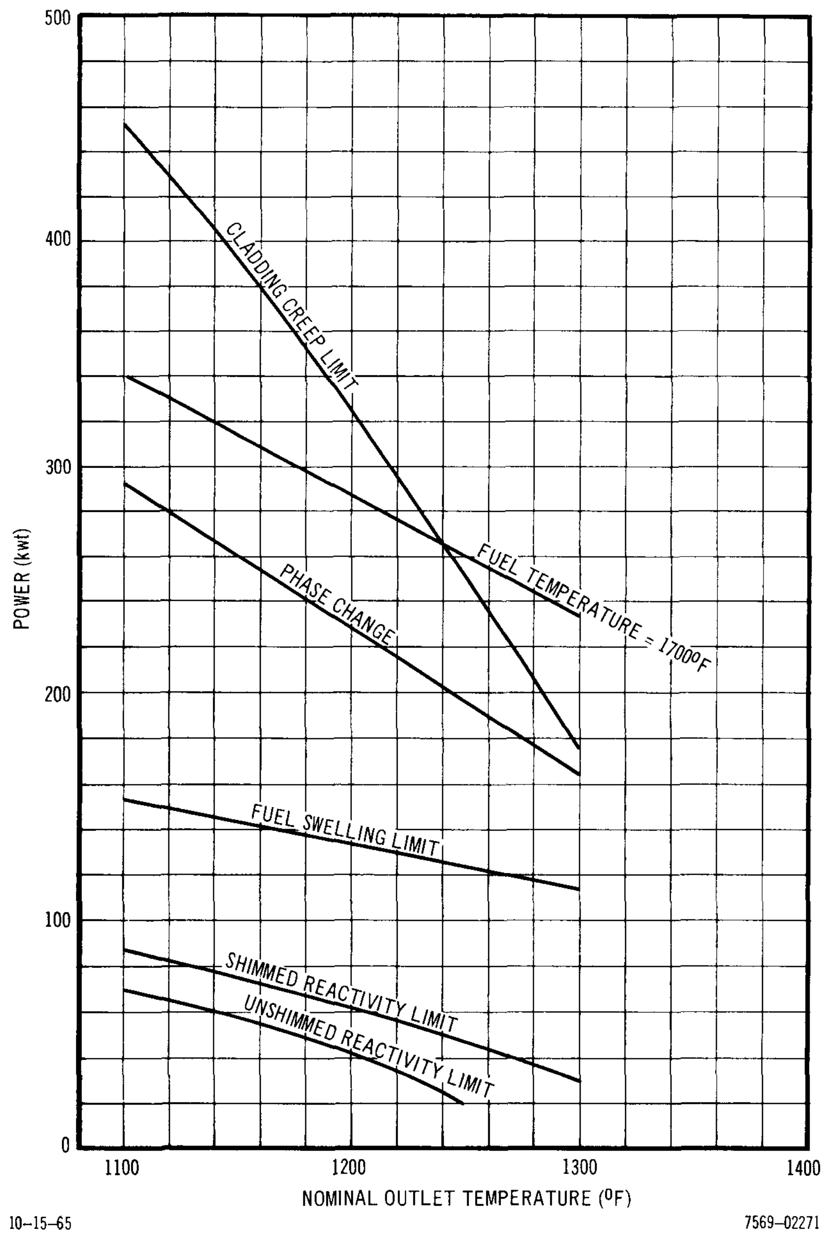

Figure III-12. Upgraded SNAP 10A/2 and (10B Basic) Parameters, 5-yr Life

a. One-Year Operation

The unshimmed reactor is limited by reactivity considerations. Operation at 295 kwt/1100°F, 210 kwt/1200°F, or 100 kwt/1300°F is possible.

If full shimming (0.5 in. of beryllium) is added to the reflector, the reactor operation is limited by fuel swelling reactivity, and cladding creep criteria. Operation is possible at 335 kwt/1100°F, 280 kwt/1200°F, and 175 kwt/1300°F.

b. Three-Year Operation

For 3-yr operation, the reactor, both shimmed and unshimmed, is limited by reactivity considerations. The unshimmed reactor can generate 125 kwt/1100°F, 75 kwt/1200°F, or 50 kwt/1250°F. The shimmed reactor can generate 150 kwt/1100°F, 110 kwt/1200°F, or 65 kwt/1300°F.

c. Five-Year Operation

The same comments apply here as in the 3-yr case. The unshimmed reactor can operate at 70 kwt/1100°F or 50 kwt/1200°F. Operation at 85 kwt/1100°F, 60 kwt/1200°F, or 20 kwt/1300°F is possible with the shimmed reactor.

4. Reliability

The Upgraded SNAP 10A/2 reactor operates under static control at 100 kwt and 1300°F coolant outlet temperature. Use of an improved hydrogen barrier material capable of operation at coolant outlet temperatures of 1300°F provides the same reactor reliability as quoted for Interim SNAP 10A/2, except that it is not required to move the control drums after the first 3 days of operation at design power and temperature.

D. SNAP 10B BASIC

1. Description

All of the preceding reactors employ gyrating drums in the reflectors for control purposes. The geometry of these drums is such that small outward rotation results in a significant radial protrusion of the drum tips. Since very small amounts of material protruding outside of the shield shadow can cause large scattered neutron dose rates to the payload, the shield must be sufficiently large to shadow the drum tips. This increased shield radius requirement contributes significantly to total system weight. (For a typical unmanned application, an increase in diameter of 1 in. in the shielded diameter at the core top corresponds to a 30-lb increase in shield weight.) In order to reduce the radial size of the shield, the 10B Basic reactor employs axially moving segments of the reflector for reactor control instead of the gyrating elements used in earlier designs. These axially moving segments always stay within the diameter of the reflector itself and, therefore, do not enlarge the required shield diameter by their motion. To further save weight, the reflector diameter tapers inward from bottom to top, to more nearly fit the shield shadow cone. In addition, the NaK coolant outlet lines and the wiring harnesses are buried in the reflector to minimize the shielded envelope. Figure III-13 shows the general arrangement of SNAP 10B, and Figure III-14 illustrates the nature of the shield savings realized with the tapered reflector.

The use of sliding control segments which are partially open during lifetime perturbs the power distribution slightly. The peak-to-average power ratio is increased by a few percent. Offsetting this effect is the flux reduction in the upper half of the core caused by the thinner reflector in this region. A detailed analysis has not yet been performed on these effects, but preliminary results indicate that few problems will be introduced by the presence of moderate tapers ($\sim$0.5-in. increase in radial beryllium thickness from the top to the bottom of the core).

There will be some practical upper limit to the allowable taper and minimum beryllium thickness however. For higher powered reactors, the buried NaK lines will replace a large amount of the beryllium reflector. The NaK in these lines is almost transparent to leaking neutrons. Sufficient beryllium thickness must, therefore, be provided at the core top to prevent large increases in neutron leakage. It is felt that a minimum reflector thickness of 1.75 in. at the core top is desirable.

The 10B Basic reactor design is based on the assumption of static control. Its core is identical with the Upgraded 10A/2 reactor core. The 10B reactor has also been designated "Advanced 10A/2" and corresponds to the static control version of a more general previous conceptual analysis designated "SNAP 10A/2 (small)."

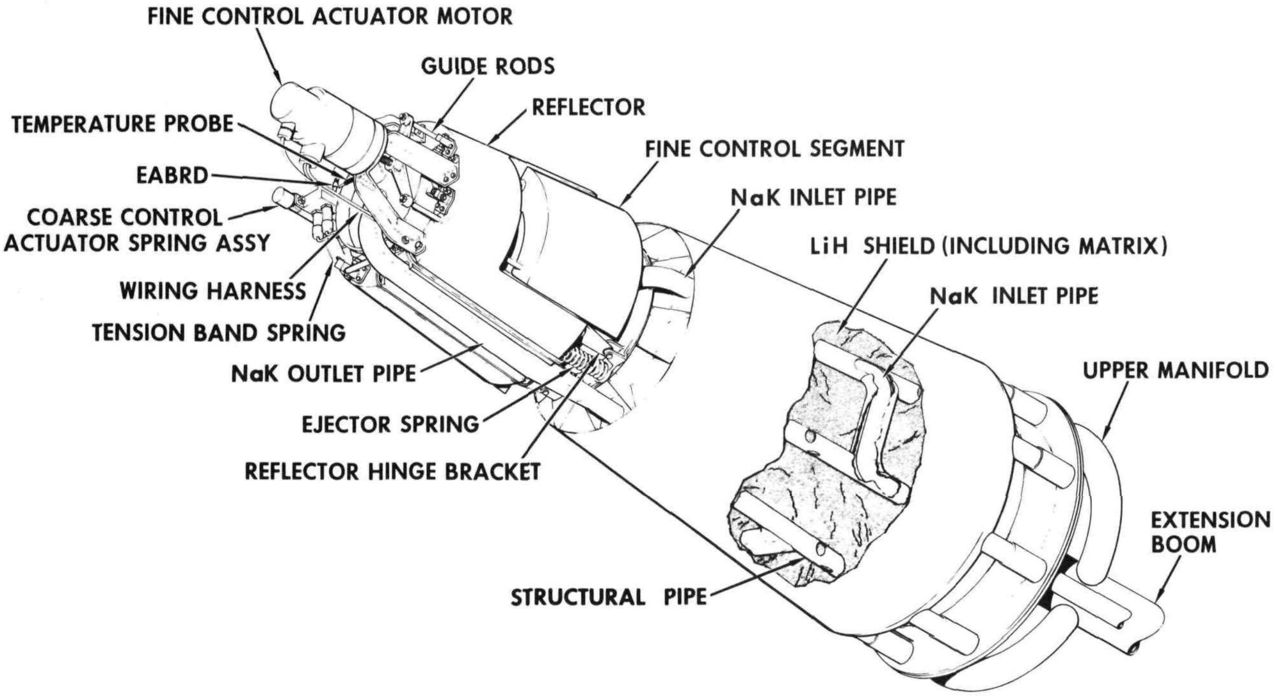

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Figure III-13. SNAP 10B Reactor Concept

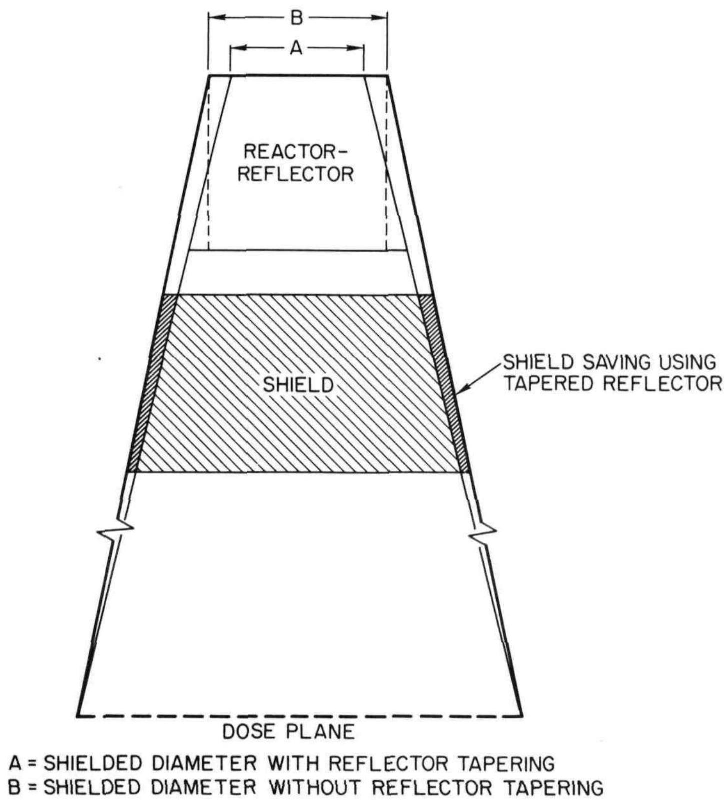

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Figure III-14.
Tapered Reflector Concept

2. Nominal Design Conditions

Table III-8 summarizes the nominal design conditions.

TABLE III-8

SNAP 10B BASIC DESIGN CHARACTERISTICS

Power level, end-of-life (kwt)	100
Outlet temperature, end-of-life (°F)	1300
Coolant temperature rise (°F)	100
Number of elements	37
Cladding material	Hastelloy N
Element OD (in.)	1.25
Maximum fuel temperature (°F)	1490
Prepoison loading ($)	4.25
N_H (10^{22} atoms H/cc fuel)	6.3
Hydrogen leakage (%/yr)	2.3
Barrier material	SCB
Core length (in.)	12.25
Core diameter (in.)	8.875
Reflector thickness (in.)	3.25 (mean)
Shielded diameter at top of core (in.)	14.6
Distance from shield top to core top (in.)	17.75
Control method	static
Active control drums	1
Reactor-reflector weight (lb)	339
Lifetime (yr)	1
Power density (kwt/lb reactor)	0.30
Power density (kwt/in.^{3} fuel)	0.19
Maximum burnup (metal at. %)	0.06
Reactivity contingency ($)	0.50

3. Performance Limitations

Since the core of the SNAP 10B Basic reactor is identical to the core of the Upgraded SNAP 10A/2, the same performance limitations apply.

4. Reliability

The SNAP 10B Basic reactor utilizes axially moving control segments. The reactor operates under static control at 100 kwt and 1300°F coolant outlet temperature. The 37-element core is the same as that specified for the SNAP 10A and SNAP 10A/2 reactor. Use of an improved hydrogen barrier material capable of operation at the higher coolant outlet temperature provides the same reactor subsystem reliability as quoted for Interim SNAP 10A/2. No appreciable change in reliability is associated with the use of axially sliding control segments, since these segments are not required to move after the first 3 days of operation at design power and temperature.

E. SNAP 10B ADVANCED

The SNAP 10B Advanced reactor concept also assumes static control and incorporates the sliding reflector control segments and tapered reflector introduced in the SNAP 10B Basic. The advanced feature of this reactor is the introduction of an additional reactivity loss compensation technique. Some method of compensating long-term reactivity losses (primarily hydrogen leakage) may be desirable to reduce the potentially large end-of-life temperature uncertainties associated with high temperature static control operation. Such reduction allows attainment of higher powers or improves the behavior of a lower-power reactor in a given system. Attainment of similar power levels can be achieved without the additional compensation technique if uncertainties in hydrogen leak rate can be reduced to approximately half of the presently predicted value.

1. Compensation Techniques

Several reactivity loss compensation techniques have been investigated and appear promising. They may be divided roughly into two categories: (1) In-core "getters" which chemically sieze and hold hydrogen moderator, leaking from fuel elements, within the active core region, thereby reducing temperature uncertainties by reducing the magnitudes of the reactivity losses and hence the uncertainties in the magnitude; and (2) Temperature Coefficient Augmenters (TCA's), thermomechanical elements which provide a gross temperature coefficient of reactivity of about -2.5¢/°F (unaugmented coefficients for SNAP reactors are -0.25 to -0.35 ¢/°F), thereby reducing the temperature uncertainties associated with a given reactivity uncertainty. The most promising getter consists of a metallic yttrium film appropriately distributed within the active core volume. At present it is envisioned that this film will be deposited on the outside of the fuel element cladding. A conceptual design of the SNAP 10B fuel element is shown in Figure III-15.

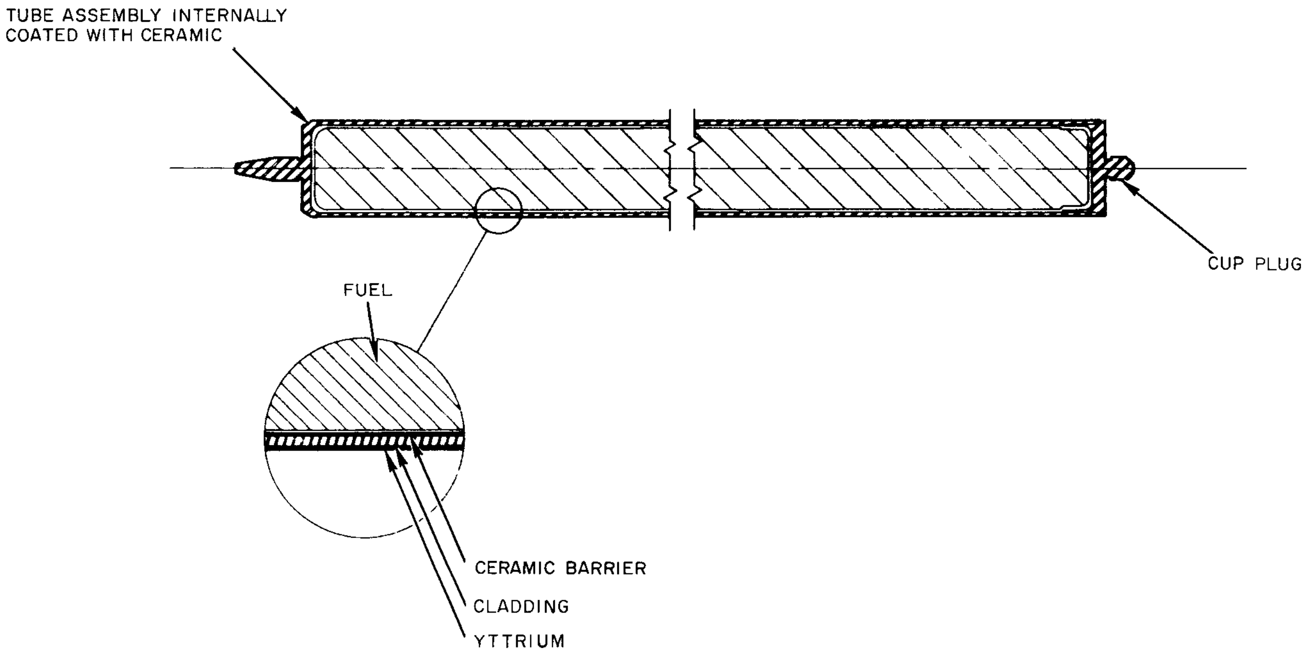

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Figure III-15. Fuel Element With Yttrium Getter

Several TCA's have been designed, employing active grid plates or mechanisms for producing reflector position perturbations. The TCA and getter approaches are viewed as two parallel alternative approaches to the same goal. Feasibility tests are now underway to determine the suitability of these techniques in SNAP reactors. The more promising of the two may then be used in the reference design for the 10B Advanced reactor. Should both approaches be capable of successful development, they might simultaneously be employed on the reactor. Preliminary analysis indicates that no significant improvement in nominal performance will be achieved by adding the TCA device to a reactor already employing the getter. Significant improvement in reliability (for example, in the case of a failed fuel element) may be gained in this manner, however, at some expense in increased reactor weight.

Use of a TCA device imposes a need for more accuracy in the startup temperature sensing. The very nature of temperature coefficient augmentation requires that a reactivity "bank" be established that can be "drawn upon" later to resist drops in temperature. If a system is to resist temperature drops by adding a certain amount of reactivity, say 2.5¢, for every degree drop in temperature of the core, then this amount of reactivity (2.5¢/°F) must have been removed from the reactor as the temperature was initially increased. Obviously such a loss of reactivity over the entire temperature rise during startup would lead to a prohibitive temperature defect. It is therefore necessary to design the TCA device to be activated at some temperature differential, say 50°F, below the anticipated maximum temperature of the reactor. Since errors in activation temperature can lead to large reactivity losses, it is essential that the device be activated at the specified temperature, therefore complicating the startup problem.

2. Concept Scope

The performance capabilities of the 10B Advanced reactor have been explored over a wide range of reactor designs. Several facts have emerged. It appears that a nominal (end-of-life) outlet temperature of 1300°F is a generally practical and satisfactory objective for static-controlled reactors, based upon the probable requirements of likely power conversion systems. Substantial advantages can be obtained by varying the size and number of fuel elements in the reactor core. The 10B Advanced reactor can be favorably designed for lifetimes of 5 yr or greater.

In the light of these conclusions, a number of specific reactor designs were examined. Reactor lifetimes of 1, 3, and 5 yr were selected as compatible with obtainable PCS and other supporting technology, and appropriate to the range of probable mission requirements. In addition to a 37-element core, lattices of 55 and 85 elements were also considered. The latter are conveniently attainable triangular arrays and represent reasonable increments in element number while holding a constant core diameter of 8.875 in. Fuel element diameters for the 55- and 85-element designs were 1.06 and 0.855 in., respectively. Active fuel element lengths were constrained to fall in the range 11 to 15 in. Variable amounts of prepoison loading were also permitted.

3. Performance Capabilities

The SNAP 10B Advanced concept has not yet evolved to the point where it is desirable to "freeze" specific performance requirements and design features. Nine tentative design power levels have been selected to illustrate the scope of the concept. These correspond to the three core arrays and three lifetimes mentioned in the preceding section. The work upon which the designs were based was preliminary in nature, and the power levels should not be construed as optimum for their respective life and geometry assumptions.

The following tables give the power levels for the tentative design selections. In each case an attempt was made to select an estimated "most favorable" power level; i.e., a practically favored tradeoff of maximum power vs weight. For the selected designs, Table III-9 shows maximum power capabilities of the reactors, operating at one or more design criterion limits. Table III-10 gives, for the same designs, nominal power ratings which include additional conservatism and provide a degree of reactivity redundancy through startup.

The numbers shown in Tables III-9 and III-10 are not intended to imply that all of the designs noted are attractive from application or cost effectiveness viewpoints. For example, the 37-element 10B Advanced does not constitute a large increase in power capability (less than 2X) over the 10B Basic or the Upgraded 10A/2, and does involve a substantial nuclear development program.

The implication that the 10B Advanced concept is limited to power levels below 450 kwt must also be avoided. Longer fuel elements or a larger number of thinner elements would permit such extension. This approach has not yet been adequately studied, and it may prove more effective to consider application of getter or TCA techniques to a member of the SNAP 8 family of reactors.

4. Specific Design Descriptions

In order to give a more detailed picture of the 10B Advanced reactor capabilities, the characteristics of a number of specific preliminary reactor designs have been calculated and are presented in the following paragraphs.

a. 85-Element Core

The 85-element core has been selected as one illustrative example, in both the getter and TCA variations. Tables III-11 and III-12 list some of the important

TABLE III-9

10B ADVANCED MAXIMUM POWER CAPABILITIES
(OPERATION AT DESIGN CRITERION LIMITS)

No. of Elements	Lifetime (yr)		
	1	3	5
	kwt		
37	240	150	100
55	325	200	140
85	450	270	190

TABLE III-10

10B ADVANCED NOMINAL POWER RATINGS

No. of Elements	Lifetime (yr)		
	1	3	5
	kwt		
37	180	125	100
55	240	150	110
85	325	190	135

TABLE III-11

10B ADVANCED (GETTER) REACTOR
NOMINAL DESIGN CHARACTERISTICS

Power level (end of life)(kwt)	325
Outlet temperature (end of life)(°F)	1300
Coolant temperature rise (°F)	200
Number of elements	85
Cladding Material	Hastelloy N
Element OD (in.)	0.855
Maximum fuel temperature (°F)	1590
Prepoison loading ($)	2.30
N_H (10^{22} atoms H/cc fuel)	6.3
Hydrogen leakage (%/yr)	3.8
Barrier Material	SCB
Core length (in.)	13.8
Core diameter (in.)	8.875
Reflector thickness (in.)	3.0 (mean)
Shielded diameter at top of core (in.)	14.6
Distance from shield top to core top (in.)	18.9
Control method	static
Active control drums	1
Reactor-Reflector weight (lb)	388
Liftime (yr)	1
Power density (kwt/lb reactor)	0.92
Power density (kwt/in.^{3} fuel)	0.54
Maximum burnup (metal at.%)	0.17
Reactivity Contingency ($)	0.50

*Adjusted by using low N_H elements and/or reflector shimming

TABLE III-12

10B ADVANCED (TCA) REACTOR
NOMINAL DESIGN CHARACTERISTICS

Power level (end of life)(kwt)	325
Outlet temperature rise (°F) (°F)	1300
Coolant temperature rise (°F)	200
Number of elements	85
Cladding Material	Hastelloy N
Element OD (in.)	0.855
Maximum fuel temperature (°F)	1590
Prepoison loading ($)	4.20
N_H (10^{22} atoms H/cc fuel)	6.3
Hydrogen leakage (%/yr)	3.8
Barrier Material	SCB
Core length (in.)	13.8
Core diameter (in.)	8.875
Reflector thickness (in.)	3.0 (mean)
Shielded diameter at top of core (in.)	14.6
Distance from shield top to core top (in.)	18.9
Control method	static
Active control drums	1
Reactor-Reflector weight (lb)	413
Lifetime (yr)	1
Power density (kwt/lb reactor)	0.86
Power density (kwt/in.^{3} fuel)	0.52
Maximum burnup (metal at.%)	0.17
Reactivity Contingency ($)	0.50

*Adjusted by using low N_H elements and/or reflector shimming design parameters for the nominal 1-yr lifetime. Figures III-16 through III-18 show off-design power capabilities as a function of outlet temperature and lifetime. The ultimate power capability is limited to 450 kwt at 1300°F by the cladding creep design criterion. Below this temperature, the reactor is reactivity limited.

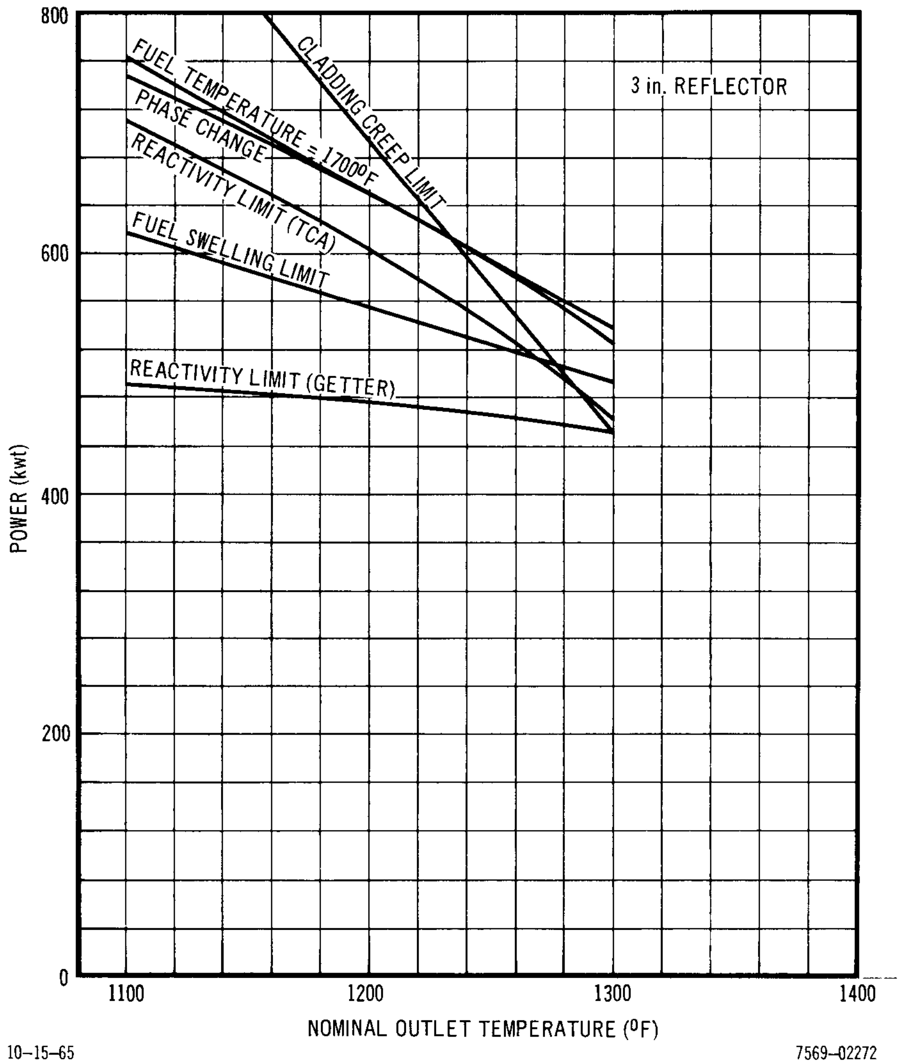

Figure III-16. SNAP 10B Advanced (85 Elements) Parameters, 1-yr Life

It should be noted that these designs were optimized for a 1-yr life, and do not necessarily constitute optimum designs for off-design longer lifetimes. One consequence of optimizing both systems to the same 1-yr lifetime is to emphasize

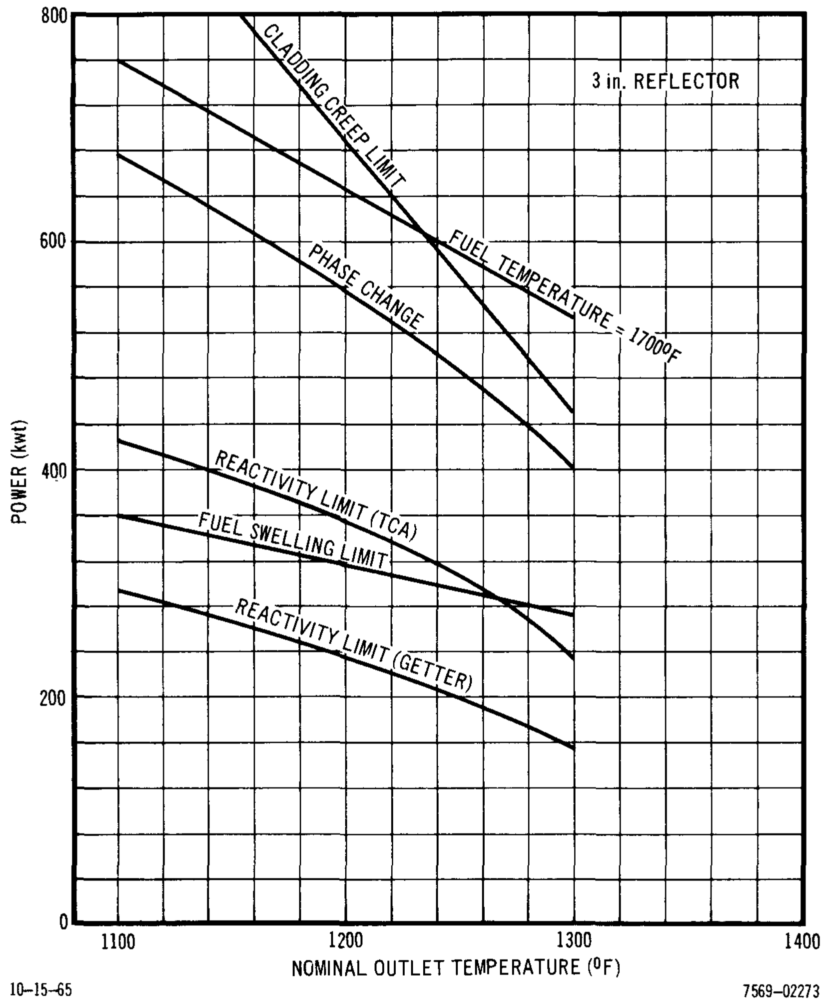

Figure III-17. SNAP 10B Advanced (85 Elements) Parameters, 3-yr Life

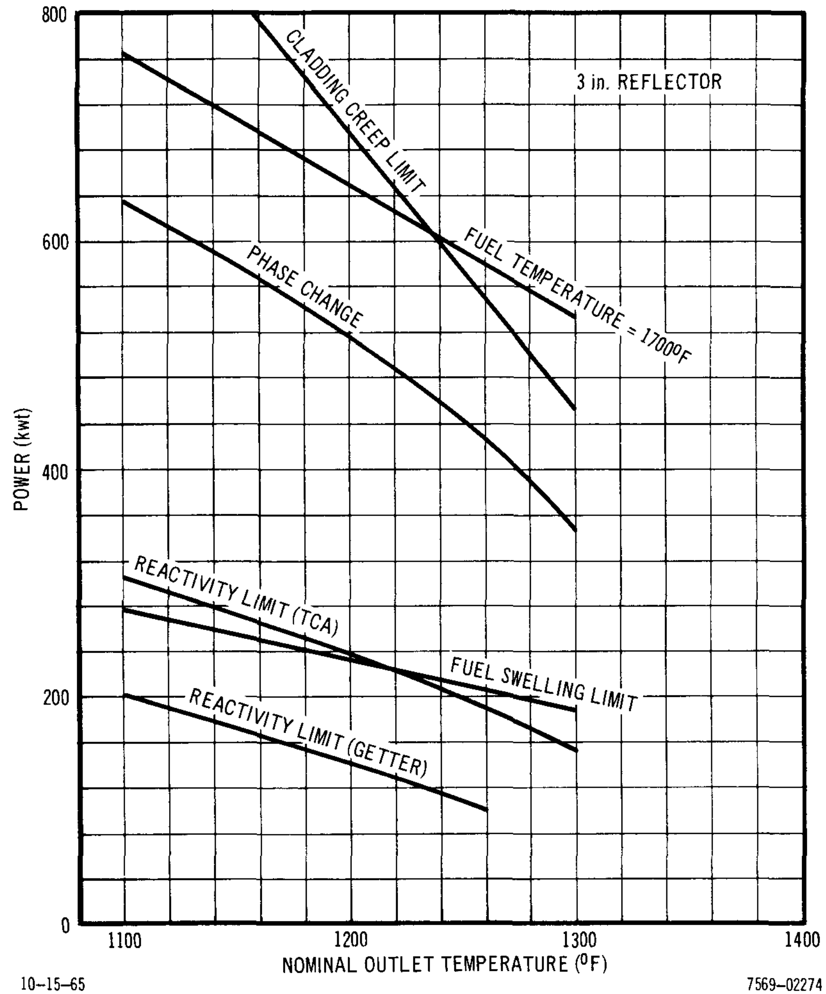

Figure III-18. SNAP 10B Advanced (85 Elements) Parameters, 5-yr Life

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the difference between the two compensation techniques for off-design extended lifetimes. Comparison of the figures shows that the getter design has lower power capabilities than the TCA design when operated under these conditions. This is a consequence of the limited hydrogen absorption capability of the optimized getter. Reoptimization of both designs at a longer lifetime, however, would be expected to yield reactor systems of comparable weights.

b. 55-Element Core Optimized at 100 kwt

Selection of the 9 tentative design points shown in Section III-E-3 was based heavily on the maximum practical power capability obtainable with a given number of fuel elements of reasonable length. For a given power level, it is possible to arrive at a design which provides a better optimization of reactor weight and shieldable envelope. Separate analyses have shown that for optimizations at power levels below about 300 kwt maximum capability, the 55-element core lattice choice provides an optimized design with a slight system weight saving over both 37- and 85-element cores.

A description of a 55-element design, optimized at 100 kwt, is presented in Tables III-13 and III-14 and in Figures III-19 through III-21. This is a design point for which considerable interest appears to exist. Its analysis here also permits comparison of the 10B Advanced concept with the 10B Basic and Upgraded 10A/2 reactors previously discussed. In these comparisons, the reactor weight saving is negligible. More significant, however, is the reduction of shieldable diameter to 12.6 in. In comparison to SNAP 10B Basic, this would result in a shield weight reduction of about 13% (see Section IV, Shielding). In comparison to Upgraded 10A/2, the shield weight saving is even more striking.

A detailed account of the optimization of this reactor is presented in a published AEC report.*

c. 85-Element Core Optimized at 100 kwt

The 85-element reactor described in (a) above may be operated at 100 kwt. It is also possible to obtain rated outlet temperature and life at this power level with a reduced reflector, such that the shielded envelope is only slightly larger than that of the 55-element reactor described in (b) above. While such a design

*R. J. Gimera, "SNAP 10B Reactor Conceptual Design," NAA-SR-10422 (October 31, 1964)

TABLE III-13

10B ADVANCED (GETTER) REACTOR
55-ELEMENT CORE OPTIMIZED AT 100 kwt
NOMINAL DESIGN CHARACTERISTICS

Power level (end of life) (kwt)	100
Outlet temperature (end of life) (°F)	1300
Coolant temperature rise (°F)	100
Number of elements	55
Element OD (in.)	1.06
Maximum fuel temperature (°F)	1440
Prepoison loading ($)	1.25
N_H (10^{22} atoms H/cc fuel)	6.3
Hydrogen leakage (%/yr)	2.5
Barrier material	SCB
Core length (in.)	13.6
Core diameter (in.)	8.875
Reflector thickness (in.)	2.0 (mean)
Shielded diameter at top of core (in.)	12.6
Distance from shield top to core top (in.)	18.0
Control method	static
Active control elements	1
Reactor-Reflector weight (lb)	337
Lifetime (yr)	1
NaK flowrate (lb/hr)	16,250
NaK ΔP (psi)	0.4
Fuel element cladding	Hastelloy N
Power density (kwt/lb reactor)	0.33
Power density (kwt/in.^{3} fuel)	0.17
Maximum burnup (metal at.%)	0.05
Reactivity Contingency ($)	0.50

TABLE III-14

10B ADVANCED (TCA) REACTOR
55-ELEMENT CORE OPTIMIZED AT 100 kwt
NOMINAL DESIGN CHARACTERISTICS

Power level (end of life) (kwt)	100
Outlet temperature (end of life) (°F)	1300
Coolant temperature rise (°F)	100
Number of elements	55
Element OD (in.)	1.06
Maximum fuel temperature (°F)	1440
Prepoison loading ($)	1.90
N_H (10^{22} atoms H/cc fuel)	6.3
Hydrogen leakage (%/yr)	2.5
Barrier material	SCB
Core length (in.)	13.6
Core diameter (in.)	8.875
Reflector thickness (in.)	2.0 (mean)
Shielded diameter at top of core (in.)	12.6
Distance from shield top to core top (in.)	18.0
Control method	static
Active control drums	1
Reactor-Reflector weight (lb)	362
Lifetime (yr)	1
NaK flowrate (lb/hr)	16,250
NaK ΔP (psi)	0.4
Fuel element cladding	Hastelloy N
Power density (kwt/lb reactor)	0.30
Power density (kwt/in.^{3} fuel)	0.16
Maximum burnup (metal at.%)	0.05
Reactivity Contingency ($)	0.50

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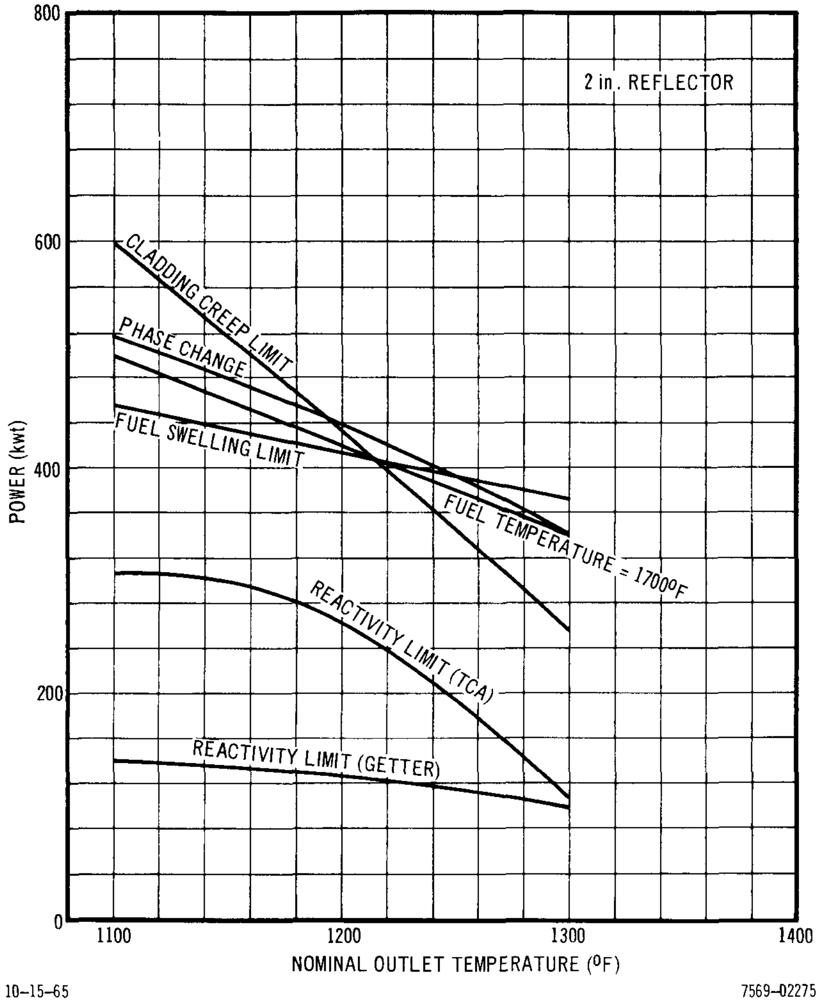

Figure III-19. SNAP 10B Advanced (55 Elements) Parameters, 1-yr Life

would not be optimum for 100-kwt application, the reactor and shield weight penalty would be small. The advantage of developing such a reactor would be its potential versatility, since its power capability could be extended to 450 kwt by the relatively inexpensive process of adding reflector thickness. In the limit, this process would approach the weight and envelope described in (a) above. A detailed evaluation of this approach has not yet been performed.

5. Performance Limitations

All potential performance limitations considered for the 10B Basic reactor were also considered here.

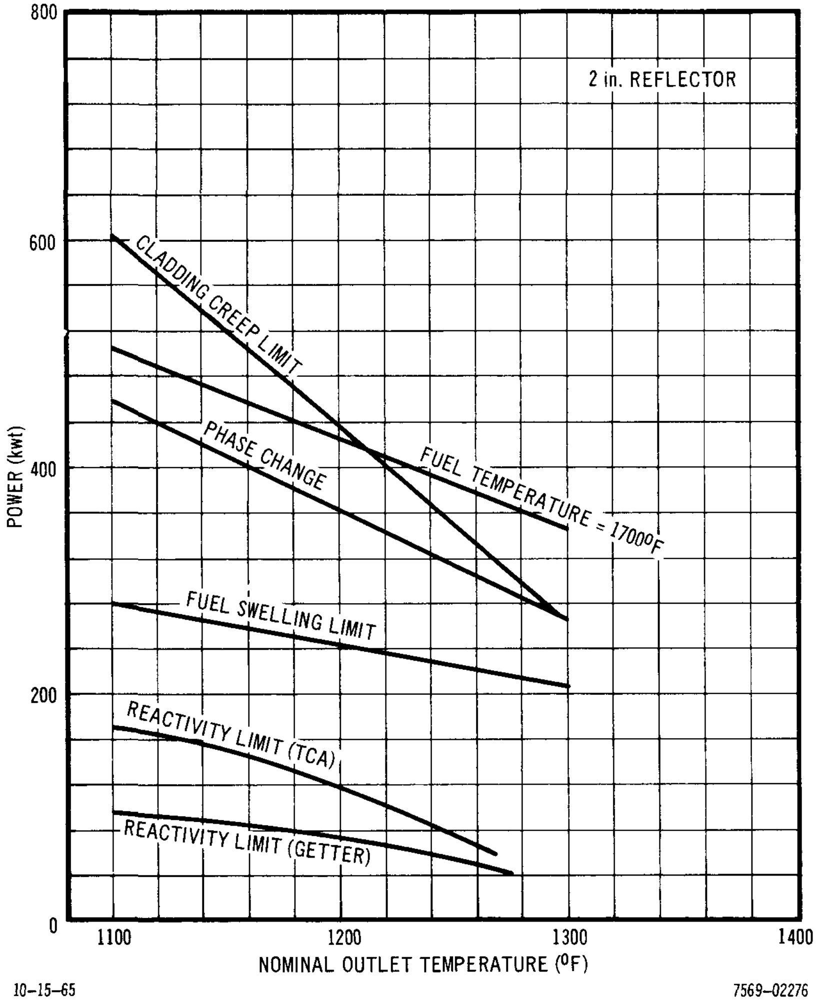

Figure III-20. SNAP 10B Advanced (55 Elements) Parameters, 3-yr Life

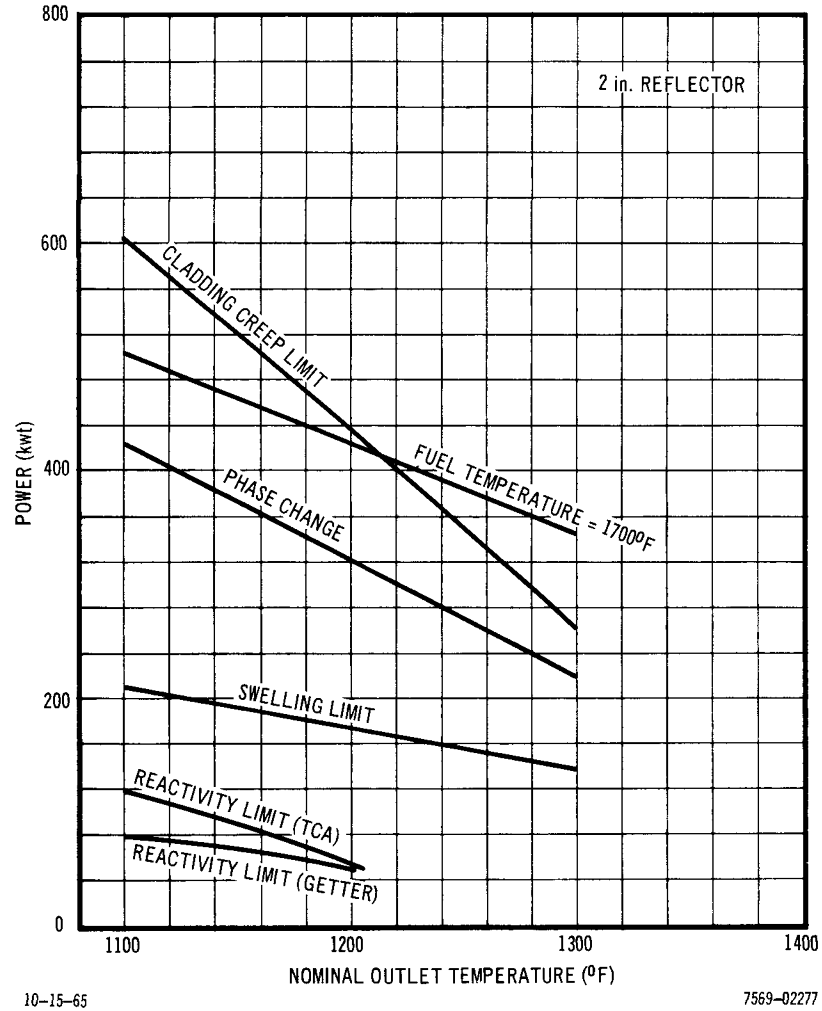

Figure III-21. SNAP 10B Advanced (55 Elements) Parameters, 5-yr Life

a. 85-Element Core

The performance limitations for the 85-element reference core are shown in Figures III-16 to III-18.

(1) One-Year Operation

The getter version is reactivity limited and is capable of operation at 500 kwt/ 1100°F, 480 kwt/1200°F, or 450 kwt/1300°F.

The TCA version is fuel-swelling-limited at temperatures below 1265°F and cladding-creep-limited above that temperature. If the creep limitation is removed by raising the NaK system pressure, the reactor can produce 620 kwt/ 1100°F, 550 kwt/1200°F, or 460 kwt/1300°F.

It is possible that some raising of the fuel swelling limit can be accomplished by optimization of the fuel-cladding gap. A value of 3-mil radial clearance (hot) was used in this study. An increase in the gap will increase fuel temperatures and therefore lower the fuel temperature limit line while raising the fuel swelling limit line somewhat. Since there is very little space between the fuel swelling and fuel temperature limit lines, probably little can be gained by attempts at gap optimization.

(2) Three-Year Operation

The getter version is completely reactivity limited. Operation at 300 kwt/ 1100°F, 230 kwt/1200°F, or 150 kwt/1300°F is possible.

The TCA version is fuel-swelling-limited over almost the entire range of operating temperatures. Operation is limited to 365 kwt/1100°F, 320 kwt/ 1200°F, or 230 kwt/1300°F. In this case, some improvement in performance may be gained by optimizing the fuel element gap, although probably at the expense of 1-yr operation characteristics.

(3) Five-Year Operation

The getter version is again completely reactivity limited. Operation at 200 kwt/1100°F, 140 kwt/1200°F, or 100 kwt/1250°F is possible.

The TCA version is fuel-swelling-limited below 1220°F and reactivity-limited above that temperature. Operation is limited to 270 kwt/1100°F, 230 kwt/1200°F, or 150 kwt/1300°F. Some slight improvement in performance may be gained by optimization of the fuel element gap, although again probably at the expense of 1-yr operation characteristics.

b. 55-Element Core Optimized at 100 kw

It should be emphasized that the limitations presented here are limitations for a particular low-powered 55-element core. Larger reactors could be built with 55 elements, but they would lack the high-power capability of the 85-element core.

Since this core is designed well below the maximum power levels attainable with a 55-element core, reactivity limitations dictate operational characteristics, given in Table III-15. All the potential limitations are shown in Figures III-19 through 21. For lifetimes longer than 1 yr, only very low powers can be extracted.

TABLE III-15

REACTIVITY LIMITATIONS 55-ELEMENT CORE OPTIMIZED AT 100 kwt

Version	Life (yr)	Temperature (°F)	Power (kwt)
Getter	1	1100	140
	1	1200	125
	1	1300	100
	3	1100	90
	3	1200	70
	3	1250	50
	5	1100	75
	5	1200	45
TCA	1	1100	310
	1	1200	265
	1	1300	110
	3	1100	165
	3	1200	115
	3	1250	70
	5	1100	115
	5	1200	50

6. Reliability

The SNAP 10B Advanced reactor employs advanced design features which nullify the nuclear effect of hydrogen leakage. The use of either yttrium as a hydrogen "getter" or employment of a mechanical temperature coefficient augmenting device is specified. Reliability allocations for the SNAP 10B reactor subsystem are shown in Table III-16.

TABLE III-16

SNAP 10B ADVANCED REACTOR SUBSYSTEM RELIABILITY ALLOCATION

Assembly	Cumulative Reliability		
	Launch and Ascent	Startup	1 yr
Reactor structure	0.99980	0.99950	0.99930
Reactor core	0.99900	0.99700	0.99460
Reflector	0.99980	0.99880	0.99860
Control equipment	0.99920	0.99620	0.99570
Radiation shield	0.99980	0.99950	0.99930
Complete reactor subsystem	0.99760	0.99100	0.98750

F. SNAP 8 REFERENCE DESIGN

1. Description

The SNAP 8 Reference Design nuclear system consists of a shielded compact nuclear reactor moderated by zirconium-hydride and cooled by liquid NaK-78. It is conservatively designed to produce 600 kwt for 10,000 hr of unattended operation in space with a 1300°F NaK outlet coolant temperature and an 1100°F inlet temperature. The reference system is designed to operate with various power conversion systems such as (1) the 35-kwe, 3-loop Hg-Rankine system being developed by NASA, (2) the compact thermoelectric power conversion system under development by the AEC as an outgrowth of the SNAP 10A program, (3) the thermoelectric SNAP 10A power conversion system, (4) the multiple installation of large combined rotating units (CRU) of the Hg-Rankine cycle being developed by the AEC as an outgrowth of the SNAP 2 program, and (5) the multiple installation of SNAP 2 type CRU's. Figure III-22 shows the SNAP 8 Reference nuclear system.

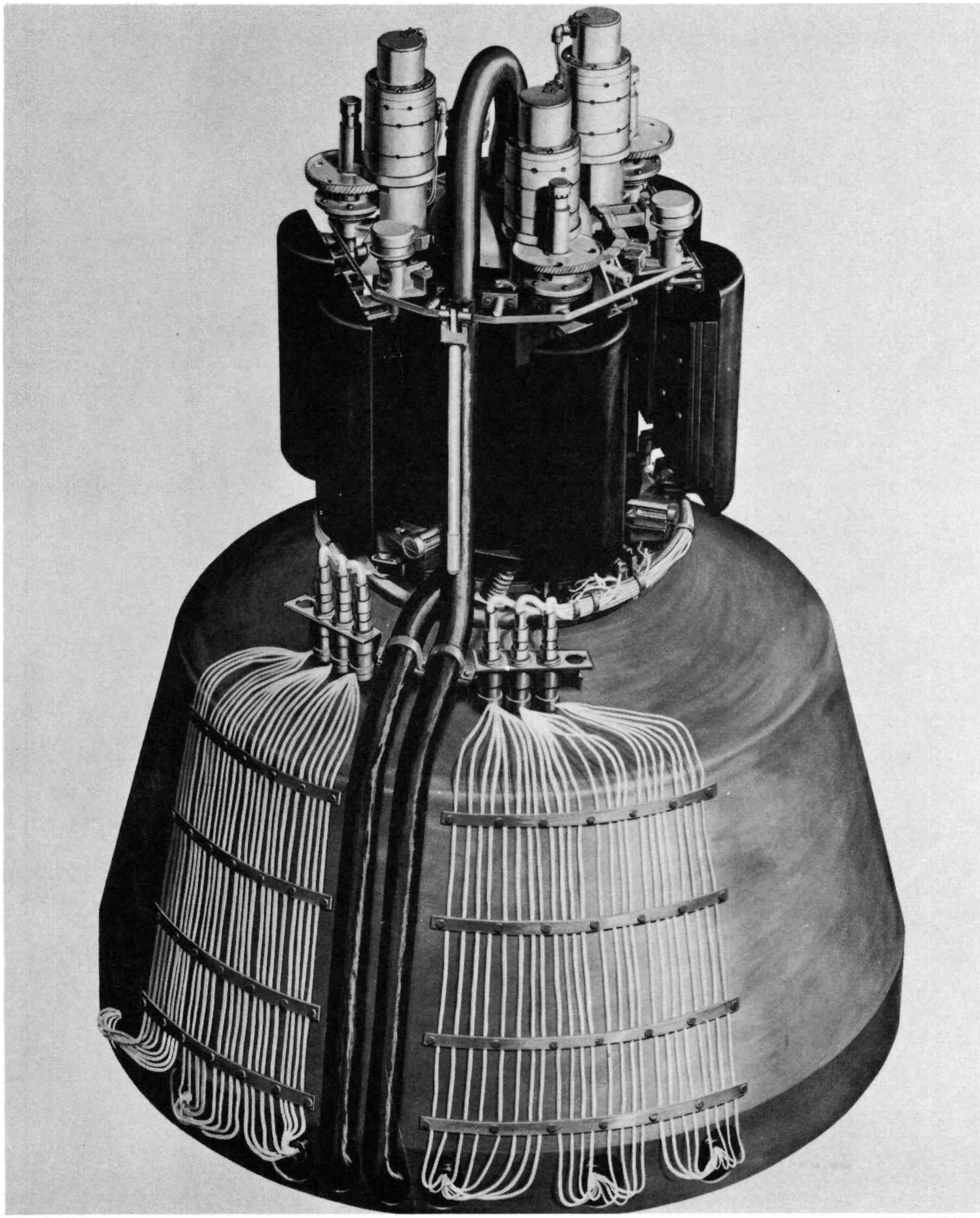

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Figure III-22. SNAP 8 Reactor and Shield

The design reliability goal for the SNAP 8 nuclear system has been tentatively established at 96.85% excluding micrometeoroid protection. The goal for non-puncture probability has been established at 99.9%. The design weight objective, exclusive of the shield, is 600 lb. The weight of the shield will depend largely upon the selected mission and payload criteria. The environmental criteria are based upon a SATURN launch vehicle and are detailed in the SNAP 8 Nuclear System Model Specification, NR7568-02. A description of the major assemblies of the SNAP 8 Reference Design nuclear system is given in a published AEC report.*

The core vessel is a right circular cylinder containing 211 fuel elements. The fuel elements are made of Zr - 10.5 wt % U alloy hydrided to a hydrogen content of 6×10^{22} hydrogen atoms/cc of $(U\text{-}Zr)H_x$. The uranium is 93% enriched. The elements are clad in 10-mil thick Hastelloy-N tubing, coated internally with a hydrogen permeation barrier. The active length of the fuel element is 16.825 in.

The beryllium reflector system consists of six stationary cusp-prisms and six rotating drums. Three reflector drums are used as startup drums and are driven by springs. The other three reflector drums are used as control drums and are driven by long-term bidirectional motors.

The shield assembly is dependent upon the payload diameter, separation distance, and allowable dose. The shield must protect the payload from scattered neutrons and therefore must shield the entire reflector assembly, 27.6 in. in diameter at approximately 22 in. above the bottom of the core vessel.

The SNAP 8 Developmental System, S8DS, based upon the Reference Design is currently being fabricated.

2. Nominal Design Conditions

Table III-17 summarizes the important design criteria of the SNAP 8 Reference Design nuclear system.

3. Parametric Capabilities of the Reference SNAP 8 Nuclear System

The power capability of the SNAP 8 reactor is a function of both required lifetime and coolant outlet temperature. Since the present reference design is

*"SNAP 8 Quarterly Progress Report, May-July 1964," NAA-SR-9992, (SRD) (September 22, 1964)

TABLE III-17

SNAP 8 REFERENCE DESIGN NUCLEAR SYSTEM

Power level (kwt)	600
Outlet temperature (°F)	1300
Coolant temperature rise (°F)	200
Number of elements	211
Element OD (in.)	0.560
Maximum fuel temperature (°F)	1520
Prepoison loading ($)	3.00
% Hydrogen leakage (yr)	2.4
Barrier material	SCB
Core length (in.)	16.825
Core ID (in.)	9.214
Reflector thickness (nominal)	3.0
Shielded diameter (shoulder) (in.)	27.6
Shoulder height (in.)	22.0
Control method	active
Controlled reflector elements	3
Reactor-reflector weight (lb)	600
Nominal lifetime (hr)	10^4
NaK flowrate	48,800
NaK ΔP (psi)	4.8
Fuel element cladding	Hastelloy-N

highly evolved, and essentially "frozen," it is possible to define an envelope of capabilities for this design, bounded by curves associated with various critical parameters. Two basic types of limiting curves may be defined; those based on "design criteria," and those based on "operational limitations."

Both design criteria and operational limitations are descriptions of reactor conditions which occur or evolve in the course of reactor operation. Design criteria limits are highly conservative estimates of the lowest power level at which, in the life-temperature domain, system performance could be expected to begin to degrade as the result of exceeding some physical condition. Design criteria are defined administratively, and do not bear any specific relationship to currently demonstrated technology or detailed engineering arguments. Operational limits are realistic but still conservative estimates, based upon currently established technology and data.

An indication of the conservatism of the design criteria will be shown by analyzing the reactor power, outlet coolant temperature, and lifetime capabilities for various physical limitations. With minor modifications to the beryllium drum size and hydrogen moderator concentration with the fuel, the existing SNAP 8 reactor can be expected to produce over 1 Mwt power for 10^4 hr with an outlet coolant temperature of 1300°F. However, at this higher power level the nuclear system's reliability would be significantly decreased due to the utilization of all excess (or redundant) reactivity (see Section III-F-4). Improved reliability and lifetime result from operation below nominal power. No weight savings can be realized, however, since the physical design is fixed.

a. SNAP 8 Reference Design Criteria

Design criterion limits are shown in Figure III-23, and are compared to operational limits in Figure III-24. All five of the basic design criteria of the reactor exceed the reactor's design point. The five basic design criteria are excess reactivity, fuel swelling, fuel phase change, cladding creep, and beryllium swelling. The assumptions and reasoning used in determining each design criterion, as well as a comparison between the conservative design criterion and the more realistic operating capability, are described in the subsequent sections.

(1) Excess Reactivity

The ultimate operational limitation of the SNAP 8 Reference Design nuclear system is excess reactivity. The reactivity lifetime is that time at which all excess reactivity has been used. The excess reactivity design criterion assumes the reactor operating with the:

1) Nominal core power fuel burnup and fission product burnup

2) Temperature defect corresponding to the hot channel fuel temperatures

3) Hydrogen loss of the hot channel fuel element with the fuel temperature remaining constant during the reactor lifetime

4) Nominal fuel specification $N_H = 6.05$

5) Nominal shim configuration.

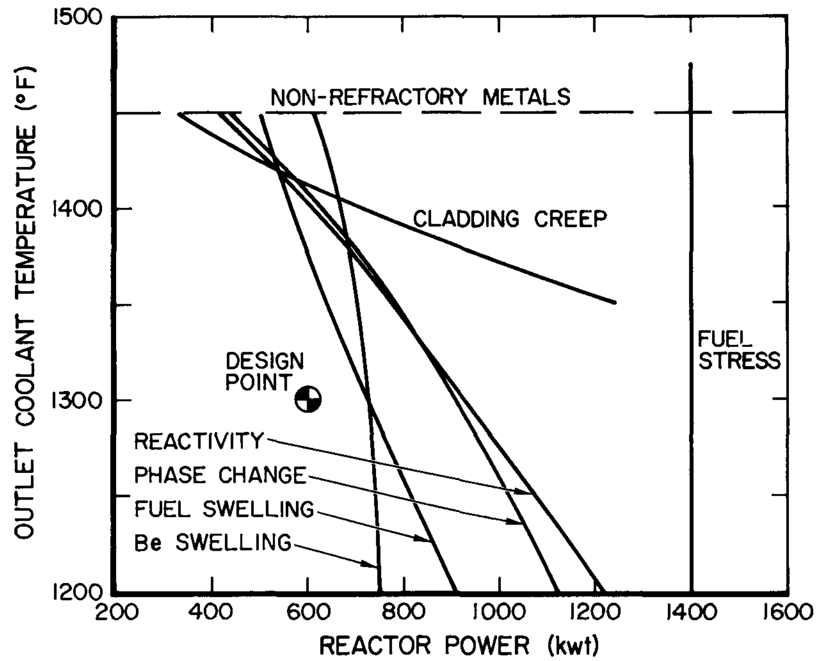

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Figure III-23. Reference SNAP 8 — Design Criteria Limits

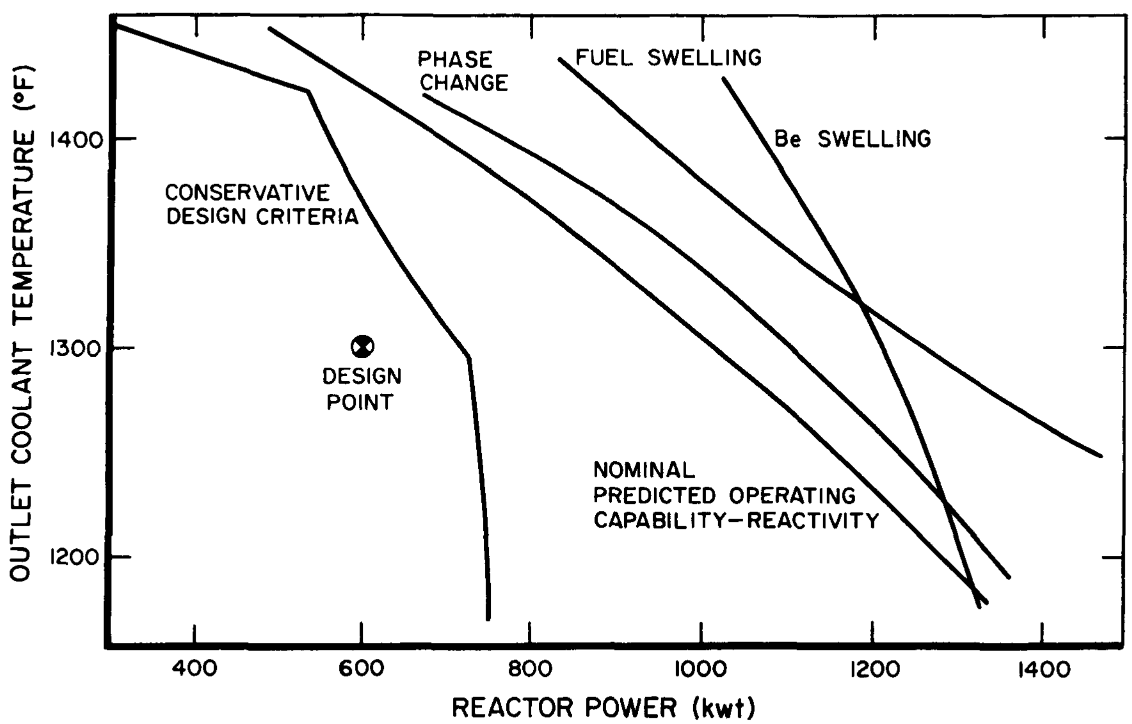

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Figure III-24. Reference SNAP 8 — Operational Limits

As shown on Figure III-23, the design criteria of the SNAP 8 Reference Design nuclear system having a 10,000-hr lifetime is not limited by excess reactivity. For outlet coolant temperatures of 1200°F, 1300°F, and 1400°F the excess reactivity design criteria limits the reactor power to approximately 1210 kwt, 930 kwt, and 600 kwt, respectively.

During reactor operation, the four major reactivity requirements are temperature defect, fission product buildup, fuel burnup, and hydrogen moderator loss. Since the only major reactivity requirement which can be varied is hydrogen loss, three reactor design criteria, fuel swelling, fuel phase change, and cladding creep are specified to maintain the hydrogen leakage within acceptable limits. These design criteria are specified to minimize or eliminate the strain on the ceramic hydrogen barrier encountered during the reactor lifetime.

(2) Fuel Swelling

The fuel swelling design criterion is defined as operation until, at the point of maximum burnup of the fuel rod under hot channel conditions, the gap between the fuel and the ceramic barrier is zero. The hot channel fuel temperature assumes that there is a uniform hydrogen gap between the fuel and the ceramic, that the fuel rod produces 10.8% more power than the nominal center rod, and that the coolant flow is 10% below the specified flow profile. The power hot channel factor is based on a statistical evaluation of the reactor specification tolerance variation of the system components.

The nominal fuel temperatures assume that the fuel touches the ceramic, nominal core power, and full coolant flowrate of the required flow profile. The coolant flow profile is based on minimizing the maximum radially-averaged fuel temperature within the reactor. This flow profile appears to be near optimum to minimize fuel and ceramic temperatures which lower fuel swelling, hydrogen dissociation pressure, and hydrogen leakage.

The restriction that the fuel does not strain the ceramic has been experimentally shown to be conservative. Tests were run to determine the effect of the ceramic barrier hydrogen leak rate by controlled straining of the ceramic by simulated fuel swelling. The tests showed that the ceramic coating could be strained 4 mils on the diameter with no detrimental effect on the hydrogen leak rate. The effect of neutron flux or fission products on the ability of the ceramic to strain without significantly increasing the hydrogen leak rate has not been directly determined. It appears that the nuclear environment should have no discernible effect on the ceramic degradation due to strain since in-pile leakage of nonstrain ceramic barriers indicated no significant change in hydrogen leakage.

A second conservative assumption in the fuel swelling design criterion is that the gas gap is closed when the fuel element has been returned to ambient temperature and the hydrogen within the fuel is uniformly distributed. When the reactor is at operating temperatures, there is approximately one additional mil of diametral clearance between the fuel and ceramic. This constraint is pertinent only to reactors which must be restarted after extensive power operation, and is incorporated principally to permit ground testing of hardware identical to flight hardware.

The third conservative assumption in the fuel swelling design criterion is that the fuel temperature remains constant; that is, independent of the gas gap during the reactor lifetime. The principal temperature drop between the fuel and the coolant is across the hydrogen gas gap.

However, during the reactor lifetime, as the fuel swells, the gas gap decreases. The fuel temperature will therefore decrease during the reactor lifetime. (The effect of fission fragments produced within the fuel on the fuel thermal conductivity will be minor.)

The fuel element design provides adequate axial clearance to accommodate the axial growth of the fuel due to irradiation damage. Because of this clearance, axial growth of the fuel is not considered in the establishment of the design limits to the performance of the system.

Figure III-23 illustrates that fuel swelling is one of the limits of the operating range of the reactor functioning within its conservative design criteria. For outlet coolant temperatures of 1200°F, 1300°F, and 1400°F, the fuel swelling design criterion would limit the reactor power to approximately 900 kwt, 720 kwt, and 560 kwt, respectively. These conditions result from the conservative fuel swelling design criterion. Operation beyond this limit should be possible without any significant decrease in the overall system performance capability.

(3) Fuel Phase Change

The second reactor design criterion required to minimize the hydrogen leak rate is the phase change of the fuel material. The fuel phase change design criterion is defined as operation until any part of the fuel element (with the lowest fuel specification hydrogen concentration operating in the hot channel condition) loses sufficient hydrogen to just reach the β-δ phase interface. Like fuel swelling, the phase change design criterion is quite conservative and assures no straining of the ceramic barrier.

The design criteria assume that:

1) The fuel temperature profile corresponds to the hot channel condition and that the fuel temperature remains constant during the reactor lifetime.

2) The hydrogen concentration (N_H) corresponds to the minimum fuel specification value $N_H = (6.05 - 0.10) = 5.95$.

3) The hydrogen leak rate through the ceramic corresponds to the S8ER median value, which is 2 to 3 times the S8DS fuel element leak rate.

4) The hydrogen leak rate of a fuel element decreases with time due to the lost hydrogen diminishing the N_H which correspondingly reduces the hydrogen dissociation pressure.

Figure III-23 exemplifies that the fuel phase change design criterion does not limit the reactor operating with a 10,000-hr lifetime. The reactor could operate at powers of 1020 kwt, 910 kwt, and 630 kwt with outlet coolant temperatures of 1200°F, 1300°F, and 1400°F, respectively, before the phase change design criterion would be exceeded.

(4) Cladding Creep

The third fuel element design criterion specified to maintain the integrity of the ceramic coating is cladding creep. The hydrogen dissociation pressure of the fuel is many atmospheres for fuel rods having hydrogen concentrations near the upper limit of the specification, having a high power density, and/or having a high outlet coolant temperature. The counteracting effect of increasing the static pressure of the NaK is bounded by another design criterion — core vessel creep. The desired cladding strain cannot be obtained by increasing the cladding wall thickness since, from a reactivity viewpoint, the cladding must be as thin as possible. The 10-mil nominal wall thickness appears to be a reasonable lower limit due to cladding extrusion and fuel element manufacturing techniques. The 0.2% lifetime creep, using a 1.4 factor of safety, is a somewhat arbitrarily small number to insure optimum hydrogen barrier operation. Excessive cladding creep could have two detrimental effects. First, strain of the ceramic barrier, and second, approaching the estimated 1% diametral cladding strain rupture limitation.

The design criterion assumes that:

1) The fuel temperature corresponds to the hot channel and that the fuel temperature remains constant during the reactor lifetime.

2) The N_H corresponds to the maximum fuel specification limitation (6.05 + 0.10 = 6.15).

3) The allowable stress has a factor of safety of 1.4.

4) The hydrogen pressure remains constant during the reactor lifetime instead of decreasing due to hydrogen leakage diminishing the N_H.

5) Minimum specification cladding thickness (10 - 1 = 9 mils).

The cladding material properties deteriorate quite rapidly at high operating temperatures. Since the cladding is in excellent thermal contact with the coolant but is separated from the fuel by the low thermal conductivity gas gap, the cladding creep design criteria of Figure III-23 is shown to be much more dependent upon the outlet coolant temperature than on the reactor power level. Only at outlet coolant temperatures above approximately 1430°F will the cladding creep limit the designed reactor operation.

(5) Beryllium Swelling (and Core Vessel Creep)

Two additional interacting design criteria are the beryllium reflector swelling and the core vessel creep. The initial gap between the core vessel and the beryllium reflectors must be large enough to insure clearance between the surfaces at all times. If all the reflector control drums touched the vessel, control of the reactor would stop and the system would start to degrade as reactivity decreases due to hydrogen loss, fuel burnup, and fission product buildup. The gap cannot be excessively large due to the decreased worth of the reflector. However, the uncertainties in the creep rate of the core vessel material and the swelling of the beryllium reflector, due to neutron irradiation under SNAP 8 conditions, are quite large. Therefore, the core vessel is designed to have a 0.2% lifetime creep using a 1.4 factor of safety. This ensures that the core vessel will not strain rupture during core operation. The design allows up to 30-mil isotropic swelling of the beryllium during the system lifetime. The beryllium swelling is caused by fast neutron production of helium atoms and has been demonstrated to be strongly dependent upon operating temperatures. Therefore, the surfaces of the reflector have been coated to obtain a high emissivity to reduce the reflector temperatures as much as possible. However, if the core power, outlet coolant temperature, and/or surrounding environmental temperatures are excessively high, the reflector would require active cooling to maintain reflector swelling within design limits for the desired lifetime.

The beryllium swelling configuration for the design criteria is assumed to correspond to 4.5-in.-radius drums which have:

1) The beryllium swelling along the entire drum length corresponding to the point of maximum temperature and maximum integrated fast neutron flux during the reactor lifetime

2) The swelling occurring on the inside (near reactor) drum surface which causes the reflector to buckle, assuming that the ends of the reflector are unrestrained

3) The beryllium deflection limited to 30 mils of the 45 mils available.

Figure III-23 shows the beryllium swelling to be much more dependent upon the reactor power level than upon the outlet coolant temperature. This is due to the high heat generation rate within the reflector and due to the vacuum gap existing between the core vessel and the reflector. For outlet coolant temperatures less than approximately 1300°F, the design criteria is limited by beryllium swelling to a power level of slightly greater than 730 kwt.

b. Reactor Operation Beyond Design Criteria

The attainment of a design limitation would have no detrimental effect on the system performance. A better assessment of the reactor operating lifetime can be found by estimating the effects of operating the reactor beyond the design criteria until the operational limitation of reactivity due to excessive hydrogen loss degrades system performance.

(1) Excess Reactivity

Figure III-25 compares the conservative excess reactivity design criteria with the progressively less conservative probable operating limit and the upper limit of operation for the reactor performing for 10,000 hr. Figures III-26 and III-27 show the same excess reactivity restraint comparison for the reactor operating 20,000 hr and 30,000 hr respectively.

(a) Probable Operating Limitation

The probable excess reactivity operating limitation of the system uses the same assumptions as the design limitation except that the N_H is increased from 5.95 to 6.30 which corresponds to the lower specification limit of the maximum N_H.

(b) Upper Limit of Operation

The upper limit reactivity limitation utilizes the same assumptions as the probable operating limitation except using the maximum shim capability of the

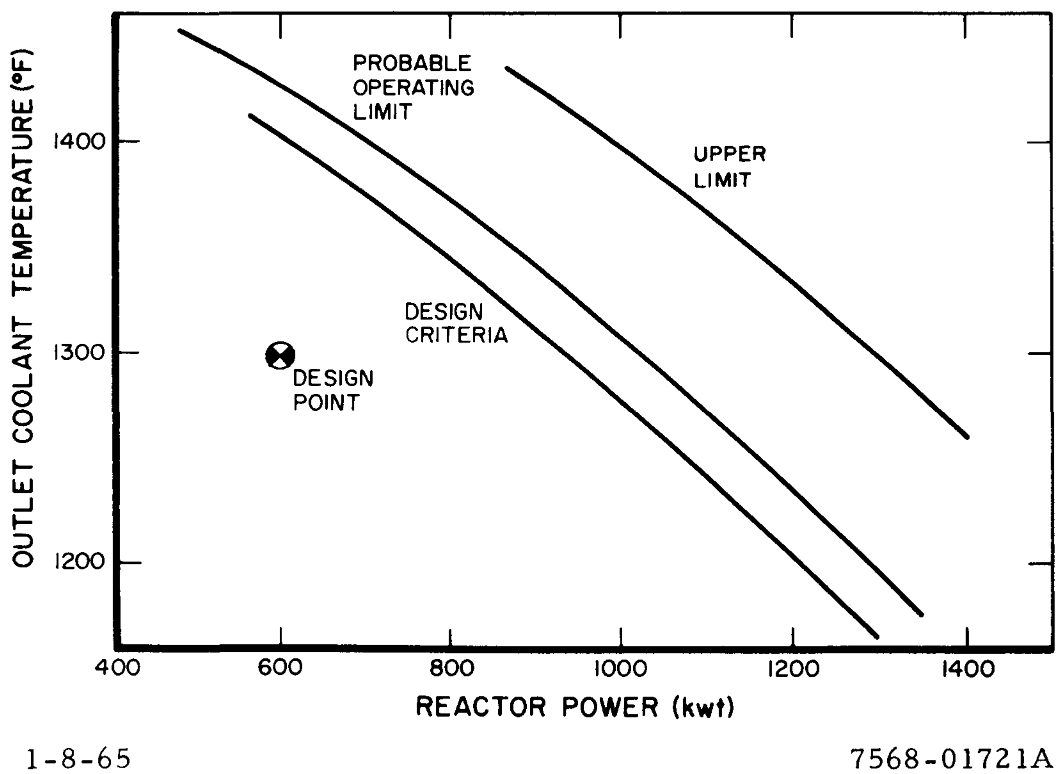

Figure III-25. Reference SNAP 8 – 10^4 hr
Reactivity Limitations

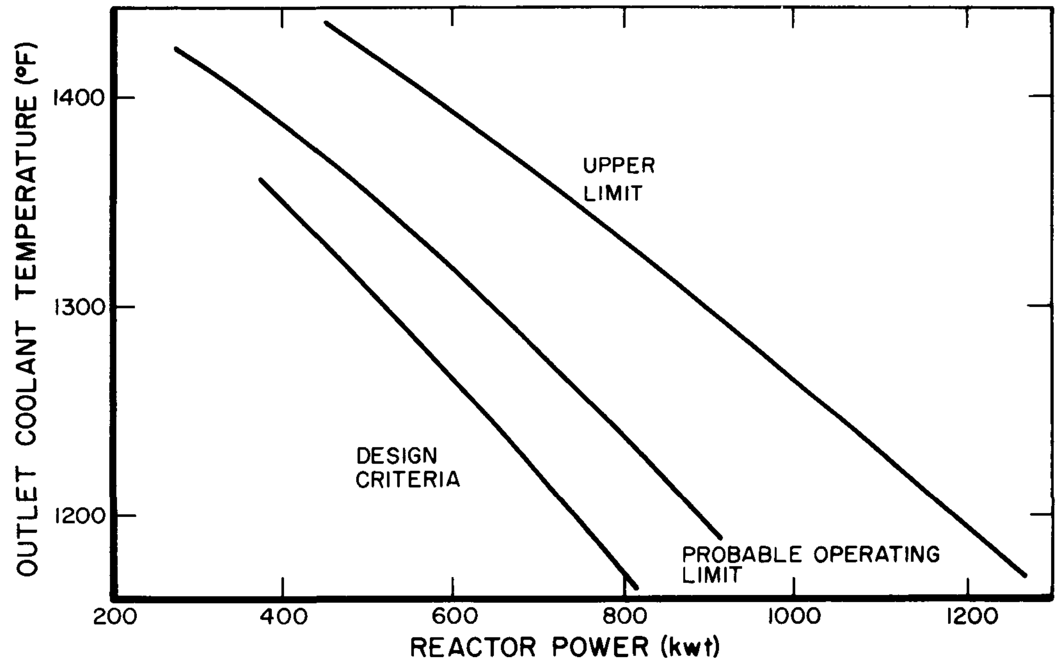

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Figure III-26. Reference SNAP 8 – 2 x 10^4 hr Reactivity Limitations

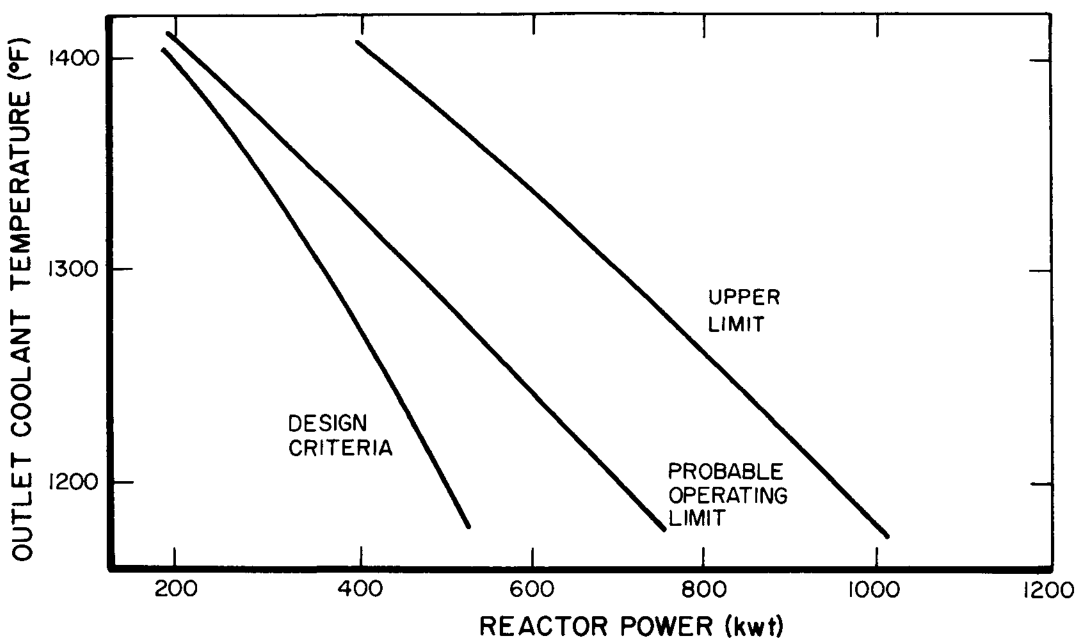

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Figure III-27. Reference SNAP 8 – 3 x 10^4 hr Reactivity Limitations beryllium drums. This restraint appears to be a truly upper limit operational capability since, in spite of the conservatism of the hot channel fuel operation, some of the reactivity would be needed to account for manufacturing tolerances, analytical uncertainties, etc.

(2) Fuel Swelling

It appears that the operational limitation of fuel swelling and cladding creep could both be extended until the cladding and ceramic have been strained 5 mils on the diameter. Although out-of-pile straining of the ceramic to 4 mils on the diameter has shown no detrimental effects, the conservative assumption is made that 4-mil in-pile straining degrades the ceramic so as to increase the hydrogen leakage by approximately an order of magnitude. This increased leakage is relatively low since the hydrogen would have to continue to permeate through the cladding. The reactor would still be able to function for well over 10^3 hr even with all fuel elements having their ceramic barrier degraded. However, a 5-mil diametral strain is approximately equal to the estimated 1% strain rupture limitation. The assumption that a 5-mil diametral straining of the cladding is a fuel swelling operational limitation is itself quite conservative since the analysis is based on the point of maximum core burnup operation in the hot channel condition.

Figure III-28 indicates the variation in the fuel swelling limitation between the conservative design criteria, the progressively less conservative modified design criteria, and the operational limitation for the reactor performing for 10,000 hr. Figures III-29 and III-30 show the same fuel swelling restraint comparison for the reactor operating 20,000 hr and 30,000 hr, respectively.

(a) Modified Design Criteria

This more reasonable swelling limitation makes the same assumptions as the design criteria except the fuel temperature is conservatively assumed to decrease due to fuel swelling decreasing the hydrogen gap. With a safety factor of 50 applied to the decrease in fuel thermal conductivity due to burnup, these conditions also result in a conservative fuel swelling limitation. Operation beyond this limit should be possible without any major decrease in the overall system performance capability.

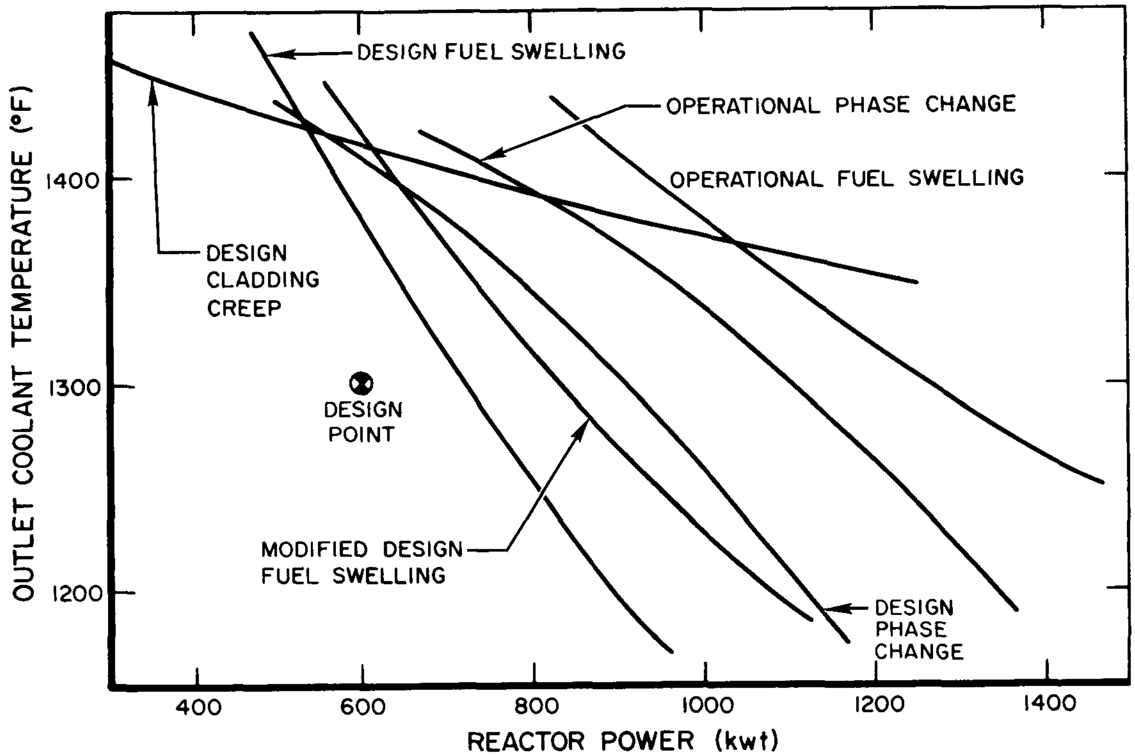

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Figure III-28. Reference SNAP 8 – 10^4 hr Fuel Element Limitations

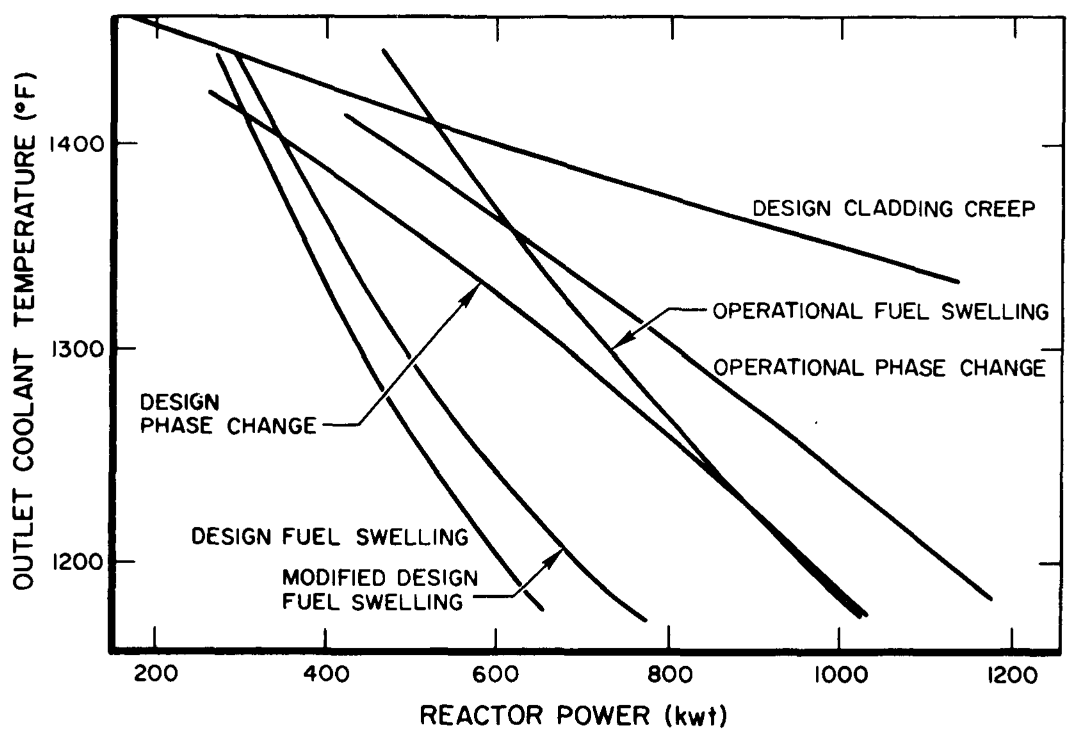

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Figure III-29. Reference SNAP 8 – 2 x 10^4 hr
Fuel Element Limitations

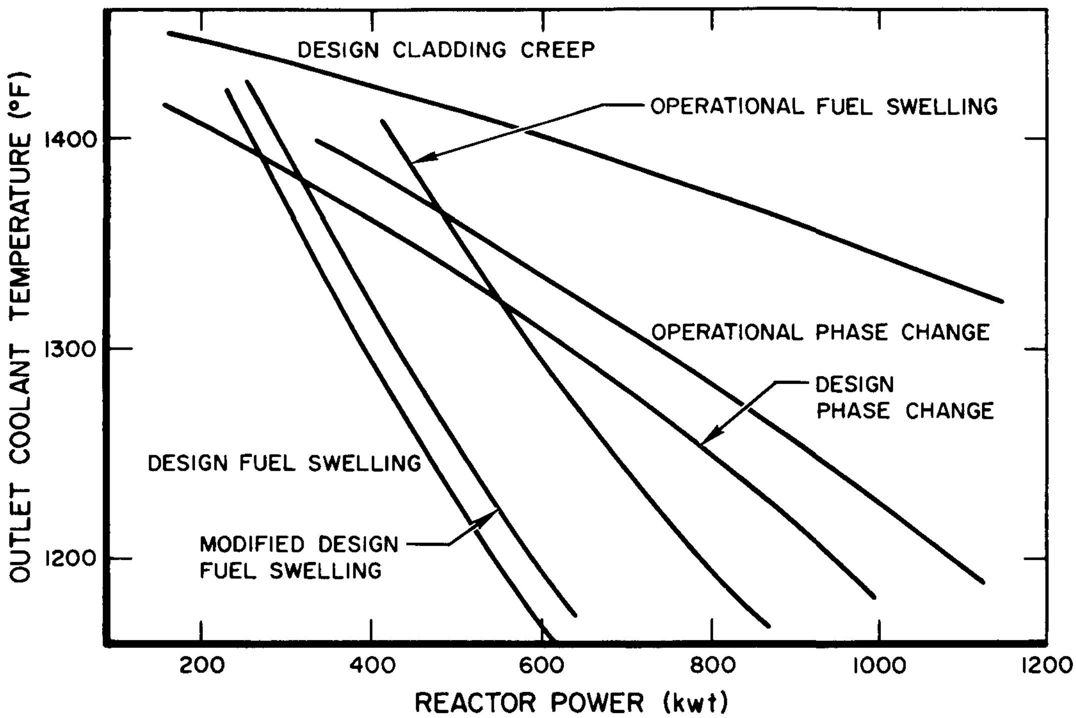

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Figure III-30. Reference SNAP 8 – 3 x 10^4 hr Fuel Element Limitations

(b) Operational Limitation

This appears to be a reasonably conservative, most probable fuel swelling limitation. Besides assuming that the fuel temperature would decrease due to fuel swelling, it is assumed the cladding can strain 5 mils in the diameter before the fuel swelling materially affects the reactor's excess reactivity. These conditions represent a conservative average operation of the nominal condition of all SNAP 8 fuel elements.

(3) Cladding Creep

Calculation of an operational limitation of the cladding creep was incorporated into the phase change limitation. By increasing the N_H to as high a value as possible, the reactivity of the system and the fuel phase change lifetime are both increased. The analysis maintained 0.2% cladding creep limitation across the 9-mil cladding wall thickness having a 1.4 factor of safety. The static NaK pressure was increased in order to counteract the higher hydrogen dissociation pressure caused by the increased N_H operating with the same hot channel fuel temperatures. The upper limit on the NaK static pressure was restricted to that value required to just strain the hottest location of the core vessel 0.2% during the reactor lifetime. The core vessel stress also had a 1.4 factor of safety. (The present core vessel thickness of 0.105 in. is required due to the launch shock and vibration specification values.) The N_H should not be increased above this corresponding upper limit NaK pressure because the reactivity loss of the beryllium reflectors, due to increasing the vessel thickness necessary to maintain the 0.2% creep limit, exceeds the reactivity gain due to the increased N_H. Cladding creep limitations for various reactor operating times are also shown in Figures III-28 through III-30.

(4) Fuel Phase Change

Figures III-28 through III-30 compare the different fuel phase change limitations of the conservative design criteria and the equally conservative operational limitation. The operational limitation utilizes all of the assumptions of the design criterion. The greater power-temperature capability is due to the longer time required for a higher initial N_H to diminish to the β-δ interface. The N_H value is the upper specification limitation (N_H = 6.35 + 0.05 = 6.4) which retains the cladding creep within the 0.2% limitation. The N_H which corresponds to the phase change design criterion is equal to 6.30, which is the lower limit fuel specification.

(5) Beryllium Swelling (and Core Vessel Creep)

It is assumed that the design criterion of the beryllium swelling and core vessel creep is the operational limitation. If the reflector and core vessel touch, the conservative assumption is made that all control drum movement stops and the system power starts to degrade. This assumption is conservative since there would still be a 15-mil gap remaining when the calculated limitation has been reached. The 15-mil tolerance is to take into account the uncertainty of irradiation flux on stainless creep properties and on beryllium metal swelling.

Figure III-31 illustrates the beryllium swelling limitations for different sized beryllium drums and for active cooling of the beryllium with the reactor operating for 10,000 hr. Figures III-32 and III-33 show the similar beryllium swelling limitations with the reactor operating for 20,000 hr and 30,000 hr, respectively.

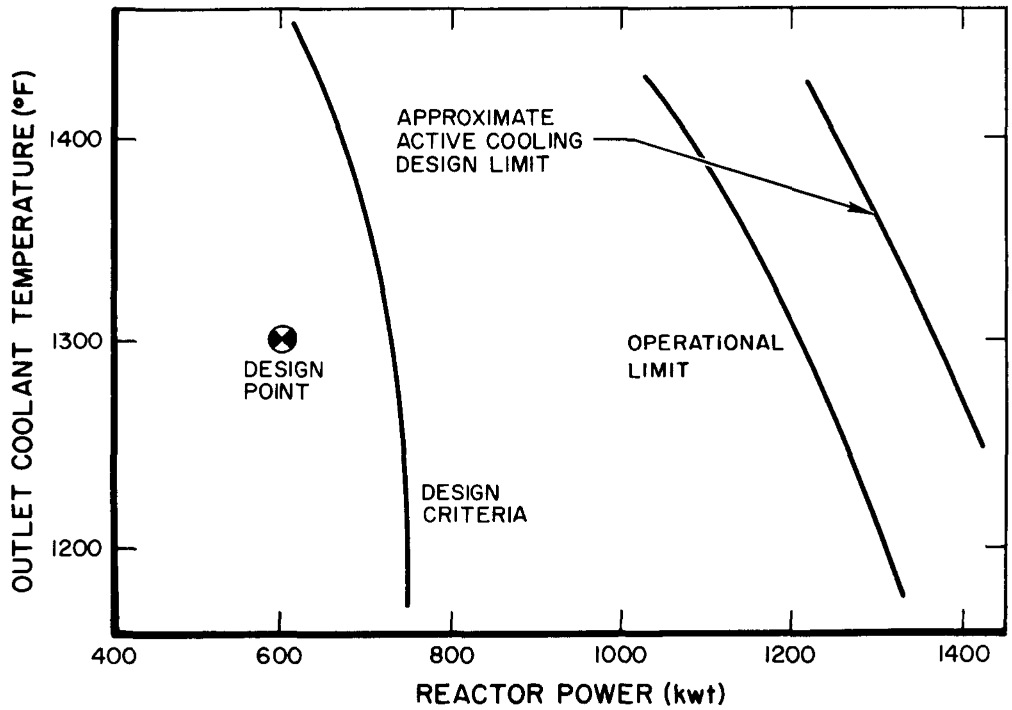

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Figure III-31. Reference SNAP 8 – 10^4 hr Beryllium Swelling Limitations

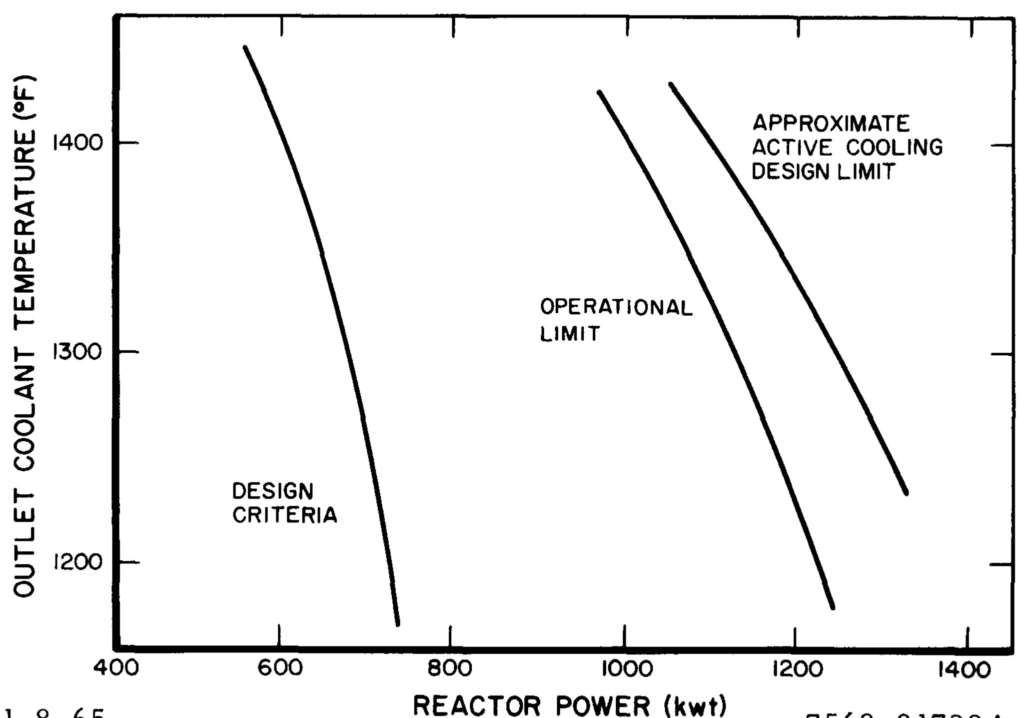

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Figure III-32. Reference SNAP 8 – 2 x 10^4 hr Beryllium Swelling Limitations

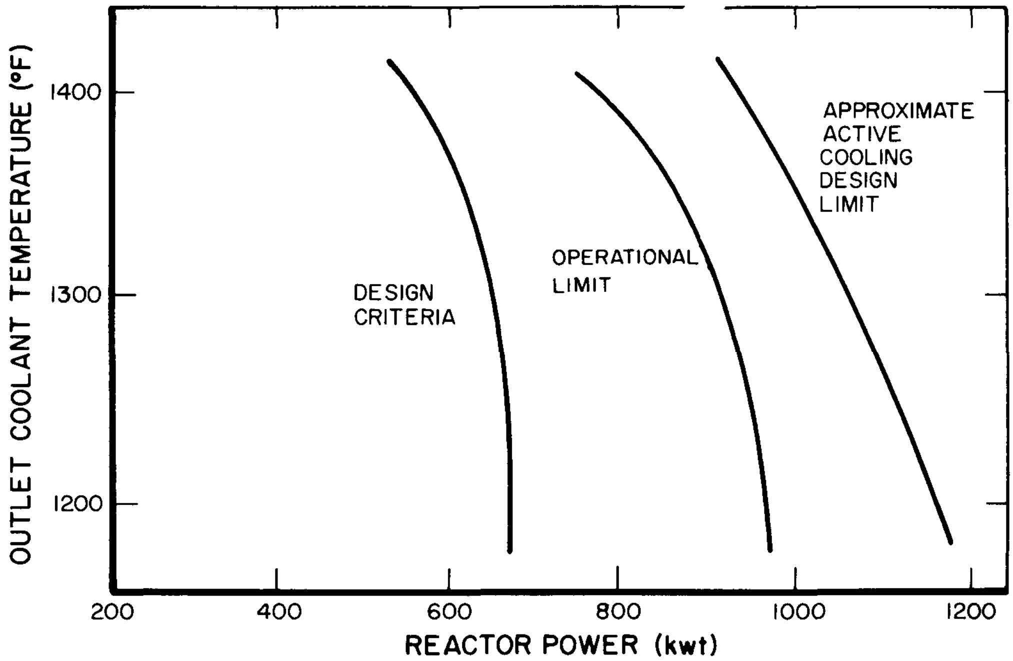

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Figure III-33. Reference SNAP 8 – 3 x 10^4 hr
Beryllium Swelling Limitations

The size of the SNAP 8 shield is determined by the total core and reflector envelope. Since the shield weight is a large fraction of the total system weight, reductions in the shielded envelope can result in significant weight savings for the system. A method of both reducing the shielded envelope and reducing the maximum beryllium temperature is to reduce the control drum radius. Reducing the control drum radius will decrease the individual drum worth, but both the present reference design as well as the small-drum designs would have more than adequate reactivity control. The operational limitation of the beryllium swelling incorporates the same assumptions as the design criteria except using a 3-in. drum radius.

As the beryllium reflector also acts as a structural member, its operating temperature should remain below approximately 1350°F. In addition, radiation damage to the beryllium is a function of operating temperature and total integrated flux. At the SNAP 8 fluxes, "breakaway" swelling is not expected to occur below approximately 1450°F. However, as the total integrated flux is increased, breakaway swelling will occur at lower temperatures. Because of these conditions it is important to maintain the beryllium reflector below approximately 1350°F.

In the reference design, the maximum operating temperature of the beryllium reflector is calculated to be slightly less than 1300°F when operating with a 150°F sink temperature, as will be the condition during ground testing or in space. Increasing the reactor power level, reactor outlet coolant temperature, or the sink temperature may result in an increase in beryllium temperature to the point where the 1350°F design limitation value is exceeded. Although exceeding this limit is not catastrophic, it is currently considered advisable to stay below it until the behavior of beryllium under SNAP 8 conditions of irradiation and temperature becomes better known.

As the expected operating temperature (~1300°F) is close to the design limitation (1350°F), a small study was conducted on means of reducing the beryllium temperature. This study considered active cooling of the reflector by NaK flowing through tubing embedded in the stationary beryllium and the changing of the control drum size from a 4-1/2-in. radius drum to a 3-in. radius drum. In addition, as 4-pi shielding was being considered for manned application of the Reference Design, the study also considered changes in the sink temperature and the use of small poison-backed control drums. The results of this study are shown in Figure III-34.

For the active cooling cases, no attempt was made to optimize either the inlet coolant temperature (assumed to be 800°F) or the flowrate (assumed to be 7100 lb/hr). Rather, the purpose of the study was to demonstrate the feasibility of using active cooling to allow operation at higher power and higher outlet coolant temperature. A conceptual layout was also made demonstrating the feasibility of the active cooling scheme.

However, adding an active reflector cooling scheme or a reactor cavity cooling scheme in the case of 4-pi shielding results in increased complexity. The high reflector temperature problem might better be solved by substitution of materials. Beryllium oxide could be used as a reflector with a very small loss in reactivity and a significant increase in temperature capability. The feasibility of using BeO has been demonstrated experimentally and analytically, but the details of the required design changes have not yet been delineated.

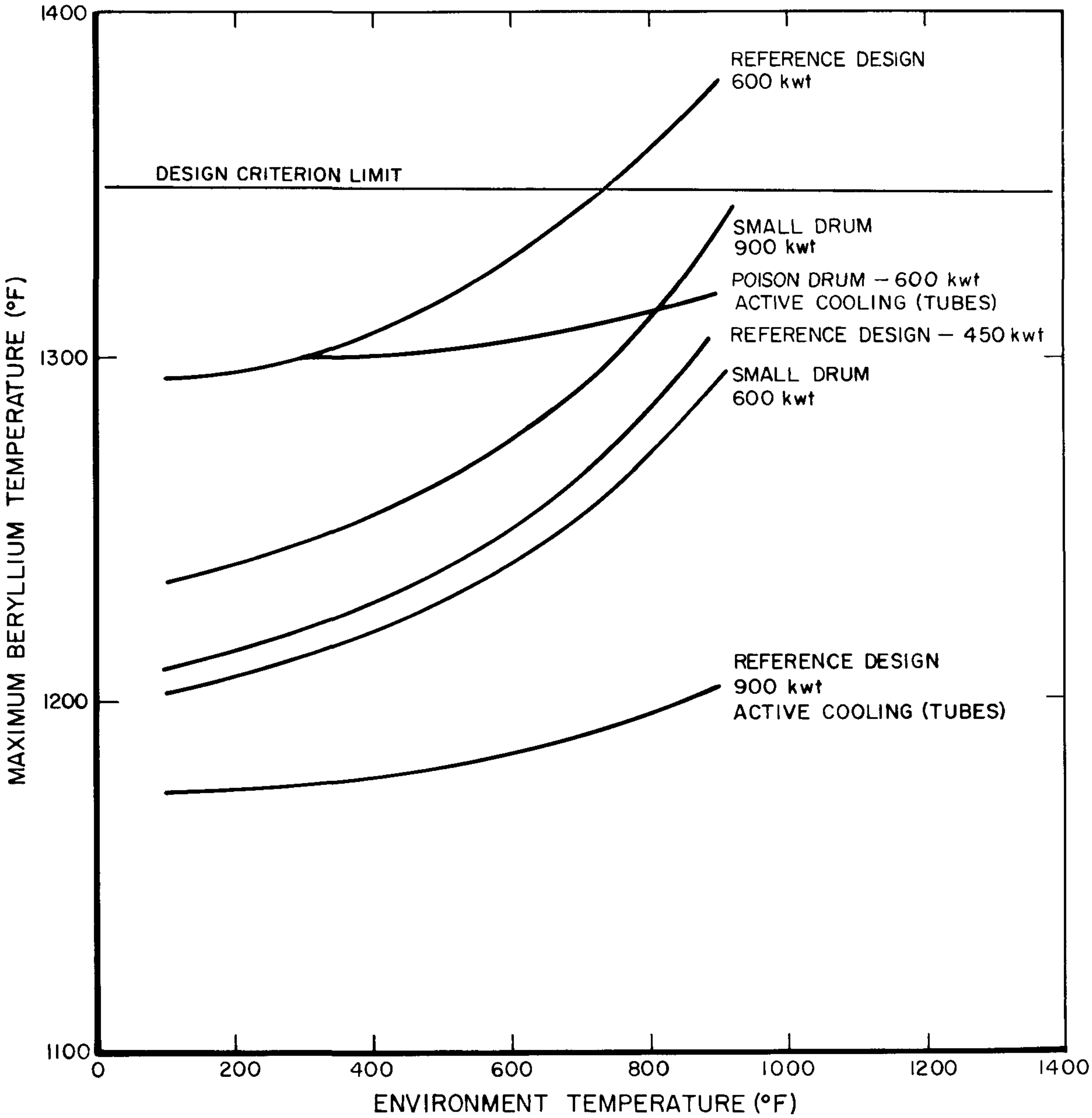

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Figure III-34. Maximum Beryllium Operating Temperatures

4. Reliability

a. Reliability Goals

One of the prime incentives for developing nuclear electric power for space applications is the long life potentially available from nuclear power plants. Attendant with this long life potential is a requirement for high reliability.

For unmanned applications, the power plant reliability should probably be at least as high as the payload reliability. The reliability goal of the reactor and its associated shield and controls would then be in the range of 0.95 to 0.98 reliability for startup in space followed by 10^4 hr of unattended operation. The reliability goal for the SNAP 8 nuclear system is $\sim$0.97.

For manned applications, the safety of the crew and great expense of each mission indicate higher reliability requirements. Current goa's for Manrated nuclear systems are in the range of 0.99 to 0.999 for 10^4 hr of operation.

b. General Approach Used to Achieve Reliability

In order to demonstrate these high reliability goals with a reasonable degree of confidence, a prohibitively expensive test program would be required. For example, Figure III-35 shows that to demonstrate with a 50% confidence level the SNAP 8 nuclear system reliability goal of 0.97, 22 reactors would have to be tested for 10,000 hr each. If one of these reactors should fail, an additional 33 reactors would have to be tested for 10,000 hr without failure. Thus, a straightforward demonstration of reactor reliability would be extremely expensive.

An alternate approach to achieve high reliability is therefore used. It employs the following plan:

1) A few developmental reactor tests are conducted culminating by one reactor qualification or demonstration test.

2) A larger number of developmental component tests are conducted followed by a limited number of component qualification tests.

3) A thorough component development program is conducted. The objective of this program is to develop and demonstrate reactor components which will operate under more severe environmental conditions than expected in actual use; for example, at higher temperatures.

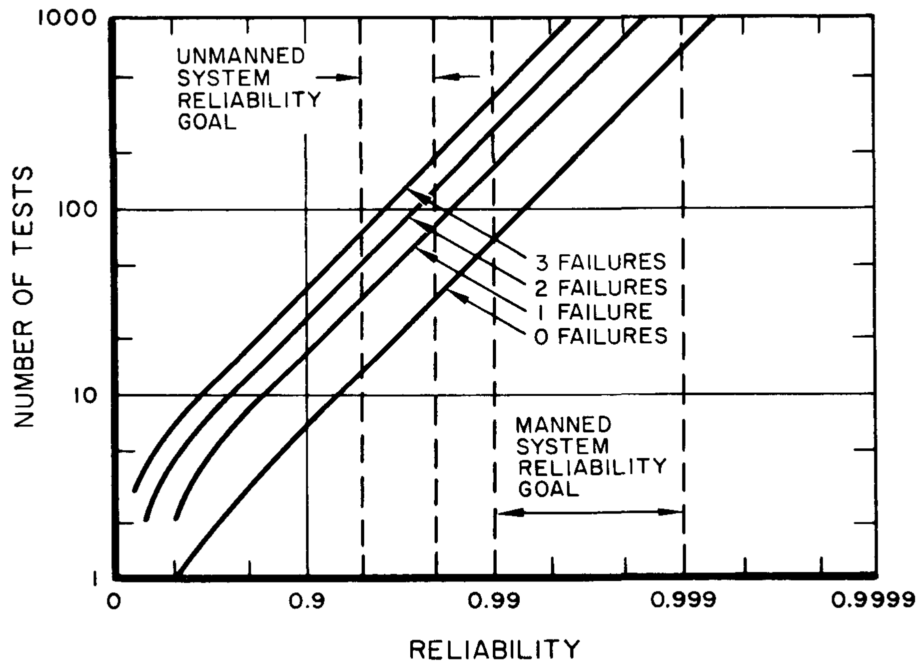

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Figure III-35. Reliability

4) The overall nuclear system is designed to achieve maximum reliability. With the low power reactors, which are inherently simple, the design stresses simplicity. For example, the SNAP 10A reactor utilizes a static, nonmoving control system. On the other hand, the design of high power reactors, which are of necessity more complex, stresses redundancy. For example, the SNAP 8 nuclear system employs active control but includes sufficient redundancy so that failure of an individual control drum drive and a limited number of fuel elements will not prevent the system from achieving rated power and life.

5) Logic circuit diagrams and reliability prediction models are formulated to assist in failure mode analyses.

6) Design reviews are conducted during the conceptual, preliminary, and final design stages to review and correct potentially weak areas in the system design.

7) High standards of fabrication quality control and inspection are established early in the program, and rigorously maintained.

c. SNAP 8 Approach to Reliability Through Redundancy

The SNAP 8 design approach to reliability is to make maximum use of redundancy. The use of redundant components allows a lower individual component reliability goal while still achieving a high system reliability goal. In the SNAP 8 nuclear system, the redundant component approach is largely through providing excess reactivity which allows failures and malfunctions of fuel elements and control drum drive assemblies without ensuing system failure.

(1) Available Excess Reactivity

Figure III-36 plots the excess reactivity needed against reactor lifetime for various discrete power levels for the SNAP 8 Reference Design nuclear system. Also shown on this figure is the nominal reactivity available, $11.00, and the reactivity available using maximum reflector shim capability, $15.00. The $4.00 difference between nominal and maximum reactivity is available to account for manufacturing tolerances, analytical inaccuracies, etc. and additional contingency.

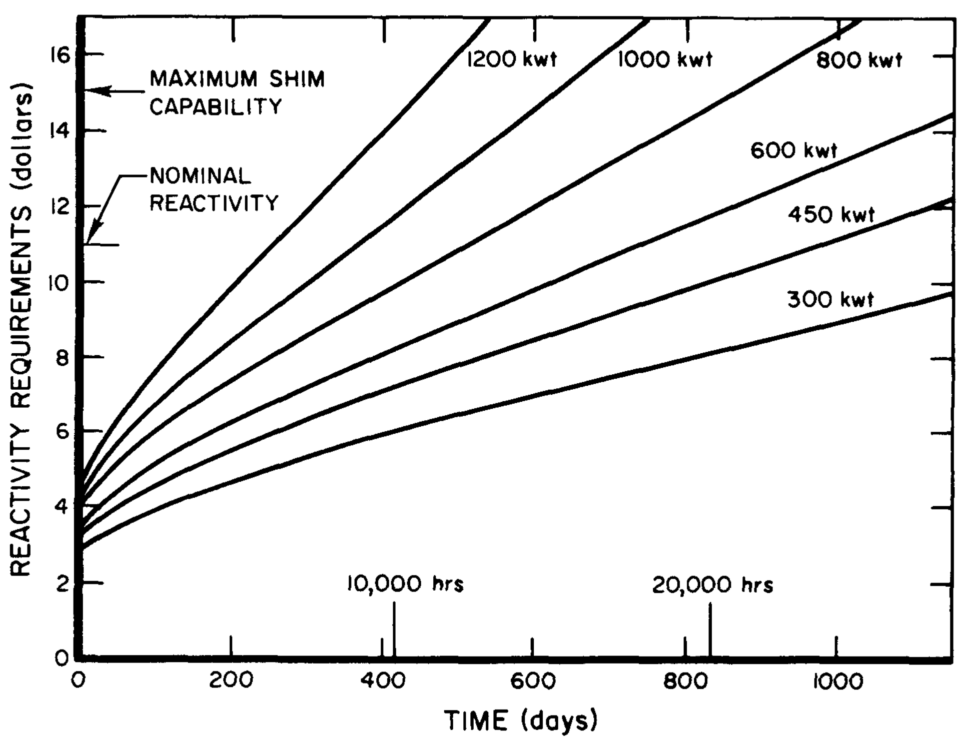

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Figure III-36. Reference SNAP 8 — Reactivity Requirements at 1300°F

Figure III-36 shows that the 600-kwt SNAP 8 Reference Design nuclear system has sufficient reactivity at the nominal value to run approximately 17,500 hr (740 days) and with maximum shim, can operate approximately 29,000 hr (1200 days).

Figure III-37 is a plot similar to Figure III-36 except the nuclear system outlet coolant temperature is reduced to 1200°F, and hence lifetime at a specific power level is increased, or the reliability at a specific power level and lifetime is increased. Figure III-38 is also a plot similar to Figure III-36 except for an outlet coolant temperature of 1400°F, and hence lifetime at a specific power level is decreased or reliability at a specific power level and lifetime is decreased.

These three figures show that the SNAP 8 nuclear system has sufficient excess reactivity to operate for 10,000 hr at various power levels from 600 to 1200 kwt at 1400 and 1200°F, respectively. However, as the nuclear system is operated at higher power and temperature, the amount of excess reactivity available for redundant components is decreased. Figure III-39 shows the nominal amount of excess reactivity available for redundancy (reliability) at the end of 10,000 hr of operation at various power levels and coolant outlet temperatures. Also shown for comparison is the maximum amount of excess reactivity available (full shim capability) at the reference outlet coolant temperature of 1300°F for various power levels. To a first order approximation, the relative reliability of operating the nuclear system at off-design conditions can be determined by the amount of excess reactivity available. For instance, the reliability of the system at 600 kwt and 1300°F outlet coolant temperature is approximately equal (same excess reactivity at the end of 10^4 hr) to the system operating at 375 kwt and 1400°F outlet coolant temperature.

(2) Available Redundancy

Figures III-40 through III-44 deal with nuclear system lifetime based upon available reactivity. The other limitations (discussed in Section III-F-3) are mainly design limitations while the reactivity limitation is an ultimate limitation. For example, when the available reactivity reaches zero, the nuclear system temperature begins to drop and the end of useful power operation occurs shortly thereafter. However, if the beryllium temperature rises from its design limit of 1350 to 1355°F or even to 1375°F, it is not likely that any immediate cessation

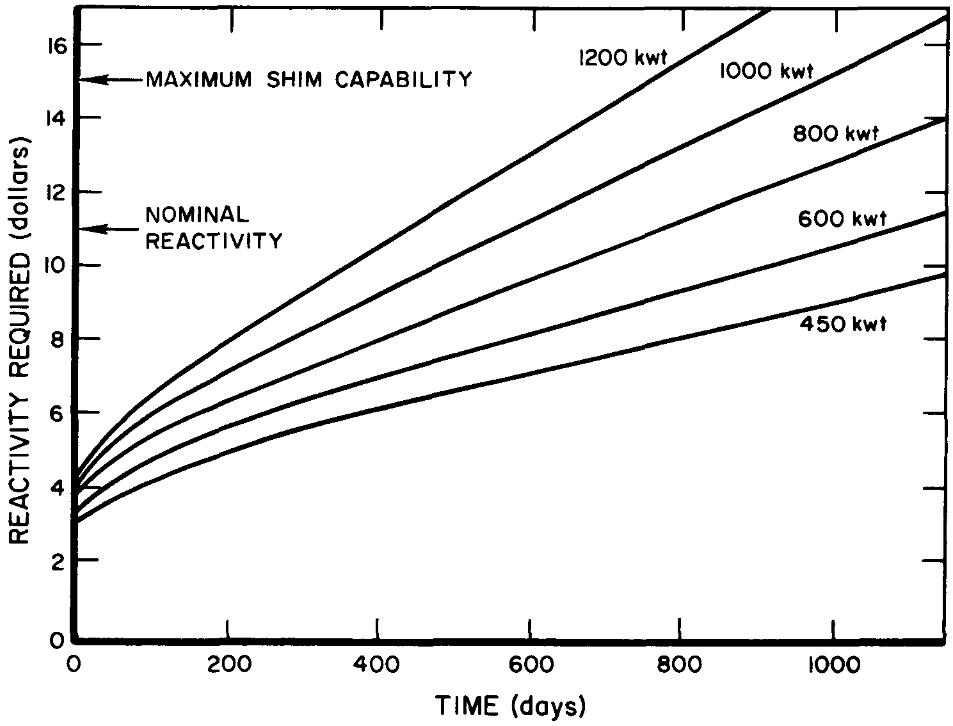

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Figure III-37. Reference SNAP 8 — Reactivity Requirements at 1200°F

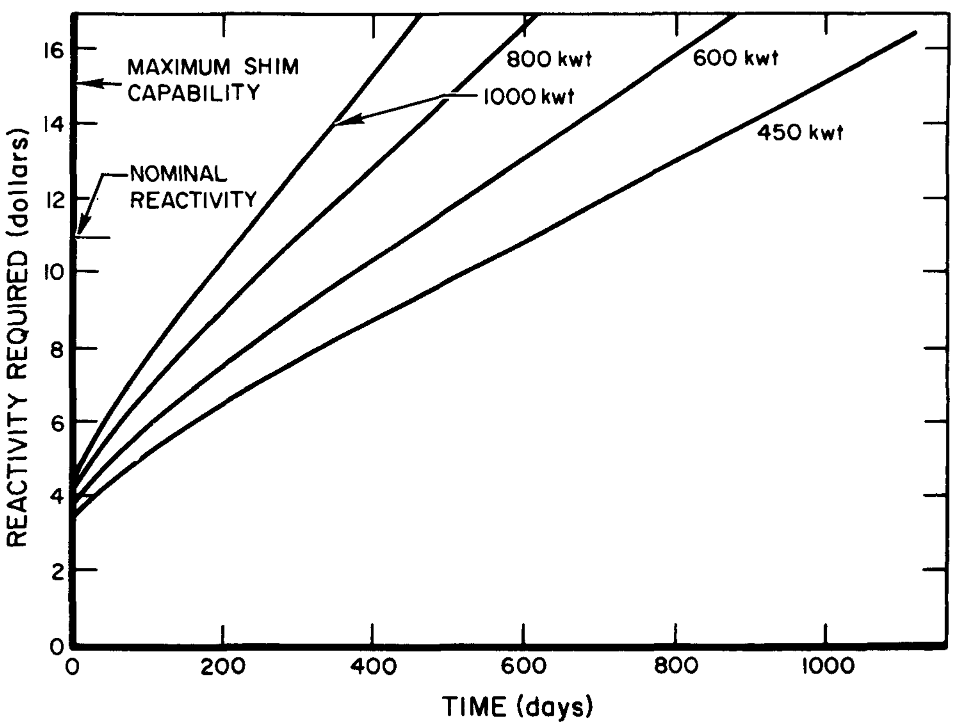

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Figure III-38. Reference SNAP 8 — Reactivity Requirements at 1400°F

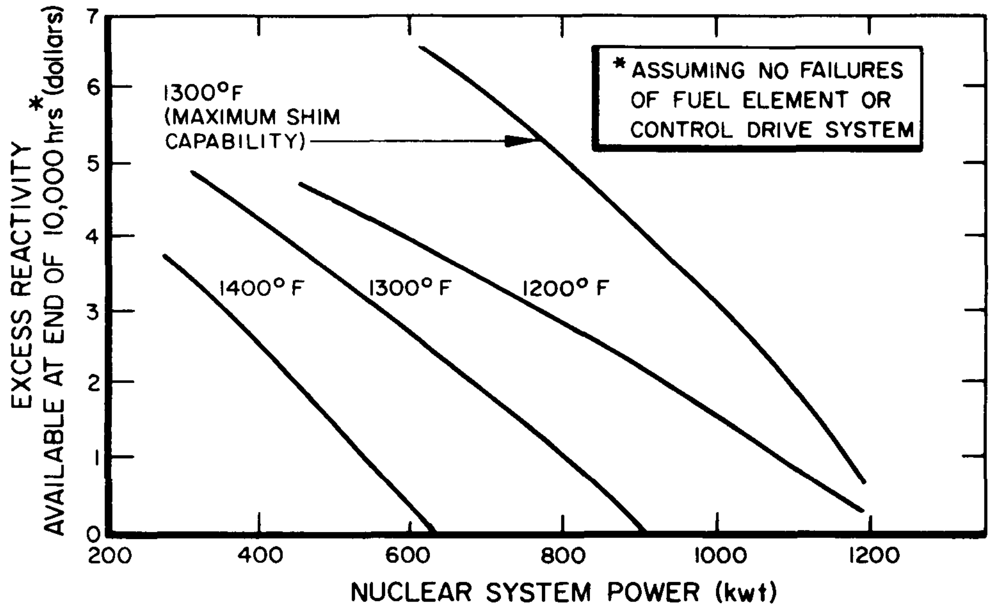

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Figure III-39. Excess Reactivity Available After 10^4 hr

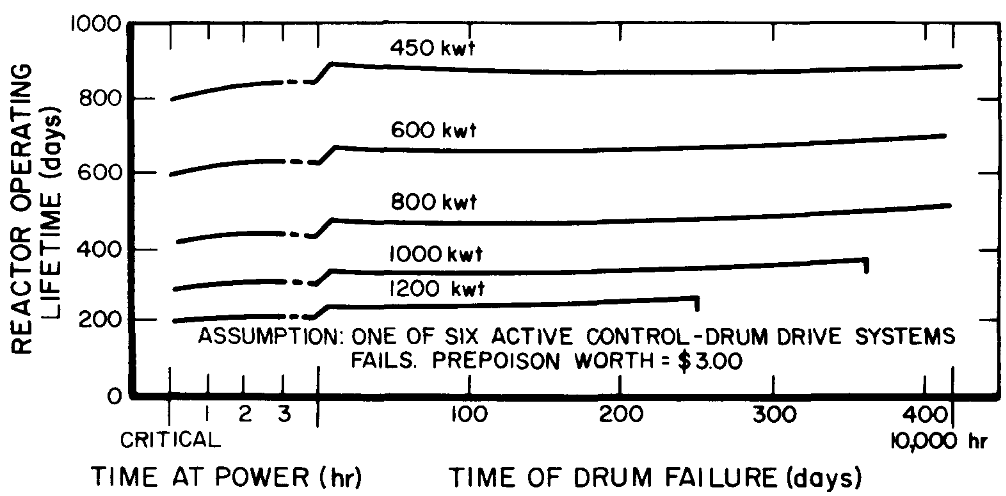

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Figure III-40. Reactivity Life vs Time of Single Drum Failure, 1300°F

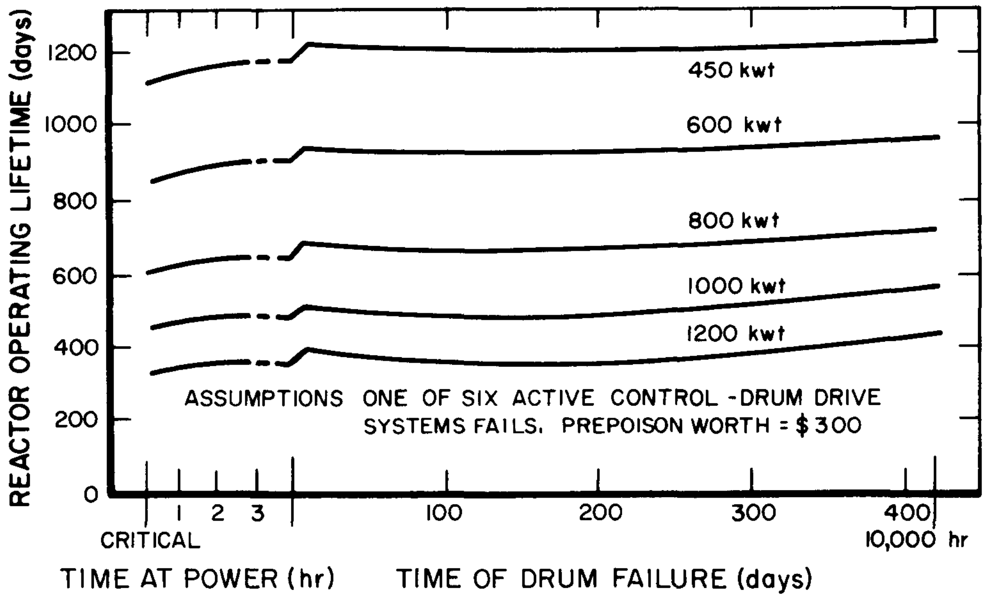

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Figure III-41. Reactivity Life vs Time of Single Drum Failure, 1200°F

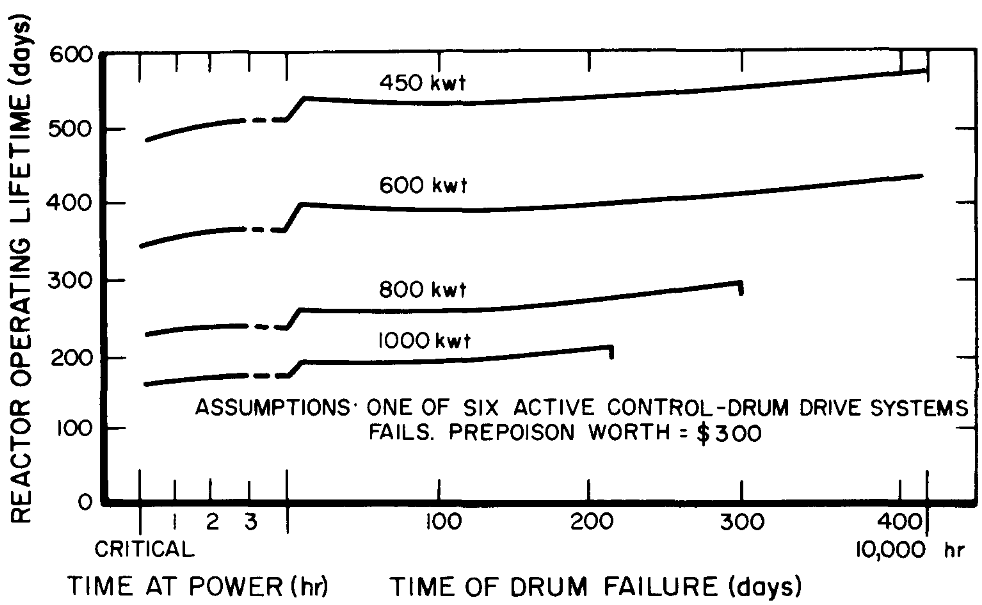

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Figure III-42. Reactivity Life vs Time of Single Drum Failure, 1400°F

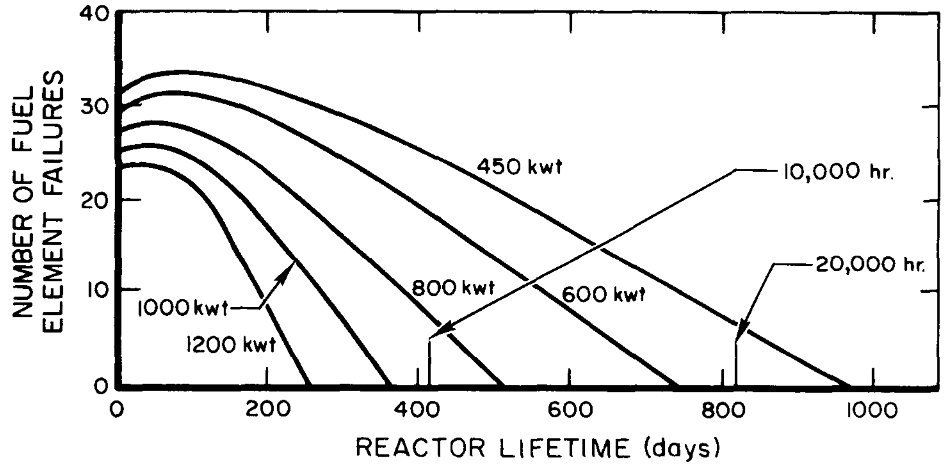

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Figure III-43. Reactivity Life vs Number of Fuel Element Failures

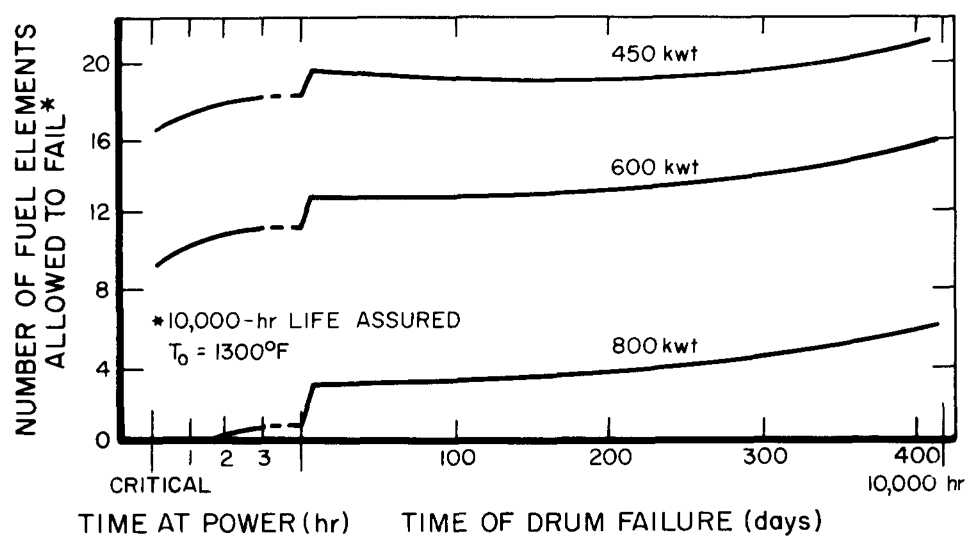

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Figure III-44. Fuel Element Redundancy With Drum Failure of operation will occur. The life of the system will be degraded but the amount of degradation or reliability decrease is difficult to predict.

Figure III-40 plots reactor lifetime as a function of time of failure of one control drum drive system for various power levels and an outlet coolant temperature of 1300°F. This figure conservatively assumes that only the nominal amount of reactivity (see Figure III-36) is available. This figure also assumes that all six control drums are used for control and that the control system philosophy used is "fail-as-is." The system is assumed to be prepoisoned ($3.00) to the Reference Design value.

Figure III-41 is similar to Figure III-40 except that the outlet coolant temperature has been decreased to 1200°F. The assumptions used to derive Figure III-40 also apply to Figure III-41. Figure III-42 is similar to Figure III-40 except that outlet coolant temperature has been increased by 100°F to 1400°F.

Figure III-43 shows the number of fuel elements that can "fail" at various power levels as a function of reactivity lifetime for a 1300°F outlet coolant temperature. "Failure" is defined as:

1) The hydrogen density in a fuel element drops instantaneously from its initial value of about 6.0×10^{22} H atoms/cc of fuel ($N_H = 6.0$) to 3.0×10^{22} atoms/cc of fuel. (This assumption is extremely conservative. Results of a recent investigation based upon 2.0 cm^2 of bare cladding surface area show that it takes 4000 hr to reach the β-δ phase boundary, $N_H \sim 5.0$. Once this phase boundary is reached, the hydrogen loss rate decreases by an order of magnitude.)

2) The reactivity worth of the lost hydrogen is equivalent to the peak value (-15¢) for any fuel element in the core.

3) All fuel element failures occur prior to the time designated reactor lifetime in Figure III-42. (If one or more of the total number of fuel element failures does not occur before this time, the operating life is actually extended beyond the time shown.)

4) There are no control drum drive system failures.

Figure III-43 shows that operating at 600 kwt, 18 fuel elements may "fail" and the reactor still has sufficient reactivity to achieve 10,000 hr.

Figure III-44 is derived from Figures III-40 and III-43 and shows the number of fuel elements that may "fail" as a function of time of control drum failure and still achieve 10,000 hr of operation. From Figure III-44 it can be seen that 9 fuel elements and 1 control drum drive system may fail and still the reactor has sufficient reactivity to operate 10,000 hr at 600 kwt with an outlet coolant temperature of 1300°F.

Figure III-45 shows the available redundancy (reliability) of the SNAP 8 nuclear system operating at off-design conditions of power and temperature for 20,000 hr or twice the design life. These curves show that the SNAP 8 nuclear system can operate for 20,000 hr at 600 kwt and 1200°F with only a small decrease in reliability over the design conditions. Operation for 20,000 hr at 300 kwt and 1300°F is possible with approximately the same reliability, as 2 control drum drive systems can malfunction. An improvement in reliability is attained when operating at 300 kwt and 1200°F outlet coolant temperature.

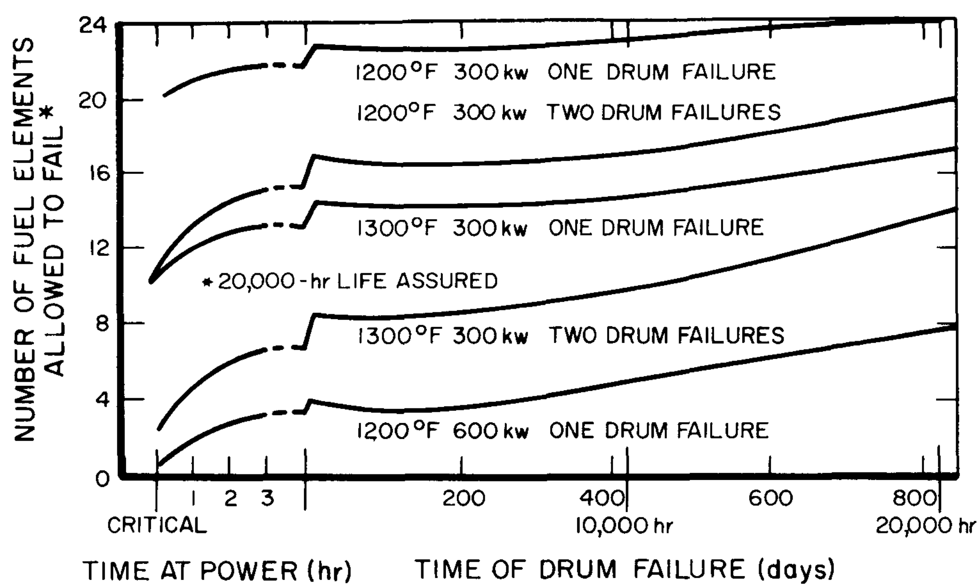

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Figure III-45. Fuel Element Redundancy With Drum Failures at Extended Life

It is possible to compensate for increases in design lifetime by reductions in the operating temperature and/or power level, since wearout of mechanical components is not a problem in the nuclear system. Degradation due to neutron flux, temperature, etc., are the most probable modes of failure. Operation at a reduced power level of 300 kwt for 20,000 hr gives the same integrated flux and hence should not significantly decrease the system reliability. Operation at the reduced temperature, 1200°F, for 20,000 hr should significantly improve the component reliability over the 10,000-hr, 1300°F operation.

(3) Passive Operation

The preceding section has discussed the effect of control drum drive failures upon the ability of the system to achieve the desired lifetime, the assumption being that loss of active control of the SNAP 8 nuclear system will result in an immediate loss of the electrical generating capability. This assumption is not strictly valid.

Since the SNAP 8 reactor has a negative temperature coefficient of reactivity, the time required for the system to degrade to the point where electrical generation ceases is directly related to the rate of change of the system reactivity. Figure III-46 shows the excess reactivity of the SNAP 8 reactor as a function of time for various power levels at a 1300°F outlet temperature. The temperature coefficient for SNAP 8 is about -0.2¢/°F so that an uncontrolled loss of 10¢ in reactivity will result in a 50°F drop in temperature. A 50°F drop in temperature would result in failure of the AGC reference design power conversion system and would occur in about 15 days if the reactor has operated more than about 200 days at 450 kwt.

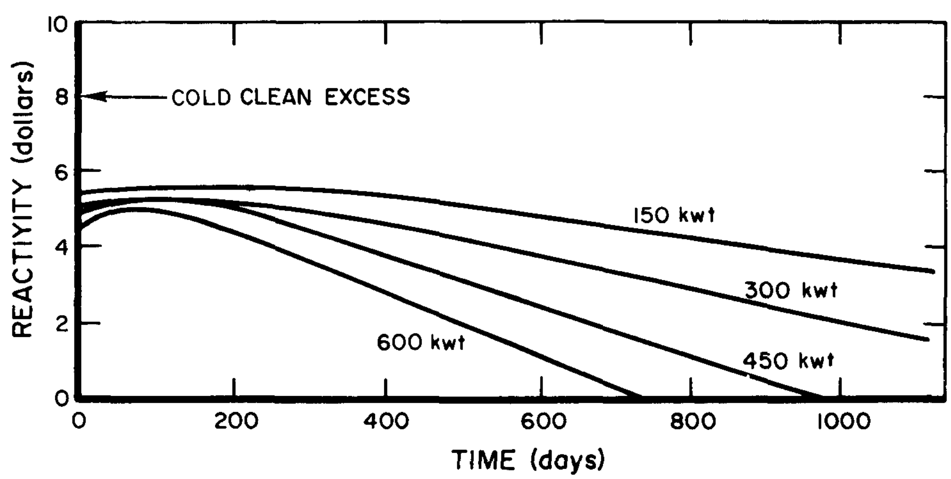

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Figure III-46. Reactivity vs Time, Frozen Control Drums

If the SNAP 8 reactor were coupled to a power conversion system which required significantly less power than the present reference value of 450 to 500 kwt, the time for the reactor temperature to degrade is considerably increased. If, for example, the reactor were operated at 150 kwt, the control system could fail after only 25 days of operation and the system would still achieve over 1 yr of operation. The reactor temperature would gradually increase about 50°F as the burnable poison is depleted. The temperature would start to decrease after about 250 days and would show a net loss of 50°F in about 400 days.

At low power levels, the use of a burnable poison in the core, in effect, provides a redundant backup to the control system.

(4) Component Reliability Goals

The excess reactivity available for redundant components and subsystems means that more reasonable reliability goals can be established for individual components. For instance, the fuel element core cluster has a reliability goal of 99%. Figure III-47 shows the single fuel element reliability necessary for a

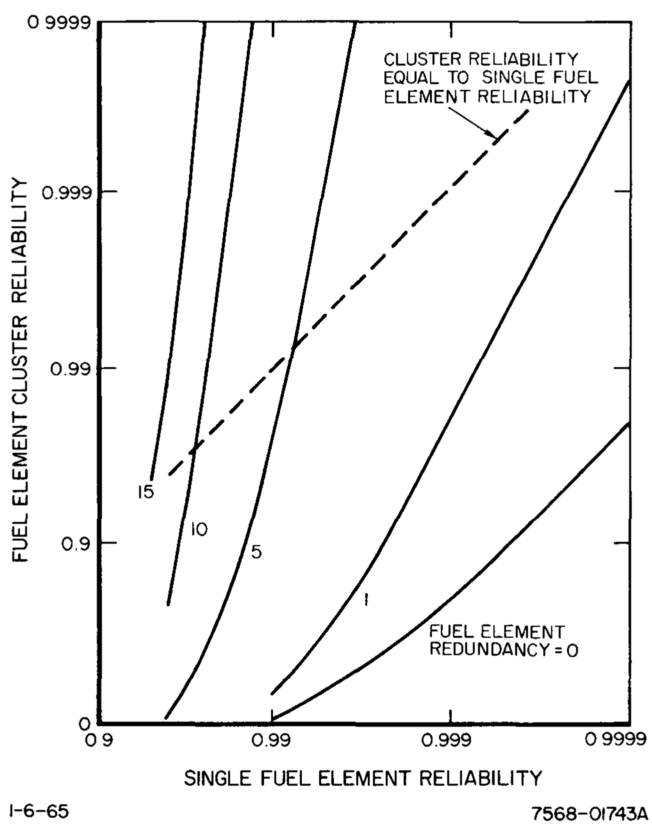

Figure III-47.
Reliability of a Cluster of 211 Fuel Elements

211-fuel element cluster reliability for various redundant fuel elements. If no redundancy were available, each fuel element would require a reliability goal in excess of 0.9999 to achieve a cluster reliability goal of 0.99; but with only 5 redundant elements, this goal is reduced to 0.99 and with 10 redundant elements the goal is reduced further to only 0.96.

The same reduction in reliability goals is also true for the control drum drive system. The design goal for the 6-drum drive system is about 0.992. Figure III-48 shows that without a redundant control drum, the individual drive systems would require a reliability of about 0.999 while with 1 redundant drum system, the individual reliability goal is reduced to about 0.975.

As previously discussed, lowering the reactor power level can also increase the system reliability, or for the same system reliability, decrease the component reliability necessary. Figure III-45 shows that 2 drum drive systems can be allowed to fail and still attain 10^4 hr of operation at 300 kwt. Referring to Figure III-48, maintaining the same 6-drum drive system reliability goal of 0.992, an individual drive system reliability goal is decreased from 0.975 goal to a goal of only 0.92. Conversely, with the same individual drive system reliability of 0.975, the overall 6-drum drive system reliability is increased from 0.992 to 0.9997 when two individual drive systems are allowed to malfunction.

In addition, a conversative philosophy is employed in the structural design of the SNAP 8 nuclear system. This conservative design results in a high system reliability. In general, design limits are set well below the point at which damage might be expected to occur and safety factors are applied to the expected loads. As an example, the design allowable creep of the core vessel is 0.2%. The core vessel would not interfere with the reflectors, however, until the core vessel creep exceeded 0.9%. In addition, a safety factor of 1.4 is used to compute the design load on the vessel. Using the maximum expected range of material strength and operating loads, the design reliability of this vessel has been computed to be in excess of 0.9999. Other components in the system are designed using the same philosophy and similar safety factors.

The SNAP 8 Reference Design nuclear system component reliability apportionment goals are shown in Table III-18. A summary of the major subsystem reliability apportionment goals are shown in Table III-19.

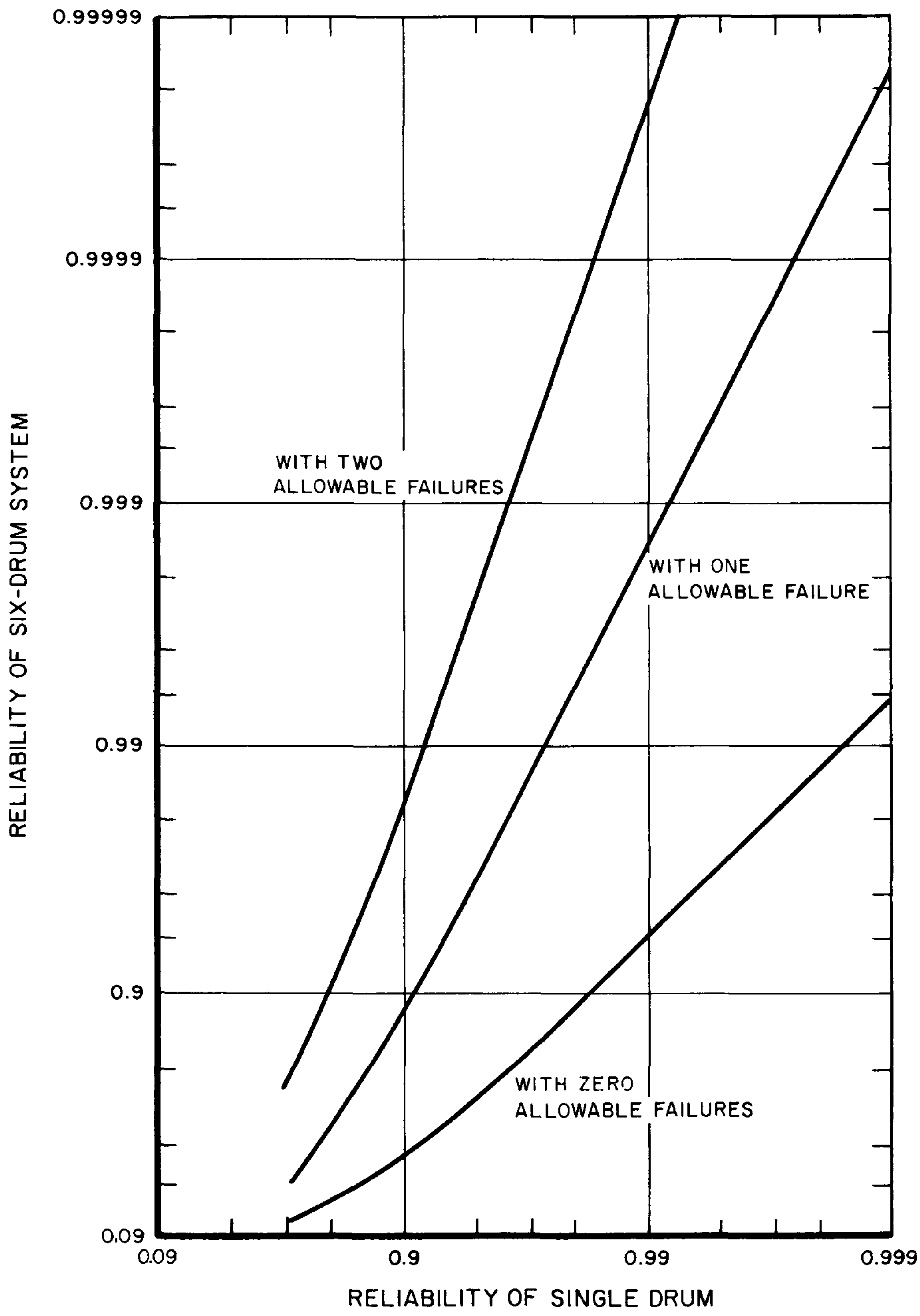

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Figure III-48. Reliability of 6-Drum System

TABLE III-18
COMPONENT RELIABILITY APPORTIONMENT GOALS

Reactor Assembly	Failure Rate		Life	Total	Reliability Goal
	Launch	Startup			
Core vessel	0.000009	0.000021	0.000075	0 000105	0.99990
NaK pipe	0.000021	0.000039	0.000050	0 000110	0.99989
Coolant — flow baffle plate	0.000006	0.000012	0 000012	0.000030	0 99997
Inlet plenum grid plate	0.000009	0.000009	0.000009	0.000027	0 999973
Outlet plenum grid plate	0 000009	0.000009	0.000009	0.000027	0.999973
Internal reflectors	0 000012	0.000030	0 000030	0.000072	0.999928
Fuel elements	0 000750	0.001500	0.007750	0.010000	0 99000
Total Subsystem	0.000816	0 001620	0 007935	0.010371	0 9896

Reactor Control and Reflector Assembly	Launch	Failure Rate		Total	Reliability Goal
		Startup	Life		
Startup control programmer	0.000135	0.001350	-	0.001485	0.9985
Position sensors-limit switches	0.000081	0.000810	-	0.000891	0.9991
Startup squibs and release actuators	0.000060	0.000300	-	0.000360	0.99964
Reflector startup drums (3)	0.000030	0.000150	-	0.000180	0.99982
Startup drum actuators	0.000108	0.000150	0.000324	0.000582	0.99942
Long term controller	0.000135	0.000100	0.002160	0.002395	0.99761
Temperature sensor switches and amplifier	0.000108	0.001080	0.000540	0.001728	0.99827
Diagnostic instrumentation	0.000060	0.000060	0.000145	0.000265	0.99974
Reflector control drums	0.000042	0.001410	0.002310	0.003762	0.99623
Reflector-control drum spacers and structure	0.000012	0.000060	0.000120	0.000192	0.99982
Control drum actuators (bidirectional)	0.000135	0.000675	0.001215	0.002025	0.99798
Electrical harness and connectors	0.000141	0.000525	0.000765	0.001431	0.99857
Total Subsystem	0.001047	0.006670	0.007579	0.01530	0.9847

Shield Assembly	Launch	Failure Rates		End-of-Life Shutdown	Reentry Shutdown	Total	Reliability Goal
		Startup	Life				
Gamma shield	0.000003	0. 000006	0.000006	-	-	0.000015	0.999985
Neutron shield	0.000003	0.000006	0.000024	-	-	0.000033	0.999967
Neutron shield vessel	0.000054	0.000078	0.000120	-	-	0.000252	0.999748
Total Subsystem	0.000060	0 000090	0.000150	-	-	0 000300	0 9997
Safety Assembly							
Scram Springs	0 000048	0.000024	0.000072	0.000120	0.000120	0.000384	0 99962
Safety system structure	0 000012	0.000006	0.000018	0.000030	0 000030	0.000096	0.999904
Mechanical End-of-life Shutdown	0.000120	0 000180	0.000150	0 000810	-	0.001260	0 99874
Electrical End-of-life Shutdown	0.000243	0 000060	0.000300	0.002100	-	0.002703	0.99736
Reentry band	0.000090	0.000060	0.000120	-	0.000600	0.000870	0.99913
Launch destruct inact	0.000081	-	-	-	-	0 000081	0.99992
Launch destruct jettison	0.000081	0.000081	-	-	-	0.000162	0.99984
Total Subsystem	0 000675	0.000411	0.000660	0.003060	0.000750	0.005556	0.9944

TABLE III-19

SNAP 8 NUCLEAR SYSTEM RELIABILITY APPORTIONMENT

Major Subsystems	Launch	Startup	10,000 hr	End-of-Life Shutdown	Reentry Shutdown	Total
Reactor Assembly	0.99912	0.99840	0.99206	-	-	0.9896
Reactor Control and Reflector Assembly	0.99895	0.99343	0.99233	-	-	0.985
Shield Assembly	0.99994	0.99991	0.99985	-	-	0.9997
Safety Assembly	0.99933	0.99959	0.99934	0.99694	0.99925	0.9944
Phase total	0.9974	0.9913	0.9836	0.99694	0.99925	
Cumulative probability of success (reliability)	0.9974	0.9887	0.9723	0.9692	0.9685	0.9685

In addition to the application of redundancy through excess reactivity, redundancy has been provided in other potential problem areas. Figure III-49 is a reliability logic diagram which has been developed to assess areas where reliability may be improved either through redundancy or increased safety factors.

As can be seen from Figure III-49, essentially all of the series elements in the logic diagram have either complete redundancy or partial redundancy. With the assistance of this type of logical approach to the problem of high reliability, the reliability of the overall nuclear system is continuously being upgraded through use of more redundancy and/or higher design margins and safety factors.

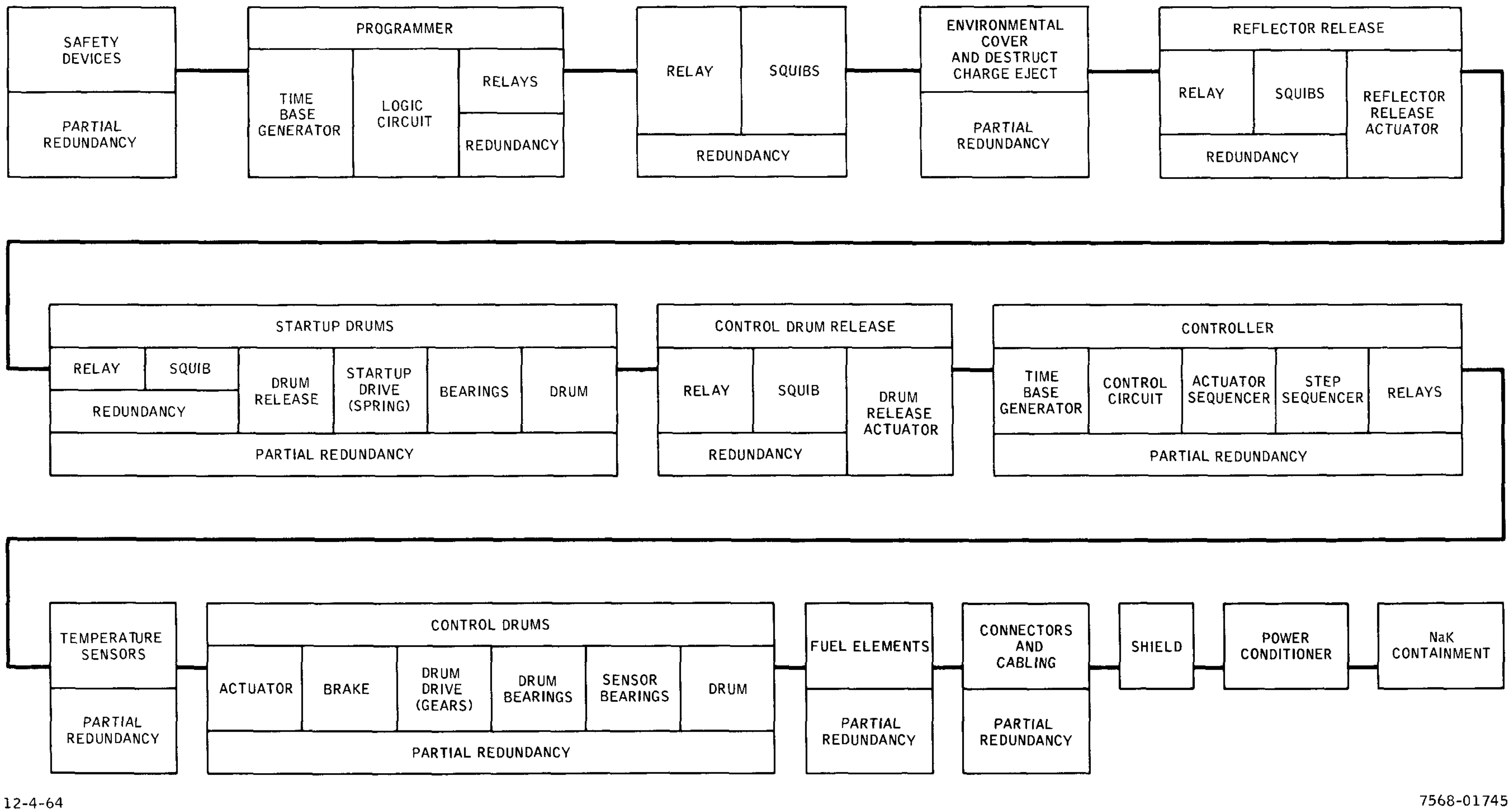

Figure III-49. Reliability Logic Diagram

G. MANRATED SNAP 8

1. Introduction

Manrating a system means taking into consideration technical factors involved with the added presence of man, his possible participation in the operation of the system, and the reliability required of the system as a consequence of the dependence of human life.

Manrating a SNAP nuclear system means optimizing the nuclear system to provide a reliable, minimum weight system requiring minimum attention by on-board personnel. The specific objectives of a SNAP manrating program are to:

1) Increase nuclear system reliability
2) Provide maintainability where possible
3) Reduce the reactor shielded envelope
4) Optimize shield weight
5) Provide shutdown and restart capability

The technical approach to obtaining the specific objectives listed above follows.

a. Reliability

The reliability goal for a manrated nuclear system should lie between 0.99 and 0.999 in order to achieve a sufficiently high overall mission reliability. To demonstrate a 0.99 reliability at the 50% confidence level would require approximately 70 system tests assuming no failures, or about 180 system tests assuming only one failure. A straightforward demonstration of nuclear system reliability would be extremely expensive. Hence an alternate approach to achieve high reliability is used. The essential features of such a program were outlined in Section III-F-4-b. Additionally, component development tests will be conducted to failure, to help uncover design weaknesses, and to experimentally determine design margins. Where possible, maintainability is also provided to improve reliability. The latter is discussed separately in the next section.

Successful completion of such a program will provide a nuclear system with a high predicted reliability and the highest demonstrated reliability consistent with a reasonable funding level.

b. Maintainability

An important task in manrating a nuclear system is to provide maintainability where practicable. However, maintenance should be kept at a minimum by providing maximum reliability for unattended, unmaintained operation. Maintenance of the electronic controls which will be located in the manned environment is, of course, highly practicable.

Although it appears impractical at first glance to perform maintenance on the reactor assembly (core and reflector assemblies) or the shield assembly, further investigation is warranted. For instance, a mechanical control drum override system might be incorporated to free a "sticky" control drum after which it could conceivably operate properly. Although this is not maintenance in the sense of replacing or repairing components, it is maintenance in the sense that further operation of the control drum may be possible as a result of this action. This type of maintenance must be investigated in order to assist in maximizing reliability. Also required is an analysis of what other types of man-intervention in the automatic plant operation might prove desirable.

c. Reactor Envelope

It is important to minimize the reactor envelope to minimize shield weight, as a significant fraction of the weight of any manrated nuclear system is due to the shield. (For example, a 10^6 reduction in gamma rays is required for manned system, 10^{-3} r/hr vs unmanned system, 10^3 r/hr.) In order to reduce the shielded diameter, design studies of alternate reflector control assemblies must be performed. These studies should consider such items as smaller diameter control drums, tapering the control drum to follow closely the angle of a typical shadow shield cone angle, routing of NaK piping and cable harness to minimize the scattering envelope caused by these appendages, etc.

d. Shield Weight

As previously noted, a significant fraction of the weight of any manrated nuclear system is due to the shield weight. The shield weight is minimized by (1) reducing the reactor assembly envelope, (2) optimizing the shield design, and (3) optimizing the separation distance between the reactor and the payload dose plane. The first item was previously discussed.

Optimization of shield design must include proper choice of shielding materials. For example, should Li^6H, Li^7H, or natural LiH be used as neutron shielding material and should depleted uranium, tungsten, lead, or some other material be used for gamma shielding? Another phase of shield optimization includes the question of how to effectively use the shielding material. For instance, should the gamma shield be uniformly dispersed throughout the neutron shield material, or should it be separated as a slab preceding the neutron shield, or split so that some precedes and some follows the neutron shield material?

Shield optimization studies also must consider how best to support the reactor assembly. For instance, should the reactor assembly leads be carried through the shield vessel wall or should they be carried by an internal structure? Another phase of the study should consider how best to route the NaK piping and cable harness. For example, should one inlet and one outlet pipe be used, or should the inlet and outlet lines be multiple and manifolded top and bottom so that a minimum of shielding material is removed from any one section of the shield?

Other shield optimization studies should consider the feasibility of reusing some or all of the shield when the rest of the power system is replaced after its useful life. Also the amount of meteoroid armor required should be investigated and possibly a "bumper" or "laminated" vessel wall design may result in a minimum weight shield.

Although the previous discussion has been mainly oriented towards a shadow shield design, a requirement for 4-pi shielding may be imposed upon the nuclear system design for certain selected mission applications. A number of the optimization study areas previously discussed are equally applicable to 4-pi shielding. However, use of 4-pi shielding will require additional studies, particularly in the area of reflector-assembly/shield-assembly interfaces.

Studies to determine optimum separation distance between the nuclear system and the payload dose plane must be conducted as part of any manned mission study. Tradeoff between boom weight and shield weight can be made somewhat independent of the mission, but due to station dynamics final optimization must await a specific mission application.

e. Shutdown and Restart

With the presence of man on-board and with the rendezvous capability undoubtably available, it probably will be required that the nuclear system have the capability of shutdown and restart. This requirement can be easily integrated into the design of the nuclear system. The areas of semiautomatic startup, manual control, and automatic protection circuits should be investigated. Using semiautomatic startup, the startup time can be reduced; however, the time gained vs the manpower required and the added control system complexity must be investigated. The amount and type of automatic protection must be investigated.

In addition, some small percentage of coolant flow may be required for a short period of time after shutdown to remove the decay heat. For extended shutdown, heat may have to be provided to prevent freezing or oxide precipitation in the primary loop. How best to perform these functions must be investigated and incorporated into the design.

Shutdown, replacement, and disposal of the spent system is an area which needs to be investigated as part of any manrating program. With men on-board, certain items could be manually accomplished which would be nearly impossible to do automatically. Utilization of men in this phase of operation should be investigated and tradeoff between automatic and manual operations conducted.

2. Nominal Design Conditions

Table III-20 lists the key design parameters of the Manrated SNAP 8 nuclear system which would result from applying a manrating program to the present SNAP 8 Reference Design. As can be seen from this table, the design performance requirements of the Manrated system are essentially identical to the unmanned reference system.

3. Preliminary Design Description

A preliminary conceptual layout of the Manrated SNAP 8 nuclear system is shown in Figure III-50. This layout is based on a TE power conversion system which requires a minimum of gallery space. (For use with Hg-Rankine system, the gallery height could be increased as necessary.)

TABLE III-20

MANRATED SNAP 8 NUCLEAR SYSTEM

Power level (kwt)	600
Outlet temperature (°F)	1300
Coolant temperature rise (°F)	200
Number of elements	211
Element OD (in.)	0.560
Maximum fuel temperature (°F)	1520
Prepoison leading ($)	3.00
Hydrogen leakage (%/yr)	2.4
Barrier material	SCB
Core length (in.)	16.825
Core ID (in.)	9.214
Reflector thickness (nominal)	3.0
Shielded diameter (shoulder) (in.)	19.0
Shoulder height (in.)	23.5
Midplane diameter (in.)	22.5
Midplane height (in.)	14.5
Control method	active
Controlled reflector elements	7
Reactor-reflector weight (lb)	600
Nominal lifetime (hr)	10^4
NaK flowrate	48,800
NaK ΔP (psi)	4.8
Fuel element cladding	Hastelloy N

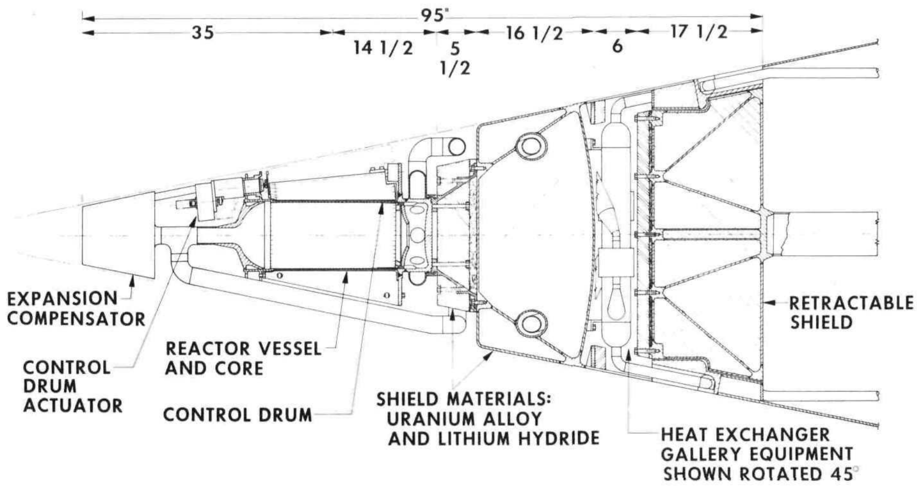

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Figure III-50. Manrated SNAP 8 — Elevation

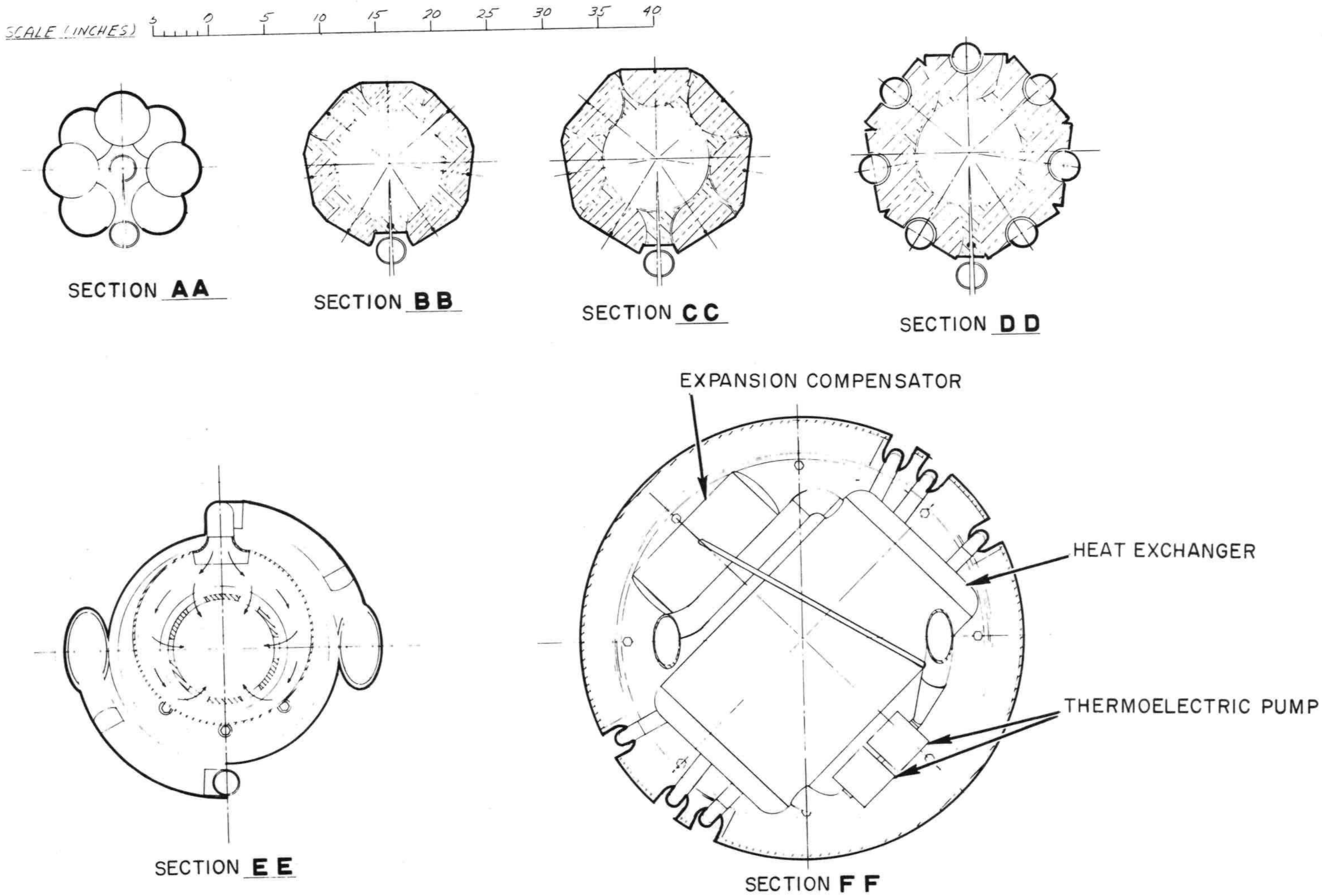

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Figure III - 51. Manrated SNAP 8 — Sectional

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The core internals and that portion of the core vessel and structure immediately adjacent are unchanged from the reference unmanned SNAP 8 nuclear system. Other portions of the core vessel and the reflector assembly have been considerably changed to reduce the overall envelope dimensions and, thus, the system weight. The changes are, however, only in shape and arrangement. No new materials, processes, or components are required in this design.

To minimize reactor assembly envelope, and hence shield weight, the reflector assembly is completely redesigned as seen in Figure III-50. Small diameter beryllium control drums replace the relatively large control drums and the number of drums is increased from six to seven. The drums, generally conical in shape, have their axes of rotation inclined approximately 6° to the axis of the core vessel. This minimizes the reactor assembly envelope by taking advantage of the natural conical shape. The flat sides of the control drums are 1/4 in. from and parallel to the axes of the drums. The conical drum shape results in various proportions of the fixed and movable reflector, with the bottom of the drums suitably cut away so as not to interfere with each other, as shown in Figure III-51. This gives an envelope diameter of 22-1/2 in. at the core midplane. Some additional improvement of the shieldable envelope might be realized by using eight or nine even smaller, tapered drums. The gain would be minor, however, and may not justify the additional system complexity.

The control drums are driven by stepping actuators of a type currently undergoing development testing at Atomics International. These actuators are mounted above the core vessel to eliminate any interference with the piping at the lower end and to place them in the most favorable position for radiation of their heat to space. The diameter of the actuators makes it necessary they be set inward from the drum axes and that a gear train be used to connect the actuator and drum shafts. The gears give a 2:1 reduction of drum motion relative to the actuator motion: the actuators move in 1/2° steps and the drums move in 1/4° steps. This limited available space also requires that the actuators be staggered in height; four are mounted in the lower position and three in the upper.

The reference unmanned SNAP 8 nuclear system control scheme was selected on the basis of its simplicity and inherent reliability. It senses the reactor outlet coolant temperature and maintains this temperature between preset limits by means of a digital control system. As this system adequately controls the reference system, its adaption to the manrated system is ideal as (1) it minimizes on-board maintenance and manual operation due to its unattended long-life (10^4 hr) design; and (2) it maximizes reliability by using an existing design which will have many thousands of test hours on components and integrated systems.

The long-term control system attempts to maximize reliability by minimizing complexity. The controller is designed to allow maximum degradation and two temperature sensors (using voting-type circuitry) are used to maximize sensor accuracy. In addition, periodic readjustment of the temperature switch is possible. Additional pairs of temperature sensors and their switches are provided which can be switched into the control system in the event of a failure of the assembly being used. The controller is of modular construction with test points brought out for ease of maintenance and trouble shooting. On-board maintenance of modules is not anticipated and they would be replaced with spare modules or a spare controller in the event of malfunctions.

The controller output switches power to the control drum motors. The drive motor is a "fail-as-is" device which maximizes nuclear system life and minimizes the probability of inadvertant shutdowns. No scram circuits are provided and shutdown of the nuclear system is accomplished by driving the control drums to their least reactive (out) position. Although no scram circuits are provided, automatic setback may be provided to protect the nuclear power plant against unforeseen incidents. If automatic setback circuits are employed, a bypass switch will be provided so that all setback circuits may be bypassed during critical phases of the mission. Also, setback, if provided, will occur only as long as the variable causing setback remains outside the specified limits and coincidence circuitry may be used to minimize false setbacks.

Consideration is also being given to a mechanical override concept in which a control drum with a failed motor could manually be put in its most reactive position (fully inserted) thereby maximizing system life.

In addition to the long-term control, a startup control mode will be provided to minimize startup time. Using the reactor outlet temperature control and a preprogrammed control drum insertion rate, startup can be automatically accomplished in less than 4 hr. Nuclear instrumentation will be provided so that an operator, if desired, could restart the plant after a shutdown in approximately 2 hr using manual control.

Repeated planned shutdown and restart of the Manrated SNAP 8 nuclear system should not impose any serious design problems. Either automatic startup or semiautomatic startup can be employed as it is anticipated that both modes of startup will be provided. The semiautomatic startup will require about 2 hr while the automatic startup will require about 4 hr. In addition, periodic changes in the automatic startup scheme will probably be required as the excess reactivity available changes during the life of the nuclear system.

Unplanned (accidental or emergency) shutdown of the Manrated SNAP 8 nuclear system may require some shutdown cooling to remove the decay heat. Although the need for this cooling has not been firmly established, there is a firm requirement to maintain the primary NaK coolant above a certain minimum temperature. (This minimum temperature will probably be in the range of 150°F so that oxides in the primary coolant system do not precipitate out of the NaK and cause plugging of the coolant passages.) Hence, cooling of the primary loop immediately after shutdown may be required, followed by heating of the primary loop after extended shutdown. (The use of an electromagnetic pump in the primary loop may be sufficient to perform both those functions although no studies of the SNAP 8 nuclear system, operating at rated conditions, have been performed to verify this.)

An important part of any manrated nuclear power system is the radiation shield. The shield must be light-weight and long-lasting, and must provide adequate protection to the crew from both neutron and gamma radiation emitted from the nuclear system. Since the shield weight is a large fraction of the total system weight, care must be taken to use the most efficient shield materials and the optimum configuration.

As the radiation dose to the payload must be kept to a minimum, no protruding scatter sources may exist. Hence all cabling and NaK piping must be hidden by the shield. Various schemes are under investigation as to ways and means of routing both the cabling and the NaK pipes through the shield. Figure III-52 and III-53 show a method of routing cabling and piping through a series of tubes spiraling up the side of the shield.

The shield designs discussed so far have dealt with the shadow shield configuration. In future applications, 4-pi shielding may also be required for large manned space stations. The problem areas attendant with 4-pi shielding have been briefly investigated. Figure III-54 shows a conceptual design of a 4-pi shield. The use of 4-pi shielding also imposes a strong desire for reduced reactor assembly envelope to minimize shield weight. The use of 4-pi shielding aids in routing cabling and piping as these need no longer be buried in the shield. However, the use of 4-pi shielding does impose a reflector assembly cooling problem and active cooling (either by providing a cool reactor cavity or by cooling the reflector assembly directly) must be provided. These areas have been briefly investigated and further study is merited. No serious development problems are apparent.

4. Reliability

The reliability goal for the Manrated SNAP 8 nuclear system should lie between 0.99 and 0.999 while for the unmanned system the goal is between 0.95 and 0.98. Hence the failure rate of the system must be decreased by a factor of 2 to 40.

Some of this increase in reliability can be accomplished by additional redundancy while some increase can be attained through maintainability. A large fraction of the nuclear system, though, because of the radioactivity associated with it, is not maintainable. The electronic controls associated with the nuclear system are about the only items which are readily maintainable (e.g., by modular replacement). However, these only have a failure rate apportionment of 0.00388 for the unmanned system compared to the total unmanned system failure rate apportionment of 0.0315. Hence, even increasing the reliability of the electronic portions of the nuclear system a factor of ten because of maintainability would only raise the overall system reliability apportionment from about 0.9685 to 0.9720. Essentially the rest of the nuclear system is unmaintainable due to the inherent high radiation.

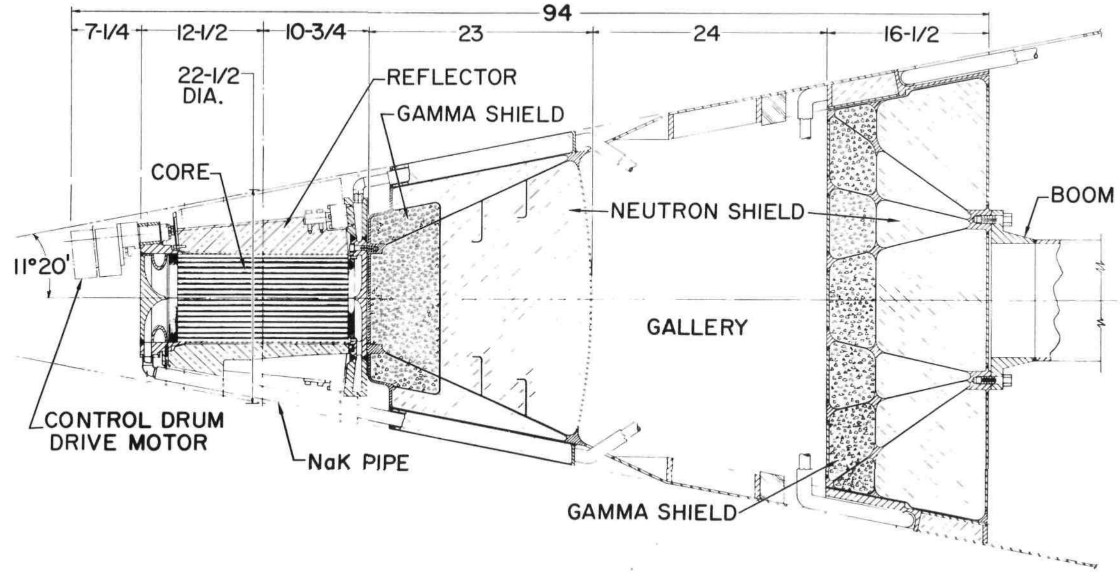

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Figure III-52. Split Shield Detail

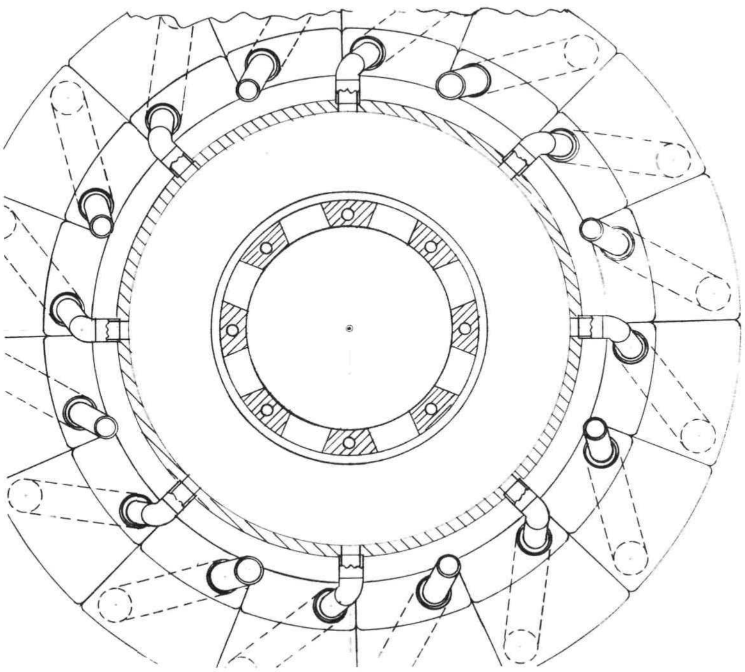

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Figure III-53. Routing of NaK Piping

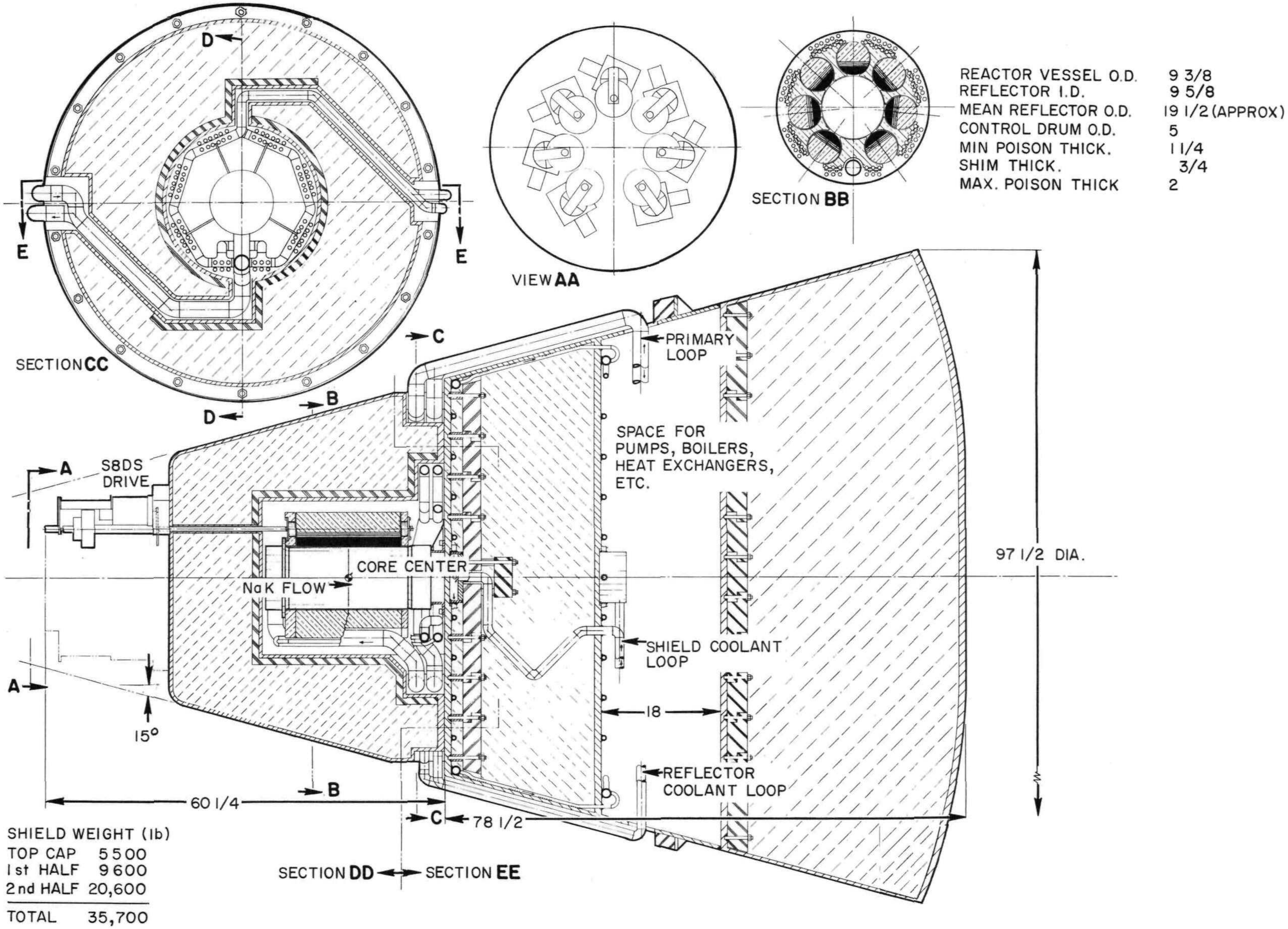

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Figure III-54. 4-pi Shielding

Redundancy is added to the Manrated SNAP 8 nuclear system by increasing the number of control drums from six to seven and providing sufficient excess reactivity to allow two control drum drives to malfunction. Figure III-55 is a plot of reliability of a single drum drive versus the reliability of the control drum system for 0, 1, 2, and 3 allowable malfunctions. If the reliability goal for a single drum system for both the manned and unmanned system is the same, i.e., 0.975, then with two allowable malfunctions in the manrated case, the overall drum drive system reliability goal can be increased from 0.992 to 0.9995. This would increase the overall system goal from 0.9720 to 0.9795.

In addition to the specific examples cited, the reliability goals of other components could be increased by additional design margins and safety factors. The removal of automatic shutdown devices increases the reliability of the nuclear system from 0.9795 to 0.9851. Another feature which will add to the overall reliability of the manned system is the use of a mechanical override concept in which a control drum with a failed motor could be manually put in its most reactive position (fully inserted) and thereby maximize the use of that drum.

Although it is impractical to demonstrate that the desired system reliability goal has been met, it is possible, through sufficient redundancy and high design margin, to design a manrated SNAP 8 nuclear system with a potential reliability in excess of 0.990. It is towards this goal that the design of the Manrated SNAP 8 nuclear system is oriented.

5. Static Control

It appears feasible to adopt the concept of static control for the SNAP 8 reactor. Since one of the most significant effects of such a change would be an increase in potential system reliability, it is appropriate to discuss this under the Manrated SNAP 8.

Detailed studies into four technical areas must be completed to determine the capabilities of the SNAP 8 reactor with static control. The four areas of investigation are: burnable poisons, hydrogen barrier improvements, SNAP 10B type hydrogen retention, and temperature coefficient augmentation.

a. Burnable Poisons

The present SNAP 8 samarium loading has not been optimized for static control. Preliminary investigations indicate that cadmium poison would provide

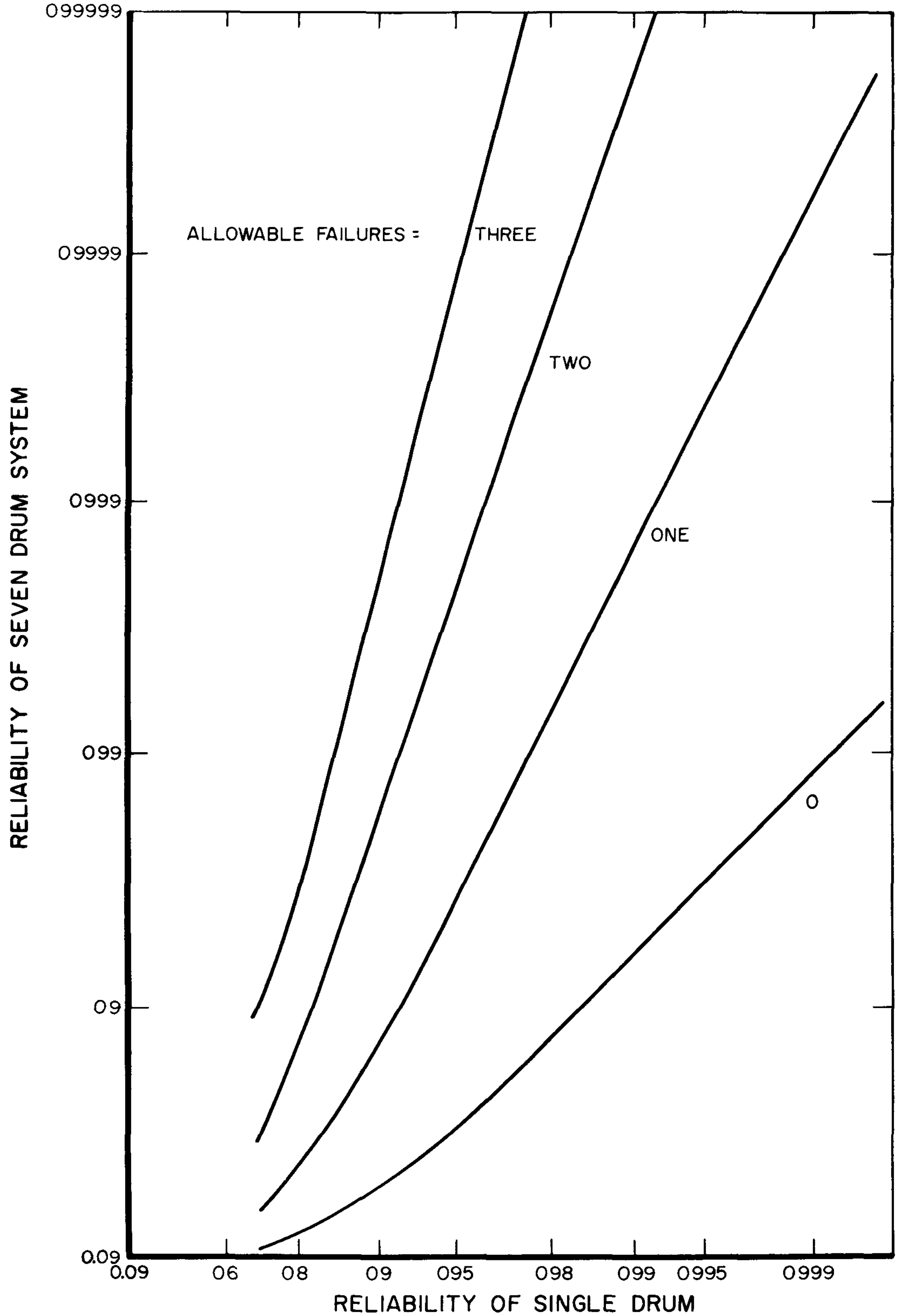

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Figure III-55. Reliability of a 7-Drum System better burnout characteristics in the SNAP 8 power range. The use of mixtures of burnable poisons such as europium and gadolinium could provide additional improvements. The concentration and type of burnable poison must be optimized for each power and outlet temperature level. Also how and where to incorporate the burnable poison in the fuel element must be investigated.

b. Hydrogen Barrier Improvements

Hydrogen loss from the fuel is the largest (and most uncertain) reactivity loss that must be compensated for by the burnable poison. A significant improvement in the fuel hydrogen barrier would greatly ease the problems associated with static control of SNAP 8. The application of a metallic coating to the fuel rod shows promise of a three-fold reduction in hydrogen loss by preventing a chemical reaction between the fuel material and the ceramic hydrogen barrier.

c. Hydrogen Retention

The SNAP 10B approach, using a yttrium getter to retain within the core the hydrogen which leaks from the fuel element, may also be applied to the SNAP 8 reactor. The reduction in the hydrogen loss from the core eases the static control requirements and makes the reactivity changes during reactor operation more predictable.

d. Temperature Coefficient Augmentation

The SNAP 8 reactor has an overall temperature coefficient of reactivity of about -0.2¢/°F. An uncontrolled 10¢ loss in reactivity therefore results in a 50°F reduction in the average core temperature. An increase in the magnitude of the temperature coefficient would reduce the sensitivity of the system to reactivity changes. Various devices to provide temperature coefficient augmentation have been studied in conjunction with the SNAP 10B reactor but none have developed to the point where they may be included in the SNAP 8 design at this time.

The use of thermoelectric pumps in the SNAP 10A and 10B systems does, however, provide about a five-fold increase in the effective power coefficient of reactivity. Since the pumping rate of a coolant excited thermoelectric pump increases with coolant temperature and (at constant power) the coolant temperature decreases with increased flow, the use of such a pump provides a very strong thermal feedback in the system and tends to stabilize the reactor outlet temperature. The use of thermoelectric pumps for the SNAP 8 nuclear system has not been investigated in detail but would probably enhance the static operation potential of the system.

H. UPGRADED SNAP 8

1. Introduction and Description

The Upgraded SNAP 8 nuclear system is a modification of the Manrated SNAP 8 design. It is being designed to produce 1,200 kwt for 10,000 hr of space operation. It is based upon the assumption of permitting only minor differences in configuration compared to the Manrated SNAP 8 design. The minor changes include lengthening the fuel element from about 17 to 24 in. and adding 30 additional fuel elements. Fuel element diameter and internal clearances are not changed. These changes allow operation of the Upgraded SNAP 8 reactor at about twice the SNAP 8 power level while maintaining approximately the same power density and fuel temperatures. The fuel element spacing is increased from 0.010 to 0.030 in. to reduce the core pressure drop. These changes require that the reflector assembly be modified slightly to recognize the increased active core length and the increased core vessel diameter.

Although the Upgraded SNAP 8 nuclear system is based upon the Manrated design, it could also be designed as an upgraded version of the reference design. The Manrated design is used as the basic building block because it is believed that the power level of the modified design will be more useful in manned applications than unmanned applications.

The Upgraded SNAP 8 nuclear system has been analyzed to determine its operating capabilities and limits. However, essentially no design effort has been expended to design the system. No layout drawings have been prepared. It is estimated, however, that the reactor assembly (core and reflector assemblies) will have a 23-1/2-in. diameter at the core centerline and follow, in general a conical shape having a cone angle of about 11°.

2. Design Point

Table III-21 lists the key design parameters of the Upgraded SNAP 8 nuclear system.

TABLE III-21

UPGRADED SNAP 8 NUCLEAR SYSTEM

Power Level (kwt)	1200
Outlet Temperature (°F)	1300
Coolant Temperature Rise (°F)	200
Number of Elements	241
Element OD (in.)	0.560
Maximum Fuel Temperature (°F)	1485
Prepoison Loading ($)	~3.00
Hydrogen Leakage (%/yr)	8.0
Barrier Material	SCB
Core Length (in.)	24.0
Core ID (in.)	9.7
Reflector Thickness (Nominal)	3.0
Shielded Diameter (shoulder), (in.)	21.0
Shoulder Height (in.)	30.0
Midplane Diameter (in.)	23.5
Midplane Height (in.)	17.0
Control Method	Active
Controlled Reflector Elements	7
Reactor-Reflector Weight (lb)	975
Nominal Lifetime	10^4 hr
NaK Flowrate	97,600
NaK ΔP (psi)	~7
Fuel Element Cladding	Hastelloy N

3. Design Point Selection

The Upgraded SNAP 8 reactor design has only minor differences relative to the SNAP 8 Reference Design. Its operational capability is improved by increasing the system's potential useful energy and reliability. The potential useful system energy is a combination of the integrated reactor power and lifetime, modified by the outlet reactor coolant temperature. The outlet coolant temperature influences the potential useful system energy as the electrical power conversion system's efficiency increases with increasing reactor outlet coolant temperature. However, higher outlet coolant temperature increases the rate of fuel degradation and hydrogen moderator leakage. This faster degradation of the reactor would lower the reactor power or lifetime design limitation to maintain the same system capability.

An improvement in the operational capability of a reactor may be considered as an increase in the reactor power level for a given outlet temperature and operating lifetime. Such an increase in reactor power level can be accomplished in two ways:

1) The core volume can be increased so that there is more fuel operating at the same maximum power density.

2) The maximum power density capability of the fuel can be increased, for example, by improving the core heat transfer characteristics.

These changes can also be interpreted as means of increasing the operating lifetime at a constant power level. The decreased fuel temperatures would reduce the amount of hydrogen loss and fuel irradiation damage.

The active core volume of the Upgraded SNAP 8 reactor has been increased by increasing both the length of the fuel rod and the number of fuel elements. The length of the fuel rods has been increased from 16.825 to about 24 in., which corresponds to an increase in reactivity of approximately $5.25. The 24-in. length limit represents the approximate dimensional limitation of the existing hydriding and permeation testing equipment. The number of fuel rods is increased from 211 to 241. These additional 30 fuel elements represent the number of fuel elements which can be added to the modified hexagonal array with a minimal increase in core diameter per fuel rod. These rods, besides adding an approximate 10% in core power generation, add approximately $3.25 in reactivity. The small increase in the core vessel diameter requires only a minor redesign of the reflector assembly. However, a minor redesign of the reflector assembly would be required in any case due to the larger spacing desired between fuel rods to decrease coolant pressure drop and increase the fuel element heat transfer coefficient to the coolant.

Increasing the fuel element power density also increases the operational capacity of a reactor. The fuel element power density can be increased by lowering the fuel rod diameter and by increasing the heat transfer coefficient between the fuel and the coolant. A smaller fuel rod would maintain nearly the same fuel temperature, for the same power per unit length of fuel, while increasing the number of fuel rods and system power in the same core volume. The development required to redesign the manufacturing and inspection equipment to accept a smaller diameter fuel element constitutes a substantial effort.

The spacing between the fuel elements in the Upgraded SNAP 8 has been increased to 30 mils from the 10 mils used in the reference design. This increased spacing is predicted to result in a 10% increase in the heat transfer coefficient between the fuel and the coolant. The fuel may, therefore, be operated at a higher power density than in the SNAP 8 reference design while maintaining the same fuel swelling and hydrogen loss criteria.

The principal reason for increasing the fuel element spacing from 0.010 in. to 0.030 in. is to decrease the coolant pressure drop along the fuel elements. This could be a significant contribution to the total reactor pressure drop, tentatively selected as 10 psi for this design. Figure III-56 shows differential plenum pressure (excluding inlet and outlet nozzle drops) as a function of element gap for the SNAP 8 Reference Design. Considering both the core diameter enlargement as well as the increased element length, it can be concluded that fuel element pressure drop will not become dominant with a 0.030-in. spacing. The detrimental effects of increasing the rod spacing are the increased core diameter, which increases the shield weight; and the decreased atomic density of the fuel and moderator, which decreases the reactor excess reactivity. The 0.030-in. spacing appears to be a near optimum condition which decreases the coolant pressure drop by over 50% while increasing the core vessel diameter less than 0.5 in. The reactivity loss associated with the increased rod spacing is approximately $2.40

The other method of extending the operational capability of the reactor is by increasing the reliability of the system. The Upgraded SNAP 8 reactor fully utilizes the SNAP 8 design approach to reliability by redundancy. The use of redundant components allows a lower individual reliability goal while still achieving a high system reliability goal. In the SNAP 8 nuclear system, the redundant component approach is possible largely by providing sufficient excess reactivity to allow for failures and malfunctions of fuel elements and control drum drive assemblies. Therefore, to a first-order approximation, the relative reliability

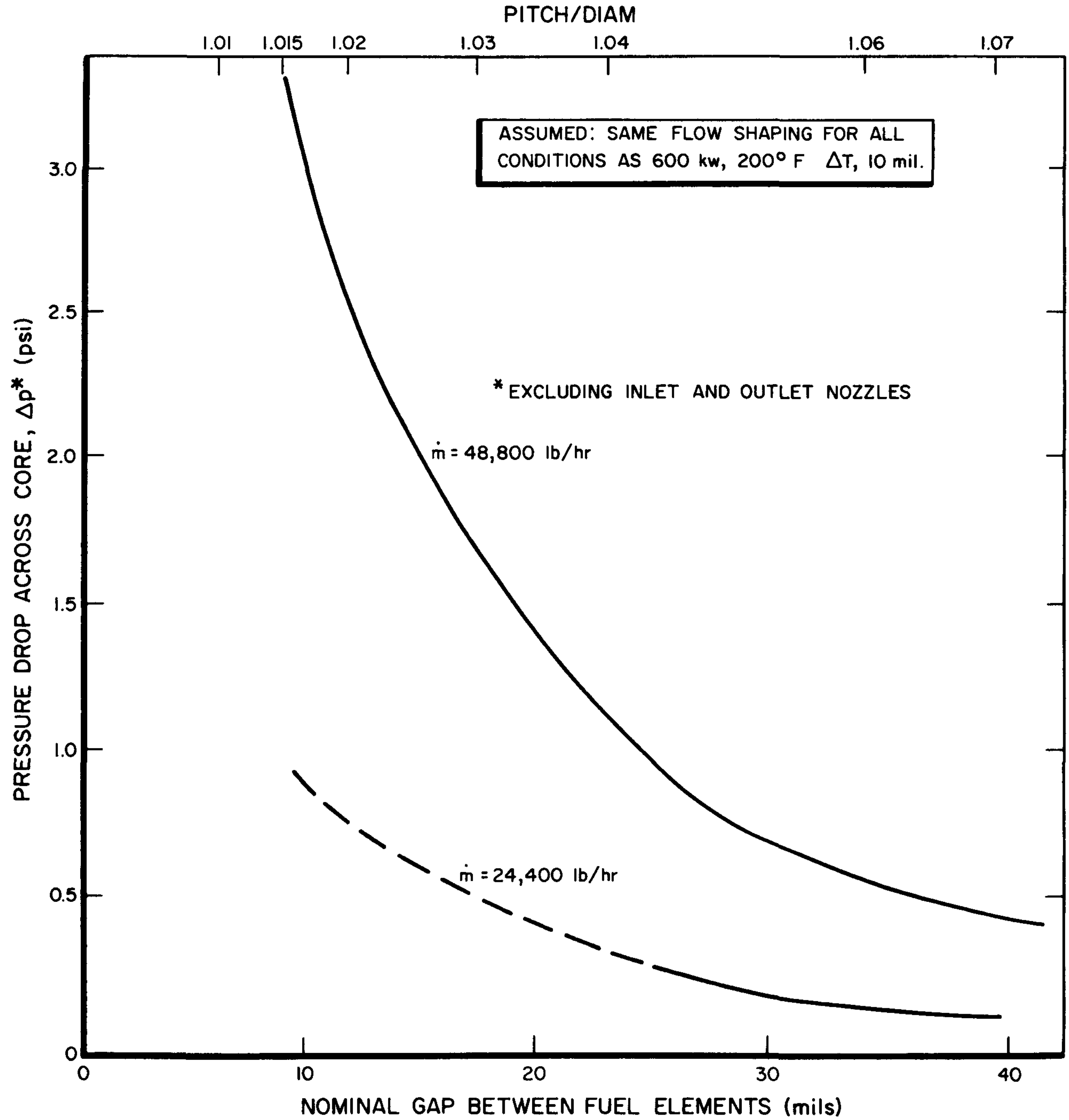

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Figure III-56. Core Pressure Drop

of operating the nuclear system at various design conditions can be determined by the amount of excess reactivity available.

For the SNAP 8 Reference Design, although the conservative design criteria is not reactivity limited, the operational capability of the reactor is reactivity limited. The three minor design changes in the modified SNAP 8 system of increased length, number of fuel rods, and increased spacing between fuel rods, results in a net increase of $6.10 in the reactor excess reactivity. With the increased excess reactivity, the Upgraded SNAP 8 reactor's operational capability is not reactivity limited under all applicable powers, outlet coolant temperatures or operating lifetimes. Therefore, fuel elements and/or control drum drive assemblies could fail without affecting the operational capability of the Upgraded SNAP 8 reactor.

4. Parametric Study of Off-Design Performance

The Upgraded SNAP 8 reactor is designed to operate at 1200 kwt for 10,000 hr with an outlet coolant temperature of 1300°F. This design point lies well within the basic design criteria limits of the reactor. The following items act as limits to the reactor performance:

a) excess reactivity,

b) fuel swelling,

c) fuel phase change,

d) fuel cladding creep, and

e) irradiation swelling of the beryllium reflector.

The performance of the Upgraded SNAP 8 reactor has been studied and compared both with the conservative design criteria limits and the more realistic operational limitations. These studies were performed over a range of reactor power levels, outlet coolant temperatures and operating lifetimes. Some of the results of these studies are provided in the following table:

	Design Point	Design Criteria Limit	Operational Limit
Power output for 10,000 hr at 1300°F	1.2 Mwt	1.5 Mwt	1.8 Mwt
Power output for 10,000 hr at 1400°F	—	1.0 Mwt	1.3 Mwt

Since the Upgraded SNAP 8 reactor is not reactivity limited, these power levels would be attainable even though some failures occurred in the fuel elements and/or control drum drive assemblies.

a. Excess Reactivity

The ultimate operational limitation of the Upgraded SNAP 8 nuclear system is excess reactivity. The reactivity lifetime is that time at which all excess reactivity has been used. Failure of a fuel element by fuel swelling, phase change or cladding creep would result in excessive hydrogen leakage from that fuel element, which is a reactivity loss. However, due to the approximate 7-in. increase in fuel length and 30 additional fuel elements, the Upgraded SNAP 8 core has a large amount of reactivity available. Figure III-57 illustrates that for the Upgraded SNAP 8 reactor operating at its design point of 1.2 Mwt and 1300°F outlet coolant temperature for 10,000 hr, the system uses less than $8.00 of the $17.00 available.

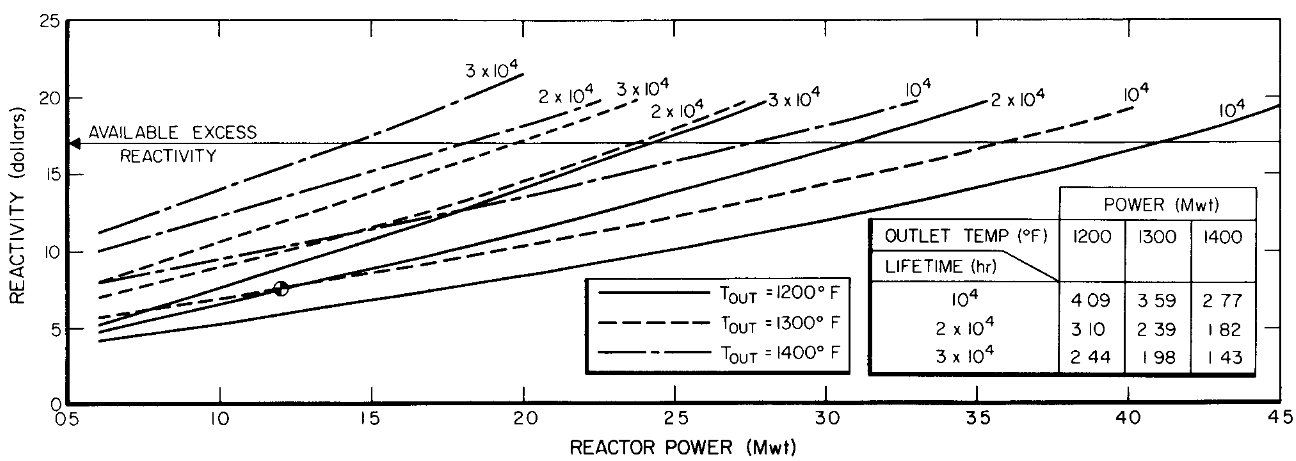

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Figure III-57. Upgraded SNAP 8 Reactivity Requirements

The principal reactivity requirements of the Upgraded SNAP 8 reactor are:

1) fission product poisons and fuel burnup,

2) temperature defect, and

3) hydrogen leakage out of the fuel elements.

The reactivity losses were based on the reactor operating with the:

1) nominal core power fuel burnup and fission product buildup,

2) temperature defect corresponding to the hot channel fuel temperatures,

3) hydrogen loss out of the hot channel fuel element with the fuel temperature remaining constant during the reactor lifetime.

4) nominal N_H, and

5) nominal control drum shim configuration.

These assumptions are the same as the SNAP 8 Reference Design except that the nominal hydrogen concentration of the fuel (N_H) was increased to 6.10, to obtain the optimum phase change and cladding creep combination instead of using the specification (6.05) of the Reference Design. As shown by Figure III-58, at 10,000 hr, the reactivity limitation of the nominally operating reactor exceeds all other design and operational limits. If it were not for other limits, the reactor would have sufficient reactivity to operate at over 2.4 Mwt with a 1400°F outlet coolant temperature for 10,000 hr. As shown by Figures III-59 and III-60, reactivity does not limit operation of the Upgraded SNAP 8 reactor for 20,000 or 30,000-hr lifetimes.

b. Fuel Swelling

The fuel swelling of the Upgraded SNAP 8 reactor is based on the following assumptions:

1) The fuel element is operating at the hot channel condition.

2) The location of the fuel corresponds to the peak burnup condition.

3) For the design criterion, no straining of the cladding is permitted when the fuel is at ambient temperature with a homogenized hydrogen concentration, i.e., isothermal shutdown of the reactor near end-of-life. For the operational limitation, it is conservatively assumed that a 5-mil diametral strain of the cladding will result in failure.

4) The fuel temperature changes due to fuel swelling are decreasing the hydrogen gap and reducing the thermal conductivity of the fuel.

The assumptions used in defining the fuel swelling limitations of the Upgraded SNAP 8 reactor are identical to the assumptions of the SNAP 8 Reference Design

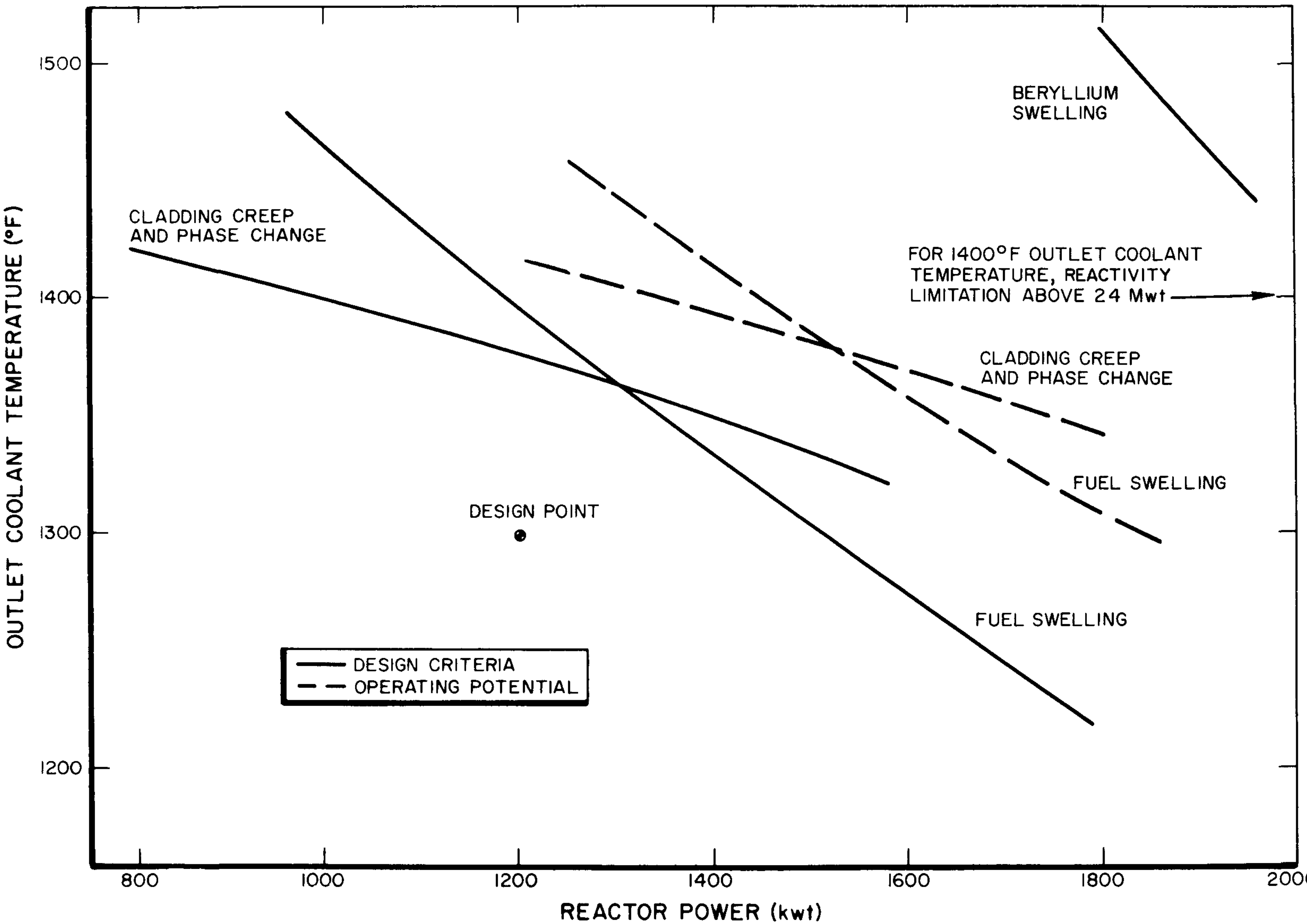

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Figure III-58. Upgraded SNAP 8 Fuel Limitations — 10^4 hr

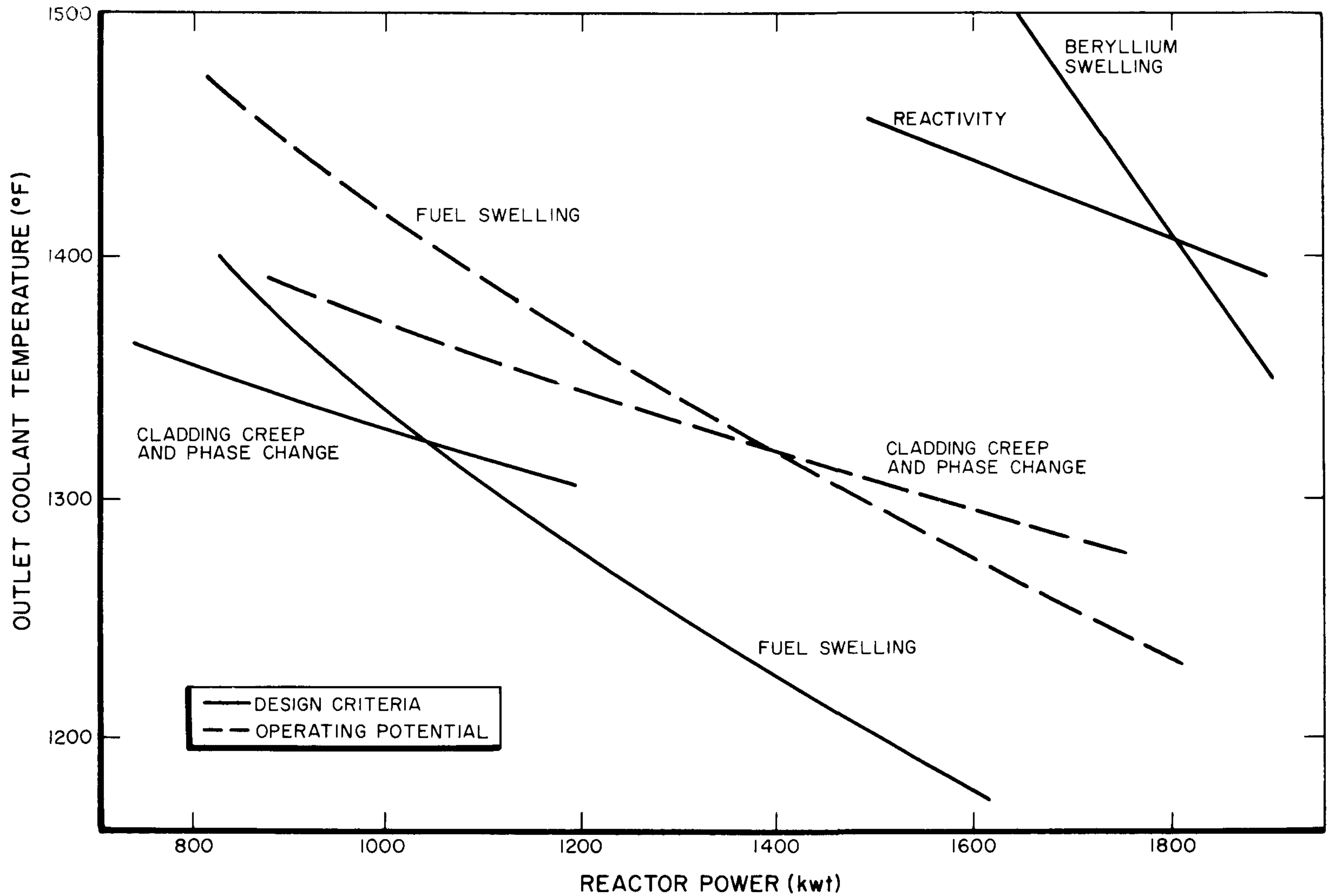

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Figure III-59. Upgraded SNAP 8 Fuel Limitations — 2 x 10^4 hr

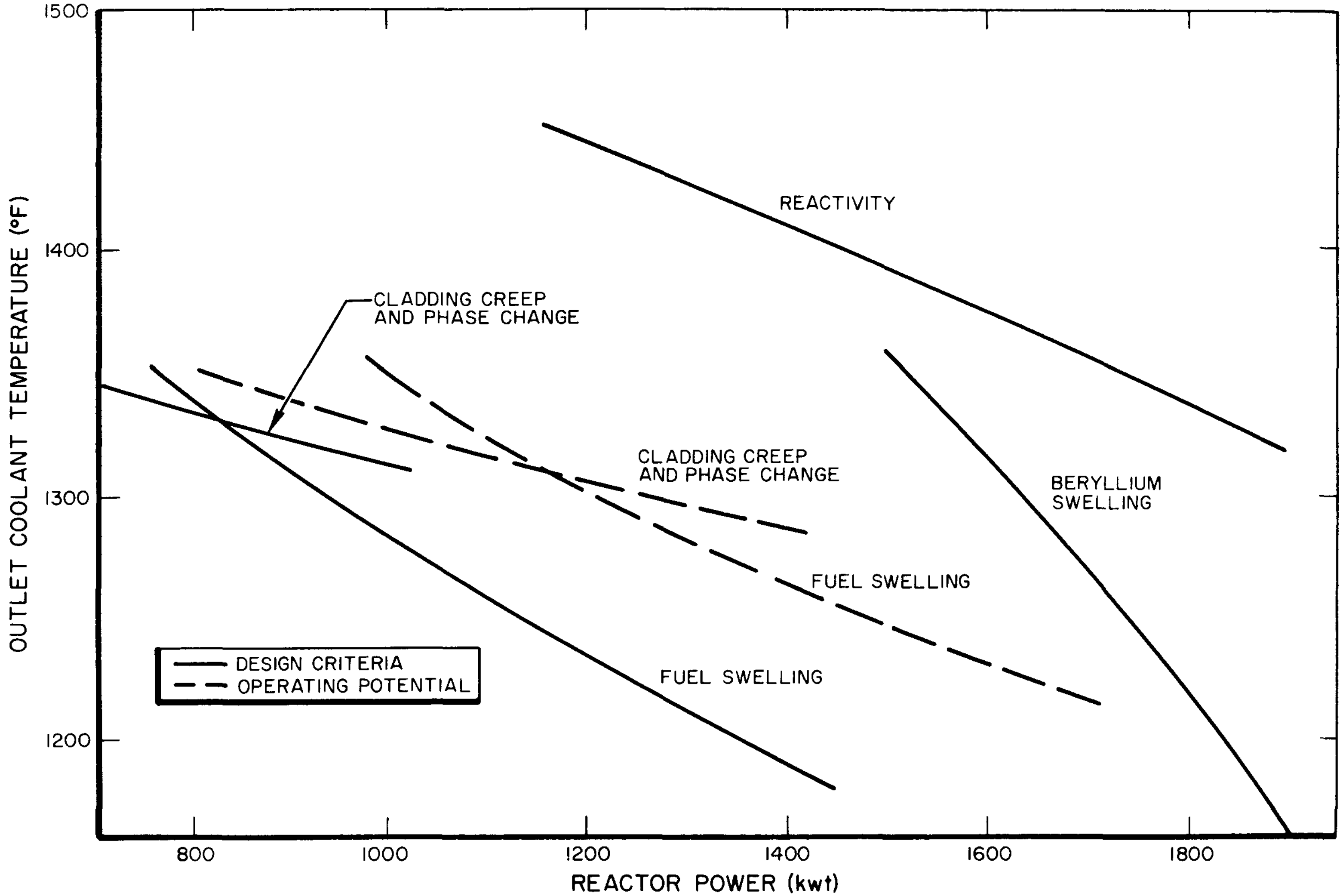

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Figure III-60. Upgraded SNAP 8 Fuel Limitations — 3×10^4 hr

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except for one major difference and one minor difference. The major difference is the much more realistic assumption that the fuel element temperature decreases as the gas gap shrinks due to fuel swelling instead of the extremely conservative assumption used in the Reference Design that there is no decrease in fuel element temperature as the gas gap decreases during the reactor lifetime. The minor difference is the change in the safety factor applied to the decrease in fuel thermal conductivity due to burnup from the extremely conservative value of 50 to a still conservative value of 10.

These two changes in the assumptions used to define the fuel swelling design limitation are considered to be justified since the present fuel irradiation program is generating experimental data which will decrease the uncertainty in the fuel swelling behavior. The fuel swelling design limitation used in the Upgraded SNAP 8 still maintains a considerable amount of the conservatism of the SNAP 8 Reference Design.

The fuel swelling operational limitation of the Upgraded SNAP 8 assumes exactly the same conservative constraint as the SNAP 8 Reference Design. The fuel swelling limitation is reached when the hot channel fuel element, having the maximum reactor burnup, has strained the cladding 5 mils on the diameter.

Figure III-58 illustrates that fuel swelling is one of the limits of the operating range of the 10,000 hr lifetime, Upgraded SNAP 8 reactor. Both the design criterion and the operational limitations are fuel swelling limited for high reactor power operation to ~ 1.5 Mwt and 1.8 Mwt, respectively, for a 1300°F outlet coolant temperature and 10,000 hr of operation. Figures III-59 and III-60 show similar high power fuel swelling limitations for the Upgraded SNAP 8 operating 20,000 and 30,000 hr, respectively.

c. Phase Change and Cladding Creep

The limitations of phase change and cladding creep are interrelated. The fuel phase change design criteria is defined as operation until any part of the fuel element (with the lowest fuel specification hydrogen concentration (N_H) operating in the hot channel condition) loses sufficient hydrogen to just reach the β-δ phase interface. The cladding creep limitation is defined as operation until the hydrogen dissociation pressure (corresponding to the highest fuel specification hydrogen concentration operating in the hot channel condition) causes the cladding to creep in excess of the desired boundary. The nominal fuel hydrogen concentration was selected so that the fuel would reach the β-δ interface when the cladding has reached the creep limitation.

The analysis of cladding creep assumed a 9-mil cladding thickness and used a 1.4 factor of safety on the imposed load. The static NaK pressure was increased in order to counteract the higher hydrogen dissociation pressure caused by the increased N_H operating with the same hot channel fuel temperatures. The upper limit on the NaK static pressure was restricted to that value required to just strain the hottest location of the core vessel 0.2% during the reactor lifetime. The core vessel stress also had a 1.4 factor of safety. A core vessel thickness of 0.12 in. was estimated to be adequate to withstand the launch shock and vibration requirements. The N_H should not be increased above that corresponding to the upper limit NaK pressure, because the reactivity loss due to increasing the vessel thickness, necessary to maintain the 0.2% creep limit, exceeds the reactivity gain due to the increased N_H.

The phase change design criterion and operational limitation are both assumed to occur when the phase change occurs, as assumed with the SNAP 8 Reference Design. The cladding creep design and operational limitations are also the same as SNAP 8 Reference Design and are estimated to occur when the cladding has undergone a creep of 0.2% and 1.0%, respectively.

Figure III-58 illustrates that cladding creep and phase change are one of the limits of the operating range of the 10,000-hr lifetime Upgraded SNAP 8 reactor. Both the design criteria and the operational limitations are cladding creep and phase change limited for high outlet coolant temperature operation at approximately 1375 and 1415°F, respectively, for a 1.2 Mwt reactor power and 10,000 hr of operation. Figures III-59 and III-60 show similar high outlet coolant temperature limitations for the Upgraded SNAP 8 operating for 20,000 and 30,000 hr, respectively.

d. Beryllium Swelling (and core vessel creep)

Two additional interacting design criteria are the core vessel creep and the beryllium swelling. The initial gap between the core vessel and the beryllium reflectors must be large enough to insure clearance between the surface at all times. The gap cannot be excessively large due to the decreased reflector worth. The operational limitation is conservatively defined as equal to the design criteria. The design criteria restricts the core vessel lifetime creep to 0.2% due to NaK static pressure and limits the beryllium deflection to 30 mils due to isotropic swelling.

The beryllium swelling configuration for the design criteria is assumed to correspond to tapered small control drums which have:

a) the beryllium swelling along the entire drum length corresponding to the point of maximum temperature and maximum integrated fast neutron flux during the reactor lifetime,

b) the swelling occurring on the inside (near reactor) drum surface which causes the reflector to buckle assuming that the ends of the reflector are unrestrained, and

c) the beryllium deflection limited to 30 mils of the 45 mils available.

As shown by Figures III-58, III-59, and III-60, the beryllium swelling limitation of the Upgraded SNAP 8 reactor operating for 10,000, 20,000, and 30,000 hr does not limit the system operation.

5. Reliability

The Upgraded SNAP 8 reactor design utilizes the components and technology developed for the Manrated and/or Reference Design SNAP 8 systems. The basic approach for achieving high reliability through the use of redundancy is maintained. In general, the system would be expected to achieve the same reliability. There are, however, two factors which will result in a significantly higher system reliability.

The individual component reliability for the Upgraded SNAP 8 should be higher than that for the Manrated or Reference Design, due to the time lag between the development of the two systems. Testing of the Reference Design or Manrated SNAP 8 system will uncover problem areas which will be corrected prior to operation of the Upgraded SNAP 8 system.

The design changes which have been incorporated in the core design have resulted in a greater increase in the excess reactivity than in the reactivity requirements. This provides a greater surplus of excess reactivity which is available to compensate for unexpected reactivity losses. The Upgraded SNAP 8, therefore, has greater redundancy than the Reference or Manrated designs in the core and reflector assemblies.

Although the Upgraded SNAP 8 reactor is designed to produce twice the power of the Reference Design, nothing has been done to degrade the system reliability. The overall system reliability has, in fact, been significantly improved.

J. ADVANCED ZrH REACTOR

1. Introduction and Description

The Advanced ZrH reactor is a design which approaches the upper limit of reactor performance which can be attained within the present state-of-the-art U-ZrH_x SNAP reactor technology. It is assumed that a rather complete redesign of the reactor can be made but no additional extensive research and development effort will be required. The Advanced ZrH reactor therefore represents the logical growth of the SNAP 8 system to higher power levels.

In order to increase the power density of the core, the fuel element diameter has been decreased to 0.428 in. from the present 0.560 in. The total power output of the reactor is further increased by increasing the overall core size. The active fuel length is 27 in. and the diameter of the core is 12 in. These changes provide a system capable of delivering 3 Mw of thermal power for 10,000 hr with an outlet coolant temperature of 1300°F.

The reflector assembly is similar to those for the Manrated SNAP 8 and the Upgraded SNAP 8 systems. The larger core diameter permits the installation of eight rather than seven control drums.

Conceptual design studies have been performed which show the feasibility of the Advanced ZrH reactor.

2. Design Point

Table III-22 lists the key design parameters of the Advanced ZrH nuclear system.

3. Design Point Selection

The Advanced ZrH reactor is a major redesign of the SNAP 8 concept which approaches the largest operational capability of a SNAP U-ZrH_x system using existing state-of-the-art research and technology. Four basic constraints are imposed upon the design:

a) the use of present SNAP 8 structural materials and U-ZrH$_x$ fuel (minor amounts of additive, for example a neutron poison, might be incorporated in the fuel)

b) the use of the rotating drum control technique (beryllium, if possible)

c) the use of NaK-78 as the coolant

d) the ability to scram and restart the reactor.

TABLE III-22

ADVANCED ZrH NUCLEAR SYSTEM

Power Level (kwt)	3000
Outlet Temperature (°F)	1300
Coolant Temperature Rise (°F)	200
Number of Elements	583
Element OD, (in.)	0.428
Maximum fuel temperature (°F)	1565
Prepoison Loading ($)	~3.00
Hydrogen Leakage (%/yr)	7.6
Barrier Material	SCB
Core Length (in.)	27.0
Core ID (in.)	12
Reflector Thickness, Nominal	3.0
Shielded Diameter, Shoulder (in.)	24.0
Shoulder Height (in.)	33.0
Midplane Diameter (in.)	26.5
Midplane Height (in.)	18.5
Control Method	Active
Controlled Reflector Elements	8
Reactor-Reflector Weight (lb)	2000
Nominal Lifetime (hr)	10^4
NaK Flowrate	224,000
NaK ΔP (psi)	~10
Fuel Element Cladding	Hastelloy N

The largest operational capability can be attained by designing the largest controllable core size containing the most fuel elements operating at their highest power density. The diameter of the core is limited primarily by control drum worth. As the core diameter is increased, the radial neutron leakage and the corresponding control drum worth is decreased.

The control drums must have enough worth at all times during the entire design lifetime to compensate for the uncertainty in the fuel reactivity due to manufacturing tolerances of the built-in poison, uranium and hydrogen concentrations; to shutdown and restart the reactor; and to offset the reactivity changes due to the counteracting forces of prepoison burnout vs fuel burnout and hydrogen loss. Based on an initial Advanced ZrH parameter study using an 0.40-in. fuel element diameter, and a 3-in. reflector thickness, an indication of the permissible nuclear design range was obtained (Figure III-61). The permissible design range is approximately bounded by a k_{eff} of the core greater than 1.10 to provide adequate lifetime, and a total reflector worth greater than about 20% $\Delta k/k$ to permit adequate control during the reactor lifetime. The somewhat conservative core diameter of 12 in. and an active core length of 27 in. were selected. This produced a core which was nearly three times the volume of the SNAP 8 Reference Design. The core length-to-diameter ratio is slightly in excess of 2. Previous design studies have shown this to be near the optimum ratio to minimize shield weight.

To obtain the maximum power density within the core, it is desired to decrease the fuel rod to as small a diameter as possible. For the same power generated per unit length of fuel, the radial temperature drop across the fuel rod is independent of the rod diameter. Therefore, a smaller fuel element can operate at the same maximum fuel temperature while increasing the number of fuel rods and system power in the same core volume.

The minimum fuel rod diameter appears to be limited by manufacturing techniques. It has been estimated that a 3/8-in. fuel rod is about the smallest size which can be extruded, hydrided, and machined while maintaining the required straightness of less than 0.010 in. along the 27-in. length. Therefore, a near optimum fuel element configuration for the Advanced ZrH reactor appears to be 583 fuel elements having a fuel rod diameter of 0.400 in. and an outside diameter of 0.428 in.

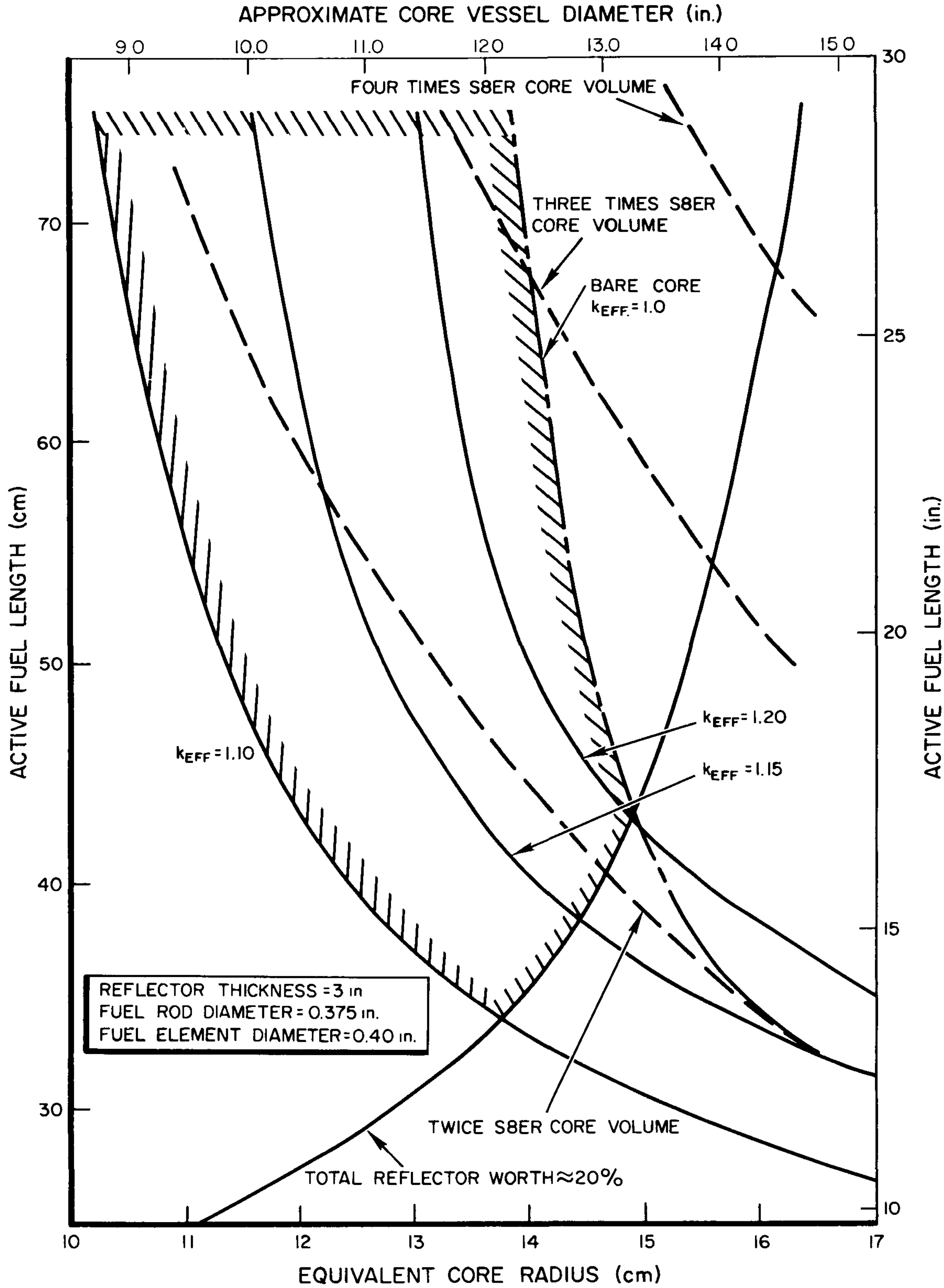

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Figure III-61. Advanced Zirconium Hydride Reactor Nuclear Design Range

A second order increase in the fuel power density is obtained due to the increase in the heat transfer coefficient resulting from an increase in the spacing between the fuel elements from 0.010 in. to 0.030 in. A 10% increase in the heat transfer coefficient was conservatively assumed in the calculations.

4. Parametric Study of Off-Design Performance

The Advanced ZrH reactor is a major system redesign and some developmental effort will be required. In the time required to design and develop the Advanced ZrH, the research and developmental work being carried on by the existing SNAP 8 Program should be able to significantly decrease the uncertainties in material behavior and improve the hydrogen barrier performance. This logical advance in the state-of-the-art is applied to liberalize some of the more conservative assumptions used in the fuel swelling and cladding creep limitations as well as for the hydrogen leak rate of the fuel elements. Therefore, only these more realistic, but still conservative, design criteria are used to indicate the system limitations of the advanced SNAP 8 reactor.

a. Excess Reactivity

Primarily due to the large core size, the Advanced ZrH reactor has a very high effective multiplication. The large core diameter, however, also results in a reduced total control drum worth. Under these conditions, effective utilization of the reactivity potential will depend upon judicious use of burnable poisons. If the burnable poison does not provide good compensation for the normal reactivity losses, the reactivity requirements of the system will approach the amount of control available. A large, unexpected reactivity loss (failure of one or more control drives, for example) would then prevent the reactor from achieving the desired lifetime. Preliminary calculations indicate that it will be possible to utilize most of the available excess reactivity.

For the system operating in its nominal hot channel condition, Figure III-62 illustrates that the Advanced ZrH reactor operating at its design point of 3 Mwt, and 1300°F outlet coolant temperature for 10,000 hr uses up less than \$11.00 of the \$25.00 available.

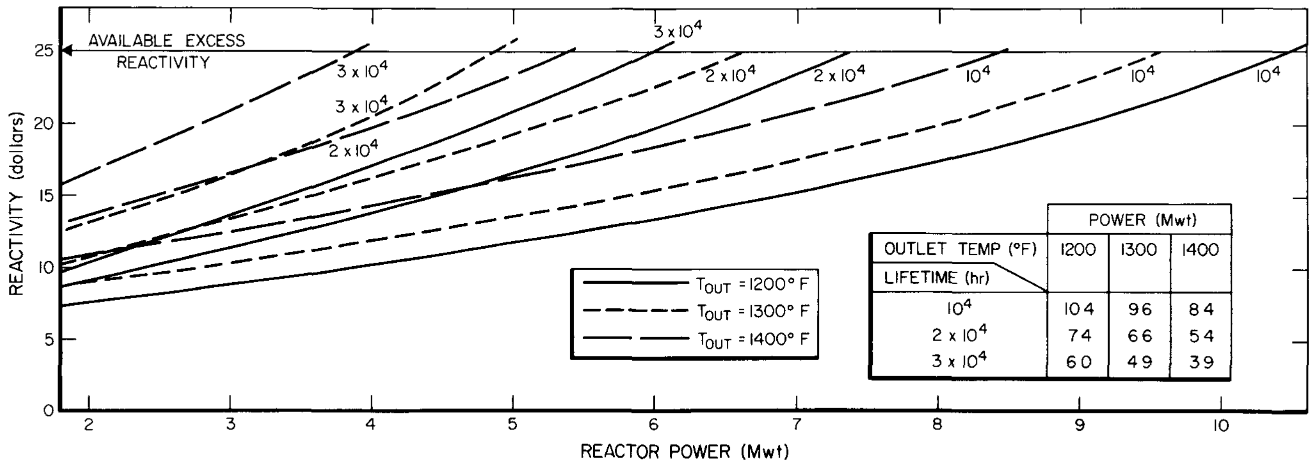

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Figure III-62. AZH Reactivity Requirements

The principal reactivity requirements of the Advanced ZrH reactor are:

1) fission products and fuel burnup,
2) temperature defect, and
3) hydrogen leakage out of the fuel elements.

The reactivity losses were based on the reactor operating with:

1) nominal core power fuel burnup and fission product buildup,

2) temperature defect corresponding to the hot channel fuel temperatures,

3) hydrogen loss out of the hot channel fuel element with the fuel temperature remaining constant during the reactor lifetime and the assumption that the hydrogen leak rate per unit of surface area is approximately equal to the value of presently manufactured fuel elements,

4) optimum N_H, and

5) nominal control drum shim configuration.

These assumptions are the same conservative ones as those used for the Upgraded SNAP 8 design except for the more realistic hydrogen leak rate from currently fabricated elements, which is 2 to 3 times better than the specification value used previously in the discussion of the design limitations of the SNAP 8 reactors. In addition, there has been an improvement in each production batch of elements to date, i.e., the S8DS prototype elements had a lower leak rate than the S8DRM1 elements which had a lower leak rate than the S8ER fuel elements. Hence, considering the scheduling time of an Advanced ZrH reactor, it appears realistic and still conservative that the hydrogen leak rate per unit surface are for the Advanced ZrH reactor be assumed to be the same as fuel elements currently being fabricated.

As shown by Figure III-63, the reactivity limitation of the normally operating reactor exceeds all design limits for the reactor operating 10,000 hr. If the reactor materials could remain integral, the reactor would have sufficient reactivity to operate at over 8.4 Mwt with a 1400°F outlet coolant temperature for 10,000 hr. As shown by Figures III-64 and III-65, the reactivity does not limit operation of the Advanced SNAP 8 system for 20,000- or 30,000-hr lifetimes.

b. Fuel Swelling

The fuel swelling of the Advanced ZrH reactor is based on the following assumptions:

1) The fuel element is operating at the hot channel condition.

2) The location of the fuel corresponds to the peak burnup condition.

3) For the design criterion, it is conservatively assumed that a 5-mil diametral strain of the cladding is permitted when the fuel is at ambient temperature with a homogenized hydrogen concentration, i.e., shutdown of the reactor near end-of-life.

4) The fuel temperature changes due to fuel swelling decreasing the hydrogen gap and reducing the thermal conductivity of the fuel.

The assumptions used in defining the fuel swelling limitations of the Advanced ZrH reactor are identical to the assumptions of the Upgraded SNAP 8 design except for one major difference. The design criterion of the Upgraded SNAP 8 design used the conservative assumption that the fuel swelling limitation was reached prior to straining the cladding, rather than allowing 5-mil diametral strain as assumed for the Advanced ZrH reactor.

The decrease in conservatism of the fuel swelling design limitation relative to the Upgraded SNAP 8 design limitation is assumed to be justified since the present fuel irradiation program is generating experimental data which will decrease the uncertainty in the fuel swelling behavior. In addition, as discussed in Section III-F-3-b(2), out-of-pile straining of the ceramic hydrogen barrier to 4 mils on the diameter showed no detrimental effects on the performance of the

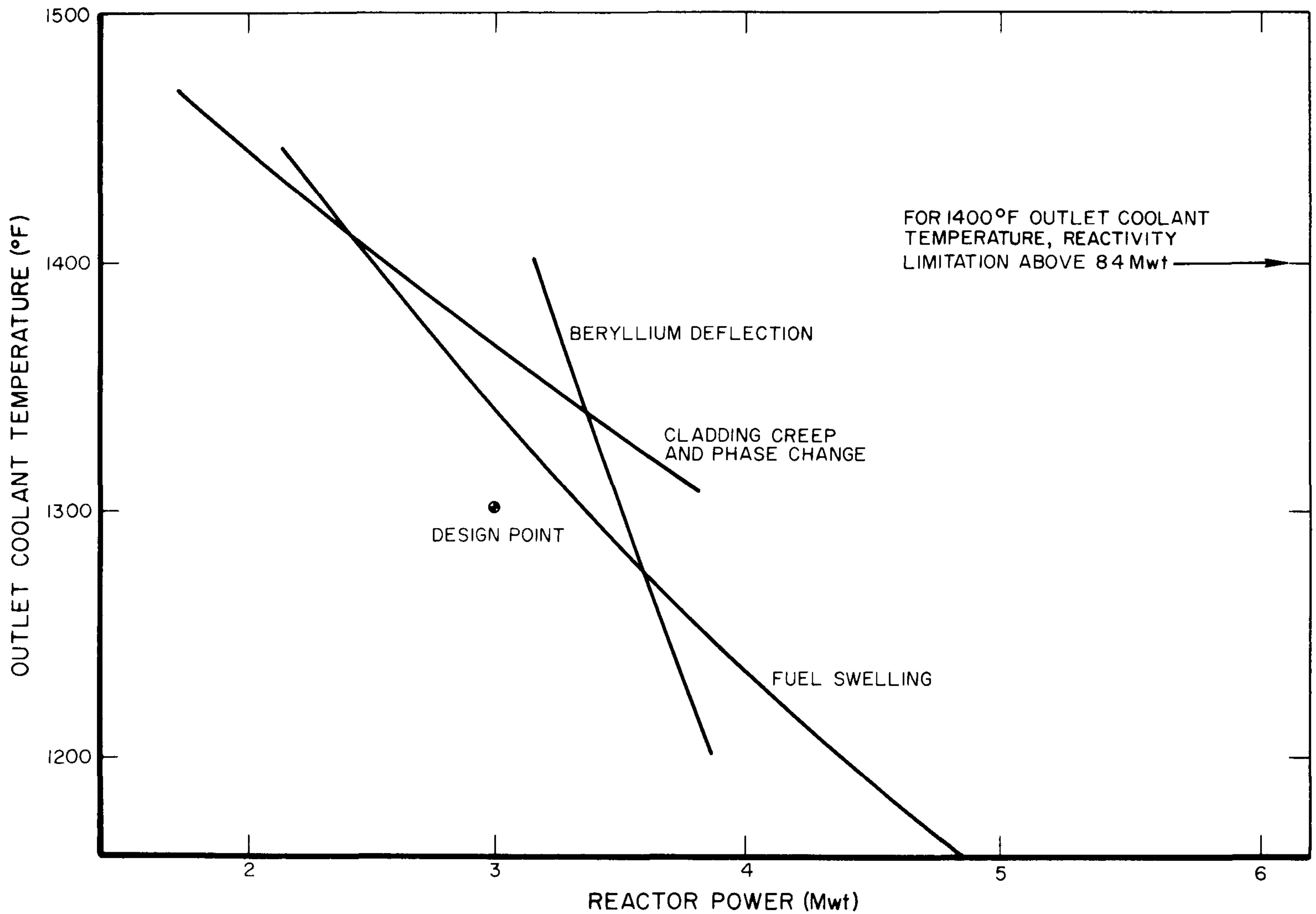

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Figure III-63. AZH Fuel Limitations — 10^4 hr

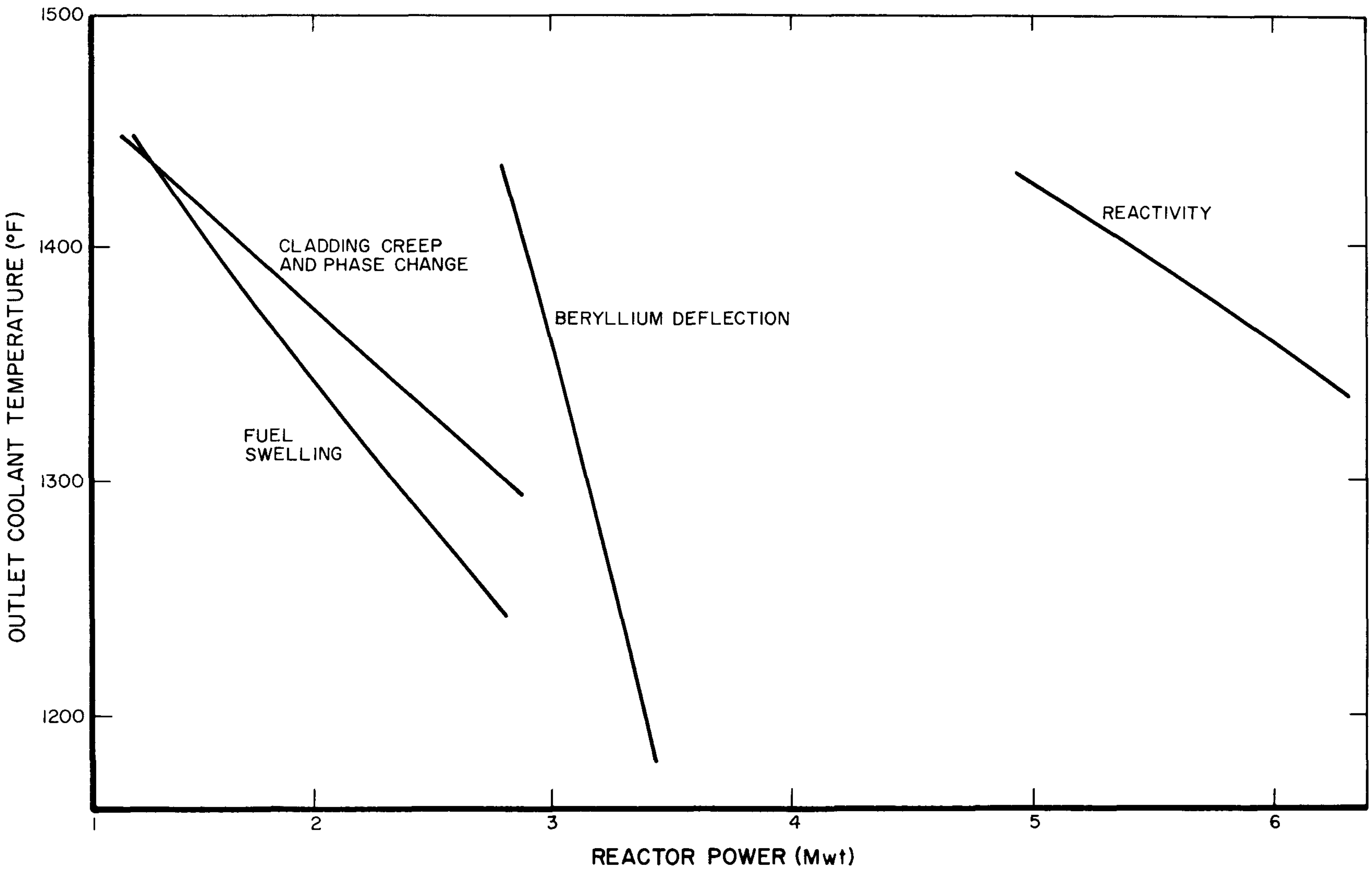

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Figure III-64. AZH Fuel Limitations – 2 x 10^4 hr

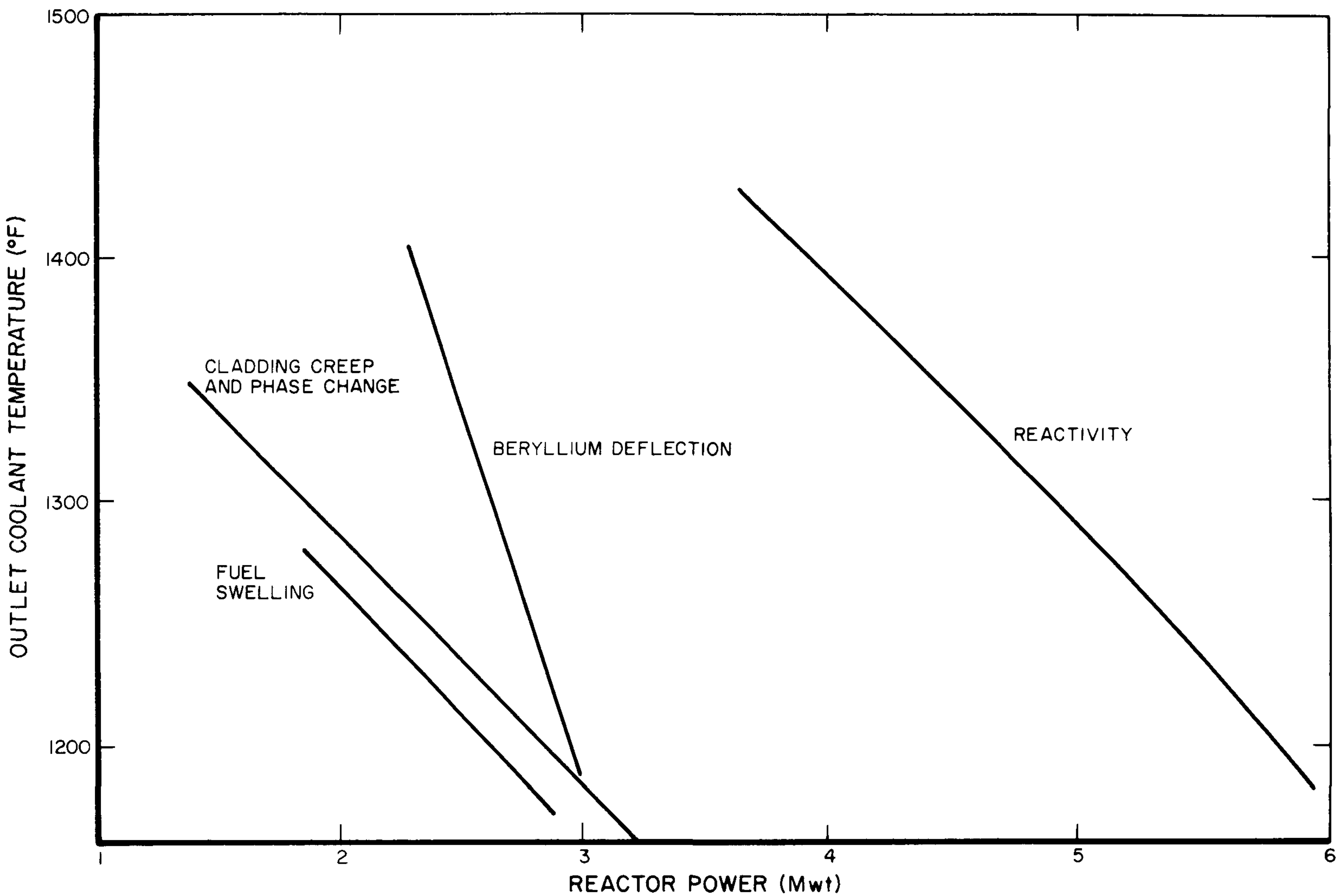

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Figure III-65. AZH Fuel Limitations — 3 x 10^4 hr barrier. Hence, the relaxation of the conservative requirement, that no straining of the cladding be allowed, appears to be reasonable.

Figure III-63 illustrates that fuel swelling is one of the limits of the operating range of the 10,000-hr lifetime Advanced ZrH reactor. The fuel swelling limit is reached at approximately 3.3 Mwt for a 1300°F outlet coolant temperature and 10,000 hr of operation. Figures III-64 and III-65 show similar high power fuel swelling limitations for the Advanced ZrH operating 20,000 and 30,000 hr, respectively.

c. Phase Change and Cladding Creep

The limitations of phase change and cladding creep are interrelated. The fuel phase change design criteria is defined as operation until any part of the fuel element (with the lowest fuel specification hydrogen concentration operating in the hot channel condition) loses sufficient hydrogen to just reach the β-δ phase interface. The cladding creep limitation is defined as operation until the hydrogen dissociation pressure (corresponding to the highest fuel specification hydrogen concentration (N_H) operating in the hot channel condition) causes the cladding to creep in excess of 1.0%. The nominal fuel hydrogen concentration was selected so that the fuel would reach the β-δ interface when the cladding has reached the creep limitation.

The analysis of cladding creep assumed a 9-mil cladding thickness and used a 1.4 factor of safety on the imposed load. The static NaK pressure was increased in order to counteract the higher hydrogen dissociation pressure caused by the increased N_H operating with the same hot channel fuel temperatures. The upper limit on the NaK static pressure was restricted to that value required to just strain the hottest location of the core vessel 0.2% during the reactor lifetime. The core vessel stress also had a 1.4 factor of safety. A core vessel thickness of 0.25 in. was estimated to be adequate to withstand the launch shock and vibration requirements. The N_H should not be increased above this corresponding upper limit NaK pressure because the reactivity loss due to increasing the vessel thickness, necessary to maintain the 0.2% creep limit, exceeds the reactivity gain due to the increased N_H.

The phase change design criterion is assumed to occur when the phase change occurs, as assumed with the Upgraded SNAP 8 design. The cladding creep limitation is estimated to occur when the cladding has undergone creep of 1.0%

instead of the 0.2% used in the Upgraded SNAP 8 criteria. The decrease in the conservatism of the cladding creep limitation relative to the Upgraded SNAP 8 design criteria is felt to be justified since the present SNAP 8 fuel materials research program will have experimental data which will decrease the uncertainty in the cladding creep behavior. This data should permit the use of this more realistic limitation.

Figure III-63 illustrates that cladding creep and phase change is one of the limits of the operating range of the 10,000-hr lifetime Advanced ZrH reactor. The design is cladding creep and phase change limited for high outlet coolant temperature operation to approximately 1440°F for a 2.0 Mwt reactor power and 10,000 hr of operation. Figures III-64 and III-65 show that cladding creep and phase change is not limiting for the Advanced ZrH operating 20,000 and 30,000 hr, respectively.

d. Beryllium Swelling (and Core Vessel Creep)

Two additional interacting design criteria are the core vessel creep and the beryllium swelling. The initial gap between the core vessel and the beryllium reflectors must be large enough to ensure clearance between the surfaces at all times. The gap cannot be excessively large due to the decrease in reflector worth. The design criteria restrict the core vessel lifetime creep to 0.2% due to NaK static pressure and limit the beryllium deflection to 50 mils due to isotropic swelling.

The beryllium swelling configuration for the design is assumed to correspond to tapered control drums which have:

1) the beryllium swelling along the entire drum length corresponding to the point of maximum temperature and maximum integrated fast neutron flux during the reactor lifetime,

2) the swelling occurring on the inside (near reactor) drum surface which causes the reflector to buckle assuming that the ends of the reflector are unrestrained, and

3) the beryllium deflection limited to 50 mils of the 80 mils available.

As shown by Figure III-63, the beryllium swelling limits the Advanced ZrH reactor performance for operation above approximately 3.6 Mwt for the 10,000 hr of core lifetime. The 10,000-hr reactor lifetime beryllium swelling limit can be increased by either recalculating the beryllium swelling for more realistic assumptions or by actively cooling the beryllium reflectors.

As shown by Figures III-64 and III-65, the beryllium swelling limitation of the Advanced ZrH reactor operating for 20,000 and 30,000 hr does not limit the system operation.

e. High Temperature Operation

The Advanced ZrH reactor is capable of operation for 10,000 hr at 2.0 Mwt with an outlet temperature of 1400°F while maintaining about the same design margins as at the design point of 3.0 Mwt, 1300°F (see Figure III-63). Some development work, particularly of components external to the core, would be required in order to realize this capability.

The effects of the higher temperature on the structural properties of the core materials have been included in the foregoing analyses, but some of the other potential problem areas have not been investigated in detail. Some of the areas which would require additional study prior to operation of the system at 1400°F are:

1) NaK mass transfer
2) Bearing performance
3) Gear performance
4) Control drum drive motors
5) Mercury corrosion (if coupled to Hg-Rankine system).

Preliminary calculations indicate that there are no problems which would limit the reactor operation to temperatures below 1400°F.

5. Reliability

The Advanced ZrH reactor design is based on the technology developed for the SNAP 8 Reference Design system. In addition, many of the components which have been developed may be used in the advanced system virtually unchanged. This state-of-the-art approach to the design should produce a system reliability equal to or greater than that of the reference design.

The large surplus of excess reactivity above that required for operation can provide a large amount of redundancy in the core and reflector systems. In addition, the knowledge gained in the testing of previous SNAP 8 systems will result in components for the advanced system which are more reliable than those used on the earlier systems.

6. Potential Design Improvements

This section briefly describes various second generation improvements which could be incorporated into SNAP reactor designs.

a. Thermal Bond

There is a large temperature drop across the hydrogen gap between the fuel rod and the ceramic. If a good thermal bond could be obtained between the fuel and the ceramic, the fuel heat generation rate, or reactor power, could be essentially doubled without increasing the maximum fuel temperatures. An improved thermal bond would then significantly improve the fuel swelling limitation. However, the cladding creep and phase change (i.e., hydrogen leak rate) limit would only improve by 10 to 20%. The only method for obtaining a good thermal bond appears to be by using a liquid and a significant research effort might be required to obtain a suitable material.

b. Fuel Coating

There is experimental evidence that the fuel rod reacts with the ceramic coating, producing a higher hydrogen leak rate out of the fuel elements. SNAP 8 is presently developing a metallic coating which is applied to the fuel and has demonstrated reductions in hydrogen leak rates of from a factor of 2 to 5 and more. When the metallic coating has demonstrated an in-pile stability, this technique will probably be incorporated in all SNAP 8 systems.

c. Improved Hydrogen Barrier

The SNAP 8 program is presently doing directed research to obtain a less permeable and more durable hydrogen barrier. The present SNAP 8 ceramic coating is a result of this effort. Improved hydrogen barriers have a potential of decreasing the hydrogen leak rate by as much as a factor of 5.

d. Yttrium Getter

Another method of decreasing the hydrogen loss from the reactor region is by the use of an yttrium getter incorporated into the fuel element cladding. A majority of the hydrogen which leaks out of the fuel element would be chemically retained by the yttrium. This technique, presently being investigated by the SNAP program, would reduce the system reactivity changes during the reactor lifetime.

e. Self-Shielded Poisons

The use of burnable poisons in a self-shielded configuration, for example, a cylindrical rod, could improve the burnout characteristics of the poison and provide a better compensation for the reactivity losses of the reactor. An analytical and developmental effort would be required.

f. Radial Power Shaping

The SNAP 8 reactors have a radial peak-to-average power distribution of about 1.3. The radial power distribution could be flattened by about 20% using, for example, a variable poison concentration radially within the core. Nearly 20% increase in reactor power level could be attained without increasing the fuel temperatures. The manufacturing and selective loading of fuel elements with varying amounts of poison would complicate the design.

g. Axial Power Shaping

The axial power distributions of the SNAP 8 cores are approximately chopped cosines with a normalized peak power of 1.4. By axially shaping the power, the fuel element temperatures can be made to approach a uniform value. A significant developmental effort would be required to determine an acceptable method, resulting in an approximate 20 to 30% increase in reactor power.

h. Segmented Fuel Rods

The reactor performance could be improved by use of isolated fuel segments within the fuel element. These fuel segments could have varying hydrogen, poison or uranium concentrations. The lower temperature sections of the fuel near the core inlet could be operated at a higher performance level so that the different sections would all be operating near their design limits. A developmental effort would be necessary to determine the required design changes. An estimated 10 to 20% increase in core performance could probably be achieved.

i. Insured Pitch/Diameter

The fuel elements could be wire-wrapped to guarantee the spacing between the fuel elements. This modification could increase the fuel and ceramic coating heat transfer coefficient, where elements presently might touch, to obtain an estimated 5 to 10% increase in reactor power with the fuel temperatures remaining constant. The improved heat transfer coefficient would be obtained at the expense of complicating the fuel manufacturing, inspection and reactor assembly.

j. Sliding Control Drums

A possible method of reducing the reactor envelope, with a corresponding reduction in shield weight, involves the use of sliding control drums. This control method is presently being investigated. It presently appears most attractive for static control systems, where the required life of sliding elements is short.

IV. SHIELDING

A. INTRODUCTION

1. Mission Criteria

The design of nuclear radiation shields for SNAP reactors is intimately dependent on the mission characteristics. At the present time, only the SNAP 10A nuclear system has been committed to a flight test program. Missions for the SNAP 8 Reference Design, the only other SNAP nuclear system committed to hardware delivery, have not been selected at this time. The other identifiable SNAP nuclear systems are design concepts without specific assigned applications; they are designed to accommodate any mission power requirement up to several thermal megawatts.

This lack of definitive mission specifications has been bypassed, to some extent, by the establishment of general shield design guidelines. These working criteria have been formulated by the AEC, NASA and Air Force, and are primarily oriented toward unmanned applications of the SNAP nuclear systems. Shield design efforts for unmanned spacecraft have not been restricted to these guidelines, however. Investigations have also been made to examine the effect on shield size and weight due to variations in the reference value of each system parameter. Similar parameter studies have been performed for manned SNAP missions. During the past year, a growing interest has been shown in manned applications, particularly for the SNAP 8 reactor. These manned system concepts cover a broader range of designs, from small cylindrical to large toroidal space stations, and for lunar surface operations.

2. Shield Configuration

The shadow shield concept has been used almost exclusively in the guideline studies on both unmanned and manned systems, since it results in a minimum weight shield. This type of shield design is basically in the form of a frustrum of a cone. The envelope of the shield is generally designed such that it protects the payload from all potential sources of radiation. In addition to the core itself, these sources include scattered radiation from the reflector assembly and other reactor components. It appears that the shadow shield concept is applicable to most space missions, so the bulk of this section is devoted to analysis and comparison of shadow shields.

The advent of interest in manrated shield designs has also stimulated the investigation of 4-pi shield concepts. This type of shield is a wrap-around design which affords radiation reduction in all, or nearly all, directions. The achievement of multidirectional attenuation of nuclear radiation may be desirable for several reasons. Among these are the increase in flexibility permitted for space rendezvous maneuvers and the attainment of radiation goals in situations where the shadow shield concept becomes untenable, such as in applications of a SNAP reactor mounted at the hub of a toroidal space station. The 4-pi shields generally carry a severe weight penalty with respect to shadow shields of comparable attenuation. Considerations pertinent to the design of 4-pi shielded systems are discussed in a separate subsection below.

3. Shield Design Methods

Analytical shielding studies for unmanned SNAP reactor applications have been made primarily with the use of simple ray-tracing techniques. Examples of the methods which have received wide use in these studies are GE's Shielding Program No. 14-0 and AI's MORTIMER and SCAR Programs. These digital computer codes have been very successful in treating the direct penetration and scattered components of the total dose at the payload. Gratifying agreement with Monte Carlo techniques and experimental results have been demonstrated.

Analytical shielding studies for manned SNAP reactor applications are considerably more complex. This is a result of the increased reduction required for gamma radiation and of the consequent increase in the importance of secondary gamma radiation. These secondary gamma sources are mainly due to neutron capture in the shield itself. The task of producing a minimum weight shield requires a fairly accurate evaluation of these secondary gamma sources. This, in turn, requires an accurate description of the neutron capture distribution in the shield. Neutron transport theory programs, such as DTF-II, are among the analytical techniques being used to supply this description. The need for more sophisticated methods of gamma-ray analyses is also being evaluated. Current efforts are being directed toward the feasibility of gamma-ray transport techniques. In addition, shield weight optimization techniques are being programmed for the digital computer. Subsequent to the conclusion of this study, the OPEX code, which uses the method of steepest descent, has been released and has been applied in a recent manrated shield design study.

TABLE IV-1

COMPARISON OF SNAP REACTOR SHIELD WEIGHTS

SNAP Reactor Designation	Reactor Power (kwt)	Outlet Temperature (°F)	Reactor Envelope		Core Diameter (in.)	Shield Weight* (lb)		Comments
			Shieldable Diameter (in.)	Shieldable Height (in.)		Unmanned	Manned	
10A	40	1000	18.0	14.75	8.8	330	3460†	See text for shield design criteria
I 10A/2	100	1200	18.4	14.75	8.8	356	3810†	
U 10A/2	100	1300	19.3	14.75	8.8	374	3980†	
10B	100	1300	14.6	17.75	8.8	254	3410	Sliding reflector concept
Advanced 10B	325	1300	14.6	18.9	8.8	357	4070	Sliding reflector concept
Reference 8	600	1300	27.6	22	9.4	945	6960	Drums-out shield protection
Manrated 8	600	1300	19	23.5	9.4	560	5380	Drums-out shield protection
Upgraded 8	1200	1300	21	30.0	10	735	6960	Drums-out shield protection
Advanced ZrH_x	3000	1400	24	33.0	12	1100	9200	Drums-out shield protection

*See text for dose criteria
†Based on standard shieldable envelope; see text for transient considerations

B. UNMANNED APPLICATIONS

Table IV-1 presents a comparison of unmanned shield weights for all SNAP reactor design concepts under discussion in this report. These results are based on the following set of shield design criteria:

dose plane diameter	5 ft
dose plane separation distance	30 ft
fast neutron dose	10^{11} nvt/yr
gamma ray dose	10^{6} rads (C)/yr

These ground rules were arbitrarily selected to provide a consistent basis for this comparison. They are representative, however, of typical shielding criteria guidelines which have been established for the SNAP nuclear systems.

The calculations of neutron shield thickness assume a fast neutron attenuation coefficient of 0.150 cm^{-1}. This value is based on recent experimental investigations* of lithium hydride.

For this tabular comparison, a constant value of 25% is assumed for the ratio of shield structure to lithium hydride weight. In practice, this weight ratio is strongly dependent on such mission characteristics as vehicle launch loads and micrometeoroid protection goals.

A summary of weight distributions in the shield designs presently associated with SNAP 10A, Interim 10A/2 and SNAP 8 Reference Design is given in Table IV-2. The higher structural weight fraction in the SNAP 8 shield is due

TABLE IV-2
SHIELD AND STRUCTURAL WEIGHTS

	10A	I10A/2	8RD
LiH wt (lb)	175	289	1162
Total Structure (lb)	42	67	374
Ratio, Structure/LiH	0.24	0.23	0.32

*S. G. Wogulis and K. L. Rooney, "Experimental Studies of Neutron Attenuation in Natural Lithium Hydride Shields," NAA-SR-9264 (February 12, 1964) to higher loadings, and more stringent specifications on shock, vibration and meteoroid protection. SNAP 10A and Interim 10A/2 have specifications which are similar in these factors, and the structural fraction of their shields are consequently similar.

A notable exception to the consistency of the tabulated unmanned shield weights is the treatment of shieldable reactor envelope. Reference is made to the shieldable reactor radius. For reflector drum control, this radius is invariably defined by the maximum radial distance from the reactor axis to any point on the drum surface. Thus drum position must be considered. The shieldable radius for all reflector drum control SNAP 10A and SNAP 10A/2 designs corresponds to the drum position at the start of passive control.* The shieldable radius for the SNAP 8 reactor versions (including the Advanced ZrH reactor, for convenience of grouping) corresponds to the full-out drum position. The SNAP 10A/2 design criteria is based on the argument that while criticality can be achieved with a drum stuck in the full-out position operation at rated conditions could be maintained for only a few days. Consequently shielding protection should not be extended to cover this no-go condition. The SNAP 8 reactor designs, on the other hand, contain sufficient reactivity to compensate for various failures such as a stuck control drum. The reference SNAP 8 reactor design, for example, could operate at rated power and outlet coolant temperature for the design life of 10,000 hr with one control drum stuck in the full-out position. Since these systems can achieve their objectives under such a condition, the shields are designed to protect against a drum in this position. The shield weight penalty for this SNAP 8 design criteria can be estimated by selecting drum position at hot criticality as an alternate basis for definition of shieldable reactor diameter. For 3-drum and 6-drum control of the SNAP 8 Reference Design, the corresponding weight penalties are 110 and 150 lb,† respectively, for the Table IV-1 design conditions.

*Valid only for unmanned applications. For manned applications, small scattering sources are important, and the drum position at first attainment of power must be considered.

†For unmanned system shields

Figures IV-1 through IV-4 represent the effects on shield size and weight due to variations in the following system parameters:

System Parameter	Parameter Range
Reactor power level	30 to 900 kwt
Dose plane diameter	5 to 100 ft
Reactor-dose plane separation	10 to 200 ft
Dose plane radiation levels	
fast neutron dose	10^{10} to 10^{14} nvt/10^4 hr
gamma ray dose	10^5 to 10^8 rads(C)/10^4 hr
Reactor envelope	
shieldable diameter	12 to 30 in.
shieldable height	15 to 40 in.

Figures IV-1, IV-2 and IV-3 present shield weight scaling factors corresponding to each system parameter.* Figure IV-4, parts A and B show similar scaling factors for maximum shield diameter and total shield thickness. The total scaling factor in each case is simply the product of each of these individual scaling factors. These scaling curves are intended primarily to illustrate trends, and the relative importance of each system parameter on shield size and weight. Graphical calculations based on these curves are accurate within about 20%.

C. MANNED APPLICATIONS

Table IV-1 also presents a comparison of manned shield weights for all SNAP reactors under consideration in this report. These results are based on the following set of shield design criteria:

dose plane diameter	40 ft
dose plane separation distance	100 ft
biological dose (RBE= 10, for neutrons)	1 mrem/hr
"gallery" thickness	24 in.

*It should be noted that logarithmic scales are used in Figure IV-1 for the system parameters of dose plane diameter, separation distance, and power, whereas linear scales are used in Figure IV-2 for the same parameters. This accounts for the apparent reverse curvatures noted in these figures.

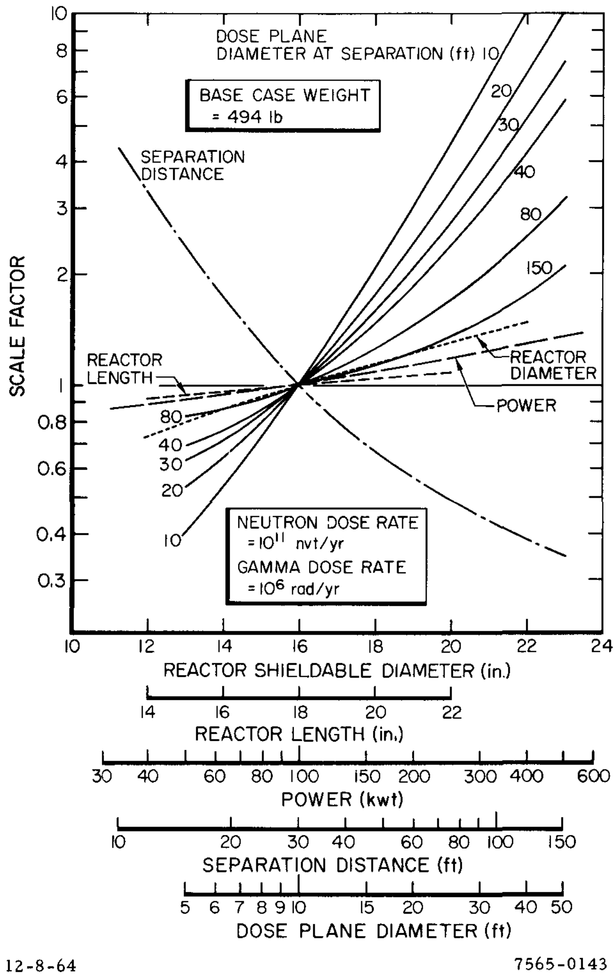


Figure IV-1.
Shield Weight Scaling Factors,
Unmanned Systems

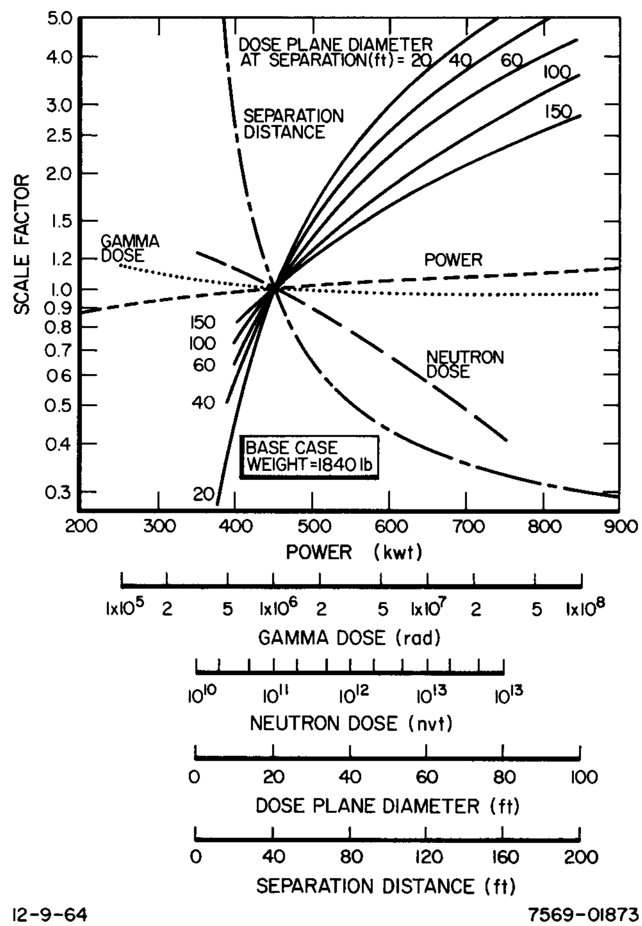
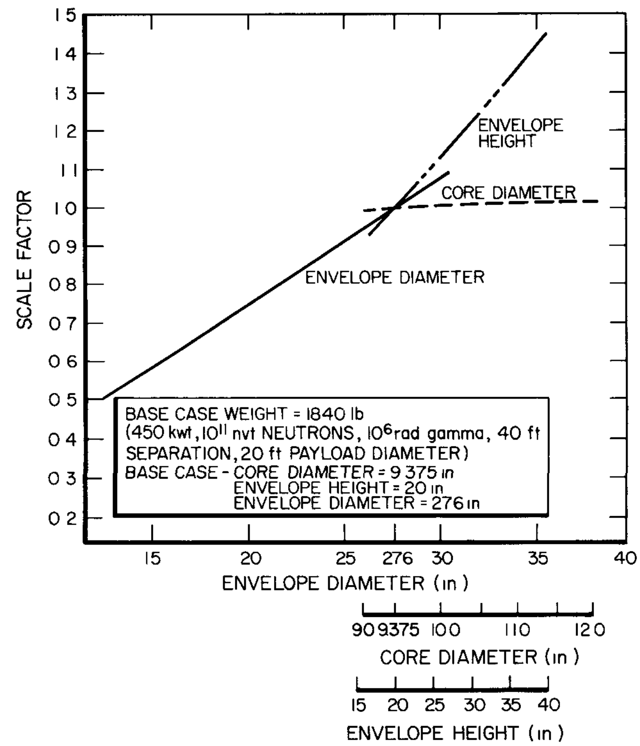


Figure IV-2.
Shield Weight Scaling Factors,
Unmanned Systems

Figure IV-3.

Shield Weight Scaling Factors, Unmanned Systems



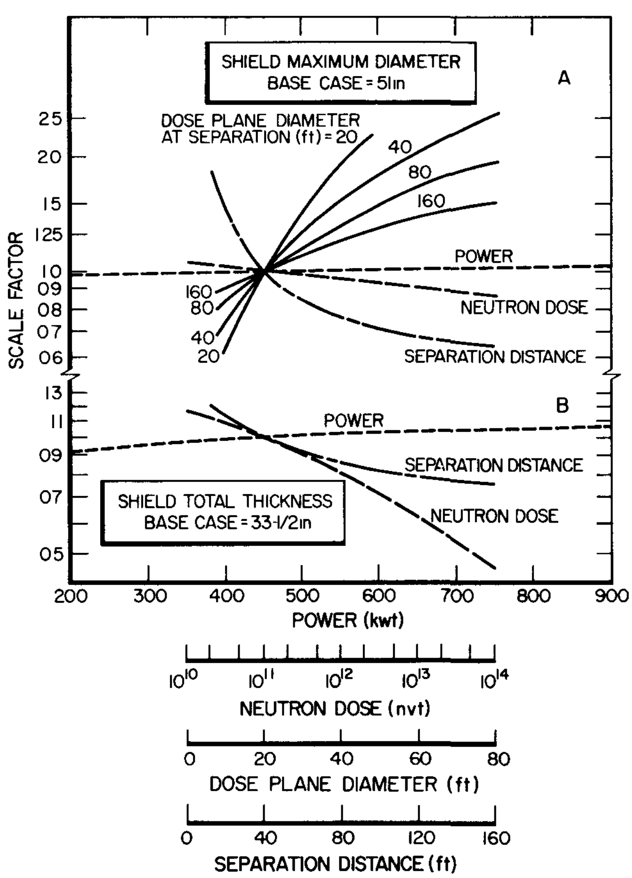


Figure IV-4.

Parametric Shield Dimension Data, Unmanned Systems

Once again these are arbitrary criteria, selected to provide a consistent basis for intercomparison. Many of the systems are not even being considered for manned missions, but shield weights are included for the sake of completeness.

The current conceptual designs of SNAP nuclear systems for manned application have the following notable features:

1) reduction in reflector control drum diameter to reduce shieldable reactor envelope and hence shield weight.

2) split shield design with "gallery" between shield halves accommodating power conversion devices which contain radioactive primary coolant.

3) primary coolant pipes coiling through top half of shield.

4) no shield penetration by control drum drive shafts.

The purpose of the gallery is to utilize the bottom shield half for double shielding duty. With such an arrangement, this section of the shield provides attenuation for gamma rays from the radioactive primary coolant (NaK) in the power conversion components as well as for the radiation emerging from the top shield section. Shield weight savings up to 33% have been found for typical shield designs using this divided configuration.

The material arrangement for split shield designs is based on the following considerations:

1) top neutron shield must be thick enough to prevent significant activation of the secondary loop fluid (for Hg-Rankine power conversion system)

2) top neutron shield must be thick enough to prevent a disproportionate amount of secondary gamma production in lower gamma shield

3) top gamma shield must not be so thick that emergent secondary gamma rays produced therein are significantly higher than transmitted gamma radiation incident on this shield

4) top gamma shield must not be so thick that total emergent gamma radiation is significantly lower than activated NaK radiation incident on lower gamma shield.

The analysis for manrated SNAP shields has assumed the selection of natural lithium hydride and depleted uranium (0.22 wt% 235) as the shield materials.

The use of Li^6H instead of natural LiH is also being considered because it offers an apparent weight reduction. The use of lead was not considered in the study reported herein, mainly due to problems associated with its low melting temperature. Investigations are being made on the feasibility of liquid lead containment, since subsequent analyses have indicated that this material can result in reduced shield weight because of its lower secondary gamma production. The use of two other promising gamma shield materials, tungsten and tungsten-aggregate/lithium hydride, was the subject of a later analytical investigation. This analyses was conducted using representative manned mission characteristics and a shield weight optimization procedure based on the method of Lagrangian Multipliers. The results of this subsequent study showed that depleted uranium shields weighed less than those in which tungsten or tungsten-aggregate/LiH were used.

The manned mission shield weights quoted in Table IV-1 should be considered as preliminary estimates. More rigorous optimization procedures and more advanced analysis of the secondary gamma sources are required before firm shield weight estimates can be established. Optimization procedures involving Lagrangian multipliers, the method of steepest descent, and dynamic programming techniques are currently under development. A dual approach is being made for the attainment of an accurate specification of secondary radiation levels. Concurrent with a scheduled experimental program at the Shield Test and Irradiation Reactor facility, improvements are being made in the analytical shielding capabilities of the available calculational tools. The shield test program is oriented toward simple, clean experiments to provide a comparison with analytical predictions. Alternate slabs in varying thicknesses of lithium hydride and heavy element materials will be investigated using a collimated detector. Extension of the accuracy in the analytical techniques is being made by improvements in the convergence technique and cross section library for the DTF code (Fortran version of the Los Alamos DTK neutron transport program). The high energy group structure is being expanded and anisotropic scattering provisions are being incorporated in the DTF library for all elements.

Figures IV-5, IV-6 and IV-7 represent the effects on manned shield weight due to variations in the following system parameters: .

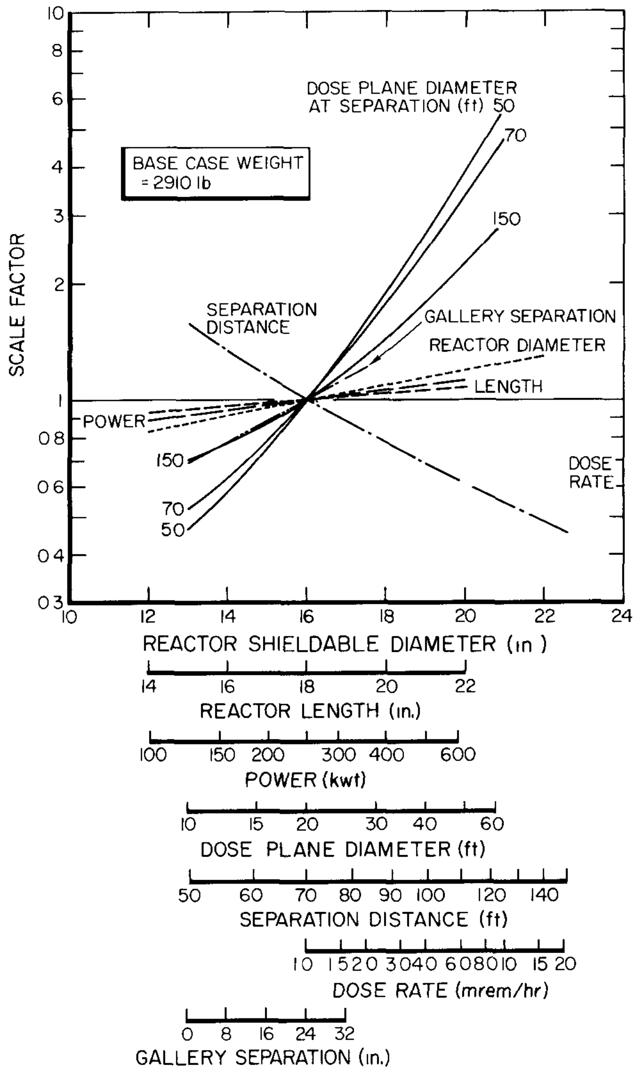


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Figure IV-5. Shield Weight Scaling Factors, Manned Systems

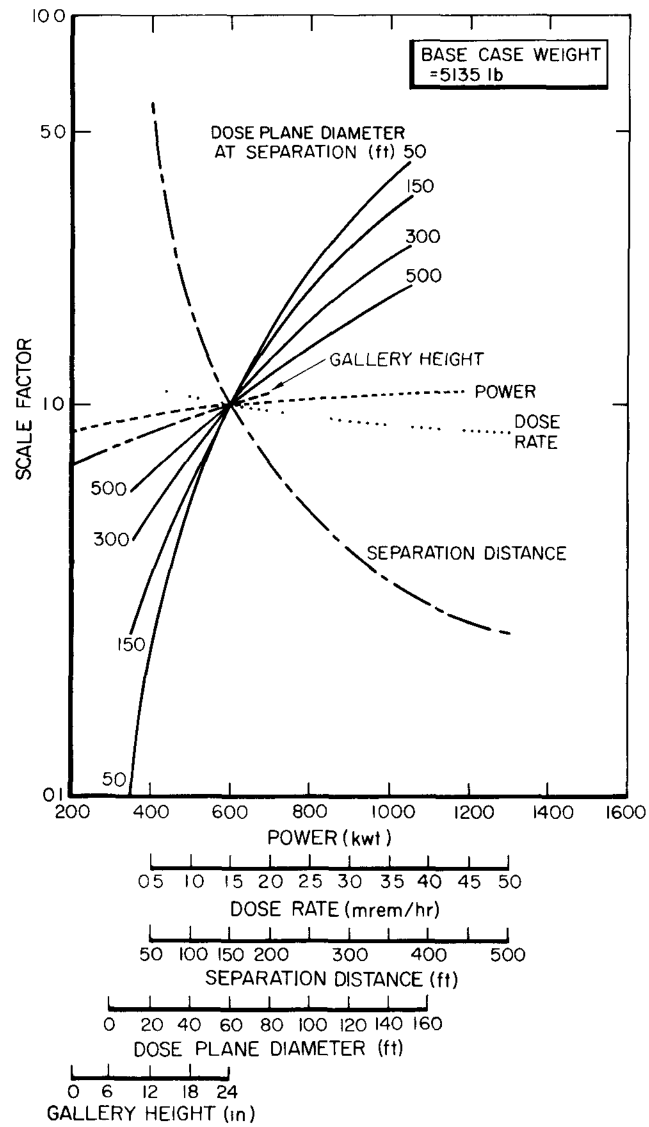

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Figure IV-6. Shield Weight Scaling Factors, Manned Systems

System Parameter	Parameter Range
Reactor power level	100 to 1200 kwt
Dose plane diameter	10 to 150 ft
Reactor-dose plane separation	50 to 500 ft
Dose plane radiation level	0.5 to 20.0 mrem/hr
Gallery height	0 to 24 in.
Reactor core diameter	9 to 12 in.
Reactor envelope	
shieldable diameter	12 to 28 in.
shieldable height	14 to 40 in.

The shield weight scaling factors presented in these figures are again intended to illustrate trends, and the relative importance of each system parameter on shield weight.

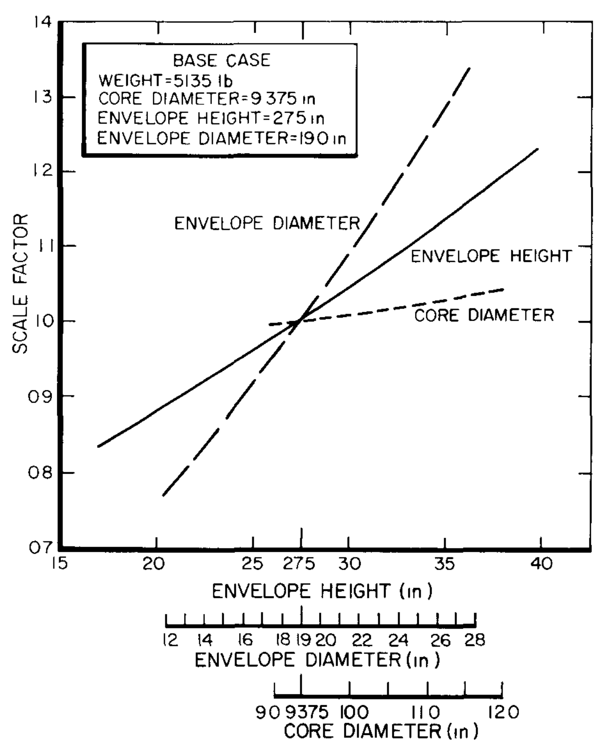

Figure IV-7. Shield Weight Scaling Factors, Manned Systems

D. SPECIAL CONSIDERATIONS FOR 4-Pi SHIELDING

In general, most SNAP shielding requirements can be met by a shadow shield which provides a fairly narrow shielded cone, and which does not encompass the reactor on its sides or top. In some cases, mission geometry may prohibit this solution. In other cases, it may prove desirable to develop information on the benefits and disadvantages of system design tradeoffs in which directional shielding properties are a parameter. Consequently, substantial interest exists in determining preliminary designs for large-angle shields, and in demonstrating a capability for performing valid design optimizations.

1. Requirements

Requirements for 4-pi or partial wrap-around shields may arise from several sources. Without wrap-around shielding, rendezvous with a space station requires that the envelope of approach paths for the shuttle craft coincide, for the most part, with the narrow shielded cone. In those cases where the approach envelope is not more stringently limited by other considerations, inclusion of a large-angle shield permits more flexibility in the docking maneuver and a higher radiation dose safety factor.

For certain space station geometries, i.e., a yaw spinning dumbbell configuration where despin for rendezvous is otherwise prohibited, 4-pi shielding is almost certainly the only alternative to temporary reactor shutdown.

Rendezvous with a spinning toroidal or Y-type space station imposes no additional shield requirements if the reactor is located at the hub, since approach will probably be limited to a particular hemisphere by gross mechanical considerations. Station operational considerations will already have imposed a shield angle of somewhat greater than 2 pi in this case. For Y-type stations where the reactor is located at the end of one leg, operational considerations do permit a shadow shielding approach. Additional shield is then required for rendezvous.

Another category where wrap-around shielding is required is the lunar* or planetary surface installation, where backscatter and territorial accessibility considerations prohibit a shadow shield approach.

*See AI-8576, "Study of SNAP Power Systems in the Lunar Environment," work performed under NASA Contract NAS-3-2530.

2. Reactor Design Interactions

Large-angle and wrap-around shields produce numerous system interactions. Besides those mission interactions implied in the preceding section, there are also a number of important interactions between the shield and the reactor design and performance.

One of the most significant interactions is the change in thermal environment imposed by closely surrounding the reactor/reflector by the shield. This can create heat balance and auxiliary cooling problems in both reactor (see Section III-F-3) and the shield. Preliminary calculations in a few specific cases have shown that these problems are subject to fairly straightforward treatment. Material substitution (e.g. BeO for metallic Be) may be desirable in some cases.

A second important area of interaction is the influence of reactor envelope dimensions on total system weight. For a fixed attenuation factor, adding a small increment to the required cavity dimension results in the addition of a similar radial increment to the outside of the entire shield. With manrated systems having large attenuations, this may constitute a large weight penalty.

A close fitting wrap-around shield also has an influence on the neutron economy of the reactor, in that it constitutes an additional neutron reflecting medium. The principal effects are to increase the overall system reactivity, and to reduce the amount of reactivity variation available in neutron leakage control systems of the SNAP type. Ground testing of SNAP systems has been successfully conducted in similar reflecting environments (concrete-walled pits), and preliminary quantitative evaluations of the "vault effect" have been performed experimentally. Additional nuclear experiments will be conducted to verify and improve present analytical treatments of this effect.

3. 4-Pi Shield Design

In general, the design of a 4-pi and other large-angle shields is subject to straightforward application of the same techniques and analytical tools used in shadow shield design. In some cases (e.g. anisotropic attenuation patterns) second order effects may become important, and must be recognized and dealt with.

When a requirement for 4-pi shielding exists, it is usually in conjunction with a manrated system where large shield attenuations are necessary. This factor, in combination with the physically wider extent of shield, makes 4-pi shields extremely heavy in comparison to unmanned shadow shields. With a 4-pi shield constituting a major fraction of the entire system weight, small variations in mission and system design can become very important in determining overall system weight. The importance of mission and system details is indeed so great that it is not presently possible to give general parametric analyses of 4-pi shield designs which are both meaningful and valid.

Conceptual designs of 4-pi shields have been performed for a number of specific application conditions. Shielding of a SNAP 8 lunar base reactor, both by burial in the lunar surface and by means of an earth fabricated 4-pi shield, has been analyzed.* In the latter case, shield weights in excess of 15,000 lb were calculated for some sets of mission and system conditions.

Modified 4-pi shielding of a reactor installed in a Y-type space station has also been studied. This reactor/shield design was intended for installation at the outer end of one of the station "legs," and was illustrated in Figure III-54. The shield provides an anisotropic attenuation pattern such that, with the reactor operating at 600 kw (thermal), the dose rate in the primary occupancy direction is 2 mrem/hr. This dose rate prevails over a 25-ft diameter reference plane located 40 ft from the reactor. Design criteria also require that dose rates in all directions outside the primary occupancy cone be simultaneously limited to a maximum of 100 rem/hr at a distance of 100 ft.

The total weight of the shield is 35,750 lb, of which approximately 26% is allocated to structural functions. The design is preliminary in nature, and has not been demonstrated to be optimum for the system criteria specified.

It is interesting to note the second order effect in the primary occupancy direction imposed by the presence of that portion of the shield which surrounds the reactor on top and sides. This "cap" has a total weight of only 5500 lb. Since it is a source of secondary and scattered radiation, however, the diameter of the main portion of the shield must be increased substantially to maintain the required primary occupancy dose rate. A shadow shield having equivalent attenuation in the primary occupancy direction would weigh about 14,000 lb, less than 40% of the weight of the 4-pi shield described, and a total weight savings of over 21,000 lb. The "second order effect," in this example, amounts to over 16,000 lb. This observation emphasizes the importance of carefully considering system and mission details in the design of 4-pi shielding.

*op. cit.

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