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REPORT

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SNAP PROGRAM
M-3679 (38th Ed.)

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FINAL SNAPSHOT SAFEGUARDS

REPORT

ADDENDUM I

(Title Unclassified)

Technical Editor

R. S. HART

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1. INTRODUCTION

The primary purpose of this document is to report and evaluate information and/or analysis pertinent to the safety of the SNAPSHOT Program which has become available since approximately May 1, 1964. May 1 represents the "cut-off" date that was imposed upon incoming data used in the preparation of the

Final SNAPSHOT Safeguards Report.¹ The information reported herein has partially originated from continuing programs of a related nature such as the Aerospace Safety Program and also from progress in the SNAPSHOT Program itself. In addition, certain new analyses also are presented.

2. SUMMARY AND CONCLUSIONS

The major aspects or phases of the SNAPSHOT Program safety evaluation which are encompassed by this report are summarized below.

2.1 Nuclear Analysis

2.1.1 Nominal Reactor Power

Recent analysis, which incorporates test data from non-nuclear systems, indicates that the nominal thermal power of the reactor will be about 39.5 kw instead of 33.5 kw as assumed in the safeguards analysis. This increase will result in a corresponding increase (18%) of the fission product inventory over the design life of the system. However, the overall effect of the safeguards analysis is negligible and no quantitative modification is warranted.

2.1.2 Long-Term Reactor Power History

A reanalysis has been made of the potential long-term operating history of the reactor in the event of reflector ejection failure. A revised temperature coefficient, based on certain material changes and the FS-1 test results, was utilized. Generally, these revised long-term reactor power levels are equal to or lower than the assumed history in the safeguards report. Consequently, the fission product inventories and/or dose rates used in the previous analysis are further confirmed as conservative values

2.1.3 Reactivity Values

A revised table of the SNAP 10A reactor reactivity status with various normal and abnormal environmental conditions has been prepared. The most significant effect of this revision is the general lessening of the reactivity worth of external sources of neutron reflection (i.e., water, personnel, etc.) Thus, increased margins of criticality safety have been gained in most areas pertinent to ground handling of the system.

2.1.4 Accidental Nuclear Transients Due to Water Immersion

On the basis of the SNAPTRAN-3 test and reductions of internal core hydrogen in the flight system, a bare SNAP 10A reactor is now calculated as subcritical when immersed in sea water (no internal water). However, for ocean impacts which could result from a launch vehicle failure, tests conducted under the Aerospace Safety Program demonstrated that sufficient damage to the core vessel and/or NaK lines would occur to permit rapid replacement of the NaK by sea water. Thus, the occurrence of self-destructive nuclear transient will remain highly probable as assumed in the safeguards analysis.

2.2 Radiological Analysis

2.2.1 Fission Product Release Fraction

Radiological sampling during the SNAP-TRAN-3 test has indicated a very small release of fission products from the fuel. No more than 5% of the rare gasses and probably considerably less than this amount of iodine, was released. As contrasted to the 100% release of these elements which was conservatively assumed in the SNAPSHOT Safeguards Report, such a reduction would have a significant effect on the potential downwind airborne doses from a nuclear excursion. However, since the potential hazards from this exposure mode were evaluated as either acceptable or not dominant in the previous analysis, no corrections have been attempted.

2.2.2 Short Term Reactor Operation

Under certain conditions, limited reactor operation terminated by ground command shutdown may be of interest. Data are presented which enables correlation between reactor operating time, orbital lifetime, and fission product inventory at the time of reentry.

2.2.3 Reentry Hazards — Localized Fallout

The probability of highly localized ground deposition of ablated SNAP 10A fuel fragments was examined. It is concluded that such deposition represents a smaller potential radiological hazard than the intact reentry survival of complete fuel elements. Since the latter exposure mode was postulated for analysis in the safeguards report, the conservatism of the selection is confirmed.

2.2.4 Reentry Hazards — Man-Rem

A population integrated dose ("man-rem") was derived for the possible reentry deposition of any fission product inventory which could be associated with the SNAP 10A reactor. Even with the assumption of ideal particle deposition of the maximum inventory, the probability of excessive man-rem doses is extremely small.

2.3 Probability Analysis

2.3.1 Self-Welding of Reflector Components

A reevaluation of the possibility of long term self-welding of the reflector which might prevent ejection and thereby enhance reentry survival of the reactor assembly has been made. The conclusions are that no reasonable argument can be made for such an event under any operating condition for any orbital lifetime (≤ 3800 yr) under consideration. Therefore, the 0.99 probability of reactor reentry disassembly for orbit lifetimes less than 100 yr, which was postulated in the previous safety analysis, should probably be extended to at least 3800 yr. However, such a modification would have a small effect on the total potential hazards as evaluated previously and, therefore, no quantitative corrections have been attempted.

2.3.2 Reactor Shutdown

A reevaluation of the probability of reactor shutdown through both automatic on-board devices (EABRD and TABRD) and ground command has resulted in the following best estimates:

- 1) If no attempt at command shutdown is ever made, a maximum reactor shutdown probability between 0.5 and 0.7 exists after about 5 yr of operation (0.8 at 3 yr was assumed in the safeguards report).

- 2) Utilization of the ground command shutdown option can considerably enhance the total shutdown probability at operating times up to about 1 yr (e.g., about 0.95 at 90 days, 0.87 at 1 yr). (No quantitative credit was taken for the ground command option after 5 days operation in the safeguards report.)

Therefore, while an increase in the probability of the no reactor shutdown case used in the previous safety analysis is indicated (i.e., from 0.2 to 0.3 - 0.5), an attempt at ground command shutdown by one year after startup would reduce the no shutdown probability to less than that previously assumed.

2.4 Miscellaneous

2.4.1 Launch Pad Procedures

Notation is made of some minor changes in launch pad procedures. These changes have no influence on the safety analysis.

2.4.2 SNAPSHOT Responsibilities

Notations are made of certain responsibility changes resulting primarily from the transfer of the Pacific Missile Range to the U.S. Air Force. These changes have no influence on the safety analysis.

With respect to the analysis and conclusions presented in the Final SNAPSHOT Safeguards Report, an overall evaluation of the information contained in the addendum results in the following conclusions:

- 1) In the area of potential nuclear accidents and the evaluation of any resultant radiological consequences, recent experimental and analytical data have generally confirmed the conservatism of the assumptions used in the SNAPSHOT Safety analysis. That is, the potential radiological hazards associated with the program are, in reality, considerably less than those postulated in the safeguards report.¹
- 2) Specifically, in the area of post-reentry hazards, the following points are significant:
 - a) The overall probability of reactor shutdown can be considerably enhanced by including the possibility of the ground command shutdown mode for reactor operating periods up to at least 1 yr.
 - b) In the event of complete shutdown failure, the power degradation of the reactor following NaK stagnation or loss is considerably faster than believed previously.
 - c) The probability of atmospheric reentry disassembly of the reactor core vessel remains high throughout the orbital lifetimes considered.
 - d) The degree of environmental contamination that may occur as a result of an accidental nuclear excursion is considerably less than previously postulated.

3. NUCLEAR ANALYSIS

3.1 Nominal Reactor Power

The nominal SNAP 10A reactor power, 33.5 kwt, quoted in the Final Safeguards Report¹ was an estimate based on calculated system heat losses. Continued analytical revisions, in combination with the experimental data from FSM-1 and FSM-4, have resulted in a reevaluated nominal reactor power of 39.5 kwt.² This power increase (18% from that assumed previously) only applies during the one-year design life of the system. The significance of the resulting increase in the fission product inventory is very small. Only for those reentry accidents which result in an excursion immediately following one year or less of reactor operation would there be any detectable change in any of the related injury probabilities calculated in Reference 1. Since for such orbital lifetimes, the possibility of direct, post-impact exposure to the fuel elements of a disassembled reactor is the dominant potential hazard, the overall significance of this power increase on the injury probabilities is negligible.

3.2 Long Term Reactor Power History

The long term reactor power history presented in the Final SNAPSHOT Safeguards Report was derived to permit an evaluation of the potential post-reentry hazards in event of a shutdown failure. Based on degradation of the NaK pump and the emissivity coating of the radiator-converter, coupled with reactivity changes, gradual power decay occurred over a period of tens of years. When a flowrate equal to 2% of full flow was reached, NaK coolant stagnation was assumed to occur, resulting in an order of magnitude drop in power. Following stagnation, gradual decrease in power continued, caused now entirely by reactivity losses. From the possible range of flow and emissivity degradation rates, two specific values were chosen (i. e., 5%/yr exponential flow and emissivity degradation) which resulted in the most conservative power history from a safety viewpoint.

However, the overall temperature coefficient has been reevaluated based on a material design change (stainless steel grid plates) and the results of the FS-1 test. Utilizing this information, new series of power histories were calculated over the degradation ranges. The results of these calculations are presented in Reference 3. For comparison purposes, the two limiting cases of this analysis (10%/yr pump and emissivity degradation and 5%/yr pump and emissivity degradation) are shown in Figure 1 along with the power history that was used in the Safeguards Report. As can be seen, the recalculated power is always equal to or lower than the previous value, except during the first year of operation. Since the fission product inventory at the time of reentry directly reflects the reactor power history, the potential injury probabilities due to possible direct exposure to fuel material have been conservatively evaluated in the Safeguards Report.

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Figure 1. Long-Term Reactor Power (Shutdown Failure)

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3.3 Reactivity Values

Appendix C, Tables C-2, C-3, and C-4 of the Final SNAPSHOT Safeguards Report presented SNAP 10A reactivity values under various normal and abnormal conditions during shipment, assembly, and reentry.

Since the creation of these tables three changes have occurred; (a) the hydrogen content of the fuel elements has been lowered, (b) later SCA-4B experiments have indicated that water reflected and water reflected and flooded SNAP 10A type reactors are less reactive than earlier experiments indicated,⁴ and (c) the reactivity worth of NaK inside the core has been reevaluated as $+\$0.25 \pm 0.05$ from FS-1 data.

The results of these changes have been incorporated into Tables 1, 2, and 3 as excerpted from Reference 3. In almost all cases these changes result in a significant decrease of reactivity. For example, the effectiveness of the void filler blocks and the shipping sleeve against external sources of reflection has been substantially increased.

3.4 Accidental Nuclear Transients Due to Water Immersion

The results of the analysis of the SNAP-TRAN-3 destructive test, along with current reactivity data on the SNAP 10A flight reactor, provide a basis for a reevaluation of the SNAP 10A water immersion accident.

The SNAPTRAN-3 reactor core, submerged in water, was subjected to a sudden (<10 msec) increase in reactivity to approximately $\$3.60$ above critical. The resulting power transient caused the core vessel to blow apart and the core to be disassembled.⁵ The disassembly of the core has been attributed to the buildup of hydrogen pressure within the system as a result of the disintegration of the fuel-moderator structure. Disintegration of the fuel-moderator structure has been deduced to occur at a temperature of approximately 1900°F .

The latest reactivity data on the water-immersed SNAP 10A flight reactor bare core can be summarized as follows:

TABLE 1
SNAP 10A REACTIVITY VALUES UNDER VARIOUS NORMAL AND ABNORMAL
CONDITIONS DURING ASSEMBLY AND SHIPMENT

Condition of Reactor		Reactivity (\$)					
On Right Side	On Left Side	In Air	Single Human Standing	Single Human Wrap	Maximum Human Wrap	Infinite Water Reflection	
						NaK in Core	H ₂ O in Core
Sleeve	Sleeve	-18.70	-18.40	-16.70	-12.70	-8.00 ± 1.20	-3.00 ± 0.50
Sleeve	Bare	-19.80	-19.50	-17.80	- 6.30	-3.35 ± 2.00	$+1.65 \pm 2.00$
			-18.70*	-12.80*			
Sleeve	Reflector with void filler blocks	-12.00	-11.70	-10.00	- 7.50	-4.60 ± 2.00	$+0.40 \pm 2.50$
Reflector with void filler blocks	Reflector with void filler blocks	-5.00 ± 1.00	- 4.70	- 4.00	- 2.00	-1.25 ± 0.50	$+3.75 \pm 1.50$
Reflector with void filler blocks	Bare	-13.10	-12.80	-12.10	- 1.10	$+0.85 \pm 1.50$	$+5.85 \pm 2.00$
			-12.00*	- 5.50*			
Bare	Bare	-21.00	-19.90	-14.00	-1.70 ± 1.00	$+1.30 \pm 0.80$	$+6.30 \pm 1.50$

*Human on the bare side

TABLE 2
REACTIVITY VALUES FOR
BERYLLIUM-REFLECTED SNAP 10A
NO VOID FILLER BLOCKS

Beryllium Reflected Reactor – All Drums Out, No Void Filler Blocks	\$-5.60
Single Human Body Reflection – Standing Adjacent	-5.25
Single Human Body Reflection – Wrap Around	-3.67 ± 1.00
Two Human Body Reflection – Wrap Around	-1.72 ± 1.00
Maximum Human Body Reflection	-0.73 ± 1.50
Water in Core – No Water Reflection	-0.60 ± 0.80
Infinite Water Reflection – No Water in Core	+1.87 ± 2.00
Infinite Water Reflection and Water in Core	+6.87 ± 3.00
Agena Nose Shroud and Cap in Place (launch configuration)	-5.55
Beryllium Reflected Reactor – Two Drums In, No Void Filler Blocks	-1.30
Single Human Body Reflection – Standing Adjacent	-0.95
Single Human Body Reflection – Wrap Around	+0.63 ± 0.50
Maximum Human Body Reflection	+2.81 ± 1.50
Infinite Water Reflection – No Water in Core	+2.23 ± 2.00
Infinite Water Reflection and Water in Core	+7.23 ± 2.00
Agena Nose Shroud and Cap in Place (launch configuration)	-1.25

TABLE 3
REACTIVITY VALUES FOR BARE SNAP 10A
REACTORS UNDER SEVERAL ABNORMAL
ENVIRONMENTAL CONDITIONS

Bare Core, Half-Plane Sand, 0% Water	\$-15.00 ± 2.00
Half-plane water	-10.00 ± 2.00
Half-plane sand, 30% H ₂ O	-10.50 ± 2.00
Half-plane concrete	-11.30 ± 2.00
Infinite Sand Reflection, Mois- ture Content for Criticality	26%
Bare Reactor, Water in Core, No Water Reflector	-16.00 ± 1.00

1) The excess reactivity of the water-immersed bare core, with NaK in the core is \$1.30.

2) Without NaK in the core, the excess reactivity is approximately \$1.0.

3) With water, both inside and outside the core, the excess reactivity is about \$6.

Recent studies indicate that the reactivity of the water-reflected SNAP 10A core is reduced by approximately \$2 when fresh water is replaced with salt water.⁶

With reference to the SNAP 10A water-immersion accidents, the above results indicate:

1) The minimum reactor excursion that will result in core disassembly, in the manner of the SNAPTRAN-3 test, is one in which the peak fuel temperature reaches a value of 1900°F.

2) In the event of water immersion in fresh water, the reactor will be between \$1 and \$1.3 supercritical with no water in the core, and will be \$6 supercritical with the core completely flooded.

3) In the case of water immersion in ocean water, the reactor will be subcritical with no water in the core and will be between \$4 and \$6 supercritical with the core completely flooded.

It is apparent that in the event of either an ocean water or a fresh water accident in which water rapidly fills the core, the consequent power excursion would cause the fuel-moderator temperature to exceed 1900°F. Rapidly, in this case, means in a shorter time than the time required to transfer heat to the water. In the event sufficient heat is transferred to boil off the incoming water, the power excursion could be terminated before the fuel-moderator temperature reached 1900°F. Subsequent operation would then depend on the dynamic stability characteristics of the system.

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It is the purpose of the following discussion to show that in the event of a water impact accident, water enters the core at a rate that is large compared to the maximum rate at which the water could be caused to vaporize.

When the center fuel-moderator temperature is 1900°F, the average temperature is 1300°F. At an average fuel-moderator temperature of 1300°F, the rate of heat transfer to the coolant is 123 (h) Btu/ft-sec, where h is the height of water in the coolant channel in feet. This estimate is based on a heat transfer coefficient (UA) of 0.095 Btu/°F-sec for the core and corresponds to film boiling heat transfer. Since approximately 970 Btu is required to boil 1 lb of water, a heat rate of 123 Btu/sec corresponds to a boiling rate of almost 0.13 lb/sec. A mass flowrate of 0.13 lb/sec corresponds to a flowrate in the coolant channels of 0.05 ft/sec. Thus, with an average fuel-moderator temperature of 1300°F, water flowing through the core at 0.05 ft/sec would boil provided water extended the entire length of the fuel-moderator elements.

In the NaK-Water Reactor Test,⁷ conducted as part of the Aerospace Safety Mechanical and Thermochemical Test Series, a mockup of the SNAP 10A reactor filled with NaK was immersed in water and the NaK lines were severed. The resulting NaK-water reaction was complete within 15 sec. This implies that the core vessel was filled with water at the end of 15 sec and indicates a flowrate of water into the reactor of at least 0.45 lb/sec, which is a factor of 3.5 greater than the rate at which water would boil with an average core temperature of 1300°F.

The pressure drop across the core for a water flowrate through the core of 1 ft/sec has been estimated to be approximately 0.43 psi, which corresponds to a 1-ft head of water. This pressure drop includes the drop across the reactor inlet nozzles, the reactor outlet nozzle, the coolant channel, the orifice plates, and both baffle plates. On the basis that the water flow-

rate into the reactor, with no NaK in the core, can be assumed to occur with a driving force equivalent to a 1-ft head of water, the above result indicates a flowrate of 2.8 lb/sec, which is a factor of 21 greater than the rate at which water would boil.

The conclusions to be drawn from these arguments are:

- 1) In the event of a water impact accident in which the NaK lines are severed, water would enter the core at a rate that is large compared to the maximum rate at which the water could be caused to vaporize.
- 2) The reactivity increase accompanying the filling of the core would cause a power excursion in which the fuel-moderator temperature would exceed 1900°F.
- 3) Disintegration of the fuel-moderator structure, under these conditions, is highly probable and would cause the core to be disassembled.

The above conclusions are based on the hypothesis that, if the fuel-moderator is heated to 1900°F, disintegration of the fuel-moderator structure will occur. Disintegration of the fuel-moderator structure is accompanied by a sudden release of hydrogen gas to the surroundings. It is the buildup of hydrogen gas pressure within the system that causes the core to disassemble. Disintegration of the fuel-moderator structure and core disassembly was observed in the SNAPTRAN-3 test. In this test the time for the fuel-moderator temperature to reach 1900°F was short compared to the time for hydrogen diffusion out of the fuel element. For slower heating rates, the buildup of hydrogen pressure within the fuel element cladding, due to diffusion of hydrogen out of the fuel element, might be sufficient to cause the cladding to rupture before the fuel-moderator temperature reached 1900°F. This could result in a termination of the power excursion without disintegration of the fuel-moderator. In either event, there will occur a

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sudden release of hydrogen from the fuel-moderator and it is highly probable that the core will be irreversibly shut down.

Terminal velocity water impact tests utilizing mockups of the SNAP 10A reactor were conducted under the Aerospace Safety Program.⁸ Gross violations of core integrity were observed in all cases. The core vessel ruptures were obviously more than adequate to allow replace-

ment of the internal NaK by water within 15 sec. Thus, it is concluded that although the SNAP 10A reactor is now considerably less reactive in a water environment, a nuclear excursion terminated by loss of hydrogen and/or core disassembly will occur following terminal velocity impacts of the reactor into water such as would result from a launch vehicle failure over ocean areas.

4. RADIOLOGICAL ANALYSIS

4.1 Fission Product Release Fraction

In the SNAPTRAN-3 experiment,^{5,9} a SNAP 10A reactor was subjected to a rapid exposure in a water-reflected environment such as might result from a launch vehicle failure. This test was primarily designed to investigate the magnitude of the nuclear energy release from such an accident and to confirm the previous hypothesis of physical self-destruction of the core by the hydrogen escaping from the fuel material. As an adjunct to these data, the opportunity afforded by this test to obtain some further information on fission product release fractions was not overlooked. A rather elaborate system of detectors and samplers for airborne radioactivity was provided. Although the evaluation of the results is not yet completed, the following conclusions appear assured:⁹

- 1) No fuel melting occurred. Maximum fuel temperature of about 1900°F.
- 2) Less than 5% of the noble gasses (Xe, Kr) formed during the excursion were released from the fuel.
- 3) No iodine release from the water tank was detected by the downwind sampling grid (other than some small amounts formed in the noble gas samplers which resulted from Xe decay).
- 4) Only trace amounts of I were measured in the tank water.

- 5) Only trace amounts of other isotopes were found in the tank water.

The low radioactivity release in the absence of fuel melting correlates well with previous experiments conducted with furnace-heated samplers of preirradiated SNAP fuel.¹⁰ Although the amount of iodine released from the fuel has not yet been qualitatively determined, it appears that the water formed a very effective trap for that which did escape. It has been postulated that any iodine released from the fuel combined with the free hydrogen releases to form hydriodic acid which is highly soluble in water. In any case, it would be difficult to imagine an iodine release from the fuel greater than that measured for the noble gases (i. e., >5%).

On the basis of the SNAPTRAN-3 test (and the previous F. P. release experiments), it would now appear that considerable support can be given for a reduction of the volatile portion of the previously assumed fission product release model; namely, a 5% release of the rare gasses and halogens instead of the 100% which was used in the Final Safeguards Report. To indicate what effect such a reduction would have in the previously calculated radiological hazards, Table 4 presents a comparison of dose-distance relationships for thyroid inhalation dose and Table 5 presents a similar comparison for the whole body cloud gamma exposure mode. Two examples of reactor power histories are

TABLE 4
DOWNWIND DISTANCES FOR INHALATION THYROID DOSES FROM
DIFFERENT FISSION PRODUCT RELEASE FRACTIONS

Reactor Power History	Fission Product Release Fraction	Dose (rem)			
		1,000	300	100	30
		Distance in Feet			
70 Mw-sec Excursion (No Prior Operation)	5% Volatiles - 1% Remainder	-	-	20	72
	100% Volatiles - 1% Remainder	44	120	275	650
70 Mw-sec Excursion (3 yr Prior Operation)	5% Volatiles - 1% Remainder	270	620	1,320	2,950
	100% Volatiles - 1% Remainder	2,100	4,100	9,500	20,000

TABLE 5
DOWNWIND DISTANCES FOR WHOLE BODY CLOUD GAMMA DOSES
FROM DIFFERENT FISSION PRODUCT RELEASE FRACTIONS

Reactor Power History	Fission Product Release Fraction	Dose (rem)			
		100	25	5	0.5
		Distance in Feet			
70 Mw-sec Excursion (No Prior Operation)	5% Volatiles - 1% Remainder	-	-	35	205
	100% Volatiles - 1% Remainder	30	95	280	1,100
70 Mw-sec Excursion (3 yr Prior Operation)	5% Volatiles - 1% Remainder	-	-	57	290
	100% Volatiles - 1% Remainder	43	130	385	2,750

shown which essentially encompass the minimum and maximum fission product inventories which could exist.

In the case of an excursion which is preceded by long-term reactor operation, it should be noted that an effect of such operation on the release fraction may occur. Thus, it would probably be prudent at this time to only apply the reduced volatile release to excursions which are postulated to occur prior to any significant reaction operation at power.

Since the previous evaluation of the hazards originating from this potential exposure mode (i. e., airborne release of radioactivity) has shown them to be either within acceptable limits or overshadowed by other modes (e. g., direct radiation), no quantitative revision of the injury probability calculations has been attempted.

4.2 Short Term Reactor Operations

Under some conditions (e. g., less than nominal orbit lifetime), limited reactor operation

terminated by ground command shutdown, may be of interest. Such a procedure permits an alternate method of limiting the fission product inventory at the time of reentry. For example, Figure 2 presents a plot of permissible reactor operating times as a function of orbital lifetime

to allow sufficient pre-reentry decay to reduce the fission product inventory to various levels. Figure 3 is a similar display with one hour at 1-ft single fuel element doses substituted for the fission product inventory values.

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Figure 2. Orbital Lifetime Required for Fission Product Inventory to Decay to Various Activity Levels

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Figure 3. Orbital Lifetime Required for the Dose 1 ft From a Fuel Element to Decay to Various Levels

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For the most significant short orbit (1 to 4 mo.) which could result from an Agena second burn failure, reactor shutdown following a minimum operating time of 2.4 days would permit sufficient decay of the fission product inventory so that the dose from the exposure to a fuel element for one hour at 1 ft would be <25 rem.

4.3 Reentry Hazards - Localized Fallout

4.3.1 Introduction

In the Final SNAPSHOT Safeguards Report, it was recognized that a number of radiation exposure modes other than those specifically analyzed could exist. The exposure modes analyzed in detail were selected as conservative representatives after examining both the likelihood of occurrence and the potential hazard presented to a population for the various modes within a specific exposure category; i.e., external or internal exposure. Included among the modes which were not specifically analyzed was the possibility of complete fuel element ablation during reentry leading to subsequent deposition of radioactive particles on the earth's surface.

From the analysis presented in the Preliminary SNAPSHOT Safety Report,¹¹ it was concluded that the hazard to the world's population from such particles would be negligible. These conclusions were based on the considerations that the size of the particles resulting from the ablation process would be small and ablation would occur at such an altitude that the meteorological environment which the particles would encounter as they settled, would cause the particles to be spread over very large areas of the earth's surface. The size of the particles would also result in relatively slow settling velocities, thus enabling considerable decay in particle activity to occur before they reached the earth's surface.

Continuing investigations in the Aerospace Safety Program on reentry burnup indicate that burnup of the fuel element could not be completely

assured for all reentry modes of the power unit. Therefore, under certain conditions, the fuel elements might possibly survive the reentry environment and impact essentially whole. With this consideration, it was postulated that whole fuel elements impacting on the earth's surface would certainly pose a greater hazard than widespread deposition of small fuel particles. However, the possibility of ablated fuel element particles or fragments settling back to earth in a highly localized area, although very remote, does exist. Therefore, the model for determining the hazard from ablated fuel element particles which have settled into highly localized areas was developed and compared with the hazards from the discovery and handling of whole fuel elements.

4.3.2 Exposure Model From Ablated Fuel Element Particles

The following assumptions were used in the analysis:

- 1) Complete ablation of the fuel elements occurs and the material is reduced to particle sizes.
- 2) Radioactive decay of the particle activity will be neglected until the particles reach the earth's surface.
- 3) During the ablation process, the entire fission product activity will be retained within the particles; i.e., no "boil off" of the volatiles. (Complete loss of the volatiles only reduces the one-year dose approximately 12%).
- 4) Uniform deposition of the materials occurs within the area under consideration.
- 5) Weathering, shielding, or removal of the deposited particles does not occur during the exposure period.
- 6) The exposure period is one year.
- 7) No reactor shutdown prior to reentry. The reactor power history is referenced from the Final SNAPSHOT Safeguards Report.

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8) The contaminated area can be represented by a uniformly contaminated disc source.

4.3.3 Discussion

The size distribution of the ablated fuel element material determines both the area extent to which contamination of the earth's surface will occur and to some degree the magnitude of the potential population exposure due to the decay of the particle activity before it reaches the earth's surface. Primarily, the magnitude of the dose is determined by the fission product inventory at the time of reentry.

Calculations in the Aerospace Safety Program and the RFD-1 flight test indicate that reentry disassembly of the SNAP 10A reactor will occur at a height such that the fuel element will be released and subjected to a considerable portion of the aerodynamic heating pulse. Data obtained from arc-jet tests on fuel elements lead to the following conclusions concerning fuel element ablation:

- 1) A comparison of the portion of the aerodynamic heating pulse to which reentering fuel elements are subjected and the heat pulse required to completely ablate a fuel element in the arc-jet shows a 30% excess in the thermal energy available during reentry aerodynamic heating to ensure complete ablation.

- 2) The time required for complete ablation as extrapolated from arc-jet tests is a 3- to 5-sec interval.

- 3) The particle size distribution of the ablated material was not determined. Some large fragments were found under the specimen holder in the arc-jet chamber. The remainder apparently escaped through the chamber air filter system.

In the absence of data for the size distribution of the particles, the potential population

exposure has been determined as a function of the area over which the particles are assumed to be distributed and the reactor power history. Figure 4 shows the potential one-year dose 3 ft above the ground as a function of a uniformly contaminated area for reentry from orbits of various lifetimes assuming no reactor shutdown prior to reentry (but reactor power degradation does occur).

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Figure 4. One-Year Dose 3 ft Above Ground as a Function of Contaminated Area

The dose shown in Figure 4 is that calculated above a smooth plane and no credit has been taken for shielding from terrain, roughness, or the limitation of the time spent out-of-doors. To provide a realistic evaluation of the exposure, an estimate of the dose reduction factor from these parameters is given.¹²

- 1) Terrain roughness — open field, 0.87; steep, rough terrain, 0.58; residential areas, 0.75.

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2) Shielding by dwellings and limitation on time spent out-of-doors, 0 66

The reduction in the dose from removal of the particles from both natural and man-made causes could be significant. Removal by wind and leaching from precipitation would result in a reduction factor of about 0.7 over a two-year period¹² (does not consider the material being washed away). Removal by man-made causes plowing, construction, street cleaning, washing down sidewalks, etc., is difficult to establish and varies widely depending upon the locale involved.

It is concluded from the above statements that a minimum reduction factor of 0.5 in the dose would be provided by terrain roughness and dwellings, and consequently the doses in Figure 4 are at least a factor of two high.

If an exposure is to be ≥ 25 rem from fallout of fuel element particles, it would be necessary for the particles to be contained within a maximum area of 2.3 mi^2 (from Figure 4 area $\times$ the reduction factor of 0.5)

The likelihood of the fuel element particles being deposited in a 2.3 mi^2 area appears to be very low. The horizontal velocity component along the reentry trajectory is computed to be approximately $2 \times 10^4 \text{ ft/sec}$. The lower time limit for complete ablation of the fuel element is 3 sec, thus enabling the particles to be dispersed along at least an 11-mi path. Unless the particles were of extremely large size, meteorological conditions, which the particles would encounter as they settled back to earth, would cause considerably more dispersion. According to recent studies completed by Travelers' Research Center, 40 km^2 (15.5 mi^2) would be a reasonable dispersion area for large (5 mm) particles. If the particles are assumed to fall

in such an area, the one-year dose, after three years of reactor operation would be approximately 3.5 rem. The dose integrated over a two-year period would be approximately 10% greater.

To compare the potential hazards from the two exposure modes it is necessary to establish a common base. This is taken to be a 25-rem whole body dose, irrespective of exposure time. The average number of people who would be expected to receive this dose are determined by the population density distribution and the power history of the reactor for both exposure modes. The results are shown in Table 6. A second method of comparing the hazards of the two exposure modes is on the basis of the probability of one or more persons receiving a dose of 25 rem or more. This comparison is shown in Figure 5. However, a comparison on this basis is actually misleading for the following reasons:

1) It has been discussed previously that the area over which the fallout could be expected to occur is at least an order of magnitude greater than the small area required to give an exposure ≥ 25 rem. The probability of the fallout being confined to these small areas must be included to arrive at a meaningful probability of exposing one or more persons. It is expected that the incorporation of this factor would result in a very significant lowering of the probability of exposing one or more persons ≥ 25 rem from a particle fallout.

2) The lack of information on further subdivision of population densities less than one person/square mile results in the assumption that the average population

¹²Nonuniform deposition could result in the possibility of ≥ 25 rem/yr doses in portions of a contaminated area greater than 2.3 mi^2 . However, any such effect would be largely negated by normal population movement during the assumed one-year exposure period.

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TABLE 6
AVERAGE NUMBER OF PEOPLE RECEIVING AN EXPOSURE
≥25 REM FROM A UNIFORMLY CONTAMINATED AREA
AND FROM EXPOSURE TO WHOLE FUEL ELEMENTS

Orbital Lifetime (yr)	Contaminated Area		Whole Fuel Elements
	Area (mi ²)*	Average No. People	Average No. People ¹
1	1.8	30	100
3	2.3	45	110
10	1.5	24	40
76	0.40	0.5	8
100	0.18	0.1	0.08
3800	0.032	0.008	0.008

*Area = 0.5 reduction factor x area from Figure 4.

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Figure 5. Probability of ≥N People Receiving
25R From Particle Deposition or From
Whole Fuel Element Impacts
(3 yr Reactor Operation)

in this region (i.e., 0 to 1) is 0.5 people/square mile.¹ When the contaminated area under consideration equals or exceeds 2 mi² the probability mathematically becomes 1.0 of exposing one or more persons for land impact and the probability model (Reference 1) is no longer applicable. A probability of exposure equal to 1 appears to be unreasonably conservative when one considers the large fraction

of sparsely populated land area which is actually completely unpopulated over many square miles (e.g., Arctic area).

4.3.4 Summary

1) From the data presented in Table 6 showing the average number of people receiving an exposure ≥ 25 rem, it is apparent that the exposure from while fuel elements for all orbital lifetimes is greater than or equal to the hazard from particle fallout in small areas.

2) From Figure 5 the probability of exposing ≥ N people to ≥ 25 rem from fallout is approximately a factor of two higher than that calculated for fuel element exposure for small numbers of people. However, this is reversed for large numbers of people where the probability of fuel element exposure is approximately a factor of ten higher.

3) In both methods of comparison, the probability of the particle fallout being contained in small areas has not been considered. The lower time limit for complete ablation of the fuel elements enables

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the particles to be dispersed along a minimum line source of 11 mi. It is extremely difficult to postulate the occurrence of complete particle deposition within the small areas being considered after falling through tens of thousands of feet from such a source. This is especially true for reentry from a polar orbit which results in a burnup path which is crosswind to the normal flow of high-altitude air movement. Travelers' Research Center estimates a 15.5 mi^2 area as reasonable area for large (5 mm) particles to be deposited.

4) The reduction in the dose from removal of the particles from both man-made and natural causes could be significant and is not considered in the calculations because of the wide variation possible in various locales.

5) No consideration has been given to the relatively greater significance of an acute dose delivered from exposure to whole fuel elements over a very short period opposed to the same dose from fallout integrated over a year's period. It is assumed that personnel spend the entire exposure period (i. e., 1 year) within the contaminated area. While this may be true in a very few cases, it is expected that the passage in and out of the small contaminated area in the normal course of events would probably occur and thus decrease the exposure.

4.3.5 Conclusions

From the data presented it is concluded that exposure to whole fuel elements which, under certain conditions could survive the reentry environment and impact the earth's surface, would present a greater potential hazard than particles from ablated fuel elements settling into small areas. When the probability of

the particles settling into such small areas after being released from a line source tens of thousands of feet above the earth's surface is considered, the true perspective of the relative hazard emerges and the hazard from particle deposition becomes relatively insignificant.

4.4 Reentry Hazards: Man-Rem Concept

Another concept in the evaluation of potential radiological hazards is through the use of the population integrated dose or, as it is usually termed, "man-rem." To determine the man-rem exposure for a uniformly contaminated area, the following relationship is used:

$$\text{Man-rem} = \text{area contaminated} \times \text{population} / \text{unit area} \times \text{dose received during exposure period}$$

For example, from Figure 4 a contaminated area of one square mile for a 3-year reactor operating period yields a dose of ~100 rem. The man-rem is:

$$\begin{aligned} &1 \text{ mi}^2 \times N \text{ people/mi}^2 \\ &\quad \times 100 \text{ rem} \\ &= 10^2 \times N \text{ man-rem} \end{aligned}$$

For a contaminated area of 100 square miles the dose is 1 rem and the man-rem is:

$$\begin{aligned} &100 \text{ mi}^2 \times N \text{ people/mi}^2 \\ &\quad \times 1 \text{ rem} \\ &= 10^2 \times N \text{ man-rem} \end{aligned}$$

Thus, in this case, man-rem is a function of the population density and a constant determined by the power history of the reactor and independent of area.

The relationship for man-rem can be rewritten as follows:

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$$\text{Man-rem} = \text{population density} \times C(t)$$

where

$C(t)$ = constant determined by the reactor power history as a function of time.

For a given power history the probability of achieving a specified man-rem is simply the probability of the particles falling into a certain population density. The probability of various man-rem exposures (assuming uniform deposition of the fission products) for the 3-year reactor operation case is shown in Figure 6.

The population density in typical residential areas averages about 3×10^3 people/square mile. The exposure in such an area from the 3-year case from particle fallout would be 1.5×10^5 man-rem with a probability of occurrence of 3×10^{-4} for this or higher man-rem doses. While it is possible that higher population densities would increase the man-rem, it should be noted that other factors such as multiple story dwelling units become significant considerations, thus providing a reduction in the doses. Also, very high population densities are usually more transitory in nature. The occurrence of a 2×10^6 man-rem exposure as suggested as an upper limit for maximum credible accidents in the "N. S. Savannah" report¹³ has an extremely small probability.

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Figure 6. Probability of $\geq(N)$ Man-Rem From Particle Deposition

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5. PROBABILITY ANALYSIS

5.1 Self-Welding of Reflector Components

The possibility of a self-welding bond forming between reflector and reactor core vessel components which, when coupled with ejection spring relaxation, might prevent reflector ejection was considered in the SNAPSHOT Safeguards Analysis. A conservative evaluation of the analytical and experimental data relative to this failure³ resulted in the following assumptions being used in this analysis:

- 1) Ejection spring relaxation and self-welding effects are insufficient to prevent reflector ejection upon release of the retention band for any reactor operating period up to 100 yr.
- 2) In these cases where the reactor operates beyond 100 yr, reflector ejection capability is assumed to be completely lost due to excessive spring relaxation.

Since the probability of reentry disassembly of the reactor to a noncritical configuration is assumed to be 0.99 in the absence of the reflector assembly and 0.01 with it in place, the assumptions listed above have considerable influence on post-impact criticality probabilities. Recent information received from the Aerospace Safety Program¹⁴ have revealed the following additional analysis:

- 1) Even if sufficient self-welding occurs at the location of interest (reflector assembly positioning stand-offs) such that the ejection springs cannot break the bond upon release of the retention band, these stand-offs will be ablated away early in the reentry phase and natural oscillations and/or aerodynamic forces will separate the reflector from the core. Moreover, this separation will occur early enough as not to significantly interfere with the normal reentry disassembly of the reactor.

- 2) Sufficient "creep" of the core vessel walls so that direct contact between the core vessel and reflector assembly might result cannot occur for thousands of years. Therefore, there is not any possibility of self-welding in these areas over the time of interest.

On the basis of this information, it now appears extremely difficult to support even a conservative assumption of reflector self-welding sufficient to interfere with reentry disassembly of the reactor. Thus, the 0.99 probability of disassembly should be extended throughout all orbital lifetimes under consideration (≤ 3800 yr). However, it should be pointed out that the resultant decrease in the probability of post-reentry criticality (for orbital lifetimes greater than 100 yr) has no pronounced effect on the total injury probability as calculated in the safeguards report. This situation results because the hazards of potential direct exposure to whole fuel elements over-shadows that from post-reentry criticality even when the latter was maximized by assuming self-welding.

5.2 Reactor Shutdown

The conclusion was reached in the SNAPSHOT Safeguards Report that, at all orbital lifetimes, the primary source of potential reentry hazards originates with the possible land impact of, and exposure of unknowing persons to, surviving fuel elements or fragments which may contain quantities of fission product and/or induced radioactivity. Thus, the importance of reactor shutdown to permit the maximum radioactive decay while the system is still in orbit is evident. Therefore, a re-evaluation has been made of the probabilities of reactor shutdown by both the "end of useful life" onboard automatic devices and the ground command mode.¹⁵ A summary of this analysis is presented below.

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The current SNAP 10A design has a high reliability for continuous operation for one year provided orbital startup can be initiated. This analysis considers the reliability of those components which can influence the probability of reflector ejection. Equations are developed for conservatively combining the logically interdependent reliabilities.

The general procedure is to set forth the reliability block diagram relating that equipment which is essential to the reflector ejection into logically alternative paths. The equations for combining these logical statements are employed over a time continuum in order to yield specific solutions for the most likely probability of ejection. The key logic diagram used in this analysis is shown in Figure 7.

NOTE: SERIES PATHS ARE FIRST COMBINED BY MULTIPLICATION, ALTERNATIVE PATHS ARE THEN COMBINED BY ADDITION.

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Figure 7. Logic Diagram for Automatic and Commanded Ejection of Reflectors

The input to all the calculations is the form of curves for the survival of components or subsystem to stated failure criteria. These curves have substantial basis in test data and reliability analysis considering launch stress and operating times of one year.^{16,17} Beyond that time the curves are determined by the engineering consideration of probable failure mechanisms, failure modes of consequence to this problem, and the assumption of a generally increasing failure rate due to environmental hazards including radiation and micrometeorites.

The resulting solutions, shown in Figures 8 and 9, are only conditionally sensitive to anyone input survival curve. However, the influence of the command system on-board the Agena is of considerable importance if it is to be relied upon as a primary shutdown means. If a conservative reliability estimate of the Agena command

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Figure 8. Probability of Reflector Ejection in Orbit

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Figure 9. Proportion of Operating NPU's Which Will be Shut Down Either Automatically or Optionally by Command at Time t

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system is employed, 87% of operating nuclear power systems can be shut down in 90 days. If a more realistic estimate is employed, the same proportion of NPU's can be shut down by command at one year. The resulting curves are weighted estimates and convey nothing about statistical confidence.

It appears likely that from 50 to 70% of all the SNAP 10A reactors which reach a significant power level will be shut down by auto-

matic reflection ejection within 5 to 10 years following startup. However, the flight systems have no automatic means of increasing the probability of reflection ejection about 70% irrespective of the time which may elapse beyond 10 yr. At operating times up to about one year, the shutdown probability can be considerably enhanced by including the possibility of the ground command shutdown mode.

6. MISCELLANEOUS

6.1 Launch Pad Procedures

The actual operations involving the 10A Nuclear Power Unit which are carried out on the launch pad remain essentially as described in the safeguards report. However, the sequence and chronology of the pre-countdown portion of the operations have changed somewhat.

Pad activation and validation tests, checkout of the support equipment, and checkout of the electrical test set TSM-1 will be accomplished several months prior to the scheduled launch date, utilizing a nonnuclear electrical mockup of the 10A power system.

Approximately 30 days before the scheduled launch date, the Nuclear Power Unit will be brought to the pad from the SPTF and mated to the Agena for checkout and validation tests. These tests will require a period of eight days, after which the Nuclear Power Unit is demated and returned to the SPTF for a storage period of approximately 15 days.

Approximately seven days before the scheduled launch date, the Nuclear Power Unit will again be brought to the launch pad from the SPTF for the final cycle of pad activities. This cycle culminates in the launch of the Nuclear Power Unit.

This sequence of operations precludes the need for storage of the Nuclear Power Unit at the launch pad. Thus, the field storage trailer described in the report has been eliminated.

6.2 SNAPSHOT Responsibilities

Subsequent to preparation of the safeguards report, the U. S. Navy has relinquished jurisdiction over the Point Arguello launch area to the U. S. Air Force. Therefore, what was formerly the Naval Missile Facility at Point Arguello (NMFPA) is now part of Vandenberg Air Force Base. The Pacific Missile Range is now known as the Air Force Western Test Range. However, complete jurisdiction over the entire range has not yet been transferred to the Air Force. Therefore, at the present time, the Navy (Commander Pacific Missile Range) has retained responsibility for missile flight safety.*

As a result of this transfer of jurisdiction, the general operational responsibilities described in Appendix F¹ have changed somewhat. The following modifications are required to bring this description to current status:

- 1) Commander, Pacific Missile Range
The responsibility of Pacific Missile Range is now limited to missile flight

*Recent information has been received that the transfer of missile flight safety responsibilities from the U. S. Navy to the Western Test Range has taken place on February 1, 1965.

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safety. This responsibility begins at the moment of liftoff and terminates at the moment the missile no longer has the capability of impacting on populated land masses. The Missile Flight Safety Officer, located in the PMR Flight Safety Facility at VAFB, monitors the missile flight and, in the event of an errant flight which threatens to violate the range safety criteria, initiates missile destruct.

2) Commander, Air Force Western Test Range

The Commander, Air Force Western Test Range is responsible for normal safety precautions within the launch complex and for providing the Launch Complex Safety Officer and technicians.

3) First Strategic Aerospace Division

The First Strategic Aerospace Division (1 STRATAD), in addition to the responsibilities previously described in the safeguards report, has assumed the responsibilities for safety criteria and disaster control for the launch complex as well as for the rest of VAFB. All disaster control operations will now be under the direction of the 1 STRATAD On-Scene Commander of the Disaster Control Team. 1 STRATAD will also furnish the Missile

Accident Emergency Team Chief, who will assure that area safety criteria are met and that specially trained persons are available for emergency operations.

4) 6595th Aerospace Test Wing

The responsibilities of the 6595 Aerospace Test Wing (6595 ATW) remain essentially as described with the exception that hazardous operations no longer require scheduling with PMR. Additionally, the 9595th ATW no longer directs the activities of the Disaster Control Team for accidents within the launch complex.

A revised chart of organization responsibilities is shown in Figure 10.

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Figure 10. SNAPSHOT Responsibilities

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