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SP-100 PROGRAM

SPACE REACTOR SYSTEM
AND
SUBSYSTEM INVESTIGATIONS

SAFETY PROGRAM REVIEW
SPACE REACTOR
ELECTRIC POWER SYSTEMS

Prepared for
the United States Department of Energy
under Contract DE-AT03-82SF11687

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1.0 INTRODUCTION

For a space reactor power system, a comprehensive safety program will be required to assure that no undue risk is present. This report summarizes the nuclear safety review/approval process that will be required for a space reactor system. The documentation requirements are presented along with a summary of the required contents of key documents. Finally, the aerospace safety program conducted for the SNAP-10A reactor system is summarized. The results of this program are presented to show the type of program that can be expected and to provide information that could be usable in future programs.

2.0 NUCLEAR SAFETY REVIEW/APPROVAL AND DOCUMENTATION

For any space reactor system program, safety will play a key role — from early concept design studies to the actual launch and operation of the reactor power system. Over the past two decades, the role of safety in space nuclear power has evolved to where predictable review/approval and documentation requirements are well established. These requirements were developed primarily for radioisotope systems; however, they have been expanded in recent years to include space nuclear reactors. The required approval steps and documentation are summarized in Figure 1. Both the Department of Energy (DOE) and the Interagency Nuclear Safety Review Panel (INSRP) are involved early in the process. DOE's involvement is obvious, since by law it is responsible for safety. The role of INSRP in the safety review and approval process is discussed below and presented in more detail in Refs. 1 and 2.

Two key documents will be required early in any space reactor power system: the Nuclear Safety Criteria and Specification and the Program Safety Plan. An early form of the Nuclear Safety Criteria and Specification is presented in Ref. 3. This form of the document will require updating as program requirements become more definite and missions are selected. The specification portion will likely become a living document used throughout the program and will require periodic updating as detail is established. The Program Safety Plan, a very important early program document, should provide an overall plan for formulating, integrating, and achieving all necessary safety elements of the program required to meet safety objectives and eventually to obtain flight approval. More specifically, it will provide the logic and strategy for safety assessment, analysis, and tests in support of the critical technology development phase of the program. Section 3.0 presents more details on this document.

The next key document required in the program will be the Preliminary Safety Analysis Report (PSAR). This document will be required for a given mission shortly after the concept design has been selected and after conceptual safety analysis and testing have been completed. The PSAR will describe

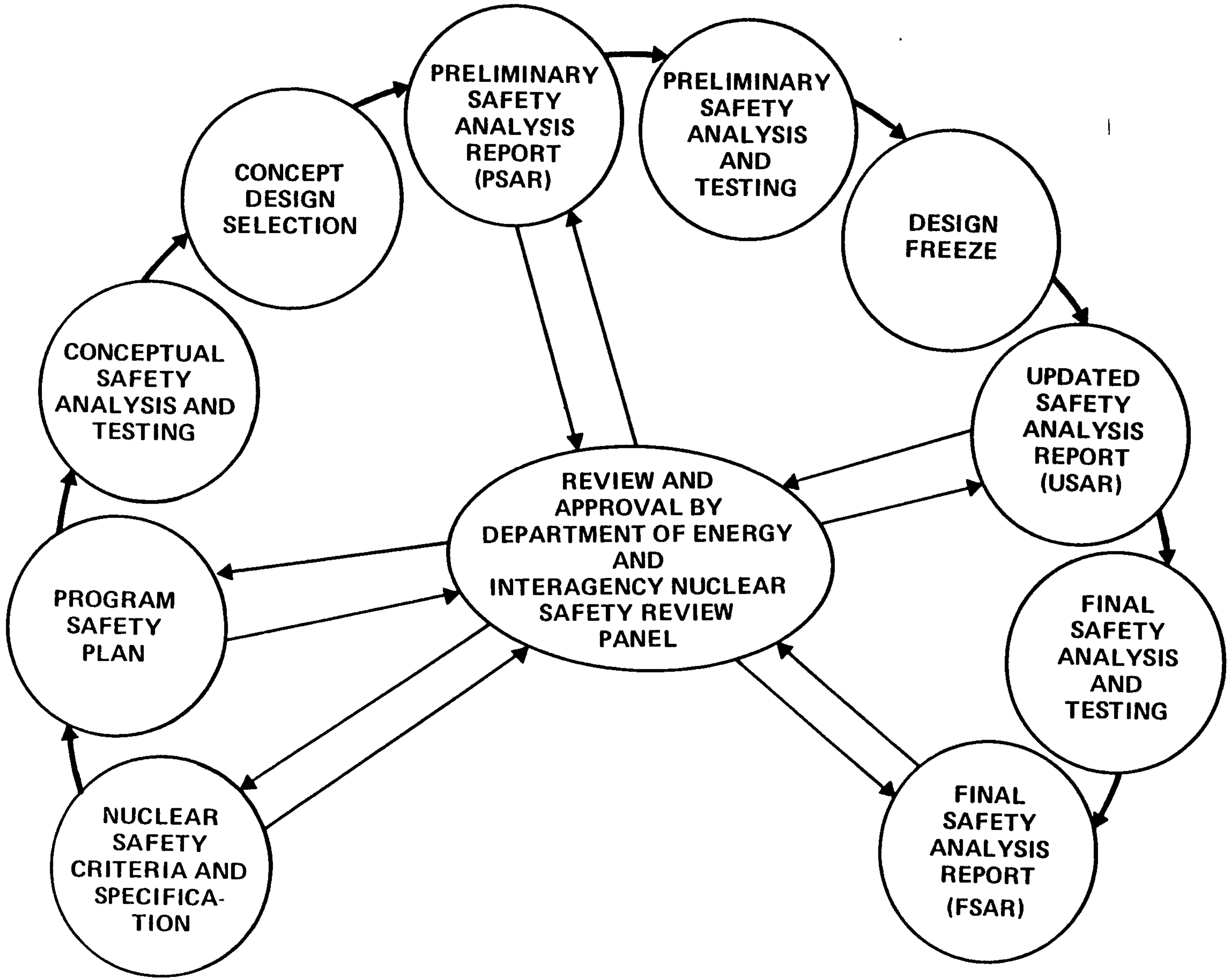

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Figure 1. Nuclear Safety Approval Steps and Documentation

the nuclear power system and the mission and will include a probabilistic radiological risk assessment supported by available conceptual design data. Section 3.0 provides more detail on the contents of a PSAR. The formal safety review process within the INSRP will start with the submittal of the PSAR. This panel, however, should be involved at the earliest possible program stage because of its expertise, the valuable data it possesses, and the contribution it can make to forming a nuclear safety program.

The second formal safety report will be the Updated Safety Analysis Report (USAR). This report will be issued as soon as practical after the power system design freeze. The USAR will include updated information on the mission, failure modes analysis, and radiological risk assessment plus any required safety tests and data. The Final Safety Analysis Report (FSAR) will normally be issued about 1 yr before the scheduled launch. The FSAR will describe the final design of the system, the mission, and radiological safety assessment data (including the results of the safety analysis tests).

The discussion above describes the key nuclear system contractor documents and their relationship to the overall program. It does not represent the overall cycle required in the flight safety review and launch approval process; the overall review process is illustrated in Figure 2. The contrac tors' Safety Analysis Reports (SARs) will not be the only inputs to the INSRP. The INSRP has members not only from DOD, NASA, and DOE, but also from various active working groups. Approximately 1200 scientists and engineers from a number of government agencies, laboratories, and universities will assist in the review. These specialists will evaluate the SARs and provide independent calculations and tests as required by the INSRP.

Once the FSAR has been reviewed and the INSRP members are satisfied, the panel will prepare a Safety Evaluation Report (SER) and submit it to agency heads for review and approval. The SER will provide necessary summary information to the offices indicated in Figure 2. Although the INSRP will not make a recommendation for launch approval or disapproval, the results of the INSRP review as documented in the SER certainly will be a factor used by agency

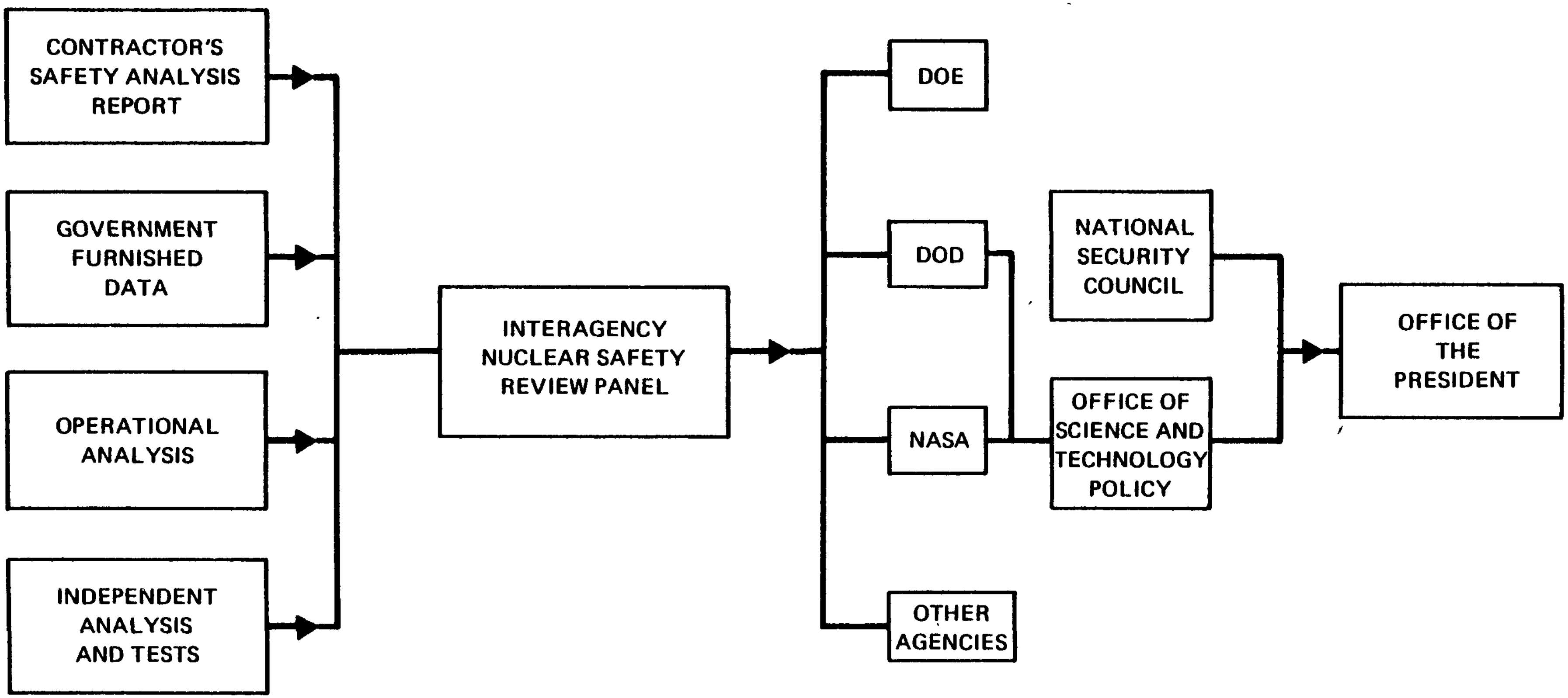

Figure 2. Safety review and launch approval process.

heads in making that decision. After the SER has gone to the agency heads and they are satisfied with the findings, the two supporting agency heads will submit letters of concurrence to the user agency. In turn, the user agency will submit a letter to the Office of Science and Technology Policy requesting launch approval. This request may in some cases be acted on at that level or in others by the Office of the President.

3.0 KEY NUCLEAR SAFETY DOCUMENTATION

As discussed above, three basic types of contractor documents will be required in a space reactor safety program:

- Nuclear Safety Criteria and Specifications
- Program Safety Plan
- Safety Analysis Reports.

This section considers the more important aspects of these documents.

3.1 NUCLEAR SAFETY CRITERIA AND SPECIFICATION

The Nuclear Safety Criteria and Specification for Space Nuclear Reactors[3] is a two-part document. Part A defines the nuclear safety criteria that must be met to implement U.S. government policy. It also provides safety documentation requirements for the program. It is not expected that Part A of the document would change unless U.S. safety policies are changed.

Part B is the nuclear safety specification that is specific for a given power system and mission. It provides the specific functional requirements for meeting the safety criteria given in Part A. The specification given in Ref. 1 is specific to the SP-100 space nuclear reactor power system technology program. Since the SP-100 program has not selected a nuclear power system concept or defined a mission, the requirements are, by necessity, very general. It can be expected that the nuclear safety specification will be periodically updated as requirements are better defined, designs are made more definite, and comments are received from INSRP.

3.2 PROGRAM SAFETY PLAN

The Program Safety Plan will be a key safety document and will require high-level attention early in the program development. It not only will provide the complete safety planning, but it also will be useful in providing early comment from INSRP as to the adequacy of the program.

The purpose of the Program Safety Plan will be to provide an overall plan for formulating, integrating, and achieving all necessary safety elements of the program required to meet safety objectives and eventually to obtain flight approval. More specifically, it will provide the logic and strategy for safety assessment, analysis, and tests in support of critical safety technology developments. The basic elements to be included in the Program Safety Plan are summarized below.

3.2.1 Event Tree Analysis

An event tree analysis will be required to identify those events that could occur over the power system life cycle that could raise nuclear safety issues. The event tree analysis will establish basic components and sequences of the safety tests and analysis requirements.

3.2.2 Program Logic

Based on the items identified in the event tree analysis, a logic diagram will be required to show the interrelationship between test needs and analysis to resolve all critical safety issues. As an aid in preparing the program logic, a table similar to Table 1 should be prepared. For each phase from factory through flight, this table will give the safety objectives, potential environments, and the methods to meet the safety objectives. Such a table will serve to focus planning on the safety objectives and how they can be met. Analyses to define the response of the reactor to these environments will then provide more detailed input to the safety specification.

From the event tree analysis and safety objectives, a time-phased program logic diagram should be prepared. The logic diagram should show how the program elements feed into each other showing how tests support analyses. Any

TABLE 1

SAFETY OBJECTIVES

Phase	Safety Objectives	Environments	Methods to Meet Objectives
Ground handling and transportation	Maintain subcriticality	Water immersion Nearness of humans Fire Dropping Puncture	
Prelaunch and launch	Maintain subcriticality	Overpressure Fireball Shrapnel Impact Afterfire (liquid and solid propellant) Thermochemical reactions Postimpact water immersion	
Ascent	Maintain subcriticality Facility recovery	Fire Explosion Shrapnel Reentry/impact Postimpact	
Orbital	Minimize biosphere and space contamination Maintain subcriticality on reentry and impact	Reentry/impact	

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items that represent potential critical technology issues should be clearly identified.

3.2.3 Safety Strategy

The Program Safety Plan should also contain a safety strategy that identifies the logic necessary to carry out the safety plan, including organizational structure, responsibilities, and authorities. Activities should be shown for each organization that clearly identify areas of responsibility.

3.3 SAFETY ANALYSIS REPORTS

The SARs, as specified in Ref. 3, should be similar in format to Table 2. Each SAR must consider all environments and be categorized by mission phase. Examples of the various environments that must be considered are listed below:

Prelaunch, Launch, and Ascent Phases

- Explosion overpressure
- Projectile impact
- Land or water impact
- Liquid propellant fire
- Solid propellant fire
- Sequential combination of the above

Orbit and/or Flight Trajectory Phases

- Reentry
- Land or water impact or collisions in space (meteroids, space debris)
- Postimpact environment (land or water)

The PSAR will normally only include the first two documents in Table 2, the Reference Design Document and the Accident Model Document. The USAR (if

TABLE 2

CONTENTS OF SAFETY ANALYSIS REPORTS

I. Reference Design Document. This part of the safety analysis report shall contain a description of:

- Mission and flight system summary
- Nuclear reactor (including type of fuel, design requirements, reactor materials and materials properties, and radiation field at launch and during operation)
- Power conversion subsystem
- Ground support equipment
- Spacecraft (including location and attachment of nuclear reactor)
- Mission profile
- Launch vehicle (including flight safety and tracking plans)
- Reference trajectory and flight characteristics (including launch conditions)
- Launch site (including demographic, topographic, and meteorological characteristics)
- Range and radiological safety operations
- Safety-related systems, subsystems, and components (engineered safety features)

II. Accident Model Document. This part of the Safety Analysis Report shall contain a description of:

- Accident and radiological models and data (including test data and verified and validated computer codes that support the analysis)
- Vehicle and reactor failure mode analysis (from prelaunch through final disposition, with a description of the potential accident environments and flight contingency options)
- Nuclear reactor response to accident environments (including prelaunch, launch, ascent, reentry, breakup, impact, post-impact--both land and water)
- Mission failure evaluation (includes accident probabilities and quantity of radioactive material potentially released to provide a risk profile)

III. Nuclear Risk Analysis Document. This final part of the PSAR, USAR, and FSAR shall be a probabilistic description of the potential radiological risk of those potential accidents which could involve the space reactor. The extent to which this final part will be included in the PSAR will be a function of the maturity of the mission data.

sufficient information is available) and the FSAR will also include the third document, the Nuclear Risk Analysis Document.

The Overall Safety Manual (OSM),[4] a four-volume manual originally prepared for plutonium-fueled isotope systems, is currently being upgraded to include nuclear reactors. Once upgraded, it will be a generalized approach to nuclear safety. As such, it will be a key document in the preparation of SARs: it will provide standardized guidance acceptable to DOE for performing risk and environmental impact analyses. The OSM will comprise:

- Volume 1, Summary, overviews the nuclear safety analysis of these systems and presents a heat-source specification that imposes requirements and guidelines on the design and testing of heat sources intended to minimize nuclear risk.
- Volume 2, Technical Models, describes each analytical model used in the nuclear safety analysis along with associated parametrics. Supporting technical reports are included as appendices.
- Volume 3, Reference Data, compiles launch area and worldwide data on meteorology, demography, oceanography, and fuels required in carrying out the analysis. Experimental data generated by test programs are also included.
- Volume 4, Supplement, presents information generated by OSM updates that cannot be readily incorporated into other sections of the OSM by replacement.

4.0 EXPERIENCE FROM SNAP-10A AEROSPACE SAFETY PROGRAM

In April 1965, the SNAP-10A space power system with a reactor heat source and SiGe thermoelectric power converters was successfully launched and operated in space. In support of this program, an Aerospace Nuclear Safety Program was conducted to assure public and worker safety. This program provided valuable information to support analyses performed for the SNAP-10A SARs. This section summarizes the SNAP-10A Aerospace Nuclear Safety Program as a helpful basis or stepping stone for the SP-100 Aerospace Nuclear Safety Program.

The objectives of the Aerospace Nuclear Safety Program were to evaluate and control the nuclear hazards associated with the transportation, launch, operation, and disposal of SNAP systems and to develop methods and designs to assure their radiological safety. The program consisted of several analytical and experimental activities that can in general be divided into the following categories:

- Reactor disintegration
- Fuel rod reentry burnup
- Critical configurations
- Reactor transient behavior
- Mechanical and thermochemical incidents
- End-of-life shutdown
- Disposal mode studies

The activities performed in each category are summarized below.[5]

4.1 REACTOR DISINTEGRATION

Safe disposal of a reentering reactor is accomplished if the reactor structure disintegrates and releases the fuel at a sufficiently high altitude for it to melt and disperse. Thermal analyses of SNAP designs supported by experiments were conducted at ESG to determine if this disposal mode was operative.

4.1.1 Preliminary Theoretical Analyses

Preliminary analytic studies were made to describe the general characteristics of a SNAP reactor reentering the atmosphere and to help define and design experiements for improving the characterization and for substantiating analytical techniques. The system considered was primarily the SNAP-10A-Agena D reactor and vehicle. Studies were made of the aerodynamic and inertial characterisitcs of the system, the reentering satellite trajectory, reentry attitude and oscillation, stagnation and local aerodynamic heat rates, flow regimes, and failure modes.

4.1.2 Initial Wind Tunnel Tests

The necessity for predicting the disintegration of space reactor systems along their reentry trajectories made it mandatory to establish the magnitude of the hypersonic heat transfer rates. This task could not have been done by theoretical work alone, since the available hypersonic boundary layer theory was limited mainly to relatively clean aerodynamic shapes. Wind tunnel tests were used to derive comparative laminar, hypersonic boundary layer heat transfer data for the irregularly shaped envelope of the SNAP-10A reactor. The data were accumulated for various angles of attack and various simulated altitudes. The results of the experiments were used in predicting the disintegration and ablation of SNAP-type reactors during reentry.

The aerospace safety wind tunnel test program had as its objectives to:

- Gather data on the heat transfer rates from laminar, hypersonic boundary layers of complex aerodynamic shapes of the SNAP-10A reactor at various flight conditions

- Provide aerodynamic heat transfer input data for calculating the temperature distribution on the reactor components during reentry
- Evaluate possible restrictions on the quality of the tunnel data.

The tunnel tests produced a simulated reentry environment for periods of 1 to 10 ms. Scale models of the SNAP-10 reactor were coated with thermopaint for testing at simulated flight environments with relative surface heat transfer rates being indicated by the discoloration of the paint. One/fiftieth-scale models of the SNAP-10A-Agena vehicle and full-scale and 1/4-scale models of components were tested. The heat transfer data from the tunnel tests were in the form of color number contours for several components of the reactor.

4.1.3 Reactor Flight Demonstration

A high-altitude flight test of a full-size, nonradioactive model of the SNAP-10A reactor was conducted to investigate aerodynamic heating effects and to demonstrate that space reactors can be designed to break apart and disintegrate when they reenter the earth's atmosphere.

The flight test, called Reentry Flight Demonstration No. 1 (RFD-1), was conducted by NASA from its station at Wallops Island, Virginia. A NASA Scout booster vehicle was used to carry the payload, which consisted of the reactor model and a reentry vehicle that contained telemetry and recovery equipment. The flight path was suborbital, with an apogee of approximately 490,000 ft and a range of approximately 800 n.mi. The impact point was in the ocean about 250 n.mi southeast of Bermuda. Valuable data were obtained by telemetry during the descent.

Figure 3 summarizes the events during the flight. Reception of the telemetry data was good until about 350 s after liftoff, at which time the ionized gas layer that normally surrounds reentry bodies caused loss of the radio

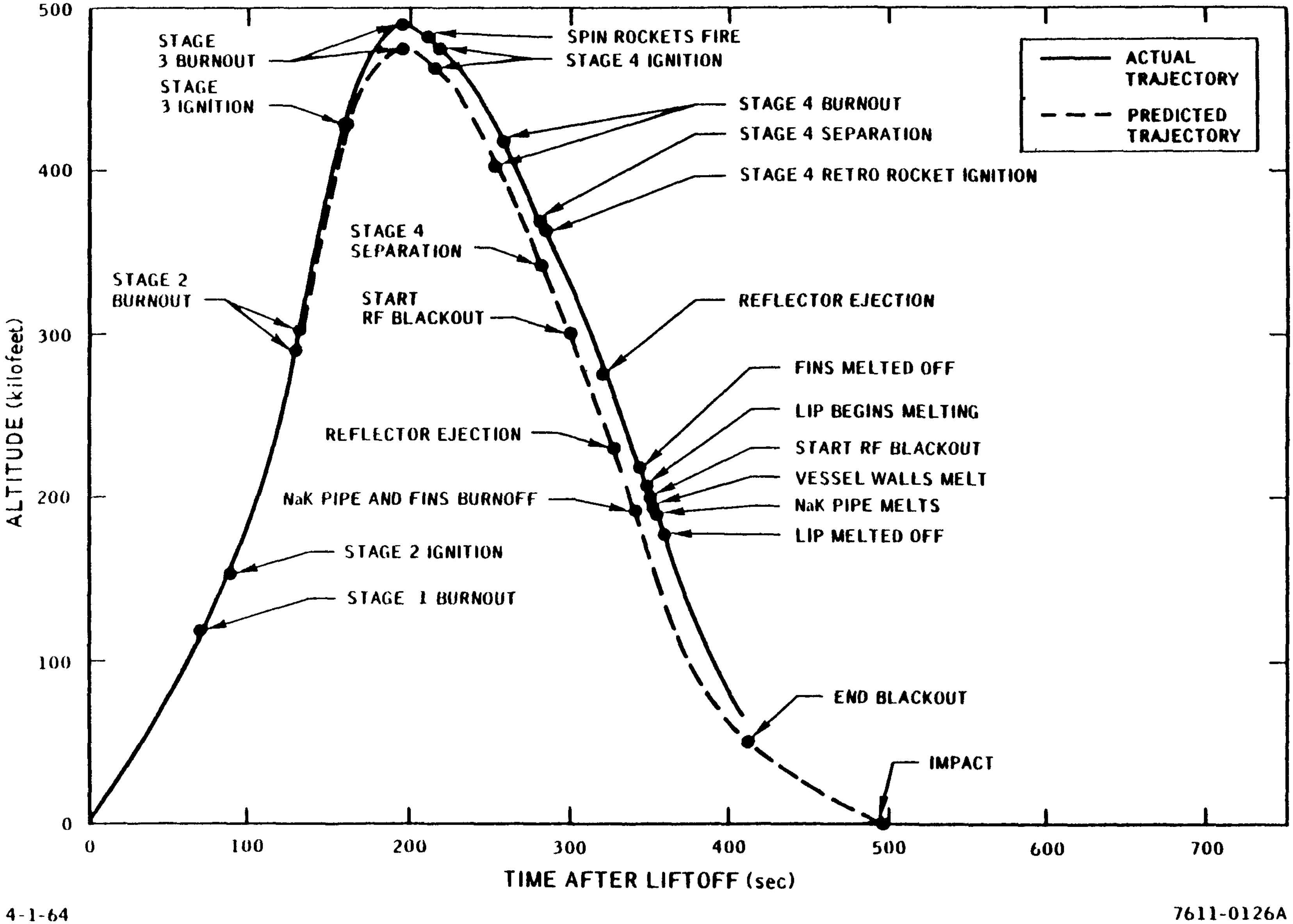

Figure 3. Reactor Flight Demonstration Trajectory

signals. The data received before rf blackout indicated significant temperature rises at the instrumented locations and separation of the reflectors as predicted. The instrumented test model was a full-scale replica of the SNAP-10A reactor, but without fuel inside the core vessel.

Heating rates were calculated from the thermocouple data for the various components. Comparison of these rates to equilibrium flow theory indicated that the theory was not well founded. An intensive study of aerodynamic heating in nonequilibrium flow and in the transition regimes between free-molecule and equilibrium flow was initiated because of the results of this test.

4.1.4 Improved Reentry Analysis

The analysis of the disintegration of the SNAP-10A-Agena during reentry was improved by using more refined data and methods for calculating aerodynamic heating rates and their effects on structural components of the reactor and vehicle. Aerodynamic heating rates obtained from the measured temperature histories of the instrumented components of RFD-1 were aplied to the case of a SNAP-10A reactor in an orbital decay trajectory, with basic corrections for differences in velocity and time of descent through various altitudes. The exact altitude of reflector ejection was shown to have little influence on the altitude of vessel melting provided the NaK is released soon enough.

A detailed thermal model was devised to calculate the altitude at which the reactor would separate from the power system. Local heating rates were based on the results of the wind tunnel tests. A similar analysis was made for the disintegration of the reactor vessel.

Based on new data and on the improved analysis, the following conclusions were drawn:

- During reentry the Agena vehicle will oscillate about an approximately nose-first attitude, with the amplitude of the oscillations tending toward smaller values as the vehicle descends. Tail-first reentry is impossible because the vehicle is aerodynamically unstable in a tail-first attitude.

- The external components of the SNAP-10A system will begin to disintegrate by melting at altitudes above 300,000 ft. Portions of the vehicle will fail at altitudes above 280,000 ft, and the reactor will separate from the vehicle at an altutude above 260,000 ft.
- The walls of the reactor vessel will melt away, releasing the partially melted remains of the top and bottom heads and allowing the core assembly to fall free at an altitude above 237,000 ft.

4.1.5 Self-Welding Experiments

Final shutdown of the SNAP reactor requires that the external beryllium reflector be ejected away from the core vessel. This must occur before reentry into the Earth's atmosphere. Should the reflector not eject, it could act as an ablative shield to the core and prevent the disintegration of the nuclear fuel rods. Although the reflector assembly is designed to be separated from the core vessel and reactor assembly, the possibility exists that thermal growth might cause the reflector assembly to be in contact with the core vessel during full-power operation of the reactor. This condition, in the combined environments of space vacuum, high temperature, and radiation, could cause the reflector to adhere or "self-weld" to the core vessel.

A special test program was conducted to determine the adhesion that could occur between the Type 316 stainless steel vessel and the beryllium reflector when subjected to the "worst case" environmental conditions. Evaluations were also made of the effect of coating the contacting surfaces with a thin film of colloidal graphite to prevent adhesion. Eight tests between Type 316 stainless steel and beryllium were conducted in the final study. Four of the tests were run for 10,000 h and four for 5,000 h. In each group of four, two were coated with the colloidal graphite and two were base metal. Each test combination was tested under vacuums of 10^{-7} to 10^{-9} Torr with the samples at 1000°F and 100 psi continuous load on the contact surface. In all cases, the data indicated that adhesion was very small or immeasurable where the contacting surfaces had been coated. But the bare, uncoated surfaces had grossly adhered to each other from contact welding; when subsequently separated, large quantities of beryllium were pulled out and transferred to the stainless steel. These data verified the preliminary adhesion test measurements made for shorter periods.

Over 200 material combinations were extensively screened in a high-temperature, ultra-high-vacuum environment to establish which combinations had the least adhesion when in contact under load for long periods. The results indicated that adhesion will be absent or minimal if dry film lubricants (such as MoS_2 plus graphite, graphite foil or cloth, or solid mechanical-grade carbon-graphite materials) are used as one of the contact surfaces, or between metal contact surfaces. On the other hand, if bare metals are allowed in contact, a five-fold increase in adhesion forces can be expected.

4.1.6 Reactor Ablation Disintegration Experiment

The Reactor Ablation Disintegration Experiment (RADE) was conducted to measure the aerodynamic heat flux to the complicated geometry of SNAP reactors and to demonstrate the capacity and accuracy of the analytical techniques used in the reactor disintegration analysis. The RADE test was performed in the hyperthermal wind tunnel at NASA's Ames Research Center.

Half-scale models of the SNAP-10A (RADE-A) and SNAP-8 (RADE-B) nuclear reactor configurations were tested in the NASA Ames hyperthermal wind tunnel under simulated atmospheric reentry conditions. Surface heat flux measurements were made with calorimeter models employing asymptotic calorimeter heat sensors. The experimental RADE models were ablated at both 0° and 30° angles of attack in the flow stream. The temperature response of the models to the aerodynamic heating of the jet stream was obtained throughout each test run from thermocouples strategically mounted in the models.

Throughout each ablation test, high-speed motion picture film coverage was obtained that provided a detailed visual description of the sequence of events during reactor ablation disintegration. These events clearly defined the meltdown and mode of disintegration of the reactor components under simulated reentry conditions. Posttest analysis showed that considerable welding between the upper grid plate and the fuel element bundle and between fuel elements probably would occur during reentry.

A computer code, the Thermal Analyzer Program (TAP), was developed to represent a 180° portion of each RADE experimental model configuration for both the head-on (0°) and 30° angle-of-attack test positions. The good agreement between the test data and the analytical code demonstrated that the preablation temperature response of SNAP reactors reentering the Earth's atmosphere could be accurately predicted.

4.1.7 Aerodynamic Heating Experiment

Reentry aerodynamic heating distributions for the SNAP-8 configuration were obtained based on hypersonic shock tunnel experiments conducted in the Cornell Aeronautical Laboratory (CAL) 48-in. hypersonic shock tunnel. The heat transfer correlations were determined as a function of angle of attack from 0° to 70°. The local aerodynamic heating factors for the 0°, 30°, and 70° angle-of-attack cases obtained in this experiment formed the basis of the heating distributions used in analytical calculations in which the shadow shield was attached to the reactor.

The aeroheating model was one-fifth scale. Test conditions were set to a simulated altitude of 250,000 ft and a free-stream Mach number of 18.4 in a hypersonic, continuum flow regime. In general, all heat transfer trends were substantiated by theory.

4.1.8 SNAP-8 Analysis

An analysis was performed to simulate the thermal behavior of a SNAP-8 reactor system during reentry. The objective of this analysis was to determine whether sufficient ablation would occur to release the reactor core at an altitude that would ensure complete burnup of the fuel elements in the upper atmosphere.

The system must undergo the following general sequence of events: (1) the melting off or removal of the upper head portion of the reactor system, (2) breakup of the reactor core by melting open the vessel wall or melting off the grid plate, (3) ablation and breakup of the fuel elements, (4) burnup of fuel elements, and (5) dispersal of the remaining fine particles into the upper atmosphere.

A detailed analytical model of the thermal behavior was developed. Unfortunately, the results of the analysis concluded that the SNAP-8 reactor vessel would not, for probable modes of reentry, ablate sufficiently to release the fuel elements at the altitudes necesary to obtain complete fuel ablation.

4.2 FUEL ROD REENTRY BURNUP

4.2.1 Arc-Jet Tests and Analysis

Arc-heated, hyperthermal wind tunnel (arc-jet) tests were performed to provide experimental support for formulating an analytical description of the SNAP fuel rod reentry behavior. Fuel rod meltdown calculations were made for orbital reentry. The wind tunnel facility at Rockwell's Los Angeles Aircraft Division was used for the tests. The test series essentially comprised the five groups of tests:

- Series 0 tests — Specimens of stainless steel and aluminum were used to check computational methods and data with materials thermally simpler than the SNAP fuel.
- Series 1 tests — Fuel material specimens were tested in a helium jet to separate from other factors the effect of surface oxidation on the fuel behavior in an aerodynamic heating environment.
- Series 2 tests — Fuel material specimens were tested in air, the runs being terminated at various time intervals to study the oxide buildup and internal state of the specimen.
- Series 3 tests — Fuel material specimens clad with Hastelloy-N were used to study the effect of cladding on reentry behavior.
- Series 4 tests — Combined phenomena tests were made of fuel material specimens in air to study the overall behavior of the fuel in the simulated reentry environment.

Specimens selected from the various groups of tests were examined after their removal from the test chamber. The most significant information was obtained from the series 2 test specimens, since the remains from these specimens essentially represented the specimens from the other series at particular time intervals during the test runs.

The initial study involved microstructural analysis with conventional metallographic techniques and spot chemical analyses. After preparation, the sections were examined, and photomicrographs and macrographs of significant structures were taken. Chemical sampling was then used to provide information on the distribution of elements within the section.

The following general observations and conclusions were drawn from this effort:

- In the sectioned specimens, the concentration of hydrogen increased with distance from the heated surface and from cracks within the specimen.
- The hydrogen was lost so that discrete interfaces were formed between the phases present.
- The zirconium-oxygen reaction at the surface was apparently preferential to the reaction of zirconium with nitrogen.
- The material in the vicinity of a molten surface was always devoid of hydrogen.

4.2.2 Rod Burnup Analysis

Analytical studies of the transient behavior of $UZrH_x$ SNAP reactor fuel elements under atmospheric reentry conditions were made to describe the conditions necessary for burnup and dispersal of the radioactive fuel material. Trajectory transient calculations were made for SNAP-10A reactor fuel elements released from the reactor vehicle at selected altitudes between 200,000 and 400,000 ft. The aerodynamic heating, transpiration cooling, and chemical reactions of the fuel with air were analyzed to determine the net surface heat flux to the elements. An analytical model was developed with which simultaneous solutions of the heat and hydrogen transport equations were obtained, including variable material properties and material phase changes.

Data from experimental tests on the transient heating and burnup of the fuel material in a hyperthermal wind tunnel were used to correlate the analytical description. Given the uncertainties in parameters, the agreement was found to be good.

4.2.3 Measurements of Thermophysical Properties

Systematic investigations of the thermal properties of SNAP fuel were made to obtain reliable thermophysical properties data needed to describe its behavior during reactor operation and reentry. Conventional measurement methods had often proved inadequate because of the high sensitivity of properties to hydrogen content coupled with the high hydrogen mobility in ZrH_x.

Heat capacity, thermal diffusivity, electrical resistivity, and thermal conductivity measurements were made on the SNAP fuel alloy, Zr-10U as well as on this alloy hydrided to H/Zr atom ratios of 0.50, 1.26, 1.58, 1.81, and 1.90.

A flash technique was used to measure thermal diffusivity. The specific heat measurements were conducted using an electrical pulse heating technique and an ice drop calorimeter. Electrical resistivity was also measured because, in the pulse heating technique, as a function of temperature it is one of the inputs required in solving for heat capacity. Thermal conductivity was calculated from the heat capacity, thermal diffusivity, and density of the materials.

4.2.4 Advanced Arc-Jet Tests

Experimental support for SNAP fuel element reentry burnup analytical programs and associated fuel particle disintegration studies was provided by a series of ablation tests on actual fuel material (nonenriched uranium) in a partially simulated reentry environment. Data were obtained on the transient temperature and hydrogen effusion behavior, surface catalytic effects on heat flux, and ablating particle size distributions and compositions.

The test series evolved from the results of previous experimental programs and from the development of analytical descriptions of the thermochemical behavior of the fuel material. The testing was necessary to provide data for the use in, and verification of, a complete description of SNAP fuel behavior during reentry.

The important feature of the test series lay in the large number of specimens tested and the large number of flow conditions and configurations used. The abundant data permitted correlations and conclusions to be made confidently. Five groups of tests were designed.

The objective of test group I was to obtain data on the aerodynamic heat flux to certain materials, primarily zirconium and zirconium oxide, when catalytic effects are significant. These data were used in correlating the thermal history data obtained in the tests.

The purpose of test group II was to evaluate the radiant heat losses for simulated reentry temperature transients. The method of evaluation was based on a comparison of thermocouple and optical pyrometer measurements. This evaluation was needed to determine the effect of surface condition (oxide buildup) on fuel emissivity. A secondary objective for these tests was to indicate the effect on the ablation characteristics of not rotating the cylindrical sample.

The primary purpose of test group III was to evaluate the effect of the radius of the fuel element on reentry behavior. This was needed because surface phenomena, as well as heat and hydrogen diffusion effects, depend on the radius of the specimen. Specimen diameters corresponding to those for the SNAP-10A and SNAP-8 fuel elements were used. A second objective in this test group was to evaluate the effects of different types of specimen holders. Both completely solid specimens and specimens with a central tungsten holder rod were made.

Test group IV was designed to study the possibility of enhancing ignition and ablation of the fuel material by surface irregularities. Grooved and cross-hatched surfaces were tried.

The primary objective of test group V was to obtain photographic data on particle size distributions during fuel ablation. Laser photographic methods were designed to supplement normal film coverage.

The arc-jet test series was performed in the hyperthermal, electric-arc wind tunnel of Rockwell's Los Angeles Aircraft Division. That wind tunnel is capable of producing nominal Mach-3 gas flows at extremely high temperatures.

Valuable data on fuel behavior during reentry conditions were obtained in the test series.

4.2.5 Particle Ablation Model

A digital computer program was written to describe the behavior of liquid metal particles in both reentry and laboratory environments. Heat, mass, and momentum transfer mechanisms were described. The program provided a means of studying particles formed from the reentry ablation of SNAP reactor fuel elements and for providing analytical support to SNAP fuel particle disintegration experiments. The code determined the temperature, radius, and velocity of a particle as functions of time.

The given particle was assumed to be initially at a temperature high enough for oxidation by a limited oxygen supply (linear oxidation). The particle was heated by aerodynamic and oxidation heating and cooled by convection, radiation, and evaporation. The oxide of the particle was assumed to form as a coating, which was assumed to evaporate, followed by evaporation of the unoxidized material, if exposed. Fission products were assumed to evaporate directly with the oxide, followed by fractional distillation from the unoxidized material, if exposed.

4.2.6 Particle Ablation Analysis

A statistical analysis was made of the behavior of reentering fuel particles from ablating SNAP-10A and SNAP-8 fuel rods, using the program described above. Sensitivity coefficients and uncertainties in the final radius resulting from variations in individual parameters were calculated. A parametric study was performed to indicate the sensitivity of particle dimension and release conditions to the percent of total aerodynamic heating received in the transition regime.

4.3 CRITICAL CONFIGURATIONS

4.3.1 Water Immersion Experiments

The overall objectives of the water immersion experiments were to determine the reactivities of SNAP-10A and SNAP-8 reactor cores immersed in water and to test the effectiveness of devices designed to maintain subcriticality.

In initial experiments, the excess reactivity of a water-reflected, bare SNAP-10A core was determined to be \$6.4 ± 0.6. The additional worth of internal water was estimated to be \$7.2 ± 0.7. Simple poison sleeves surrounding the core were found to render the water-reflected dry core subcritical, but were not adequate to prevent criticality in the water-flooded-and-reflected case.

Subsequent tests showed that:

- A combination void and poison sleeve (designated a shipping sleeve) surrounding the bare core was capable of maintaining SNAP-10A and SNAP-8 reactor cores subcritical when they were immersed in and internally flooded by water.
- Combination void and poison filler blocks that fill the four control drum voids and cover two radial beryllium faces of the reactor are capable of maintaining a beryllium-reflected SNAP-10A reactor subcritical when it is immersed in, but not internally flooded by, water. Such a device was used when working around the reactor during erection on the Agena launch vehicle. It was removed before launch.

4.3.2 Intrinsic Subcriticality Experiments and Analysis

The water critical experiments demonstrated the feasibility of using void-poison sleeves and void-poison filler blocks to preclude the possibility of accidental criticality of SNAP reactors due to water immersion and/or human body reflection during reactor shipment, ground handling, and prelaunch checkout operations. Such devices did not, however, preclude the possibility of accidental criticality during a launch abort into water. Consequently, the concept of intrinsic water subcriticality was developed to provide effective safeguards against accidental criticality during the launch phase, as well as to simplify and provide economical core shipment.

The intrinsic water subcriticality (IWS) concept provided safety in a water environment, yet maintained adequate core performance under normal operating conditions. Two methods were investigated in detail:

- Use of highly spectrum-dependent thermal-resonance neutron absorbers in the fuel elements and in the core reflector interface region
- Use of modified core and reflection geometries.

The first method takes advantage of the shape of the low-thermal-energy neutron cross-section characteristic of gadolinium and other rare-earth elements. The special shape of this cross section increases the poison effectiveness of the material in a water-immersed SNAP reactor relative to that in the operational mode. This occurs because the water softens the thermal neutron spectrum. The reactivity change that results from increasing the poison effectiveness can be sufficient to render a reactor subcritical in water while maintaining operational capabilities. In addition, burnout of the poison in the operational mode provides compensation for normal reactivity losses. Thus, prepoisoning a SNAP system produces two desired results: (1) an engineered safeguard against launch aborts and (2) a burnable poison for normal operation.

The second method exploits the high neutron leakage characteristic of SNAP reactors and the superior neutron reflecting properties of beryllium relative to water. In general, the core geometry is designed to make the reactor configuration subcritical in water yet supercritical when beryllium reflected. This technique offers the advantage of permanent water immersion safety (i.e., the reactor is always subcritical in water regardless of whether it is clean or has been operated). It has the disadvantage that the system mass is higher and the power lifetime capabilities are lower.

Initial experiments were performed to provide data for the evaluation of the use of natural gadolinium poison and core geometry changes as potential techniques for the design of SNAP reactors that would be subcritical when immersed in water, yet retain their operational capabilities. Reactivity worths of various concentrations of gadolinium, applied to the surface of the fuel elements and separately to the surface of the core vessel, were measured in both the water-immersed and the operational environments. Critical dimensions in water, as a function of core geometry, were also determined.

Analytical studies, using the experimental results as a basis for normalization, were performed in which the system weight and power lifetime penalties associated with each technique were evaluated. These studies showed that the penalties associated with the use of core geometry changes for attaining IWS were signficantly more severe than those associated with the use of a prepoison. They also showed that the dominating isotope, ^{157}Gd, had a burnout rate that was too rapid for use as a prepoison material.

Subsequent experiments were performed to evaluate the effectiveness of ^{155}Gd, ^{149}Sm, and ^{151}Eu as prepoison materials. These studies included tests to determine the effect of the prepoison on the reactor prompt-temperature coefficient of reactivity. The results showed that gadolinium and samarium reduced the negative magnitude of the coefficient and that europium increased the negative magnitude of the coefficient.

On the basis of the experiments and analysis, it was concluded that none of the prepoison materials should be used alone in the quantity (approximately \$8) required for sustained (20,000 h) SNAP reactor operation. The preferred approach would be to use combinations of two or three of the rare earths. The basic choice appears to be between these two schemes:

- \$6 of ^{155}Gd and \$2 of ^{151}Eu
- \$6 of ^{151}Eu and \$2 of ^{149}Sm.

The choice between these schemes depends on the effect of ^{151}Eu on the prompt-temperature coefficient. The rationale for either option is as follows:

- Neither ^{155}Gd or ^{149}Sm can be used alone because each has a positive effect on the prompt-temperature coefficient.
- ^{151}Eu should not be used by itself because its burnout rate is low and possible large increase in the temperature defect.
- A combination of ^{151}Eu and either ^{155}Gd or ^{149}Sm (or possibly both) could preserve the unpoisoned temperature coefficient and defect.

4.4 REACTOR TRANSIENT BEHAVIOR

4.4.1 SNAPTRAN-3 Experiment

The SNAPTRAN-3 destructive experiment was conducted at the National Reactor Testing Station (NRTS) as one of the major projects in the Aerospace Safety Program. In this test, a NaK-filled SNAP-10A reactor was submerged in water and subjected to a \$3.60 step transient above critical, which was terminated by core disassembly.

The SNAPTRAN-3 reactor consisted of a standard SNAP-10A core containing 37 fuel-moderator ($UZrH_x$) elements. The elements were positioned within the core vessel with upper and lower grid plates. The reactor vessel was positioned in an environmental tank constructed of steel-reinforced concrete, 14 ft in diameter by 10 ft deep. Since the reactor vessel was not fitted with external beryllium reflectors and control drums, reactor power was controlled with a cylindrical binal (B_4C) sleeve. The binal control sleeve was provided with two in-line drive systems: (1) motor drive for low-speed movement (2.3 in./min) and precise positioning and (2) a pyrotechnic actuator for rapid removal of the sleeve (approximately 90 ft/s) for the destructive testing. The SNAPTRAN-3 reactor was provided with instrumentation to monitor the functions listed in Table 3.

TABLE 3

SNAPTRAN-3 INSTRUMENTATION

Function	Measurement	Comment
Nuclear	Fast neutron Thermal neutron Gamma	External External External
Thermal	Energy probe, in-core Chromel-Alumel thermocouple	Radial and axial External system
Strain	Reactor disassembly	Vessel and external, in-tank
Displacement	Reactor disassembly	External, in-tank
Acceleration	Reactor disassembly	External, in-tank
Pressure	Reactor disassembly	External, in-tank
Acoustic	Reactor disassembly time of event	External
Photography	Reactor disassembly – slow, intermediate, and high speed	Documentary
Radiological	Fission product release	External

Reactivity as a function of time was obtained from power history (energy probe) data by solving the reactor-kinetics equations. Continuous energy deposition was indicated up to fuel disintegration or massive meachanical disassembly. Temperatures measured at disassembly ranged from 1800 to 2000°F. Photographs showed the vessel to bulge at the center, becoming circular as expansion proceeded. Sequential photographs of this core expansion are shown in Figure 4. Radial and axial temperature distributions were obtained from fitted energy probe data. The energy probe data were integrated to obtain the total core energy release of 32.5 ± 3.6 MW s. A fuel-moderator temperature coefficient of -0.22¢/°F up to about 1400°F was derived from the data. A reactivity feedback coefficient of expansion was derived using expansion photographs. The value was -$6.65/in. up to 0.12-in. expansion, and -$11.05/in. thereafter.

Specific results were calculated using theoretical models. Good agreements were obtained for power, energy, inverse period, reactivity, and maximum radial expansion as functions of time. Using improved analytical models based on SNAPTRAN-1 and -2 experimental results (described later in this section), the temperature coefficient was calculated to be -0.24¢/°F, in good agreement with the experiment.

An analysis made of the remains of intact fuel elements showed that fuel disintegration was caused by the generation of hydrogen pressure.

4.4.2 Electrical Pulse-Heating Experiments

The specific heat, thermal expansion, and hydrogen evolution rates for $UZrH_x$ fuel material were determined using electrically induced heating rates. Composition of the fuel was varied from an H/Zr ratio of 0.50 to 1.81. Heating rates to be studied were to range from 7000 to 100,000°C/s.

The test apparatus for the $ZrUH_x$ electrical pulse-heating experiments consisted of a three-bank 12-V nickel-cadmium battery power supply, a circuit breaker, a sample mounting system, and associated instrumentation. Maximum power transfer from the energy source to the sample during pulse heating was accomplished by impedance-matching sample and internal battery resistance. Heating rates were controlled by changing the number of batteries connected in parallel.

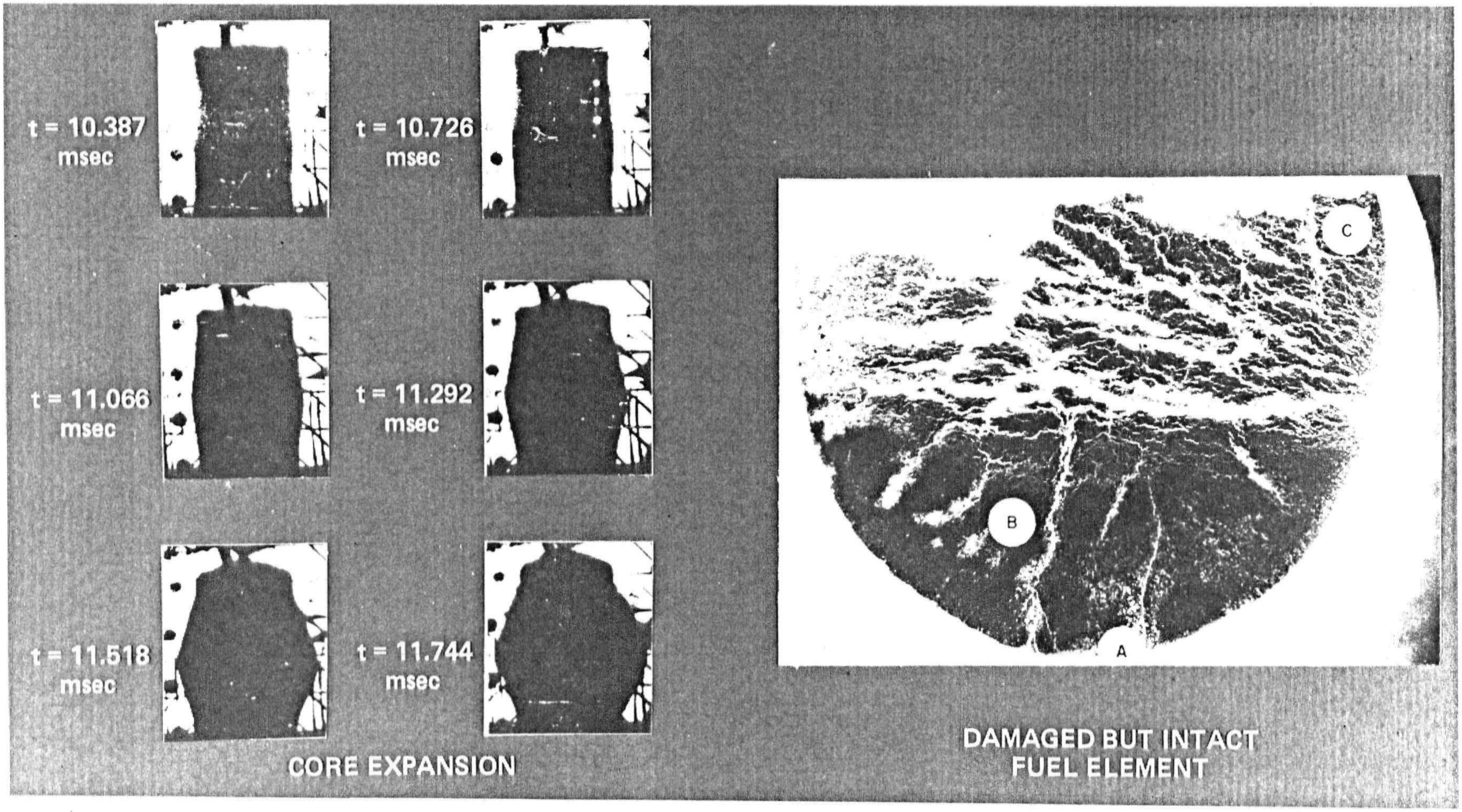

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Figure 4. Core Vessel During Expansion

Twenty-one specimens were pulse heated during the tests. The experiments were grouped according to the initial heating rate: rates greater than 80,000°C/s were in group I, from 20,000 to 80,000°C/s were in group II, and below 20,000°C/s were in group III. Specimens in group I exploded between 20 and 30 ms without evolving hydrogen; the specimens in group II exploded before 40 ms and began evolving hydrogen just before failure; specimens in group III evolved hydrogen over a longer period of time and exhibited characteristics in three time regions due to phase changes.

4.4.3 TREAT Nuclear Test Capsule Experiment

The purpose of the transient reactor test (TREAT) nuclear test capsule experiment was to measure the hydrogen diffusion coefficient, thermal expansion, and burnup of $ZrUH_x$ fuel at high temperatures. These objectives were accomplished by a series of neutron bursts obtained in the TREAT reactor at NRTS.

Each of the 17 capsules prepared for this test series consisted of a 3-in. OD by 5-in.-long by 0.188-in.-thick stainless steel shell with an internal tantalum foil used as a heat shield. These capsules were instrumented for temperature and pressure. Suspended inside each capsule was a 1.212-in.-diameter by 0.100-in.-thick wafer of $ZrUH_x$ which contained 10 wt. % ^{235}U. Mounted against the edge of the fuel wafer was a linear displacement transducer for measuring thermal expansion of the fuel. These prepared capsules received from 1 to 5 bursts in the TREAT reactor.

The fuel varied from intact to powdered and granulated form after irradiation. Not all capsules yielded temperature data, but those that did recorded sample temperatures from 1100 to 1500+°C. Postirradiation hydrogen analysis was made on each of the fuel wafers. Final values ranged from 0.14 to 1.50 wt. % for all samples originally containing 1.8 wt. %. The burnup analysis was obtained by counting and integrating over the 1.60-MeV Ba-La 140 photo peak. Samples receiving a single TREAT burst showed 6.7 ($\pm$0.6) x 10^{-6} at. % burnup.

4.4.4 Fission Product Release Tests

A series of experiments was performed to determine the fission product release from SNAP-10A $UZrH_x$ fuel samples. No prior experimental data were available for this material, and extrapolation from other materials was not possible because of the complex solid-state processes involved. Otherwise, overly conservative assumptions would have had to be made.

Twelve tests were run using six samples with standard SNAP-10A Hasteloy-N clad fuel and six samples with no cladding. Samples were irradiataed in the materials testing reactor (MTR) of NRTS in Idaho for the equivalent of about 1 yr of operation at 60 kW. Later, each sample was heated in a closed induction furnace to temperatures from 2000 to 3500°F. Temperatures were maintained from a few minutes for melts to 8 h for nonmelts. All fission products were collected in specially designed particulate collectors, iodine traps, and gas traps. The entire system was closed to the atmosphere and contained helium gas as a carrier.

Activities of isotopes of interest were determined for the fuel residue and evolved material through radiochemical analyses. Since decay times between irradiation and testing were on the order of 3 to 8 weeks, short-lived isotopes were not studied. Gross fission product release for various temperature-time conditions is summarized in Table 4.

SNAPTRAN-3 tests had indicated that, during a maximum excursion, the maximum fuel temperature in a SNAP-10A reactor was 2000°F. If this were followed by a rocket fuel fire during launch, the temperature could be maintained for about 10 min. Table 4 shows the small release for clad samples at 2000°F (for 6 h). Had these release experiments and the SNAPTRAN-3 tests not been performed, the radiation hazard from fission product release, based on conservative assumptions, would have been overestimated by several orders of magnitude.

TABLE 4

FISSION PRODUCT RELEASE FROM URANIUM-ZIRCONIUM HYDRIDE FUEL RODS IN HELIUM ATMOSPHERE[a]

Fission Product	Unclad Samples	Clad Samples[b]
^{140}Ba	10% evolved at 3200°F; less than 1% evolved at 2000°F	70 to 95% evolved from melts; 40% evolved at 1450°F; less than 0.01% evolved at 2000°F
^{137}Cs	12% evolved at 3200°F; 2% evolved at 2800°F; 0.04% evolved at 2000°F	70 to 97% evolved from melts; 25% evolved at 2450°F; less than 0.01% evolved at 2000°F
^{103}Ru, ^{106}Ru	Approximately 0.001% evolved	20% evolved from melts; less than 0.001% evolved at 2000 to 2450°F
^{141}Ce, ^{144}Ce	0.01% evolved at 3200°F; less than 0.001% evolved at 2000 to 2800°F	25 to 35% evolved from melts; 0.004% or less evolved at 2000 to 2450°F
^{89}Sr, ^{90}Sr	3 to 5% evolved at 2800 to 3200°F; 0.2% evolved at 2500°F; 0.003% evolved at 2000°F	80 to 100% evolved from melts; 25% evolved at 2450°F; 0.002% evolved at 2000°F
^{131}I	10% evolved at 3200°F; 0.2% evolved at 2000°F	100% evolved from melts; 0.2% evolved at 2000°F

[a]Nonmelt temperatures were maintained 4 to 8 h; melts were maintained approximately 20 min.

[b]At 2450°F, a clad sample evidently melted partially. After the test, the cladding appeared discolored, but there were no large cracks. Minute cracks or holes must have allowed expulsion of hydrogen and fission products. At 2000°F, the sample was not discolored, but minute openings must have occurred in the cladding.

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4.4.5 SNAPTRAN-1 Experiment

The SNAPTRAN-1 test series included several nondestructive power transients, the purpose of which was to enable a detailed study to be made of the prompt neturon kinetics of the SNAP core with a berylliumm reflector. Extensive analyses of these tests were performed to extract basic kinetics data over a wide range of variables.

The SNAPTRAN-1 reactor consisted of a SNAP-10A core and reflector. Control of the four reflector control drums was by means of special drive assemblies. Two diametrically opposed drive assemblies were provided with pneumatically driven, rack-and-pinion operation through a 45° arc, allowing step reactivity insertions ($14/s·single drum) to be made. The other two control drums were provided with impulse control (i.e., 450° drum rotation from the least reactive position through the most reactive position, then to rest at the least reactive position). The impulse mode of operation was provided with rack-and-pinion movement of the control drums. Reactivity insertion rates of $185/s·single drum were attainable. Either single- or coupled-drum step and impulse operation were provided. All four control drums were equipped with variable-speed drive motors for reactivity insertions up to $0.06/s·drum in the normal mode. Instrumentation for the SNAPTRAN-1 reactor included power, temperature, pressure, strain, and drum-position sensors.

Twenty-nine step transients were initiated from ambient starting temperatures. Reactivity inputs ranging from $0.27 to $1.50 produced reactor periods ranging from 6670 to 1.7 ms. During the initial tests, where the reactor period was greater than 500 ms, the transient was terminated with a manual scram. All other transients were terminated by a programmed scram.

One SNAPTRAN-1 objective was to extend the average core temperature as high as possible to determine the magnitude of the temperature feedback mechanisms. The previous test series was extended until maximum core temperatures were raised to approximately 1300°F as a result of a single $1.70 coupled drum transient. To accomplish this, transients were run in which the average core temperature was raised above ambient temperature using nuclear heat. The first transient involved higher reactivity inputs ($1.50 to $1.68). After thermal equilibrium in the core had been established, the reactor was subjected to a second transient, then a third and fourth of lesser reactivity inputs ($1.40 to $1.10). As a result of the transients, the maximum core temperatures were approximately the same as the previous transient, while the average core temperature at peak power increased significantly over that obtained from a single-step transient. Five step-tests were run, and temperature feedback mechanisms and reactor characteristics were determined over that elevated temperature range.

Single-drum impulse transient tests were performed to extend reactivity inputs from the maximum obtained in the step-transient series ($1.70) to $2.34. The impulse mode of reactivity insertion provided a well-defined 28-ms reactivity pulse from -135 to 0 to +135° drum rotation. Kinetic behavior of the reator was determined using a single reactivity insertion starting from various subcritical and critical power levels. The first 8 of the 27 single-drum impulse tests were initiated from milliwatt power levels. Four tests were run with a reactivity input of $2.16, and the ratio of peak power to starting power was determined. These tests provided a basis for performing similar transients at $2.34 input and higher starting power levels. The remaining 19 transients were performed with $2.34 reactivity input. During this latter series, the starting power level was increased from 10 mW to 260 W.

A series of transients was performed using the highest reactivity insertion mode of the SNAPTRAN-1 test series. Reactivities ranging from $2.00 to $4.15 were inserted by means of coupled-drum impulse tests. The influence of reflector neutron delay on dynamic behavior was observed, and the prompt-temperature coefficient of reactivity was measured in transients where significant nuclear heat was developed.

4.4.6 SNAPTRAN-2 Experiment

The analytical model refined during the nondestructive SNAPTRAN-1 tests was used to predict the results of the planned SNAPTRAN-2 destruction test. Input parameters to the analytical model were based on the capabilities of the stepper and impulse drive mechanism. The mode of reactivity insertion for the destruct test was to first insert as much reactivity as possible with the stepper drums to bring the reactor slightly supercritical and then to insert the remaining reactivity with the impulse drums. These reactivity values were programmed into the analytical model, and the results were computer analyzed. Analysis showed that the SNAPTRAN-2 reactor would undergo complete destruction with a stepwise reactivity insertion of approximately $5. By performing a destructive transient of this magnitude, knowledge of the neutronic behavior of the reactor was extended from the range observed during the nondestructive transients. Also, the mechanical and fission-product-release behaviors of this type of reactor were determined. The foregoing results and consequences were then factored into the analytical model for determining maximum reactivity accidents in SNAP-10A reactors.

The SNAPTRAN-2 destruct machine was essentially the same as the SNAPTRAN-1 reactor previously described. The impulse drum drive mechanisms were modified to allow rapid (approximately 12 ms) step reactivity insertion. To accomplish this, the impulse drums were stopped at their full-in position, 0°, using pneumatic shock absorbers. A new core was used for the destructive test. Instrumentation for the SNAPTRAN-2 reactor included: (1) reactor power; (2) temperature (energy probe); (3) radial and axial fuel element, vessel, and reflector strains; (4) acoustics; (5) motion detectors; and (6) motion pictures.

The destruct test was initiated by the full insertion of the four reflector control drums. Examination of the drum position data and slow-motion pictures showed that all drums were fully inserted before the time interval considered in these analyses. The total drum-inserted reactivity was determined to be \$5.05. Energy probe data were used to reconstruct power and energy release from the destructive test. The maximum reactor power calculated from EP154 and EP155 was 7.31×10^4 MW released compared to 7.24×10^4 MW obtained from the nuclear detectors. The released energy total was 47.25 MW s (energy probe) compared to 46.78 MW s (nuclear detectors). The temperature coefficient calculated from the energy probes was constant over the entire range of calculations and was determined to be -0.208¢/°F compared to -0.207¢/°F from the nuclear detectors.

Analysis of the fuel debris found after the SNAPTRAN-2 destruct test showed that a significant portion of the fuel material had undergone sufficient heating to melt. Verification that melting had occurred was based on: (1) spheroidization of materials found and (2) metallographic studies on fragments that found dendritic structures free of hydrogen, cracks, and voids. Electron microscope data showed that the spheroids contained only uranium and zirconium. Other identifiable fuel fragments (fuel element and pieces) showed little more than mechanical damage and minor peripheral dehydriding.

4.5 MECHANICAL AND THERMOCHEMICAL INCIDENTS

4.5.1 Phase I Tests for SNAP-10A

A Phase I mechanical and thermochemical test series was conducted in conjunction with the Air Force Special Weapons Center at Holloman Air Force Base, New Mexico. The principal objective of the tests was to obtain data and information that could be used to evaluate the hazards that might occur before, during, and after the flight of the SNAP-10A reactor.

During the factory-to-orbit sequence, the SNAP-10A reactor and its related systems were to be subjected to various handling, transportation, and launch conditions. As a result of these conditions, potential accidents, such as impacts, chemical interactions, explosions, and fires could be postulated.

To provide adequate nuclear safeguards, the potential radiological hazards that could result from the postulated thermochemical and mechanical incidents were evaluated.

The selection of the individual tests was quite critical because of the large number of accident conditions, attitudes, and geometries that could be postulated. Fourteen tests were selected that would give a statistically representative sampling of the credible conditions and would develop the most meaningful information. Where possible, limiting conditions were selected in each test. These tests were divided into three categories: (1) chemical interaction, (2) fire and explosion, and (3) impact.

The tests performed are discussed below.

4.5.1.1 Liquid Oxygen Spray Tests

The purpose of this test was to simulate the reactor being exposed to a liquid oxygen (LOX) spray from a ruptured Atlas fuel tank. It was conducted to measure the effects that the thermal shock created by such a LOX spray would have on the reactor vessel, grid plates, and fuel element array, and to observe the general behavior of the reactor for evidence of structural failure.

For 40 s, a dense spray of LOX enveloped the reactor. There was no external or internal physical damage to the reactor. A similar test was made with NaK also exposed to the LOX spray. The NaK solidified at the nozzles and no damage resulted.

4.5.1.2 Explosion Test

The purpose of this test was to measure the geometric changes of the fuel element array and to observe structural failure of the reactor and fuel element dispersal resulting from an explosion. This test simulated prelaunch or postlaunch Atlas abort and subsequent propellant explosion.

The suspended reactor was subjected to an overpressure of 400 psi for 2.8 ms caused by an explosion of 156 lb of TNT 13 ft below it. The reflectors were blown off, and the bottom of the reactor vessel deformed inwardly.

4.5.1.3 NaK-Water Interaction Immersion Test

The purpose of this test was to observe the NaK-H_2O chemical reaction and its effect on the reactor. The reactor was placed in a 36-ft^3 tank filled with water. The exposure of NaK to water caused an explosion that ruptured the tank. The core vessel was slightly ballooned by pressure buildup inside.

4.5.1.4 Drop Tests from 100 ft

The purpose of these tests was to determine the mode of failure of the reactor assembly when subjected to impact with concrete at an impact velocity of 70 ft/s. These tests were intended to simulate a reactor being dropped from the top of the launch vehicle during the mating operation. To simulate various modes of impact, test articles were dropped: (1) side-on, (2) tail-on, and (3) nose-on.

In the side-on drop test, the model impacted at 68.8 ft/s. The reflectors were knocked off and the vessel sustained a small puncture. All the support legs failed. One reflector rotated inward before the reflectors separated.

On tail-on impact, the reactor fell into the converter conical structure. All reflector drum brackets were severed, and all top pins were bent outward. The core vessel buckled circumferentially.

The velocity of the test article in the nose-on impact was 64.4 ft/s. The impact caused complete failure of the pump radiators and severed all of the reflector drums. After the vessel support legs failed, the reflector blocks were pushed into the converter structure and were held around the reactor vessel by the structure. Examination indicated that no nuclear hazard would result except for water immersion.

4.5.1.5 Fire Test

The purpose of this test was to measure differential expansion and to observe evidence of structural failure of the reactor when subjected to a high thermal flux environment. It was intended to simulate a prelaunch or postlaunch abort and subsequent propellant fire.

The reactor was surrounded by a burning cycle of 400°F flames for 2.2 s and 1500°F flames for 15 min based on launch-abort fire information. The reflector separated during the initial high-temperature burn. A local deflection inversion occurred in the lower vessel head.

4.5.1.6 Water Impact Tests

The purpose of these tests was to observe the reactor assembly during and after water impact. These tests were intended to simulate an abort after launch of 10,000 ft or above, including a free fall from the apogee.

Three reactor assemblies and one unit, including the converter and shield, were impacted at different attitudes into an 8- by 8- by 16-ft deep water tank.

A side-on water impact test was performed at 500 ft/s; the reactor passed through the water tank. The fixed reflector blocks and the reflector drums were completely separated from the reactor vessel in the water. The vessel was found intact.

The nose-on water impact test was performed at 573 ft/s; the reactor passed through the water tank. The reflector assembly separated completely from the reactor vessel in the water. Also, the NaK pump, the upper head, and the upper grid plate were severed from the test article. Fuel elements were displaced longitudinally in such a manner as to form a spherical pattern at the ends. Sufficient disassembly of the core did not occur to prevent its criticality in the water.

The tail-on water impact test was performed at 596 ft/s. The reflector assembly was separated, and the core vessel split open axially. Fuel elements were displaced longitudinally through the bottom of the vessel, and two elements were completely ejected from the array. A fueled core would have gone critical in water.

The unit with converter cone structure and shield was impacted at 428 ft/s. The reflector assembly, NaK pump, shield, converter structure, bottom grid plate, and bottom head separated from the core vessel. Fuel elements were displaced through the open end of the vessel. Core disassembly was not sufficient to prevent criticality in water.

4.5.1.7 Concrete Impact Test

The purpose of this test was to observe the reactor assembly during and after a high-speed concrete impact. This test was intended to simulate an abort after launch at 10,000 ft or higher, including a free fall from the apogee.

A concrete-faced monorail sled was impacted at 560 ft/s onto the pump end of the test article. The impact completely destroyed the reactor. The vessel was in three parts: the upper head, lower head, and the shell. The shell was completely flattened and had many local failures.

4.5.2 Phase II Tests for SNAP-10A

4.5.2.1 NaK-Water Interaction Immersion Test

The Phase II test series was a rerun of the Phase I test series under improved conditions. An accident in which the reactor is dropped into water with subsequent rupture of the NaK nozzles was successfully simulated.

Two explosive pulses were followed by a gradual pressure buildup ending at 15 s. Strain measurements showed that the stresses were generally below the yield stress of the Type 316 stainless steel vessel. The core vessel was not damaged during the test. The measured time of the initial pressure pulse was slower than the calculated time to peak power of a water-immersed core.

4.5.2.2 Fire Test on Squibs

A test was performed to evaluate the possibility of a missile abort fireball causing ignition of the SNAP-10A drum lockout pin puller squibs during a free fall of the reactor to the pad. To simulate the abort environment, the squibs were subjected to an incident heat flux of greater than 360 $Btu/s{\cdot}ft^2$ for 3 s. Three sets of two squibs each were tested. At the end of the 3-s period, the temperature of the actuator assembly had risen to approximately 600°F, but none of the squibs tested had ignited. Squib ignition took place at 8.7 s.

The significance of this test is that it demonstrates that abort heat ignition of the squibs and release of the drum lockout pins cannot occur before impact of the reactor on the pad surface. Therefore, barring some other mechanism for squib firing, the reactor assembly will be in a configuration with all four control drums locked out at the moment of impact.

4.5.2.3 Soil Impact Tests

A thermal-velocity soil impact test was performed to establish the condition of the SNAP-10A core vessel and fuel element array after a terminal-velocity head-on impact on compacted soil.

The test article, consisting of the core vessel with internals, reflector assembly, pump and fins, shield, and top part of the converter structure, was impacted into a rectangular parallelopiped of compact soil that was 126 ft wide, 8 ft high, and 8 ft deep.

The reflector assembly separated from the core vessel. Very little damage was incurred by the vessel, the most noticeable being a 1-in.-long opening in the vicinity of the top head weld. The fuel element array was not appreciably changed. The test indicated that the core vessel would survive a terminal-velocity land impact.

4.5.3 Cylindrical Vessel Disassembly Tests

In the event of launch abort or following atmospheric reentry, an intact SNAP-10A reactor could be subject to a nuclear excursion if subsequently immersed in water. Impact disassembly of the reactor core would disrupt the potential critical configuration and prevent such excursions. The purpose of these tests was to determine the effect of various postulated impact disassembly features that would render the reactor sensitive to impact, yet maintain the strength required for normal operation.

For this study, no attempt was made to simulate any particular reactor vessel. Rather, cylindrical vessels, with loads of steel bars simulating fuel rods and water simulating NaK, were used to represent probable reactor vessels.

Since most launch towers place the reactor at a minimum launch height of about 100 ft, this was selected for the test drop height. Further, as most launch pads are made of concrete, the impact target was so selected. The vessels were dropped and fin-guided to produce side-on impact. This was considered to be the least damaging to the vessel. Data on the impact dynamics were derived from transducer and photographic recordings taken from three series of impact tests.

The first series was performed at Rockwell's Rocketdyne vertical test stand, VTS-1, and at Rockwell's Atomics International Field Laboratory.

The test vessels consisted of enclosed cylinders, 9.5 in. in diameter and 12 in. long. End closures consisted of flat disk plugs 0.25 in. thick. Vessel material was Type 316 stainless steel. The vessel wall thickness and special machined stress concentration zones varied with the specimens. Each cylinder contained 31 simulated fuel elements radially restrained by wood wedges that had been machined to simulate reflector structure geometry. No grid plates were used. The elements were made from cold-rolled steel bars 1.25 in. in diameter and 11.5 in. long. Two stainless steel accelerometer blocks were welded on three of the specimens.

From the six vessels tested from this series, it was concluded that the design of a vessel definitely contributes to its impact strength properties. The judicious use of stress concentration grooves can weaken a vessel structure so that prompt disassembly will occur on impact. A pattern of three equally spaced longitudinal 90° V-grooves, 0.016 in. deep (0.032 in. wall thickness), produced the most pronounced disassembly.

A second series of tests was initiated to develop some means for enhancing impact disassembly through the use of the following features:

- Minimum-strength design
- Brittle vessel material
- Special effects such as local embrittlement, load direction, and stress concentration.

The objective of the second test series was to provide data about the relationship of stress concentration groove depth and root radius to the impact disassembly of 0.032-in.-thick wall stainless steel cylindrical vessels containing mock reactor cores. Improved testing methods based on the first series of tests were used. Of the nine vessels tested, none destructively failed on impact; however, the use of grooves were effective in causing fracture.

This second series of tests showed that cleavage fracture can be initiated and made to propagate along selected paths (grooves) in normally ductile Type 316 stainless steel. The depth of the grooves affects the extent of the fracture along the grooves. The value of the calculated elastic stress concentration factor in the range investigated does not noticeably affect the extent of rupture. A larger core void area within the vessel increases the damage to that vessel on impact.

A third series of tests was made to study the effect of further modifications to the vessel and its internals to improve impact disassembly. Two modified designs were tested: type 1, a peripheral element modification, and type 2, an external end-ring modification. As in previous tests, one criterion was to minimize the modification of the basic design. Test conditions were similar to previous tests. Seven vessels were tested in this series. Of the two types of vessels tested, type 2 was more completely disassembled for any groove depth than was type 1. When compared with previous test series, both types of vessels disassembled more completely at impact.

This third series of tests showed that it is difficult to cause disassembly of a SNAP-type reactor vessel made of Type 316 stainless steel without significantly modifying the core elements and reactor vessel. A more promising approach to enhancing the impact disassembly of compact reactor cores may be to make the reactor vessel of a less ductile material.

4.5.4 Impact Tests of SNAP Component Materials

To further study the effect of impact on the SNAP-10A reactor, two series of impact tests were performed on its component materials. Specimens were shot from a gas gun onto a granite block. The first series was conducted on solid cylinders of stainless steel, lithium hydride, tungsten, tantalum-10 tungsten, and normal zirconium hydride fuel material. Orbit reentry velocities were used. The results were used to check assumptions made in the analytical model developed to predict the postimpact configuration of the reactor. The experiments were designed to measure ductility, the ratio of dynamic-to-normal energy absorption per pound (K), and size-dependent effects.

The first series showed stainless steel to deform plastically and to be amenable to scale-model testing. Tungsten, LiH, and fuel material failed brittlely. Tantalum-10 tungsten deformed plastically and was suggested for use as the radiation shield material instead of tungsten. Velocity-dependent K values were measured for plastically deforming materials.

The second series of tests was conducted on stainless steel cylindrical shells, LiH, stainless steel-LiH composites, and Inconel 800 cylinders. Models with different scales were used to test modeling laws. Stainless steel tests verified that scale-model testing is feasible for predicting postimpact deformation. LiH tests showed that modeling laws do apply to such materials and that LiH absorbs considerable energy on impact. Postimpact deformations and the mode of failure of composite samples scaled very closely. Inconel 800 deformed plastically and absorbed more energy before rupture than stainless steel.

4.6 END-OF-LIFE SHUTDOWN

In many aerospace reactor missions, a reliable shutdown device was needed to reduce the potential radiological hazards from a reentering reactor. From early studies it was concluded that:

- From the potential radiological consequences of reactor operation, orbital shutdown is always desirable and in some cases is necessary for safety in reentering missions in which complete burnup and dispersal in the upper atmosphere is not achieved.
- Shutdown was not a natural consequence of reactor operation or space environments and hence a highly reliable shutdown mechanism is needed.
- The incorporation of radioactive tritium, which decays with a half life of 12.26 yr to ^{3}He and is a strong neutron absorber, provided an inherently reliable mechanism for shutdown of the SNAP aerospace reactors.
- For orbit lifetimes greater than $\sim$5 yr, the tritium shutdown mechanism is practically as effective in reducing radiological hazards from reentering SNAP reactors as prompt shutdown at the end of a 1-yr mission.
- The tritium-helium shutdown mechanism would essentially eliminate radiological hazards from reactor missions in orbits of greater than 400 yr.
- If tritiding is feasible, the tritium could be incorporated in the present SNAP reactors with a minimum of design modifications or reactivity penalties.
- Reactivity penalties of the tritium system would not significantly affect reactor performance for most missions.
- Incorporating tritium in a reactor would not significantly increase the radiological hazards of reactor handling after fabrication.
- From preliminary development considerations, incorporating tritium as a tritide appeared feasible, but experimental information was needed on the tritiding process to demonstrate the technique.

Later detailed studies were made of preorbital radiological safety and the feasibility of incorporating tritium into SNAP fuel. The radiological safety study concluded that hazards would not be materially increased by the adding of tritium to a SNAP reactor.

The fuel incorporation studies included:

- The effect of adding tritium on the disassociation pressure of SNAP fuel
- The amount of fuel swelling that would occur due to the presence of ^3He in the fuel
- The permeation rates of hydrogen and tritium through a SNAP fuel cladding assembly
- The redistribution rate of tritium with SNAP fuel as a function of temperature
- The rate of ^3He evolution from SNAP fuel as a function of temperature
- The retention of tritium in the cladding
- A practical process to incorporate tritium into SNAP fuel elements.

Dissociation pressure and annealing experiments showed that expected tritium additions or resulting ^3He would not appreciably affect the total dissociation pressure or fuel swelling and growth.

Knowledge of permeation and redistribution rates was needed to determine the distribution and amount of nuclear poisons in order to determine the level and distribution of power in the reactor. The retention of tritium in the cladding was needed to determine whether excessive helium embrittlement of the cladding would occur. Experimental studies showed that:

- The loss of ^3He was easily tolerable (less than 5% in 5 yr at operating temperatures).
- Tritium diffused sufficiently rapidly to maintain desired power distributions (uniform distribution in less than 72 h at 1300°F).

- The loss of tritium from fuel elements was negligible for proposed loadings (cladding permeation rate less than 10% that of hydrogen).
- No discernible effects on the mechanical properties of the cladding would occur due to the decay of dissolved tritium.

The feasibility of a process for incorporating tritium into SNAP fuel was also demonstrated experimentally. Thus, based on nonirradiated fuel measurements, the feasibility and practicality of incorporating a tritium-helium shutdown mechanism into SNAP fuel was established.

4.7 DISPOSAL MODE STUDIES

Methods for safely disposing of SNAP reactors after space operation were evaluated. Three principles were employed: decay, dilution, and recovery. Three basic methods had evolved for implementing the disposal principles: (1) boost to a long-lived orbit, (2) destruct, and (3) controlled reentry. The value of each of the methods depended on the characteristics of the reactor.

Boost to a sufficiently long-lived orbit would allow fission products to decay until no reentry hazard exists. If a reliable end-of-life shutdown device were used, orbit liftimes as short as 50 yr could be adequate. For shorter-lived orbits, reentry radiation hazards are functions of the reactor power history, shutdown mechanism, and orbit lifetime.

The effect of solar flux on orbital decay was studied as functions of shutdown date and ballistic coefficient. The solar flux was found to have a period of about 11.4 yr, during which the magnitude varied by about a factor of 4. An expression for the incremental time for a vehicle to sink from a given altitude to another was derived as a function of solar flux density. A body tended to orbit at a nearly constant altitude during periods of low solar activity but to decay substantially as activity increased. Reentry usually occurred following periods of high solar activity. For a NASA space station, the reentry time varied from 1.2 to 8.0 yr depending on what year orbit stabilization was discontinued.

The hazards associated with the radiation field produced by large particles from the ablation of a SNAP-10A core reentering the atmosphere were evaluated. Fallout calculations were made using conservative assumptions. It was shown that the equivalent residual dose limit would not be exceeded unless the activity concentration was at least one core per square mile. It was concluded that reentry burnup would virtually eliminate SNAP-10A radiological hazards.

A study was directed toward the disposal problems associated with orbits of short duration (<50 yr). Combinations of disposal methods and system failures were considered. Eight combinations were examined in detail to determine the probability of overexposure. The estimated overexposures for these combinations are given in Table 5.

The study showed that reentry burnup, as a backup to orbital boost or controlled reentry, may provide a two-order-of-magnitude improvement over all other disposal methods. Development of an effective destruct device was recommended.

TABLE 5

COMPARISON OF DISPOSAL METHODS FOR REACTORS

Disposal Method	Average Number of Overexposures per Mission
1) Orbital boost to parking orbit with reentry burnup	0.0036
2) Orbital boost to parking orbit with immediate destruct and reentry burnup	0.001
3) Controlled reentry with intact impact in deep ocean	0.36
4) Controlled reentry with reentry disassembly and burnup and ocean impact	0.0046
5) Controlled reentry with destruct and reentry burnup and ocean impact	0.001
6) Controlled reentry with intact impact on land; recovery and storage	0.26
7) In-orbit destruct (random reentry) with reentry burnup	0.14
8) Random reentry with reentry burnup	0.36

5.0 REFERENCES

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