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SNAP
PROGRAMS
BIBLIOGRAPHY

AEC Research and Development Report

Atoms International Division
Rockwell International

P.O. Box 309
Canoga Park, California 91304

MASTER

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SNAP
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BIBLIOGRAPHY

R. R. EGGLESTON
Editor

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Rockwell International

P.O. Box 309
Canoga Park, California 91304

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I. INTRODUCTION

The SNAP program at Atomics International was active from January 1957 through June 1973. This bibliography was prepared to supplement previously prepared and published bibliographies.

Section II of this report contains the annotated bibliography of "pertinent" SNAP documents arranged by subject matter. This annotated bibliography does not duplicate the SNAP 10A program documents that were annotated in NAA-SR-Memo-12023. Additional SNAP 10A annotated bibliographies do occur in this section to complete the record of this program.

The test for pertinency for inclusion in the annotated bibliography was applied: (1) to all topical reports prepared for the entire program period, (2) to all NAA-SR-Memo's issued since January 1, 1966, and (3) to the AI reports (i. e. , those reports prepared with company funds). Also included are selected engineering supporting documents that were selected by the project management staff as containing pertinent information.

Section III contains the SNAP Program Progress Reports arranged by project. The reports are listed chronologically according to the report period for each of the programs.

Section IV is a bibliography of all SNAP reports listed numerically -- topical reports, memo's, TDR's, and AI reports that have been issued or were not included in NAA-SR-TDR-8301, Rev 3, and NAA-SR-TDR-8301, Supplement 1 (to Revision 3).

II. ANNOTATED BIBLIOGRAPHY – SUBJECT INDEX

A. GENERAL

1. Program, General

NAA-SR-11685, Vol I

TITLE: SNAP SYSTEMS CAPABILITIES, VOLUME I, STUDY SUMMARY

AUTHOR: Staff 12-1-65

ABSTRACT: This report is a summary of a document which compiles nuclear reactor space power system capabilities. It includes the reactor, shielding, power conversion system, and, finally, the complete systems and their variations.

NAA-SR-Memo-11916, Parts I and II

TITLE: PRELIMINARY FY 1967 SNAP PROGRAM REVIEW, WASHINGTON, D.C.,
APRIL 13-14, 1966

AUTHOR: Staff 4-22-66

ABSTRACT: A briefing book of accomplishments, status, and future plans of the SNAP programs is presented. Included are the Mercury-Rankine, Advanced TE, Aerospace Safety, SNAP 8 Reactor, SNAP Reactor Improvement, and General Supporting Technology Programs.

NAA-SR-Memo-12207

TITLE: SNAP PROGRAM REVIEW AT ATOMICS INTERNATIONAL, SEPTEMBER 1966

AUTHOR: Staff 5-31-66

ABSTRACT: This report is a briefing book for the presentation of the accomplishments, status, and future plans of the SNAP program. Included are the SNAP 8, General Supporting Technology, Aerospace Safety, Advanced Thermoelectric, and Advanced Systems Programs.

NAA-SR-Memo-12476

TITLE: SNAP PROGRAM REVIEW PRESENTATION TO THE AEC, APRIL 1967

AUTHOR: Staff 5-24-67

ABSTRACT: This report is a briefing book for the presentation of the SNAP program organizations, historical development, and achievements. The specifics on the technical status and future plans for the SNAP 8, Aerospace Safety, Advanced Thermoelectric, and Advanced Reactor Programs are included.

AI-AEC-13066

NAA-SR-Memo-12522, Volume I and II

TITLE: ADVANCED SNAP REACTOR CAPABILITY AND PROGRAM STUDY

AUTHOR: Staff

10-20-67

ABSTRACT: This report presents the results of a study that was made to determine the performance capabilities and development requirements for several, alternate, flight-rated reactor subsystem designs based on the SNAP 8, zirconium-hydride reactor concept.

The report is resented in two volumes. Volume I contains a summary of the projected mission and system requirements and the technical evaluation of the reactor performance capabilities and limits. The development program descriptions, costs, and schedules are presented and summarized in Volume II.

NAA-SR-Memo-12545

TITLE: SNAP REACTOR REVIEW BRIEFING AT ATOMICS INTERNATIONAL,
JUNE-JULY 1967

AUTHOR: Staff

9-25-67

ABSTRACT: This report is a briefing book on the accomplishments, status, and future plans of the SNAP 8 and advanced system programs.

AI-AEC-Memo-12902

TITLE: NUCLEAR SPACE POWER PROGRAM PRESENTATION, NASA-MSFC,
NOVENBER 4, 1969

AUTHOR: Staff

11-4-69

ABSTRACT: This report is a briefing book describing the SNAP system, program objectives, status, test results, and future plans.

AI-AEC-13067

TITLE: SNAP PROGRAMS SUMMARY REPORT

AUTHORS: D. Staub et al.

6-73

ABSTRACT: This report is a summary of the overall SNAP Reactor Programs, covering the period from 1951 through 1973. The following programs are reviewed: SNAP 2, SNAP 4, SNAP 8, SNAP 10A, 5-kwe Program, and the SNAP Reactor Aerospace Safety Program.

AI-Memo-7585

TITLE: NUCLEAR SPACE POWER SYSTEMS

AUTHOR: H. M. Dieckamp

9-64

ABSTRACT: This document is a primer for space nuclear power systems. It discusses the basic nuclear physics, heat transfer and power conversion systems, shielding, vehicle integration, meteroid protection, safety, and factory-to-flight sequence. The history of the nuclear space power program, future direction, applications, competitive position, and the status of the SNAP Hydride Reactor design are presented.

AI-AEC-13066

2. Reliability

NAA-SR-7355

TITLE: THERMOELECTRIC CONVERSION FOR ADVANCED SNAP SYSTEMS

AUTHOR: C. J. Morse

12-15-62

ABSTRACT: The use of thermoelectric conversion in advanced nuclear space power systems is investigated. A converter reliability analysis is presented, and a discussion of applications for thermoelectric systems is given.

AI-AEC-13098

TITLE: SYSTEM STATISTICAL RELIABILITY MODEL AND ANALYSIS

AUTHORS: V. S. Kekach and H. Rood

5-73

ABSTRACT: A digital computer code was developed to simulate the time-dependent behavior of the 5-kwe Reactor Thermoelectric System. The code was used to determine life-time sensitivity coefficients for a number of system design parameters, such as thermoelectric module efficiency and degradation rate, radiator absorptivity and emissivity, fuel element barrier defect constant, beginning-of-life reactivity, etc. A probability distribution (mean and standard deviation) was estimated for each of these design parameters. Then, error analysis was used to obtain a probability distribution for the system lifetime (mean = 7.7 yr, standard deviation = 1.1 yr). From this, the probability that the system will achieve the design goal of 5 yr lifetime is 0.993. This value represents an estimate of the degradation reliability of the system.

RAR-696-13-001

TITLE: RELIABILITY ANALYSIS OF ZrH REACTOR THERMOELECTRIC SYSTEM

AUTHOR: D. S. Burgess

5-29-68

ABSTRACT: The reliability of the ZrH reactor thermoelectric system is determined on the basis of reasonable attainable component and subsystem estimates. The system reliability is presented for full and partial power and for both 10,000- and 20,000-hr missions.

3. Materials Data

NAA-SR-2133

TITLE: THE GENERAL FEATURES OF THE ZIRCONIUM RICH CORNER OF THE ZIRCONIUM-URANIUM-YTTERIUM PHASE DIAGRAM

AUTHOR: J. C. Bokros

10-15-57

ABSTRACT: The general features of the zirconium-uranium-ytterium phase diagram are described.

NAA-SR-4898

TITLE: THE PERMEATION OF HYDROGEN THROUGH HASTELLOY B

AUTHORS: D. W. Rudd, D. W. Vose, and J. B. Vertrano

2-15-61

ABSTRACT: The flux of hydrogen gas through Hastelloy B alloy at temperatures from 640 to 820°C and pressures from 1 to 2 atm was determined as a function of alloy thickness, pressure differential, and temperature.

AI-AEC-13066

NAA-SR-5066

TITLE: THERMAL DIFFUSION OF HYDROGEN IN NONSTOICHIOMETRIC ZIRCONIUM-DEHYDRIDE

AUTHOR: A. W. Sommer and W. F. Dennison 10-1-60

ABSTRACT: Measurements of the Ludwig-Soret effect in delta-phase ZrHx in the temperature range of 1000 to 1300°F have been made. The "heat of transport," Q^* , of hydrogen in ZrHx was determined to be 1.3 ± 0.4 kcal/mole. A mathematical analysis of the kinetics of hydrogen redistribution is presented and shown to compare favorably with experimentally observed results. A similarity in the magnitude of Q^* and ΔH , enthalpy of solution, was noted and a rationalization of this similarity is presented.

NAA-SR-5413

TITLE: COMPATIBILITY OF SNAP FUEL AND CLAD MATERIALS

AUTHOR: J. K. Balkwill 10-15-60

ABSTRACT: Sample buttons of materials of interest for SNAP fuel elements were held at temperatures up to 1600°F in hydrogen under a 20-psi load for 200+ hr. The hydrided Zr-U alloys formed significant diffusion zones with Hastelloy B, Inconel X, Type 347 stainless steel, gold, and copper at 1200°F, but did not alloy with molybdenum, beryllium, or nickel. At 1600°F, diffusion was also found between the fuel, molybdenum, and nickel. The fuel did not react with Solaramic S-1435A glass at temperatures of 1200 and 1500°F. At 1300°F, beryllium formed a diffusion zone with Hastelloy B and Inconel X and also reacted with Solaramic, while BeO did not react with any of these.

NAA-SR-6264

TITLE: THEORETICAL KINETICS AND STRESSES FOR DELTA-PHASE ISOTHERMAL HYDRIDING OF URANIUM-ZIRCONIUM

AUTHOR: M. E. Nathan 7-15-62

ABSTRACT: The diffusion equation for isothermal single-phase hydriding of infinitely long cylinders is solved to investigate the theoretical kinetics and stresses of delta-phase by hydriding of a Zr-10U alloy. Results are correlated to indicate the effect of temperature, time, and radial hydrogen concentration gradients.

NAA-SR-6311

TITLE: THE SNAP 2 POWER CONVERSION SYSTEM TOPICAL REPORT No. 19
MERCURY MATERIALS EVALUATION AND SELECTION

AUTHOR: J. F. Mejedlik 12-63

ABSTRACT: SNAP 2 is the designation for a 3-kw nuclear auxiliary power unit to be used in a satellite vehicle. The SNAP 2 system consists of a reactor heat source, a boiler, a condenser, a mercury rankine engine, and an alternator. The corrosion and subsequent mass transfer resulting from the use of mercury as the thermodynamic working fluid are important considerations in the selection of materials for the SNAP 2 system. Consequently, corrosion and mass transfer behavior have been under study for the past 4 years. This report presents the results of this study for fiscal year 1962. The program was performed under a subcontract to Atomics International as part of the Atomic Energy Commission Contract AT(11-1)-GEN-8.

NAA-SR-6316

TITLE: SNAP-2 POWER CONVERSION SYSTEM TOPICAL REPORT No. 24,
MERCURY MATERIALS EVALUATION AND SELECTION

AUTHOR: Staff -63

ABSTRACT: A large number of metals and braze alloys were corrosion tested in mercury at 700 to 1400°F to determine the extent of material damage and mass transport of corrosion products. The observations suggest that judicious choice of an alloy may be made only for lower temperature use. At higher temperatures, differences in alloy behavior are small and the choice is between conventional alloys as a group and the refractory metals.

NAA-SR-6439

TITLE: SNAP 2 PRIMARY COOLANT DEVELOPMENT

AUTHOR: M. A. Perlow 7-15-61

ABSTRACT: Efforts in FY 1961 are described and summarized in the design development, testing, and selection of components integral to the primary coolant system for the SNAP 2 (Systems for Nuclear Auxiliary Power) reactor for space applications. Included are core hydraulic studies, heater development, corrosion studies, NaK hydride precipitation studies, and lithium hydride studies.

NAA-SR-6476

TITLE: BEARING MATERIALS COMPATIBILITY FOR SPACE NUCLEAR AUXILIARY
POWER SYSTEMS

AUTHOR: W. J. Kurzeka 9-1-61

ABSTRACT: The coefficients of friction for a large number of material combinations were measured in a 10^{-6} torr vacuum at 1000°F for 200 hr. Seven combinations were found to have friction coefficients less than 0.5. They are graphite vs Haynes 90, Stellite No. 3, and Al_2O_3 and Al_2O_3 vs Cr_2C_3 , TiC, and two Deva Metals Corp. alloys

NAA-SR-6732

TITLE: MECHANICAL STRENGTH OF HYDRIDES OF ZIRCONIUM AND ZIRCONIUM-BASE
ALLOYS

AUTHORS: J. W. Raymond and S. M. Tog 4-15-62

ABSTRACT: The compressive strength and elevated temperature transverse rupture strength of zirconium and zirconium-base alloy hydrides of compositions 2.2 wt % Nb, 3.9 wt % La, 1.4 wt % Cr, and 7 and 10 wt % U were measured. Alloying essentially doubled the transverse rupture strength for approximately the same hydrogen content, probably due to a grain refining effect. The strength of Zr-10 wt % U hydride increased with temperature to at least 1600°F and with increasing hydrogen content. Above 1.75 H/M a significant increase in compressive strength over zirconium hydride was found for all alloy additions.

NAA-SR-7529

TITLE: EFFECTS OF VIBRATION AND SHOCK ON HYDROGEN PERMEATION ON
SNAP 2/10A FUEL ELEMENTS

AUTHOR: A. J. Fitzgerad 8-1-63

ABSTRACT: The hydrogen retention qualities of SNAP 2/10A development fuel elements was determined before and after being subjected to simulated launch vibration and shock conditions. A method of normalizing the data to correct for variations in hydrogen content and test temperature profiles is presented.

NAA-SR-8533

TITLE: STABILITY OF A LIGHT METAL HYDRIDE

AUTHOR: J. B. Vetrano 8-1-63

ABSTRACT: The thermal stability of beryllium hydride, BeH_2 , was investigated. After a short induction period, the compound was found to decompose irreversibly at 250°C . The kinetics of decomposition fit a second order autocatalytic rate expression.

NAA-SR-8617, VOL. I

TITLE: SNAP TECHNOLOGY HANDBOOK, VOLUME I, LIQUID METALS

AUTHOR: G. F. Burdi 8-1-64

ABSTRACT: Important excerpts from commonly used references along with recent development have been arranged for the convenience of the design engineer. Data selected for presentation has passed the test of being in reasonable agreement with the majority of published work. The liquid metals covered by this handbook are mercury, sodium, sodium-potassium, potassium, rubidium, and lithium.

NAA-SR-8617, VOL. I, ADD. 1

TITLE: SNAP TECHNOLOGY HANDBOOK, VOL. I, ADD. 1, LIQUID METALS

AUTHOR: G. F. Burdi 5-30-64

ABSTRACT: Heat and mass transfer in liquid metals are discussed. Mass transfer and corrosion data for Hf, K, and Li are presented. This report contains certain data omitted from the main report, Vol. I.

NAA-SR-8617, VOL. II

TITLE: SNAP TECHNOLOGY HANDBOOK, VOL. II, HYDRIDE FUEL AND CLADDINGS

AUTHOR: G. F. Burdi 11-15-64

ABSTRACT: Physical, chemical, mechanical, and irradiation properties of Zr and Y hydrides, Hastelloy N, Hastelloy X, R-235, René 41, TD Nickel and T 2M are compiled. The design, fabrication, and performance of various fuel elements for the SNAP 2, 10A, and 8 reactors are presented.

AI-AEC-13066

NAA-SR-8617, VOL. III

TITLE: SNAP TECHNOLOGY HANDBOOK, VOLUME III, REFRACTORY FUELS AND CLADDING

AUTHOR: G. F. Burdi

4-30-65

ABSTRACT: This report contains the then currently available properties data for ceramic (UC, UN, and UO₂), ceramic dispersion (Al₂O₃-UO₂ and BeO-UO₂) and cermet (Mo-UC, -UN, and -UO₂) fuels. Also included are data for the refractory cladding materials (Cb, Ir, Mo, Re, Ta, Ti, V, and W).

NAA-SR-9295

TITLE: HYDROGEN DISSOCIATION PRESSURES OF MODIFIED SNAP FUELS

AUTHOR: H. E. Johnson

3-15-64

ABSTRACT: Hydrogen dissociation pressures for modified zirconium - 10 wt % uranium alloys are presented for H/Zr ratios from 1.4 to 2.0. Dissociation pressures for the modified material were found to be predictably higher than that for the unmodified hydride.

NAA-SR-9374

TITLE: EQUILIBRIUM DISSOCIATION PRESSURE OF THE DELTA AND EPSILON PHASES IN THE ZIRCONIUM-HYDROGEN SYSTEM

AUTHOR: J. W. Raymond

5-31-64

ABSTRACT: Pressure-temperature isochores were obtained for zirconium alloys of composition 1.430 to 1.910 H/Zr ratio in the temperature range of 300 to 900°C within pressure limits of 0.01 to 10.0 atm. An expression relating equilibrium dissociation pressure, temperature, and composition was determined, as was the heat of solution function.

NAA-SR-9400, VOL I

TITLE: LITHIUM HYDRIDE TECHNOLOGY, VOL I, PROPERTIES OF LITHIUM HYDRIDE AND CORROSION STUDIES

AUTHORS: F. H. Welch, E. C. Philips, J. G. Asquith, and N. F. Davies

5-8-64

ABSTRACT: This first volume presents experimentally determined properties and behaviors of various forms of LiH, which are of interest for LiH neutron shield. The physio-chemical properties reported include thermal expansion, thermal conductivity, dissociation rate in vacuum, dissociation pressures, outgassing, sinterability, and reactions with Gd and Gd₂O₃, liquid oxygen, and silicon. No deleterious effects were reported on the 304, 316, 347, and 19-9D1 containers for cast LiH after 4,000 hr exposure at 900°F, 1200°F, and 1400°F. From the somewhat contradictory results of limited irradiation experiments, the authors conclude that LiH can be used for neutron shields without any difficulty up to total dose of approximately 10¹⁸ nvt thermal neutrons. Further closely controlled in-pile experiments are recommended to verify LiH use in shields at higher doses. Any irradiation effects are concluded to be significantly reduced by the use of high-purity Li⁷H.

NAA-SR-9525, VOL I AND II

TITLE: BIBLIOGRAPHICAL SURVEY OF SNAP FUEL MATERIALS - ZIRCONIUM-URANIUM HYDRIDE AND RELATED ZIRCONIUM AND URANIUM SYSTEMS, VOLS I AND II

AUTHORS: A. W. Barsell, L. D. Montgomery, and J. T. Roberts 7-15-64

ABSTRACT: A bibliography of SNAP fuel materials papers.

NAA-SR-9618

TITLE: SNAP 8 EXPERIMENTAL REACTOR (S8ER) HYDROGEN BARRIER COATING DEVELOPMENT

AUTHOR: R. W. Webb 6-15-64

ABSTRACT: The development and evaluation of barium-silicate ceramic coating for use on the S8ER cladding as a hydrogen barrier is presented. Coating requirements, compositions, procedures, and test results are included.

NAA-SR-9643

TITLE: SNAP REACTOR MATERIALS DEVELOPMENT SELF-WELD STUDIES

AUTHOR: L. G. Kellogg 6-1-64

ABSTRACT: Self-welding tests were conducted on 110 material combinations held at 700 to 1300°F for times to 3400 hr in vacuums of less than 10^{-7} torr. The tests are classified into five basic groups: metal vs metal, metal vs ceramic, metal vs graphite, ceramic vs ceramic, and ceramic vs graphite. A sixth group consisted of a contaminant film between the contacting surfaces of the combinations from the other groups. Adhesion was noted for some combinations in all groups tested. The data indicate that to minimize adhesion some contaminant film should be placed between the two surfaces.

NAA-SR-9644

TITLE: SNAP REACTOR MATERIAL DEVELOPMENT, ULTRAHIGH VACUUM FRICTION STUDIES

AUTHOR: L. G. Kellogg 6-2-64

ABSTRACT: Studies conducted to evaluate various potential material combinations for bearing and braking surfaces for components to be operated at high temperature ($\sim 1000^\circ\text{F}$) in ultrahigh vacuum environment are described. The material combinations evaluated are categorized as metal vs metal (with and without dry film lubricants), metal vs ceramic, metal vs carbon, ceramic vs ceramic (with and without dry film lubricants), and ceramic vs carbon. It is concluded that ceramic vs carbon-graphite ($\sim 40\%$ graphite) provides the optimum friction characteristics for bearings ($\mu \leq 0.4$). The addition of dry film lubricant, particularly MoS_2 , reduces the friction coefficient as long as the lubricant remains at the bearing surfaces. For braking purpose the combination of metal vs carbon provides the most uniform friction coefficient (0.4 - 0.8).

NAA-SR-9672

TITLE: BERYLLIUM OXIDATION RESEARCH RELATIVE TO THE SNAP 8 REACTOR

AUTHORS: G. Ervin, Jr. and T. L. Mackay 7-24-64

ABSTRACT: Oxidation of beryllium in the temperature range 500 to 900°C proceeds slowly at a parabolic rate during an initial protective period, but proceeds catastrophically during a subsequent accelerating period. The period of stability before catastrophic failure was shortened significantly by the presence of moisture; on the other hand, the presence of 1% calcium in the beryllium provided a high degree of stability. The cause of the catastrophic failure is attributed to pits that grow in the metal under the protective film of adherent oxides.

AI-AEC-13066

NAA-SR-9673

TITLE: FISSION PRODUCT RELEASE FROM IRRADIATED URANIUM-ZIRCONIUM HYDRIDE SNAP FUEL

AUTHOR: W. M. Cegelski 7-15-64

ABSTRACT: Samples of uranium-zirconium hydride fuel (4 in. long by 1-1/4 in. diameter) were irradiated in the MTR and subsequently heated to 2000 to 3500°F for periods of a few minutes to 8 hr. The fission product release data obtained show steadily decreasing release rates for periods of several hours. Also, 100% release of volatile fission products only occurred when a clad sample was melted at 3500°F and then dropped to 2000°F for 15 minutes.

NAA-SR-9704

TITLE: MICROSTRUCTURE STUDIES OF SNAP FUELS

AUTHOR: H. Taketani 3-30-65

ABSTRACT: The microstructure of epsilon-phase zirconium - 10 wt % uranium hydride fuel was studied.

NAA-SR-9861

TITLE: HYDROGEN PERMEATION THROUGH SNAP 8 FUEL ELEMENT CLADDING ASSEMBLIES

AUTHOR: C. J. Engberg 9-1-64

ABSTRACT: A barium silicate enamel (AI-8763D) used as a hydrogen barrier on S8ER fuel elements is evaluated, and an empirical equation is presented expressing hydrogen flowrates through SNAP 8 cladding assemblies as a function of temperature and hydrogen dissociation pressure.

NAA-SR-9875

TITLE: PARAMETRIC STUDIES ON THE SHORT-TERM TENSILE MECHANICAL PROPERTIES OF Zr-10U ALLOY HYDRIDES

AUTHOR: L. B. Lundberg 7-15-64

ABSTRACT: The effects were studied on mechanical behavior of zirconium - 10 wt % uranium hydride resulting from variations in hydrogen content in the range of 1.45 to 1.94 effective H/Zr and carbon content in the range of 0.03 to 0.47% from room temperature to 1450°F.

NAA-SR-9908

TITLE: THE DEVELOPMENT AND QUALIFICATION OF THERMAL CONTROL COATING FOR SNAP SYSTEMS

AUTHOR: J. R. Crosby 9-30-65

ABSTRACT: The results of evaluation and testing conducted to select thermal control coatings for various components of the SNAP 10A and SNAP 2 systems are reported. Out of the hundreds of coatings and coating systems investigated, the following were selected or developed: (1) an aluminum-phosphate-bonded Cr-Co-Ni spinel with an overcoat of stannic oxide (AI-93) for the high thermal emittance/low solar absorptance ($\epsilon_{TH} > 0.90$, $\alpha_S < 0.30$) surfaces of the thermoelectric radiator and mercury radiator condenser; (2) commercial black enamel (PT-404, $\epsilon_{TH} > 0.90$) on surfaces requiring only high thermal emittance with plasma-applied Cr_2O_3 ($\epsilon_{TH} > 0.80$) as an alternate; (3) gold on molybdenum foil ($\epsilon_{TH} < 0.06$) on surfaces requiring low thermal emittance such as the core vessel and NaK piping; and (4) a pattern of 44% of the area covered with clear chemical film, 54% silicone paint and 2% white acrylic lacquer for a low thermal emittance/high solar absorptance coating system ($\epsilon_{TH} < 0.20$, $\alpha_S/\epsilon_{TH} > 1.6$) on the heat shield to retain the system heat during the launch and prior to the startup. Test data are presented to verify the stability of these coatings under the simulated system operational conditions, including exposure to irradiation.

AI-AEC-13066

NAA-SR-10027

TITLE: OBSERVATIONS ON THE PROPERTIES AND BEHAVIOR OF EARLY SNAP 8 HYDROGEN PERMEATION BARRIERS

AUTHOR: H. W. Carpenter 4-15-65

ABSTRACT: Small specimens of Hastelloy N tubing coated with Solaramic 14-35A or AI-8763D glass were aged at 1400 and 1500°F at 1 atm of hydrogen for 50 to 2000 hr. A number of observations were made regarding the properties of these coatings.

NAA-SR-10515

TITLE: FISSION GAS RELEASE FROM LIGHTLY IRRADIATED SNAP 8 FUEL MATERIALS

AUTHORS: A. H. Willis and A. J. Moses 9-15-65

ABSTRACT: SNAP 8 fuel materials were studied by low-level irradiation followed by annealing treatments. From these tests, it is concluded that the $\epsilon \rightarrow \delta$ phase transformation does not promote gas release, but the $\delta \rightarrow \beta$, $\alpha \rightarrow \beta$, and $\alpha \rightarrow \delta$ phase transformations cause massive gas release. When phase transformations do occur, sudden fission gas release can be anticipated for any reasonable burnup level. This suggests a swelling region at high burnup levels.

NAA-SR-10516

TITLE: THE DIFFUSION COEFFICIENT OF HYDROGEN IN DELTA-PHASE ZIRCONIUM HYDRIDE

AUTHORS: S. D. Harkness and W. A. Young 2-22-65

ABSTRACT: Diffusion coefficients of hydrogen in δ -phase zirconium - 10 wt % uranium hydride were determined for H/Zr ratios of 1.50 to 1.80 and for temperatures of 650 to 900°C, with an uncertainty of $\pm 25\%$. No concentration dependence was observed and neither carbon nor uranium have any effect on the diffusion of hydrogen in the δ -phase material.

NAA-SR-10851

TITLE: IRRADIATION STABILITY OF AI-8763D HYDROGEN BARRIER MATERIAL

AUTHOR: C. J. Engberg 4-15-65

ABSTRACT: Two S8ER cladding tubes coated with a ceramic, hydrogen-barrier coating (AI-8763D) were irradiated in the ETR to a peak integrated dose of 1.03×10^{21} nvt ($E_n > 1$ Mev) in 2337 hr at temperatures from 750 to 1250°F. The hydrogen permeation rates of these tubes did not change significantly, but devitrification of the coating may have been accelerated due to this treatment.

NAA-SR-10965

TITLE: PHASE RELATIONSHIPS AND MICROSTRUCTURES IN THE ZIRCONIUM - 10 wt% URANIUM ALLOY HYDRIDE SYSTEM

AUTHOR: J. W. Raymond 11-30-65

ABSTRACT: A microstructural survey was conducted in the zirconium - 10 wt% uranium - hydrogen ternary system over the composition range H/Zr ratios from 0 to 2. Structures were prepared under both equilibrium and nonequilibrium conditions. A series of photomicrograph demonstration phase relationships and microstructures in the system are presented and a revision of the Zr-H binary phase diagram is proposed.

AI-AEC-13066

NAA-SR-Memo-11048

TITLE SNAP 8 DEVELOPMENT REACTOR HYDROGEN BARRIER COATING DEVELOPMENT
AUTHOR R. W. Webb 11-4-66
ABSTRACT The development and evaluation of a barium-silicate ceramic coating for use as a hydrogen barrier on the S8DS cladding is presented. Coating requirements, compositions, procedures, and test results are included.

NAA-SR-11164

TITLE HYDROGEN IN SNAP SYSTEMS
AUTHOR D. F. Atkins (Editor) 4-30-65
ABSTRACT The effects of hydrogen on materials, components, and overall system performance under reactor service conditions is presented. Hydrogen permeation absorption and influence on strength and corrosion are considered. Hydrogen migration and accumulation do not appear to cause significant problems for any of the materials in the SNAP systems, with the possible exception of corrosion and wetting by mercury.

NAA-SR-11316

TITLE EFFECTS OF PROCESSING AND OPERATING PARAMETERS ON THE CREEP PROPERTIES OF SNAP REACTOR FUEL ELEMENT CLADDING
AUTHORS G. E. Culley and J. T. Berling 10-15-65
ABSTRACT A series of 1000-hr biaxial creep tests at 1400°F on SNAP 8 fuel element cladding tubes were conducted to determine the effects of various coatings, grain size, hydrogen permeation, stress, and flowing NaK on the creep characteristics of INOR 8 (Hastelloy N). No gross deviations from the basic data were observed.

NAA-SR-11519

TITLE IRRADIATION TEST OF SNAP 8 CANDIDATE SPRING MATERIALS
AUTHORS D. S. Cowgill, N. F. Davies, and W. J. Kurzeka 6-30-67
ABSTRACT This report describes the in-pile test conducted on mechanical compression springs formed from René 41 (Ni-Cr-Fe) and Stauffer 88 (88Ta-12W) materials. Various spring measurements are presented that were made prior to and after the 3250-hr experiment during which the springs were held at constant length in a 700 to 1300°F and 10⁻⁴ torr environment while being exposed to total dose of 2 x 10¹⁷ to 3 x 10¹⁹ nvt fast neutrons and 5 x 10⁹ to 7 x 10¹⁰ r gamma. Comparisons are made to characteristics of similar springs tested under similar temperature, vacuum, and nonnuclear environments. It is concluded that the irradiation decreases the force relaxation rate of the René 41 springs, while its effects on Stauffer 88 springs were not established. It is also concluded that René 41 is the preferred material up to 1200°F and Stauffer 88 for temperatures of 1200 to 1300°F.

NAA-SR-11959

TITLE THERMOPHYSICAL PROPERTIES OF SNAP FUEL MATERIALS
AUTHORS M. M. Nakata, C. J. Ambrose, and R. A. Finch 9-25-67
ABSTRACT Thermal diffusivity, heat capacity, thermal conductivity, and electrical resistivity measurements, made on SNAP fuel materials, are presented. Compositions investigated include the unhydrided zirconium - 10 wt % uranium alloy and the alloy hydrided to 0.50, 1.26, 1.58, 1.81, and 1.90 H/Zr atomic ratios. Anomalies observed in these properties-vs-temperature curves are discussed and related to the available phase diagrams.

AI-AEC-13066

NAA-SR-Memo-11966

TITLE: LEAST SQUARES DETERMINATION OF ISOTHERMAL RATE CONSTANTS FOR THE OXIDATION OF METALS

AUTHOR: W. A. Young 6-14-66

ABSTRACT: A least squares computer program is described which permits the solution of the rate expression, $(\Delta m + c)^n = kt$, for all combinations of the parameters c , n , and k . The program is written in FORTRAN IV and provides graphic display of data points and the least squares curve.

NAA-SR-12077

TITLE: ROOM TEMPERATURE COMPRESSIVE PROPERTIES OF Zr-10U ALLOY HYDRIDE AT HIGH HYDROGEN CONTENTS

AUTHORS: L. B. Lundberg and J. R. Berling 8-30-66

ABSTRACT: The room-temperature compressive strengths of Zr-10U alloy hydrided to H/Zr values from 0.5 to 1.90 were measured. An unexpected discovery was the release of hydrogen as a result of room-temperature compressive testing. The amount of hydrogen released seems to be a function of strain rate, hydride phase, and grain structure. The highest release observed was 0.03 wt % hydrogen. A mechanism for the release is proposed.

NAA-SR-Memo-12125

TITLE: DEVELOPMENT OF AI-1 CEMENT ENCAPSULANT

AUTHOR: W. G. Long 8-22-66

ABSTRACT: The development of an improved inorganic cement, AI-1, for wet-winding and encapsulating high-temperature electrical coils is described. Unlike the previously used proprietary cements, this phosphate-bonded, silica-base cement can be formulated from basic materials and is also less corrosive to copper conductors in its wet state. Test data are presented to show that the insulation resistance values at room and elevated temperatures are comparable to the previously utilized cements and well above the minimum required values. Limited life test data are also shown.

NAA-SR-Memo-12364, Rev A

TITLE: HYDROGEN RETENTION OF SNAP 8 FUEL ELEMENTS CONTAINING NICKEL-PLATED FUEL RODS

AUTHORS: W. F. Dennison and J. L. Nummelin 2-19-68

ABSTRACT: Fuel elements with nickel-plated fuel were found to have low hydrogen permeation rates as compared to nonplated fuel. Diffusion of the nickel into the fuel was studied for periods up to 10,000 hr at 1400°F. The maximum diffusion depth observed was 6 mils after 7104 hr at 1400°F.

NAA-SR-12607

TITLE: THERMOPHYSICAL PROPERTIES OF UNIRRADIATED SNAP FUELS

AUTHORS: W. A. Young, C. C. Weeks, M. M. Nakata, and C. A. Smith 2-23-68

ABSTRACT: Enthalpy and thermal diffusion measurements were made on zirconium-10 wt % uranium hydride samples with 1.58, 1.65, 1.70, and 1.81 H/Zr ratios from room temperature to 900°C. Specific heat and thermal conductivity were calculated from these data and polynomial expressions were obtained for each property.

AI-AEC-13066

NAA-SR-12449

TITLE ELECTRON MICROSCOPY ANALYSIS OF IRRADIATED SNAP 8 FUEL

AUTHOR K. J. Miller 8-20-67

ABSTRACT Irradiated fuel from the NAA 115-1 experiment was examined by electron microscopy replication techniques. Fuel microcracking was observed, but it was not excessive. Bubble densities were on the order of 10^{13} bubbles/cc with an average diameter of 900 Å. Fission gas bubble formation accounted for approximately 50% of the observed hydrogen-connected volume change. Bubble size and density appeared to depend on temperature but not burnup. There was no evidence of fission gas bubble sweeping during phase change, and no correlation of bubble sites with epsilon banding or microcracks. Bubble-depleted regions were found around large particles

NAA-SR-12587

TITLE PHASE RELATIONSHIPS AT HIGH HYDROGEN CONTENTS IN THE SNAP FUEL SYSTEM

AUTHORS K. E. Moore, W. A. Young, M. M. Nakata, and C. A. Smith 2-15-68

ABSTRACT Phase boundaries in the $\delta \rightleftharpoons \epsilon$ region of the zirconium-10 wt % uranium-hydrogen system have been defined. Experimental methods employed included high-temperature X-ray diffraction, electrical resistivity, dilatometry, and hot-stage microscopy. The $(\delta + \epsilon)$ two-phase field was found to exist at room temperature over the range of 1.64 to 1.74 H/Zr and to diminish in width with increasing temperature, forming a closed loop at $\sim 850^\circ\text{F}$ and 1.70 H/Zr ratio. From this point, a single boundary, sloping toward higher hydrogen contents, to at least 1280°F and 1.81 H/Zr ratio separates the δ and ϵ single-phase regions.

AI-AEC-12703

TITLE PHASE RELATIONSHIPS IN THE $(\alpha + \delta)$ REGION OF THE ZrH SYSTEM

AUTHORS K. E. Moore and M. M. Nakata 9-30-68

ABSTRACT The $(\alpha + \delta) \rightleftharpoons \delta$ phase boundary of the ZrH system has been defined on the basis of studies performed on zirconium hydrides containing 4 at. % U. The methods used include optical hot-stage microscopy and room-temperature metallography, dilatometry, and X-ray diffraction. This boundary was found to exist between about 1.33 H/Zr ratio at 1020°F and 1.60 H/Zr ratio at 75°F . The metastable phase, γ' , coexisting with α and δ phases, was found to transform to δ at about 560°F . The relative proportions of α and γ' phases at a particular H/Zr ratio are dependent on the size of the hydrided sample. The amount of γ' present is virtually independent of cooling rates.

AI-AEC-Memo-12782

TITLE CHROMIUM REDISTRIBUTION EVALUATION OF S8DR COATED TUBING

AUTHORS J. R. Miller and J. T. Roberts 2-5-69

ABSTRACT The extent of chromium diffusion from the surface of chromized and coated Hastelloy N tubing was determined after aging up to 10,000 hr at 1400 to 1600°F . Changes in the coating of substrate were also studied.

AI-AEC-13066

AI-AEC-13089

TITLE: SPACE NUCLEAR SYSTEM SHEATHED ELECTRICAL CABLES SUMMARY REPORT

AUTHOR: L. E. Donelan

6-73

ABSTRACT: The development of a metallic-sheathed electrical cable is described for use with the ZrH reactor power system. The positive protection to the conductor and the insulation, provided by the metallic sheath, could be expected to increase the cable performance reliability from that of the previously utilized braided-wire-covered cables. Material and design studies were conducted to select the reference cable configuration. The reference design selected consisted of a single stainless steel clad 0.040-in.-diameter conductor, encased within a 0.090-in.-diameter stainless steel sheath, with compacted MgO insulation between the conductor and the sheath. Methods of terminating the cable to the actuator and the limit switch were also selected. Preliminary tests showed that this type of cable will maintain the insulation resistance well above the minimum acceptable level under long-time exposure to the fabrication, system assembly, and operational environments and conditions.

AI-AEC-13102

TITLE: MATERIALS INTERACTIONS BETWEEN THE THERMOELECTRIC CONVERTER AND THE 5-kwe REACTOR SYSTEM

AUTHOR: P. B. Ferry

6-73

ABSTRACT: Integration of a compact thermoelectric converter with the 5-kwe reactor system requires introduction of Ta - 10 W piping into some portion of the primary coolant loop. The accompanying material interaction uncertainties could affect the present system design. These include degradation of the required austenitic - refractory metal transition joint during operation at high temperatures, loss of corrosion resistance, embrittlement by the presence of hydrogen, and loss of design margin by transport of interstitial elements. Analysis and limited experimental evidence indicates that these potential materials interactions can be adequately controlled. It appears that the Group Va refractory metal can be utilized without unacceptable adverse effect on system reliability.

TI-696-240-080

TITLE: STRAIN VS STRAIN RATE RELATIONSHIPS FOR INCOLOY 800, HASTELLOY X, AND HASTELLOY N

AUTHOR: N. F. Davies

11-18-70

ABSTRACT: Strain versus strain rate data for unirradiated and irradiated Incoloy 800, Hastelloy X, and Hastelloy N were collected and plotted. All readily available data, both biaxial and uniaxial, between 1200 and 1400°F were included regardless of chemistries, heat treatment, aging effects, etc. Several general conclusions were drawn from these data.

TI-696-26-038

TITLE: TEST REPORT DEVELOPMENT. EMITTANCE OF Ta-10W, EMITTANCE AND COATING ADHESION OF AN ALUMINIDE COATING ON A Ta-10W SUBSTRATE

AUTHOR: G. F. Queener

8-8-69

ABSTRACT: This document contains the appendices for Test Report AI-AEC-TDR-12876 on the emittance of Ta-10W and the emittance and adhesion of an aluminide coating on a Ta-10W substrate.

AI-AEC-13066

TI-696-26-046

TITLE EXAMINATION OF ZrH CONTROL DRUM MATERIAL COMPATIBILITY SAMPLES
AUTHOR C. C. Woolsey 2-18-70
ABSTRACT Examination of a compatibility stack revealed no problems between BeO and Ta-10W, BeO and TZM, BeO and Hastelloy N, BeO and Cb-752, Ta-10W and Cb-752, TZM and Cb-752. Severe reactions were found at interfaces between Hastelloy N and Ta-10W. Exposure was for 4000 hr at 1800°F with 160 hr at 2000°F.

TR-696-26-015

TITLE MAGNETIC MATERIALS TEST REPORT
AUTHOR L. Donelan 7-18-69
ABSTRACT Two magnetic materials, Hipercro 27 and Armco Magnetic Ingot Iron, were tested for their magnetic stability with time at temperature (1200°F). Both materials are magnetically stable with time. Hipercro 27 has better magnetic performance than Armco Magnetic Ingot Iron, both at high temperature and at high flux densities. Impact tests made indicate that Hipercro 27 is more brittle at room temperature, but less brittle at elevated temperature than Armco Magnetic Ingot Iron.

TI-759-260-001

TITLE STABILITY OF BERYLLIUM IN NaK
AUTHOR R. Van Houten 9-21-70
ABSTRACT The successful long-time use (30,000 to 40,000 hr) of Be in high-temperature NaK (nominal 1200°F) is problematical. Particular concerns are (1) the effects of even trace quantities of oxygen on Be corrosion, (2) Be erosion by classic mass transfer, and (3) possible negative consequences associated with the formation of brittle phases resulting from contact of Be (direct or by deposition) with many of the materials in the reactor/heat exchanger loop.

4. Radiation Damage and Hardening

NAA-SR-5141

TITLE A HEAT TRANSFER CONTROL DEVICE FOR IRRADIATION EXPERIMENTS
AUTHORS W. K. McCarty and R. C. Noyes 6-30-60
ABSTRACT A device which controls temperatures by varying the heat transfer across an annular space has been designed and tested. The central temperature of a nuclear fuel test specimen has been controlled at $1165 \pm 10^\circ\text{F}$ by use of this device. This mechanism makes possible temperature control under conditions where the heat source, heat sink, or both vary between 50 and 150% of the nominal value. Out-of-pile tests and three in-pile experiments have confirmed the usefulness of this heat transfer control device.

AI-AEC-13066

NAA-SR-9634

TITLE LOW FLUX RADIATION EFFECTS ON ELECTRICAL AND ELECTRONIC COMPONENTS

AUTHORS M. N. Robinson, S. G. Kimble, and D. M. Walker 12-1-64

ABSTRACT The methods and results of irradiation effects tests conducted on 5 types of transistors, 3 sizes of capacitors, several electrical insulation samples, and a SNAP 10A temperature switch are described. The performance of the electrical components measured periodically during the 11-day test at 140°F in a 10^{-4} torr environment which accumulated total dose of 10^{16} nvt fast neutrons and 4×10^8 r gamma are presented in series of computer-produced graphs. These data indicate that the H_{FE} and V_{CE} of the 2N1506A, 2N1711, SN101, 2N2865, and TIX 693 transistors are significantly affected in the 10^{12} to 10^{13} nvt range, tolerance of the 2.5 microfarad (29F155) capacitor was exceeded in the 10^{-2} to 10^{-3} nvt range, the capacitance of the 0.50 microfarad (M1-504) and of the 0.25 microfarad (M1-254) had decreased less than 15% at the end of the test, all 7 of the epoxy-type insulation samples changed in dimensions, became more brittle, and exhibited increase in electrical resistance at approximately 10^{14} nvt, and the performance of the SNAP 10A temperature switch was unaffected to approximately 10^{14} nvt.

NAA-SR-9684

TITLE LOW FLUX NUCLEAR RADIATION EFFECTS ON SEMICONDUCTOR DEVICES

AUTHOR M. N. Robinson 12-1-64

ABSTRACT Tests and results of subjecting 2 types of transistors (2N744 and 2N1072), diode (PS760), and zenor diode (1N2032) to low flux nuclear radiation are described. Multiple graphs showing the performance of the devices as a function of dose (up to 1×10^{16} fast neutrons and 4.5×10^8 r gamma) are presented. It was concluded that both diode and zenor diode are relatively unaffected by dose up to 10^{13} nvt while the threshold for transistors may be approximately 1 decade lower.

NAA-SR-10612

TITLE RADIATION DAMAGE IN SILICON NEUTRON SPECTRUM CONSIDERATIONS

AUTHOR M. N. Robinson 6-10-65

ABSTRACT A spectrum normalization scheme for radiation damage in silicon is proposed and applied to some representative test spectra. It is shown that significant differences may be predicted for data obtained in different environments

NAA-SR-11519

TITLE IRRADIATION TEST OF SNAP 8 CANDIDATE SPRING MATERIALS

AUTHORS D. S. Cowgill, N. F. Davies, and W. J. Kurzeka 6-30-67

ABSTRACT This report describes the in-pile test conducted on mechanical compression springs formed from René 41 (Ni-Cr-Fe) and Stauffer 88 (88Ta-12W) materials. Various spring measurements are presented that were made prior to and after the 3250-hr experiment during which the springs were held at constant length in a 700 to 1300°F and 10^{-4} torr environment while being exposed to total dose of 2×10^{17} to 3×10^{19} nvt fast neutrons and 5×10^9 to 7×10^{10} r gamma. Comparisons are made to characteristics of similar springs tested under similar temperature, vacuum, and nonnuclear environments. It is concluded that the irradiation decreases the force relaxation rate of the René 41 springs, while its effects on Stauffer 88 springs were not established. It is also concluded that René 41 is the preferred material up to 1200°F and Stauffer 88 for temperatures of 1200 to 1300°F.

AI-AEC-13066

NAA-SR-11924

TITLE: DEVELOPMENT IRRADIATION TEST OF SNAP 8 ELECTRICAL COMPONENTS

AUTHOR: N. F. Davies

7-15-66

ABSTRACT: In-pile testing of four types of electrical connectors, two types of electrical cables, and five electrical coil samples is described. Test data, including the insulation resistance, measured before, during, and/or after the 3000-hr test are presented. During this test, the components were exposed to irradiation doses of 10^{-7} to 10^{-9} nvt fast neutrons and 8×10^9 to 9×10^{10} r gamma in 900 to 1200°F and 10^{-6} torr environment. These data indicated that the insulation resistances of all components were essentially unaffected by the test. The insulation around the contact pins of one type connector was severely degraded. Visual appearance of all other components was excellent.

NAA-SR-12042

TITLE: IRRADIATION TEST OF SNAP 8 ACTUATOR, POSITION SENSOR, AND LIMIT SWITCH

AUTHORS: N. F. Davies, W. J. Kurzeka, and M. Warren

11-5-66

ABSTRACT: In reactor test in which two each of the SNAP 8 control drum actuator, position sensor, and limit switch were exposed to nuclear irradiation while being operated periodically in an 800 to 1000°F and 10^{-5} torr environment is described. Test data measured periodically during the 3,000-hr test wherein the total irradiation dose ranged from 1.0 to 8.9×10^{19} nvt fast neutrons and 0.6 to 1.5×10^{10} r gamma are presented in series of graphs. It is reported that all component performance remained stable or within design requirements. The reduction of insulation resistance, attributed to oil from the vacuum system, was corrected by controlled high-temperature oxidation.

TI-653-240-027, Rev C

TITLE: DEVELOPMENT OF A FORMULA FOR CALCULATING ZIRCONIUM HYDRIDE FUEL SWELLING

AUTHOR: W. J. Roberts

6-7-72

ABSTRACT: A model is developed to describe irradiation-induced growth in zirconium hydride or for heavy ion accelerator irradiation of structural materials. The model is based on the phenomenon of volume saturation occurring as a result of annealing of irradiation-produced dislocations. The model was proposed by Hudson, Mazey, and Nelson for swelling in Ni irradiated with C^{++} ions. The model predicts a strong temperature and weak flux sensitivity. A sensitivity is shown to lower temperature, higher power operation following an initial steady operation.

5. Computer Programs

NAA-SR-7357

TITLE: TYCHE, A MONTE CARLO SLOWING DOWN CODE

AUTHOR: R. A. Blaine

6-30-62

ABSTRACT: TYCHE is a Monte Carlo code designed to find the second, fourth, and sixth moments of the neutron slowing down density distribution in an infinite homogeneous medium. Analytic techniques are used to reduce the number of random variables and recursive relations to generate the moments. The code was written in the FORTRAN language for the IBM 7090 computer; typical running time is 5 to 10 minutes for 5000 neutron histories. Cross sections and anisotropic scattering coefficients for ten elements are attached to the deck as a binary library.

AI-AEC-13066

NAA-SR-9190

TITLE: THE SNAPKIN VI COMPUTER PROGRAM FOR SNAP REACTOR KINETICS CALCULATIONS

AUTHOR: W. A. Rhoades

4-15-64

ABSTRACT: A computer program is described which solves the space-independent reactor kinetics equations with feedback equations representing the principal shutdown mechanisms in a SNAP reactor. Spatially dependent feedback effects may be considered. Printed, CRT, and punchcard output may be obtained optionally. Sample problems are demonstrated for both versions of the code.

NAA-SR-9233

TITLE: QUIKIE - A COMPUTER PROGRAM FOR SPATIALLY INDEPENDENT MULTIGROUP SLOWING DOWN AND THERMALIZATION CALCULATIONS

AUTHORS: M. Boling and W. Rhoades

4-15-64

ABSTRACT: In the case of an infinite homogeneous medium or a finite medium, QUIKIE code calculates the neutron energy spectrum, multiplication factor, and total reaction rates. It also calculates few-group cross sections and punches libraries for the spatially dependent diffusion and transport codes.

AI-AEC-13065

TITLE: GEOM - A THERMAL-HYDRAULIC STRESS CODE

AUTHOR: E. Moody

5-10-73

ABSTRACT: This report documents the GEOM computer code (IBM 360) as it existed at the close-out of the SNAP program. The code performs thermal, hydraulic, and stress analysis on a liquid metal ZrH reactor core. The theory and empirical data used are described in detail. Instructions for using the code are given with a sample problem for illustration.

AI-AEC-13076, Vol I

TITLE: SELECTED COMPUTER CODES AND LIBRARIES, VOLUME I, ZIP - A TIMESHARE PROGRAM FOR SNAP REACTOR PARAMETRIC STUDIES

AUTHOR: H. Rood

6-73

ABSTRACT: ZIP is a FORTRAN program written for the Honeywell 440 timeshare computer. It is a parametric survey code which applies specifically to SNAP-type nuclear reactors; i. e., compact U-ZrH-fueled, beryllium-reflected, NaK-cooled reactors. As input, the code requires information about the mission requirements (power level, operating lifetime, coolant outlet temperature, etc.) and sufficient dimensional information to describe the reactor (fuel length, cladding thickness, reflector thickness, etc.). Then, using simple equations fit to results obtained using more complex codes, ZIP determines all the significant nuclear and thermal performance characteristics of the reactor. Also included is a shield thickness calculation, and a calculation of the reactor and shield weights.

AI-AEC-13066

AI-AEC-13076, Vol II

TITLE: SELECTED COMPUTER CODES AND LIBRARIES, VOLUME II, APC - A MULTI-GROUP TRANSPORT BUCKLING ITERATION CODE

AUTHOR: D. S. Bost 5-2-73

ABSTRACT: APC and APC II are both buckling (B^2) or leakage (DB^2) iteration codes for reactor calculations, based on the one-dimensional transport theory code, ANISN. In APC, a uniform composition (single material) must be specified for all core zones. In APC II, the composition in core zones may be different. Options for flux and importance weighting of the different compositions are provided. In both codes, two complete ANISN problems are specified. They are processed alternately, transferring B^2 or DB^2 from each to the other until the results are mutually consistent. A description of the input data, sample problems, and FORTRAN listing are given for both codes.

AI-AEC-13076, Vol III

TITLE: SELECTED COMPUTER CODES AND LIBRARIES, VOLUME III, COMPOZ - A MULTIGROUP TRANSPORT PERTURBATION CODE

AUTHOR: V. S. Lekach 6-73

ABSTRACT: COMPOZ is a transport theory perturbation code designed to be used with the ANISN-based multigroup transport code, APC. The code calculates the reactivity worth of cross section sets included in the APC calculation, or sets included in the APC library.

AI-AEC-13076, Vol IV

TITLE: SELECTED COMPUTER CODES AND LIBRARIES, VOLUME IV, A 28-GROUP CROSS SECTION LIBRARY FOR SNAP REACTOR ANALYSIS

AUTHOR: J. M. Otter 5-31-73

ABSTRACT: A library of 28-group neutron cross sections designed for SNAP reactor analyses is described. Sources of microscopic cross section data, flux weighting spectra, and data processing methods are given. A listing of the library in the ANISN code format is included.

AI-AEC-13076, Vol V

TITLE: SELECTED COMPUTER CODES AND LIBRARIES, VOLUME V, TEMPRO - A HEAT TRANSFER-LIFETIME CODE FOR SNAP REACTOR ANALYSIS

AUTHORS: L. D. Felten and E. Moody 6-73

ABSTRACT: The TEMPRO computer program does heat transfer and reactivity calculations as a function of time for SNAP-type reactors. It determines hydrogen loss rates and the associated reactivity loss. Fuel swelling, uranium burnup, and fission product buildup are considered.

AI-AEC-13094

TITLE: VOLUME ACCUMULATOR DESIGN ANALYSIS COMPUTER CODES

AUTHORS: W. D. Whitaker and T. T. Shimazaki

6-73

ABSTRACT: The computer codes, VANEP and VANES, were written and used to aid in the design and performance calculation of the Volume Accumulator Units (VAU's) for the 5-kwe Reactor Thermoelectric System. VANEP computes the VAU design which meets the primary coolant loop VAU volume and pressure performance requirements. VANES computes the performance of the VAU design, determined from the VANEP code, at the conditions of the secondary coolant loop. The codes can also compute the performance characteristics of the VAU's under conditions of possible modes of failure which still permit continued system operation.

AI-AEC-13095

TITLE: THERMOELECTRIC PUMP PERFORMANCE ANALYSIS COMPUTER CODE

AUTHOR: J. L. Johnson

5-73

ABSTRACT: This report presents the computer program that has been used to analyze and design dual-throat electromagnetic dc conduction pumps for the 5-kwe ZrH Reactor Thermoelectric System. In addition to a listing of the code and corresponding identification of symbols, a discussion of the bases for this analytical model is provided.

AI-AEC-13066

B. SYSTEMS

1. Systems Design and Analysis

NAA-SR-5613

TITLE: SNAP-2 SYSTEM STUDIES

AUTHOR: R. D. Welsh

10-15-61

ABSTRACT: A study was made of the SNAP 2 nose cone configuration design point for reliability and minimum weight. Included in the study were: steady-state, full-power, and orbital operation. Brief consideration was given to a cylindrical SNAP 2 configuration. Also considered were thermoelectric, canned-rotor, and EM NaK pumps instead of the shaft-driven permanent magnet configuration.

NAA-SR-6312

TITLE: SNAP 2 POWER CONVERSION SYSTEM TOPICAL REPORT No. 20
CRU V DESIGN

AUTHOR: Staff - TRW

5-31-63

ABSTRACT: CRU V design and development was undertaken to provide a SNAP 2 turbo-alternator package that will meet SNAPSHOT flight objectives. This report describes the selection of the design concept, CRU detail design features, performance estimates, and the plumbing module design. The layout design fulfills the majority of the specific CRU V design objectives. Potential problems are discussed but they are not considered to jeopardize accomplishment of current SNAP 2 program objectives.

NAA-SR-6735

TITLE: THE PRELIMINARY DESIGN OF A 150 kw SNAP REACTOR

AUTHOR: Staff

5-15-62

ABSTRACT: This report documents the thoughts and analysis originating from Aerojet-General Corporation regarding the preliminary design of a 150 kwe SNAP reactor.

NAA-SR-9056

TITLE: NUCLEAR MERCURY RANKINE POWERPLANT (3 to 20 kwe) CAPABILITY

AUTHOR: R. L. Wallerstedt

6-15-64

ABSTRACT: The design characteristics and advanced capabilities of a 3-to 20-kwe nuclear mercury-rankine power plant (using a SNAP 2 or 8 reactor) are summarized and discussed. Program objectives and development status are presented. Reliability criteria are outlined.

NAA-SR-11685, Vol I

TITLE: SNAP SYSTEMS CAPABILITIES, VOLUME I, STUDY SUMMARY

AUTHOR: Staff

12-1-65

ABSTRACT: This report is a summary of a document which compiles nuclear reactor space power system capabilities. It includes the reactor, shielding, power conversion system, and, finally, the complete systems and their variations.

NAA-SR-11935

TITLE: SNAP 10A SYSTEM OPTIMIZATION

AUTHOR: P. L. Soske (Editor)

8-1-66

ABSTRACT: Since the configuration of the SNAPSHOT-1 system was established, many technological advances have been made in related programs. These programs include the SNAP 2 and SNAP 8 Reactor Development Programs, the SNAP Reactor Improvement Program, and the Advanced Thermoelectric Development Program. These advances in the technology and the results of the experience gained in the SNAP 10A Program are reviewed and their applications projected to SNAP 10A and 10A-type direct-radiating thermoelectric power systems.

NAA-SR-11984

TITLE: DESIGN STUDY OF SNAP 8 COMPACT CONVERTER

AUTHOR: AI Staff

6-30-66

ABSTRACT: The detailed parametric analysis and design data for potential compact nuclear-powered thermoelectric power systems are presented. The systems studied include unmanned shadow-shielded configurations for 5- and 15-kwe power and manned shadow and 4-Pi-shielded configurations with 15-kwe power. Five reactors (S8DS, Man-Rated SNAP 8, SNAP 8 Universal, SNAP 8 Low Power, and SNAP 10B) and three thermoelectric converter systems (RCA Compact Converter, Westinghouse Compact, and RCA/AI Directing Radiating) in various combinations are evaluated in this study. The lifetime (2.5 to 5 yr), system weight, system configuration and performance, and operational data are presented for the resulting potential systems.

AI-AEC-Memo-12717

TITLE: REFERENCE ZIRCONIUM HYDRIDE REACTOR THERMOELECTRIC SYSTEM

AUTHOR: Staff

6-15-69

ABSTRACT: This report presents the results of a 9-month engineering study of nuclear reactor-thermoelectric (TE) power systems. This general type of space power system, which is similar to the 500-watt (SNAP 10A) reactor-TE system, was flight-tested in space in 1965. Additional objectives set for this study were to:

- 1) Establish a conceptual design of a reference system suitable for adaption to either manned or unmanned space missions with a nominal power level of 25 kwe and a minimum lifetime of 20,000 hr.
- 2) Prepare parametric performance data over a suitable power range above and below 25 kwe to illustrate the trade-offs available between power, weight, area, temperature, etc.

AI-AEC-13090

TITLE: SPACE NUCLEAR SYSTEM VOLUME ACCUMULATOR DEVELOPMENT SUMMARY REPORT

AUTHOR: W. D. Whitaker

6-73

ABSTRACT: The engineering, design, and fabrication status of the volume accumulator units (VAU's) to be employed in the NaK primary and secondary coolant loops of the 5-kwe Reactor Thermoelectric System are described. Three identical VAU's are required - two for the primary coolant loop, and one for the secondary coolant loop. The VAU's utilize nested-formed bellows as the flexing member, are hermetically sealed, provide double containment, and utilize a combination of gas-pressure force and bellows-spring force to obtain the desired pressure regulation of the coolant loops. All parts of the VAU, except the NaK inlet tube, are to be fabricated from Inconel 718.

AI-AEC-13066

AI-AEC-13094

TITLE: VOLUME ACCUMULATOR DESIGN ANALYSIS COMPUTER CODES

AUTHORS: W. D. Whitaker and T. T. Shimazaki

6-73

ABSTRACT: The computer codes, VANEP and VANES, were written and used to aid in the design and performance calculation of the Volume Accumulator Units (VAU's) for the 5-kwe Reactor Thermoelectric System. VANEP computes the VAU design which meets the primary coolant loop VAU volume and pressure performance requirements. VANES computes the performance of the VAU design, determined from the VANEP code, at the conditions of the secondary coolant loop. The codes can also compute the performance characteristics of the VAU's under conditions of possible modes of failure which still permit continued system operation.

AI-AEC-13098

TITLE: SYSTEM STATISTICAL RELIABILITY MODEL AND ANALYSIS

AUTHORS: V. S. Kekach and H. Rood

5-73

ABSTRACT: A digital computer code was developed to simulate the time-dependent behavior of the 5-kwe Reactor Thermoelectric System. The code was used to determine life-time sensitivity coefficients for a number of system design parameters, such as thermoelectric module efficiency and degradation rate, radiator absorptivity and emissivity, fuel element barrier defect constant, beginning-of-life reactivity, etc. A probability distribution (mean and standard deviation) was estimated for each of these design parameters. Then, error analysis was used to obtain a probability distribution for the system lifetime (mean = 7.7 yr, standard deviation = 1.1 yr). From this, the probability that the system will achieve the design goal of 5 yr lifetime is 0.993. This value represents an estimate of the degradation reliability of the system.

AI-Memo-9155

TITLE: NUCLEAR POWER SYSTEM CONCEPTS FOR A LUNAR BASE

AUTHOR: Staff

10-20-63

ABSTRACT: Four specific plant concepts for lunar base applications have been designed using 5-kwe, 10-kwe, and 100-kwe turboelectric and 10-kwe thermoelectric power conversion systems. The four plants are merely examples of the range of capabilities of the SNAP system components. Thermal power is supplied by either a SNAP 2, SNAP 8, or upgraded SNAP 8 reactor as required.

AI-64-106

TITLE: SNAP 8 MARINE PROPUSION SYSTEM

AUTHOR: H. R. Brager

5-11-64

ABSTRACT: The feasibility of using the SNAP 8 reactor as an energy source to supply 30-kwe primary propulsion power for the Reynolds Aluminaut is demonstrated. The system, using eight SNAP 2 mercury power conversion systems, would be 8 ft in diameter by 15-1/2 ft long and weigh 36,600 lb, including shielding. A 120-kwe system could be built that would be 6% larger and weigh 46,000 lb.

AI-AEC-13066

AI-64-Memo-129

TITLE: INTERIM STUDY SUMMARY - MORL NUCLEAR POWER SYSTEMS

AUTHOR: W. V. Botts

6-18-64

ABSTRACT: A comprehensive study of the use of nuclear power for the six-man MORL mission includes preliminary optimization of both thermoelectric (TE) and mercury rankine (MR) nuclear power packages for direct intergration with the MORL. Both systems are conceptually designed for continuous net power of 15 kwe for 2 years but are also evaluated for 7.5 and 10 kwe. The conclusions reached in the study are that the nuclear power plants offer significant weight and operational advantages over all other competitive power sources and are well within the projected state-of-the-art. Criteria for choosing between the TE or MR system based on system design constraints are presented.

AI-65-Memo-9

TITLE: PRELIMINARY STUDY OF PROPERTIES OF COMPACT FAST REACTORS

AUTHORS: G. B. Zwetzig, L. S. Mims, and C. R. Harder

1-8-65

ABSTRACT: Compact fast reactor designs, based on PuN fuel, are examined parametrically and three reference designs are examined in detail. The three designs are for 29 kw, 250 kw, and 3 Mw. This study indicates that for unmanned missions these designs offer an improvement in specific weight compared to hydride fueled reactors. Improvements can be obtained in reliability, high-temperature capability, higher power conversion efficiency, and lower weights. The detailed investigation of a 3 Mw reactor is recommended.

AI-66-4

TITLE: SPACE POWERPLANT WORKING FLUIDS

AUTHOR: I. E. Randall

1-6-66

ABSTRACT: Space power requirements for the late 1970's and early 1980's are related to the available power sources. The performance characteristics of the sources are compared. The direct Rankine cycle is shown to be the most practical. Power systems using this cycle for a 40-kwe plant are optimized using each of several working fluids and their performances compared. The conclusion is drawn that there is very little difference in primary system factors for rubidium, potassium, cesium, mercury, sodium, or lithium working fluids. Selection must be made based on secondary matters such as the state of technology and fluid costs.

AI-66-TDR-25

TITLE: PERFORMANCE CHARACTERISTICS OF MERCURY AND ORGANIC RANKINE CYCLES AND THE ARGON BRAYTON CYCLE

AUTHOR: P. Barbour

2-28-66

ABSTRACT: A comparison of the mercury and organic Rankine cycles and the argon Brayton cycle over the temperature ranges applicable to the systems indicates that: (1) the Brayton cycle is not attractive when coupled to reactor systems unless extremely large radiator areas or very high temperatures (1800°F) can be tolerated, (2) organic Rankine cycles are well suited to long-life passive reactor control applications, and (3) isotope capsule temperatures are not materially reduced by changing from mercury to organic coolant when heat transfer from the capsule is by radiation.

AI-AEC-13066

AI-66-Memo-44, Rev 1

TITLE: NUCLEAR POWER SYSTEMS FOR LUNA EXPLORATION

AUTHOR: Staff

4-18-66

ABSTRACT: Nuclear power systems using both reactors and isotopes are designed for lunar applications and then evaluated on the bases of mission productivity and safety. Other factors such as maintainability and deployment are also considered. Both static and dynamic power conversion systems are coupled to the reactor and isotope heat sources.

AI-66-Memo-154

TITLE: REACTOR THERMOELECTRIC SYSTEM DESIGN STUDIES (0.5 to 10 kwe)

AUTHOR: Staff

-66

ABSTRACT: The design of low-to-medium power reactor thermoelectric systems utilizing SNAP 10A experience and technology is presented. Configuration, weight, and performance as well as a variety of systems and design criteria including material protection, system reliability, dose criteria, system life, separation distance, reactor control and optional features are considered. The detailed design for a 2-kwe system is projected over a power range of 500 watts to 10 kwe. The reference system has a 3-yr life and a reliability of 0.99, weighs 1081 lb, and has an overall height of 13 ft, 7 in.

AI-66-206

TITLE: 10 to 30 kwe NUCLEAR UNDERWATER POWER SYSTEM

AUTHOR: Staff

10-17-66

ABSTRACT: A natural-circulation-cooled reactor to deliver 10 to 30 kwe is designed for independent underwater use and its performance compared with competitive systems such as batteries and fuel cells. The reactor system is compact, light-weight (14,000 lb), safe, easy to handle and ship, and does not require refueling in its 20-yr life at 50% usage factor. The 30-kwe system will cost on the order of \$10⁶ per unit.

AI-67-TDR-102

TITLE: CHARACTERISTICS OF MARINE AND TERRESTRIAL LONG LEVEL UNATTENDED POWER SOURCES (<10 kw)

AUTHOR: J. E. Stewart

7-5-67

ABSTRACT: The characteristics of electrochemical, fossil fuel, and nuclear power generating systems are evaluated on the ability to continuously supply primary power of 1 to 10,000 watts for periods up to 5 years. With the information on weight, volume, lifetime, reliability, and cost presented one should be able to select the most suitable power supply for a particular unattended application.

AI-68-89

TITLE: SYSTEM DESCRIPTION AND JUSTIFICATION - STUDY OF A NUCLEAR POWER SYSTEM IN A HARDENED UNDERGROUND MISSILE FACILITY

AUTHOR: Staff

7-26-68

ABSTRACT: The design of an unattended 12-kwe nuclear power source with a 10-yr life and post-nuclear attack survivability is described. Comparisons with alternate power sources and justification of the design selected are also presented.

AI-AEC-13066

AI-68-131, Volumes I through VI

TITLE: HARDENED MISSILE BASE NUCLEAR POWER SYSTEM STUDY

AUTHOR: Staff

12-31-68

ABSTRACT: Volume I Design Concept
Volume II Power Plant and Heat Sink Detailed Support Data
Volume III UZrH Reactor Detailed Support Data
Volume IV Budgetary and Planning Estimates
Volume V Reactor Demonstration Power Plant
Volume VI Summary Evaluation

Detailed design and evaluation of a 12 kwe, 10 year life unattended power system is presented. Both functional and economic comparisons with alternate power sources are considered. Detailed schedules and costs for fabrication and testing of a demonstration plant and follow-on production are also shown.

AI-69-85, Volume I, II, and III

TITLE: FOLLOW-ON HARDENED MISSILE BASE NUCLEAR POWER SYSTEM STUDY
FINAL REPORT

AUTHOR: Staff

10-31-69

ABSTRACT: Volume I Design Concept Evaluation
Volume II Appendices
Volume III Power System Summary Description

The reference reactor-converter system for the Hardened Missile Base (HMB) mission consists of a U-ZrH reactor coupled to a Pb-Te thermoelectric converter by a natural circulation NaK-78 coolant loop. A recent change of the requirement from a 400-kwt, 10-year life system to a 200-kwt, 15-year lifetime prompted a reevaluation of the HMB reactor. The results of this reevaluation indicate that the core design represents a conservative application of existing ZrH reactor technology, resulting in a highly reliable powerplant system.

AI-70-88

TITLE: ZIRCONIUM HYDRIDE REACTOR-POWER SYSTEMS FOR UNDERWATER PULSE
POWER APPLICATIONS

AUTHOR: Staff

12-70

ABSTRACT: The spectrum of UZrH reactor thermoelectric, organic Rankine, and liquid metal MHD concepts can all become available during the 1975 to 1980 period to provide a continuous power range of 0.5 to 500 kwe with capability of generating pulse power levels of 1 to 10 Mwe for highly reliable, autonomous, 10-year lifetime without refueling. Various methods of energy storage are evaluated and the optimum method is selected for each combination of pulse power and repetition rate. The ZrH reactor power system can be designed to meet any of the expected operational criteria associated with large pulse power undersea applications.

RAR-652-340-002, Rev A

TITLE: PRELIMINARY FAILURE MODE AND EFFECT ANALYSIS

AUTHOR: J. V. Addison

9-8-72

ABSTRACT: A preliminary Failure Mode and Effect Analysis (FMEA) was made on the overall 5-kwe system. A general discussion of the system and failure effect is given in addition to the tabulated FMEA and a primary block diagram of the system.

AI-AEC-13066

TI-652-310-001

TITLE: PIPING THERMAL STRESS

AUTHOR: S. Westerman

10-27-72

ABSTRACT: Results of a thermal stress analysis on the reference design piping system are presented. Conclusions verify the use of expansion joints in the primary piping system. A backup concept utilizing Hastelloy N is also presented.

TI-652-310-003

TITLE: MASS TRANSFER AND OXYGEN ANALYSIS IN THE SNAP 5-kwe GROUND TEST REACTOR

AUTHOR: R. L. Eichelberger

1-9-73

ABSTRACT: Mass transfer in the SNAP 5-kwe Ground Test Reactor System is discussed. It is shown that because the primary system consists of several different alloys, it is predisposed to be subject to mass transfer processes. For a given combination of materials, the rate at which mass transfer will occur depends upon the maximum temperature and the temperature difference in the loop, and upon the oxygen concentration of the NaK. In order that the system produce its rated power, the maximum temperature and the specified temperature difference must be obtained in operation, so that the only controllable factor affecting mass transfer is the oxygen impurity concentration in the NaK. Because the system does not include a cold trap, it is necessary to effect the control of the oxygen concentration during the NaK charging and cleanup operation. The cleanup operation should involve cold trapping with an auxiliary, removable unit, and it should be operated until the oxygen concentration is no more than 5 wppm, with the system at at least 600°F, and preferably at 800°F. Then the loading and cleanup unit can be removed and the reactor system sealed off.

TI-652-610-003

TITLE: PRESTART ORBITAL OPERATING CHARACTERISTICS

AUTHOR: J. Beoletto

12-4-72

ABSTRACT: The performance characteristics of the 5-kwe reactor-TE system were evaluated under prestart orbital operating conditions at a reference 600-n. mi orbit. Shroud temperatures and NaK temperatures were evaluated for maximum and minimum thermal input orbits assuming varying conditions of NaK flow and shroud absorptivity/emissivity ratios. To maintain desirable minimum NaK temperatures, a shroud α/ϵ ratio of 2.5 to 3.0 with approximately 5% NaK flow in the system is required.

TSR-652-640-002

TITLE: 5-kwe REACTOR THERMOELECTRIC SYSTEM DESIGN POINT AND CONFIGURATION MODIFICATION STUDY

AUTHOR: R. V. Anderson

9-20-72

ABSTRACT: A configuration and thermal performance statepoint had been established for the 5-kwe Reactor Thermoelectric System. Design modifications to this system have resulted from further analyses and trade studies of component performance parameters and designs. Preliminary margin assessments were incorporated into several component designs. The resultant system was established as the conceptual design. Associated performance and weight summaries were prepared for this system.

2. Systems Testing

NAA-SR-11397

TITLE: SNAP 10A FS-3 REACTOR PERFORMANCE

AUTHORS: J. P. Hawley and R. A. Johnson

8-15-66

ABSTRACT: SNAP 10A FS-3 was the first flight-qualified SNAP reactor system to be operated in a simulated space environment. Prestartup qualification testing, automatic startup, endurance period performance, extended operation test, and reactor shutdown are described. Performance of the reactor control system and the diagnostic instrumentation is evaluated.

NAA-SR-11814

TITLE: SNAP 10A FSM-4 SYSTEM TEST

AUTHOR P. L. Soske

3-25-66

ABSTRACT: The FSM-4 test system duplicated in every detail the SNAP 10A unit except for the nuclear heat source, which was simulated in the FSM-4 tests by electrical heating. Before the flight system could be launched, FSM-4 was required to demonstrate successful performance in surviving the startup temperature transient. The FSM-4 testing had four objectives: (1) shock and vibration testing of a NaK-loaded system, (2) simulated orbital operation before startup, (3) simulated orbital startup sequence with emphasis on the thermal transient, and (4) thermal reference endurance testing.

NAA-SR-12504

TITLE: THE DISASSEMBLY AND POSTOPERATION COMPONENT EXAMINATION OF THE SNAP 10A FS-3

AUTHORS: J. P. Dooley and J. P. Beall

9-15-67

ABSTRACT: The SNAP 10A Flight System was operated as a ground demonstration test of the SNAP 10A system. The test was conducted by Atomics International at the Santa Susana Field Test Laboratory for the Atomic Energy Commission. After operating continuously for over 10,000 hr, the test was terminated, the system was disassembled, and critical system components were examined to determine the effect of the system operating environment. This report covers the disassembly of the system and the postoperation examination of the system components, with the exception of the fuel.

3. SNAP 2

NAA-SR-1840

TITLE: A 3-kw NUCLEAR AUXILIARY POWER UNIT FOR THE 117L ADVANCED RECONNAISSANCE SYSTEM

AUTHOR: J. R. Wetch (Editor)

1-31-57

ABSTRACT: This report presents a summary of a 9-month feasibility study for nuclear auxiliary power plant. The reactor consists of a metal hydride, uranium-235 fueled, beryllium-reflected, and sodium-cooled thermal reactor producing 45 kw. The heat is transferred to a Rankine cycle mercury power plant delivering 3 kwe. The whole system weighs about 185 lb.

NAA-SR-3465

TITLE: SNAP 2 EXPERIMENTAL REACTOR HAZARDS REPORT

AUTHOR: R. R. Eggleston and G. L. Schmidt (Editors)

4-1-59

ABSTRACT: A description of the SNAP Experimental Reactor (SER), the site, and facility for the operation of the SER is presented. A number of accident studies are made and evaluated.

NAA-SR-4831

TITLE: CLADDING DEVELOPMENT FOR THE SER FUEL ELEMENTS

AUTHORS: J. Romero and S. Elchyshyn

2-15-61

ABSTRACT: Cladding and hydrogen-barrier materials were evaluated for use on SER fuel elements. Factors considered were hydrogen retention, coatability, weldability, availability, and sodium corrosion. The necessary test equipment was designed and built and the test techniques were developed. A glass coating, Solaramic, applied to the inside of Type 347 stainless steel tubing was selected for the SER fuel elements. Processes were developed for the SER core fabrication.

NAA-SR-5317

TITLE: THE SNAP-2 RADIATIVE CONDENSER ANALYSIS

AUTHORS: M. G. Coombs, R. A. Stone, and T. Kapus

-61

ABSTRACT: This report summarizes the design analysis for the SNAP-2 direct-condensing radiator. The design point selected in this report utilizes mercury as the working fluid to be condensed at 600°F.

NAA-SR-5613

TITLE: SNAP-2 SYSTEM STUDIES

AUTHOR: R. D. Welsh

10-15-61

ABSTRACT: A study was made of the SNAP 2 nose cone configuration design point for reliability and minimum weight. Included in the study were: steady-state, full-power, and orbital operation. Brief consideration was given to a cylindrical SNAP 2 configuration. Also considered were thermoelectric, canned-rotor, and EM NaK pumps instead of the shaft-driven permanent magnet configuration.

NAA-SR-6310

TITLE: SNAP 2 POWER CONVERSION SYSTEM ROTATIONAL SPEED CONTROL:
TOPICAL REPORT No. 18

AUTHORS: W. E. Dauterman and E. J. Viton

12-1-63

ABSTRACT: The parasitic load speed control for the SNAP 2 power conversion system consists of a frequency-discriminator low-level magnetic amplifier, magnetic amplifier power stages, and a resistive load. The design and development of this parasitic load speed control is described. Experimental results are given to confirm the breadboard design approach.

AI-AEC-13066

NAA-SR-6311

TITLE: THE SNAP 2 POWER CONVERSION SYSTEM TOPICAL REPORT No. 19
MERCURY MATERIALS EVALUATION AND SELECTION

AUTHOR: J. F. Mejedlik 12-63

ABSTRACT: SNAP 2 is the designation for a 3-kw nuclear auxiliary power unit to be used in a satellite vehicle. The SNAP 2 system consists of a reactor heat source, a boiler, a condenser, a mercury rankine engine, and an alternator. The corrosion and subsequent mass transfer resulting from the use of mercury as the thermodynamic working fluid are important considerations in the selection of materials for the SNAP 2 system. Consequently, corrosion and mass transfer behavior have been under study for the past 4 years. This report presents the results of this study for fiscal year 1962. The program was performed under a subcontract to Atomics International as part of the Atomic Energy Commission Contract AT(11-1)-GEN-8.

NAA-SR-6312

TITLE: SNAP 2 POWER CONVERSION SYSTEM TOPICAL REPORT No. 20
CRU V DESIGN

AUTHOR: Staff - TRW 5-31-63

ABSTRACT: CRU V design and development was undertaken to provide a SNAP 2 turbo-alternator package that will meet SNAPSHOT flight objectives. This report describes the selection of the design concept, CRU detail design features, performance estimates, and the plumbing module design. The layout design fulfills the majority of the specific CRU V design objectives. Potential problems are discussed but they are not considered to jeopardize accomplishment of current SNAP 2 program objectives.

NAA-SR-6314

TITLE: SNAP 2 POWER CONVERSION SYSTEM TOPICAL REPORT No. 22
JET-CENTRIFUGAL MERCURY PUMP DESIGN AND PERFORMANCE ANALYSIS

AUTHORS: C. W. Grennan and R. A. Lewis 2-64

ABSTRACT: Two different mercury pumping units were designed and tested for the SNAP 2 system. These units have missions in the ground test and prototype flight units, respectively. The pumps are somewhat different in detail because the requirements of the ground and flight units differ. This report presents the design analysis and performance results obtained during the 1963 fiscal year.

NAA-SR-6315

TITLE: SNAP 2 POWER CONVERSION SYSTEM TOPICAL REPORT No. 23
COMPACT BOILER DEVELOPMENT

AUTHORS: R. J. Zibbra and D. A. Oberacker 2-64

ABSTRACT: This report describes selection of the compact (5C) boiler design concept as well as thermal, hydrodynamic, and mechanical design and stress analysis of the 5C boiler. Results of thermal performance and acceptance tests on the 5C-1, 5C-2, 5C-4, and 5C-5 boilers and the acceptance vibration test of the 5C-3 boiler are also presented. The compact boiler effort produced the highly successful Model 5C boiler that weighs only 50% of the specification weight.

NAA-SR-6317

TITLE: SNAP 2 POWER CONVERSION SYSTEM TOPICAL REPORT No. 25
THREE-PHASE ALTERNATOR ELECTRICAL DESIGN AND PERFORMANCE
TESTING

AUTHOR: Z. L. Reineks

10-15-64

ABSTRACT: The CRU V design and development program initiated in 1962 was directed toward elimination of inadequacies noted in CRU III and CRU IV designs and toward development of a combined rotating unit fully qualified for flight. This report discusses rotor and stator design changes incorporated in the CRU V alternator and the effect of these modifications on electrical performance. Design and development of electrical output terminals are also briefly reviewed.

NAA-SR-6318

TITLE: SNAP-2 POWER CONVERSION SYSTEM TOPICAL REPORT No. 26
DESIGN, DEVELOPMENT AND PERFORMANCE OF CRU IV AND CRU IVM
MERCURY VAPOR TURBINES

AUTHOR: A. J. Stromquist

3-64

ABSTRACT: The CRU IV turbine was designed to provide a definite and substantial gain in efficiency over previous SNAP 2 turbines. The CRU IV turbine had a partial-admission first stage and a full-admission second stage with optimum ratios of wheel speed to vapor velocity. The efficiency of the initial CRU IV design was 53%, a 10% gain in efficiency. Fabrication deviations, internal leakage, and pressure losses were corrected in the CRU IVM redesign, and an additional gain of greater than 10% in turbine power and efficiency was realized, as evidenced by testing. CRU IVM turbine efficiency averaged 58% on a total-to-static, blade-to-blade basis, and 57% on a flange-to-flange basis. Although the CRU IVM turbine exceeded specification efficiency, calculations reveal that potential efficiencies of greater than 60% are possible at design point with an improved interstage collector.

The techniques used to calculate turbine performance at design point were applied to exact test data conditions and fabricated turbine geometry and produced results that agree within 2% with the test results.

NAA-SR-6319

TITLE: SNAP 2 POWER CONVERSION SYSTEM MODEL 6 TWO-PHASE FLIGHT-PACKAGED
SPEED CONTROL, TOPICAL REPORT 27

AUTHOR: W. E. Dauterman

6-64

ABSTRACT: This report summarizes the packaging effort to achieve the Model 6 flight-speed control. The heat-transfer analysis conducted prior to design of the Model 6 speed control is included.

NAA-SR-6320

TITLE: MERCURY RANKINE PROGRAM (SNAP-2) DEVELOPMENT OF LIQUID-MERCURY-LUBRICATED BEARINGS, TOPICAL REPORT NO. 28

AUTHOR: O. Decker

6-63

ABSTRACT: This report defines the analytical and experimental results obtained with fluid film bearings lubricated by liquid mercury, including the design approach, analytical techniques, procedures, test results, and experimental activity performed by TRW Inc. as a facet of the Mercury Rankine Power Conversion Program during the period of January 1961 through June 1963.

This report consist of the following six volumes:

Volume I	Analytical Design Approach and Status of Bearing Systems (With Appendixes)
Volume II	Plain Bearing Experimental Results
Volume III	Tilting-Pad Bearing Experimental Results
Volume IV	Three-Sector Bearing Experimental Results
Volume V	Three-Pad Bearing Experimental Results
Volume VI	Spiral-Groove Thrust Bearing Experimental Results

NAA-SR-6439

TITLE: SNAP 2 PRIMARY COOLANT DEVELOPMENT

AUTHOR: M. A. Perlow

7-15-61

ABSTRACT: Efforts in FY 1961 are described and summarized in the design development, testing, and selection of components integral to the primary coolant system for the SNAP 2 (Systems for Nuclear Auxiliary Power) reactor for space applications. Included are core hydraulic studies, heater development, corrosion studies, NaK hydride precipitation studies, and lithium hydride studies.

NAA-SR-6561

TITLE: THERMAL DESIGN OF SNAP REACTORS

AUTHOR: P. D. Cohn

2-1-62

ABSTRACT: This report describes the analytical techniques used in the preliminary thermal design of the SNAP 8 and SNAP 2 reactors. These analytical procedures are applicable to most core designs which incorporate a close-packed array of fuel rods.

NAA-SR-7048

TITLE: ASSEMBLY OF S2DR FUEL ELEMENTS

AUTHORS: W. F. Dennison and T. S. Jakubowski

12-15-62

ABSTRACT: The materials procurement, fabrication, and assembly of 51 fuel elements for the S2DR is described.

NAA-SR-8090

TITLE: POST-IRRADIATION EVALUATION OF SER FUEL ELEMENTS

AUTHOR: J. R. Miller

5-1-63

ABSTRACT: Selected fuel elements from the SNAP 2 Experimental Reactor were examined to determine their performance capability. The examination indicated that these fuel elements are capable of satisfactory performance at SNAP 2 full power design conditions.

NAA-SR-8303

TITLE: AEROSPACE SAFETY REENTRY ANALYTICAL AND EXPERIMENTAL PROGRAM, SNAP 2 and 10A: INTERIM REPORT

AUTHOR: R. D. Elliott

9-30-63

ABSTRACT: This report describes the analytical and experimental effort pursued by the Aerospace Safety Reentry project to investigate potential hazards to the population during the reentry from orbit of SNAP reactors.

NAA-SR-9054

TITLE: SNAP 2 STRUCTURAL AND DYNAMIC ANALYSIS

AUTHOR: R. A. Castle

5-1-64

ABSTRACT: The structural design criteria and the system basic loads for the SNAP 2 compact power unit are presented. Included in these structural criteria are the following environments: ground, prelaunch, launch, preorbital startup, and orbital startup and continuous operation. These data were then used to determine the structural integrity of the system and to satisfy the reliability requirements. A summary is presented of the more significant system structural and dynamic loads generated or analysis techniques developed on the SNAP 2 compact power unit.

NAA-SR-9056

TITLE: NUCLEAR MERCURY RANKINE POWERPLANT (3 to 20 kwe) CAPABILITY

AUTHOR: R. L. Wallerstedt

6-15-64

ABSTRACT: The design characteristics and advanced capabilities of a 3-to 20-kwe nuclear mercury-rankine power plant (using a SNAP 2 or 8 reactor) are summarized and discussed. Program objectives and development status are presented. Reliability criteria are outlined.

NAA-SR-9078

TITLE: SNAP 2 STARTUP ANALYSIS

AUTHOR: P. M. Gresho et al.

1-20-64

ABSTRACT: This report summarizes the calculational procedures and results which are used to define the orbital startup methods for the SNAP 2 unit.

AI-AEC-13066

NAA-SR-9087

TITLE: SNAP 2 PERFORMANCE ANALYSIS

AUTHOR: R. T. Lancet et al.

2-15-64

ABSTRACT: The most significant steady-state thermodynamic and fluid dynamic analyses of the SNAP 2, 3- to 5-kw nuclear power supply is presented. In addition to the general system and design point analysis, the detailed analysis of the thermal and fluid dynamic performance of the radiator-condenser and the thermal performance of the instrument compartment are presented.

NAA-SR-9252

TITLE: MULTIPLE-UNIT SNAP 2 POWER SYSTEMS FOR MANNED APPLICATION

AUTHOR: H. N. Rosenberg, (Editor)

1-28-64

ABSTRACT: A study has been made of the feasibility of utilizing multiple-unit SNAP 2 systems to provide power in the 10- to 40-kwe range for manned space stations. Principal areas described are reliability, packaging, weight, and operational considerations.

NAA-SR-9715

TITLE: APPLICATION OF SNAP REACTOR POWER SYSTEMS TO MANNED SPACE STATIONS

AUTHOR: J. McDonald

6-15-64

ABSTRACT: This report describes the zirconium-hydride-uranium thermal reactor coupled through a NaK loop to a mercury-rankine power-conversion system to be used as the power source for a manned space station. Preliminary designs of this system are given for 5-, 10-, 15-, and 20-kwe power levels.

NAA-SR-9801

TITLE: SNAP MERCURY RANKINE PROGRAM MERCURY CONDENSING EXPERIMENTS

AUTHOR: R. P. Forslund

8-1-64

ABSTRACT: An experimental study program of the various basic operating characteristics of mercury condensers for space applications is described. Included are techniques for predicting pressure drops in a variety of environments, condenser stability, the effects of noncondensibles, startup dynamics, and the differences between wetting and nonwetting condensing.

NAA-SR-9906

TITLE: SNAP MERCURY RANKINE PROGRAM SNAP-2 STRUCTURAL EVALUATION PROTOTYPE SYSTEM

AUTHOR: W. B. Gruenberg

1-1-64

ABSTRACT: This report defines the SNAP 2, PSM-2, Phase-II prototype system program. The need for a structural test item is discussed and the resulting objectives presented. A summation of the design, fabrication, and assembly efforts is given. The test program is outlined and some preliminary results are discussed.

NAA-SR-12181, Vol I

TITLE: FINAL SUMMARY REPORT - SNAP 2/MERCURY RANKINE PROGRAM REVIEW,
VOLUME I

AUTHORS: R. L. Walkerstedt, J. S. McDonald, and D. E. Ellis 6-15-67

ABSTRACT: This summary report details the technical aspects of the development program for the SNAP 2/Mercury Rankine System from the start of the program in 1958 through FY 1967. The individual system components descriptions, test results, and status of the development are presented together with a description of, and test results from, the system tests operated during the program. The engineering analytical accomplishments made during the program are also described.

NAA-SR-Memo-12181, Vol II

TITLE: FINAL SUMMARY REPORT - SNAP 2/MERCURY RANKINE PROGRAM REVIEW,
VOLUME II, ADMINISTRATIVE SUMMARY

AUTHORS: R. L. Wallerstedt, J. S. McDonald, and D. E. Ellis 6-15-67

ABSTRACT: This report summarizes the program funding and the subcontract costs from GFY 1956 through 1967.

NAA-SR-12186

TITLE: SNAP 2 DEVELOPMENTAL REACTOR MOCKUP NO. 1 FINAL REPORT

AUTHORS: L. Maki, J. F. Milner, and B. R. Granich 6-15-68

ABSTRACT: The series of developmental tests conducted on the nonnuclear SNAP 2 reactor assembly are described. The complete reflector system and the shield assembly were initially acceptance tested and then mated to a nonnuclear (dummy fuel elements) core assembly. Subsequently, the reactor assembly successfully withstood the launch environment-simulating shock and vibration tests and performed as required to simulate the startup and long-term operation in a vacuum chamber that simulated space conditions while the assembly was being heated with hot NaK circulated through the core vessel. The detailed description of the test and performance of the system and the various components are reported.

NAA-SR-12233

TITLE: MERCURY RANKINE PROGRAM BOILER DEVELOPMENT SUMMARY

AUTHOR: E. B. Baumeister 1-15-67

ABSTRACT: The Mercury Rankine Program NaK-to-mercury boiler development program history and present status are presented. Also included are the results of boiler conditioning studies, thermal-hydraulic boiler design procedures, and advanced application conceptual designs.

TITLE: SNAP 2 SUMMARY REPORT

AUTHOR: A. A. Jarrett

6-73

ABSTRACT: The development history of the SNAP 2 concept is presented. The program was originally oriented toward the development of a specific 3-kwe Mercury Rankine cycle system powered by a hydride reactor. Subsequently, the program was directed toward a broader technology development of the engineering feasibility of larger systems for space applications into the 100- to 200-kwe range. The report describes the initial critical assembly of the reactor concept, and the power, temperature, and life demonstration by the SNAP Experimental Reactor (SER) and the SNAP 2 Developmental Reactor (S2DR). The report summarizes the development of components of the power conversion system, called the combined rotating unit (CRU), which has all rotating components mounted on a single common shaft. Other major component developments described are: (1) the mercury boiler-superheater, (2) the radiator-condenser heat rejection system, and (3) the thermo-electromagnetic pump. In addition, the report describes the development and status of other components required for startup, steady-state operation, shutdown, and restart. System testing is summarized which demonstrated compatibility, structural integrity, and zero gravity characteristics. Potential applications of the Mercury Rankine system are presented.

4. SNAP 4

NAA-SR-8590

TITLE: SNAP 4 SUMMARY REPORT

AUTHOR: L. Bernath (Editor)

7-12-63

ABSTRACT: This report summarizes the technical activities conducted by Atomics International for the AEC during the past four fiscal years on the SNAP-4 program. The overall objective of the SNAP-4 program was the development of a multipurpose, compact, nuclear power system which incorporates a reflector-controlled, boiling-water-cooled, uranium-zirconium hydride-fueled-and-moderated reactor in a direct-cycle saturated steam power conversion system.

NAA-SR-8657

TITLE: PROJECT PROPOSALS FISCAL YEARS 1964 AND 1965 SNAP 4 PROTOTYPE SYSTEM DEVELOPMENT PROGRAM

AUTHOR: Staff

6-28-63

ABSTRACT: This report summarizes the research, design, and development effort required on the SNAP 4 system. The system consists of a uranium-zirconium-hydride-fueled-and-moderated, beryllium-reflector-controlled, forced convection boiling-water-cooled compact reactor.

NAA-SR-10699

TITLE: SNAP-4 COMPACT REACTOR PROGRAM SUMMARY REPORT, JULY 1, 1963 THROUGH SEPTEMBER 30, 1964

AUTHOR: Staff

1-8-65

ABSTRACT: This report presents the results of analytical, design, and development activities for the SNAP 4 Compact Reactor Program from July 1963 through September 1964.

AI-AEC-13069

TITLE: SNAP 4 SUMMARY REPORT

AUTHOR: A. R. Marchese

6-73

ABSTRACT: This report summarizes the technical activities conducted by Atomics International for the Atomic Energy Commission, during the period from 1960 to 1965, on the SNAP 4 Program. The overall objective of this program was to develop an automatically controlled, transportable, compact nuclear power system for multipurpose applications requiring power in the range of 500 to 4000 kwe. This system was designed to be capable of unattended remote operation and automatic startup and shutdown, and to be compatible with all terrestrial environments, either water or atmospheric.

AI-6915

TITLE: A COMPACT 2-MEGAWATT ELECTRICAL TRANSPORTABLE NUCLEAR POWER GENERATION UNIT

AUTHOR: Staff

2-15-62

ABSTRACT: A conceptual design for a compact transportable nuclear power generation unit is described. A SNAP-4 reactor configuration is set forth that will generate 2 megawatts of electrical power.

5. SNAP 8

NAA-SR-6207

TITLE: SNAP 8 REACTOR PRELIMINARY DESIGN SUMMARY

AUTHOR: H. Rosenberg (Editor)

6-15-61

ABSTRACT: The purpose of this report is to describe the present concept of the SNAP 8 reactor, reactor control system, safety systems, and shield and to outline the reasoning behind the design point selection.

NAA-SR-6561

TITLE: THERMAL DESIGN OF SNAP REACTORS

AUTHOR: P. D. Cohn

2-1-62

ABSTRACT: This report describes the analytical techniques used in the preliminary thermal design of the SNAP 8 and SNAP 2 reactors. These analytical procedures are applicable to most core designs which incorporate a close-packed array of fuel rods.

NAA-SR-7329

TITLE: SNAP 8 TERRESTRIAL SYSTEM

AUTHOR: S. R. Nixon et al.

5-7-62

ABSTRACT: This report describes a design and engineering program to produce a light-weight terrestrial power plant utilizing a SNAP 8 nuclear reactor core.

NAA-SR-8287

TITLE: CLADDING AND ASSEMBLY DEVELOPMENT FOR THE S8ER FUEL ELEMENTS
AUTHOR: W. F. Dennison 11-15-63
ABSTRACT: Several end closure designs were examined and one was selected to meet the temperature and hydrogen loss rate requirements of the S8ER reactor. A semi-automatic process was developed for making the ceramic seals in the end closure.

NAA-SR-9618

TITLE: SNAP 8 EXPERIMENTAL REACTOR (S8ER) HYDROGEN BARRIER COATING DEVELOPMENT
AUTHOR: R. W. Webb 6-15-64
ABSTRACT: The development and evaluation of barium-silicate ceramic coating for use on the S8ER cladding as a hydrogen barrier is presented. Coating requirements, compositions, procedures, and test results are included.

NAA-SR-9626

TITLE: ANALYSIS OF THE RESPONSE OF THE SNAP 8 NUCLEAR SYSTEM DURING STARTUP OF THE POWER CONVERSION SYSTEM
AUTHOR: C. J. Code, Jr. 6-8-64
ABSTRACT: Detailed results of the parameter studies to determine the response of the SNAP 8 system to power conversion system startup are presented. The analog simulation is described and discussed.

NAA-SR-9648

TITLE: EVALUATION OF FUEL ELEMENTS FROM SNAP 8 DEVELOPMENT REACTOR CORE
AUTHOR: P. S. Olson 1-15-65
ABSTRACT: Eight hydrided 90 Zr-10U fuel/moderator rods with BeO end reflectors, cladd with ceramic-coated Hastelloy N which had been used in the S2DR were examined after 11,290 reactor operation hours at a peak temperature of 1280°F. Apart from a small reaction with the beryllium internal reflectors, all elements were in excellent condition after 0.027 metal at. % burnup.

NAA-SR-9861

TITLE: HYDROGEN PERMEATION THROUGH SNAP 8 FUEL ELEMENT CLADDING ASSEMBLIES
AUTHOR: C. J. Engberg 9-1-64
ABSTRACT: A barium silicate enamel (AI-8763D) used as a hydrogen barrier on S8ER fuel elements is evaluated, and an empirical equation is presented expressing hydrogen flowrates through SNAP 8 cladding assemblies as a function of temperature and hydrogen dissociation pressure.

NAA-SR-10859

TITLE: SNAP 8 NUCLEAR SYSTEMS FOR UNMANNED AND MANNED APPLICATIONS

AUTHOR: D. G. Mason

5-15-65

ABSTRACT: An unmanned SNAP 8 reference design and a man-rated SNAP 8 reference concept are described in detail. Curves to estimate shield weights for each design are presented as a function of payload separation distance, payload diameter, neutron and gamma doses, dose rate, and reactor power level.

NAA-SR-10903

TITLE: SNAP 8 EXPERIMENTAL REACTOR OPERATIONS AND TEST RESULTS

AUTHORS: L. M. Fead, L. D. Felten, and V. L. Rooney

6-28-65

ABSTRACT: The test results of the SNAP 8 Experimental Reactor are summarized in this report. The test program provided experimental evaluations of the reactor performance characteristics, including power and isothermal temperature coefficients, poisoning effects, and hydrogen loss reactivity effects.

NAA-SR-11520

TITLE: DISASSEMBLY AND POSTOPERATION EXAMINATION OF THE SNAP 8 EXPERIMENTAL REACTOR

AUTHOR: R. E. Dyer

ABSTRACT: A discussion is presented of the shutdown and disassembly of the SNAP 8 Experimental Reactor and of the postoperation evaluation of the heat transfer system, the reflectors, and portions of the core.

NAA-SR-11637

TITLE: NON-NUCLEAR PERFORMANCE TESTING - SNAP 8 EXPERIMENTAL REACTOR FUEL ELEMENTS - FINAL REPORT

AUTHORS: A. J. Fitzgerald and F. A. La Scala

3-25-66

ABSTRACT: Non-nuclear testing of S8ER fuel elements at simulated reactor conditions indicate that thermal cycling and ramp heating significantly affect hydrogen permeation rates; but vibration, shock, and accelerations do not. Long-term exposure at operating conditions causes an improvement (decrease) in the permeation rate.

NAA-SR-Memo-12097

TITLE: SNAP 8 FUEL ELEMENT STRESS BRIEFING, JULY 19 AND 20, 1966, CANOGA PARK, CALIFORNIA

AUTHOR: Staff

8-5-66

ABSTRACT: This report is a briefing book for the presentation of the status of SNAP 8 fuel element development, the test results from the S8ER and in-pile irradiations, and the fuel element stress analysis.

AI-AEC-13066

NAA-SR-Memo-12099

TITLE: SNAP 8 SUMMARY — S8ER and S8DS

AUTHOR: Staff 8-30-66

ABSTRACT: A brief summary is presented of the operational and post-operational results of the SNAP 8 Experimental Reactor (S8ER) and the relationship of these results with the SNAP 8 Developmental Reactor (S8DS) design.

NAA-SR-Memo-12122

TITLE: SNAP 8 BRIEFING, WASHINGTON, D. C., AUGUST 15-16, 1966

AUTHOR: Staff 8-31-66

ABSTRACT: This report is a briefing book presenting the results of the SNAP 8 fuel element stress analysis, SNAP 8 fuel studies from the S8ER, and in-pile fuel irradiations.

NAA-SR-Memo-12210

TITLE: SNAP 8 EXPERIMENTAL REACTOR FUEL ELEMENT BEHAVIOR — ATOMICS INTERNATIONAL TASK FORCE REVIEW

AUTHOR: H. Pearlman (Chairman) 11-30-66

ABSTRACT: The task force scope was to review, evaluate, and make recommendations on Atomics International post-irradiation examination of SNAP 8 Experimental Reactor (S8ER) fuel elements. Emphasis was on density changes, dimensional changer, hydrogen compensation, and microstructure. The report includes those evaluations, recommendations, and other findings resulting from this task force review.

NAA-SR-Memo-12272

TITLE: SNAP 8 BRIEFING TO AEC WASHINGTON, D. C., DECEMBER 1, 1966

AUTHOR: Staff 12-14-66

ABSTRACT: This report is a briefing book presenting information on the SNAP 8 fuel swelling analysis, fuel element life behavior, core design, Hastelloy N materials data, cladding stresses, and the S8DR test program.

NAA-SR-12300

TITLE: FUNCTIONAL DESIGN DESCRIPTION AND TEST SUMMARY OF SNAP 8 DEVELOPMENTAL REACTOR MOCKUP - 1

AUTHOR: W. J. Kurzeka 2-24-67

ABSTRACT: The design features and the testing of the S8DRM-1 are described. S8DRM-1 is a nonnuclear version of the first SNAP 8 reactor designed for 1300°F outlet NaK and 600 kw power output. The system tested was completed, except that dummy fuel element assemblies were installed during the shock and vibration tests and an electrical heater was installed in the core during the thermal testing.

Summary of the performance record, simulated reactor startups, thermal vacuum and launch shock, and vibration tests are presented. All of these tests were successfully completed with only minor malfunctions. Design improvements to overcome these problems are recommended.

AI-AEC-13066

NAA-SR-Memo-12339

TITLE: SNAP 8 DEVELOPMENTAL REACTOR PROGRAM REVIEW

AUTHOR: Staff

2-28-67

ABSTRACT: This report is a briefing book that presents the S8DR test objectives, component studies and development, fuel element studies, and development programs.

NAA-SR-12355

TITLE: THERMAL BEHAVIOR OF SNAP 8 REACTOR DURING ATMOSPHERIC REENTRY

AUTHOR: L. D. Montgomery, et al.

5-1-67

ABSTRACT: An analysis was conducted to simulate the thermal behavior of a SNAP 8 type reactor system during reentry into the earth's atmosphere. From the results of this analysis, it is concluded that the SNAP 8 reactor vessel will not ablate to a sufficient extent, for probable modes of reentry, to release the fuel elements at altitudes necessary to obtain complete fuel ablation.

NAA-SR-12449

TITLE: ELECTRON MICROSCOPY ANALYSIS OF IRRADIATED SNAP 8 FUEL

AUTHOR: K. J. Miller

8-20-67

ABSTRACT: Irradiated fuel from the NAA 115-1 experiment was examined by electron microscopy replication techniques. Fuel microcracking was observed, but it was not excessive. Bubble densities were on the order of 10^{13} bubbles/cc with an average diameter of 900 Å. Fission gas bubble formation accounted for approximately 50% of the observed hydrogen-connected volume change. Bubble size and density appeared to depend on temperature but not burnup. There was no evidence of fission gas bubble sweeping during phase change, and no correlation of bubble sites with epsilon banding or microcracks. Bubble-depleted regions were found around large particles.

AI-AEC-Memo-12702

TITLE: S8DR FUEL ELEMENT QUALITY ASSURANCE RECORD

AUTHORS: C. Donnelly and W. Nagel

3-21-68

ABSTRACT: This report summarizes the major inspection, test, and traceability information for the 221 enriched uranium fuel elements released for use in the SNAP 8 Developmental Reactor.

AI-AEC-Memo-12775

TITLE: S8DR FUEL ROD FABRICATION

AUTHOR: D. T. McClelland

1-29-69

ABSTRACT: The detailed process parameters and product inspection data for fabrication of the S8DR fuel rods are presented.

AI-AEC-13066

AI-AEC-Memo-12782

TITLE: CHROMIUM REDISTRIBUTION EVALUATION OF S8DR COATED TUBING
AUTHORS: J. R. Miller and J. T. Roberts 2-5-69
ABSTRACT: The extent of chromium diffusion from the surface of chromized and coated Hastelloy N tubing was determined after aging up to 10,000 hr at 1400 to 1600°F. Changes in the coating of substrate were also studied.

AI-AEC-Memo-12784

TITLE: SNAP PROGRAM REVIEW, WASHINGTON, D.C., JANUARY 30, 1969
AUTHOR: Staff -69
ABSTRACT: This report is a briefing book for the presentation of the SNAP 8 Developmental Reactor operating results, fuel development test results, and intrinsic water subcriticality test results.

AI-AEC-12850

TITLE: SNAP 8 AUTOMATIC STARTUP AND PCS MERCURY INJECTION SIMULATION TESTS
AUTHOR: R. F. Audette 10-31-69
ABSTRACT: SNAP 8 reactor dynamic tests have demonstrated the capability for fully automatic startup over eight decades of power from shutdown, and for automatically furnishing the heat load demand imposed by automatic startup of the flight system power conversion loop following reactor startup. The tests performed approximate predicted flight system operation.

AI-AEC-13002

TITLE: PROTOTYPE S8DR FUEL ELEMENT PERFORMANCE TEST (NAA 121-1 EXPERIMENT)
AUTHOR: J. G. LaBlanc 7-15-71
ABSTRACT: The NAA 121-1 experiment consisting of four S8DR prototype fuel elements was irradiated in the Hanford K-East Reactor for 12,000 hr at cladding temperatures up to 1300°F and fuel temperatures to 1450°F. These fuel elements achieved peak burnups of about 0.3 metal at. %. Post-irradiation examination indicated that the three normal-hydrogen-content elements behaved satisfactorily. However, these full-length experiments produced families of swelling data that were inconsistent with prior short-length (single-point) specimens. The fuel in the fourth, low-hydrogen-content fuel element was deliberately caused to transform from δ to β zirconium hydride. This fuel element was severely damaged. A mechanism of fuel shrinkage followed by excessive swelling is postulated to explain the behavior of this element.

AI-AEC-13070

TITLE: SNAP 8 SUMMARY REPORT
AUTHORS: A. W. Thiele and S. Berger 6-73
ABSTRACT: The SNAP 8 Program was directed toward the development of lightweight nuclear reactors which would operate in space for greater than 10,000 hr at a power level of 600 kw. Toward this objective, two reactors were designed, built, and successfully operated. The experimental reactor (S8ER) was an engineering test of a prototype core, but did not have flight-developed hardware. The developmental reactor (S8DR) was a nuclear system test to verify the operability of the integrated flight configuration. This report discusses, in detail, for each of the two reactors, their development, design characteristics, operating experience, experimental results, and supporting tests.

AI-AEC-13066

6. SNAP 10A

NAA-SR-3473

TITLE: FEASIBILITY STUDY AND PRELIMINARY DESIGN OF A LOW POWER DIRECT CONVERSION NUCLEAR AUXILIARY POWER UNIT

AUTHOR: A. W. Thiele

11-1-59

ABSTRACT: The preliminary design of a reactor auxiliary power unit for space application is presented. The unit utilizes direct conversion of the reactor heat to electricity by thermoelectric methods. The power is 250 w electric. The unit is completely static, requiring no moving parts or fluids and no active control devices during its operating life.

NAA-SR-9892

TITLE: SNAP 10A POWERED METEOROLOGICAL SATELLITE STUDY

AUTHOR: Staff

6-15-64

ABSTRACT: The applicability of reactor-powered SNAP systems for advanced Operational Meteorological Satellites (OMS) is summarized. An optimum configuration for such a satellite is given.

NAA-SR-11397

TITLE: SNAP 10A FS-3 REACTOR PERFORMANCE

AUTHORS: J. P. Hawley and R. A. Johnson

8-15-66

ABSTRACT: SNAP 10A FS-3 was the first flight-qualified SNAP reactor system to be operated in a simulated space environment. Prestartup qualification testing, automatic startup, endurance period performance, extended operation test, and reactor shutdown are described. Performance of the reactor control system and the diagnostic instrumentation is evaluated.

NAA-SR-11814

TITLE: SNAP 10A FSM-4 SYSTEM TEST

AUTHOR: P. L. Soske

3-25-66

ABSTRACT: The FSM-4 test system duplicated in every detail the SNAP 10A unit except for the nuclear heat source, which was simulated in the FSM-4 tests by electrical heating. Before the flight system could be launched, FSM-4 was required to demonstrate successful performance in surviving the startup temperature transient. The FSM-4 testing had four objectives: (1) shock and vibration testing of a NaK-loaded system, (2) simulated orbital operation before startup, (3) simulated orbital startup sequence with emphasis on the thermal transient, and (4) thermal reference endurance testing.

NAA-SR-11934

TITLE: FINAL SNAPSHOT PERFORMANCE REPORT

AUTHOR: G. E. Berg

8-15-66

ABSTRACT: An evaluation of overall system and individual component performance during all phases of the flight test is presented for SNAP 10A. In the evaluation, special emphasis is placed on maximizing the amount of information obtained regarding the effect of the launch and space environments on the operation and performance of a nuclear thermoelectric system. In general, the in-flight data agree with ground test results or extrapolation of those results to space.

AI-AEC-13066

NAA-SR-12073

TITLE: SNAP 10A SUMMARY REPORT

AUTHOR: D. W. Staub

3-25-67

ABSTRACT: The SNAP 10A/SNAPSHOT Program objectives and system requirements are presented. A technical description of the system and its major components is given together with a technical experience summary derived from the developmental engineering and test program. A summary of the safety evaluation and a description of the flight test are also included.

NAA-SR-12504

TITLE: THE DISASSEMBLY AND POSTOPERATION COMPONENT EXAMINATION OF THE SNAP 10A FS-3

AUTHORS: J. P. Dooley and J. P. Beall

9-15-67

ABSTRACT: The SNAP 10A Flight System was operated as a ground demonstration test of the SNAP 10A system. The test was conducted by Atomics International at the Santa Susana Field Test Laboratory for the Atomic Energy Commission. After operating continuously for over 10,000 hr, the test was terminated, the system was disassembled, and critical system components were examined to determine the effect of the system operating environment. This report covers the disassembly of the system and the postoperation examination of the system components, with the exception of the fuel.

NAA-SR-Memo-12614

TITLE: SNAP ENGINEERING TEST FACILITY SAFETY ANALYSIS REPORT ADDENDUM FOR INTRINSIC SUBCRITICALITY EXPERIMENTS

AUTHORS: K. G. Golliher, V. A. Swanson, A. W. Thiele, and B. J. Verna

11-30-67

ABSTRACT: The use of the SNAPTRAN reactor in the SNAP Engineering and Test Facility (SETF) for a series of experiments in support of the Intrinsic Subcriticality Project of the Aerospace Safety Program is described. A summary description of the reactor and facility, an experimental program description, the proposed operating limits, and the safety analysis are included. It is concluded that the proposed operation of the SNAPTRAN reactor in the SETF is within the scope of existing safeguards approvals for the facility.

NAA-SR-Memo-12614, Supplement 1

TITLE: SNAP ENGINEERING TEST FACILITY SAFETY ANALYSIS REPORT ADDENDUM FOR INTRINSIC SUBCRITICALITY EXPERIMENTS, SUPPLEMENT 1

AUTHOR: W. Heneveld

2-8-71

ABSTRACT: The material in this report is a description of a new system for the instrumentation and control of the SNAPTRAN reactor. It is intended to replace Appendix B of NAA-SR-Memo-12614. The purpose of this newer equipment is to provide greater reliability.

AI-AEC-13066

7. SNAP 2/10A

NAA-SR-7529

TITLE: EFFECTS OF VIBRATION AND SHOCK ON HYDROGEN PERMEATION ON
SNAP 2/10A FUEL ELEMENTS

AUTHOR: A. J. Fitzgerald 3-1-63

ABSTRACT: The hydrogen retention qualities of SNAP 2/10A development fuel elements was determined before and after being subjected to simulated launch vibration and shock conditions. A method of normalizing the data to correct for variations in hydrogen content and test temperature profiles is presented.

NAA-SR-8303

TITLE: AEROSPACE SAFETY REENTRY ANALYTICAL AND EXPERIMENTAL PROGRAM,
SNAP 2 and 10A: INTERIM REPORT

AUTHOR: R. D. Elliott 9-30-63

ABSTRACT: This report describes the analytical and experimental effort pursued by the Aerospace Safety Reentry project to investigate potential hazards to the population during the reentry from orbit of SNAP reactors.

8. 5 kwe — Thermoelectric

AI-AEC-13058

TITLE: PERFORMANCE MODELING OF THE 5-kwe REACTOR THERMOELECTRIC
SYSTEM

AUTHOR: R. V. Anderson 4-3-73

ABSTRACT: A time-dependent model has been developed for the 5-kwe Reactor Thermoelectric (TE) System. This model incorporates bulk simulation of the TE modules and pump/piping system, with nodal modeling of the radiator subsystem. Temperature-dependent degradation of the power and pump TE modules is provided for time-dependent performance characteristics. The model is fully parametric and all pertinent variables may be externally controlled. The model may be used for system design specification and system performance mapping analyses. Assumptions incorporated into the model are discussed.

AI-AEC-13073

TITLE: SCRAM CONTROLLER AND AUTOMATIC CONTROLLER, 5-kwe REACTOR
THERMOELECTRIC SYSTEM

AUTHOR: R. J. Tuttle 5-25-73

ABSTRACT: The design objectives and limitations developed for the scram and automatic controllers are discussed in detail and, to a lesser extent, for the manual and computer control systems and the backup shutdown mechanism. Digital computer simulations used to analyze startup methods are described and the results presented for several candidate methods. The selected startup method uses three reactivity insertion rates to provide a startup satisfying the design objectives and limitations. Improvements in this startup method are recommended. Functional requirements of the automatic and scram controllers are discussed and means of implementing these functions by use of integrated circuits are proposed.

AI-AEC-13066

AI-AEC-13074

TITLE OPERATIONAL HAZARDS EVALUATION FOR 5-kwe REACTOR THERMOELECTRIC SYSTEM GROUND TEST

AUTHOR R. J. Tuttle 6-73

ABSTRACT Selected reactor accidents, analyzed by means of a digital computer simulation program, and the requirements for scram protection are discussed. Prevention of undesirable interaction between the control and safety systems is described. Operational limitations on segment positions are presented that afford adequate shutdown protection throughout the reactor lifetime. Potential operator errors and equipment malfunctions and failures are discussed.

AI-AEC-13075

TITLE REACTOR ASSEMBLY, 5-kwe REACTOR THERMOELECTRIC SYSTEM

AUTHOR H. Rood 6-73

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

AI-AEC-13077

TITLE FUEL ELEMENT AND CORE BUNDLE, 5 kwe REACTOR THERMOELECTRIC SYSTEM

AUTHORS L. D. Felten and E. Moody 6-73

ABSTRACT This report discusses the design of the core bundle for the 5-kwe reactor. The design thermal power is 110 kw at 1200°F outlet temperature for 5 yr. Most of the thermal, hydraulic, and stress calculations have been completed for the core bundle. Operational stresses for the fuel elements show positive margins of safety. Launch vibration loads have yet to be specified so that the fuel slug vibration suppressor design can be verified.

AI-AEC-13093

TITLE DESIGN AND ANALYSIS OF RADIATOR-STRUCTURE FOR SPACE POWER SYSTEMS

AUTHORS W. H. Dauterman and L. D. Montgomery 6-6-73

ABSTRACT The design, analysis, fabrication, and development of the 5-kwe radiator-structure is shown. Thermal performance, meteoroid protection, structural capability during launch, development testing and space operation, material evaluation, and the configuration selection are described. The fin-tube development depends on the relative values of the thermal coefficients of expansion. The initial selection of aluminum fins and Type 316 stainless steel tubes was based on previous experience, however, the large differential in their expansion rates showed that an alternate, more compatible, combination was needed. Copper, stainless steel clad copper, boron-impregnated aluminum, and an independent radiator with a titanium structure were all considered as alternate materials. The final selection was Lockalloy fins with Type 304 stainless steel 'D' tubes.

AI-AEC-13066

AI-AEC-13096

TITLE 5 kwe REACTOR THERMOELECTRIC SYSTEM SUMMARY

AUTHOR T H Springer

6-73

ABSTRACT Design of the 5 kwe Reactor Thermoelectric System was initiated in February 1972, and extended through the conceptual design phase and into the preliminary design phase. Design effort was terminated in January 1973. This report documents the system and component requirements, design approaches, and performance and design characteristics for the 5-kwe system. Included is summary information on the reactor, radiation shields, power conversion systems, thermoelectric pump, radiator-structure, liquid metal components, and the control system.

AI-AEC-13102

TITLE MATERIALS INTERACTIONS BETWEEN THE THERMOELECTRIC CONVERTER AND THE 5-kwe REACTOR SYSTEM

AUTHOR P B. Ferry

6-73

ABSTRACT Integration of a compact thermoelectric converter with the 5-kwe reactor system requires introduction of Ta - 10 W piping into some portion of the primary coolant loop. The accompanying material interaction uncertainties could affect the present system design. These include degradation of the required austenitic - refractory metal transition joint during operation at high temperatures, loss of corrosion resistance, embrittlement by the presence of hydrogen, and loss of design margin by transport of interstitial elements. Analysis and limited experimental evidence indicates that these potential materials interactions can be adequately controlled. It appears that the Group Va refractory metal can be utilized without unacceptable adverse effect on system reliability.

RAR-652-340-002, Rev A

TITLE PRELIMINARY FAILURE MODE AND EFFECT ANALYSIS

AUTHOR J. V. Addison

9-8-72

ABSTRACT A preliminary Failure Mode and Effect Analysis (FMEA) was made on the overall 5-kwe system. A general discussion of the system and failure effect is given in addition to the tabulated FMEA and a primary block diagram of the system.

TI-652-260-001

TITLE 5-kwe REACTOR POWER DISTRIBUTIONS

AUTHOR D. S. Bost

6-26-72

ABSTRACT Power distributions representative of beginning and end of life are presented for the 5-kwe reactor. These distributions were obtained from DOT calculations in R,Z geometry. Reactivity worth of the sliding control segments was also obtained.

TI-652-260-002

TITLE 5-kwe REACTOR NUCLEAR PERFORMANCE ANALYSIS

AUTHOR D. S. Bost

9-19-72

ABSTRACT A summary is given of the nuclear analysis of the conceptual design 5-kwe reactor TE system.

AI-AEC-13066

TI-652-310-002

TITLE 5-kwe REFERENCE CONCEPTUAL DESIGN - SYSTEM HYDRUALICS

AUTHOR A. F. Lillie 9-25-72

ABSTRACT The hydraulic analyses, including power module hydraulics, for the 5-kwe Space Power System reference conceptual design (Dwg N652630001) are presented.

TI-652-310-003

TITLE MASS TRANSFER AND OXYGEN ANALYSIS IN THE SNAP 5-kwe GROUND TEST REACTOR

AUTHOR R. L. Eichelberger 1-9-73

ABSTRACT Mass transfer in the SNAP 5-kwe Ground Test Reactor System is discussed. It is shown that because the primary system consists of several different alloys, it is predisposed to be subject to mass transfer processes. For a given combination of materials, the rate at which mass transfer will occur depends upon the maximum temperature and the temperature difference in the loop, and upon the oxygen concentration of the NaK. In order that the system produce its rated power, the maximum temperature and the specified temperature difference must be obtained in operation, so that the only controllable factor affecting mass transfer is the oxygen impurity concentration in the NaK. Because the system does not include a cold trap, it is necessary to effect the control of the oxygen concentration during the NaK charging and cleanup operation. The cleanup operation should involve cold trapping with an auxiliary, removable unit, and it should be operated until the oxygen concentration is no more than 5 wppm, with the system at at least 600°F, and preferably at 800°F. Then the loading and cleanup unit can be removed and the reactor system sealed off.

TI-652-320-003

TITLE 5-kwe SYSTEM RADIATOR THERMAL ANALYSIS

AUTHOR L. D. Montgomery 8-7-72

ABSTRACT The 5-kwe system thermal radiator reference design is presented in two parts. The first portion describes the reference meteoroid armor equation used in preliminary design. The second portion describes the current reference design.

TI-652-320-005, Rev A

TITLE 5-kwe SYSTEM LOCKALLOY RADIATOR THERMAL ANALYSIS

AUTHOR L. Montgomery 9-13-72

ABSTRACT The reference design of the 5-kwe reactor system thermal radiator using Lockalloy as the fin material is presented in two parts. The first portion describes the reference meteoroid armor equation used in preliminary design. The second portion describes the selection of the current reference design.

AI-AEC-13066

TI-652-510-001

TITLE: 5-kwe SYSTEM MODEL AND TRANSIENTS

AUTHOR: F. J. Halfen

5-16-72

ABSTRACT: A system model has been prepared for the 5-kwe SNAP system with thermoelectric power conversion. This model includes descriptions of all the processes judged to significantly affect system temperature transients. Several startup transients and several protected accident transients were calculated using the model to provide transient data for design analysis purposes. The model and transients are documented in this report.

TI-652-510-002

TITLE: ADDITIONAL 5-kwe THERMOELECTRIC SYSTEM TEMPERATURE TRANSIENTS

AUTHOR: F. J. Halfen

5-25-72

ABSTRACT: Several additional system transients have been calculated for the 5-kwe TE system and are reported in this document. They include a startup transient with a reactivity rate of 0.005 β /sec, several startup accidents, a step reactivity insertion at full power, and a loss of electrical load. These data are intended for input to system design analyses and for possible use in the protected accident section of the safety report.

TI-652-610-001

TITLE: 5-kwe SYSTEM THERMAL LOSSES AND PIPING INSULATION PERFORMANCE

AUTHOR: J. A. Beoletto

9-19-72

ABSTRACT: Heat balances performed for components of the 5-kwe TE SNAP System during full power, steady-state operation are reported and summarized. Radiation shunt losses from the primary NaK system to the secondary, heat rejection loop require the application of a low-emittance foil on the primary loop piping to minimize thermal shunting.

TI-652-610-003

TITLE: PRESTART ORBITAL OPERATING CHARACTERISTICS

AUTHOR: J. Beoletto

12-4-72

ABSTRACT: The performance characteristics of the 5-kwe reactor-TE system were evaluated under prestart orbital operating conditions at a reference 600-n. mi orbit. Shroud temperatures and NaK temperatures were evaluated for maximum and minimum thermal input orbits assuming varying conditions of NaK flow and shroud absorptivity/emissivity ratios. To maintain desirable minimum NaK temperatures, a shroud α/ϵ ratio of 2.5 to 3.0 with approximately 5% NaK flow in the system is required.

TSR-652-640-002

TITLE: 5-kwe REACTOR THERMOELECTRIC SYSTEM DESIGN POINT AND CONFIGURATION MODIFICATION STUDY

AUTHOR: R. V. Anderson

9-20-72

ABSTRACT: A configuration and thermal performance statepoint had been established for the 5-kwe Reactor Thermoelectric System. Design modifications to this system have resulted from further analyses and trade studies of component performance parameters and designs. Preliminary margin assessments were incorporated into several component designs. The resultant system was established as the conceptual design. Associated performance and weight summaries were prepared for this system.

AI-AEC-13066

9. 150 kwe

NAA-SR-6735

TITLE: THE PRELIMINARY DESIGN OF A 150 kw SNAP SYSTEM

AUTHOR: Staff

1-22-62

ABSTRACT: The results of preliminary design study of a 150-kwe nuclear space power system are presented. A mission and application analysis is included showing the capabilities of this system over the design power range.

10. 25 kwe

AI-AEC-Memo-12717

TITLE: REFERENCE ZIRCONIUM HYDRIDE REACTOR THERMOELECTRIC SYSTEM

AUTHOR: Staff

6-15-69

ABSTRACT: This report presents the results of a 9-month engineering study of nuclear reactor-thermoelectric (TE) power systems. This general type of space power system, which is similar to the 500-watt (SNAP 10A) reactor-TE system, was flight-tested in space in 1965. Additional objectives set for this study were to:

- 1) Establish a conceptual design of a reference system suitable for adaption to either manned or unmanned space missions with a nominal power level of 25 kwe and a minimum lifetime of 20,000 hr.
- 2) Prepare parametric performance data over a suitable power range above and below 25 kwe to illustrate the trade-offs available between power, weight, area, temperature, etc.

AI-AEC-12839

TITLE: REACTOR-THERMOELECTRIC SYSTEM FOR NASA SPACE STATION

AUTHOR: J. D. Gylfe and R. A. Johnson

7-1-69

ABSTRACT: The results of a study regarding the design, integration, and operation of nuclear reactor-thermoelectric power systems for earth-orbiting manned space stations are presented.

AI-AEC-Memo-12840

TITLE: REACTOR-THERMOELECTRIC SYSTEM DEVELOPMENT PLANS

AUTHOR: J. D. Gylfe and R. A. Johnson

7-1-69

ABSTRACT: During GFY 1968 a joint study was made by NASA's Marshall Space Flight Center (MSFC) and the Atomic Energy Commission (AEC) contractor, Atomics International, to define a 25-kwe reactor-thermoelectric power system for an orbiting space station. This report describes the development plans. The technical results of this study are reported in AI-AEC-12839 and a report being prepared by MSFC, the former primarily describing the power system and the latter the integration considerations. One of the constraints on the study was that the system be ready for flight in 1975.

AI-AEC-13066

11. PSM-3

NAA-SR-11945, Volume I

TITLE: MERCURY RANKINE PROGRAM INTEGRATED SYSTEM TEST (PSM-3),
VOLUME I - SYSTEM OPERATIONS

AUTHORS: J. G. LeBlanc and F. W. Poucher 10-25-66

ABSTRACT: Tests conducted to verify the performance of the Mercury Rankine Power Conversion System under simulated space operational conditions are described. To simulate the significant zero-g effects on the system parameters, a flat horizontal version of the radiator-condenser was configured and utilized with the elevational location of the various components carefully determined. Initial tests were conducted on the horizontal radiator-condenser to determine its transient and steady-state heat transfer characteristics and later its interaction with the mercury pump of the Combined Rotating Unit (CRU). The PSM-3 was assembled from available SNAP 2 Power Conversion System (PCS) components with appropriate simulation made for the unavailable components, such as the thermoelectric NaK pump and initially, the turbine and mercury pump of the CRU. Data recorded during the extensive series of startup, shutdown, and steady-state tests are presented and evaluated. It is concluded that this relatively simple arrangement enabled the performance of the Mercury Rankine PCS under zero-g conditions to be measured, verifying the design of the major components, mainly the mercury boiler, CRU, and the radiator-condenser. Additional tests are recommended to assess off-design conditions and long-term endurance.

NAA-SR-11945, Volume II

TITLE: MERCURY RANKINE PROGRAM INTEGRATED SYSTEM TEST (PSM-3),
VOLUME II - DATA CORRELATION

AUTHORS: V. Heiskala, et al. 10-25-66

ABSTRACT: The test data from the tests described in Volume I are compared and evaluated against the analytically predicted performance of the Mercury Rankine system and its boiler, CRU, and the radiator-condenser. In general, close correlations were achieved. Where discrepancies occurred, shortcomings of the instrumentation utilized or the lack of sufficient sophistication in the analytical models are recognized. Further tests to generate the necessary data to improve the analytical methods are recommended.

12. Zirconium-Hydride

AI-AEC-13072

TITLE: DESIGN AND INTEGRATION OF ZIRCONIUM HYDRIDE REACTOR POWER
SYSTEMS

AUTHOR: J. H. VanOsdol 6-73

ABSTRACT: During the 1960's and 1970's, the SNAP Zirconium Hydride (ZrH) Reactor Program supported reactor and associated component development, system development, and a variety of application and integration studies. The purposes of this report are to: (1) summarize the key technical aspects of each of these components and subsystems, (2) present the technology status of each, (3) discuss operational characteristics of ZrH reactor systems, and (4) summarize the application studies which were conducted. The components and subsystems presented are: reactor, shielding, various power conversion systems, heat rejection systems, liquid metal components, and support structure. The applications summarized are: (1) NASA - lunar base, space station, space base, communications satellites, and space shuttle integration; (2) DOD - surveillance satellites and hardened missile base; and (3) General - terrestrial power systems and marine systems.

AI-AEC-13066

BLANK

C. REACTOR

1. Reactor, General

NAA-SR-3465

TITLE: SNAP 2 EXPERIMENTAL REACTOR HAZARDS REPORT

AUTHOR: R. R. Eggleston and G. L. Schmidt (Editors)

4-1-59

ABSTRACT: A description of the SNAP Experimental Reactor (SER), the site, and facility for the operation of the SER is presented. A number of accident studies are made and evaluated.

2. Reactor Design

NAA-SR-6207

TITLE: SNAP 8 REACTOR PRELIMINARY DESIGN SUMMARY

AUTHOR: H. Rosenberg (Editor)

6-15-61

ABSTRACT: The purpose of this report is to describe the present concept of the SNAP 8 reactor, reactor control system, safety systems, and shield and to outline the reasoning behind the design point selection.

NAA-SR-6561

TITLE: THERMAL DESIGN OF SNAP REACTORS

AUTHOR: P. D. Cohn

2-1-62

ABSTRACT: This report describes the analytical techniques used in the preliminary thermal design of the SNAP 8 and SNAP 2 reactors. These analytical procedures are applicable to most core designs which incorporate a close-packed array of fuel rods.

NAA-SR-6735

TITLE: THE PRELIMINARY DESIGN OF A 150 kw SNAP SYSTEM

AUTHOR: Staff

1-22-62

ABSTRACT: The results of preliminary design study of a 150-kwe nuclear space power system are presented. A mission and application analysis is included showing the capabilities of this system over the design power range.

NAA-SR-9062

TITLE: PRELIMINARY CONSIDERATION OF THE USE OF U^{233} IN SNAP REACTORS

AUTHOR: R. D. Doctor

1-29-64

ABSTRACT: The use of U^{233} in place of, and in mixture with, U^{235} in SNAP reactors fuels is considered. It was found that a reduction as great as 7% hydrogen density and 60% in uranium loading can be achieved. Presence of U^{232} as a contaminant in U^{233} may present a serious radiological health hazard. U^{233} handling problems are discussed.

AI-AEC-Memo-12716

TITLE: ZrH REACTOR DESIGN STUDIES

AUTHORS: E. J. Donovan and L. M. Maki

6-15-69

ABSTRACT: The results of a number of detailed design and performance studies conducted in the process of establishing the conceptual design of the Reference Zirconium Reactor are presented.

AI-AEC-13087

TITLE: ZIRCONIUM HYDRIDE REACTOR CORE HYDRAULIC STUDIES SUMMARY REPORT

AUTHOR: B. J. Ostermier

6-73

ABSTRACT: A summary is presented for the series of water-hydraulic tests performed to evaluate the flow characteristics of the various zirconium hydride reactors. Test models, test methods, instrumentation, and results are described. Tests were conducted to develop inlet plenum configurations and fuel element fin configurations. Inlet plenum configurations were evaluated on their ability to produce desired in-core flow distributions, and to provide a uniform pressure at the inlet grid plate surface without any undesirable flow characteristics. A series of inlet plenum tests was also conducted to determine any flow behavior which might have contributed to the operational problems of the S8DR. Fuel element fin configurations were evaluated on their ability to produce interchannel flow mixing within the reactor core. Pressure drop determinations were made for all plenum and fuel element fin configurations.

TI-653-220-002

TITLE: THERMAL ANALYSIS OF THE SPF VOID BACKED REFLECTOR ASSEMBLY

AUTHOR: E. M. Larson

11-10-71

ABSTRACT: The results of a study to assess the effect of cold wall and reflector emissivity on SPF reflector temperature are presented. The results indicate that cold wall and reflector emissivities of 0.6 will yield reflector temperatures under 1200°F.

ORR-696-13-001

TITLE: INTERIM DESIGN CRITERIA FOR THE REFERENCE ZrH REACTOR

AUTHOR: G. E. German

3-20-70

ABSTRACT: Design criteria for the reference ZrH reactor and the underlying considerations for many of the more obscure requirements are presented in this document. The scope includes general criteria tailored for space operations now being defined as well as detailed criteria considered necessary to meet the operational requirements described in specifications SS696N10001 and SS696N20001.

TI-696-13-006

TITLE: REACTOR DESIGN EVOLUTION, REFERENCE DESIGN ZrH REACTOR-TE

AUTHOR: H. E. Richter, Jr.

2-26-68

ABSTRACT: The selected reference design ZrH Reactor is described. Conceptual design arrangements studied are described and their effect on selection of the reference design is documented.

AI-AEC-13066

TI-696-24-027

TITLE: REVIEW OF METHODS FOR INTRODUCING BURNABLE POISON INTO THE
REFERENCE ZrH REACTOR

AUTHOR: T. G. Parker, Jr. 6-17-69

ABSTRACT: It has been decided to replace Gd^{155} with another rare earth as the burnable poison for the Reference ZrH Reactor. Different methods of introducing the burnable poison into the ZrH Reactor core are reviewed. Evaluation of these techniques indicates that vapor deposition on the fuel rod surface still has the greatest advantage.

TI-700-13-013

TITLE: IMPACT TEST DATA OF ZrH REACTOR SYSTEM

AUTHOR: S. L. Lee 3-4-70

ABSTRACT: As part of the post-impact disassembly studies of the ZrH Reactor, a series of impact tests have been conducted on the principal reactor materials. This report presents the basic equations used in determining the impact energy balance between the test specimens and the target at impact. With the aid of these equations, the energy absorption capacity of the materials tested is determined.

The effects of irradiation on Type 304 SS is presented indicating that the energy-absorbing capacity of the reactor vessel will not be affected by irradiation. Modeling laws are shown that indicate that it is possible to predict the post-impact configuration of the reactor by scale-model testing.

3. Reactor Nuclear Analysis

NAA-SR-3607

TITLE: SNAP-2 EXPERIMENTAL REACTOR PHYSICS ANALYSIS

AUTHOR: L. A. Wilson 6-15-59

ABSTRACT: This report presents the physics analysis of the SNAP 2 Experimental Reactor (SER). The reactor is fueled with highly enriched uranium-235 alloyed with zirconium hydride. The reactor coolant is eutectic alloy of sodium and potassium. The reflector is made of beryllium. The control elements consist of rotating semicylindrical drums of beryllium. The SER is a test of only the reactor portion of the SNAP-2 power plant.

NAA-SR-5991

TITLE: PRELIMINARY RESULTS OF THE SNAP-2 EXPERIMENTAL REACTOR

AUTHOR: M. W. Hulin 4-1-61

ABSTRACT: This report presents the operating history of the SNAP 2 Experimental Reactor (SER) and the preliminary results from the testing program. The total energy generated during the life of the reactor was 224,650 kilowatt hours. This is equivalent to approximately one-half year of full-power operation. The methods used to obtain the reactor parameters are also described. The experimental data obtained were generally in excellent agreement with calculated values. The principal comparisons are tabulated.

AI-AEC-13066

NAA-SR-7140

TITLE: TEMPERATURE COEFFICIENTS AND SPECTRA IN THE HYDRIDE MODERATED SNAP REACTORS

AUTHORS: J. Miller, R. L. Brehm, and W. J. Roberts 12-30-62

ABSTRACT: A summary of SNAP temperature coefficient measurements accomplished on the SNAP Experimental and Developmental Reactors is presented. Isothermal measurements, transient experiments, and power coefficient measurements are included. A new model for the energy transfer process in zirconium hydride is presented which reproduces measured spectra and results in temperature coefficients of the magnitude measured.

NAA-SR-7786

TITLE: KINETICS, STABILITY AND CONTROL, A SELECTED BIBLIOGRAPHY

AUTHOR: R. L. Johnson 3-15-63

ABSTRACT: This report is a bibliography of literature pertaining to nuclear reactor kinetics, control, and stability published prior to the autumn of 1962. The purpose of this bibliography is to provide a ready source of references by which analysis of the dynamic behavior of SNAP reactor systems can be planned, guided, and evaluated. As a result, the emphasis is on theory and calculation. A broader coverage of reactor kinetics is given in NAA-SR-3808, a bibliography of all papers on the subject published through July 1958.

NAA-SR-7938

TITLE: A REACTOR TRANSIENT HEAT TRANSFER MODEL

AUTHORS D. G. Mason and R. W. Winson 3-1-64

ABSTRACT: A transient heat transfer model of a SNAP 8 reactor core is described which can be programmed on a general purpose electronic analog computer. The transient responses of two models, both formulated from the physical system using the laws of conservation of mass and energy, are compared for step changes in power and inlet coolant temperature. One of the models is selected as having the desired transient accuracy in response to the input disturbances.

NAA-SR-9078

TITLE: SNAP 2 STARTUP ANALYSIS

AUTHOR: P. M. Gresho et al. 1-20-64

ABSTRACT: This report summarizes the calculational procedures and results which are used to define the orbital startup methods for the SNAP 2 unit.

NAA-SR-9087

TITLE: SNAP 2 PERFORMANCE ANALYSIS

AUTHOR: R. T. Lancet et al. 2-15-64

ABSTRACT: The most significant steady-state thermodynamic and fluid dynamic analyses of the SNAP 2, 3- to 5-kw nuclear power supply is presented. In addition to the general system and design point analysis, the detailed analysis of the thermal and fluid dynamic performance of the radiator-condenser and the thermal performance of the instrument compartment are presented.

NAA-SR-9368

TITLE: SNAP REACTOR HANDBOOK - TRANSIENT ANALYSIS

AUTHORS: W. A. Rhoades and W. B. Green

7-15-64

ABSTRACT: This report discusses some of the problems involved in the kinetics of uncontrolled power transients in SNAP reactors, and some of the techniques available for solving them. The report includes a review of space-independent kinetics equations for bare and reflected reactors, the more important feedback effects present in SNAP reactors, analytical solutions to the equations applicable in certain special cases, and a review of the computer codes available for solution of the equations in the general case.

NAA-SR-9626

TITLE: ANALYSIS OF THE RESPONSE OF THE SNAP 8 NUCLEAR SYSTEM DURING STARTUP OF THE POWER CONVERSION SYSTEM

AUTHOR: C. J. Code, Jr.

6-8-64

ABSTRACT: Detailed results of the parameter studies to determine the response of the SNAP 8 system to power conversion system startup are presented. The analog simulation is described and discussed.

NAA-SR-9646

TITLE: RESPONSE OF THE SNAP-8 REACTOR DURING AUTOMATIC STARTUP

AUTHOR: S. Birken

9-24-64

ABSTRACT: The SNAP 8 nuclear system was simulated using both digital and analog models to determine the response of the system to automatic startup. Parameter studies were made to determine the effects of startup source size, NaK coolant flow, temperature coefficients, and additional drum rates on both fuel temperature transients and start-up time.

NAA-SR-9754

TITLE: SNAP-10A REACTOR NUCLEAR ANALYSIS

AUTHORS: A. R. Dayes, W. A. Flynn, and J. P. Hawley

1-15-65

ABSTRACT: A physical description of the SNAP-10A reactor and a description of the calculational models employed for nuclear analysis are presented. Analytical methods and sources of nuclear constants are discussed. Where available, experimental results are compared to analytical results.

NAA-SR-9780

TITLE: ANALYSIS OF SNAPTRAN-3 DESTRUCTIVE EXPERIMENT

AUTHOR: W. A. Rhoades

3-31-67

ABSTRACT This report presents data obtained from the SNAPTRAN (SNAP Transient) -3 destructive experiment that was conducted at the National Reactor Testing Station, and discusses the analysis of the data from the standpoint of determining the nuclear energy release and reactivity feedback effects.

AI-AEC-13066

NAA-SR-10899

TITLE: AN INTRODUCTORY SURVEY OF THERMALIZATION THEORY FOR ZIRCONIUM HYDRIDE

AUTHOR: W. J. Roberts 4-20-65

ABSTRACT: A brief introduction to the main features of the theory of thermal neutron scattering is presented; and the approaches, or models, used specifically for hydrides by various investigators are surveyed. The hydrogen models considered are: (1) the rigid equilibrium point harmonic oscillator model; (2) its variant, the phonon model; and (3) the conduction-state, or periodic-potential wave-state model.

NAA-SR-Memo-12099

TITLE: SNAP 8 SUMMARY - S8ER and S8DS

AUTHOR: Staff 8-30-66

ABSTRACT: A brief summary is presented of the operational and post-operational results of the SNAP 8 Experimental Reactor (S8ER) and the relationship of these results with the SNAP 8 Developmental Reactor (S8DS) design.

NAA-SR-Memo-12562

TITLE: SNAP TEMPERATURE COEFFICIENT LITERATURE ABSTRACT

AUTHOR: L. A. Wilson 9-15-67

ABSTRACT: The experimental and theoretical literature on isothermal and partial temperature coefficients and power coefficients are reviewed and abstracted for SNAP reactors SER, S2DR, S10A-FS-3, SCA-4A, S8ER, and SNAPTRAN.

AI-AEC-12864

TITLE: SNAP 8 DEVELOPMENT REACTOR NUCLEAR ANALYSIS

AUTHOR: L. D. Swenson 10-31-69

ABSTRACT: The S8DR reactor nuclear analyses are described including the determination of reactivity requirements, nuclear control requirements, material nuclear control requirements, material nuclear worths, and system excess reactivity. Comparison and normalization of analytical calculations with experimental results are discussed. The S8DR has a goal of 12,000 hr of operation at 600 kw, with NaK coolant temperatures of 1100°F inlet and 1300°F outlet.

AI-AEC-12909

TITLE: REACTIVITY EFFECTS OF GADOLINIUM-155 ON A SNAP REACTOR

AUTHOR: B. J. Dray, et al. 4-15-70

ABSTRACT: The effects of gadolinium-155 on both the reactivity and the temperature coefficient of reactivity and the temperature coefficient of reactivity of a SNAP reactor were measured. This is part of a continuing program to assure SNAP reactor subcriticality on water immersion.

AI-AEC-13066

AI-AEC-13072

TITLE: DESIGN AND INTEGRATION OF ZIRCONIUM HYDRIDE REACTOR POWER SYSTEMS

AUTHOR: J. H. VanOsdol

6-73

ABSTRACT: During the 1960's and 1970's, the SNAP Zirconium Hydride (ZrH) Reactor Program supported reactor and associated component development, system development, and a variety of application and integration studies. The purposes of this report are to: (1) summarize the key technical aspects of each of these components and subsystems, (2) present the technology status of each, (3) discuss operational characteristics of ZrH reactor systems, and (4) summarize the application studies which were conducted. The components and subsystems presented are: reactor, shielding, various power conversion systems, heat rejection systems, liquid metal components, and support structure. The applications summarized are: (1) NASA - lunar base, space station, space base, communications satellites, and space shuttle integration; (2) DOD - surveillance satellites and hardened missile base; and (3) General - terrestrial power systems and marine systems.

AI-AEC-13077

TITLE: FUEL ELEMENT AND CORE BUNDLE, 5 kwe REACTOR THERMOELECTRIC SYSTEM

AUTHORS: L. D. Felten and E. Moody

6-73

ABSTRACT: This report discusses the design of the core bundle for the 5-kwe reactor. The design thermal power is 110 kw at 1200°F outlet temperature for 5 yr. Most of the thermal, hydraulic, and stress calculations have been completed for the core bundle. Operational stresses for the fuel elements show positive margins of safety. Launch vibration loads have yet to be specified so that the fuel slug vibration suppressor design can be verified.

AI-AEC-13086

TITLE: CONTROL WORTH OF SLIDING REFLECTORS FOR ZIRCONIUM HYDRIDE REACTORS

AUTHOR: D. S. Bost

6-73

ABSTRACT: Results of measured and calculated worth of sliding reflectors on the SCA-4A critical assembly are presented.

Two quadrants on opposite sides of SCA-4A were fitted with new tapered beryllium reflectors. The existing reflectors and control drums were used in two quadrants. The new reflectors were cut to allow measurement of the worth of sliding reflectors over a range of angles (30, 60, and 90 deg) and strokes (2, 2-3/4, and 3-1/2 in.).

A maximum worth of \$8.01 was obtained for both 90 deg segments, open 3-1/2 in. Two-dimensional calculations in R,Z geometry gave a worth of \$6.54.

AI-AEC-13087

TITLE: ZIRCONIUM HYDRIDE REACTOR CORE HYDRAULIC STUDIES SUMMARY REPORT

AUTHOR: B. J. Ostermier

6-73

ABSTRACT: A summary is presented for the series of water-hydraulic tests performed to evaluate the flow characteristics of the various zirconium hydride reactors. Test models, test methods, instrumentation, and results are described. Tests were conducted to develop inlet plenum configurations and fuel element fin configurations. Inlet plenum configurations were evaluated on their ability to produce desired in-core flow distributions, and to provide a uniform pressure at the inlet grid plate surface without any undesirable flow characteristics. A series of inlet plenum tests was also conducted to determine any flow behavior which might have contributed to the operational problems of the S8DR. Fuel element fin configurations were evaluated on their ability to produce interchannel flow mixing within the reactor core. Pressure drop determinations were made for all plenum and fuel element fin configurations.

AI-AEC-13066

TI-652-240-010, Rev A

TITLE: HOT CHANNEL FACTORS FOR 5-kwe TE REACTOR

AUTHOR: L. D. Felten 9-6-72

ABSTRACT: Uncertainty factors for the fuel and coolant temperatures are presented. The maximum individual channel outlet temperature uncertainty is 15.8°F and the maximum fuel temperature uncertainty is 21.0°F. Without the fuel swelling uncertainty, the fuel temperature uncertainty falls to 13.6°F.

TI-652-240-011

TITLE: FUEL ELEMENT GAP SELECTION FOR THE 5-kwe REACTOR

AUTHOR: H. Rood 9-20-72

ABSTRACT: The recommended fuel element gap for the 5-kwe reactor is 12 mils, cold, diametral. This provides space for fuel swelling, fuel segment bowing, dimensional tolerances, and hydrogen redistribution. In addition, enough margin is provided so that full circumferential fuel/cladding contact at end of life is very unlikely.

TI-652-260-001

TITLE: 5-kwe REACTOR POWER DISTRIBUTIONS

AUTHOR: D. S. Bost 6-26-72

ABSTRACT: Power distributions representative of beginning and end of life are presented for the 5-kwe reactor. These distributions were obtained from DOT calculations in R,Z geometry. Reactivity worth of the sliding control segments was also obtained.

TI-652-260-002

TITLE: 5-kwe REACTOR NUCLEAR PERFORMANCE ANALYSIS

AUTHOR: D. S. Bost 9-19-72

ABSTRACT: A summary is given of the nuclear analysis of the conceptual design 5-kwe reactor TE system.

TI-652-260-004

TITLE: PARTIAL TEMPERATURE COEFFICIENTS FOR THE 5-kwe REACTOR

AUTHOR: V. S. Lekach 2-8-73

ABSTRACT: Temperature coefficients for the 5-kwe reactor are calculated and compared to previous experiments and reactors. The recommended isothermal temperature coefficient is $-0.18\%/^{\circ}\text{F}$.

TI-652-510-001

TITLE: 5-kwe SYSTEM MODEL AND TRANSIENTS

AUTHOR: F. J. Halfen 5-16-72

ABSTRACT: A system model has been prepared for the 5-kwe SNAP system with thermoelectric power conversion. This model includes descriptions of all the processes judged to significantly affect system temperature transients. Several startup transients and several protected accident transients were calculated using the model to provide transient data for design analysis purposes. The model and transients are documented in this report.

AI-AEC-13066

TI-652-510-002

TITLE: ADDITIONAL 5-kwe THERMOELECTRIC SYSTEM TEMPERATURE TRANSIENTS

AUTHOR: F. J. Halfen

5-25-72

ABSTRACT: Several additional system transients have been calculated for the 5-kwe TE system and are reported in this document. They include a startup transient with a reactivity rate of 0.005 β /sec, several startup accidents, a step reactivity insertion at full power, and a loss of electrical load. These data are intended for input to system design analyses and for possible use in the protected accident section of the safety report.

TI-653-240-010

TITLE: CLUSTER/ASYMMETRY EFFECTS ON SPF REACTOR, 0.750 PITCH

AUTHOR: E. Moody

11-1-71

ABSTRACT: Eleven cluster/asymmetry configurations have been examined in an attempt to bound realistic thermal gradients for stress analysis. A maximum equivalent ΔT of 32°F was found in these cases. The study showed that asymmetry is the controlling factor in both magnitude and direction of cross-element ΔT .

TI-653-260-001

TITLE: COMPARISON OF CALCULATED AND EXPERIMENTAL REACTIVITIES OF SNAP REACTORS

AUTHOR: D. S. Bost

8-4-71

ABSTRACT: The reactivities of S8ER, the S8ER core in SCA-4B, S8DR, and SCA-4A were calculated using current methods and cross-section data for comparison with experiment. A variety of core loadings and reflector shim configurations were used, making a total of 12 calculations. They may be grouped into six near-critical and six significantly supercritical configurations. The mean error and standard deviation was $-0.11 \pm \$1.53$ for all calculations, $+0.51 \pm \$1.05$ for the near criticals, and $-0.73 \pm \$1.76$ for the supercritical.

TI-653-260-003

TITLE: EFFECT OF PREPOISONS ON THE NUCLEAR BEHAVIOR OF THE 450-kwt and 300-kwt ZrH REACTORS

AUTHOR: D. S. Bost

9-28-71

ABSTRACT: Results of nondimensional burnup calculations on 450-kwt and 300-kwt reactors are presented. Varying amounts and combinations of Sm, Gd¹⁵⁵, and Eu¹⁵¹ prepoisons were investigated.

TI-696-13-014

TITLE: NUCLEAR MODEL FOR A 295 ELEMENT REFERENCE ZrH REACTOR AND SHIELD, DRUM POISON ROTATED OUT

AUTHOR: E. H. Ottewitte

10-15-68

ABSTRACT: A two-dimensional nuclear model was prepared for current studies of a 295-element Reference ZrH Reactor and Shield.

AI-AEC-13066

TI-696-13-032

TITLE: REFERENCE 295 ELEMENT ZrH REACTOR STARTUP TRANSIENTS WITH FLOW RATE AND UA VARIATIONS

AUTHOR: F. J. Halfen

4-24-69

ABSTRACT: An analysis of startup transients for the 295-element Reference ZrH Reactor with a thermoelectric power conversion system has been made in an effort to determine the effects of S8DR-type UA variation on these transients. In general, the S8DR-type UA variation causes larger and more rapid temperature transients. These can be mitigated by increasing initial NaK flow or decreasing startup reactivity rate moderately.

TI-696-13-045

TITLE: POWER HISTORIES FOR 295 ELEMENT ZrH REACTOR DURING STATIC CONTROL DRIFTDOWN

AUTHOR: T. A. Clemmer

12-29-69

ABSTRACT: Power histories have been calculated for the 295-element Reference ZrH Reactor. The power histories include cases for varying loadings of burnable poisons and tritium-helium for EOL shutdown. Power histories generated may be considered conservative from the standpoint of safety analyses.

TI-696-13-050

TITLE: REACTOR PERFORMANCE CHANGES NECESSARY TO COUNTERACT THERMO-ELECTRIC CONVERTER DEGRADATION

AUTHOR: G. E. German

6-19-70

ABSTRACT: The effects that degrading thermo-electric (TE) converters have upon reactor performance are calculated. It is shown that beginning-of-life (BOL) reactor coolant outlet temperature and TE converter degradation rate are key design factors that must be properly set to avoid reactor operation at outputs and temperatures where accelerated reactor degradations occur. It is also shown that the rise time to reach a specific reactor coolant outlet temperature is independent of the TE converter conversion efficiency. It is concluded that 5-yr power systems using state-of-the-art TE converters are feasible if the proper BOL reactor coolant outlet temperature is selected. A data display suitable for adjusting BOL reactor temperature and TE converter degradation rate objective for 5-yr power systems is provided.

TI-696-13-125

TITLE: REFERENCE 259 ZrH REACTOR STARTUP TRANSIENTS THROUGH THE FIRST POWER BURST TO 300°F NaK OUTLET TEMPERATURE

AUTHOR: F. J. Halfen

1-9-69

ABSTRACT: An analysis of initial startup transients through the first power burst to 300°F NaK outlet temperature has been made for the Reference 295-element ZrH Reactor with a thermoelectric power conversion system. Allowable reactivity insertion rate during this phase of startup is reported as a function of initial NaK flowrate and temperature coefficients of reactivity. The reactor model used is also reported in some detail along with selected temperature transients at several locations in the system.

TI-696-160-001

TITLE: WATER FLOODING CRITICALITY STUDY FOR ZrH FLIGHT REACTOR

AUTHOR: R. V. Anderson

10-29-70

ABSTRACT: Five analytical criticality calculations were performed to study the effects of: (1) water reflection only (no core flooding), (2) water reflection with 10% core flooding, (3) water reflection with 35% flooding, (4) water reflection plus complete core flooding, and (5) the negative reactivity feedback associated with rapid core expansion induced by a destructive transient.

TI-696-20-001

TITLE: THERMAL ANALYSIS - ADVANCED ZIRCONIUM HYDRIDE REACTOR

AUTHOR: F. P. Beiriger, Jr.

5-8-68

ABSTRACT: Preliminary thermal analysis results are presented for the Advanced ZrH Reactor vessel, drum drive bearings, and shield at reactor power levels of 600 and 1200 kw.

TI-696-21-003

TITLE: ZrH REACTOR VESSEL STRESSES

AUTHOR: W. H. Dauterman

4-22-69

ABSTRACT: A summarization of the ZrH reactor vessel stress and design margins are presented.

TI-700-14-004

TITLE: THE INFLUENCE OF POISON MATERIALS ON THE PROMPT TEMPERATURE COEFFICIENT OF A SNAP REACTOR - THEORY VS EXPERIMENT

AUTHORS: B. J. Dray, W. Rhoades, and H. Rood

6-29-70

ABSTRACT: Recent transient experiments were performed to determine the effect of Gd^{155} , Sm^{149} , and Eu^{151} prepoison materials on the prompt temperature coefficient of the SNAPTRAN-1 machine. Calculations were performed using AI's 28-group nuclear cross-section library in a variant of the APC neutron transport code. These calculated coefficients are in rough agreement with the experiments between 200 and 500°F. Above 500°F, the calculations indicate coefficients which are too positive for all the poisoned configurations.

TI-759-130-001

TITLE: CONTROL OF ZrH REACTOR REACTIVITY PERTURBATIONS DURING ORBITAL MANEUVERS

AUTHOR: R. F. Audette

9-9-70

ABSTRACT: This analysis compares potential uncontrolled reactor transients with allowable transients for uninterrupted electrical power generation from a Brayton system, and presents a control scheme to limit transient reactor outlet temperatures to 1250°F for a system designed to operate at a nominal 1200°F reactor outlet. Potential uncontrolled transients could result in a reactor outlet temperature swing of $\pm 77^\circ\text{F}$ about a nominal 1200°F and a reactor power swing of +92 kw and -67 kw about a nominal 130 kw for the Brayton system.

TI-759-200-001

TITLE: CONCEPTUAL CORE MODEL FOR THE REACTOR CORE TEST

AUTHOR: L. D. Swenson

11-16-70

ABSTRACT: Several design options for the ZrH Flight System Reactor were investigated which involved tradeoffs of core excess reactivity, reactor control, coolant mixing, and cladding thickness. A design point was selected which is to be the basis for more detailed evaluation in the preliminary design phase. The selected design utilizes 295 elements with 0.670-in. element-to-element pitch, 32-mil-thick Incoloy cladding, 18.00-in.-long fuel meat, hydrogen content of 6.3×10^{22} atoms/cc fuel, 10.5 wt % uranium, and a spiraled fin configuration with alternate elements having fins with spiral to the right, spiral to the left, and no fin at all (R-L-N fin configuration). Fin height is 30 mils for the center region of the core and 15 mils for the outer region.

TI-759-200-002

TITLE: USE OF HASTELLOY X-280 IN REACTOR CORE TEST ALTERNATE CONCEPTUAL DESIGN

AUTHOR: L. D. Swenson

11-24-70

ABSTRACT: A cursory investigation was performed to assess the feasibility of using 20-mil-thick Hastelloy X-280 cladding in place of the 32-mil-thick Incoloy 800 in the conceptual core model for the reactor core test. Fuel and cladding dimensions are presented as are the extrapolated changes in core excess and control worth.

TI-759-200-008

TITLE: HASTELLOY X FUEL ELEMENT CREEP RELAXATION AND RESIDUAL EFFECTS

AUTHOR: R. A. Castle

1-28-71

ABSTRACT: A worst case, seven-element, asymmetric fuel, thermal environment was assumed and a creep relaxation analysis generated. The fuel element cladding is 0.020-in. Hastelloy X. The contact load decreased from 11.6 lb to 5.87 lb. in 100,000 hr. The residual stresses were then computed for various shutdown times.

TI-759-240-007

TITLE: ZrH REACTOR LATTICE SPACING (HEAT TRANSFER CONSIDERATIONS)

AUTHOR: L. D. Felten

9-25-70

ABSTRACT: Temperature calculations for a 295-element ZrH reactor at fuel element spacings from 0.010 to 0.065 in. showed a very small dependence of reactor temperature on element spacing. It was found that one variation in coolant channel area (2 zones) was sufficient to satisfactorily shape the radial flow profile for the core.

TI-759-240-012

TITLE: DETERMINATION OF EXCESS REACTIVITY AND CONTROL WORTH FOR THE CORE TEST REACTOR

AUTHORS: H. Rood and R. Roberts

10-12-70

ABSTRACT: One-dimensional scoping calculations were performed to determine the reactivity and control worth characteristics of several core and reflector configurations for the Core Test Reactor. A comparison was made of three reflector designs: (1) six S8DR drums, 4.5-in. radius; (2) seven S8DR drums, 4.5-in. radius; and (3) six new drums, 6.25-in. radius. Only the third option, six new drums, appears to have sufficient control and reactivity to provide adequate margins to achieve the design lifetime. Parametric studies of the nuclear effects of the internal reflector composition indicated that the Be/BeO area should be maximized in order to increase reactivity, and that the steel/nickel area should be minimized in order to increase reactivity and control.

TI-759-240-016

TITLE: EFFECT OF CLADDING WALL THICKNESS AND GAS GAP WIDTH ON CIRCUMFERENTIAL TEMPERATURE GRADIENTS AND FILM COEFFICIENTS FOR ROD BUNDLES HAVING A P/D = 1.00

AUTHOR: A. R. Marchese

10-29-70

ABSTRACT: For the case of NaK flowing in-line through a rod bundle with P/D = 1.00, calculations are presented showing the effect of cladding wall thickness and gas gap width on the azimuthal temperature gradient and convective film coefficient. Calculations are given for a typical set of test reactor conditions.

TI-759-240-023

TITLE: EFFECT OF P/D RATIO AND CLADDING THICKNESS ON THE 60 TEMPERATURE RIPPLE FOR S8DR AND THE TEST REACTOR

AUTHOR: A. R. Marchese

12-21-70

ABSTRACT: For the case of NaK flowing in-line through a bundle of SNAP-type fuel elements, calculations are presented showing the effects of pitch-to-diameter (P/D) ratio and the cladding wall thickness on the 60 temperature ripple for conditions of interest to both S8DR and the test reactor. The maximum 60 temperature ripple for the inner zone (P/D = 1.045) and outer zone (P/D = 1.022) of the test core are predicted to be 28 and 34°F, respectively. The effects of fins and fin-induced mixing are not included in the analysis.

TI-759-240-038

TITLE: NaK FILM COEFFICIENTS AND 60 TEMPERATURE RIPPLE FOR THE ZrH REFERENCE REACTOR

AUTHOR: L. D. Felten

4-1-71

ABSTRACT: The TAP computer program was used to model a 180-degree segment of a fuel cell of the reference reactor. Thermal analyses of nominal conditions were made to determine 60 temperature ripples and NaK film heat-transfer coefficients. Effects of cladding thickness and coolant mixing are treated.

AI-AEC-13066

4. Reactor Testing

NAA-SR-3620

TITLE: A SNAP-2 HYDRIDE-MODERATED CRITICAL ASSEMBLY

AUTHORS: A. W. Thiele and C. A. Guderjahn 6-1-59

ABSTRACT: A small, homogeneous, metal-hydride-moderated, critical assembly has been constructed to verify the feasibility of the SNAP 2 compact hydride-moderated reactor. The critical assembly is pseudospherical with a core made of powder-compacted homogeneous mixture of $ZrH_{1.88}$ and enriched UO_2 . The reflector is beryllium and graphite. The core is in two hemispheres, and criticality is achieved when the two halves are brought together.

NAA-SR-5610

TITLE: AN ENRICHED UO_2 -ZrH CRITICAL ASSEMBLY

AUTHOR: M. V. Davis, et al. 11-1-60

ABSTRACT: This was the first SNAP Critical Assembly (SCA-1) and consisted of two hemispheres having horizontal faces. The central part is cold pressed ZrH with 8 wt % enriched UO_2 . This is surrounded by ~1-in. beryllium followed by an outer reflector of graphite. The upper hemisphere is stationary, and the lower hemisphere is raised to achieve criticality. The SCA-1 was used to obtain nuclear data that was required for the design of the SNAP Experimental Reactor (SER).

NAA-SR-5619

TITLE: PHYSICS MEASUREMENTS OF THE SNAP EXPERIMENTAL REACTOR (SER)

AUTHOR: F. H. Clark, S. G. Wogulis, and M. V. Davis 1-30-61

ABSTRACT: Prior to operation of the SER at temperature and power and with NaK coolant present, a series of measurements were performed at room temperature and at powers less than 1 watt. These measurements were designed to establish the characteristics of the system under controlled conditions and to develop safe operating procedures. This report is a description of these measurements.

NAA-SR-5991

TITLE: PRELIMINARY RESULTS OF THE SNAP-2 EXPERIMENTAL REACTOR

AUTHOR: M. W. Hulin 4-1-61

ABSTRACT: This report presents the operating history of the SNAP 2 Experimental Reactor (SER) and the preliminary results from the testing program. The total energy generated during the life of the reactor was 224,650 kilowatt hours. This is equivalent to approximately one-half year of full-power operation. The methods used to obtain the reactor parameters are also described. The experimental data obtained were generally in excellent agreement with calculated values. The principal comparisons are tabulated.

NAA-SR-7088

TITLE THE FINAL REPORT ON THE SNAP 2 EXPERIMENTAL REACTOR (SER)
OPERATION AND TEST PROGRAM

AUTHORS J. P. Beall and M. W. Hulin 4-30-62

ABSTRACT This report presents the operating history of SNAP 2 Experimental Reactor (SER) from initial fuel loading until final disassembly and removal of fuel. The results of the testing program are also presented. The total energy generated during the life of the reactor is equivalent to approximately 6 months of full-power operation. The methods used to obtain the reactor parameters are also described. The experimental data obtained were generally in excellent agreement with calculated values.

NAA-SR-7261

TITLE SNAP-8 EXPERIMENTAL REACOR, A DESCRIPTION

AUTHOR Staff 6-15-62

ABSTRACT This report presents a description of the SNAP 8 Experimental Reactor (S8ER), including the facility, and associated process and control systems. The S8ER is a test of the core only, i. e. , without power conversion system equipment. The objectives of the test are to determine the operating characteristics of the reactor, and to demonstrate reactor performance.

NAA-SR-7368

TITLE SNAP SHIELD TEST EXPERIMENT REACTOR PHYSICS TESTS

AUTHORS R. L. Tomlinson, R. P. Johnson, and S. G. Wogulis 7-15-62

ABSTRACT The initial physics tests on the Shield Test Experiment reactor and the precriticality rod-drop test data are presented in this report.

NAA-SR-7786

TITLE KINETICS, STABILITY AND CONTROL, A SELECTED BIBLIOGRAPHY

AUTHOR R. L. Johnson 3-15-63

ABSTRACT This report is a bibliography of literature pertaining to nuclear reactor kinetics, control, and stability published prior to the autumn of 1962. The purpose of this bibliography is to provide a ready source of references by which analysis of the dynamic behavior of SNAP reactor systems can be planned, guided, and evaluated. As a result, the emphasis is on theory and calculation. A broader coverage of reactor kinetics is given in NAA-SR-3808, a bibliography of all papers on the subject published through July 1958.

NAA-SR-8295

TITLE FINAL REPORT ON THE SNAP 2 DEVELOPMENT REACTOR, S2DR, TEST PROGRAM

AUTHORS L. D. Felten, J. P. Beall, and J. H. Burton 4-3-64

ABSTRACT This report describes the test program for the SNAP-2 Development Reactor, summarizes the operating history of the reactor, and presents significant reactor engineering methods developed during the test program

AI-AEC-13066

NAA-SR-9642

TITLE: SNAP-8 EXPERIMENTAL REACTOR CRITICAL EXPERIMENT

AUTHOR: D. E. Crouter

9-15-64

ABSTRACT: This report describes the reactor engineering efforts and criticality studies involved in the SNAP-8 Experimental Reactor Critical Experiment.

NAA-SR-9646

TITLE: RESPONSE OF THE SNAP-8 REACTOR DURING AUTOMATIC STARTUP

AUTHOR: S. Birken

9-24-64

ABSTRACT: The SNAP 8 nuclear system was simulated using both digital and analog models to determine the response of the system to automatic startup. Parameter studies were made to determine the effects of startup source size, NaK coolant flow, temperature coefficients, and additional drum rates on both fuel temperature transients and start-up time.

NAA-SR-9871

TITLE: SNAP CRITICAL ASSEMBLY - 4B PHASE III WATER IMMERSION EXPERIMENTS

AUTHOR: S. Yee

12-1-64

ABSTRACT: The Phase III SNAP Critical Assembly - 4B (SCA-4B) experiments demonstrated the effectiveness of: (1) a void-poison sleeve (shipping sleeve) in maintaining a bare core subcritical when it is internally flooded and fully reflected with water and (2) void-poison filler blocks in maintaining a beryllium-reflected core subcritical when it is submerged in, but not internally flooded with, water.

NAA-SR-10903

TITLE: SNAP 8 EXPERIMENTAL REACTOR OPERATIONS AND TEST RESULTS

AUTHORS: L. M. Fead, L. D. Felten, and V. L. Rooney

6-28-65

ABSTRACT: The test results of the SNAP 8 Experimental Reactor are summarized in this report. The test program provided experimental evaluations of the reactor performance characteristics, including power and isothermal temperature coefficients, poisoning effects, and hydrogen loss reactivity effects.

NAA-SR-11175

TITLE: STARTUP AND OPERATION OF THE ONE-MEGAWATT SHIELD TEST AND IRRADIATION REACTOR

AUTHORS: K. G. Randen and K. G. Golliher

3-25-66

ABSTRACT: This report presents reactor physics data on the Shield Test and Irradiation Reactor (STIR). MTR-type fuel elements were replaced in the Shield Test Experiment (STE) facility and the reactor power was increased from 50 kw to 1 Mwt. The STIR is to be used for shielding experiments. In addition, it will be used in support of a number of other programs in such areas as radiation damage, reactor physics, fuel material research, radiochemistry, and instrumentation development.

AI-AEC-13066

NAA-SR-11303

TITLE ARGON GENERATION IN NaK-COOLED REACTORS

AUTHOR S. L. Rosen

11-10-65

ABSTRACT Argon-41, a gamma emitter, was experimentally observed in the primary coolant system of the SNAP 8 Experimental Reactor. A fast neutron reaction with the potassium-41 in the NaK-78 eutectic alloy coolant is believed to be the mechanism for the generation of the argon-41. This mechanism led to the prediction of the presence of argon-39. This prediction was verified experimentally as described in this report.

NAA-SR-11520

TITLE DISASSEMBLY AND POSTOPERATION EXAMINATION OF THE SNAP 8 EXPERIMENTAL REACTOR

AUTHOR R. E. Dyer

ABSTRACT A discussion is presented of the shutdown and disassembly of the SNAP 8 Experimental Reactor and of the postoperation evaluation of the heat transfer system, the reflectors, and portions of the core.

NAA-SR-11528

TITLE CAPABILITIES OF THE 1-Mw SHIELD TEST AND IRRADIATION REACTOR

AUTHORS K. G. Golliher and K. G. Randen

11-10-65

ABSTRACT This report describes the physical layout of the Shield Test and Irradiation Reactor (STIR) facility. This facility is designed to provide a flexible source of neutrons and gamma rays for experimental work on shield development, for irradiation damage studies on electronic systems, and for the nuclear qualification of SNAP components. Neutron fluxes and gamma ray dose rates at various locations are presented.

NAA-SR-Memo-12099

TITLE SNAP 8 SUMMARY - S8ER and S8DS

AUTHOR: Staff

8-30-66

ABSTRACT: A brief summary is presented of the operational and post-operational results of the SNAP 8 Experimental Reactor (S8ER) and the relationship of these results with the SNAP 8 Developmental Reactor (S8DS) design.

NAA-SR-Memo-12210

TITLE SNAP 8 EXPERIMENTAL REACTOR FUEL ELEMENT BEHAVIOR - ATOMICS INTERNATIONAL TASK FORCE REVIEW

AUTHOR: H. Pearlman (Chairman)

11-30-66

ABSTRACT: The task force scope was to review, evaluate, and make recommendations on Atomics International post-irradiation examination of SNAP 8 Experimental Reactor (S8ER) fuel elements. Emphasis was on density changes, dimensional changes, hydrogen compensation, and microstructure. The report includes those evaluations, recommendations, and other findings resulting from this task force review.

AI-AEC-13066

NAA-SR-12504

TITLE: THE DISASSEMBLY AND POSTOPERATION COMPONENT EXAMINATION OF THE SNAP 10A FS-3

AUTHORS: J. P. Dooley and J. P. Beall 9-15-67

ABSTRACT: The SNAP 10A Flight System was operated as a ground demonstration test of the SNAP 10A system. The test was conducted by Atomics International at the Santa Susana Field Test Laboratory for the Atomic Energy Commission. After operating continuously for over 10,000 hr, the test was terminated, the system was disassembled, and critical system components were examined to determine the effect of the system operating environment. This report covers the disassembly of the system and the postoperation examination of the system components, with the exception of the fuel.

AI-AEC-12864

TITLE: SNAP 8 DEVELOPMENT REACTOR NUCLEAR ANALYSIS

AUTHOR: L. D. Swenson 10-31-69

ABSTRACT: The S8DR reactor nuclear analyses are described including the determination of reactivity requirements, nuclear control requirements, material nuclear control requirements, material nuclear worths, and system excess reactivity. Comparison and normalization of analytical calculations with experimental results are discussed. The S8DR has a goal of 12,000 hr of operation at 600 kw, with NaK coolant temperatures of 1100°F inlet and 1300°F outlet.

AI-AEC-13071

TITLE: SUMMARY OF SNAP 8 DEVELOPMENTAL REACTOR (S8DR) OPERATIONS

AUTHORS: L. D. Felten and H. D. May 6-73

ABSTRACT: The SNAP 8 Developmental Reactor (S8DR) achieved criticality on June 22, 1968. Power operation was initiated in January of 1969, and continued throughout the year. Final shutdown was in December of 1969. The reactor logged 6688 hr at nominal 600 kw, and 429 hr at 1000 kw. Total energy release was 4.35×10^6 kwh thermal.

AI-AEC-13086

TITLE: CONTROL WORTH OF SLIDING REFLECTORS FOR ZIRCONIUM HYDRIDE REACTORS

AUTHOR: D. S. Bost 6-73

ABSTRACT: Results of measured and calculated worth of sliding reflectors on the SCA-4A critical assembly are presented.

Two quadrants on opposite sides of SCA-4A were fitted with new tapered beryllium reflectors. The existing reflectors and control drums were used in two quadrants. The new reflectors were cut to allow measurement of the worth of sliding reflectors over a range of angles (30, 60, and 90 deg) and strokes (2, 2-3/4, and 3-1/2 in.).

A maximum worth of \$8.01 was obtained for both 90 deg segments, open 3-1/2 in. Two-dimensional calculations in R,Z geometry gave a worth of \$6.54.

AI-AEC-13066

TR-652-360-001

TITLE: NaK HEAT TRANSFER STUDY FOR A 19-ROD BUNDLE (P/D = 1.05)
EMPLOYING HELICAL-FIN SPACERS

AUTHOR: A. R. Marchese

7-26-72

ABSTRACT: An experimental heat transfer study is reported for the case of longitudinal flow of a liquid metal (NaK-78) through a closely spaced rod bundle. The rod bundle consisted of 19 electrically heated elements which were arranged on hexagonal arrays having a pitch-to-diameter (P/D) ratio of 1.05. Helical-fin spacers were attached to the cladding tubes to promote increased intrachannel turbulence and interchannel mixing, or crossflow. In addition, the fins served to maintain symmetric and positive rod spacing. Rod azimuthal temperature distributions were determined from thermocouples which were welded to the inside cladding tubes. Instrumentation was also provided to measure the coolant temperature distribution at the inlet and outlet to the rod bundle, the power supplied to each rod, and the coolant flowrate. The instrumentation provided the necessary measurements for determining the rod-average heat transfer coefficient.

Heat transfer results for the cladding circumferential temperature distributions and the rod-average film coefficients were obtained over a range of heating and flow conditions of interest to a compact ZrH reactor core. Because of the increased intrachannel turbulence created by the helical-fins, negligible rod azimuthal temperature variations and significantly improved rod-average film coefficients were observed. In the flow range investigated, the heat transfer results were found to be independent of the Peclet number. In addition, the effect of fin-induced interchannel mixing, or crossflow, was to significantly reduce coolant channel-to-channel temperature differences. The increased mixing should do a great deal in reducing cross fuel element ΔT 's and the resulting bending stresses in an actual ZrH reactor core.

TI-696-13-003

TITLE: NORMALIZED POISON-BACK CONTROL DRUM WORTH FOR THE REFERENCE
DESIGN ZrH REACTOR

AUTHOR: H. Rood

2-1-68

ABSTRACT: The worth of ten 4.5-in.-diameter poison-backed drums around the Reference Advanced ZrH Reactor is estimated to be $\$10.00 \pm \1.70 . These drums consist of a 1.5-in.-thick poison segment of 60 vol % Eu_2O_3 in nickel, and a 3.0-in.-thick reflecting segment of BeO.

TI-696-13-013

TITLE: REACTIVITY EFFECTS OF GADOLINIUM-155 ON A SNAP REACTOR

AUTHORS: W. H. Heneveld and V. A. Swanson

12-23-69

ABSTRACT: Results of experimental measurements of reactivity effects of various concentrations of gadolinium-155 on a SNAP reactor are presented. Included are reactivity worth measurements with a water-immersed core, temperature defect measurements, and transient test results in the SNAPTRAN machine.

TI-759-510-001

TITLE: PRELIMINARY DESCRIPTION OF AUTOMATION SYSTEM FOR BLDG T059

AUTHOR: L. S. Beller

10-12-70

ABSTRACT: A summary conceptual design and proposed specification is presented for automating major functions of the ZRT in Bldg T059.

AI-AEC-13066

5. Fuel Element Development

NAA-SR-4831

TITLE: CLADDING DEVELOPMENT FOR THE SER FUEL ELEMENTS

AUTHORS: J. Romero and S. Elchyshyn

2-15-61

ABSTRACT: Cladding and hydrogen-barrier materials were evaluated for use on SER fuel elements. Factors considered were hydrogen retention, coatability, weldability, availability, and sodium corrosion. The necessary test equipment was designed and built and the test techniques were developed. A glass coating, Solaramic, applied to the inside of Type 347 stainless steel tubing was selected for the SER fuel elements. Processes were developed for the SER core fabrication.

NAA-SR-5413

TITLE: COMPATIBILITY OF SNAP FUEL AND CLAD MATERIALS

AUTHOR: J. K. Balkwill

10-15-60

ABSTRACT: Sample buttons of materials of interest for SNAP fuel elements were held at temperatures up to 1600°F in hydrogen under a 20-psi load for 200+ hr. The hydrided Zr-U alloys formed significant diffusion zones with Hastelloy B, Inconel X, Type 347 stainless steel, gold, and copper at 1200°F, but did not alloy with molybdenum, beryllium, or nickel. At 1600°F, diffusion was also found between the fuel, molybdenum, and nickel. The fuel did not react with Solaramic S-1435A glass at temperatures of 1200 and 1500°F. At 1300°F, beryllium formed a diffusion zone with Hastelloy B and Inconel X and also reacted with Solaramic, while BeO did not react with any of these.

NAA-SR-8287

TITLE: CLADDING AND ASSEMBLY DEVELOPMENT FOR THE S8ER FUEL ELEMENTS

AUTHOR: W. F. Dennison

11-15-63

ABSTRACT: Several end closure designs were examined and one was selected to meet the temperature and hydrogen loss rate requirements of the S8ER reactor. A semi-automatic process was developed for making the ceramic seals in the end closure.

NAA-SR-9953

TITLE: SNAP 8 DRM-1 FUEL ELEMENT END CLOSURE DEVELOPMENT

AUTHORS: W. F. Dennison and C. P. Messina

12-15-64

ABSTRACT: The description and the test results of four designs of an enclosure for the SNAP 8 fuel element assembly are discussed. These designs, two modifications of the S8ER and two new, were prepared to alleviate the fuel element problems encountered on the original S8ER. The fabrication yield and permeation test results on samples of the four designs are presented to support the selection of a new design as reference for the SNAP 8 fuel element. The permeation rate on 25 production fuel element assemblies built for the S8DR with the reference design enclosure are reported as ranging from 0.12 to 0.52 cc/hr at 1400°F, well below the allowable limit of 1.50 cc/hr.

NAA-SR-10482

TITLE: RAPID PULSE HEATING OF MODIFIED ZIRCONIUM-URANIUM HYDRIDES TO DESTRUCTION

AUTHOR: R. A. Finch

7-25-65

ABSTRACT: Twenty-one samples of zirconium - 10 wt% uranium hydride were pulse heated to destruction to study the behavioral characteristics of the fuel when it is subjected to severe thermal transients. The duration of the pulses ranged from 28 to 450 msec with initial heating rates from 7,000 to 100,000°C/sec. Sample compositions ranged from 0.5 to 1.81 H/Zr. These tests verified the theoretical predictions of failure mechanisms.

NAA-SR-10515

TITLE: FISSION GAS RELEASE FROM LIGHTLY IRRADIATED SNAP 8 FUEL MATERIALS

AUTHORS: A. H. Willis and A. J. Moses

9-15-65

ABSTRACT: SNAP 8 fuel materials were studied by low-level irradiation followed by annealing treatments. From these tests, it is concluded that the $\epsilon \rightarrow \delta$ phase transformation does not promote gas release, but the $\delta \rightarrow \beta$, $\alpha \rightarrow \beta$, and $\alpha \rightarrow \delta$ phase transformations cause massive gas release. When phase transformations do occur, sudden fission gas release can be anticipated for any reasonable burnup level. This suggests a swelling region at high burnup levels.

AI-AEC-Memo-12653

TITLE: INCOLOY 800 - CLADDING BARRIER DEVELOPMENT TESTS

AUTHOR: J. L. Nummelin

2-16-68

ABSTRACT: Commercial grade Incoloy 800 was evaluated as a strength member and substrate for the hydrogen-barrier coating for SNAP reactor fuel elements. Adherence of the ceramic coating, hydrogen permeation characteristics, and short-term tensile properties were within acceptable limits for SNAP cladding materials. The feasibility of a butt-weld end closure of the coated cladding was also demonstrated.

AI-AEC-Memo-12784

TITLE: SNAP PROGRAM REVIEW, WASHINGTON, D.C., JANUARY 30, 1969

AUTHOR: Staff

-69

ABSTRACT: This report is a briefing book for the presentation of the SNAP 8 Developmental Reactor operating results, fuel development test results, and intrinsic water subcriticality test results.

AI-AEC-13066

AI-AEC-13082

TITLE: PROCESS DEVELOPMENT AND FABRICATION OF ZIRCONIUM HYDRIDE
HASTELLOY X CLAD FUEL ELEMENTS

AUTHOR: T. G. Parker, Jr.

6-73

ABSTRACT: This report summarized the development work performed, and the processes established, for the fabrication of Hastelloy X clad, U-ZrH_x fuel elements. Processes described include: U-Zr alloy melting and extrusion, U-Zr hydriding, Gd₂O₃ vapor deposition on U-ZrH_x fuel rods, machining helical fins in Hastelloy X cladding, Hastelloy X electron beam welding, ceramic coating of Hastelloy X, and final assembly operations. Process and test data on developmental fuel elements are presented, which demonstrate that the dimensional and performance test requirements of the 5-kwe Zirconium Hydride Reactor could be met.

AI-AEC-13083

TITLE: HASTELLOY X CLADDING MATERIAL EVALUATION

AUTHORS: K. E. Moore, R. G. Brengle, and T. G. Parker, Jr.

6-73

ABSTRACT: Hastelloy X is the reference fuel element cladding for the 5-kwe space power reactor. The effects of fuel element processing, and exposure to the reactor environment, were evaluated by conducting mechanical property and corrosion tests on three heats of Hastelloy X in various heat-treat conditions. Mechanical properties of processed material are in good agreement with literature values. Some ductility loss was observed in tensile tests. This is attributed to interstitial contamination resulting from the SCB ceramic hydrogen barrier coating process. Hastelloy X samples were exposed to flowing 1400°F NaK in a loop operated with a 90°F ΔT to obtain accelerated corrosion effects. The exposed surfaces of these samples were affected to a depth of ~2 mil after 2000 hr. The affected area had lost chromium, and contained intergranular microvoiding. Biaxial stress-rupture tests, performed in flowing NaK at 1300°F, indicate no significant effect on properties.

Creep tests were performed on unirradiated and irradiated material. Irradiated material showed slightly greater creep strength; ductilities were 1.0% or higher.

Transmission electron microscopy was performed on processed and aged Hastelloy X. Only slight differences in precipitation reactions could be attributed to the processing environment. The predominant phase resulting from thermal aging at 1200 and 1300°F was found to be M₆C.

AI-AEC-13066

AI-AEC-13085

TITLE: BURNABLE POISON DEPOSITION ON ZIRCONIUM HYDRIDE FUEL - PROCESS DEVELOPMENT AND EVALUATION

AUTHOR: R. Van Houten 6-73

ABSTRACT: This report reviews the development and evaluation of a high-vacuum process for depositing thin rare earth oxide burnable poison films on fuel rods. The burnable poison films are formed by controlled-rate, electron-beam-heated, high-vacuum evaporation of the oxides from water-cooled crucibles. The report reviews poison placement screening criteria, recommended coating and measuring equipment performance, and swelling and aging test results. All test results for Gd_2O_3 , Er_2O_3 , and Y_2O_3 coatings are promising. Tests extend to 18,584 hr at 1300°F or higher.

TI-696-24-045

TITLE: He and H₂ FREE MOLECULE AND CONTINUUM HEAT CONDUCTION

AUTHOR: E. Moody 11-26-69

ABSTRACT: Gas conduction characteristics in the regime pertinent to SNAP 8 technology are derived using current values of mean-free path lengths and continuum conductances. Helium is included to permit determination of loading characteristics for startup conduction. The effect of the helium on gas gap conduction during steady-state operation is considered and found to be fairly significant, even for low-pressure loading at end of life.

6. Fuel Element Analysis

NAA-SR-7752

TITLE: A METHOD FOR DIFFERENTIATING BETWEEN MATERIALS CONTAINING NORMAL AND FULLY ENRICHED URANIUM

AUTHORS: R. C. Barry and A. R. Mooers 4-15-63

ABSTRACT: A system to allow rapid discrimination of whether a uranium-zirconium alloy contains normal or fully enriched uranium was developed. This autoradiometric method is based on the fact that the U^{238} decay chain includes emission of a 2.31-Mev beta particle which is more easily absorbed by an aluminum filter than gamma photons emitted by U^{235} . The instrumentation required and the procedures employed should be readily adaptable to other uranium-bearing compounds and alloys.

NAA-SR-9600, VOL II

TITLE: NONDESTRUCTIVE DETERMINATION OF URANIUM-235 CONTENT OF SNAP FUEL RODS, VOL II, STATISTICAL METHODS FOR EVALUATING DATA OBTAINED FROM COUNTING EXPERIMENTS

AUTHORS: M. A. Greenfield, C. T. Nelson, and R. R. Koontz 7-15-64

ABSTRACT: This report describes a statistical method for evaluating the homogeneity of U^{235} in reactor fuel rods as the basis of measuring the characteristic radiation emitted by this uranium isotope. Analysis of SNAP fuel rods showed that the counting instrument was capable of determining the U^{235} content for each rod within 0.1% and changes in concentration of 0.3% at a 99% level of confidence.

AI-AEC-13066

NAA-SR-9600, VOL III

TITLE: NONDESTRUCTIVE DETERMINATION OF U^{235} CONTENT OF SNAP FUEL RODS, VOL III, EQUIPMENT DESCRIPTION AND PRODUCTION QUALITY CONTROL APPLICATIONS

AUTHORS: A. R. Mooers and R. C. Barry 5-15-65

ABSTRACT: An autoradiometric test method and associated production inspection equipment have been developed for nondestructively measuring the incremental and total U^{235} content in cylindrical SNAP fuel rods. A prototype single-channel analyzer scintillation counting system and special faced rod scanning mechanism are described. A statistical comparison of the autoradiometric and destructive chemical methods disclosed the correlation coefficient between the two methods to be 0.976. The precision of the autoradiometric method was 0.023 wt % uranium for fully enriched zirconium - 10 wt % uranium hydride.

NAA-SR-Memo-12097

TITLE: SNAP 8 FUEL ELEMENT STRESS BRIEFING, JULY 19 AND 20, 1966, CANOGA PARK, CALIFORNIA

AUTHOR: Staff 8-5-66

ABSTRACT: This report is a briefing book for the presentation of the status of SNAP 8 fuel element development, the test results from the S8ER and in-pile irradiations, and the fuel element stress analysis.

NAA-SR-Memo-12122

TITLE: SNAP 8 BRIEFING, WASHINGTON, D. C., AUGUST 15-16, 1966

AUTHOR: Staff 8-31-66

ABSTRACT: This report is a briefing book presenting the results of the SNAP 8 fuel element stress analysis, SNAP 8 fuel studies from the S8ER, and in-pile fuel irradiations.

NAA-SR-Memo-12210

TITLE: SNAP 8 EXPERIMENTAL REACTOR FUEL ELEMENT BEHAVIOR - ATOMICS INTERNATIONAL TASK FORCE REVIEW

AUTHOR: H. Pearlman (Chairman) 11-30-66

ABSTRACT: The task force scope was to review, evaluate, and make recommendations on Atomics International post-irradiation examination of SNAP 8 Experimental Reactor (S8ER) fuel elements. Emphasis was on density changes, dimensional changer, hydrogen compensation, and microstructure. The report includes those evaluations, recommendations, and other findings resulting from this task force review.

NAA-SR-Memo-12272

TITLE: SNAP 8 BRIEFING TO AEC WASHINGTON, D. C., DECEMBER 1, 1966

AUTHOR: Staff 12-14-66

ABSTRACT: This report is a briefing book presenting information on the SNAP 8 fuel swelling analysis, fuel element life behavior, core design, Hastelloy N materials data, cladding stresses, and the S8DR test program.

NAA-SR-Memo-12384

TITLE: DETAILED THERMAL ANALYSIS OF SNAP REACTOR FUEL ELEMENTS INCLUDING ASYMMETRY AND SPACING EFFECTS

AUTHOR: P. M. Magee

3-20-67

ABSTRACT: The detailed heat transfer characteristics of fuel elements in tightly packed, liquid-metal-cooled SNAP reactors are analyzed. The effects of element spacing, gap conductance, cladding thickness, and fuel rod asymmetry within the cladding are investigated.

NAA-SR-Memo-12507

TITLE: PERFORMANCE OF THE S8DR ZrH FUEL ELEMENTS

AUTHORS: J. G. Asquith, G. W. Meyers, and W. J. Roberts

6-20-68

ABSTRACT: A summary of the performance of the S8DR ZrH fuel elements is presented. This report is divided into four topics: fuel swelling correlation, S8DR cladding performance, fuel element in-core performance, and preproduction fuel element results.

TI-652-210-004

TITLE: STRUCTURAL ANALYSIS OF FUEL ELEMENT CLADDING FOR LONG TERM FUEL SLUG GROWTH

AUTHORS: R. E. Leary and J. Meertens

2-5-73

ABSTRACT: An elastic and creep structural analysis of the Hastelloy X cladding of the 1-in.-diameter fuel element has been conducted. The distortion of the fuel element is calculated from a projected 5-yr swelling of the fuel slug which is assumed initially offset laterally in the cladding. The creep strain capacity of the cladding is found to be structurally adequate under the assumed loading criteria. Several additional loading criteria and possible structural relief mechanisms are proposed for further evaluation.

TI-652-240-005

TITLE: ANALYSIS OF OFFSET FUEL SLUG STRAIN TEST DATA

AUTHOR: R. A. Castle

7-26-72

ABSTRACT: The fuel element test data from the offset fuel slug (bearing on the cladding inner surface) stress test is analyzed. Extrapolation of these data from the 0.540 ID by 0.020 tube tested to the current design fuel element tube, 0.9485 ID by 0.025, is accomplished by separating the stresses into their components. For a hypothetical 30-lb side load directly under the clad, the clad stresses are 17,700-psi tension on the outer surface, -12,400-psi compression on the inner surface, and a bearing pressure of approximately 1000 psi. Ninety degrees from the load point the stresses are $\pm 39,400$ psi peak. The stresses at 90 degrees command concern because of the tensile stress induced into the SCB coating on the inner surface of the fuel element.

TI-652-240-008

TITLE: FUEL ELEMENT COMPUTERIZED STRESS ANALYSIS WITH CORE FORCE BALANCE

AUTHOR: R. A. Castle

8-25-72

ABSTRACT: A computer program was written to compute the fuel element stresses from the GEOM heat transfer code output. The forces on all 85 elements are balanced. Then the detailed bending six-theta and local contact load stresses are computed for each element and the associated margin of safety is calculated.

AI-AEC-13066

TI-652-240-012

TITLE: THERMAL PERFORMANCE AND FUEL ELEMENT STRESS - CONCEPTUAL DESIGN

AUTHOR: E. Moody 10-3-72

ABSTRACT: The reactor performance and fuel element stresses are analyzed over a range of operating conditions. The effects of manufacturing bow, clearances, and fuel asymmetry are considered. In analysis of statistically random parameter variations, element loads and stresses were quite low. Analysis of forced worst case conditions still showed substantial margin for all but neutral edge elements with half the core unloaded. Use of hot end clearance on edge neutrals eliminated the high loads providing a highly adequate core design from a stress standpoint.

TI-652-240-014

TITLE: FUEL ELEMENT STRESS ANALYSIS STATUS

AUTHOR: R. A. Castle 9-25-72

ABSTRACT: The stress analysis for the 5-kwe TE System Reactor Fuel Element is presented. The residual manufacturing stresses, operational stresses utilizing the 85-element core force balance technique, creep, and fatigue analyses are presented.

TI-652-240-016

TITLE: LOADS INTRODUCED IN THE COLLAR BUTTON DEVICE BY FUEL SWELLING AND CREEP RELAXATION

AUTHOR: C. T. Kudija, Jr. 12-1-72

ABSTRACT: Load-vs-time graphs are presented for several designs of the Collar Button Device (CBD). Effects, such as pre-kink magnitude and retarded fuel swelling under load, are investigated.

TI-653-240-019

TITLE: FUEL ELEMENT STRESS ANALYSIS STATUS FOR SPF REACTOR CONCEPTUAL DESIGN

AUTHOR: R. A. Castle 12-3-71

ABSTRACT: A summary of the fuel element stress analysis conducted to date for the SPF Reactor conceptual design is presented. Two basic fuel element configurations were analyzed: one is a 0.027-in. wall thickness Hastelloy X design, the other a 0.032-in. wall thickness Incoloy 800 design. Both elements are interchangeable in the same core assembly. The analysis to date shows that the structural integrity of either element is maintained through the expected environment.

TI-653-240-027, Rev C

TITLE: DEVELOPMENT OF A FORMULA FOR CALCULATING ZIRCONIUM HYDRIDE FUEL SWELLING

AUTHOR: W. J. Roberts 6-7-72

ABSTRACT: A model is developed to describe irradiation-induced growth in zirconium hydride or for heavy ion accelerator irradiation of structural materials. The model is based on the phenomenon of volume saturation occurring as a result of annealing of irradiation-produced dislocations. The model was proposed by Hudson, Mazey, and Nelson for swelling in Ni irradiated with C^{++} ions. The model predicts a strong temperature and weak flux sensitivity. A sensitivity is shown to lower temperature, higher power operation following an initial steady operation.

AI-AEC-13066

TI-759-240-031

TITLE: FUEL ELEMENT VIBRATION STUDY - DESIGN AND ANALYSIS OF A PRE-KINKED COLUMN BUCKLING DEVICE

AUTHOR: R. V. Anderson 3-3-71

ABSTRACT: The column buckling device (CBD) fuel element vibration suppressor has been analyzed and re-designed. Pre-kinking of the column shows resultant lowering of the strain hardening effects of a straight column and the axial deformation required to sustain the total column load. Design of the new CBD is presented and further recommendations are included.

TI-759-240-035

TITLE: FUEL ROD FINITE ELEMENT STRESS ANALYSIS: SEGMENTED FLIGHT REACTOR FUEL VERSUS S8DR FUEL

AUTHOR: R. V. Anderson 3-30-71

ABSTRACT: A finite element stress analysis has been performed to define the stresses which exist near the delta/epsilon boundary in the fuel rod. Seven analyses were performed including an S8DR fuel rod at BOL and during 1-Mw operation, and a core test segmented fuel rod with 3.0-, 4.5-, and 6.0-in. segments and worst-case boundary locations. Results indicate that maximum tensile stresses induced in the core test fuel segments are at least 20% lower than those induced in the S8DR rod during BOL (600 kw) operation and at least 35% lower than the S8DR rod during megawatt operation.

7. Fuel Element Fabrication

NAA-SR-7048

TITLE: ASSEMBLY OF S2DR FUEL ELEMENTS

AUTHORS: W. F. Dennison and T. S. Jakubowski 12-15-62

ABSTRACT: The materials procurement, fabrication, and assembly of 51 fuel elements for the S2DR is described.

NAA-SR-7293

TITLE: FABRICATION OF CERAMIC INTERNAL REFLECTOR FOR THE SNAP 8 EXPERIMENTAL REACTOR

AUTHOR: K. Langrod 7-15-62

ABSTRACT: BeO, with and without the addition of Sm₂O₃ as a nuclear poison, was fabricated for use as internal reflectors for the S8ER core. The hot pressing of blanks followed by diamond grinding results in a high yield of close tolerance parts with greater than 98% theoretical density. The low eutectic temperature in the BeO-Sm₂O₃ system (~1420°F) necessitated special pressing parameters to achieve the required densities.

NAA-SR-7305

TITLE: DEVELOPMENT OF A HIGH YIELD 90 ZR-10U ALLOY MASSIVE HYDRIDE

AUTHOR: J. W. Raymond 1-1-63

ABSTRACT: Through addition of approximately 0.4 wt % carbon (as ZrC) to a zircorium - 10 wt % uranium alloy, a 100% yield of uncracked, massively-hydrided fuel rods of high hydrogen density ($N_H = 5.7$ to 6.6) was obtained. The resulting hydride structure was fine grained and equiaxed.

AI-AEC-13066

NAA-SR-8589

TITLE: SNAP 8 EXPERIMENTAL REACTOR FUEL ELEMENT ACCEPTANCE TESTING

AUTHORS: H. J. Hubbell and W. E. Nagel 3-1-64

ABSTRACT: The inspection and test methods applied and the results obtained from the S8ER fuel elements accepted for use in the S8ER core are presented.

AI-AEC-Memo-12702

TITLE: S8DR FUEL ELEMENT QUALITY ASSURANCE RECORD

AUTHORS: C. Donnelly and W. Nagel 3-21-68

ABSTRACT: This report summarizes the major inspection, test, and traceability information for the 221 enriched uranium fuel elements released for use in the SNAP 8 Developmental Reactor.

AI-AEC-Memo-12775

TITLE: S8DR FUEL ROD FABRICATION

AUTHOR: D. T. McClelland 1-29-69

ABSTRACT: The detailed process parameters and product inspection data for fabrication of the S8DR fuel rods are presented.

AI-AEC-13082

TITLE: PROCESS DEVELOPMENT AND FABRICATION OF ZIRCONIUM HYDRIDE HASTELLOY X CLAD FUEL ELEMENTS

AUTHOR: T. G. Parker, Jr. 6-73

ABSTRACT: This report summarized the development work performed, and the processes established, for the fabrication of Hastelloy X clad, U-ZrH_x fuel elements. Processes described include: U-Zr alloy melting and extrusion, U-Zr hydriding, Gd₂O₃ vapor deposition on U-ZrH_x fuel rods, machining helical fins in Hastelloy X cladding, Hastelloy X electron beam welding, ceramic coating of Hastelloy X, and final assembly operations. Process and test data on developmental fuel elements are presented, which demonstrate that the dimensional and performance test requirements of the 5-kwe Zirconium Hydride Reactor could be met.

AI-AEC-13085

TITLE: BURNABLE POISON DEPOSITION ON ZIRCONIUM HYDRIDE FUEL - PROCESS DEVELOPMENT AND EVALUATION

AUTHOR: R. Van Houten 6-73

ABSTRACT: This report reviews the development and evaluation of a high-vacuum process for depositing thin rare earth oxide burnable poison films on fuel rods. The burnable poison films are formed by controlled-rate, electron-beam-heated, high-vacuum evaporation of the oxides from water-cooled crucibles. The report reviews poison placement screening criteria, recommended coating and measuring equipment performance, and swelling and aging test results. All test results for Gd₂O₃, Er₂O₃, and Y₂O₃ coatings are promising. Tests extend to 18,584 hr at 1300°F or higher.

AI-AEC-13066

TR-653-240-004

TITLE: MACHINING CAPABILITY OF HOBBING SNAP CLADDING FINS

AUTHOR: F. C. Schrag

7-7-71

ABSTRACT: Three helical fins were machined on 51 Incoloy 800 cladding tubes by hobbing. This machining technique was to meet SNAP dimensional requirements. The inspection data are presented along with a summary dimensional analysis. A number of actual dimensions did not meet drawing requirements; however, a statistical analysis of the data showed that the hobbing operation did control all critical dimensions within print tolerances. It is concluded that a high yield can be achieved by improving the machine setup technique and performing first-piece inspection before machining a quantity of parts.

8. Fuel Element Testing

NAA-SR-5066

TITLE: THERMAL DIFFUSION OF HYDROGEN IN NONSTOICHIOMETRIC ZIRCONIUM-DEHYDRIDE

AUTHOR: A. W. Sommer and W. F. Dennison

10-1-60

ABSTRACT: Measurements of the Ludwig-Soret effect in delta-phase ZrH_x in the temperature range of 1000 to 1300°F have been made. The "heat of transport," Q^* , of hydrogen in ZrH_x was determined to be 1.3 ± 0.4 kcal/mole. A mathematical analysis of the kinetics of hydrogen redistribution is presented and shown to compare favorably with experimentally observed results. A similarity in the magnitude of Q^* and ΔH , enthalpy of solution, was noted and a rationalization of this similarity is presented.

NAA-SR-5141

TITLE: A HEAT TRANSFER CONTROL DEVICE FOR IRRADIATION EXPERIMENTS

AUTHORS: W. K. McCarty and R. C. Noyes

6-30-60

ABSTRACT: A device which controls temperatures by varying the heat transfer across an annular space has been designed and tested. The central temperature of a nuclear fuel test specimen has been controlled at $1165 \pm 10^\circ F$ by use of this device. This mechanism makes possible temperature control under conditions where the heat source, heat sink, or both vary between 50 and 150% of the nominal value. Out-of-pile tests and three in-pile experiments have confirmed the usefulness of this heat transfer control device.

NAA-SR-6264

TITLE: THEORETICAL KINETICS AND STRESSES FOR DELTA-PHASE ISOTHERMAL HYDRIDING OF URANIUM-ZIRCONIUM

AUTHOR: M. E. Nathan

7-15-62

ABSTRACT: The diffusion equation for isothermal single-phase hydriding of infinitely long cylinders is solved to investigate the theoretical kinetics and stresses of delta-phase by hydriding of a Zr-10U alloy. Results are correlated to indicate the effect of temperature, time, and radial hydrogen concentration gradients.

AI-AEC-13066

NAA-SR-7736

TITLE: PULSE HEATING OF MODIFIED ZIRCONIUM-URANIUM HYDRIDE

AUTHOR: R. E. Taylor

2-15-63

ABSTRACT: Fourteen samples of modified zirconium-uranium hydride ($ZrU_{0.04}H_{1.8 \pm 0.1}$) were pulse heated to destruction. Data on power input, electrical resistivity, temperature, and hydrogen release during the 450-msec pulses are presented.

NAA-SR-7752

TITLE: A METHOD FOR DIFFERENTIATING BETWEEN MATERIALS CONTAINING NORMAL AND FULLY ENRICHED URANIUM

AUTHORS: R. C. Barry and A. R. Mooers

4-15-63

ABSTRACT: A system to allow rapid discrimination of whether a uranium-zirconium alloy contains normal or fully enriched uranium was developed. This autoradiometric method is based on the fact that the U^{238} decay chain includes emission of a 2.31-Mev beta particle which is more easily absorbed by an aluminum filter than gamma photons emitted by U^{235} . The instrumentation required and the procedures employed should be readily adaptable to other uranium-bearing compounds and alloys.

NAA-SR-8090

TITLE: POST-IRRADIATION EVALUATION OF SER FUEL ELEMENTS

AUTHOR: J. R. Miller

5-1-63

ABSTRACT: Selected fuel elements from the SNAP 2 Experimental Reactor were examined to determine their performance capability. The examination indicated that these fuel elements are capable of satisfactory performance at SNAP 2 full power design conditions.

NAA-SR-9600, VOL II

TITLE: NONDESTRUCTIVE DETERMINATION OF URANIUM-235 CONTENT OF SNAP FUEL RODS, VOL II, STATISTICAL METHODS FOR EVALUATING DATA OBTAINED FROM COUNTING EXPERIMENTS

AUTHORS: M. A. Greenfield, C. T. Nelson, and R. R. Koontz

7-15-64

ABSTRACT: This report describes a statistical method for evaluating the homogeneity of U^{235} in reactor fuel rods as the basis of measuring the characteristic radiation emitted by this uranium isotope. Analysis of SNAP fuel rods showed that the counting instrument was capable of determining the U^{235} content for each rod within 0.1% and changes in concentration of 0.3% at a 99% level of confidence.

NAA-SR-9600, VOL III

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AUTHORS: A. R. Mooers and R. C. Barry

5-15-65

ABSTRACT: An autoradiometric test method and associated production inspection equipment have been developed for nondestructively measuring the incremental and total U^{235} content in cylindrical SNAP fuel rods. A prototype single-channel analyzer scintillation counting system and special faced rod scanning mechanism are described. A statistical comparison of the autoradiometric and destructive chemical methods disclosed the correlation coefficient between the two methods to be 0.976. The precision of the autoradiometric method was 0.023 wt % uranium for fully enriched zirconium - 10 wt % uranium hydride.

AI-AEC-13066

NAA-SR-9633

TITLE: EFFECTS OF IRRADIATION ON HYDRIDED ZIRCONIUM-URANIUM ALLOY --
NAA-53 EXPERIMENTS

AUTHORS: J. R. Miller and D. A. Polak

12-30-64

ABSTRACT: The effects of irradiation on the SNAP 8 uranium-zirconium hydride ($N_H = 6.0$) fuel was evaluated in three capsules exposed in the MTR to burnups of from 0.3 to 0.95 metal atom % at average peak temperatures from 1210 to 1470°F. Fuel swelling, fission gas release, microstructure, and hydrogen barrier coating effectiveness were measured and correlated with specimen operating conditions.

NAA-SR-9648

TITLE: EVALUATION OF FUEL ELEMENTS FROM SNAP 8 DEVELOPMENT REACTOR
CORE

AUTHOR: P. S. Olson

1-15-65

ABSTRACT: Eight hydrided 90 Zr-10U fuel/moderator rods with BeO end reflectors, clad with ceramic-coated Hastelloy N which had been used in the S2DR were examined after 11,290 reactor operation hours at a peak temperature of 1280°F. Apart from a small reaction with the beryllium internal reflectors, all elements were in excellent condition after 0.027 metal at. % burnup.

NAA-SR-9673

TITLE: FISSION PRODUCT RELEASE FROM IRRADIATED URANIUM-ZIRCONIUM HYDRIDE
SNAP FUEL

AUTHOR: W. M. Cegelski

7-15-64

ABSTRACT: Samples of uranium-zirconium hydride fuel (4 in. long by 1-1/4 in. diameter) were irradiated in the MTR and subsequently heated to 2000 to 3500°F for periods of a few minutes to 8 hr. The fission product release data obtained show steadily decreasing release rates for periods of several hours. Also, 100% release of volatile fission products only occurred when a clad sample was melted at 3500°F and then dropped to 2000°F for 15 minutes.

NAA-SR-10183

TITLE: SNAP 8 FUEL ELEMENT FABRICATION AND PERFORMANCE TESTING

AUTHOR: G. W. Meyers

7-23-64

ABSTRACT: The process flow diagram for fabrication and nonnuclear performance testing of SNAP 8 fuel elements is shown in diagrammatic and pictorial form.

NAA-SR-11637

TITLE: NON-NUCLEAR PERFORMANCE TESTING -- SNAP 8 EXPERIMENTAL REACTOR
FUEL ELEMENTS -- FINAL REPORT

AUTHORS: A. J. Fitzgerald and F. A. La Scala

3-25-66

ABSTRACT: Non-nuclear testing of S8ER fuel elements at simulated reactor conditions indicate that thermal cycling and ramp heating significantly affect hydrogen permeation rates; but vibration, shock, and accelerations do not. Long-term exposure at operating conditions causes an improvement (decrease) in the permeation rate.

AI-AEC-13066

NAA-SR-12032

TITLE: IN-PILE BEHAVIOR OF SNAP 8 EXPERIMENTAL REACTOR TYPE SUBLENGTH FUEL ELEMENTS (NAA-117 EXPERIMENT)

AUTHORS: R. C. Courtright and R. E. Forrester 11-30-67

ABSTRACT: Sixteen sublength fuel elements of the S8ER design were irradiated in the Hanford K-West Reactor over a range of atom burnups at temperatures of 1300 or 1500°F in approximately 10,000 hr. Their compositions ranged from 1.60 to 1.85 H/Zr with 0.05 to 0.4% carbon. The following observations were made during post-irradiation examination:

- 1) Fuel element integrity was good except where phase-change-dominated swelling was excessive.
- 2) Carbon level significantly affects fuel swelling.
- 3) Both temperature and burnup significantly affect fuel swelling.
- 4) The cladding withstood 1% strain without evidence of cracking.
- 5) No significant defects were noted in the coating, welds, or closures.

NAA-SR-Memo-12097

TITLE: SNAP 8 FUEL ELEMENT STRESS BRIEFING, JULY 19 AND 20, 1966, CANOGA PARK, CALIFORNIA

AUTHOR: Staff 8-5-66

ABSTRACT: This report is a briefing book for the presentation of the status of SNAP 8 fuel element development, the test results from the S8ER and in-pile irradiations, and the fuel element stress analysis.

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AUTHOR: Staff 8-31-66

ABSTRACT: This report is a briefing book presenting the results of the SNAP 8 fuel element stress analysis, SNAP 8 fuel studies from the S8ER, and in-pile fuel irradiations.

NAA-SR-Memo-12210

TITLE: SNAP 8 EXPERIMENTAL REACTOR FUEL ELEMENT BEHAVIOR - ATOMICS INTERNATIONAL TASK FORCE REVIEW

AUTHOR: H. Pearlman (Chairman) 11-30-66

ABSTRACT: The task force scope was to review, evaluate, and make recommendations on Atomics International post-irradiation examination of SNAP 8 Experimental Reactor (S8ER) fuel elements. Emphasis was on density changes, dimensional change, hydrogen compensation, and microstructure. The report includes those evaluations, recommendations, and other findings resulting from this task force review.

AI-AEC-13066

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AUTHOR: Staff 12-14-66

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NAA-SR-Memo-12507

TITLE: PERFORMANCE OF THE S8DR ZrH FUEL ELEMENTS

AUTHORS: J. G. Asquith, G. W. Meyers, and W. J. Roberts 6-20-68

ABSTRACT: A summary of the performance of the S8DR ZrH fuel elements is presented. This report is divided into four topics: fuel swelling correlation, S8DR cladding performance, fuel element in-core performance, and preproduction fuel element results.

NAA-SR-Memo-12588

TITLE: POST-IRRADIATION EXAMINATION OF THE NAA 115-2 IRRADIATION EXPERIMENT

AUTHOR: R. E. Forrester 11-7-67

ABSTRACT: The NAA 115-2 was irradiated at the Hanford Reactor Facility for ~ 10,000 hr at 1300 to 1600°F to achieve a range of metal atomic % burnups. This report documents the detailed post-irradiation examinations that were performed.

NAA-SR-12625

TITLE: IN-PILE BEHAVIOR OF SNAP 8 EXPERIMENTAL REACTOR TYPE SUBLENGTH FUEL ELEMENTS (NAA 115-2 EXPERIMENT)

AUTHORS: R. E. Forrester and W. J. Roberts 6-15-68

ABSTRACT: Twelve sublength fuel elements of the S8DR design were irradiated in the Hanford K-East Reactor to a burnup of 0.3 metal atomic % at temperatures from 1300 to 1600°F in approximately 10,000 hr. Their compositions ranged from 1.60 to 1.85 H/Zr ratio with 0.02 to 0.4 wt % carbon. Four of the elements had longitudinal cladding cracks caused by fuel swelling. Three of these were late-in-life failures. The swelling, temperature, and composition data appeared to correlate with the prior data. A "prior history" dependence of fuel swelling was indicated.

AI-AEC-Memo-12697

TITLE: PERFORMANCE TEST AND EVALUATION OF "SPECIAL SEVEN" PRE-S8DR FUEL ELEMENTS

AUTHORS: W. D. Fuller and K. E. Moore 5-6-68

ABSTRACT: Seven developmental fuel elements were fabricated to evaluate two SNAP 8 process improvements. These fuel elements were subjected to 10,000 hr of out-of-pile performance testing including vibration, thermal cycling, and thermal endurance at 1400°F. The test results indicated a significant improvement over prior fuel elements. These tests were the basis of the S8DR fuel element and core design.

AI-AEC-13066

AI-AEC-Memo-12883

TITLE: POST-IRRADIATION EXAMINATION OF THE NAA 120-4 IRRADIATION EXPERIMENT

AUTHOR: R. E. Forrester 12-11-69

ABSTRACT: The detailed post-irradiation examination data for the NAA 120-4 experiment is presented. Neutron radiographic techniques were developed for nondestructive examination of the irradiated fuel elements. An unsuccessful attempt to use neutron radiography for nondestructive hydrogen analysis was made.

AI-AEC-12963

TITLE: EFFECTS OF IRRADIATION ON HYDRIDED ZIRCONIUM-URANIUM ALLOY -
NAA 120-4 EXPERIMENT

AUTHORS: N. F. Davies and R. E. Forrester 6-30-70

ABSTRACT: In the NAA 120-4 experiment, 17 fuel specimens were irradiated simultaneously in the Hanford K-East Reactor. The specimens with H/Zr ratio ranging from 1.70 to 1.85 experienced temperatures from 1300 to 1600°F and achieved burnups of about 0.9 metal at. %. This experiment confirmed the existence of the postulated "offset" fuel growth which saturates early in life. As expected, fuel which entered β phase exhibited larger swelling than δ phase fuel. Specimens of ϵ phase material also showed higher swelling than δ phase material. The test results were consistent with earlier data.

AI-AEC-13002

TITLE: PROTOTYPE S8DR FUEL ELEMENT PERFORMANCE TEST (NAA 121-1 EXPERIMENT)

AUTHOR: J. G. LaBlanc 7-15-71

ABSTRACT: The NAA 121-1 experiment consisting of four S8DR prototype fuel elements was irradiated in the Hanford K-East Reactor for 12,000 hr at cladding temperatures up to 1300°F and fuel temperatures to 1450°F. These fuel elements achieved peak burnups of about 0.3 metal at. %. Post-irradiation examination indicated that the three normal-hydrogen-content elements behaved satisfactorily. However, these full-length experiments produced families of swelling data that were inconsistent with prior short-length (single-point) specimens. The fuel in the fourth, low-hydrogen-content fuel element was deliberately caused to transform from δ to β zirconium hydride. This fuel element was severely damaged. A mechanism of fuel shrinkage followed by excessive swelling is postulated to explain the behavior of this element.

AI-AEC-13066

AI-AEC-13083

TITLE: HASTELLOY X CLADDING MATERIAL EVALUATION

AUTHORS: K. E. Moore, R. G. Brengle, and T. G. Parker, Jr.

6-73

ABSTRACT: Hastelloy X is the reference fuel element cladding for the 5-kwe space power reactor. The effects of fuel element processing, and exposure to the reactor environment, were evaluated by conducting mechanical property and corrosion tests on three heats of Hastelloy X in various heat-treat conditions. Mechanical properties of processed material are in good agreement with literature values. Some ductility loss was observed in tensile tests. This is attributed to interstitial contamination resulting from the SCB ceramic hydrogen barrier coating process. Hastelloy X samples were exposed to flowing 1400°F NaK in a loop operated with a 90°F ΔT to obtain accelerated corrosion effects. The exposed surfaces of these samples were affected to a depth of ~2 mil after 2000 hr. The affected area had lost chromium, and contained intergranular microvoiding. Biaxial stress-rupture tests, performed in flowing NaK at 1300°F, indicate no significant effect on properties.

Creep tests were performed on unirradiated and irradiated material. Irradiated material showed slightly greater creep strength; ductilities were 1.0% or higher.

Transmission electron microscopy was performed on processed and aged Hastelloy X. Only slight differences in precipitation reactions could be attributed to the processing environment. The predominant phase resulting from thermal aging at 1200 and 1300°F was found to be M_6C .

AI-AEC-13084

TITLE: ZIRCONIUM HYDRIDE FUEL ELEMENT PERFORMANCE CHARACTERISTICS

AUTHOR: A. F. Lillie

6-73

ABSTRACT: Design of the (U-Zr)H-fueled SNAP compact reactor requires prediction of: (1) the reactivity decrease over the core lifetime associated with the loss of hydrogen moderator from the fuel elements, and (2) the anticipated fuel growth for the particular reactor operating conditions to assure integrity of the fuel element cladding. Extensive data were obtained during the course of the SNAP Reactor Development Program regarding the hydrogen-retention characteristics of the ceramic-lined fuel element and the fuel growth characteristics of the (U-Zr)H fuel. The ceramic liner was found to be a very effective and highly durable hydrogen barrier. The fuel growth, observed to consist of a rapid early-life growth followed by a much slower long-term growth, was found to be temperature-, flux-, and fluence-dependent.

TR-653-240-006, Change No. 1

TITLE: EXPERIMENTAL STRESSES FOR A CONTACT LOAD ON SIMULATED SPF REACTOR FUEL ELEMENTS

AUTHORS: R. A. Castle and R. J. Cardunas

3-27-72

ABSTRACT: As part of the SNAP reactor program, a structural test was conducted on simulated fuel elements to determine experimentally the stresses on the inner surface of the cylinder due to local contact loads. These local loads were applied by simulated fuel elements with helical fins 0.072 to 0.160 in. wide on a 5-1/2-in. pitch, acting on 3/4-in.-diameter, thin-walled cylinders, 21 in. long which were internally strain gaged. The results were correlated with analytical efforts utilizing a Bijlaard analysis. Stresses were also determined for the case where a plain tube was loaded against a bowed plain tube.

AI-AEC-13066

TI-759-240-030

TITLE: DIMENSIONAL STABILITY OF HEAVY WALLED INCOLOY 800 CLADDING TUBES - DATA

AUTHOR: J. T. Roberts 2-25-71

ABSTRACT: The effects of SCB coating, thermal shock, and thermal soak on the dimensional stability of four heavy walled Incoloy 800 cladding tubes, ranging in thickness from 20 to 44 mils, was investigated. The results of the study were summarized in TI-759-240-029. This report serves as a repository for the detailed data and profilometer traces used in the preparation of that summary.

9. Reflector Assembly

AI-AEC-13078

TITLE: ZIRCONIUM HYDRIDE REACTOR CONTROL REFLECTOR SYSTEMS SUMMARY REPORT

AUTHORS: P. H. Horton and W. J. Kurzeka 6-73

ABSTRACT: The beryllium reflector control system development for SNAP reactors is documented, from the initial SNAP 10A System through the current 5-kwe Thermoelectric System. Described are the various reflector concepts used in these systems for shadow-shielded and 4π -shielded nuclear systems. The development of the key components, such as the actuators, bearings, and drive mechanisms for these systems, is also traced from the SNAP 10A concept through to the current system. Developmental test results are outlined, showing the performance capability improvements made throughout the life of the SNAP programs. Component development was highly successful, as proven by a number of reactor systems tests, including the launch and operations of the SNAP 10A flight.

TI-652-220-006

TITLE: DESIGN SUMMARY REFLECTOR DRIVE SYSTEM

AUTHORS: O. P. Steele III and E. J. Donovan 10-10-72

ABSTRACT: The design requirements, design description, and expected performance of the reflector and drive control system are summarized.

TI-652-220-007

TITLE: 5-kwe REACTOR TE REFLECTOR SECTOR DRIVE ACTUATOR DESIGN SUMMARY

AUTHOR: L. E. Donelan 10-4-72

ABSTRACT: The results of the 5-kwe Reactor TE System actuator parametric study, and subsequent load and performance calculations, are presented, together with the results of the stress analysis.

TI-653-220-005

TITLE: SPF REFLECTOR AND CONTROL DRUM DRIVE SYSTEM CALCULATIONS

AUTHOR: R. T. Thexton

2-10-72

ABSTRACT: This report is a summary of all calculations performed in development of the Reflector and Control Drum Drive System concept for the SPF Reactor.

TI-653-220-010

TITLE: SPF - REFLECTOR AND CONTROL DRUM DRIVE - SUMMARY STATUS

AUTHOR: R. T. Thexton

4-3-72

ABSTRACT: This report is a summary of drawings and documents with their status for the SPF Reflector and Control Drum Drive System.

TI-653-260-001

TITLE: COMPARISON OF CALCULATED AND EXPERIMENTAL REACTIVITIES OF SNAP REACTORS

AUTHOR: D. S. Bost

8-4-71

ABSTRACT: The reactivities of S8ER, the S8ER core in SCA-4B, S8DR, and SCA-4A were calculated using current methods and cross-section data for comparison with experiment. A variety of core loadings and reflector shim configurations were used, making a total of 12 calculations. They may be grouped into six near-critical and six significantly supercritical configurations. The mean error and standard deviation was $-0.11 \pm \$1.53$ for all calculations, $+0.51 \pm \$1.05$ for the near criticals, and $-0.73 \pm \$1.76$ for the supercritical.

TR-696-26-003

TITLE: TEST REPORT - DEVELOPMENT THERMAL CYCLE, SHOCK, ACCELERATION, AND VIBRATION TESTING OF A BeO REFLECTOR SPECIMEN
PART NO. EX-N69626085

AUTHOR: F. H. Welch

4-24-69

ABSTRACT: This report presents a summary of test results and recommendations. A BeO block representing a section of the 4-Pi reflector was thermal cycled to 1300°F, and then subjected to a series of shock, vibration, and acceleration tests which simulated the prelaunch thermal cycles and launch shock and vibration condition. Examination of the block revealed that no damage occurred after vibration tests up to 85 g. The tests demonstrated that the BeO block and its mountings are adequate for the design loads.

AI-AEC-Memo-12725

AUTHOR P. H. Horton

6-27-68

ABSTRACT This report covers the thermal cycle tests, between ambient room temperature and 1500°F, conducted on an advanced ZrH reactor control drum bearing, consisting of a solid graphite monoball within an alumina-coated Ta-10W socket. The measured frictional torque and the corresponding calculated friction coefficient are presented graphically for the 27 thermal cycles conducted over the 2300-hr test while the bearing was loaded with 25-lb radial and 75-lb thrust loads in 10^{-8} torr pressure environment. Pre- and post-test photographs of the various bearing components are also included. The bearing components are reported to be in excellent condition after the test with graphite film being formed on the alumina friction surfaces of the socket and thrust washers. The calculated friction coefficient is reported as 0.70 at room temperature, 0.5 at 800°F, and 0.3 at 1500°F.

AI-AEC-13078

TITLE: ZIRCONIUM HYDRIDE REACTOR CONTROL REFLECTOR SYSTEMS SUMMARY REPORT

AUTHORS: P. H. Horton and W. J. Kurzeka

6-73

ABSTRACT: The beryllium reflector control system development for SNAP reactors is documented, from the initial SNAP 10A System through the current 5-kwe Thermoelectric System. Described are the various reflector concepts used in these systems for shadow-shielded and 4π -shielded nuclear systems. The development of the key components, such as the actuators, bearings, and drive mechanisms for these systems, is also traced from the SNAP 10A concept through to the current system. Developmental test results are outlined, showing the performance capability improvements made throughout the life of the SNAP programs. Component development was highly successful, as proven by a number of reactor systems tests, including the launch and operations of the SNAP 10A flight.

AI-AEC-13080

TITLE: ZIRCONIUM HYDRIDE REACTOR CONTROL DRIVE ACTUATOR DEVELOPMENT
SUMMARY REPORT

AUTHOR: L. E. Donelan

6-73

ABSTRACT. Over the years, actuators for compact space nuclear reactor reflector control have progressed from modified commercial units with relatively short life and low reliability at moderate temperature to units custom-designed for specific application. These latter units have demonstrated long life and high reliability at high temperatures. The design for a specific application now includes the ability to design for various torque and step sizes and the ability to control the relative proportions (length and diameter) of the actuator. The present status of design calculation capability incorporates reliable computer programs to facilitate the design and selection of the actuator, in the form of: (1) a stepper motor program for calculating actuator performance based on physical configuration, (2) an actuator brake program for calculating the magnetic performance based on a physical brake configuration, and (3) a bearing program for calculating loads and clearances for initial end bell and bearing interferences and journal dimensions. Using these programs and various other supporting calculations, a control drive actuator has been designed for the 5-kwe Thermoelectric System reactor to fit in the available space within the reflector and shield configuration. This actuator will develop sufficient torque for both the specified stepping drive control and the scram requirement, at current densities no greater than the proven S8DR design.

The report contains a brief history of the progress of actuator design, development, and tests to date, as well as the design calculations applicable to the actuator designed for the 5-kwe Thermoelectric System.

AI-AEC-13066

AI-AEC-13079

TITLE: ZIRCONIUM HYDRIDE REACTOR CONTROL SYSTEM BEARING DEVELOPMENT
SUMMARY REPORT

AUTHOR: P. H. Horton

6-73

ABSTRACT: The development of the bearings for use on control reflector systems of various SNAP reactors is described. These bearings must operate unattended for long duration under the severe environment of high temperature, space vacuum, and nuclear irradiation from the reactor. Initial self-weld and friction studies lead to the selection of flame-sprayed Al_2O_3 coating and carbon graphite as the candidate bearing couple. Frictional characteristics and the material compatibility of this combination (various substrate materials for the Al_2O_3 coating) were demonstrated by tests at temperatures up to 1600°F and space-simulated vacuum. Self-aligning monoball bearings for control reflector use, and bushing-type bearings for other reactor components, have been successfully operated as long as 20,000 hr at temperatures up to 1500°F.

TI-653-220-010

TITLE: SPF - REFLECTOR AND CONTROL DRUM DRIVE - SUMMARY STATUS

AUTHOR: R. T. Thexton

4-3-72

ABSTRACT: This report is a summary of drawings and documents with their status for the SPF Reflector and Control Drum Drive System.

TI-696-22-003

TITLE: CONTROL DRUM DRIVE TRAIN AND ACTUATOR DESIGN

AUTHORS: K. Bilibin and D. Harwood

4-10-70

ABSTRACT: The design analysis of the control drum drive train, including the mockup actuator assemblies, is documented.

TI-696-220-012

TITLE: DEVELOPMENT OF AN IMPROVED ACTUATOR FOR ZrH REACTOR SYSTEMS;
A HISTORICAL REFERENCE

AUTHOR: L. E. Donelan

7-7-71

ABSTRACT: The history of the actuator for the ZrH Reactor system is presented. The present design status is summarized.

TI-696-26-046

TITLE: EXAMINATION OF ZrH CONTROL DRUM MATERIAL COMPATIBILITY SAMPLES

AUTHOR: C. C. Woolsey

2-18-70

ABSTRACT: Examination of a compatibility stack revealed no problems between BeO and Ta-10W, BeO and TZM, BeO and Hastelloy N, BeO and Cb-752, Ta-10W and Cb-752, TZM and Cb-752. Severe reactions were found at interfaces between Hastelloy N and Ta-10W. Exposure was for 4000 hr at 1800°F with 160 hr at 2000°F.

AI-AEC-13066

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D. NUCLEAR RADIATION SHIELD DESIGN, ANALYSIS, AND TESTING

NAA-SR-5897

TITLE: SNAP SHIELD TEST EXPERIMENT PRELIMINARY OPERATIONS MANUAL

AUTHOR: R. P. Johnson, et al. 5-8-61

ABSTRACT: This is a preliminary operations manual for the SNAP Shield Test Experiment (STE). A brief description of the facility is presented. Procedures for precritical operation, reactor loading and testing, and shield test equipment operation are presented. In addition, reactor servicing, radiological safety, and industrial hygiene are discussed. Finally an STE operator training program is presented.

NAA-SR-6207

TITLE: SNAP 8 REACTOR PRELIMINARY DESIGN SUMMARY

AUTHOR: H. Rosenberg (Editor) 6-15-61

ABSTRACT: The purpose of this report is to describe the present concept of the SNAP 8 reactor, reactor control system, safety systems, and shield and to outline the reasoning behind the design point selection.

NAA-SR-7432

TITLE: SNAP SHIELD TEST EXPERIMENT FISSION-PLATE POWER CALIBRATION

AUTHOR: R. L. Tomlinson et al. 8-15-62

ABSTRACT: Three techniques were employed in measuring the fission rate of the STE fission-source plate: thermal neutron absorption rate, calorimetric method and fission-neutron production technique. All three methods are discussed. The calorimetric calibration technique proved to be the most satisfactory.

NAA-SR-9364

TITLE: EXPERIMENTAL STUDIES OF NEUTRON ATTENUATION IN NATURAL LITHIUM HYDRIDE SHIELDS

AUTHORS: S. G. Wogulis and K. L. Rooney 2-12-64

ABSTRACT: Results of a program to measure the attenuation of fission neutrons in lithium hydride at the SNAP Shield Test Facility are presented, with the methods used. Experimental and analytical corrections to the data are given to allow reduction to the infinite medium case.

NAA-SR-9400, VOL I

TITLE: LITHIUM HYDRIDE TECHNOLOGY, VOL I, PROPERTIES OF LITHIUM HYDRIDE AND CORROSION STUDIES

AUTHORS: F. H. Welch, E. C. Philips, J. G. Asquith, and N. F. Davies 5-8-64

ABSTRACT: This first volume presents experimentally determined properties and behaviors of various forms of LiH, which are of interest for LiH neutron shield. The physio-chemical properties reported include thermal expansion, thermal conductivity, dissociation rate in vacuum, dissociation pressures, outgassing, sinterability, and reactions with Gd and Gd₂O₃, liquid oxygen, and silicon. No deleterious effects were reported on the 304, 316, 347, and 19-9D1 containers for cast LiH after 4,000 hr exposure at 900°F, 1200°F, and 1400°F. From the somewhat contradictory results of limited irradiation experiments, the authors conclude that LiH can be used for neutron shields without any difficulty up to total dose of approximately 10¹⁸ nvt thermal neutrons. Further closely controlled in-pile experiments are recommended to verify LiH use in shields at higher doses. Any irradiation effects are concluded to be significantly reduced by the use of high-purity Li⁷H.

NAA-SR-9400, VOL II

TITLE: LITHIUM HYDRIDE TECHNOLOGY, VOL II, FABRICATION OF SHADOW SHIELDS FOR SNAP APPLICATIONS

AUTHOR: F. H. Welch 9-30-65

ABSTRACT: The methods of fabricating the stainless steel shield container and the internal honeycomb matrix, and the equipment and technique of filling and casting LiH within the container are described. The radiographic and void volume inspection methods to verify the integrity of the cast LiH are presented. The configurations and special features, such as two separate LiH regions, internal structures and steel wool matrix, inspection data, and test results on 9 cast LiH shadow shields are reported.

NAA-SR-9400, VOL III

TITLE: LITHIUM HYDRIDE TECHNOLOGY, VOL III, PROPERTIES OF LITHIUM HYDRIDE FOR SNAP SHIELDING APPLICATIONS

AUTHOR: F. H. Welch 5-1-67

ABSTRACT: This third volume describes the additional experiments conducted to evaluate the use of LiH as a neutron shield for SNAP reactor systems. Conclusions reached include: the migration of hydrogen from the hot to colder section of LiH is minimal; LiH casting process does not affect the hydrogen permeability of the stainless steel container material; hydrogen loss through meteroid puncture is less than calculated; metallic lithium does not hydride to any degree at 800°F or less when contained with LiH; a hydrostatic weighing method can be used to distinguish Li⁷H from LiH; compressive creep of cast LiH at about 1000°F is less than that of cold-pressed LiH; modulus of rupture of cast LiH is equivalent to that of cold-pressed and sintered LiH; samples of cast LiH can be easily decanned if cast in titanium tubes; backfilling cast LiH shield assembly with metallic lithium may be used to improve the structural capability; no evidence of corrosion of the 304 stainless steel container after 4000-hr exposure at 600, 900, and 1200°F to Li/LiH mixture.

NAA-SR-9400, VOL IV

TITLE LITHIUM HYDRIDE TECHNOLOGY, VOL IV, A NOVEL NEUTRON-GAMMA MATERIAL FOR SNAP SHIELDING APPLICATIONS

AUTHOR F. H. Welch

5-25-67

ABSTRACT The development of a novel SNAP reactor shield in which the gamma shielding material is selectively dispersed in the cast LiH is described. Reported laboratory-scale efforts include casting of sampled rods, stability testing of composites for 4000 hr at 800 and 1000°F to establish the material compatibility between tungsten and ZrH and LiH, and measuring of the modulus of rupture. The design, fabrication, and disassembly examination of a SNAP 10A size cast LiH shield assembly containing approximately 200 lb of tungsten granules near one end as gamma shielding to demonstrate this shield concept are presented.

NAA SR-9400, VOL V

TITLE LITHIUM HYDRIDE TECHNOLOGY, VOL V, TESTING AND EXAMINATION OF SNAP SHADOW SHIELDS

AUTHOR F. H. Welch

8-31-67

ABSTRACT The results of nondestructive tests on 12 SNAP reactor-type cast LiH shadow shields and final disassembly examination on 7 of these are described. On the basis of the results, the following conclusions are reported: the inspection of the cast LiH for cracks and voids can be made by careful radiographic examination; shields with two separate LiH zones can be cast in a single run; lightweight internal structure supported by the LiH provides light methods of supporting the reactor mass; stainless steel wool can be used as LiH matrix material to control cracking where it is impossible or difficult to install the preferred honeycomb matrix; tungsten or zirconium hydride granules can be cast in a selected layer within the LiH for gamma shielding; backfilling cast LiH shield assembly with metallic lithium strengthens the LiH structure; ties the cast LiH to the shield container to reduce launch loading; and increases the bulk thermal conductivity; curved tubular penetration for NaK pipes and electrical cables can be built into the shield assembly.

NAA-SR-9418

TITLE ANALYSIS OF NEUTRON SCATTERING FROM SMALL VOLUMES OF CARBON, BERYLLIUM, ALUMINUM, IRON, AND POLYETHYLENE

AUTHOR R. S. Hubner

3-1-64

ABSTRACT The results of the analyses of the ORNL Tower Shield experiments on neutron scattering from small volumes are presented and compared to the experimental data. Satisfactory agreement in the scattered dose is noted with both experiments and ORNL analytic methods. The Monte Carlo techniques used in the analysis are described in the appendix.

NAA-SR-10853

TITLE ANALYSIS OF NEUTRON TRANSMISSION THROUGH SLABS OF LITHIUM HYDRIDE

AUTHOR R. S. Hubner

7-25-65

ABSTRACT The results of the analyses of the ORNL Tower Shielding experiments on fast neutron transmission through lithium hydride slabs are presented and compared to the experimental data. Satisfactory agreement in the scattered dose is noted in all cases with the maximum deviation at the 30-in. thickness being only 25%. Modifications to the Monte Carlo techniques used in the analyses are presented in the Appendix.

AI-AEC-13066

NAA-SR-11175

TITLE: STARTUP AND OPERATION OF THE ONE-MEGAWATT SHIELD TEST AND IRRADIATION REACTOR

AUTHORS: K. G. Randen and K. G. Golliher 3-25-66

ABSTRACT: This report presents reactor physics data on the Shield Test and Irradiation Reactor (STIR). MTR-type fuel elements were replaced in the Shield Test Experiment (STE) facility and the reactor power was increased from 50 kw to 1 Mwt. The STIR is to be used for shielding experiments. In addition, it will be used in support of a number of other programs in such areas as radiation damage, reactor physics, fuel material research, radiochemistry, and instrumentation development.

NAA-SR-11528

TITLE: CAPABILITIES OF THE 1-Mw SHIELD TEST AND IRRADIATION REACTOR

AUTHORS: K. G. Golliher and K. G. Randen 11-10-65

ABSTRACT: This report describes the physical layout of the Shield Test and Irradiation Reactor (STIR) facility. This facility is designed to provide a flexible source of neutrons and gamma rays for experimental work on shield development, for irradiation damage studies on electronic systems, and for the nuclear qualification of SNAP components. Neutron fluxes and gamma ray dose rates at various locations are presented.

NAA-SR-Memo-11838

TITLE: DESIGN, FABRICATION, TESTING, AND FINAL EXAMINATION OF SNAP EXPERIMENTAL SHIELD NUMBER 1

AUTHOR: F. H. Welch 2-22-66

ABSTRACT: The design features, the fabrication (including the loading and casting of the LiH), and the subsequent tests and examination of the experimental shield assembly are described. The following conclusions are reported: A light-weight tubular internal structure supported by the compressive strength of the LiH can be utilized to transmit the reactor loads to the shield attachment points. A stainless steel liner on the inside of the shield vessel provides better separation of the LiH from the vessel wall and also reduces the flaking of the LiH at surfaces during the vibration environment. Stainless steel wool can be substituted as the matrix material where it is impossible or impractical to use the honeycomb matrix. The shield vessel can be designed essentially as a safety container for the integral LiH shielding-tubular structural assembly.

NAA-SR-Memo-12168

TITLE: SNAP REACTOR COMPONENT PERFORMANCE CAPABILITIES - HIGH-TEMPERATURE SHIELD MATERIALS

AUTHOR: F. H. Welch 9-22-66

ABSTRACT: The studies of material properties and behaviors of candidate neutron and gamma shielding materials for high-temperature applications are described. LiH cast within stainless steel container is reported as the best neutron shield for temperatures up to 1150°F and total irradiation dose of 10^{19} nvt, provided that the temperature is above 700°F. Tungsten alloy (90W, 7.5 Ni, 2.5 Cu) is preferred over lead, depleted uranium, or uranium alloys. The low melting point of lead, thermal cycle and irradiation instability of depleted uranium, and need for canning the more stable uranium alloys are factors favoring tungsten.

AI-AEC-13066

NAA-SR-Memo-12183

TITLE: THE DESIGN, FABRICATION, REMELT, AND EXAMINATION OF SNAP
EXPERIMENTAL SHIELD NUMBER 2

AUTHOR: F. H. Welch

10-4-66

ABSTRACT: The design features, fabrication methods including the LiH loading and casting, and the disassembly examination of the second experimental shield assembly are reported. The shield design feasibilities established by this shield include: (1) internal layer of granular tungsten alloy materials for gamma shielding, (2) curved penetration axially through the shield for NaK pipes and electrical cables, (3) lightweight tubular internal structures supported by the LiH to transfer the reactor loads to the shield mounting points, and (4) stainless steel vessel liner for cleaner separation between the LiH and the vessel wall.

NAA-SR-Memo-12189

TITLE: BULK THERMAL CONDUCTIVITY OF SNAP 8DS SHIELD

AUTHOR: Phillip H. Horton

10-5-66

ABSTRACT: The test methods and the calculated bulk thermal conductivities of the cast LiH S8DS shield assembly are reported. Graphs of thermocouple data vs time and the TAP code-computed bulk thermal conductivity vs shield temperature are included for both increasing and decreasing temperature conditions. Thermal conductivity values obtained were 2.3 to 3.8 Btu/hr-ft-°F in the temperature range of 600 to 1150°F.

AI-AEC-13081

TITLE: RADIATION SHIELDING FOR ZIRCONIUM HYDRIDE REACTORS SYSTEMS

AUTHORS: V. Keshishian, et al.

6-73

ABSTRACT: This report is a summary of the radiation shielding for zirconium hydride (ZrH) reactors. Shield nuclear calculational methods are presented. The shield variations for unmanned and manned space systems, in addition to nonspace systems, are presented. Both neutron and gamma-ray shield fabrication methods for ZrH reactor shields are discussed in detail. Finally, the nuclear and mechanical aspects of the 5-kwe Reactor Thermoelectric System shield are presented in detail.

AI-7950

TITLE: APPLICATION OF SNAP 2 TO MANNED ORBITAL SPACE STATIONS

AUTHOR: H. N. Rosenberg (Editor)

12-16-62

ABSTRACT: Installation and operation of an 11-kwe version of a SNAP 2 system in a manned space station is studied. Of prime importance is the radiation shielding required for the manned system and the use of design features which minimize shield weight. Two space station configurations are considered. Parametric curves of shield weight are presented to enable the space vehicle designer to estimate system weights for other configurations.

AI-Memo-8838

TITLE: PROTON SOURCES FOR APOLLO SHIELDING CALCULATIONS

AUTHOR: V. E. Kistler

8-2-63

ABSTRACT: Basic proton sources from 3 imaginary model flares, 10 actual flares, and 1 Van Allen proton belt are presented for use in radiation shielding analyses. High energy angular distributions are also presented for the Van Allen belt and a typical solar flare.

AI-AEC-13066

TI-652-230-004

TITLE SHIELD NUCLEAR DESIGN FOR THE 5-kwe TE SYSTEM

AUTHOR V. Keshishian 9-20-72

ABSTRACT The nuclear analysis of the 5-kwe reactor shield is presented. Calculation methods and optimization techniques used are presented. Borated stainless steel was selected for the gamma ray shield with tungsten alloy as an alternate. The total shield weight was calculated to be 355 lb.

TI-652-230-005

TITLE LiH SHIELD ANALYSIS

AUTHOR L. D. Felten 10-5-72

ABSTRACT Thermal analyses of the LiH neutron shield are presented. Acceptable temperature profiles are obtained when the shield containment shell is coated with reflective insulation and conduction from the gamma shield is eliminated.

TI-696-23-004

TITLE Na-24 ACTIVITY IN THE REFERENCE ZrH REACTOR SHIELD GALLERY

AUTHOR V. Keshishian 8-27-68

ABSTRACT Preliminary saturated Na-24 activity in the Reference ZrH Reactor primary coolant system was calculated to be 594.4 curies. Of this amount, 260 curies are in the gallery region. The unshielded dose rate from Na-24 in the gallery was calculated to be 158 mrem/hr at a separation distance of 150 ft. The above is for 600 kw.

TI-696-23-023

TITLE REFERENCE ZrH REACTOR DOCUMENTATION OF SHIELD CONFIGURATION AND NUCLEAR HEATING

AUTHOR C. A. Goetz 4-11-69

ABSTRACT Preliminary design values are provided for the following Reference ZrH Reactor subsystem parameters: (1) shield dimensions and materials, (2) shield nuclear heating, and (3) control drum nuclear heating.

TI-696-23-025

TITLE REFERENCE ZrH REACTOR SHIELD DESIGN CALCULATIONS

AUTHOR C. A. Goetz 4-18-69

ABSTRACT Details of the calculations in support of the reference shield configuration for the 295 ZrH Reactor are reported.

TI-696-23-040

TITLE NUCLEAR HEATING IN REFERENCE ZrH REACTOR SHIELD

AUTHORS R. J. Thomson and G. V. Schwab

12-17-69

ABSTRACT Internal heat generation in the conceptual design shield was found to be 7.82 kw. This is about half the preliminary estimate made during the shield materials trade study. Spatial distribution of this heating is presented so that heat transfer calculations may be performed and shield operating temperatures determined.

TI-696-23-041

TITLE RADIAL NEUTRON FAST FLUX IN ZrH REFERENCE REACTOR

AUTHOR G. V. Schwab

1-2-70

ABSTRACT Fluence, or integrated flux, for high-energy neutrons is presented radially throughout reactor regions and shields for 20,000-hr operating time.

TI-696-23-043

TITLE TANTALUM GAMMA SHIELD

AUTHORS G. V. Schwab and R. J. Thomson

1-13-70

ABSTRACT The reference gamma shield material for the Reference ZrH Reactor is tungsten. Since this material is brittle, tantalum-10W has been suggested as an alternate. This study found that in the reference system, the two materials are essentially identical from a nuclear and weight standpoint.

TI-700 13-010

TITLE PREDICTION OF ABLATION DEPTHS OF LITHIUM HYDRIDE SHIELD DURING REENTRY, INCLUDING A STUDY OF THE EFFECTS OF NODE SIZE ON CALCULATED ABLATION RATES

AUTHOR R. D. Elliott

11-11-69

ABSTRACT Calculations were made using a one-dimensional thermal model of a section through the lithium hydride shield surrounding the ZrH Reactor. It was found that, for a representative reentry trajectory and for an assumed aerodynamic heating factor of 0.535, about 4.8 in. of hydride would ablate before the end of the heating period. The validity of the heating factor used has not yet been established, so these results should be considered as preliminary. The sensitivity of the calculated ablation rates to node thickness of interest in the thermal model was established.

AI-AEC-13066

BLANK

E. THERMOELECTRIC CONVERTER

1. Thermoelectric Converter Design, Analysis, and Testing

NAA-SR-7355

TITLE THERMOELECTRIC CONVERSION FOR ADVANCED SNAP SYSTEMS

AUTHOR C. J. Morse 12-15-62

ABSTRACT The use of thermoelectric conversion in advanced nuclear space power systems is investigated. A converter reliability analysis is presented, and a discussion of applications for thermoelectric systems is given.

NAA-SR-11955

TITLE SNAP 10A POWER CONVERSION SYSTEM DESIGN AND TEST

AUTHORS S. R. Rocklin, et al. 6-30-66

ABSTRACT The design features, fabrication processes, and the testing of the SNAP 10A PCS System with the direct-radiating SiGe thermoelectric elements are discussed. Performance data and an evaluation of the various laboratory tests of the PCS modules and of the SNAP 10A system operation, including the 50-day operation of FSM-4, 10,000-hr nuclear ground test of FS-3, and the 43-day space operation of FS-4, are presented. It is concluded that the PCS system of the SNAP 10A should operate successfully for a minimum of 2 yr with power output of 3.25 watts/lb with a degradation rate of less than 6%/yr.

AI-AEC-Memo-12762

TITLE CORRELATION OF DATA FROM NaK HEATED AND ELECTRICALLY HEATED THERMOELECTRIC MODULE TESTS

AUTHOR J. L. Johnson 11-18-68

ABSTRACT The performance data of 12 direct radiating thermoelectric modules tested with electrically heated static NaK in the hot-leg tube are compared with that of 10 modules tested under similar conditions but with flowing heated NaK in the hot-leg tube. Data are presented to show that the module performance variances were less than 1%. The slightly lower degradation rate of the modules tested with flowing NaK is attributed to approximately 50% less axial temperature gradient than with static NaK and is more representative of the actual operational conditions. It is concluded that the electrically heated static NaK test method is an economical method to generate conservative performance data.

AI-AEC-13099

TITLE NaK LOOP TESTING OF THERMOELECTRIC CONVERTER MODULES

AUTHOR J. L. Johnson 5-73

ABSTRACT The history of testing of Westinghouse compact tubular modules in flowing NaK loops at Atomic International is summarized. Test procedures, data handling, and instrument calibration are discussed. Also included are descriptive information of the test facilities, operational problems encountered, and recommendations for testing

TI-696-13-050

TITLE REACTOR PERFORMANCE CHANGES NECESSARY TO COUNTERACT THERMOELECTRIC CONVERTER DEGRADATION

AUTHOR G. E. German 6-19-70

ABSTRACT The effects that degrading thermo-electric (TE) converters have upon reactor performance are calculated. It is shown that beginning-of-life (BOL) reactor coolant outlet temperature and TE converter degradation rate are key design factors that must be properly set to avoid reactor operation at outputs and temperatures where accelerated reactor degradations occur. It is also shown that the rise time to reach a specific reactor coolant outlet temperature is independent of the TE converter conversion efficiency. It is concluded that 5-yr power systems using state-of-the-art TE converters are feasible if the proper BOL reactor coolant outlet temperature is selected. A data display suitable for adjusting BOL reactor temperature and TE converter degradation rate objective for 5-yr power systems is provided.

AI-AEC-13066

2. Advanced Thermoelectric Converter Design, Analysis, and Testing

NAA-SR-Memo-11885

TITLE SUMMARY OF ADVANCED SNAP 10A HIGH TEMPERATURE THERMOELECTRIC MODULE TESTING

AUTHORS A. B. Sorkin and S. J. Block 3-25-66

ABSTRACT Test data and performance evaluation of eight advanced high-temperature, direct-radiating (SNAP 10A type) thermoelectric (TE) modules are described. These modules, based on the SNAP 10A TE technology, were designed for 1300°F hot-junction operation for increased power output for given size and weight. Tests include qualification (initial 100-hr steady-state, 10 thermal cycles, 100-hr steady-state, shock and vibration, and endurance at steady-state with 3 thermal cycles/1000 hr), shock and vibration evaluation, and multiple thermal cycles and over-temperature operation with 1400°F hot junction temperature. One qualification unit, continuing on test with over 14,000 hr of operation, was still producing 0.90 watt compared to the average initial output of 1.2 to 1.4 watts. Test data and post-test examination results are presented.

NAA-SR-Memo-12151

TITLE INTERIM TEST REPORT - DEVELOPMENT, ADVANCED TE 8-COUPLE MODULE

AUTHOR M. Mike 8-22-66

ABSTRACT The results of tests being conducted on 20 advanced thermoelectric 8-couple modules (DTA Series) are reported. Tests include shock and vibration to the acceptance level on all units and to higher levels on selected units, thermal cycling at 20°F/min rate, and steady-state endurance at 1275 or 1475°F hot-junction temperature. The only reported failure occurred during the vibration testing at twice the acceptance level. Performance data are presented graphically.

AI-AEC-Memo-12672

TITLE THERMOELECTRIC SYSTEM AEC REVIEW, MARCH 29, 1968, WASHINGTON, D. C.

AUTHOR Staff 3-27-68

ABSTRACT This report is a briefing book for the presentation of the thermoelectric system and advanced subsystem engineering for manned and unmanned space applications.

TI-696-78-001

TITLE POWER TRANSMISSION OPTIMIZATION

AUTHOR J. A. Sapia 5-20-70

ABSTRACT The optimum converter interconnect pattern and transmission scheme to meet the conflicting requirements of system weight and system reliability have been determined for the 33-kwe (nominal) space station power system having the converter located either on the reactor side (2 loop) or the station side (3 loop) of the lower shield. A general equation involving the influencing factors is developed, and parametric curves are presented and discussed.

AI-AEC-13066

F. HEAT TRANSFER AND HYDRAULICS

1. Heat Transfer and Hydraulic Design and Analysis

NAA-SR-6439

TITLE SNAP 2 PRIMARY COOLANT DEVELOPMENT

AUTHOR M A. Perlow

7-15-61

ABSTRACT Efforts in FY 1961 are described and summarized in the design development, testing, and selection of components integral to the primary coolant system for the SNAP 2 (Systems for Nuclear Auxiliary Power) reactor for space applications. Included are core hydraulic studies, heater development, corrosion studies, NaK hydride precipitation studies, and lithium hydride studies.

NAA-SR-7938

TITLE A REACTOR TRANSIENT HEAT TRANSFER MODEL

AUTHORS D. G. Mason and R. W. Winson

3 1 64

ABSTRACT A transient heat transfer model of a SNAP 8 reactor core is described which can be programmed on a general purpose electronic analog computer. The transient responses of two models, both formulated from the physical system using the laws of conservation of mass and energy, are compared for step changes in power and inlet coolant temperature. One of the models is selected as having the desired transient accuracy in response to the input disturbances.

NAA-SR-8171

TITLE CORRELATION OF LIQUID FRACTION IN TWO-PHASE FLOW WITH APPLICATION TO LIQUID METALS

AUTHOR C. J. Baroczy

4-15-63

ABSTRACT This report presents a generalized correlation for liquid fraction in two-phase flow which is proposed for use with all fluids including liquid metals. The correlation is based on isothermal two-phase, two-component liquid fraction data for liquid mercury-nitrogen and water-air. Prediction of liquid fraction by this method is shown for sodium, potassium, rubidium, and mercury. Application of the method to boiling mercury, for a range of temperatures and exit qualities, is demonstrated for SNAP systems.

NAA-SR-8468

TITLE STATISTICAL TREATMENT OF HOT CHANNEL FACTORS FOR COMPACT REACTORS

AUTHORS P. D. Cohn and H. A. Evans

7-15-63

ABSTRACT This report presents the theoretical development, statistical treatment, and application of hot channel factors in compact SNAP reactors as represented by the SNAP 8 experimental core loading. Presented are the channel and rod power effects of variation in the concentrations of uranium, hydrogen, and poison in the fuel rods. The statistical distributions of all variables are examined. A random selection of rods and channels is compared to the actual SNAP 8 configuration, with no difference noted. Confidence limits are set on hot rod and hot channel factors. Various other factors and aspects of the problem are discussed.

AI-AEC-13066

AI-9068

TITLE: SPACE HEAT REJECTION

AUTHOR: M. G. Coombs and R. A. Stone

10-63

ABSTRACT: The two available means of space heat rejection are radiant heat transfer and expulsion of a heated substance. Only radiation is practical for rejection of large quantities of heat and only this is considered. The different aspects of radiator design are studied and applied to illustrate design procedures leading to minimum-weight heat rejection systems.

NAA-SR-9087

TITLE: SNAP 2 PERFORMANCE ANALYSIS

AUTHOR: R. T. Lancet et al.

2-15-64

ABSTRACT: The most significant steady-state thermodynamic and fluid dynamic analyses of the SNAP 2, 3- to 5-kw nuclear power supply is presented. In addition to the general system and design point analysis, the detailed analysis of the thermal and fluid dynamic performance of the radiator-condenser and the thermal performance of the instrument compartment are presented.

NAA-SR-9425

TITLE: A TRANSIENT MULTICHANNEL TWO-PHASE HYDRAULIC CODE

AUTHORS: L. R. Steele and R. F. Berland

12-1-64

ABSTRACT: A description of a computer program that calculates both steady-state and transient distribution of flow in a reactor core is presented in this report. Results are compared to experimental data for the transient characteristics and to another program for the steady-state distribution of flow. The program may handle a single channel or multiple channels with crossflow under either natural or forced circulation.

NAA-SR-9885

TITLE: DEVELOPMENT, FABRICATION AND TESTING OF HONEYCOMB RADIATOR CONDENSER

AUTHORS: C. Scott and W. Engle

3-1-65

ABSTRACT: The extensive development efforts conducted to establish the feasibility of a unique honeycomb radiator condenser concept for the SNAP 2 system are reported. The basic design consists of a conical honeycomb shell structure with condensing tubes placed within the sandwich and copper radiator attached to the outer surface. An evaluation is made of the braze alloys to join the PH15-7 components of the honeycomb, the PH15-7 tubes, and the copper fins in a single brazing operation. The development of thin-wall tapered rectangular tubes, fabrication and evaluation of test panels, and the fabrication and structural and thermal testing of three full-size radiator condensers are described. It is concluded that although additional fabrication refinements are necessary to upgrade the overall quality, the feasibility of this radiator condenser was established.

AI-AEC-13066

NAA-SR-9908

TITLE THE DEVELOPMENT AND QUALIFICATION OF THERMAL CONTROL COATING FOR SNAP SYSTEMS

AUTHOR J. R. Crosby

9-30-65

ABSTRACT The results of evaluation and testing conducted to select thermal control coatings for various components of the SNAP 10A and SNAP 2 systems are reported. Out of the hundreds of coatings and coating systems investigated, the following were selected or developed (1) an aluminum-phosphate-bonded Cr-Co-Ni spinel with an overcoat of stannic oxide (AI-93) for the high thermal emittance/low solar absorptance ($\epsilon_{TH} > 0.90$, $\alpha_S < 0.30$) surfaces of the thermoelectric radiator and mercury radiator condenser, (2) commercial black enamel (PT-404, $\epsilon_{TH} > 0.90$) on surfaces requiring only high thermal emittance with plasma-applied Cr_2O_3 ($\epsilon_{TH} > 0.80$) as an alternate, (3) gold on molybdenum foil ($\epsilon_{TH} < 0.06$) on surfaces requiring low thermal emittance such as the core vessel and NaK piping and (4) a pattern of 44% of the area covered with clear chemical film, 54% silicone paint and 2% white acrylic lacquer for a low thermal emittance/high solar absorptance coating system ($\epsilon_{TH} < 0.20$, $\alpha_S/\epsilon_{TH} > 1.6$) on the heat shield to retain the system heat during the launch and prior to the startup. Test data are presented to verify the stability of these coatings under the simulated system operational conditions, including exposure to irradiation.

NAA-SR-11193

TITLE DEVELOPMENT OF NOISE ANALYSIS TECHNIQUES FOR MEASURING REACTOR COOLANT VELOCITIES

AUTHORS R. L. Randall, P. J. Pekrul, and G. R. Grayban

3-25-66

ABSTRACT The development of cross-correlation techniques and equipment for measurement of localized fluid velocities and flow distribution in hydraulic test models of SNAP reactor cores is described. The basic approach is experimental with supplemental theory introduced as needed. A number of experiments are discussed in addition to the S8DS core baffle and velocity profile studies.

NAA-SR-12564

TITLE SNAP 8 DEVELOPMENT REACTOR (S8DR) CORE THERMAL AND HYDRAULIC PERFORMANCE

AUTHOR P. M. Magee

3-10-68

ABSTRACT This report summarizes the anticipated steady-state thermal and hydraulic performance of the SNAP 8 Development Reactor (S8DR). The average thermal performance of the reactor is defined in terms of an overall heat-transfer coefficient (UA). This is presented as a function of temperature for beginning-of-life and end-of-life conditions.

AI-AEC-13066

AI-AEC-13087

TITLE ZIRCONIUM HYDRIDE REACTOR CORE HYDRAULIC STUDIES SUMMARY REPORT

AUTHOR B. J. Ostermier

6-73

ABSTRACT: A summary is presented for the series of water-hydraulic tests performed to evaluate the flow characteristics of the various zirconium hydride reactors. Test models, test methods, instrumentation, and results are described. Tests were conducted to develop inlet plenum configurations and fuel element fin configurations. Inlet plenum configurations were evaluated on their ability to produce desired in-core flow distributions, and to provide a uniform pressure at the inlet grid plate surface without any undesirable flow characteristics. A series of inlet plenum tests was also conducted to determine any flow behavior which might have contributed to the operational problems of the S8DR. Fuel element fin configurations were evaluated on their ability to produce interchannel flow mixing within the reactor core. Pressure drop determinations were made for all plenum and fuel element fin configurations.

AI-AEC-13088

TITLE ZIRCONIUM HYDRIDE REACTOR CORE HEAT TRANSFER STUDIES SUMMARY REPORT

AUTHORS A. N. Gallegos and A. R. Marchese

6-73

ABSTRACT This report presents the results of an experimental study showing the influence of rod spacers on the heat transfer to a liquid metal (NaK-78) in a turbulent longitudinal flow through a closely spaced rod bundle. The rod bundle consisted of 19 electrically heated elements, which were arranged on hexagonal arrays having a pitch-to-diameter (P/D) ratio of 1.05. Two rod configurations, each with a different type of rod spacer, were investigated. The first configuration utilized ring spacers which were banded to the outer surface of each test rod. For the second configuration, helical-fin spacers were attached to the cladding tubes. Besides maintaining uniform rod spacing, the purpose of the rod spacers was to promote increased fluid turbulence and mixing.

Heat transfer results for the cladding circumferential temperature distribution and the rod-average film coefficients were obtained over a range of heating and flow conditions of interest to compact reactor cores. For the rod-average film coefficient, the helical fin configuration showed a significant improvement over the predicted smooth tube results. In the flow range investigated, the heat transfer results were found to be essentially independent of the coolant Peclet number.

AI-AEC-13066

2-106

TI-652-240-006

TITLE THERMAL TRANSIENT AND DEFLECTION ANALYSIS OF INTERNAL REFLECTOR - 5-kwe REACTOR

AUTHORS E. J. Donovan and C. T. Kudija

8-2-72

ABSTRACT Thermal transient analyses were performed on the stainless steel clad internal reflector of the 5-kwe reactor. The assumed thermal transient was a linear NaK temperature ramp from 116 to 260°F at a rate of 2.4°F/sec. This approximates the most severe portion of a reactor startup resulting from a 0.0025-¢/sec insertion with a 20-min hold at 15¢ excess

Three internal reflector designs were evaluated (1) NaK-filled, (2) helium back-filled, and (3) evacuated. Although all designs would probably prove to be acceptable, the benign behavior of the NaK-filled design makes it the preferred choice

TI-652-240-007

TITLE RESULTS OF THE ADVANCED DIFFUSION-DILUTION MIXING TEST MODEL AS APPLIED TO THE HOBBED-FIN 19 ELEMENT BUNDLE DATA

AUTHOR R. V. Anderson

9-1 72

ABSTRACT Thorough investigation of results from all previous testing of Right-Left-Neutral (R-L-N) fin geometries has led to the development of a mixing theory which postulates the existence of two distinct types of flow within the tricusp. This advanced mixing theory predicts axial saline concentration distributions which agree well with experimental data. A mixing correlation has resulted from this theory which is empirically matched to several sets of experimental data. This mixing correlation is dependent only upon the geometry of the R-L-N finned element being considered.

TI-652-310-002

TITLE 5-kwe REFERENCE CONCEPTUAL DESIGN - SYSTEM HYDRUALICS

AUTHOR A. F. Lillie

25-72

ABSTRACT The hydraulic analyses, including power module hydraulics, for the 5-kwe Space Power System reference conceptual design (Dwg N652630001) are presented.

AI-AEC-13066

2-107

TI-652-610-001

TITLE: 5-kwe SYSTEM THERMAL LOSSES AND PIPING INSULATION PERFORMANCE

AUTHOR: J. A. Beoletto

9-19-72

ABSTRACT: Heat balances performed for components of the 5-kwe TE SNAP System during full power, steady-state operation are reported and summarized. Radiation shunt losses from the primary NaK system to the secondary, heat rejection loop require the application of a low-emittance foil on the primary loop piping to minimize thermal shunting.

TR-652-360-001

TITLE: NaK HEAT TRANSFER STUDY FOR A 19-ROD BUNDLE (P/D = 1.05)
EMPLOYING HELICAL-FIN SPACERS

AUTHOR: A. R. Marchese

7-26-72

ABSTRACT: An experimental heat transfer study is reported for the case of longitudinal flow of a liquid metal (NaK-78) through a closely spaced rod bundle. The rod bundle consisted of 19 electrically heated elements which were arranged on hexagonal arrays having a pitch-to-diameter (P/D) ratio of 1.05. Helical-fin spacers were attached to the cladding tubes to promote increased intrachannel turbulence and interchannel mixing, or crossflow. In addition, the fins served to maintain symmetric and positive rod spacing. Rod azimuthal temperature distributions were determined from thermocouples which were welded to the inside cladding tubes. Instrumentation was also provided to measure the coolant temperature distribution at the inlet and outlet to the rod bundle, the power supplied to each rod, and the coolant flowrate. The instrumentation provided the necessary measurements for determining the rod-average heat transfer coefficient.

Heat transfer results for the cladding circumferential temperature distributions and the rod-average film coefficients were obtained over a range of heating and flow conditions of interest to a compact ZrH reactor core. Because of the increased intrachannel turbulence created by the helical-fins, negligible rod azimuthal temperature variations and significantly improved rod-average film coefficients were observed. In the flow range investigated, the heat transfer results were found to be independent of the Peclet number. In addition, the effect of fin-induced interchannel mixing, or crossflow, was to significantly reduce coolant channel-to-channel temperature differences. The increased mixing should do a great deal in reducing cross fuel element ΔT 's and the resulting bending stresses in an actual ZrH reactor core.

TI-653-210-004

TITLE: TEMPERATURE DISTRIBUTION IN THE SPF REACTOR SHIELD PLUG DURING OPERATION

AUTHOR: L. D. Montgomery

1-17-72

ABSTRACT: Steady-state temperature distributions in the reactor support ring, shield plug, and actuator portions of the SPF reactor have been determined. The reactor was assumed to be operating at 300 kw with a coolant inlet temperature of 1125°F. Heat generation rates within the reactor support plate, shield material, and actuator motors were included. The results of this analysis indicate that a cold plate will not be necessary between the shield plug and the actuator and gear set support ring.

AI-AEC-13066

TI-653-220-002

TITLE THERMAL ANALYSIS OF THE SPF VOID BACKED REFLECTOR ASSEMBLY

AUTHOR E. M. Larson

11-10-71

ABSTRACT The results of a study to assess the effect of cold wall and reflector emissivity on SPF reflector temperature are presented. The results indicate that cold wall and reflector emissivities of 0.6 will yield reflector temperatures under 1200°F

TI-653 240-005

TITLE 300-kwt CORE CONCEPTUAL MODEL THERMAL/HYDRAULIC CHARACTERISTICS

AUTHOR E. Moody

7-27-71

ABSTRACT The 300-kwt, 199-element, NASA Lewis/AEC core conceptual model has been analyzed to determine its thermal-hydraulic characteristics using the GEOM-3 code. Stackups of tolerances and fuel rod asymmetry patterns were used to ascertain cross-element ΔT 's. Both zoned and uniform spacing were analyzed with a somewhat lower fuel temperature and cross-element ΔT found for zoned spacing. With the models considered, the core design appears adequate to limit thermal gradients to approximately 32°F. Bypass flow should be controlled to prevent excessive edge element ΔT 's.

TI-653-500-003

TITLE FURTHER TRANSIENT INVESTIGATIONS FOR THE SPF REACTOR/BRAYTON PCS COMBINED SYSTEMS TEST

AUTHOR R. V. Anderson

2-4-72

ABSTRACT The transient simulation model for the SPF Reactor/Brayton PCS Combined System has been further modified to include scram simulations, PCS shutdown simulations, NASA reference PCS startup simulations, and room temperature reactor startup simulations. Several additional NASA reference PCS startups have been performed to scope the effects from the Brayton ramp-up. Several scram simulations have been completed to show interactive effects between the reactor and PCS during system failure transients.

TR-653-210-002

TITLE OPEN CHANNEL PLENUM MODEL HYDRAULIC DEVELOPMENT TEST - FLARED INLET LINES

AUTHOR C. R. Davidson

3-14-72

ABSTRACT This report presents the test results of the hydraulic studies on the SPF open channel inlet plenum model with flared inlet lines. The work is a continuation of a series of tests, the initial phase, of which, was reported in TR-653-210-001. An evaluation was made of the effectiveness of 4- and 5-vane diffusers in 2-by-3 straight flared inlets, 5-vane diffusers in 2-by-3 curved flared inlets, and 5-vane diffusers in 2-by-4 straight flared inlets. The 5-vane diffusers in 2-by-3 straight flared inlets provided the flattest pressure profiles at the grid plate and were recommended for incorporation into the plenum design.

TR-653-360-001

TITLE EXPERIMENTAL STUDY OF HEAT TRANSFER TO NaK FLOWING IN-LINE
THROUGH TIGHTLY PACKED ROD BUNDLES

AUTHOR A. R. Marchese

10-10-71

ABSTRACT An experimental heat transfer study is reported for the case of longitudinal flow of NaK through tightly packed rod bundles having pitch-to-diameter (P/D) ratios of 1.00 and 1.05. Although the flow was turbulent, the Peclet number range was low enough such that eddy conduction was negligible. Fully established heat transfer results for the cladding azimuthal temperature distributions and rod-average film coefficients were obtained over a range of power and flow conditions of interest to the ZrH reactor

TI-696-21-004

TITLE FLOW LEAK THROUGH RECTANGULAR SLITS IN ZrH REACTOR OUTLET
PLENUM

AUTHOR A. R. Marchese

11-12-69

ABSTRACT Results of calculations are presented here for the inlet NaK flow leak through rectangular slits in the Reference ZrH Reactor outlet plenum. For a total reactor pressure loss of 0.65 psi, the flow leak is 1.74% of the total flowrate. The flow leak causes the NaK mixed-mean outlet temperature to be reduced from 1300 to 1298°F. Also, for a fixed core thermal power of 600 kw, the NaK temperature difference across the core is increased from 200 to 203°F, which means a lifetime penalty of about 300 hr.

TI-696-24-045

TITLE He and H₂ FREE MOLECULE AND CONTINUUM HEAT CONDUCTION

AUTHOR E. Moody

11-26-69

ABSTRACT Gas conduction characteristics in the regime pertinent to SNAP 8 technology are derived using current values of mean-free path lengths and continuum conductances. Helium is included to permit determination of loading characteristics for startup conduction. The effect of the helium on gas gap conduction during steady-state operation is considered and found to be fairly significant, even for low-pressure loading at end of life.

TI-759-220-002

TITLE THERMAL ANALYSIS OF THE ZrH REACTOR 4-P₁ AND VOID-BACKED
REFLECTOR ASSEMBLIES

AUTHOR E. M. Larson

6-21-71

ABSTRACT A steady-state thermal analysis is presented for (1) a BeO-Ta₁₀W-B₄C poison back drum and (2) a Be void back drum. TAP listings and a results summary are presented for both drums. The results indicate that the poison back drum may be used with an 1150°F cold wall and the void back drum may be used with a 1000°F cold wall.

TI-759-230-003

TITLE CORE TEST REACTOR SHIELD COOLING SYSTEM ANALYSIS

AUTHORS E. M. Larson and R. D. Elliott

5-5-71

ABSTRACT System requirements for cooling the shield within the vacuum vessel for the core test reactor are analyzed. The total heat to be removed by the coolant system is <22,700 Btu/hr, with an additional 4600 Btu/hr to be removed by the 2-in.-thick steel plate below the shield. This latter heat is radiated to the walls of the vacuum chamber where the heat is removed by the cooling coils wrapped around the outside of the chamber. The maximum temperature of the concrete in the shield can be kept below 200°F if the shield plug walls are kept below 160°F. The walls of the two 'donut' shaped shield segments, which are cooled by the water from the shield and vessel cooling system, should operate below 95°F. The walls of the center plug, which are cooled with nitrogen, should operate below 100°F.

TI-759-240-007

TITLE ZrH REACTOR LATTICE SPACING (HEAT TRANSFER CONSIDERATIONS)

AUTHOR L. D. Felten

9-25-70

ABSTRACT Temperature calculations for a 295-element ZrH reactor at fuel element spacings from 0.010 to 0.065 in. showed a very small dependence of reactor temperature on element spacing. It was found that one variation in coolant channel area (2 zones) was sufficient to satisfactorily shape the radial flow profile for the core.

TI-759-240-016

TITLE EFFECT OF CLADDING WALL THICKNESS AND GAS GAP WIDTH ON CIRCUMFERENTIAL TEMPERATURE GRADIENTS AND FILM COEFFICIENTS FOR ROD BUNDLES HAVING A P/D = 1.00

AUTHOR A. R. Marchese

10-29-70

ABSTRACT: For the case of NaK flowing in-line through a rod bundle with P/D = 1.00, calculations are presented showing the effect of cladding wall thickness and gas gap width on the azimuthal temperature gradient and convective film coefficient. Calculations are given for a typical set of test reactor conditions.

TI-759-240-023

TITLE EFFECT OF P/D RATIO AND CLADDING THICKNESS ON THE 6θ TEMPERATURE RIPPLE FOR S8DR AND THE TEST REACTOR

AUTHOR A. R. Marchese

12-21-70

ABSTRACT For the case of NaK flowing in-line through a bundle of SNAP-type fuel elements, calculations are presented showing the effects of pitch-to-diameter (P/D) ratio and the cladding wall thickness on the 6θ temperature ripple for conditions of interest to both S8DR and the test reactor. The maximum 6θ temperature ripple for the inner zone (P/D = 1.045) and outer zone (P/D = 1.022) of the test core are predicted to be 28 and 34°F, respectively. The effects of fins and fin-induced mixing are not included in the analysis.

2. Pump

NAA-SR-5244

TITLE SNAP 2 REACTOR PUMP DEVELOPMENT PROGRAM (RADIAL GAP PERMANENT MAGNET PUMP)

AUTHOR S. Sudar 4-1-61

ABSTRACT The design basis, design details, and test results on four variations of radial gap permanent magnet rotor electromagnetic pump for the SNAP 2 system are described. Three of the designs utilized the conventional annular pump section while the fourth unit had a helically coiled flow passage. The performance of the latter pump exceeded the original design requirements and agreed reasonably well with the design predictions. The pump performance data in sodium or NaK test loop are presented in graphical form

NAA-SR-6313

TITLE SNAP 2 POWER CONVERSION SYSTEM TOPICAL REPORT NO. 21 NaK PUMP DESIGN AND TESTING

AUTHORS D. E. Foltz, B. A. Snoko, and J. Tiltins 6-27-63

ABSTRACT The development of a permanent-magnet, axial-gap, centrifugal NaK pump for the SNAP 2 Power Conversion System is reported. The design analysis procedures and mechanical features of several variations of the pump are described in detail. Data from performance and parametric tests, including the effects of several design parameters, are presented. The final design verification unit is reported to have achieved 4 psi head rise at the rated flow of 72 lb/min at 1000°F, with an efficiency of 23%. Recommendations for design improvements and a program to demonstrate reliability are also given.

NAA-SR-11198

TITLE DEVELOPMENT OF THE THERMOELECTROMAGNETIC PUMP FOR THE 3-kwe MERCURY-RANKINE SYSTEM

AUTHOR H. H. Hanlin 9-1-65

ABSTRACT The development history of the prototype electromagnetic NaK pump for the SNAP Mercury-Rankine Power System is reported. The prototype unit is a dc electromagnetic pump, self-powered by chromel-constantan thermoelectric (TE) elements cooled on the cold leg by circulating mercury of the PCS. The basic engineering analysis, design configuration, and material selection studies are included. Extensive braze material and fabrication methods evaluation for the joining of the TE elements to the throat and the cold plate, and of the cold plates to the cooling pipes and the return buses is described. Fabrication of ten prototype units and the testing on these units are described. From the results of extensive performance tests conducted on ten units, including approximately 18,000 hr of operation on four units (648 to 6110 hr) and severe thermal shocking cycles, it was concluded that the prototype pump design satisfied the design objectives, and that the analytical design methods were verified.

NAA-SR-Memo-11841

TITLE POST-EXAMINATION OF SNAP 10A THERMOELECTRIC PUMPS NUMBERS 16, 31, AND 32

AUTHOR R M. Willard 2-9-66

ABSTRACT The post-test examination of the SNAP 10A pumps, built with and without a diffusion barrier at the copper shoe-iron shoe interface of the thermoelectric element, are reported. Without the diffusion barrier, considerable change in the microstructure of the element near the junction to the iron shoe on the hot leg were evident after 8200 hr of operation with 1100°F inlet NaK on one unit and 9388 hr with 1010°F inlet NaK on another unit. The source of contaminants was concluded to be the braze alloy used at the copper shoe-iron shoe bond on the "N" elements and the copper shoe of the "P" elements (no braze alloy is used at the copper-iron interface). On one unit built with an iron foil diffusion barrier at the copper shoe-iron shoe interface, no indications of microstructure changes were found after 17,000 hr of operation with 1010°F NaK and additional 360 hr with 1100°F NaK.

NAA-SR-Memo-11930

TITLE PUMPS, LIQUID METAL, EVALUATION FOR USE IN SNAP 8 COMPACT THERMOELECTRIC CONVERTER NPU CONFIGURATION

AUTHOR J V. Addison 4-22-66

ABSTRACT A literature survey is described that was made to determine the state-of-the-art on liquid metal pumps for space applications. A trade-off study of pumps was made with respect to various SNAP 8 Compact Thermoelectric Converter System configurations. The results of this trade-off study are presented.

AI-AEC-Memo-12692

TITLE SNAP 8 PUMP DESIGN AND ANALYSIS

AUTHOR Staff 4-30-68

ABSTRACT The parametric analytical studies conducted to select the conceptual design of the dc conduction pumps for the primary, intermediate, and the heat rejection NaK loops of the SNAP 8 reactor power system are reported. Evaluation of integral and separate source pumps, direct-radiating and compact thermoelectric power modules, and series and parallel flows through multiple pump throats are included. A common dual throat (series flow) pump designed for the most critical requirements is recommended for each quadrant of the three NaK loops. For the primary loop, two of these pumps would be placed in series with the power from the two-module compact thermoelectric converter flowing in series through the two pumps. A single pump would be placed in each of the other two loops with the power from a second two-module thermoelectric converter flowing in series through these two pumps.

AI-AEC-13091

TITLE SPACE NUCLEAR SYSTEM THERMOELECTRIC NaK PUMP DEVELOPMENT SUMMARY REPORT

AUTHOR J. I. Johnson 6-73

ABSTRACT This report summarizes the engineering, design, fabrication, and test history of the dual-throat thermoelectric NaK development pump, and the engineering and design status of a similar prototype pump intended for use on the 5-kwe Reactor Thermoelectric System. The history of dc pump development and testing on previous SNAP programs is also summarized.

AI-AEC-13066

TI-652-330-005

TITLE DUAL THROAT NaK PUMP PERFORMANCE EVALUATION
(CONCEPTUAL DESIGN OF PROTOTYPE)

AUTHOR J. L. Johnson 10-2-72

ABSTRACT: An evaluation of the performance of the dual throat NaK pump for the 5-kwe Reactor TE System is discussed. Performance at nominal operating conditions and at startup conditions is predicted. The basis for selection of this pump design is presented.

TI-696-13-002

TITLE REFERENCE DESIGN ZrH REACTOR-TE - DC CONDUCTION PUMP

AUTHOR W. D. Leonard 1-3-68

ABSTRACT: The results of a test to determine the performance of a triple-pass pump are reported. The testing included a magnet system test, the pump resistance test, the pump performance test, and the pump hydraulic loss test. The results are compared to predicted performance.

TI-696-30-001

TITLE AUXILIARY PUMP CONCEPTUAL DESIGN

AUTHOR W. D. Leonard 3-13-69

ABSTRACT A conceptual design has been prepared for the auxiliary pump for the ZrH Reactor Thermoelectric System. These pumps maintain flow during system standby, provide startup flow, and provide a dc signal proportional to flow.

TI-696-31-002

TITLE HELICAL INDUCTION PUMP DESIGN

AUTHOR L. E. Donelan 7-17-70

ABSTRACT Preliminary design work resulting in the calculation of a pump based on the helical induction principle is reported.

TI-696-41-010

TITLE NaK PUMP PERFORMANCE

AUTHOR W. D. Leonard 2-11-69

ABSTRACT: The performance of the dc conduction pumps for the ZrH Reactor thermoelectric system has been calculated for operation both at design conditions and off-design conditions. The results are presented in terms of pressure developed and voltage required for the pump.

3. NaK

TR-653-360-001

TITLE EXPERIMENTAL STUDY OF HEAT TRANSFER TO NaK FLOWING IN-LINE
THROUGH TIGHTLY PACKED ROD BUNDLES

AUTHOR A. R. Marchese

10-10-71

ABSTRACT An experimental heat transfer study is reported for the case of longitudinal flow of NaK through tightly packed rod bundles having pitch-to-diameter (P/D) ratios of 1.00 and 1.05. Although the flow was turbulent, the Peclet number range was low enough such that eddy conduction was negligible. Fully established heat transfer results for the cladding azimuthal temperature distributions and rod-average film coefficients were obtained over a range of power and flow conditions of interest to the ZrH reactor.

TI-759-260-001

TITLE STABILITY OF BERYLLIUM IN NaK

AUTHOR R. Van Houten

9-21-70

ABSTRACT The successful long-time use (30,000 to 40,000 hr) of Be in high-temperature NaK (nominal 1200°F) is problematical. Particular concerns are (1) the effects of even trace quantities of oxygen on Be corrosion, (2) Be erosion by classic mass transfer, and (3) possible negative consequences associated with the formation of brittle phases resulting from contact of Be (direct or by deposition) with many of the materials in the reactor/heat exchanger loop.

4. Mercury Boiler

NAA-SR-6315

TITLE SNAP 2 POWER CONVERSION SYSTEM TOPICAL REPORT No 23
COMPACT BOILER DEVELOPMENT

AUTHORS R. J. Zibbra and D. A. Oberacker

2-64

ABSTRACT This report describes selection of the compact (5C) boiler design concept as well as thermal, hydrodynamic, and mechanical design and stress analysis of the 5C boiler. Results of thermal performance and acceptance tests on the 5C-1, 5C-2, 5C-4, and 5C-5 boilers and the acceptance vibration test of the 5C-3 boiler are also presented. The compact boiler effort produced the highly successful Model 5C boiler that weighs only 50% of the specification weight.

NAA-SR-11949

TITLE EFFECTS OF SOME SURFACE AND SYSTEM CONTAMINANTS ON MERCURY
POOL BOILING HEAT TRANSFER

AUTHOR T. T. Shimazaki

7-15-66

ABSTRACT The effects of some surface and system contaminants on mercury pool boiling heat transfer were investigated to (1) explore the validity of the loss-of-adequate-wetting hypothesis proposed as an explanation for Mercury Rankine Program boiler deconditioning, and (2) determine to which contaminants the loss of wetting is attributable. The results indicate that loss of adequate wetting is the most likely cause of boiler deconditioning, and that the effect of oil contamination on loss of wetting is much more severe and permanent than air or water contamination.

AI-AEC-13066

TITLE: A STUDY OF EXPANSION OF MERCURY VAPOR

AUTHORS: B. J. Ostermier, R. D. Keen, and V. H. Heiskala

3-15-66

ABSTRACT Series of tests conducted to determine the supersaturation of mercury vapor attainable by isentropic expansion through a nozzle are described. The details of the test assembly, methods of testing, the recorded test data, and evaluation of the test results are presented. It is concluded that although relatively high levels of supersaturation were achieved during the experiments, the theoretical limits were not attained. The presence of mercury droplets, of sufficient size to initiate nucleation, or the error in the value of the surface tension of the droplet used in the theoretical calculation are two possible explanations of the divergence. Modification of the test assembly and further testing to evaluate these factors are recommended.

G. MERCURY-RANKINE (CRU) CONVERTER DESIGN, ANALYSIS, AND TESTING

NAA-SR-6308

TITLE SNAP 2 POWER CONVERSION SYSTEM TOPICAL REPORT No. 16, 2500-HOUR ENDURANCE TEST OF MERCURY RANKINE CYCLE POWER SYSTEM

AUTHOR P. E. Grevstad 9-61

ABSTRACT The 2500-hr endurance performance test conducted by TRW on the mercury rankine cycle power conversion machinery is described. The power conversion system consisted of the turbomachinery (turbine, alternator, and pump mounted on a common mercury bearing supported shaft) originally designed for ground testing of the SNAP 1 (radioisotope power) system, electrically heated mercury boiler, and air cooled condenser. The 2500-hr run was completed with only minor problems and the deterioration of the alternator output was the only significant performance anomaly. Post-test inspection indicated all components to be in good condition except for erosion of the pump impeller attributed to air entrained in the mercury. The need to develop methods to hermetically seal the mercury was concluded. If this were accomplished, the goal of 1-hr unattended operation of the power conversion system should be attained.

NAA-SR-6310

TITLE SNAP 2 POWER CONVERSION SYSTEM ROTATIONAL SPEED CONTROL TOPICAL REPORT No. 18

AUTHORS W. E. Dauterman and I. J. Viton 12-1-63

ABSTRACT The parasitic load speed control for the SNAP 2 power conversion system consists of a frequency discriminator low-level magnetic amplifier, magnetic amplifier power stages, and a resistive load. The design and development of this parasitic load speed control is described. Experimental results are given to confirm the breadboard design approach.

NAA-SR 6311

TITLE THE SNAP 2 POWER CONVERSION SYSTEM TOPICAL REPORT No. 19 MERCURY MATERIALS EVALUATION AND SELECTION

AUTHOR J. F. Mejedlik 12-63

ABSTRACT SNAP 2 is the designation for a 3-kw nuclear auxiliary power unit to be used in a satellite vehicle. The SNAP 2 system consists of a reactor heat source, a boiler, a condenser, a mercury rankine engine, and an alternator. The corrosion and subsequent mass transfer resulting from the use of mercury as the thermodynamic working fluid are important considerations in the selection of materials for the SNAP 2 system. Consequently, corrosion and mass transfer behavior have been under study for the past 4 years. This report presents the results of this study for fiscal year 1962. The program was performed under a subcontract to Atomics International as part of the Atomic Energy Commission Contract AT(11-1)-GEN-8.

NAA-SR-6312

TITLE SNAP 2 POWER CONVERSION SYSTEM TOPICAL REPORT No. 20
CRU V DESIGN

AUTHOR Staff TRW 5-31-63

ABSTRACT CRU V design and development was undertaken to provide a SNAP 2 turbo-alternator package that will meet SNAPSHOT flight objectives. This report describes the selection of the design concept, CRU detail design features, performance estimates, and the plumbing module design. The layout design fulfills the majority of the specific CRU V design objectives. Potential problems are discussed but they are not considered to jeopardize accomplishment of current SNAP 2 program objectives.

NAA-SR-6314

TITLE SNAP 2 POWER CONVERSION SYSTEM TOPICAL REPORT No. 22
JET-CENTRIFUGAL MERCURY PUMP DESIGN AND PERFORMANCE ANALYSIS

AUTHORS C. W. Grennan and R. A. Lewis 2-64

ABSTRACT Two different mercury pumping units were designed and tested for the SNAP 2 system. These units have missions in the ground test and prototype flight units, respectively. The pumps are somewhat different in detail because the requirements of the ground and flight units differ. This report presents the design analysis and performance results obtained during the 1963 fiscal year

NAA-SR-6315

TITLE SNAP 2 POWER CONVERSION SYSTEM TOPICAL REPORT No. 23
COMPACT BOILER DEVELOPMENT

AUTHORS R. J. Zibbra and D. A. Oberacker 2-64

ABSTRACT This report describes selection of the compact (5C) boiler design concept as well as thermal, hydrodynamic, and mechanical design and stress analysis of the 5C boiler. Results of thermal performance and acceptance tests on the 5C-1, 5C-2, 5C-4, and 5C-5 boilers and the acceptance vibration test of the 5C-3 boiler are also presented. The compact boiler effort produced the highly successful Model 5C boiler that weighs only 50% of the specification weight.

NAA-SR-6317

TITLE SNAP 2 POWER CONVERSION SYSTEM TOPICAL REPORT No. 25
THREE-PHASE ALTERNATOR ELECTRICAL DESIGN AND PERFORMANCE TESTING

AUTHOR Z. L. Reineks 10-15-64

ABSTRACT The CRU V design and development program initiated in 1962 was directed toward elimination of inadequacies noted in CRU III and CRU IV designs and toward development of a combined rotating unit fully qualified for flight. This report discusses rotor and stator design changes incorporated in the CRU V alternator and the effect of these modifications on electrical performance. Design and development of electrical output terminals are also briefly reviewed.

AI-AEC-13066

NAA-SR-6318

TITLE SNAP-2 POWER CONVERSION SYSTEM TOPICAL REPORT No. 26
DESIGN, DEVELOPMENT AND PERFORMANCE OF CRU IV AND CRU IVM
MERCURY VAPOR TURBINES

AUTHOR A. J. Stromquist

3-64

ABSTRACT The CRU IV turbine was designed to provide a definite and substantial gain in efficiency over previous SNAP 2 turbines. The CRU IV turbine had a partial-admission first stage and a full-admission second stage with optimum ratios of wheel speed to vapor velocity. The efficiency of the initial CRU IV design was 53%, a 10% gain in efficiency. Fabrication deviations, internal leakage, and pressure losses were corrected in the CRU IVM redesign, and an additional gain of greater than 10% in turbine power and efficiency was realized, as evidenced by testing CRU IVM turbine efficiency averaged 58% on a total-to-static, blade-to-blade basis, and 57% on a flange-to-flange basis. Although the CRU IVM turbine exceeded specification efficiency, calculations reveal that potential efficiencies of greater than 60% are possible at design point with an improved interstage collector.

The techniques used to calculate turbine performance at design point were applied to exact test data conditions and fabricated turbine geometry and produced results that agree within 2% with the test results.

NAA-SR-6320

TITLE MERCURY RANKINE PROGRAM (SNAP-2) DEVELOPMENT OF LIQUID-MERCURY-LUBRICATED BEARINGS, TOPICAL REPORT NO 28

AUTHOR O. Decker

6-63

ABSTRACT This report defines the analytical and experimental results obtained with fluid film bearings lubricated by liquid mercury, including the design approach, analytical techniques, procedures, test results, and experimental activity performed by TRW Inc. as a facet of the Mercury Rankine Power Conversion Program during the period of January 1961 through June 1963.

This report consist of the following six volumes

Volume I	Analytical Design Approach and Status of Bearing Systems (With Appendixes)
Volume II	Plain Bearing Experimental Results
Volume III	Tilting-Pad Bearing Experimental Results
Volume IV	Three-Sector Bearing Experimental Results
Volume V	Three-Pad Bearing Experimental Results
Volume VI	Spiral-Groove Thrust Bearing Experimental Results

NAA-SR-9801

TITLE SNAP MERCURY RANKINE PROGRAM MERCURY CONDENSING EXPERIMENTS

AUTHOR R. P. Forslund

8-1-64

ABSTRACT An experimental study program of the various basic operating characteristics of mercury condensers for space applications is described. Included are techniques for predicting pressure drops in a variety of environments, condenser stability, the effects of noncondensibles, startup dynamics, and the differences between wetting and nonwetting condensing.

AI-AEC-13066

2-119

NAA-SR-11597

TITLE: MERCURY RANKINE PROGRAM ALTERNATOR BARE SEAL-UNSEALED ALTERNATOR

AUTHORS: H. Horii and A. I. Wesolowski 3-15-66

ABSTRACT: This report describes an unsealed stator concept that may be used to exclude mercury from the Combined Rotating Unit of the Mercury-Rankine Power Conversion system.

NAA-SR-Memo-11895

TITLE: CRU-V BEARING LIFE ESTIMATE

AUTHOR: J. S. McDonald 5-17-66

ABSTRACT: The factors affecting the life of the mercury-lubricated bearings on the SNAP 2 CRU-V are analyzed and the available test data are evaluated to derive a model to predict lifetime. The operational envelope applicable to this prediction model is defined. Because the failure criteria are based on worst conditions observed to date on bearings still operating and the degradation rates are more severe than experimentally observed, the model is reportedly very conservative. On the basis of this model, a 50,000-hr minimum life of the CRU-V bearing is predicted and a failure rate of $2.17 \times 10^{-5}/\text{hr}$ at a 50% confidence level is postulated. Design modifications which should improve these values are also recommended.

NAA-SR-11943

TITLE: AN ANALOG COMPUTER STUDY OF THE DYNAMICS OF THE MRP COMBINED ROTATING UNIT SHAFT-BEARING SYSTEM

AUTHORS: J. H. Bick and A. G. Presson 7-15-66

ABSTRACT: A mathematical model formulated for the Combined Rotating Unit shaft-bearing system is discussed. The model includes the effects of shaft imbalance, magnetic rotational load, unidirectional load, and spin speed on the shaft dynamics. This report includes all the theory and assumptions used in the derivations. Recommendations are made for improvements and suggested future investigations.

NAA-SR-11944

TITLE: A DIGITAL COMPUTER STUDY OF THE DYNAMICS OF THE COMBINED ROTATING UNIT SHAFT-BEARING SYSTEM

AUTHOR: J. M. Howard 7-15-66

ABSTRACT: This report summarizes the objectives, method of approach, and results of a digital computer rotordynamic analysis of the SNAP 2 CRU-V. Bearing characteristics are in the form of curve fit equations based on empirical data. The effects of shaft bending, differences in fabrication and balancing tolerances, and varying orientation of the rotating magnetic load are evaluated.

AI-AEC-13066

NAA-SR-11945, Volume II

TITLE: MERCURY RANKINE PROGRAM INTEGRATED SYSTEM TEST (PSM-3),
VOLUME II - DATA CORRELATION

AUTHORS: V. Heiskala, et al.

10-25-66

ABSTRACT: The test data from the tests described in Volume I are compared and evaluated against the analytically predicted performance of the Mercury Rankine system and its boiler, CRU, and the radiator-condenser. In general, close correlations were achieved. Where discrepancies occurred, shortcomings of the instrumentation utilized or the lack of sufficient sophistication in the analytical models are recognized. Further tests to generate the necessary data to improve the analytical methods are recommended.

NAA-SR-11949

TITLE: EFFECTS OF SOME SURFACE AND SYSTEM CONTAMINANTS ON MERCURY
POOL BOILING HEAT TRANSFER

AUTHOR: T. T. Shimazaki

7-15-66

ABSTRACT: The effects of some surface and system contaminants on mercury pool boiling heat transfer were investigated to: (1) explore the validity of the loss-of-adequate-wetting hypothesis proposed as an explanation for Mercury Rankine Program boiler deconditioning, and (2) determine to which contaminants the loss of wetting is attributable. The results indicate that loss of adequate wetting is the most likely cause of boiler deconditioning, and that the effect of oil contamination on loss of wetting is much more severe and permanent than air or water contamination.

NAA-SR-Memo - 12080

TITLE: CRU-V TURBINE LIFE ESTIMATE

AUTHOR: J. S. McDonald

7-11-66

ABSTRACT: An estimate was developed of the expected operating life of the CRU-V mercury-vapor turbine. It was estimated that turbine life will be 50,000 hr, or more, provided that boiler performance is satisfactory and that excessive corrosion product carryover from the boiler does not occur.

NAA-SR-Memo-12134

TITLE: CRU-V ALTERNATOR LIFE ESTIMATE

AUTHOR: J. S. McDonald

8-29-66

ABSTRACT: An estimate was developed of the expected operating life of the CRU-V alternator. It was concluded that the CRU-V alternator, operated with the stator evacuated, should be suitable for at least 100,000-hr operation.

NAA-SR-Memo-12139

TITLE: RADIATOR CONDENSER DESIGN MANUAL

AUTHORS: R. C. Smith, J. R. Crosby, E. A. Elliott, and R. D. Galletly

11-1-66

ABSTRACT: The design, analytical, and fabrication techniques used at Atomic International in the development of three types of radiator condensers for use on the SNAP 2 Mercury Rankine Program are presented. The successful testing of each type indicates the appropriateness of the techniques presented. The use of the manual for future radiator designs should aid in achieving an optimum design with a minimum expenditure of effort in the development phase of future programs.

AI-AEC-13066

NAA-SR-12181, Vol I

TITLE: FINAL SUMMARY REPORT - SNAP 2/MERCURY RANKINE PROGRAM REVIEW,
VOLUME I

AUTHORS: R. L. Walkerstedt, J. S. McDonald, and D. E. Ellis 6-15-67

ABSTRACT: This summary report details the technical aspects of the development program for the SNAP 2/Mercury Rankine System from the start of the program in 1958 through FY 1967. The individual system components descriptions, test results, and status of the development are presented together with a description of, and test results from, the system tests operated during the program. The engineering analytical accomplishments made during the program are also described.

NAA-SR-12233

TITLE: MERCURY RANKINE PROGRAM BOILER DEVELOPMENT SUMMARY

AUTHOR: E. B. Baumeister 1-15-67

ABSTRACT: The Mercury Rankine Program NaK-to-mercury boiler development program history and present status are presented. Also included are the results of boiler conditioning studies, thermal-hydraulic boiler design procedures, and advanced application conceptual designs.

NAA-SR-12411

TITLE: ANALOG COMPUTER STUDY OF THE DYNAMICS OF THE MRP COMBINED
ROTATING UNIT SHAFT-BEARING SYSTEM

AUTHOR: D. Terris 6-29-67

ABSTRACT: Further study of the bearing and shaft dynamics of the Mercury Rankine Program combined rotating unit (CRU V) was undertaken. The analog computer simulation of this high-speed machine containing mercury-lubricated bearings was extended and improved.

NAA-SR-12437

TITLE: CRU-V DEVELOPMENT STATUS AND CAPABILITY

AUTHORS: D. E. Ellis, J. S. McDonald, and J. M. Howard 6-29-67

ABSTRACT: This report reviews in some detail the present development status of the SNAP 2/Mercury Rankine Program combined rotating unit (CRU-V). Included are the test results through September 1966. The pertinent operational characteristic of not only the individual components but also the results of component interactions are included. Fabrication and quality control considerations are also discussed. The additional development which would be required before the CRU would be considered operational is delineated.

AI-AEC-12850

TITLE: SNAP 8 AUTOMATIC STARTUP AND PCS MERCURY INJECTION SIMULATION TESTS

AUTHOR: R. F. Audette 10-31-69

ABSTRACT: SNAP 8 reactor dynamic tests have demonstrated the capability for fully automatic startup over eight decades of power from shutdown, and for automatically furnishing the heat load demand imposed by automatic startup of the flight system power conversion loop following reactor startup. The tests performed approximate predicted flight system operation.

AI-AEC-13066

H. THERMIONIC CONVERTER DESIGN, ANALYSIS, AND TESTING

NAA-SR-6077

TITLE: DESIGN STUDY FOR A THERMIONIC REACTOR POWER PLANT FOR SPACE APPLICATIONS

AUTHORS: C. K. Smith, J. W. Holland, and R. Hirsch 5-1-61

ABSTRACT: This report presents the results of a study to determine the design of a nuclear power plant for space applications. The advantages of thermionic power sources are summarized and the problem areas are defined.

NAA-SR-7661

TITLE: PERFORMANCE OF CESIUM THERMIONIC DIODES OPERATED IN SERIES - PARALLEL CIRCUITS

AUTHOR: J. W. Holland 2-1-63

ABSTRACT: Electrical power degradation from the operation of many series-parallel circuited cesium diodes in a thermionic reactor must be considered when a nonflattened nuclear power distribution exists over the volume of the reactor core. This experiment measures the loss of power and efficiency due to unequal heat inputs to series- or parallel-connected diodes, and studies the operating characteristics of a multiple-diode system.

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I. STRUCTURES

1. Structure Design, Analysis, and Testing

NAA-SR-8127

TITLE: SURVEY OF METEOROID PROPERTY AND DISTRIBUTION DATA RELEVANT TO SNAP 8 REACTOR AND SHIELD DESIGN

AUTHOR: C. R. Harden

4-30-63

ABSTRACT: This report is a survey of existing literature relative to the properties and distributions of meteoroids. Its objective is to predict the criteria which may be used to calculate the impairing effect of this meteoroidal phenomenon on the SNAP 8 space system and/or its components and other space vehicle structures.

NAA-SR-9054

TITLE: SNAP 2 STRUCTURAL AND DYNAMIC ANALYSIS

AUTHOR: R. A. Castle

5-1-64

ABSTRACT: The structural design criteria and the system basic loads for the SNAP 2 compact power unit are presented. Included in these structural criteria are the following environments: ground, prelaunch, launch, preorbital startup, and orbital startup and continuous operation. These data were then used to determine the structural integrity of the system and to satisfy the reliability requirements. A summary is presented of the more significant system structural and dynamic loads generated or analysis techniques developed on the SNAP 2 compact power unit.

NAA-SR-9906

TITLE: SNAP MERCURY RANKINE PROGRAM SNAP-2 STRUCTURAL EVALUATION PROTOTYPE SYSTEM

AUTHOR: W. B. Gruenberg

1-1-64

ABSTRACT: This report defines the SNAP 2, PSM-2, Phase-II prototype system program. The need for a structural test item is discussed and the resulting objectives presented. A summation of the design, fabrication, and assembly efforts is given. The test program is outlined and some preliminary results are discussed.

NAA-SR-12186

TITLE: SNAP 2 DEVELOPMENTAL REACTOR MOCKUP NO. 1 FINAL REPORT

AUTHORS: L. Maki, J. F. Milner, and B. R. Granich

6-15-68

ABSTRACT: The series of developmental tests conducted on the nonnuclear SNAP 2 reactor assembly are described. The complete reflector system and the shield assembly were initially acceptance tested and then mated to a nonnuclear (dummy fuel elements) core assembly. Subsequently, the reactor assembly successfully withstood the launch environment-simulating shock and vibration tests and performed as required to simulate the startup and long-term operation in a vacuum chamber that simulated space conditions while the assembly was being heated with hot NaK circulated through the core vessel. The detailed description of the test and performance of the system and the various components are reported.

AI-AEC-13066

AI-AEC-13090

TITLE: SPACE NUCLEAR SYSTEM VOLUME ACCUMULATOR DEVELOPMENT SUMMARY REPORT

AUTHOR: W. D. Whitaker 6-73

ABSTRACT: The engineering, design, and fabrication status of the volume accumulator units (VAU's) to be employed in the NaK primary and secondary coolant loops of the 5-kwe Reactor Thermoelectric System are described. Three identical VAU's are required - two for the primary coolant loop, and one for the secondary coolant loop. The VAU's utilize nested-formed bellows as the flexing member, are hermetically sealed, provide double containment, and utilize a combination of gas-pressure force and bellows-spring force to obtain the desired pressure regulation of the coolant loops. All parts of the VAU, except the NaK inlet tube, are to be fabricated from Inconel 718.

AI-AEC-13093

TITLE: DESIGN AND ANALYSIS OF RADIATOR-STRUCTURE FOR SPACE POWER SYSTEMS

AUTHORS: W. H. Dauterman and L. D. Montgomery 6-6-73

ABSTRACT: The design, analysis, fabrication, and development of the 5-kwe radiator-structure is shown. Thermal performance, meteoroid protection, structural capability during launch, development testing and space operation, material evaluation, and the configuration selection are described. The fin-tube development depends on the relative values of the thermal coefficients of expansion. The initial selection of aluminum fins and Type 316 stainless steel tubes was based on previous experience; however, the large differential in their expansion rates showed that an alternate, more compatible, combination was needed. Copper, stainless steel clad copper, boron-impregnated aluminum, and an independent radiator with a titanium structure were all considered as alternate materials. The final selection was Lockalloy fins with Type 304 stainless steel "D" tubes.

AI-67-TDR-35

TITLE: FLEXIBLE NaK ELEMENTS FOR DEPLOYABLE SPACE POWER SYSTEMS

AUTHOR: J. V. Facha 3-1-67

ABSTRACT: A variety of flexible piping elements are examined to assess their potential for application to deployable, fluid-carrying radiators. Flexibility of the elements, expressed as radius of curvature or angle of rotation, is related to material peak total strain so that the information developed can be applied without regard to specific material properties and to avoid the problems of non-linearity beyond elastic limits. Material properties of specific interest to the design of flexible elements are discussed and a number of possible radiator deployment concepts are shown.

AI-AEC-13066

TI-652-240-009, Rev A

TITLE: STRUCTURAL EVALUATION OF THE REACTOR VESSEL AND PIPING -
CONCEPTUAL DESIGN

AUTHORS: J. V. Facha and R. Leary

9-5-72

ABSTRACT: The structural design of the reactor vessel and piping (conceptual design) are evaluated for conformance to the requirements of the ASME code for nuclear vessels, incorporating Code Case 1331-5.

TI-652-320-007

TITLE: STRESS ANALYSIS OF RADIATOR/STRUCTURE DURING LAUNCH ACCELERATION

AUTHOR: S. Lee

10-3-72

ABSTRACT: The structural analysis of the Lockalloy/stainless-fin/tube-radiator/structure is shown. The radiator is capable of maintaining structural integrity when the system is subjected to TITAN IIIb launch accelerations.

TI-700-13-010

TITLE: PREDICTION OF ABLATION DEPTHS OF LITHIUM HYDRIDE SHIELD DURING
REENTRY, INCLUDING A STUDY OF THE EFFECTS OF NODE SIZE ON CALCULATED ABLATION RATES

AUTHOR: R. D. Elliott

11-11-69

ABSTRACT: Calculations were made using a one-dimensional thermal model of a section through the lithium hydride shield surrounding the ZrH Reactor. It was found that, for a representative reentry trajectory and for an assumed aerodynamic heating factor of 0.535, about 4.8 in. of hydride would ablate before the end of the heating period. The validity of the heating factor used has not yet been established, so these results should be considered as preliminary. The intensitivity of the calculated ablation rates to node thickness of interest in the thermal model was established.

TI-700-13-013

TITLE: IMPACT TEST DATA OF ZrH REACTOR SYSTEM

AUTHOR: S. L. Lee

3-4-70

ABSTRACT: As part of the post-impact disassembly studies of the ZrH Reactor, a series of impact tests have been conducted on the principal reactor materials. This report presents the basic equations used in determining the impact energy balance between the test specimens and the target at impact. With the aid of these equations, the energy absorption capacity of the materials tested is determined.

The effects of irradiation on Type 304 SS is presented indicating that the energy-absorbing capacity of the reactor vessel will not be affected by irradiation. Modeling laws are shown that indicate that it is possible to predict the post-impact configuration of the reactor by scale-model testing.

TR-700-13-003

TITLE: TEST REPORT ON IMPACT TESTS CONDUCTED ON SAMPLES OF 136 SS, LITHIUM HYDRIDE, TUNGSTEN, TANTALUM-10 TUNGSTEN, AND NORMAL ZIRCONIUM HYDRIDE FUEL MATERIAL

AUTHOR: J. M. Brammer

2-27-70

ABSTRACT: This report describes the results of the impact tests conducted on the principal materials of which the Reference ZrH Reactor system is composed. These materials were tested at impact velocities of from 100 to 850 ft/sec. The LiH, ZrH, and W specimens all fractured and broke up at an impact velocity of 100 ft/sec. The 316 SS and the Ta-10 specimens deformed plastically at impact velocities from 100 to 850 ft/sec. The SS specimens did, however, fracture at impact velocities above 700 ft/sec.

TR-700-13-004

TITLE: TEST REPORT OF IMPACT TEST CONDUCTED ON SS SHELLS, LiH PADS, INCO 800, AND COMPOSITE STRUCTURES FOR DETERMINATION OF K FACTORS AND VERIFICATION OF MODELING LAWS

AUTHOR: J. M. Brammer

8-17-70

ABSTRACT: This report describes the results of impact tests conducted on SS shells, LiH pads, Inco 800 cylinders, and composite models similar to the geometries and loading conditions found in the ZrH Reactor system. K factors were determined for these specimens. The test results were used to verify the modeling laws for impact testing and as a check of the applicability of an analytical model for predicting post-impact configurations.

2. Radiator Condenser Design, Analysis, and Testing

NAA-SR-5317

TITLE: THE SNAP-2 RADIATIVE CONDENSER ANALYSIS

AUTHORS: M. G. Coombs, R. A. Stone, and T. Kapus

-61

ABSTRACT: This report summarizes the design analysis for the SNAP-2 direct-condensing radiator. The design point selected in this report utilizes mercury as the working fluid to be condensed at 600°F.

AI-AEC-13093

TITLE: DESIGN AND ANALYSIS OF RADIATOR-STRUCTURE FOR SPACE POWER SYSTEMS

AUTHORS: W. H. Dauterman and L. D. Montgomery

6-6-73

ABSTRACT: The design, analysis, fabrication, and development of the 5-kwe radiator-structure is shown. Thermal performance, meteoroid protection, structural capability during launch, development testing and space operation, material evaluation, and the configuration selection are described. The fin-tube development depends on the relative values of the thermal coefficients of expansion. The initial selection of aluminum fins and Type 316 stainless steel tubes was based on previous experience; however, the large differential in their expansion rates showed that an alternate, more compatible, combination was needed. Copper, stainless steel clad copper, boron-impregnated aluminum, and an independent radiator with a titanium structure were all considered as alternate materials. The final selection was Lockalloy fins with Type 304 stainless steel "D" tubes.

AI-AEC-13066

TI-652-320-003

TITLE: 5-kwe SYSTEM RADIATOR THERMAL ANALYSIS

AUTHOR: L. D. Montgomery

8-7-72

ABSTRACT: The 5-kwe system thermal radiator reference design is presented in two parts. The first portion describes the reference meteoroid armor equation used in preliminary design. The second portion describes the current reference design.

TI-652-320-005, Rev A

TITLE: 5-kwe SYSTEM LOCKALLOY RADIATOR THERMAL ANALYSIS

AUTHOR: L. Montgomery

9-13-72

ABSTRACT: The reference design of the 5-kwe reactor system thermal radiator using Lockalloy as the fin material is presented in two parts. The first portion describes the reference meteoroid armor equation used in preliminary design. The second portion describes the selection of the current reference design.

AI-AEC-13066

BLANK

J. CONTROL AND INSTRUMENTATION

1. Component Development and Testing

NAA-SR-6490

TITLE DEVELOPMENT OF A LARGE METAL ULTRAHIGH VACUUM SIMULATION CHAMBER

AUTHOR F. J. Kamensky

11-1-61

ABSTRACT: The design and operational performance of a large 30-in. diameter by 48-in.-long space simulation chamber for the high-temperature high-vacuum environment testing of SNAP reactor components are described. The system reported is an all-metallic sealed unit pumped by a combination of water and liquid-nitrogen cold-trapped oil-diffusion pump, liquid nitrogen cryopanel, and ion pump. Internal quartz lamps are capable of heating the test components to 1000°F. Strip heaters and water cooling coils are built into the chamber walls for the initial system bakeout and subsequent chamber cooling. Continuous operation at pressure of 8×10^{-8} torr with specimen heated to 1000°F and without the operation of the cryopanel is reported. Pressure reduction to the 10^{-9} torr design range with the operation of the cryopanel is predicted.

NAA-SR-9634

TITLE LOW FLUX RADIATION EFFECTS ON ELECTRICAL AND ELECTRONIC COMPONENTS

AUTHORS M. N. Robinson, S. G. Kimble, and D. M. Walker

12-1-64

ABSTRACT: The methods and results of irradiation effects tests conducted on 5 types of transistors, 3 sizes of capacitors, several electrical insulation samples, and a SNAP 10A temperature switch are described. The performance of the electrical components measured periodically during the 11-day test at 140°F in a 10^{-4} torr environment which accumulated total dose of 10^{16} nvt fast neutrons and 4×10^8 r gamma are presented in series of computer-produced graphs. These data indicate that the H_{FE} and V_{CE} of the 2N1506A, 2N1711, SN101, 2N2865, and TIX 693 transistors are significantly affected in the 10^{12} to 10^{13} nvt range, tolerance of the 2.5 microfarad (29F155) capacitor was exceeded in the 10^{-2} to 10^{-3} nvt range, the capacitance of the 0.50 microfarad (M1-504) and of the 0.25 microfarad (M1-254) had decreased less than 15% at the end of the test, all 7 of the epoxy-type insulation samples changed in dimensions, became more brittle, and exhibited increase in electrical resistance at approximately 10^{14} nvt, and the performance of the SNAP 10A temperature switch was unaffected to approximately 10^{14} nvt.

NAA-SR-9644

TITLE: SNAP REACTOR MATERIAL DEVELOPMENT, ULTRAHIGH VACUUM FRICTION STUDIES

AUTHOR: L. G. Kellogg

6-2-64

ABSTRACT: Studies conducted to evaluate various potential material combinations for bearing and braking surfaces for components to be operated at high temperature ($\sim 1000^\circ\text{F}$) in ultrahigh vacuum environment are described. The material combinations evaluated are categorized as metal vs metal (with and without dry film lubricants), metal vs ceramic, metal vs carbon, ceramic vs ceramic (with and without dry film lubricants), and ceramic vs carbon. It is concluded that ceramic vs carbon-graphite ($\sim 40\%$ graphite) provides the optimum friction characteristics for bearings ($\mu \leq 0.4$). The addition of dry film lubricant, particularly MoS_2 , reduces the friction coefficient as long as the lubricant remains at the bearing surfaces. For braking purpose the combination of metal vs carbon provides the most uniform friction coefficient (0.4 - 0.8).

AI-AEC-13066

NAA-SR-9645

TITLE: SNAP 8 CONTROL DRUM ACTUATOR

AUTHORS: S. Giles and A. A. Marcus

12-15-64

ABSTRACT: The design description and results of limited tests on 3 types of control drum drive actuators for the SNAP 8 reactors are discussed. Included are the S8DRM actuator (essentially an improved SNAP 10A device), 1/2-degree stepper (designed to direct-drive the control drum), and the S8DR actuator. The detail design and selection basis of the S8DR actuator and the planned development and verification test program are presented.

NAA-SR-9684

TITLE: LOW FLUX NUCLEAR RADIATION EFFECTS ON SEMICONDUCTOR DEVICES

AUTHOR: M. N. Robinson

12-1-64

ABSTRACT: Tests and results of subjecting 2 types of transistors (2N744 and 2N1072), diode (PS760), and zenor diode (1N2032) to low flux nuclear radiation are described. Multiple graphs showing the performance of the devices as a function of dose (up to 1×10^{16} fast neutrons and 4.5×10^8 r gamma) are presented. It was concluded that both diode and zenor diode are relatively unaffected by dose up to 10^{13} nvt while the threshold for transistors may be approximately 1 decade lower.

NAA-SR-9778

TITLE: SNAP 10A EXPANSION COMPENSATOR

AUTHORS: S. Sudar, G. Paine, and L. Stone

10-1-65

ABSTRACT: The development and qualification test programs of the SNAP 10A NaK loop expansion compensator are reported. The details of the test fixtures and test methods are described and the status of the testings and preliminary test results are presented. It is stated that through the adoption of rigid in-process controls and development of relevant inspection methods and criteria, verified by testing, the quality of the AM-350 welded bellows has been improved significantly. This improvement, coupled with the conservatism of the elastic fixed beam theory used for the stress analysis of the bellows is expected to demonstrate the high performance reliability of the double containment expansion compensators.

NAA-SR-9885

TITLE: DEVELOPMENT, FABRICATION AND TESTING OF HONEYCOMB RADIATOR CONDENSER

AUTHORS: C. Scott and W. Engle

3-1-65

ABSTRACT: The extensive development efforts conducted to establish the feasibility of a unique honeycomb radiator condenser concept for the SNAP 2 system are reported. The basic design consists of a conical honeycomb shell structure with condensing tubes placed within the sandwich and copper radiator attached to the outer surface. An evaluation is made of the braze alloys to join the PH15-7 components of the honeycomb, the PH15-7 tubes, and the copper fins in a single brazing operation. The development of thin-wall tapered rectangular tubes, fabrication and evaluation of test panels, and the fabrication and structural and thermal testing of three full-size radiator condensers are described. It is concluded that although additional fabrication refinements are necessary to upgrade the overall quality, the feasibility of this radiator condenser was established.

AI-AEC-13066

NAA-SR-10511

TITLE INCREASING THERMOCOUPLE RELIABILITY FOR IN-PILE EXPERIMENTS

AUTHOR: E. L. Babbe

4-20-65

ABSTRACT: A series of laboratory thermal cycling tests indicates that the reliability of metal-sheathed, MgO-insulated, chromel-alumel thermocouples can be significantly improved by increasing the packed density of the insulation and using a sheath with a low coefficient of thermal expansion.

NAA-SR-Memo-11841

TITLE: POST-EXAMINATION OF SNAP 10A THERMOELECTRIC PUMPS NUMBERS 16, 31, AND 32

AUTHOR: R. M. Willard

2-9-66

ABSTRACT: The post-test examination of the SNAP 10A pumps, built with and without a diffusion barrier at the copper shoe-iron shoe interface of the thermoelectric element, are reported. Without the diffusion barrier, considerable change in the microstructure of the element near the junction to the iron shoe on the hot leg were evident after 8200 hr of operation with 1100°F inlet NaK on one unit and 9388 hr with 1010°F inlet NaK on another unit. The source of contaminants was concluded to be the braze alloy used at the copper shoe-iron shoe bond on the "N" elements and the copper shoe of the "P" elements (no braze alloy is used at the copper-iron interface). On one unit built with an iron foil diffusion barrier at the copper shoe-iron shoe interface, no indications of microstructure changes were found after 17,000 hr of operation with 1010°F NaK and additional 360 hr with 1100°F NaK.

NAA-SR-11924

TITLE: DEVELOPMENT IRRADIATION TEST OF SNAP 8 ELECTRICAL COMPONENTS

AUTHOR: N. F. Davies

7-15-66

ABSTRACT: In-pile testing of four types of electrical connectors, two types of electrical cables, and five electrical coil samples is described. Test data, including the insulation resistance, measured before, during, and/or after the 3000-hr test are presented. During this test, the components were exposed to irradiation doses of 10^{-7} to 10^{-9} nvt fast neutrons and 8×10^9 to 9×10^{10} r gamma in 900 to 1200°F and 10^{-6} torr environment. These data indicated that the insulation resistances of all components were essentially unaffected by the test. The insulation around the contact pins of one type connector was severely degraded. Visual appearance of all other components was excellent.

NAA-SR-12038

TITLE: S8DR TEMPERATURE SWITCH DEVELOPMENT AND TESTING

AUTHOR: J. L. Powers

5-15-66

ABSTRACT: The selection and the development of a magnetic amplifier device for signalling the preselected temperatures of SNAP 8 reactor coolant being monitored by a platinum resistance temperature detector are described. The results of various short-term evaluation and 12,000-hr endurance tests are presented. The switch exhibited excellent resolution (0.25°F), stability (2.0°F), and response (1.0 sec), with minimal hysteresis (1.5°F).

AI-AEC-13066

NAA-SR-12042

TITLE: IRRADIATION TEST OF SNAP 8 ACTUATOR, POSITION SENSOR, AND LIMIT SWITCH

AUTHORS: N. F. Davies, W. J. Kurzeka, and M. Warren 11-5-66

ABSTRACT: In reactor test in which two each of the SNAP 8 control drum actuator, position sensor, and limit switch were exposed to nuclear irradiation while being operated periodically in an 800 to 1000°F and 10⁻⁵ torr environment is described. Test data measured periodically during the 3,000-hr test wherein the total irradiation dose ranged from 1.0 to 8.9 x 10¹⁹ nvt fast neutrons and 0.6 to 1.5 x 10¹⁰ r gamma are presented in series of graphs. It is reported that all component performance remained stable or within design requirements. The reduction of insulation resistance, attributed to oil from the vacuum system, was corrected by controlled high-temperature oxidation.

NAA-SR-Memo-12086

TITLE: SUMMARY OF FAILURE OF HIGH-TEMPERATURE COILS DUE TO CONDUCTOR RUPTURE

AUTHOR: W. G. Long 7-22-66

ABSTRACT: The failures of coils in the SNAP 10A, SNAP 2, and S8DS actuators and S8DS position sensors are described and evaluated. It is concluded that all failures are attributable to the corrosion fatigue rupture of the magnet wire conductor. The materials used for the conductor and its insulation, the solvent of the encapsulating cement, and the processes utilized for the fabrication of the coils are conducive to corrosion fatigue. Because the basic fabrication processes cannot be changed, substitution of more corrosion-resistant, stainless-steel-clad, copper conductor and the less acidic, newly developed AI-1 encapsulating cement are recommended to minimize the corrosion fatigue factors.

NAA-SR-Memo-12125

TITLE: DEVELOPMENT OF AI-1 CEMENT ENCAPSULANT

AUTHOR: W. G. Long 8-22-66

ABSTRACT: The development of an improved inorganic cement, AI-1, for wet-winding and encapsulating high-temperature electrical coils is described. Unlike the previously used proprietary cements, this phosphate-bonded, silica-base cement can be formulated from basic materials and is also less corrosive to copper conductors in its wet state. Test data are presented to show that the insulation resistance values at room and elevated temperatures are comparable to the previously utilized cements and well above the minimum required values. Limited life test data are also shown.

NAA-SR-12605

TITLE: S8DR ACTUATOR INTERIM REPORT

AUTHOR: W. G. Long 12-30-67

ABSTRACT: The design of the S8DR actuator and methods and preliminary results of developmental and verification testing are discussed. Development tests with the S8DR scram kits at actuator temperatures of 900 to 1000°F had progressed through 10,000 to 13,000 hr. Other tests, including irradiation effects, are being conducted at actuator design temperature of 1025°F and had progressed through 6,000 hr. Minor problems encountered during fabrication and early testing and their assessment and design or process corrections are described. Two companion reports are AI-AEC-12779 and AI-AEC-12925.

AI-AEC-13066

AI-AEC-12779

TITLE: S8DR ACTUATOR SECOND INTERIM REPORT

AUTHOR: L. E. Donelan

3-69

ABSTRACT: This second report covers the successful completion of the S8DR design verification testing at the design temperature of 1025°F, the failures and problems encountered during the design margin testing at 1125°F, evaluation of these problems, and the design modification of the actuator. These modifications included the selection of stainless-steel-clad copper magnet wire (in lieu of the chrome-plated copper wire), the use of the newer, less corrosive (prior to drying and curing) wet winding encapsulant, rearrangement of the coil leads to eliminate the coil failures, electrical connection of the two coils in each phase in parallel and the direction termination of cable assembly conductors to the coil leads to improve reliability, and selection of material with more stable constituents for the chip shield to minimize outgassing at higher temperatures. The successful results of a series of thermal cycle tests on the improved design coils to 1025, 1125, and 1225°F are also described. The original and final reports of this series are NAA-SR-12065 and AI-AEC-12925, respectively.

AI-AEC-12925

TITLE: S8DR ACTUATOR FINAL REPORT

AUTHOR: L. E. Donelan

3-15-70

ABSTRACT: This final report describes the method and results of design verification tests conducted at 1125°F on three of the four modified S8DR actuators (design used in the S8DS experiment). The tests on these units, which had been subjected to the original tests prior to the rebuild, included 200 thermal cycles to 1125°F during the approximately 14,000-hr test period. The mechanical interference between the rotor and stator pole face which occurred on one unit was attributed to mechanical misalignment due to the disassembly and reassembly. The results of extended tests on 6 stator coils and 1 brake coil at 1125°F for nearly 14,000 hr including 541 thermal cycles and maximum current capacity tests on 3 stator and 1 brake coils are also presented. The recommended limit of the coils is 1225°F with current density limits of 5350 amp/in.² for the stator and 3450 amp/in.² for the brake coils, respectively. The other two reports of this series are NAA-SR-12605 and AI-AEC-12779.

AI-AEC-12964

TITLE: SNAP 8 CONTROL DRUM BEARING DEVELOPMENT AND EVALUATION HISTORY

AUTHOR: P. H. Horton

8-15-70

ABSTRACT: The design descriptions and the developmental and verification test results on the control drum bearing for the SNAP 8 reactors are reported. The initial successful short-time tests of the S8DRM bearing were followed by failure of the alumina coatings on the S8DR bearing shafts during long-time testing. From the failure analysis and evaluation testing, it was concluded that coating failure resulted from phase change of the alumina, which was catalyzed by the presence of the LiF impregnant in the carbon graphite bushings of the bearing ball. The developmental studies to select an alternate coating for the alumina are described. The selection basis of the S8DR bearing design with a solid unimpregnated carbon graphite ball and the results of the successful 12,000-hr design verification tests are presented.

AI-AEC-13066

AI-AEC-13073

TITLE: SCRAM CONTROLLER AND AUTOMATIC CONTROLLER, 5-kwe REACTOR THERMOELECTRIC SYSTEM

AUTHOR: R. J. Tuttle

5-25-73

ABSTRACT: The design objectives and limitations developed for the scram and automatic controllers are discussed in detail and, to a lesser extent, for the manual and computer control systems and the backup shutdown mechanism. Digital computer simulations used to analyze startup methods are described and the results presented for several candidate methods. The selected startup method uses three reactivity insertion rates to provide a startup satisfying the design objectives and limitations. Improvements in this startup method are recommended. Functional requirements of the automatic and scram controllers are discussed and means of implementing these functions by use of integrated circuits are proposed.

AI-AEC-13080

TITLE: ZIRCONIUM HYDRIDE REACTOR CONTROL DRIVE ACTUATOR DEVELOPMENT SUMMARY REPORT

AUTHOR: L. E. Donelan

6-73

ABSTRACT: Over the years, actuators for compact space nuclear reactor reflector control have progressed from modified commercial units with relatively short life and low reliability at moderate temperature to units custom-designed for specific application. These latter units have demonstrated long life and high reliability at high temperatures. The design for a specific application now includes the ability to design for various torque and step sizes and the ability to control the relative proportions (length and diameter) of the actuator. The present status of design calculation capability incorporates reliable computer programs to facilitate the design and selection of the actuator, in the form of: (1) a stepper motor program for calculating actuator performance based on physical configuration, (2) an actuator brake program for calculating the magnetic performance based on a physical brake configuration, and (3) a bearing program for calculating loads and clearances for initial end bell and bearing interferences and journal dimensions. Using these programs and various other supporting calculations, a control drive actuator has been designed for the 5-kwe Thermoelectric System reactor to fit in the available space within the reflector and shield configuration. This actuator will develop sufficient torque for both the specified stepping drive control and the scram requirement, at current densities no greater than the proven S8DR design.

The report contains a brief history of the progress of actuator design, development, and tests to date, as well as the design calculations applicable to the actuator designed for the 5-kwe Thermoelectric System.

AI-AEC-13089

TITLE: SPACE NUCLEAR SYSTEM SHEATHED ELECTRICAL CABLES SUMMARY REPORT

AUTHOR: L. E. Donelan

6-73

ABSTRACT: The development of a metallic-sheathed electrical cable is described for use with the ZrH reactor power system. The positive protection to the conductor and the insulation, provided by the metallic sheath, could be expected to increase the cable performance reliability from that of the previously utilized braided-wire-covered cables. Material and design studies were conducted to select the reference cable configuration. The reference design selected consisted of a single stainless steel clad 0.040-in.-diameter conductor, encased within a 0.090-in.-diameter stainless steel sheath, with compacted MgO insulation between the conductor and the sheath. Methods of terminating the cable to the actuator and the limit switch were also selected. Preliminary tests showed that this type of cable will maintain the insulation resistance well above the minimum acceptable level under long-time exposure to the fabrication, system assembly, and operational environments and conditions.

AI-AEC-13066

AI-AEC-13092

TITLE: SPACE NUCLEAR SYSTEM EXPANSION JOINT SUMMARY REPORT

AUTHORS: W. D. Whitaker and T. T. Shimazaki

5-73

ABSTRACT: The engineering, design, and fabrication status of the expansion joint unit (EJU) to be employed in the NaK primary coolant piping loop of the 5-kwe Reactor Thermo-electric System are described. Four EJU's are needed in the NaK primary coolant piping loop. The four EJU's, which will be identical, utilize bellows as the flexing member, are hermetically sealed, and provide double containment. The bellows are of a nested-formed design, and are to be constructed of 1-ply thickness of 0.010-in. Inconel 718. The EJU's provide a minimum piping load margin of safety of +0.22.

TI-652-220-007

TITLE: 5-kwe REACTOR TE REFLECTOR SECTOR DRIVE ACTUATOR DESIGN SUMMARY

AUTHOR: L. E. Donelan

10-4-72

ABSTRACT: The results of the 5-kwe Reactor TE System actuator parametric study, and subsequent load and performance calculations, are presented, together with the results of the stress analysis.

TI-653-220-004

TITLE: ACTUATOR FOR THE SPF REACTOR CONTROL SYSTEM

AUTHOR: L. E. Donelan

2-2-72

ABSTRACT: This report documents the wiring and coil decisions made on the SPF actuator, based upon the existing system. It also describes a system of electrical separation of scram and control functions utilizing the same actuator. By attachment, the dual (redundant) brake coil concept for the SPF reactor is included, as well as a discussion of actuator coil connections.

TR-653-240-005

TITLE: RESULTS OF ADDITIONAL STUDIES OF INSTRUMENTATION AND FLOW MIXING IN THE ZrH 19-ELEMENT TEST BUNDLE

AUTHOR: B. J. Ostermier

11-24-71

ABSTRACT: Results are reported for a series of tests recording injected salt concentrations in the instrumented ZrH 19-element test bundle. These results are presented as plots vs axial location of the fraction of calculated injected salt tracer detected by probes along the bundle flow channels. Data were recorded for different probe-to-flow-channel orientations and are interpreted both to evaluate the ability of probes to detect average channel salt concentration and to identify flow characteristics in the test bundle. A portion of the results presented are for improved data recorded after instrumentation was modified to eliminate signal interactions in the electrode leads.

TI-696-220-012

TITLE: DEVELOPMENT OF AN IMPROVED ACTUATOR FOR ZrH REACTOR SYSTEMS; A HISTORICAL REFERENCE

AUTHOR: L. E. Donelan

7-7-71

ABSTRACT: The history of the actuator for the ZrH Reactor system is presented. The present design status is summarized.

AI-AEC-13066

2. CRU Control

NAA-SR-6310

TITLE: SNAP 2 POWER CONVERSION SYSTEM ROTATIONAL SPEED CONTROL:
TOPICAL REPORT No. 18

AUTHORS: W. E. Dauterman and E. J. Viton 12-1-63

ABSTRACT: The parasitic load speed control for the SNAP 2 power conversion system consists of a frequency-discriminator low-level magnetic amplifier, magnetic amplifier power stages, and a resistive load. The design and development of this parasitic load speed control is described. Experimental results are given to confirm the breadboard design approach.

NAA-SR-6319

TITLE: SNAP 2 POWER CONVERSION SYSTEM MODEL 6 TWO-PHASE FLIGHT-PACKAGED
SPEED CONTROL, TOPICAL REPORT 27

AUTHOR: W. E. Dauterman 6-64

ABSTRACT: This report summarizes the packaging effort to achieve the Model 6 flight-speed control. The heat-transfer analysis conducted prior to design of the Model 6 speed control is included.

AI-AEC-12850

TITLE: SNAP 8 AUTOMATIC STARTUP AND PCS MERCURY INJECTION SIMULATION TESTS

AUTHOR: R. F. Audette 10-31-69

ABSTRACT: SNAP 8 reactor dynamic tests have demonstrated the capability for fully automatic startup over eight decades of power from shutdown, and for automatically furnishing the heat load demand imposed by automatic startup of the flight system power conversion loop following reactor startup. The tests performed approximate predicted flight system operation.

AI-AEC-13066

K. SAFETY

1. General Safety

NAA-SR-3465

TITLE SNAP 2 EXPERIMENTAL REACTOR HAZARDS REPORT

AUTHOR R R Eggleston and G. L. Schmidt (Editors) 4-1-59

ABSTRACT A description of the SNAP Experimental Reactor (SER), the site, and facility for the operation of the SER is presented. A number of accident studies are made and evaluated

NAA-SR 3513

TITLE SNAP 2, ENVIRONMENTAL TEST FACILITY HAZARDS REPORT

AUTHOR G. H. Anno (Editor) 5-1-59

ABSTRACT This report is a facility hazards study and describes the protective features of the SNAP 2 Environmental Test Facility (SETF). The facility is basically a reactor shield with the associated containment and protective systems. To support the required test program, the installation includes control systems, heat removal systems, remote handling systems, ventilation systems, waste storage facilities, and general laboratory facilities.

NAA-SR-5483

TITLE SUMMARY SAFEGUARD REPORT FOR SNAP 2 DEVELOPMENTAL SYSTEM

AUTHOR H. N. Rosenberg (Editor) 11-23-60

ABSTRACT This report presents a study of the hazards associated with the operation of the SNAP 2 Developmental System (S-2-DS). A comprehensive examination is made of the potential hazards and the provisions for protection against, or elimination of, these hazards are outlined. This study shows that even under the most adverse circumstances of faulty operation and design, the S-2-DS is safe and represents no significant danger either to operation personnel or to the surrounding population.

NAA-SR-6181

TITLE THE SNAP 8 DEVELOPMENT SYSTEM (S8DS) TEST FACILITY PRELIMINARY SAFEGUARDS STUDY

AUTHOR A. Piccot (Editor) 9-1-61

ABSTRACT This report is a preliminary facility hazards summary for the SNAP 8 ground test facility. The facility will house the SNAP 8 reactor operating at 600 kw.

NAA-SR-6207

TITLE SNAP 8 REACTOR PRELIMINARY DESIGN SUMMARY

AUTHOR H. Rosenberg (Editor) 6-15-61

ABSTRACT The purpose of this report is to describe the present concept of the SNAP 8 reactor, reactor control system, safety systems, and shield and to outline the reasoning behind the design point selection.

AI-AEC-13066

AI-AEC-13074

TITLE: OPERATIONAL HAZARDS EVALUATION FOR 5-kwe REACTOR THERMOELECTRIC SYSTEM GROUND TEST

AUTHOR: R. J. Tuttle 6-73

ABSTRACT: Selected reactor accidents, analyzed by means of a digital computer simulation program, and the requirements for scram protection are discussed. Prevention of undesirable interaction between the control and safety systems is described. Operational limitations on segment positions are presented that afford adequate shutdown protection throughout the reactor lifetime. Potential operator errors and equipment malfunctions and failures are discussed.

AI-69-Memo-19

TITLE: SNAP REACTOR SAFETY IN THE HARDENED MISSILE BASE APPLICATION

AUTHOR: Staff 3-19-69

ABSTRACT: Key elements of the SNAP reactor safety program are reviewed in the context of an Air Force ground application for Hardened Missile Base (HMB) power. Bases for safety evaluations are discussed, with emphasis on the unique properties of zirconium-uranium hydride fuel as measured in actual accident simulations. Important safety factors affecting delivery, installation, operation, and post-operation disposal of the HMB powerplant are summarized. The HMB application of SNAP reactors can be made safely and within the scope of existing pertinent regulations, and in conformance with policies and criteria already reviewed, established, and approved by both USAF and AEC.

AI-70-58

TITLE: THE SAFE USE OF NaK LIQUID METAL COOLANT FOR A MARINE UNDERSEA REACTOR POWERPLANT (PRELIMINARY EVALUATION)

AUTHOR: R. L. Wallerstedt 7-30-70

ABSTRACT: The potential for NaK-water reaction in an undersea powerplant, the magnitude of the reaction, and the effects and consequences of such a reaction were studied to establish general design criteria for a NaK-cooled reactor marine power system. The NaK coolant required by a SNAP-type compact reactor can be safely accommodated in an undersea environment. The potential energy availability from the NaK-water reaction is of equal or less magnitude than certain other explosive or combustible products (TNT, hydrogen, etc.) handled undersea by the Navy. Nuclear safety factors have not been included in this study.

TI-696-160-001

TITLE: WATER FLOODING CRITICALITY STUDY FOR ZrH FLIGHT REACTOR

AUTHOR: R. V. Anderson 10-29-70

ABSTRACT: Five analytical criticality calculations were performed to study the effects of: (1) water reflection only (no core flooding), (2) water reflection with 10% core flooding, (3) water reflection with 35% flooding, (4) water reflection plus complete core flooding, and (5) the negative reactivity feedback associated with rapid core expansion induced by a destructive transient.

AI-AEC-13066

2-140

2. Aerospace Safety

NAA-SR-4714

TITLE SNAP 2 SAFETY EVALUATION ANALYSIS

AUTHOR F D. Anderson

1-1-60

ABSTRACT Preliminary SNAP 2 safety evaluation analysis was performed to determine the severity of possible hazards that might be encountered during the SNAP 2 flight program and to investigate the possible solutions to these problems. Four major periods of possible fission product release from the SNAP 2 reactor were analyzed (1) shipment and integration, (2) launch pad operations, (3) launch operations, and (4) reentry.

NAA-SR-8303

TITLE AEROSPACE SAFETY REENTRY ANALYTICAL AND EXPERIMENTAL PROGRAM, SNAP 2 and 10A INTERIM REPORT

AUTHOR R. D. Elliott

9-30-63

ABSTRACT This report describes the analytical and experimental effort pursued by the Aerospace Safety Reentry project to investigate potential hazards to the population during the reentry from orbit of SNAP reactors.

NAA-SR-9479

TITLE AEROSPACE SAFETY PROGRAM, SNAP 10A REENTRY FLIGHT DEMONSTRATION (RFD-1), ANALYSIS OF FLIGHT TEST DATA

AUTHOR R. D. Elliott

7-15-64

ABSTRACT A full-scale nonradioactive model of the SNAP-10A reactor was launched into a sub-orbital trajectory to obtain data on reentry heating that could be used to predict the mode of disintegration of the actual reactor returning from a satellite orbit. The results of the analysis of the data obtained from this flight test are the subject of this report

NAA-SR-11353

TITLE AEROSPACE SAFETY PROGRAM - SAFE DISPOSAL OF SNAP REACTORS

AUTHORS R L. Detterman, A. Weitzberg, and C. A. Willis

7-25-65

ABSTRACT Methods for safely disposing of SNAP reactors after space operation are evaluated. Three basic methods have evolved for implementing the disposal principles, the basic methods are (1) boost to a long-lived orbit, (2) destruct, and (3) controlled reentry. Combinations of these methods for the short orbit case are considered and several systems are found to be promising.

NAA-SR-11502

TITLE: THERMAL BEHAVIOR OF SNAP REACTOR FUEL ELEMENTS DURING
ATMOSPHERIC REENTRY

AUTHORS: A. W. Barsell, L. D. Montgomery, and J. E. Arnold

3-25-66

ABSTRACT: An analytical model is developed to describe the thermochemical behavior of SNAP reactor fuel elements during atmospheric reentry. The aerodynamic heating, transpiration cooling, and the chemical reactions of the fuel with air are investigated. A detailed evaluation of zirconium-uranium hydride fuel material properties is presented. Results using the analytical model are presented and compared with experimental data.

NAA-SR-11704

TITLE: METHODS FOR PREDICTING AERODYNAMIC HEATING RATES TO HIGH-VELOCITY
BODIES IN THE UPPER ATMOSPHERE

AUTHOR: R. D. Elliott

1-15-66

ABSTRACT: The basic physical phenomena associated with aerodynamic heating are described. The methods currently used at Atomics International for predicting heating rates to bodies moving at high velocities through the earth's atmosphere are summarized. The methods described provide the principal input data for the analytical studies of SNAP reentry disintegration in the Aerospace Nuclear Safety Program.

NAA-SR-Memo-11833

TITLE: BIBLIOGRAPHY - SNAP AEROSPACE NUCLEAR SAFETY PROGRAM REPORTS

AUTHOR: R. L. Detterman

4-1-66

ABSTRACT: The formal reports published by Atomics International relating to the work performed under the Aerospace Safety Program are compiled. Abstracts are included for the convenience of the reader.

NAA-SR-Memo-11876

TITLE: DEVELOPMENT PLAN - SNAP REACTOR EXPLOSIVE DESTRUCT TECHNIQUES

AUTHOR: R. H. Prowett

5-11-66

ABSTRACT: A development plan, based on a systematic series of parametric studies, is discussed for an explosive method of destructing and dispersing the SNAP-type reactor at the end of useful life. The overall concept is an explosive charge, stored below the neutron shield during useful life, which is delivered toward the reactor and detonated at the proper moment to rip apart and disperse the reactor, particularly the fuel, into small particles. The general plan, with estimated cost and schedule, is proposed for the selection and evaluation of explosives, determining the optimum detonation for reactor destruct, development of the transport system and controls, and a full-scale demonstration.

AI-AEC-13066

2-142

NAA-SR-12082

TITLE: EXPERIMENTAL INVESTIGATION OF THE THERMOCHEMICAL BEHAVIOR
OF SNAP FUEL MATERIAL IN A HYPERTHERMAL WIND TUNNEL

AUTHORS: J. E. Arnold and A. W. Barsell 6-20-67

ABSTRACT: Experimental support for SNAP fuel element reentry burnup analytical programs and associated fuel particle disintegration studies were 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. It was found that good temperature correlation was possible for those test conditions in which nonequilibrium flow effects were small.

NAA-SR-12355

TITLE: THERMAL BEHAVIOR OF SNAP 8 REACTOR DURING ATMOSPHERIC REENTRY

AUTHOR: L. D. Montgomery, et al. 5-1-67

ABSTRACT: An analysis was conducted to simulate the thermal behavior of a SNAP 8 type reactor system during reentry into the earth's atmosphere. From the results of this analysis, it is concluded that the SNAP 8 reactor vessel will not ablate to a sufficient extent, for probable modes of reentry, to release the fuel elements at altitudes necessary to obtain complete fuel ablation.

NAA-SR-12382

TITLE: SNAP REACTOR ABLATION - DISINTEGRATION EXPERIMENT (RADE) IN A
HYPERTHERMAL WIND TUNNEL

AUTHORS: J. E. Arnold and L. D. Montgomery 6-30-67

ABSTRACT: Experimental support for current SNAP reactor reentry burnup analytical programs was provided by a series of ablation tests on half-scale reactor models exposed to a partially simulated reentry environment. Measurements of the aerodynamic heat flux to the complicated geometry of SNAP reactors were obtained with gold-plated calorimeter models. Data were also obtained on the transient temperature response of the reactor models and on the mode of disintegration of the forward components of the reactor, thus satisfying these primary test objectives. Application of the measured heat flux data to the analytical model resulted in good correlation of the analytical calculations with the experimental temperature response of the models.

NAA-SR-12383

TITLE: SNAP 8 CONFIGURATION REENTRY HYPERSONIC HEATING RATES DERIVED
FROM A SHOCK TUNNEL TEST

AUTHOR: D. K. Nelson 5-1-67

ABSTRACT: Reentry aerodynamic heating distributions for the SNAP 8 configuration have been obtained based on hypersonic shock tunnel experiments. The heat-transfer correlations are given as a function of angle of attack from 0 to 70 degrees. The perturbative effect of piping on the flow field, and consequently the heat-transfer distributions, was also investigated at the 30- and 60-degree angle-of-attack positions.

AI-AEC-13066

NAA-SR-12433

TITLE: EVALUATION OF SNAP REACTOR DISPOSAL TECHNIQUES

AUTHORS: D. K. Nelson and R. L. Detterman 6-30-67

ABSTRACT: Active and passive reactor disposal modes have been evaluated to establish their relative merits and conditions of best application. The active disposal modes included deorbit for ocean disposal, intact earth impact and recovery, orbit enhancement, and earth escape. Random reentry was also considered as a disposal mode in the event of active disposal failure and as a primary passive mode.

NAA-SR-Memo-12614

TITLE: SNAP ENGINEERING TEST FACILITY SAFETY ANALYSIS REPORT ADDENDUM FOR INTRINSIC SUBCRITICALITY EXPERIMENTS

AUTHORS: K. G. Golliher, V. A. Swanson, A. W. Thiele, and B. J. Verna 11-30-67

ABSTRACT: The use of the SNAPTRAN reactor in the SNAP Engineering and Test Facility (SETF) for a series of experiments in support of the Intrinsic Subcriticality Project of the Aerospace Safety Program is described. A summary description of the reactor and facility, an experimental program description, the proposed operating limits, and the safety analysis are included. It is concluded that the proposed operation of the SNAPTRAN reactor in the SETF is within the scope of existing safeguards approvals for the facility.

NAA-SR-Memo-12614, Supplement 1

TITLE: SNAP ENGINEERING TEST FACILITY SAFETY ANALYSIS REPORT ADDENDUM FOR INTRINSIC SUBCRITICALITY EXPERIMENTS, SUPPLEMENT 1

AUTHOR: W. Heneveld 2-8-71

ABSTRACT: The material in this report is a description of a new system for the instrumentation and control of the SNAPTRAN reactor. It is intended to replace Appendix B of NAA-SR-Memo-12614. The purpose of this newer equipment is to provide greater reliability.

AI-AEC-12738

TITLE: SNAP REACTOR AEROSPACE NUCLEAR SAFETY PROGRAM ANNUAL REPORT, GFY 1968

AUTHOR: R. L. Detterman 10-15-68

ABSTRACT: Techniques were developed for calculating the effects of Gd-155 addition to SNAP reactors. An operating manual for the intrinsic subcriticality experiments using the SNAPTRAN machine was completed. Test procedures for the experimental program were completed and approved, and checkout was begun.

AI-AEC-Memo-12784

TITLE: SNAP PROGRAM REVIEW, WASHINGTON, D.C., JANUARY 30, 1969

AUTHOR: Staff -69

ABSTRACT: This report is a briefing book for the presentation of the SNAP 8 Developmental Reactor operating results, fuel development test results, and intrinsic water subcriticality test results.

AI-AEC-13066

AI-AEC-12909

TITLE: REACTIVITY EFFECTS OF GADOLINIUM-155 ON A SNAP REACTOR

AUTHOR: B. J. Dray, et al. 4-15-70

ABSTRACT: The effects of gadolinium-155 on both the reactivity and the temperature coefficient of reactivity and the temperature coefficient of reactivity of a SNAP reactor were measured. This is part of a continuing program to assure SNAP reactor subcriticality on water immersion.

AI-AEC-Memo-12917, Volume I

TITLE: PRELIMINARY REPORT, POSTOPERATIONAL SAFETY OF REACTOR POWER SYSTEM FOR NASA SPACE STATION, VOLUME I, INTRODUCTION AND SUMMARY

AUTHORS: J. D. Gylfe, W. R. Lahs, and R. B. Paulson 6-30-70

ABSTRACT: This report summarizes the results of a preliminary study of the postoperational safety aspects of nuclear reactor power systems for NASA's Space Station Mission. The study showed the importance of the cyclic solar activity on the normal orbit-decay time before reentry. Another important conclusion was that intact reentry survival of the reactor would be achieved by proper reactor shield design and/or disposal technique.

AI-AEC-Memo-12917, Volume II

TITLE: PRELIMINARY REPORT, POSTOPERATIONAL SAFETY OF REACTOR POWER SYSTEM FOR NASA SPACE STATION, VOLUME II, MISSION AND SYSTEM DESCRIPTION

AUTHORS: J. D. Gylfe and R. A. Johnson 7-1-69

ABSTRACT: The design, integration, and operation of a nuclear reactor-thermoelectric power system for earth-orbiting manned space stations are presented. This was a joint study where AI performed the design and analysis of the nuclear power system and NASA performed the space station design, integration, and mission analysis. This report emphasizes only the nuclear power system design and operation.

AI-AEC-Memo-12917, Volume III

TITLE: PRELIMINARY REPORT, POSTOPERATIONAL SAFETY OF REACTOR POWER SYSTEM FOR NASA SPACE STATION, VOLUME III, SAFETY ANALYSIS

AUTHORS: W. R. Lahs, R. B. Paulson, and W. B. Sayer 6-30-70

ABSTRACT: The objective of the analysis in this report is to provide a quantitative measure of the risk to man from the disposal of a space nuclear power system after its useful lifetime. The study covers mission safety only with respect to the general public after orbital operation. Safety of shipping, launch, startup, and orbital operation are not included. The following three potential disposal modes were evaluated: random reentry from the operating orbit, orbital boost to a long-life orbit, and orbital deboost into the deep ocean.

AI-AEC-13066

AI-AEC-Memo-12917, Volume IV

TITLE: PRELIMINARY REPORT, POSTOPERATIONAL SAFETY OF REACTOR POWER SYSTEM FOR NASA SPACE STATION, VOLUME IV, SUPPORTING ANALYSIS

AUTHORS: R. F. Audette, et al. 6-30-70

ABSTRACT: This report presents the details of specific analyses that were presented in Volume III. It includes the following: effects and results due to normal and abnormal reactor histories with regard to fission product inventories, orbital decay analysis, the behaviour of the power system as it reenters the earth's atmosphere, models used to assess the extent and significance of release of radioactivity to the environment during and following reentry, and results of a shutdown reliability analysis.

AI-AEC-Memo-12917, Volume V

TITLE: PRELIMINARY REPORT, POSTOPERATIONAL SAFETY OF REACTOR POWER SYSTEM FOR NASA SPACE STATION, VOLUME V, NUCLEAR POWER SYSTEM DISPOSAL METHODS

AUTHOR: F. G. Etheridge (Editor) 6-30-70

ABSTRACT: A conceptual design study was conducted to define a system with a 5-year orbital storage life capable of disposing of a nuclear power system by either deorbiting into preselected ocean areas or by insertion into an orbit with a lifetime of 100 years. This system uses four solid propellant motors.

AI-AEC-13100

TITLE: AEROSPACE SAFETY PROGRAM SUMMARY REPORT

AUTHORS: J. M. Otter, K. E. Buttrey, and R. P. Johnson 6-73

ABSTRACT: Projects in support of the Aerospace Nuclear Safety Programs are summarized. The objective of the program was to evaluate the radiological safety of SNAP systems, and to develop methods and designs to assure that safety. Projects summarized are experiments and analyses for determining: (1) risks, by various exposure modes, from different methods of reactor disposal, (2) the disintegration of a reactor re-entering the atmosphere, (3) the burnup of fuel elements reentering the atmosphere, including ablation and dispersal of particles, (4) the nuclear behavior of reactors on impact with the earth and in handling accidents, (5) the behavior of reactors in transient power operation, (6) reactor criticality conditions, and the assurance of sub-criticality in water, (7) the assurance of shutdown at the end of the power production lifetime, and (8) thermophysical properties of SNAP fuel materials.

SA-696-13-005

TITLE: RADIOACTIVITY FROM NEUTRON ACTIVATION OF NUCLEAR POWER SYSTEM COMPONENTS IN NASA SPACE STATIONS

AUTHOR: A. R. Piccot 1-5-70

ABSTRACT: Neutron activation products produced in a space station nuclear power system are determined. The inventories accumulated in the system components and the resulting radiation source strengths are presented for a variety of operating and accident conditions. The important isotopes with respect to direct radiation, inhalation, or ingestion hazards are identified.

SA-696-13-006

TITLE: EXPOSURE MODEL FOLLOWING RESERVOIR IMPACT OF THE NASA SPACE STATION REACTOR POWER SOURCE

AUTHOR: W. R. Lahs

1-13-70

ABSTRACT: This safety analysis documents the reservoir contamination model which will be incorporated into the NASA Space Station Safety Evaluation Report. The long-term integrated bone dose resulting from an instantaneous release of Sr⁹⁰ to the reservoir is concluded to be the controlling hazard.

TI-700-13-001

TITLE: DESCRIPTION OF A LAUNCH PAD FIRE ENVIRONMENT

AUTHOR: R. B. Paulson

6-30-67

ABSTRACT: A description of the characteristics of a launch pad fire environment is presented which can be used to determine the survival capability of isotope fuel capsules when they are subjected to the temperatures, pressures, and chemical environment of a launch fire.

3. SNAPSHOT Safety

NAA-SR-9781

TITLE: SNAP AEROSPACE SAFETY PROGRAM PHASE I WIND TUNNEL TESTS

AUTHOR: L. E. Shpiner

2-1-65

ABSTRACT: The operation, parameter selection, and thermopaint technique are described for a series of tests conducted in a hotshot wind tunnel. These tests were performed to determine the distribution of aerodynamic heating intensities on the surfaces of the SNAP 10A reactor during orbital reentry. This project was undertaken in support of the Aerospace Safety Program.

NAA-SR-11361

TITLE: WATER IMMERSION SAFETY FOR SNAP REACTORS

AUTHOR: J. P. Hawley

9-30-67

ABSTRACT: Potential radiological hazards associated with the construction, launch, operation, and disposal of SNAP reactors are summarized. Accidental preorbital criticality and end-of-life disposal are the two most difficult safety problems which were identified. The prevention of accidental criticality is the subject of this report.

NAA-SR-11584

TITLE: ANALYSIS OF SNAPTRAN-3 DESTRUCTIVE EXPERIMENT DEBRIS

AUTHOR: G. E. Culley

12-15-65

ABSTRACT: A metallographic survey of the SNAPTRAN-3 Destructive Experiment debris is presented. A description is also provided of the reactor fuel elements and core components following disintegration during the experiment.

AI-AEC-13066

NAA-SR-12073

TITLE: SNAP 10A SUMMARY REPORT

AUTHOR: D. W. Staub

3-25-67

ABSTRACT: The SNAP 10A/SNAPSHOT Program objectives and system requirements are presented. A technical description of the system and its major components is given together with a technical experience summary derived from the developmental engineering and test program. A summary of the safety evaluation and a description of the flight test are also included.

AI-AEC-13066

2-148

L. APPLICATIONS

1. Manned-Space

NAA-SR-9252

TITLE MULTIPLE-UNIT SNAP 2 POWER SYSTEMS FOR MANNED APPLICATION

AUTHOR H. N. Rosenberg, (Editor) 1-28-64

ABSTRACT A study has been made of the feasibility of utilizing multiple-unit SNAP 2 systems to provide power in the 10- to 40-kwe range for manned space stations. Principal areas described are reliability, packaging, weight, and operational considerations.

NAA-SR-9715

TITLE APPLICATION OF SNAP REACTOR POWER SYSTEMS TO MANNED SPACE STATIONS

AUTHOR J. McDonald 6-15-64

ABSTRACT This report describes the zirconium-hydride-uranium thermal reactor coupled through a NaK loop to a mercury-rankine power-conversion system to be used as the power source for a manned space station. Preliminary designs of this system are given for 5-, 10-, 15-, and 20-kwe power levels.

NAA-SR-10318

TITLE APPLICATION OF NUCLEAR POWER PLANTS (SNAP UNITS) TO THE MANNED ORBITING RESEARCH LABORATORY (MORL)

AUTHOR W. V. Botts, et al. 2-1-65

ABSTRACT This report describes in detail two designs of a nominal 6-kwe Nuclear Power Plant (NPP), one using thermoelectrics for power conversion and the other using the mercury-rankine cycle NPP. The integration of the NPP into the Manned Orbiting Research Laboratory (MORL) is given. In addition, detailed discussions are presented on the NPP operational and integration requirements as defined by the MORL mission, nuclear shielding parametric and design data, nuclear hazards and safety criteria and design, NPP deployment subsystem design and analysis, and the developmental requirements to meet the MORL schedule.

NAA-SR-10859

TITLE SNAP 8 NUCLEAR SYSTEMS FOR UNMANNED AND MANNED APPLICATIONS

AUTHOR D. G. Mason 5-15-65

ABSTRACT An unmanned SNAP 8 reference design and a man-rated SNAP 8 reference concept are described in detail. Curves to estimate shield weights for each design are presented as a function of payload separation distance, payload diameter, neutron and gamma doses, dose rate, and reactor power level.

AI-AEC-Memo-12672

TITLE THERMOELECTRIC SYSTEM AEC REVIEW, MARCH 29, 1968, WASHINGTON, D.C.

AUTHOR Staff 3-27-68

ABSTRACT This report is a briefing book for the presentation of the thermoelectric system and advanced subsystem engineering for manned and unmanned space applications.

AI-AEC-13066

AI-AEC-Memo-12717

TITLE: REFERENCE ZIRCONIUM HYDRIDE REACTOR THERMOELECTRIC SYSTEM

AUTHOR: Staff 6-15-69

ABSTRACT: This report presents the results of a 9-month engineering study of nuclear reactor-thermoelectric (TE) power systems. This general type of space power system, which is similar to the 500-watt (SNAP 10A) reactor-TE system, was flight-tested in space in 1965. Additional objectives set for this study were to:

- 1) Establish a conceptual design of a reference system suitable for adaption to either manned or unmanned space missions with a nominal power level of 25 kwe and a minimum lifetime of 20,000 hr.
- 2) Prepare parametric performance data over a suitable power range above and below 25 kwe to illustrate the trade-offs available between power, weight, area, temperature, etc.

AI-AEC-12839

TITLE: REACTOR-THERMOELECTRIC SYSTEM FOR NASA SPACE STATION

AUTHOR: J. D. Gylfe and R. A. Johnson 7-1-69

ABSTRACT: The results of a study regarding the design, integration, and operation of nuclear reactor-thermoelectric power systems for earth-orbiting manned space stations are presented.

AI-AEC-Memo-12917, Volume I

TITLE: PRELIMINARY REPORT, POSTOPERATIONAL SAFETY OF REACTOR POWER SYSTEM FOR NASA SPACE STATION, VOLUME I, INTRODUCTION AND SUMMARY

AUTHORS: J. D. Gylfe, W. R. Lahs, and R. B. Paulson 6-30-70

ABSTRACT: This report summarizes the results of a preliminary study of the postoperational safety aspects of nuclear reactor power systems for NASA's Space Station Mission. The study showed the importance of the cyclic solar activity on the normal orbit-decay time before reentry. Another important conclusion was that intact reentry survival of the reactor would be achieved by proper reactor shield design and/or disposal technique.

AI-AEC-Memo-12917, Volume II

TITLE: PRELIMINARY REPORT, POSTOPERATIONAL SAFETY OF REACTOR POWER SYSTEM FOR NASA SPACE STATION, VOLUME II, MISSION AND SYSTEM DESCRIPTION

AUTHORS: J. D. Gylfe and R. A. Johnson 7-1-69

ABSTRACT: The design, integration, and operation of a nuclear reactor-thermoelectric power system for earth-orbiting manned space stations are presented. This was a joint study where AI performed the design and analysis of the nuclear power system and NASA performed the space station design, integration, and mission analysis. This report emphasizes only the nuclear power system design and operation.

AI-AEC-13066

AI-7950

TITLE: APPLICATION OF SNAP 2 TO MANNED ORBITAL SPACE STATIONS

AUTHOR: H. N. Rosenberg (Editor) 12-16-62

ABSTRACT: Installation and operation of an 11-kwe version of a SNAP 2 system in a manned space station is studied. Of prime importance is the radiation shielding required for the manned system and the use of design features which minimize shield weight. Two space station configurations are considered. Parametric curves of shield weight are presented to enable the space vehicle designer to estimate system weights for other configurations.

AI-64-Memo-129

TITLE: INTERIM STUDY SUMMARY -- MORL NUCLEAR POWER SYSTEMS

AUTHOR: W. V. Botts 6-18-64

ABSTRACT: A comprehensive study of the use of nuclear power for the six-man MORL mission includes preliminary optimization of both thermoelectric (TE) and mercury rankine (MR) nuclear power packages for direct intergration with the MORL. Both systems are conceptually designed for continuous net power of 15 kwe for 2 years but are also evaluated for 7.5 and 10 kwe. The conclusions reached in the study are that the nuclear power plants offer significant weight and operational advantages over all other competitive power sources and are well within the projected state-of-the-art. Criteria for choosing between the TE or MR system based on system design constraints are presented.

2. Unmanned-Space

NAA-SR-1840

TITLE: A 3-kw NUCLEAR AUXILIARY POWER UNIT FOR THE 117L ADVANCED RECONNAISSANCE SYSTEM

AUTHOR: J. R. Wetch (Editor) 1-31-57

ABSTRACT: This report presents a summary of a 9-month feasibility study for nuclear auxiliary power plant. The reactor consists of a metal hydride, uranium-235 fueled, beryllium-reflected, and sodium-cooled thermal reactor producing 45 kw. The heat is transferred to a Rankine cycle mercury power plant delivering 3 kwe. The whole system weighs about 185 lb.

NAA-SR-9892

TITLE: SNAP 10A POWERED METEOROLOGICAL SATELLITE STUDY

AUTHOR: Staff 6-15-64

ABSTRACT: The applicability of reactor-powered SNAP systems for advanced Operational Meteorological Satellites (OMS) is summarized. An optimum configuration for such a satellite is given.

NAA-SR-10859

TITLE: SNAP 8 NUCLEAR SYSTEMS FOR UNMANNED AND MANNED APPLICATIONS

AUTHOR: D. G. Mason 5-15-65

ABSTRACT: An unmanned SNAP 8 reference design and a man-rated SNAP 8 reference concept are described in detail. Curves to estimate shield weights for each design are presented as a function of payload separation distance, payload diameter, neutron and gamma doses, dose rate, and reactor power level.

AI-AEC-13066

AI-AEC-Memo-12672

TITLE: THERMOELECTRIC SYSTEM AEC REVIEW, MARCH 29, 1968, WASHINGTON, D. C.
AUTHOR: Staff 3-27-68
ABSTRACT: This report is a briefing book for the presentation of the thermoelectric system and advanced subsystem engineering for manned and unmanned space applications.

AI-AEC-Memo-12717

TITLE: REFERENCE ZIRCONIUM HYDRIDE REACTOR THERMOELECTRIC SYSTEM
AUTHOR: Staff 6-15-69
ABSTRACT: This report presents the results of a 9-month engineering study of nuclear reactor-thermoelectric (TE) power systems. This general type of space power system, which is similar to the 500-watt (SNAP 10A) reactor-TE system, was flight-tested in space in 1965. Additional objectives set for this study were to:

- 1) Establish a conceptual design of a reference system suitable for adaption to either manned or unmanned space missions with a nominal power level of 25 kwe and a minimum lifetime of 20,000 hr.
- 2) Prepare parametric performance data over a suitable power range above and below 25 kwe to illustrate the trade-offs available between power, weight, area, temperature, etc.

3. Underwater

NAA-SR-5358

TITLE: SELECTION OF A NUCLEAR POWER PLANT CONCEPT FOR UNDERWATER APPLICATIONS
AUTHOR: R. L. Mela 9-1-60
ABSTRACT: This report presents the results of a study conducted to determine the most suitable nuclear system to provide a reliable power source for operation in an underwater environment.

NAA-SR-5405

TITLE: PRELIMINARY DESIGN OF A COMPACT NUCLEAR POWER PLANT FOR UNDERWATER APPLICATION
AUTHOR: Department Staff 7-15-60
ABSTRACT: This report presents the results of a study to determine the preliminary design of a nuclear power plant for operation in an underwater environment. The power plant design presented is based upon existing technology and stresses both maximum reliability and reasonable cost.

AI-66-206

TITLE: 10 to 30 kwe NUCLEAR UNDERWATER POWER SYSTEM
AUTHOR: Staff 10-17-66
ABSTRACT: A natural-circulation-cooled reactor to deliver 10 to 30 kwe is designed for independent underwater use and its performance compared with competitive systems such as batteries and fuel cells. The reactor system is compact, light-weight (14,000 lb), safe, easy to handle and ship, and does not require refueling in its 20-yr life at 50% usage factor. The 30-kwe system will cost on the order of \$10⁶ per unit.

AI-AEC-13066

AI-67-TDR-102

TITLE: CHARACTERISTICS OF MARINE AND TERRESTRIAL LONG LEVEL UNATTENDED POWER SOURCES (<10 kw)

AUTHOR: J. E. Stewart

7-5-67

ABSTRACT: The characteristics of electrochemical, fossil fuel, and nuclear power generating systems are evaluated on the ability to continuously supply primary power of 1 to 10,000 watts for periods up to 5 years. With the information on weight, volume, lifetime, reliability, and cost presented one should be able to select the most suitable power supply for a particular unattended application.

AI-70-58

TITLE: THE SAFE USE OF NaK LIQUID METAL COOLANT FOR A MARINE UNDERSEA REACTOR POWERPLANT (PRELIMINARY EVALUATION)

AUTHOR: R. L. Wallerstedt

7-30-70

ABSTRACT: The potential for NaK-water reaction in an undersea powerplant, the magnitude of the reaction, and the effects and consequences of such a reaction were studied to establish general design criteria for a NaK-cooled reactor marine power system. The NaK coolant required by a SNAP-type compact reactor can be safely accommodated in an undersea environment. The potential energy availability from the NaK-water reaction is of equal or less magnitude than certain other explosive or combustible products (TNT, hydrogen, etc.) handled undersea by the Navy. Nuclear safety factors have not been included in this study.

AI-70-88

TITLE: ZIRCONIUM HYDRIDE REACTOR-POWER SYSTEMS FOR UNDERWATER PULSE POWER APPLICATIONS

AUTHOR: Staff

12-70

ABSTRACT: The spectrum of UZrH reactor thermoelectric, organic Rankine, and liquid metal MHD concepts can all become available during the 1975 to 1980 period to provide a continuous power range of 0.5 to 500 kwe with capability of generating pulse power levels of 1 to 10 Mwe for highly reliable, autonomous, 10-year lifetime without refueling. Various methods of energy storage are evaluated and the optimum method is selected for each combination of pulse power and repetition rate. The ZrH reactor power system can be designed to meet any of the expected operational criteria associated with large pulse power undersea applications.

4. Terrestrial

NAA-SR-7329

TITLE: SNAP 8 TERRESTRIAL SYSTEM

AUTHOR: S. R. Nixon et al.

5-7-62

ABSTRACT: This report describes a design and engineering program to produce a light-weight terrestrial power plant utilizing a SNAP 8 nuclear reactor core.

AI-AEC-13066

AI-6915

TITLE: A COMPACT 2-MEGAWATT ELECTRICAL TRANSPORTABLE NUCLEAR POWER GENERATION UNIT

AUTHOR: Staff 2-15-62

ABSTRACT: A conceptual design for a compact transportable nuclear power generation unit is described. A SNAP-4 reactor configuration is set forth that will generate 2 megawatts of electrical power.

AI-67-TDR-102

TITLE: CHARACTERISTICS OF MARINE AND TERRESTRIAL LONG LEVEL UNATTENDED POWER SOURCES (<10 kw)

AUTHOR: J. E. Stewart 7-5-67

ABSTRACT: The characteristics of electrochemical, fossil fuel, and nuclear power generating systems are evaluated on the ability to continuously supply primary power of 1 to 10,000 watts for periods up to 5 years. With the information on weight, volume, lifetime, reliability, and cost presented one should be able to select the most suitable power supply for a particular unattended application.

AI-68-89

TITLE: SYSTEM DESCRIPTION AND JUSTIFICATION - STUDY OF A NUCLEAR POWER SYSTEM IN A HARDENED UNDERGROUND MISSILE FACILITY

AUTHOR: Staff 7-26-68

ABSTRACT: The design of an unattended 12-kwe nuclear power source with a 10-yr life and post-nuclear attack survivability is described. Comparisons with alternate power sources and justification of the design selected are also presented.

AI-68-131, Volumes I through VI

TITLE: HARDENED MISSILE BASE NUCLEAR POWER SYSTEM STUDY

AUTHOR: Staff 12-31-68

ABSTRACT: Volume I Design Concept
Volume II Power Plant and Heat Sink Detailed Support Data
Volume III UZrH Reactor Detailed Support Data
Volume IV Budgetary and Planning Estimates
Volume V Reactor Demonstration Power Plant
Volume VI Summary Evaluation

Detailed design and evaluation of a 12 kwe, 10 year life unattended power system is presented. Both functional and economic comparisons with alternate power sources are considered. Detailed schedules and costs for fabrication and testing of a demonstration plant and follow-on production are also shown.

AI-AEC-13066

TITLE: SNAP REACTOR SAFETY IN THE HARDENED MISSILE BASE APPLICATION

AUTHOR: Staff 3-19-69

ABSTRACT: Key elements of the SNAP reactor safety program are reviewed in the context of an Air Force ground application for Hardened Missile Base (HMB) power. Bases for safety evaluations are discussed, with emphasis on the unique properties of zirconium-uranium hydride fuel as measured in actual accident simulations. Important safety factors affecting delivery, installation, operation, and post-operation disposal of the HMB powerplant are summarized. The HMB application of SNAP reactors can be made safely and within the scope of existing pertinent regulations, and in conformance with policies and criteria already reviewed, established, and approved by both USAF and AEC.

AI-69-85, Volume I, II, and III

TITLE: FOLLOW-ON HARDENED MISSILE BASE NUCLEAR POWER SYSTEM STUDY
FINAL REPORT

AUTHOR: Staff 10-31-69

ABSTRACT: Volume I Design Concept Evaluation
Volume II Appendices
Volume III Power System Summary Description

The reference reactor-converter system for the Hardened Missile Base (HMB) mission consists of a U-ZrH reactor coupled to a Pb-Te thermoelectric converter by a natural circulation NaK-78 coolant loop. A recent change of the requirement from a 400-kwt, 10-year life system to a 200-kwt, 15-year lifetime prompted a reevaluation of the HMB reactor. The results of this reevaluation indicate that the core design represents a conservative application of existing ZrH reactor technology, resulting in a highly reliable powerplant system.

5. Electric Propulsion

NAA-SR-7085

TITLE: ELECTRIC PROPULSION APPLICATIONS FOR SNAP SYSTEMS

AUTHOR: C. J. Morse 4-6-62

ABSTRACT: The application of SNAP systems to electric propulsion has been investigated. A review is given of the basic analysis involved in establishing optimum power levels and payload capabilities for electric spacecraft. Several typical missions are analyzed to determine the usefulness of the SNAP systems which are under development. In general, it is found that SNAP power units in the range of 60 to 80 kw offer significant mission capability when used in conjunction with initial spacecraft weights of about 10,000 lb in a low-level orbit.

6. Communications

NAA-SR-7408

TITLE: APPLICATIONS OF SNAP REACTOR SYSTEMS TO COMMUNICATIONS SATELLITES

AUTHOR: R. E. Wimmer

7-30-62

ABSTRACT: The field of satellite communications is one of intense and widespread interest, since potential military communications uses are numerous and it also appears to be the only field in space technology with reasonably immediate prospects for commercial exploitation. This report deals primarily with the technological considerations of satellite communication systems, and determination of the satellite power supply requirements for such systems.

Special attention has been given to the present SNAP reactor systems, since they represent a range of power output capabilities suitable to a variety of communications missions, and will be available early in the schedule of attractive satellite communication vehicles.

7. Orbiting Space Station

AI-AEC-Memo-12840

TITLE: REACTOR-THERMOELECTRIC SYSTEM DEVELOPMENT PLANS

AUTHOR: J. D. Gylfe and R. A. Johnson

7-1-69

ABSTRACT: During GFY 1968 a joint study was made by NASA's Marshall Space Flight Center (MSFC) and the Atomic Energy Commission (AEC) contractor, Atomics International, to define a 25-kwe reactor-thermoelectric power system for an orbiting space station. This report describes the development plans. The technical results of this study are reported in AI-AEC-12839 and a report being prepared by MSFC, the former primarily describing the power system and the latter the integration considerations. One of the constraints on the study was that the system be ready for flight in 1975.

8. Lunar Base

AI-Memo-9155

TITLE: NUCLEAR POWER SYSTEM CONCEPTS FOR A LUNAR BASE

AUTHOR: Staff

10-20-63

ABSTRACT: Four specific plant concepts for lunar base applications have been designed using 5-kwe, 10-kwe, and 100-kwe turboelectric and 10-kwe thermoelectric power conversion systems. The four plants are merely examples of the range of capabilities of the SNAP system components. Thermal power is supplied by either a SNAP 2, SNAP 8, or upgraded SNAP 8 reactor as required.

AI-AEC-13066

M. FACILITIES

NAA-SR-3465

TITLE: SNAP 2 EXPERIMENTAL REACTOR HAZARDS REPORT

AUTHOR: R. R. Eggleston and G. L. Schmidt (Editors) 4-1-59

ABSTRACT: A description of the SNAP Experimental Reactor (SER), the site, and facility for the operation of the SER is presented. A number of accident studies are made and evaluated.

NAA-SR 3513

TITLE: SNAP 2, ENVIRONMENTAL TEST FACILITY HAZARDS REPORT

AUTHOR: G. H. Anno (Editor) 5-1-59

ABSTRACT: This report is a facility hazards study and describes the protective features of the SNAP 2 Environmental Test Facility (SETF). The facility is basically a reactor shield with the associated containment and protective systems. To support the required test program, the installation includes control systems, heat removal systems, remote handling systems, ventilation systems, waste storage facilities, and general laboratory facilities.

NAA-SR-6181

TITLE: THE SNAP 8 DEVELOPMENT SYSTEM (S8DS) TEST FACILITY PRELIMINARY SAFEGUARDS STUDY

AUTHOR: A. Piccot (Editor) 9 1 61

ABSTRACT: This report is a preliminary facility hazards summary for the SNAP 8 ground test facility. The facility will house the SNAP 8 reactor operating at 600 kw.

NAA-SR-6733

TITLE: PRELIMINARY HAZARDS SUMMARY - QUALIFICATION TEST BUILDING

AUTHOR: Staff 9-20-61

ABSTRACT: The Nuclear Qualification Test Building (Bldg 19 located at the Santa Susana Field Laboratory Test Site) is designed as a facility in which final assembly and adjustment of SNAP 2 and SNAP 10A power systems will be performed, prior to shipment to the launch site. This report presents a preliminary hazards summary of this facility.

AI-AEC-13066

III. SNAP PROGRAM PROGRESS REPORTS

A. SNAP 2 AUXILIARY POWER

Nuclear Auxiliary Power Unit, Quarterly Progress Report

January-March 1957	NAA-SR-1965	9-1-57
April-June 1957	NAA-SR-2126	10-1-57

Annual Technical Report, SNAP 2 Program

January-December 1957	NAA-SR-2402	5-1-58
-----------------------	-------------	--------

Nuclear Auxiliary Power Unit Development Program, SNAP 2 Bi-Monthly Progress Report

October-December 1957	NAA-SR-2385	1-1-58
December 1957 - January 1958	NAA-SR-2555	3-1-58
February-March 1958	NAA-SR-2715	3-15-58
April-May 1958	NAA-SR-2863	7-1-58
June-July 1958	NAA-SR-3048	10-1-58
August-September 1958	NAA-SR-3216	11-1-58
October-November 1958	NAA-SR-Memo-3399	1-1-59
December 1958 - January 1959	NAA-SR-3589	3-59
February-March 1959	NAA-SR-Memo-3745	5-1-59
April-May 1959	NAA-SR-Memo-4002	7-15-59
June-July 1959	NAA-SR-4326	10-15-59
August-September 1959	NAA-SR-4584	12-1-59
October-November 1959	NAA-SR-4767	1-15-60
December 1959 - January 1960	NAA-SR-Memo-4953	3-1-60

Bi-Monthly Progress Report, Systems for Nuclear Auxiliary Power (SNAP)

February-March 1960	NAA-SR-5225	6-1-60
April-May 1960	NAA-SR-5450	8-15-60

Bi-Monthly Progress Report, SNAP 2 Nuclear Auxiliary Power Unit Development

June-July 1960	NAA-SR-5691	1-10-60
August-September 1960	NAA-SR-5929	1-15-61
October-November 1960	NAA-SR-6021	3-1-61

SNAP PROGRESS REPORTS

CALENDAR YEAR

AI-AEC-13066

3-2

Progress Report, SNAP 2 Nuclear Auxiliary Power Unit Development

December 1960 - March 1961	NAA-SR-6292	6-15-61
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SNAP 2 Quarterly Progress Report

April-June 1961	NAA-SR-6691	10-15-61
July-September 1961	NAA-SR-6791	2-1-62
October-December 1961	NAA-SR-6991	5-16-62
January-March 1962	NAA-SR-7191	10-11-62
April-June 1962	NAA-SR-7491	12-15-62
July-September 1962	NAA-SR-7791	2-22-63
October-December 1962	NAA-SR-8091	3-31-62
January-March 1963	NAA-SR-8391	5-15-63
April-June 1963	NAA-SR-8091	8-31-63
July-September 1963	NAA-SR-8991	11-15-63
October-December 1963	NAA-SR-9291	2-14-64

B. SNAP REACTOR IMPROVEMENT PROGRAM - MERCURY RANKINE

Progress Report, SNAP Systems Improvement Program, Mercury-Rankine Program

January-March 1964	NAA-SR-9591	5-15-64
April-June 1964	NAA-SR-9991	8-15-64

Progress Report, Mercury Rankine Program

July-September 1964	NAA-SR-10491, V. 1-2	11-15-64
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Progress Report, SNAP Systems Improvement Program, Mercury Rankine Program

October-December 1964	NAA-SR-10791	2-11-65
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Progress Report, Mercury Rankine

January-March 1965, V. 2 (TRW)	NAA-SR-11091, V. 2	5-15-65
April-June 1965, V. 1 (AI)	NAA-SR-11491, V. 1-2	8-24-65
V. 2 (TRW)		10-8-65
July-September 1965	NAA-SR-11691	11-15-65

SNAP Systems Improvement Program, Mercury Rankine Program, Progress Report

October-December 1965	NAA-SR-11755, V. 1-2	3-15-66
January-March 1966	NAA-SR-11886, V. 1-2	5-30-66
April-June 1966	NAA-SR-12084, V. 1-2	8-15-66 11-15-66
July-September 1966	NAA-SR-12256	1-15-66

C. SNAP 2 REACTOR DEVELOPMENT (IMPROVEMENT) PROGRAM

SNAP 2 Reactor, Quarterly Progress Report

July-September 1963	NAA-SR-8990	11-15-63
October-December 1963	NAA-SR-9290	2-18-64

Progress Report, SNAP 2A Reactor

January-March 1964	NAA-SR-9590	6-9-64
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Progress Report, SNAP Reactor Development Program

April-June 1964	NAA-SR-9990	8-31-64
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Progress Report, SRIP Reactor Development Program

July-September 1964	NAA-SR-10490	12-31-64
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SNAP Reactor Improvement Program, Progress Report

October-December 1964	NAA-SR-10790	2-19-65
January-March 1965	NAA-SR-11090	5-15-65
April-June 1965	NAA-SR-11490	9-31-65
July-September 1965	NAA-SR-11690	11-22-65
October-December 1965	NAA-SR-11754	2-15-66
January-March 1966	NAA-SR-11882	5-23-66
April-June 1966	NAA-SR-12020	8-15-66
July-September 1966	NAA-SR-12157	11-15-66
October-December 1966	NAA-SR-12292	2-15-67
January-March 1967	NAA-SR-12378	5-19-67

D. SNAP 2/10A REACTOR PROGRAM

SNAP 2/10A Reactor, Quarterly Progress Report

-	NAA-SR-6694	12-1-61
July-September 1961	NAA-SR-6794	2-15-62
October-December 1961	NAA-SR-6994	6-30-62
January-March 1962	NAA-SR-7194	10-26-62
April-June 1962	NAA-SR-7494	2-5-63
July-September 1962	NAA-SR-7794	4-22-63
October-December 1962	NAA-SR-8094	5-7-63
January-March 1963	NAA-SR-8394	6-7-63
April-June 1963	NAA-SR-8694	10-9-63

E. SNAP 4 PROGRAM

Bi-Monthly Progress Report for SNAP Supporting R&D and SNAP 4

June-July 1960	NAA-SR-5694	1-10-61
August-September 1960	NAA-SR-5932	1-15-61
October-November 1960	NAA-SR-6024	3-1-61

Progress Report, SNAP 4

December 1960 - March 1961	NAA-SR-6295	6-15-61
April-June 1961	NAA-SR-6695	11-1-61
July-September 1961	NAA-SR-6795	12-15-61
October-December 1961	NAA-SR-6995	7-1-62
January-March 1962	NAA-SR-7195	6-1-62
April-June 1962	NAA-SR-7495	4-3-63
July-September 1962	NAA-SR-7795	4-15-63
October-December 1962	NAA-SR-8095	5-31-63
January-March 1963	NAA-SR-8395	7-15-63
July-September 1963	NAA-SR-8995	11-26-63

F. SNAP 8 PROGRAM

Bi-Monthly Progress Report, SNAP 8 Nuclear Auxiliary Power Unit Development

February-May 1960	NAA-SR-Memo-5715	
June-July 1960	NAA-SR-5692	1-10-61
August-September 1960	NAA-SR-5930	1-15-61
October-November 1960	NAA-SR-6022	3-15-61

Progress Report, SNAP 8 Reactor Development

December 1960 - March 1961	NAA-SR-6293	8-15-61
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SNAP 8 Quarterly Progress Report

April-June 1961	NAA-SR-6692	12-1-61
July-September 1961	NAA-SR-6792	7-16-62

Progress Report, SNAP 8

October-December 1961	NAA-SR-6992	8-15-62
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SNAP 8 Quarterly Progress Report

January-March 1961	NAA-SR-7192	12-1-62
April-June 1962	NAA-SR-7492	3-1-63
July-September 1962, Vol. I	NAA-SR-7792, V. I, II, & III	4-22-63
Vol. II		4-29-63
Vol. III		5-10-63
October-December 1962	NAA-SR-8092	5-20-63
January-March 1963	NAA-SR-8392	8-20-63
April-June 1963	NAA-SR-8692	9-27-63
July-September 1963	NAA-SR-8992	11-15-63
October 1963 - January 1964	NAA-SR-9392	3-16-64
February-April 1964	NAA-SR-9592	6-12-64
May-July 1964	NAA-SR-9992	9-22-64
August-October 1964	NAA-SR-10492	12-29-64

SNAP 8 Progress Report

November 1964 - January 1965	NAA-SR-10792	3-15-65
February-April 1965	NAA-SR-11092	6-15-65
May-July 1965	NAA-SR-11492	9-24-65
August-October 1965	NAA-SR-11692	12-29-65
October 1965 - January 1966	NAA-SR-11758	4-8-66
February-April 1966	NAA-SR-11898	7-15-66
May-July 1966	NAA-SR-12068	11-1-66
August-October 1966	NAA-SR-12188	1-24-67
November 1966 - January 1967	NAA-SR-12293	4-7-67
February-April 1967	NAA-SR-12414	6-15-67
May-July 1967	NAA-SR-12521	6-15-67

G. ADVANCED THERMOELECTRIC DEVELOPMENT PROGRAM

Advanced Thermoelectric Development Program

October-December 1963	NAA-SR-9298	2-21-64
January-April 1964	NAA-SR-9598	6-30-64
May-July 1964	NAA-SR-9998	9-30-64

Progress Report, Advanced Thermoelectric Development Program

August-October 1964	NAA-SR-10498	1-15-65
October-December 1964	NAA-SR-10798, V. 2	3-15-65
November 1964 - January 1965	NAA-SR-10798, V. 1	3-15-65
February-May 1965	NAA-SR-11098, V. 1-2	7-15-65
May-July 1965	NAA-SR-11498, V. 1-2	10-19-65
August-October 1965	NAA-SR-11698, V. 1-2	1-10-66
November 1965 - January 1966	NAA-SR-11760, V. 1-2	3-31-66
February 1966 - April 1966	NAA-SR-12016, V. 1-2	7-15-66
May 1966 - July 1966	NAA-SR-12158, V. 1-2	10-24-66
August 1966 - October 1966	NAA-SR-12245, V. 1-2	1-30-67
November 1966 - January 1967	NAA-SR-12436, V. 1-2	6-20-67
May-July 1967	NAA-SR-12541	9-11-67
July-December 1967	AI-AEC-12646	2-20-68
1968 Annual	AI-AEC-12786	5-12-69
1969 Annual (Final)	AI-AEC-12934	5-30-70

AI-AEC-13066

H. SNAP SUPPORTING R&D PROGRAMS

SNAP Supporting R&D, Quarterly Progress Report

April-June 1961	NAA-SR-6696	11-1-61
July-September 1961	NAA-SR-6796	11-15-61
October-December 1961	NAA-SR-6996	2-15-62
January-March 1962	NAA-SR-7196	7-1-62
April-June 1962	NAA-SR-7496	8-15-62

Progress Report for SNAP General Supporting Technology

July-September 1962	NAA-SR-7796	12-15-62
October-December 1962	NAA-SR-8096	2-15-63
January-March 1963	NAA-SR-8395	5-15-63
April-June 1963	NAA-SR-8696	8-15-63
July-September 1963	NAA-SR-8996	11-15-63
October 1963 - January 1964	NAA-SR-9396	3-16-64
February-April 1964	NAA-SR-9596	6-30-64
May-July 1964	NAA-SR-9996	9-30-64
August-October 1964	NAA-SR-10496	12-22-64
November 1964 - January 1965	NAA-SR-10796	4-1-65
February-April 1965	NAA-SR-11096	6-30-65
May-July 1965	NAA-SR-11496	9-30-65
August-October 1965	NAA-SR-11696	1-15-66
November 1965 - January 1966	NAA-SR-11756	3-15-66
February-April 1966	NAA-SR-11962	6-15-66
May-July 1966	NAA-SR-12055	9-20-66
August-October 1967	NAA-SR-12193	12-15-66
November 1966 - January 1967	NAA-SR-12316	5-5-67
February-April 1967	NAA-SR-12454	6-20-67

SNAP Reactor Programs Progress Report

August-October 1967	NAA-SR-12583	12-15-67
November 1967 - January 1968	AI-AEC-12641	3-20-68
February-April 1968	AI-AEC-12674	6-15-68
May-July 1968	AI-AEC-12729	9-15-68
August-October 1968	AI-AEC-12750	12-13-68

AI-AEC-13066

SNAP Reactor Programs Progress Report (Continued)

November 1968 - January 1969	AI-AEC-12773	3-18-69
February-April 1969	AI-AEC-12819	7-14-69
May-July 1969	AI-AEC-12870	9-23-69
August-October 1969	AI-AEC-12894	12-22-69
November 1969 - January 1970	AI-AEC-12923	4-1-70
February-April 1970	AI-AEC-12953	6-15-70
May-July 1970	AI-AEC-12975	9-8-70
August-October 1970	AI-AEC-12982	1-15-71
November 1970 - January 1971	AI-AEC-12989	3-15-71
February-April 1971	AI-AEC-12998	6-18-71
May-July 1971	AI-AEC-13011	9-15-71
August-October 1971	AI-AEC-13017	12-23-71
November 1971 - January 1972	AI-AEC-13023	4-15-72
February-April 1972	AI-AEC-13032	8-15-72
May-July 1972	AI-AEC-13039	10-15-72
August-October 1972	AI-AEC-13046	4-73

I. AEROSPACE SAFETY PROGRAM

Aerospace Safety Progress Report

July-September 1962	NAA-SR-7797	3-20-63
October-December 1962	NAA-SR-8097	4-1-63
January-March 1963	NAA-SR-8397	6-1-63
April-June 1963	NAA-SR-8697	8-15-63
July-September 1963	NAA-SR-8997	11-15-63
October-December 1963	NAA-SR-9297	2-14-64
January-March 1964	NAA-SR-9597	5-13-64
April-June 1964	NAA-SR-9997	8-15-64
July-September 1964	NAA-SR-10497	11-15-64
October-December 1964	NAA-SR-10797	2-15-65
January-March 1965	NAA-SR-11097	5-15-65
April-June 1965	NAA-SR-11497	8-4-65
July-September 1965	NAA-SR-11697	11-15-65
October-December 1965	NAA-SR-11759	2-15-66

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Aerospace Safety Progress Report (Continued)

January-March 1966	NAA-SR-11884	5-15-66
April-June 1966	NAA-SR-12021	8-15-66
July-September 1966	NAA-SR-12182	11-15-66
October-December 1966	NAA-SR-12291	2-15-67
January-March 1967	NAA-SR-12385	5-15-67

SNAP Reactor Aerospace Nuclear Safety Program Annual Report

GFY 1968	AI-AEC-12738	10-15-68
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Progress Report for SNAP Reactor Improvement Program, Aerospace Nuclear Safety Project

April-June 1967	NAA-SR-12517	8-23-67
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IV. BIBLIOGRAPHY - NUMERICAL LISTING

A. TOPICAL REPORTS, MEMOS, AND TDR'S

NAA-SR-10277

"Room Temperature Compressive Properties of Zr-10U Alloy Hydride and High Hydrogen Contents," L. B. Lundberg and J. R. Berling, 8-30-66

NAA-SR-10903

"SNAP 8 Experimental Reactor Operations and Test Results," L. M. Fead, L. D. Felten, and V. L. Rooney, 6-28-65

NAA-SR-Memo-11393

"Reactor Shield Weight Studies of Intrinsically Subcritical Low Power Systems," R. H. Norman and D. G. Oliver, 4-8-66

NAA-SR-Memo-11526

"High Temperature Coil Design," L. Donolan, 7-12-65

NAA-SR-Memo-11551

"A Suggested Program for Atmospheric Fallout Experiments," C. G. Hendricks and C. A. Willis, 12-1-65

NAA-SR-Memo-11574

"Experimental Study of Intrinsic Core Subcriticality by Neutron Spectral Changes," D. W. Clifford and S. Yee, 8-16-65

NAA-SR-11597

"Mercury Rankine Program Alternator Bare Seal-Unsealed Alternator," H. Horii and A. I. Wesolowski, 3-15-66

NAA-SR-Memo-11688

"Design Study of a 4π Shielded Manrated Reactor," R. Simms, R. Tobler, and J. Crano, 10-13-65

NAA-SR-Memo-11713

"Comparison of Void and Poison-Backed Control Drums for SNAP Reactors," R. Simms, 11-16-65

NAA-SR-Memo-11841

"Post Examination of SNAP 10A Thermoelectric Pump No. 16, 31, and 32," R. M. Willard, 2-9-66

NAA-SR-TDR-11954

"Particle Heating Regime Investigation and Description of the Particle Aerodynamic Trajectory and Heating (PATH) Computer Code," J. Rohlf's and S. R. Fields, 5-3-66

NAA-SR-11956

"End-of-Life Shutdown Studies," V. E. Moore, 6-30-66

NAA-SR-11959

"Thermophysical Properties of SNAP Fuel Materials," M. M. Nakata, C. J. Ambrose, and R. A. Finch, 9-25-67

NAA-SR-Memo-11966

"Least Squares Determination of Isothermal Rate Constants for the Oxidation of Metals," W. A. Young, 6-14-66

NAA-SR-12032

"In-Pile Behavior of SNAP 8 Experimental Reactor Type Sublength Fuel Elements (NAA 117 Experiment)," R. C. Courtright and R. E. Forrester, 11-30-67

NAA-SR-12038

"S8DR Temperature Switch Development and Testing," J. L. Powers, 5-15-68

NAA-SR-Memo-12113

"Studies of Core Geometry and Axial Reflector Effectiveness on Water Immersion of Medium Power (~600 kw) Reactors," D. G. Oliver, 8-11-66

NAA-SR-Memo-12125

"Development of AI-1 Cement Encapsulant," W. G. Long, 8-22-66

NAA-SR-12186

"SNAP 2 Developmental Reactor Mockup No. 1 Final Report," J. Maki, J. F. Milner, and B. R. Granich, 6-15-68

NAA-SR-Memo-12364

"Hydrogen Retention of SNAP 8 Fuel Elements Containing Nickel-Plated Fuel Rods," W. F. Dennison and J. L. Nommelin, 2-19-68

NAA-SR-Memo-12384

"Detailed Thermal Analysis of SNAP Reactor Fuel Elements Including Asymmetry and Spacing Effects," P. M. Magee, 3-20-67

NAA-SR-Memo-12392

"NAA 120-4 Heat Transfer and Stress Analysis," L. L. Lowe, 4-7-67

NAA-SR-12449

"Electron Microscopy Analysis of Irradiated SNAP 8 Fuel," K. J. Miller, 8-20-67

NAA-SR-TDR-12480

"NAA-120-4 Irradiation Experiment Assembly," E. L. Babbe, 5-26-67

NAA-SR-12504

"The Disassembly and Postoperation Component Examination of the SNAP 10A FS-3," J. P. Dooley and J. P. Beall, 9-15-67

NAA-SR-Memo-12507

"Performance of the S8DR Zirconium Hydride Fuel Element," J. G. Asquith, G. W. Myers, and W. T. Roberts, 6-20-68

NAA-SR-TDR-125110

"Electron Microprobe Analysis of Irradiated S8ER Cladding," W. E. Krupp and L. Cooper, 8-24-67

NAA-SR-12517

"Progress Report SRIP, Aerospace Nuclear Safety Project, April-June 1967," L. M. Maki, 8-23-67

NAA-SR-Memo-12521

"SNAP 8 Progress Report, May-July 1967," D. G. Mason, 9-5-67

NAA-SR-12522

"Advanced SNAP Reactor Capability and Program Study - Vol I, Technical Requirements and Reactor Capabilities; Vol II, Development Program Alternatives," Staff, 10-20-67

NAA-SR-12541

"Progress Report Advanced Thermoelectric Development Program, May-July 1967," Staff, 9-11-67

NAA-SR-TDR-12543

"NAA-121 - As Built Assembly Procedures," L. Morrison, 8-15-67

NAA-SR-TDR-12553

"Low Cycle Fatigue Testing of Irradiated Hastelloy N," W. E. Krupp and L. L. Johnson, 9-28-67

NAA-SR-TDR-12557

"Alternate Bearing Materials for the ZrH Reactor," G. F. Queener, 9-11-67

NAA-SR-TDR-12558

"Reverse Bend Fatigue Test of Unirradiated S8DS Strip Specimens at 1400°F," J. W. Wunderlich, D. F. Atkins, and L. C. Metcalf, 11-1-67

NAA-SR-Memo-12562

"SNAP Temperature Coefficient Literature Abstract," L. A. Wilson, 9-15-67

NAA-SR-12564

"SNAP 8 Development Reactor (S8DR) Core Thermal and Hydraulic Performance," P. M. Magee, 3-10-68

NAA-SR-Memo-12576

"Nuclear Analysis for the NAA-121-1 Irradiation Experiments," F. C. Franklin, 9-29-67

NAA-SR-TDR-12577

"Post-Irradiation Examination of In-Pile Stress-Rupture Specimens ORR S-2 and ORR S-3," L. L. Johnson, 9-29-67

NAA-SR-TDR-12579

"Structural Analysis of the NAA-121-1 Irradiation Experiment Assembly," A. W. Dalcher, 10-6-67

NAA-SR-12583

"SNAP Reactor Programs, Progress Report, August-October 1967," D. G. Mason, 12-15-67

NAA-SR-Memo-12584

"NAA-121-1 Heat Transfer Analysis," T. A. Park and R. Roberts, 10-5-67

NAA-SR-12587

"Phase Relationships at High Hydrogen Contents in the SNAP Fuel System,"
K. E. Moore, W. A. Young, and M. M. Nakata, 2-15-68

NAA-SR-Memo-12588

"Post-Irradiation Examination of the NAA-115-2 Irradiation Experiment,"
R. E. Forrester, 10-7-67

NAA-SR-TDR-12589

"Electrical Resistivity of SNAP Fuels," C. A. Smith, 10-18-67

NAA-SR-TDR-12590

"Computed Stress, Relaxation Curves for Hastelloy N," A. W. Dalcher,
11-1-67

NAA-SR-Memo-12591

"Development Test Report - S8DR Pre-Production Fuel Elements,"
W. D. Fuller, 10-20-67

NAA-SR-TDR-12592

"Transport-Theory Angular Quadrature Coefficients for SNAP Reactor
Analysis," E. H. Ottewitte, 10-19-67

NAA-SR-TDR-12594

"Space Power System Pump Selection," W. D. Leonard, 10-19-67

NAA-SR-TDR-12596

"Plan for Ten-Module Converter NaK Loop Test," S. J. Block and
W. D. Leonard, 10-25-67

NAA-SR-TDR-12597

"Evaluation of Residual Stresses in S8DR Fuel Element Cladding. Test
Report," L. R. Stone, 10-27-67

AI-AEC-Memo-12598

"NAA 121-1 Pre-Irradiation Measurement," H. L. Hodges, 8-23-68

NAA-SR-12601

"Helium Embrittlement in Type 304 Stainless Steel," D. Kramer,
H. R. Brager, and C. G. Rhodes, 12-5-67

NAA-SR-12605

"S8DR Actuator Interim Report," W. G. Long, 12-30-67

NAA-SR-12607

"Thermophysical Properties of Unirradiated SNAP Fuels," W. A. Young, C. C. Weeks, and M. M. Nakata, 2-23-68

NAA-SR-TDR-12609

"Rotating Beam Fatigue Testing of SNAP 8 Fuel Cladding," S. K. Lee, 11-10-67

NAA-SR-Memo-12610

"Fuel Rod Fabrication," C. A. Martin, 11-7-67

NAA-SR-TDR-12611

"S8DR Flow Distributor Design," R. H. Tromel, 11-10-67

NAA-SR-Memo-12612

"Coating Adherence Studies for S8DR Fuel Elements," G. W. Meyers, 12-12-67

NAA-SR-Memo-12614

"SNAP Engineering Test Facility Safety Analysis Report. Addendum for Intrinsic Subcriticality Experiments," K. G. Golliher, V. A. Swanson, and A. W. Thiele, 11-30-67

NAA-SR-Memo-12614

"SNAP Engineering Test Facility Safety Analysis Report. Addendum for Intrinsic Subcriticality Experiments. Supplement 1," W. Heneveld, 2-8-71

NAA-SR-TDR-12615

"BeO Literature Survey," R. M. Willard, 11-17-67

NAA-SR-TDR-12623

"Thermal Behavior of Manned Orbiting Research Laboratory and Lunar Base Reactor Systems During Atmospheric Reentry," L. D. Montgomery, 12-14-67

NAA-SR-12625

"In-Pile Behavior of SNAP 8 Experimental Reactor Type Sublength Fuel Elements (NAA 115-2 Experiment)," R. E. Forrester and W. J. Roberts, 6-15-68

NAA-SR-TDR-12631

"Determination of Stress-Rupture Properties of SNAP-8 Fuel Cladding in a NaK Environment," J. D. Stearns, 12-4-67

NAA-SR-TDR-12632

"Development Test Report - Some Effects of Thermal Cycling and Compressive Stress on the Stability of Zirconium Hydride Fuel Rods," L. R. Stone, 12-1-67

NAA-SR-TDR-12633

"Studies of Samarium Sesquioxide Used for S8DR," J. T. Robert, 12-4-67

NAA-SR-TDR-12634

"Preparation of Layers of Oxides of Th, U, Pu, and Rare Earth Elements by Electrodeposition from Organic Solutions," W. A. Young, 12-6-67

AI-AEC-TDR-12640

"Effect of Computational Uncertainties on SNAP Reactor Shield Weight," C. A. Goetz, 1-15-68

AI-AEC-12641

"SNAP Reactor Programs Progress Report, November 67 - January 68," R. L. Detterman, 3-20-68

AI-AEC-TDR-12644

"S8DR Journal Bearing Test," P. H. Horton, 1-12-68

AI-AEC-12646

"Progress Report Advanced Thermoelectric Development Program July-December 1967," S. J. Block and L. P. Garvey, 2-20-68

AI-AEC-TDR-12651

"Operating Temperatures of S8DR Reflectors, Actuators, and Other Components," P. M. Magee, 2-7-68

AI-AEC-Memo-12653

"Incoloy 800 - Cladding Barrier Development Tests," J. L. Nummelin,
2-16-68

AI-AEC-TDR-12654

"Operational Life Testing of an S8DR Bearing Set in 10 to the -5th Power
torr Vacuum at 1150 Degrees F with Ball Assemblies Having ...,"
P. H. Horton and L. G. Kellogg, 2-20-68

AI-AEC-TDR-12655

"High Temperature SNAP Reactor Shielding Studies, the Corrosion of
1020 Carbon Steel, Croloy 2-1/4, and Type 446 Stainless," F. H. Welch,
2-27-68

AI-AEC-TDR-12656

"Quality of Piping Welds in the S8DR Test Facility," J. W. Wunderlich,
2-27-68

AI-AEC-TDR-12658

"S8DR Control Drum Bearing Operational Test Without Dry Film Lubricant
at 1200°F Alumina Coated Inconel-750 Ball Design," P. H. Horton,
2-28-68

AI-AEC-TDR-12665

"Failure Analysis of the Alumina Coated S8DR Control Drum Bearings,"
P. H. Horton, 4-30-68

AI-AEC-TDR-12666

"Hydrogen Redistribution in the NAA-121-1 Irradiation Experiments,"
M. E. Nathan, 3-15-68

AI-AEC-TDR-12669

"S8DR Thermal Cycle Test No. 3, Inconel-750 Ball with P5-N Carbon-
Graphite Bushings," P. H. Horton, 3-15-68

AI-AEC-12670

"Helium Embrittlement of Type 316 Stainless Steel," D. Kramer,
K. R. Garr, C. G. Rhodes, and A. G. Pard, 4-30-68

AI-AEC-12671

"Solubility of Copper in Sodium," R. L. Eichelburger and R. L. McKisson,
5-31-68

AI-AEC-Memo-12672

"Thermoelectric System AEC Review, March 29, 1968, Washington, D. C.,"
Staff, 3-27-68

AI-AEC-TDR-12673

"New Cross Section Needs for Zirconium Hydride SNAP Reactors,"
E. H. Ottewitte, 4-2-68

AI-AEC-12674

"SNAP Reactor Programs Progress Report, February-April 1968,"
D. G. Mason, 6-15-68

AI-AEC-TDR-12675

"NaK Compatibility of S8DR Brazed Thermocouple Bulkhead Joints,"
S. K. Lee, 4-3-68

AI-AEC-Memo-12678, Vol 1

"Project Proposals for FY 1969-1970, SNAP Reactor Programs,"
Staff, 5-1-68

AI-AEC-12681

"High Heat Flux Heater Development Status Report," R. L. McKisson and
E. L. Babbe, 6-15-68

AI-AEC-TDR-12683

"S8DR Operations Manual," E. M. Larson, 5-1-68

AI-AEC-TDR-12684

"SNAP Ground Prototype Test Facility (Bldg 059) Operations Manual,"
R. M. Micklich and E. M. Larson, 4-29-68

AI-AEC-TDR-12690

"S8DR Control Drum Bearing Thermal Cycle Test No. 2, Aluminum Oxide
Coated Inconel-750 Ball Design with P5-N Graphite Bushings," P. H. Horton,
5-6-68

AI-AEC-Memo-12692

"SNAP 8 Pump Design and Analysis," W. D. Leonard, 4-30-68

AI-AEC-12695

"Thermal Gradient Effects on Stress-Rupture Behavior of Thin-Walled
Tubing," J. H. Shively, 6-25-68

AI-AEC-Memo-12697

"Performance Test and Evaluation of Special Seven Pre-S8DR Fuel Elements," W. D. Fuller and K. E. Moore, 5-6-68

AI-AEC-TDR-12698

"High Emissivity Coating for Beryllium," S. J. Block, 5-1-68

AI-AEC-TDR-12699

"A Method for the Precision Measurement of the Thickness and Uniformity of Thin Films and Foils," W. A. Young, 5-6-68

AI-AEC-TDR-12701

"Endurance Tests of S8DR Preproduction Fuel Elements - Interim Report," L. D. Dean and W. H. Friske, 5-8-68

AI-AEC-Memo-12702

"S8DR Fuel Element Quality Assurance Record," C. Donnelly and W. Nagel, 3-21-68

AI-AEC-12703

"Phase Relationship in the ($\alpha + \delta$) Region of the ZrH System," K. E. Moore and M. M. Nakata, 9-30-68

AI-AEC-TDR-12704

"Test Report - Development, SNAP 8 Scram Mechanism 7568-917054, S/N 003," G. F. Queener, 5-20-68

AI-AEC-12706

"Load Sensitive, Low Noise Pulse Preamplifier for Use with Semiconductor Detectors of Nuclear Radiation," K. Langrod, 5-24-68

AI-AEC-Memo-12715

"Reference Zirconium Hydride Reactor," L. M. Maki and E. J. Donovan, Jr., 6-15-69

AI-AEC-Memo-12716

"ZrH Reactor Design Studies," E. J. Donovan and L. M. Maki, 6-15-69

AI-AEC-Memo-12717

"Reference Zirconium Hydride Reactor Thermoelectric System," Staff, 6-15-69

AI-AEC-TDR-12720

"High Temperature SNAP Reactor Shielding Studies — Aggregate Shield Materials — A Summary Report," F. H. Welch, 6-19-68

AI-AEC-TDR-12723

"Zirconium Hydride Reactor Technology Actuator Coil Reliability Report," L. Donelan, 7-3-68

AI-AEC-Memo-12725

"Thermal Cycle Testing of the Advanced ZrH Reactor Control Drum Bearings," P. H. Horton, 6-27-68

AI-AEC-TDR-12728

"SNAP 8 Material Evaluation Shaft Tests at 1250°F and 10^{-5} torr," P. H. Horton, 8-14-68

AI-AEC-12729

"SNAP Reactor Programs Progress Report, May-July 1968," J. G. Asquith, 9-15-68

AI-AEC-TDR-12732

"Irradiation Effects Upon the S8DR Reflector and Control Components," J. R. Miller and T. S. Nakae, 9-3-68

AI-AEC-TDR-12734

"Material Compatibility and Sliding Friction Studies on S8DR Reference and Alternate Bearing Couples," P. H. Horton, 8-28-68

AI-AEC-12738

"SNAP Reactor Aerospace Nuclear Safety Program Annual Report GFY 1968," R. L. Detterman, 10-15-68

AI-AEC-TDR-12740

"S8DR Fuel Element Performance Tests — Interim Report," W. H. Friske, 8-26-68

AI-AEC-TDR-12745

"Review of Radiation Effects on Incoloy 800," J. R. Miller, 1-21-69

AI-AEC-Memo-12746

"S8DR Normal Uranium Fuel Element Quality Assurance Record,"
C. W. Donnelly, 9-26-68

AI-AEC-12750

"SNAP Reactor Programs Progress Report, August-October 1968,"
Staff, 12-13-68

AI-AEC-TDR-12752

"Radiator Panel Folding Mechanisms," J. Facha, 10-23-68

AI-AEC-TDR-12755

"Parametric Load Testing of the Advanced Zirconium Hydride Reactor
Control Drum Bearings," P. H. Horton, 11-4-68

AI-AEC-TDR-12758

"Performance Summary of Four Double Clad S8DS Developmental
Elements," W. F. Dennison, 11-6-68

AI-AEC-Memo-12762

"Correlation of Data from NaK Heated and Electrically Heated Thermo-
electric Module Tests," J. L. Johnson, 11-18-68

AI-AEC-TDR-12766

"High Temperature SNAP Reactor Shielding Studies; Preparation of Dense
LiH/ZrH and LiH/W Aggregate Shield Materials," F. H. Welch, 12-4-68

AI-AEC-TDR-12768

"Friction Studies of Mo S₂-Ta Compact Lubricant Materials,"
P. H. Horton, 5-12-69

AI-AEC-TDR-127690

"High Temperature SNAP Reactor Shielding Studies: The Effect of Time
and Temperature on the Liquid Lead Corrosion of 1020 Carbon,"
F. H. Welch, 12-17-68

AI-AEC-TDR-12772

"Intrinsic Water Subcriticality Transient Data Analysis — First Status
Report," W. A. Rhoades and B. J. Dray, 1-6-68

AI-AEC-12773

"SNAP Reactor Programs Progress Report, November 1968 - January 1969,"
Staff, 3-18-69

AI-AEC-Memo-12775

"S8DR Fuel Rod Fabrication," D. T. McClelland, 1-29-69

AI-AEC-12779

"S8DR Actuator Second Interim Report," L. E. Donelan, 3-1-69

AI-AEC-Memo-12782

"Chromium Redistribution Evaluation of S8DR Coated Tubing,"
J. R. Miller and J. T. Roberts, 5-2-69

AI-AEC-Memo-12784

"SNAP Program Review, Washington, D. C., January 30, 1969,"
Staff, -69

AI-AEC-12786

"Progress Report Advanced Thermoelectric Development Program,
1968 Annual," R. L. Detterman, 5-12-69

AI-AEC-Memo-12789

"NAA 120-4 Operations Report," N. F. Davies, 2-14-69

AI-AEC-Memo-12795

"Nuclear Analysis for the NAA 140-1 Irradiation Experiment,"
F. C. Franklin, 2-21-69

AI-AEC-TDR-12796

"High Temperature SNAP Reactor Shielding Studies. The Tensile Properties
of 1020 Carbon Steel, Croloy 2-1/4 Steel and Type . . . ,"
F. H. Welch and J. A. Quealy, 2-24-69

AI-AEC-TDR-12797

"Performance Tests of S8DR Preproduction Fuel Elements," W. H. Friske,
2-24-69

AI-AEC-TDR-12803

"ED2IT - Computer Program to Process Data from the S8DR Data Logger,"
L. D. Felten, 3-14-69

AI-AEC-12806

"SNAP 8 Development Reactor Fuel Element Fabrication," D. T. McClelland,
6-30-69

AI-AEC-Memo-12811

"Project Proposals FY 1970-1971. Space Electric Power Program,"
Staff, 5-69

AI-AEC-12819

"SNAP Reactor Programs Progress Report, February-April 1969,"
Staff, 7-14-69

AI-AEC-TDR-12824

"Structural Analysis of the SNAP-8 Developmental Reactor Fuel Element
Cladding," A. W. Dalcher, 4-15-69

AI-AEC-12839

"Reactor - Thermoelectric System for NASA Space Station," J. D. Gylfe
and R. A. Johnson, 7-1-69

AI-AEC-Memo-12840

"Reactor - Thermoelectric System Development Plans," J. D. Gylfe and
R. A. Johnson, 7-1-69

AI-AEC-TDR-12842

"Material Properties of BeO Reflector EX-N69622007-4, Test Report -
Development," G. F. Queener, 6-2-69

AI-AEC-12850

"SNAP 8 Automatic Startup and PCS Mercury Injection Simulation Tests,"
R. F. Audette, 10-31-69

AI-AEC-TDR-12851

"High Temperature SNAP Reactor Shielding Studies. The Effect of a
Thermal Gradient on the Relocation of Hydrogen in a Lithium . . .,"
F. H. Welch, 6-3-69

AI-AEC-TDR-12852

"SNAPTRAN Operations Manual," W. H. Heneveld, 6-10-69

AI-AEC-12864

"SNAP 8 Development Reactor Nuclear Analysis," L. D. Swenson, 10-30-69

AI-AEC-TDR-12865

"Sliding Friction and Wear Testing of Advanced ZrH Bearing Couples,"
P. H. Horton, 6-26-69

AI-AEC-TDR-12867

"High Temperature SNAP Reactor Shielding Studies," F. H. Welch, 6-24-69

AI-AEC-12870

"SNAP Reactor Programs Progress Report, May-July 1969," Staff, 9-23-69

AI-AEC-TDR-12874

"Evaluation of a New 28-Group Cross-Section Library for SNAP Reactor
Analysis," J. M. Otter, 7-21-69

AI-AEC-TDR-12876

"Emitance of Ta-10W, Emittance and Coating Adhesion of an Aluminide
Coating on a Ta-10W Substrate, Test Report Development," G. F. Queener,
8-7-69

AI-AEC-TDR-12881

"High Temperature SNAP Reactor Shielding Studies: The Effect of Carbon
Additions to Lead to Suppress the Decarburization of ...," F. H. Welch,
9-9-69

AI-AEC-Memo-12883

"Post-Irradiation Examination of the NAA 120-4 Irradiation Experiment,"
R. E. Forrester, 12-11-69

AI-AEC-12894

"SNAP Reactor Program Progress Report, August-October 1969," Staff,
12-22-69

AI-AEC-12896

"Thermal Gradient Effects on Stress-Rupture Behavior of Thin-Walled
Tubing," J. H. Shively and M. W. Mahoney, 1-2-70

AI-AEC-TDR-12900

"Advanced ZrH Reactor Bearings 1600°F 5000-Hour Material Compatibility Test," P. H. Horton, 2-16-70

AI-AEC-Memo-12902

"Nuclear Space Power Program Presentation, NASA-MSFC, November 4, 1969, Atomics International," Staff, 11-4-69

AI-AEC-TDR-12903

"High Temperature SNAP Reactors Shielding Studies: The Effect of Carbon Additions to Lead to Suppress the Decarburization of . . .," F. H. Welch, 11-17-69

AI-AEC-TDR-12905

"Lithium Hydride Technology: Design Casting, and Examination of an Annular LiH Shield (Experimental Shield No. 3)," F. H. Welch, 12-3-69

AI-AEC-Memo-12908

"S8DR Operational Review Presentation," Staff, 11-69

AI-AEC-12909

"Reactivity Effects of Gadolinium-155 on a SNAP Reactor," B. J. Dray, et al, 4-15-70

AI-AEC-Memo-12917, (Volumes I through IV)

"Postoperational Safety of Reactors Power System for NASA Space Station - Preliminary Report"

Volume I, "Introduction and Summary," J. D. Gylfe, et al., 6-30-70

Volume II, "Mission and System Description," J. D. Gylfe, et al., 7-1-69

Volume III, "Safety Analysis," W. R. Lahs, et al., 6-30-70

Volume IV, "Support Analysis," R. F. Audette, et al., 6-30-70

AI-AEC-TDR-12919

"Shock, Acceleration, and Vibration Testing of Advanced ZrH Reactor Drum Bearings," P. H. Horton, 1-26-70

AI-AEC-Memo-12920

"SNAP Critical Facility Safety Analysis Report. Addendum for SCA-4A Operation," W. H. Heneveld, 1-15-70

AI-AEC-TDR-12921

"S8DR 12,000 Hour Drum Bearing Design Verification Tests,"
P. H. Horton, 3-12-70

AI-AEC-12923

"SNAP Reactor Programs Progress Report, November 1969-January 1970,"
Staff, 4-1-70

AI-AEC-12925

"S8DR Actuator Final Report," L. E. Donelan, 3-15-70

AI-AEC-12932

"Effect of Helium on Stress-Rupture Behavior of Type 304 and 316 Stainless
Steels," K. R. Garr, D. Kramer, and C. G. Rhodes, 4-15-70

AI-AEC-12934

"Advanced Direct-Radiating Thermoelectric Program, Final Report,"
J. L. Johnson (AI) and L. P. Garvey (RCA), 5-30-70

AI-AEC-Memo-12936, Vol I

"Project Proposals FY 1971-1972. Space Electric Power Program,"
Staff, 6-70

AI-AEC-Memo-12940

"Post-Irradiation Examination of NAA-121 Irradiation Experiment,"
R. E. Forrester, 3-24-70

AI-AEC-12944

"Biaxial Creep and Stress Rupture of Wire-Wrapped Cladding in 1200°F
Sodium Under ΔT Conditions," J. H. Shively, 4-15-70

AI-AEC-12953

"SNAP Reactor Programs Progress Report, February-April 1970,"
Staff

AI-AEC-12963

"Effects of Irradiation on Hydrided Zirconium-Uranium Alloy - NAA 120-4
Experiment," N. F. Davies and R. E. Forrester, 6-30-70

AI-AEC-12964

"SNAP 8 Control Drum Bearing Development and Evaluation History,"
P. H. Horton, 8-15-70

AI-AEC-12975

"SNAP Reactor Programs Progress Report, May-July 1970," Staff, 9-8-70

AI-AEC-12976

"Stress-Rupture Behavior of Types 304 and 316 Stainless Steel Cladding in
High-Temperature Static Sodium," D. F. Atkins, 9-30-70

AI-AEC-12982

"SNAP Reactor Programs Progress Report, August-October 1970,"
Staff, 1-15-71

AI-AEC-12989

"SNAP Reactor Programs Progress Report, November 1970 - January 1971,"
3-15-71

AI-AEC-12997 Volumes III and IV

"Project Proposals FY 1972-1973 Space Electric Power Program,"
Staff, 7-71

AI-AEC-12998

"SNAP Reactors Programs Progress Report, February-April 1971,"
Staff, 6-18-71

AI-AEC-13002

"Prototype S8DR Fuel Element Performance Test (NAA 121-1 Experiment),"
J. G. LaBlanc, 7-15-71

AI-AEC-13003

"Findings of SNAP 8 Developmental Reactor Post Test Examination,"
A. H. Lillie and V. L. Rooney, Jr., 6-30-71

AI-AEC-13011

"SNAP Reactor Programs Progress Report, May-July 1971," Staff, 9-15-71

AI-AEC-13017

"SNAP Reactor Programs Progress Report, August-October 1971,"
Staff, 12-23-71

AI-AEC-13023

"SNAP Reactor Programs Progress Report, November 1971 - January 1972,
Staff, 4-15-72

AI-AEC-13032

"SNAP Reactor Programs Progress Report, February-April 1972,"
Staff, 8-15-72

AI-AEC-13039

"SNAP Reactor Programs Progress Report, May-July 1972," Staff,
10-15-72

AI-AEC-13046

"SNAP Reactor Progress Report, August-October 1972," Staff

AI-AEC-13056

"Reactor Safety Report Bibliography, 1961-1972," Staff, 3-9-73

AI-AEC-13058

"Performance Modeling of the 5-kwe Reactor Thermoelectric System,
R. V. Anderson, 4-3-73

AI-AEC-13065

"GEOM - A Thermal-Hydraulic Stress Code, E. Moody, 5-10-73

AI-AEC-13067

"SNAP Programs Summary Report," D. Staub et al., 6-73

AI-AEC-13069

"SNAP 4 Summary Report," A. R. Marchese, 6-73

AI-AEC-13070

"SNAP 8 Summary Report," A. W. Thiele and S. Berger, 6-73

AI-AEC-13071

"Summary of SNAP 8 Developmental Reactor (S8DR) Operations,"
L. D. Felten and H. D. May, 6-73

AI-AEC-13072

"Design and Integration of Zirconium Hydride Reactor Power Systems,"
J. H. VanOsdol, 6-73

AI-AEC-13073

"Scram Controller and Automatic Controller, 5-kwe Reactor Thermoelectric System," R. J. Tuttle, 5-25-73

AI-AEC-13074

"Operational Hazards Evaluation for 5-kwe Reactor Thermoelectric System Ground Test," R. J. Tuttle, 6-73

AI-AEC-13075

"Reactor Assembly, 5-kwe Reactor Thermoelectric System,

AI-AEC-13076, Volumes I through V

"Selected Computer Codes and Libraries,"

Volume I, "ZIP - A Timeshare Program for SNAP Reactor Parametric Studies," H. Rood, 6-73

Volume II, "APC - A Multigroup Transport Buckling Iteration Code," D. S. Bost, 5-2-73

Volume III, "COMPOZ - A Multigroup Transport Perturbation Code," V. S. Lekach, 6-73

Volume IV, "A 28-Group Cross Section Library for SNAP Reactor Analysis," J. M. Otter, 5-31-73

Volume V, "TEMPRO - A Heat Transfer-Lifetime Code for SNAP Reactor Analysis," L. D. Felten and E. Moody, 6-73

AI-AEC-13077

"Fuel Element and Core Bundle, 5 kwe Reactor Thermoelectric System, L. D. Felten and E. Moody, 6-73

AI-AEC-13078

"Zirconium Hydride Reactor Control Reflector Systems Summary Report," P. H. Horton and W. J. Kurzeka, 6-73

AI-AEC-13079

"Zirconium Hydride Reactor Control System Bearing Development Summary Report," P. H. Horton, 6-73

AI-AEC-13080

"Zirconium Hydride Reactor Control Drive Actuator Development," L. E. Donelan, 6-73

AI-AEC-13081

"Radiation Shielding for Zirconium Hydride Reactor Systems,"
V. Keshishian, et al., 6-73

AI-AEC-13082

"Process Development and Fabrication of Zirconium Hydride Hastelloy X
Clad Fuel Elements," T. G. Parker, Jr., 6-73

AI-AEC-13083

"Hastelloy X Cladding Material Evaluation," K. E. Moore, R. G. Brengle,
and T. G. Parker, Jr., 6-73

AI-AEC-13084

"Zirconium Hydride Fuel Element Performance Characteristics,"
A. F. Lillie, 6-73

AI-AEC-13085

"Burnable Poison Deposition on Zirconium Hydride Fuel — Process
Development and Evaluation," R. Van Houten, 6-73

AI-AEC-13086

"Control Worth of Sliding Reflectors for Zirconium Hydride Reactors,"
D. S. Bost, 6-73

AI-AEC-13087

"Zirconium Hydride Reactor Core Hydraulic Studies Summary Report,"
B. J. Ostermier, 6-73

AI-AEC-13088

"Zirconium Hydride Reactor Core Heat Transfer Studies Summary Report,"
A. N. Gallegos and A. R. Marchese, 6-73

AI-AEC-13089

"Space Nuclear System Sheathed Electrical Cables Summary Report,"
L. E. Donelan, 6-73

AI-AEC-13090

"Space Nuclear System Volume Accumulator Development Summary Report,"
W. D. Whitaker, 6-73

AI-AEC-13091

"Space Nuclear System Thermoelectric NaK Pump Development Summary Report," J. L. Johnson, 6-73

AI-AEC-13092

"Space Nuclear System Expansion Joint Summary Report," W. D. Whitaker and T. T. Shimazaki, 5-73

AI-AEC-13093

"Design and Analysis of Radiator-Structure for Space Power Systems," W. H. Dauterman and L. D. Montgomery, 6-73

AI-AEC-13094

"Volume Accumulator Design Analysis Computer Code," W. D. Whitaker and T. T. Shimazaki, 6-73

AI-AEC-13095

"Thermoelectric Pump Performance Analysis Computer Code," J. L. Johnson, 5-73

AI-AEC-13096

"5-kwe Reactor Thermoelectric System Summary," T. H. Springer, 6-73

AI-AEC-13098

"System Statistical Reliability Model and Analysis," V. S. Lekach and H. Rood, 5-73

AI-AEC-13099

"NaK Loop Testing of Thermoelectric Converter Modules," J. L. Johnson, 5-73

AI-AEC-13100

"Aerospace Safety Program Summary Report," J. M. Otter, K. E. Buttrey, and R. P. Johnson, 6-73

AI-AEC-13102

"Materials Interactions Between the Thermoelectric Converter and the 5-kwe Reactor System," P. B. Ferry, 6-73

B. AI REPORTS

AI-65-268

"A Proposal for the Study of Nuclear Power Systems for the Voice Broadcast Statellite," Staff, 12-20-65

AI-66-Memo-4

"Space Powerplant Working Fluids," I. E. Randall, 1-6-66

AI-66-TDR-5

"Structural Analysis of SNAP Shielding Research Reactor System," J. V. Facha, 1-7-66

AI-66-TDR-8

"A Comparison of Refrigeration Systems for Space Vehicle Application," P. Barbour, 1-21-66

AI-66-TDR-11

"Radiator Design 3 kwe RIG-AAP," A. P. Gelman, 1-13-66

AI-66-Memo-23

"Evaluation of Space Power Conversion Systems," L. Bernath, J. L. Grimaldi, and H. N. Rosenberg, 2-28-66

AI-66-TDR-25

"Performance Characteristics of Mercury and Organic Rankine Cycles and the Argon Brayton Cycle," P. Barbour, 2-28-66

AI-66-Memo-44

"Nuclear Power Systems for Lunar Exploration," Staff, 4-8-66

AI-66-Memo-44, Rev 1

"Nuclear Power Systems for Lunar Exploration," Staff, 4-18-66

AI-66-50-1

"Technical Proposal to U. S. Army Engineer Research and Development Laboratories for a Study of an Advanced Nuclear Power System," Staff, 4-8-66

AI-66-50-2

"Cost Proposal to U. S. Army Engineer Research and Development Laboratories for a Study of an Advanced Nuclear Power System Concept 2," Staff, 4-8-66

AI-66-68-1

"Proposal for Nuclear Power Supply for an Acoustic Transponder," Staff, 4-25-66

AI-66-71-1

"Cost Proposal to the Office of Naval Research for an Extension of the Direct Conversion Reactor Study," Staff, 4-27-66

AI-66-71-2

"Technical Proposal to the Office of Naval Research for an Extension of the Direct Conversion Reactor Study," Staff, 4-27-66

AI-66-Memo-79

"SNAP Programs Briefing for Presidential Scientific Advisory Committee, Washington, D. C., May 1966," Staff, 5-6-66

AI-66-Memo-99

"SNAP 8 Briefing, Washington, D. C., May 17, 1966," Staff, -66

AI-66-108

"Emittance Coatings," Staff, 8-2-66

AI-66-TDR-118

"SNAP Systems Structural Development," E. A. Robb, 6-30-66

AI-66-129

"Atomics International Space Structures Capabilities," Staff, -66

AI-66-Memo-154

"Reactor Thermoelectric System Design Studies, 0.5 to 10 kwe," Staff, 11-18-66

AI-66-206

"10 to 30-kwe Nuclear Underwater Power System," Staff, 10-17-66

AI-67-30, Volumes I and II

"Proposal for Nuclear Power Systems Application Study for TV Broadcast Satellites," Staff, 2-23-67

AI-67-33, Volumes I and II

"Proposal to the Office of Naval Research for an Extension of the Direct Conversion Reactor Study," Staff, 3-15-67

AI-67-TDR-35

"Flexible NaK Elements for Deployable Space Power Systems," J. V. Facha, 3-1-67

AI-67-TDR-37

"October 1965 - September 1966 Progress Report TPA-101, Advanced Nuclear Thermoelectric Technology," S. R. Rocklin, 1-13-67

AI-67-67

"Lightweight High-Power Thermoelectric Module Development, Quarterly Progress Report, January - March 1967," S. Rocklin, D. Leonard, and S. Block, 5-1-67

AI-67-TDR-102

"Characteristics of Marine and Terrestrial Long-Lived Unattended Power Sources Less than 10 kw," J. E. Stewart, 7-5-67

AI-67-Memo-160, Volumes I and II

"Preproposal Evaluation for an Organic Rankine Nuclear System Study," Staff, 10-67

AI-68-Memo-7

"Preliminary Review of a 2-Mwe Nuclear -MHD Space Power System for Extended Mission Applications," L. Prem, R. Wimmer, and M. Burg, 1-12-68

AI-68-Memo-8

"SNAP Reactor - Thermoelectric Power Systems, Progress Report, Presented to National Aeronautics and Space Council," Staff, 1-23-68

AI-68-46, Volumes I, II, and III

"A Nuclear Power System for a Hardened Underground Missile Facility," Staff, 5-6-68

AI-68-Memo-81, Parts 1 and 2

"Compact Systems Presentation to AEC Headquarters - NASA Huntsville
June 19-20, 1968," Staff, -68

AI-68-89

"System Description and Justification - Study of a Nuclear Power System
in a Hardened Underground Missile Facility," Staff, 7-26-68

AI-68-Memo-98

"25-kwe Reactor-Thermoelectric Power System for NASA-Orbital Work-
shop Presentation Made to NASA-MSFC on August 1, 1968," Staff, 8-12-68

AI-68-Memo-124

"Selection of the Reference Core Design for the Hardened Missile Base
Power Plant," L. D. Swenson, 10-10-68

AI-68-131, Volumes I through VI

"Hardened Missile Base Nuclear Power System Study,"

Volume I, "Design Concept Evaluation," Staff, 12-13-68

Volume II, "Powerplant and Heat Sink," Staff, 12-31-68

Volume III, "U-ZrH Reactor Detailed Support Data," Staff, 12-31-68

Volume IV, "Budgetary and Planning," Staff, 12-31-68

Volume V, "Reactor Demonstration," Staff, 12-31-68

Volume VI, "Summary Evaluation," Staff, 12-31-68

AI-69-Memo-19

"SNAP Reactor Safety in the Hardened Missile Base Application," Staff,
3-19-69

AI-69-22, Volumes I and II

"Heavy-Metal-Reflected, Fast Spectrum Critical Experiments," A Proposal
to the National Aeronautics and Space Administration, Staff, 4-11-69

AI-69-Memo-27

"Marketing Plan for the Application of a Nuclear Power System to an Air
Force Hardened Missile Base," Staff, 4-4-69

AI-69-Memo-79

"SNAP Critical Facility Safety Analysis Report, Addendum for NASA
Critical Assembly Experiments," W. Heneveld, V. Swanson, and A. Thiele,
Staff, 11-24-69

AI-69-85, Volume II

"Follow-On Hardened Missile Base Nuclear Power System Study Final Report, Volume II, Appendices," Staff, 10-31-69

AI-69-85, Volume III

"Follow-On Hardened Missile Base Nuclear Power System Study Final Report, Volume III, Power System Summary Description," Staff, 10-31-69

AI-70-27, Volumes I and II

"HMB Reactor Power System Program Definition Phase,"

Volume I, "Technical Proposal," Staff, 5-20-70

Volume II, "Cost Proposal," Staff, 5-20-70

AI-70-28, Volumes I and II

"HMB Reactor Power System Nuclear Safety Evaluation,"

Volume I, "Technical Proposal," Staff, 5-20-70

Volume II, "Cost Proposal," Staff, 5-20-70

AI-70-58

"The Safe Use of NaK Liquid Metal Coolant for a Marine Undersea Reactor Power Plant (Preliminary Evaluation)," R. L. Wallerstedt, 7-30-70

AI-70-88

"Zirconium Hydride Reactor-Power Systems for Underwater Pulsed Power Applications," Staff, 12-70

AI-72-33

"Summary Report No. 2, Experimental Physics Characteristics of a Heavy-Metal-Reflected Fast-Spectrum Critical Assembly," W. H. Heneveld, R. K. Paschall, et al., 1-28-72

V. REFERENCES

(Including errata information for previously published bibliographies)

NAA-SR-TDR-8301, Revision 3, "Bibliography, SNAP Program Reports, Memos, and Motion Pictures,"
Staff (April 15, 1966)

NOTE: The following documents should be deleted from the above report:

Motion Pictures:

NAA-SR-5788
NAA-SR-7100
NAA-SR-9350

Topical and Progress Reports:

NAA-SR-6395	NAA-SR-11232
NAA-SR-8608	NAA-SR-11244
NAA-SR-9621	NAA-SR-11378
NAA-SR-11145	NAA-SR-11732
NAA-SR-11162	NAA-SR-11763

NAA-SR-TDR-8301, Supplement 1, "Bibliography, SNAP Program Reports, Memos, TDR's, and Motion Pictures - January 1, 1966-September 30, 1967," (Supplement to Revision 3)

NOTE: The following documents should be deleted from the above report:

Topical and Progress Reports:

NAA-SR-Memo-11782	NAA-SR-TDR-11874
NAA-SR-11805, Rev 1	NAA-SR-Memo-12002
NAA-SR-TDR-11811	NAA-SR-12017
NAA-SR-11813	NAA-SR-Memo-12091
NAA-SR-TDR-11816, Rev 1	NAA-SR-Memo-12369
NAA-SR-Memo-11840, Rev	NAA-SR-TDR-12531
NAA-SR-Memo-11858, Rev	NAA-SR-TDR-12532
NAA-SR-TDR-11866	

NAA-SR-Memo-11833, "Bibliography, SNAP Aerospace Safety Program,"
R. L. Detterman (April 1, 1966)

NAA-SR-TDR-11983, "Annotated Bibliography, SNAP 10A Program Technical
Data Reports," D. J. Lilja (June 30, 1966)

NAA-SR-Memo-12023, "Annotated Bibliography, SNAP 10A Program,"
D. J. Lilja (October 1, 1966)

AI-AEC-13066