INEL-95/0131
UC-510

A Document Review to Characterize Atomic International SNAP Fuels Shipped to INEL 1966–1973

D. W. Kneff
W. E. Nagel
H. Pearlman
V. J. Schaubert
S. D. Wahnschaffe, Editor
R. E. Lords, Editor

Published September 1995

Idaho National Engineering Laboratory
INEL Spent Nuclear Fuel Program
Lockheed Idaho Technologies Company
Idaho Falls, Idaho 83415

Prepared for the
U.S. Department of Energy
Assistant Secretary for Environmental Management
Under DOE Idaho Operations Office
Contract DE-AC07-94ID13223

TABLE OF CONTENTS

DISCLAIMER

DISCLAIMER

A DOCUMENT REVIEW TO CHARACTERIZE ATOMICS INTERNATIONAL SNAP FUELS SHIPPED TO INEL 1966 - 1973

EXECUTIVE SUMMARY

This report provides the results of a document search and review study to obtain information on the spent fuels for the following six Nuclear Auxiliary Power (SNAP) reactor cores now stored at the Idaho National Engineering Laboratory (INEL):

SER	SNAP-2 Experimental Reactor
S2DR (SDR)	SNAP-2 Developmental Reactor
10FS-3	SNAP-10A Ground Test Reactor
S8ER	SNAP-8 Experimental Reactor
S8DR	SNAP-8 Developmental Reactor
STR	Shield Test Reactor

The report also covers documentation on SNAP fuel materials from four in-pile materials tests: NAA-82-1, NAA-115-2, NAA-117-1, and NAA-121. Pieces of these fuel materials are also stored at INEL as part of the SNAP fuel shipments.

The following highlights provide a summary of the results of this study:

1. We have reviewed all of the original records of irradiated fuel shipments, with independent verification of mass sums by computer analysis. Correlations were made with the INEL receiving records that were available to us. We believe we have accounted for all material involved, with only minor discrepancies.

2. We have reviewed the Rocketdyne/ETEC documents that deal with fuel fabrication, quality assurance, operating experience, and post-irradiation hot cell evaluation, and have obtained characterization information for all of the SNAP fuels presently stored at INEL. The most extensive information available is for the S8ER, S8DR, and 10FS-3 cores.

3. It has not been possible to send to INEL "intelligible copies of all relevant fuel information" for above items 1 and 2. This is because of the nature of the documents (some hand-written and not readily reproducible) and the large volume of material.

4. SNAP fuels were originally intended for reprocessing, and were never intended to be an acceptable waste form for a repository. It is our opinion that at least some conditioning is necessary to make them acceptable for repository disposal. Precise criteria for such waste forms are not now available, and knowledge of the present state of the fuel is very limited. It is our recommendation that further work include sampling of the fuels to establish the range of their condition, code calculations to estimate the present inventories of fission products and transuranics (if any), and investigation of compact waste forms (such as Tailored Ceramics or Synroc) for final repository emplacement.

5. Recommendations for follow-on activities to continue the characterization of the SNAP fuels were developed during the course of the work and are included in this report.

1. INTRODUCTION

1.1 OVERVIEW

North American Aviation's Atomics International Division (AI) was under contract with the former U.S. Atomic Energy Commission for about 25 years to perform research and development work on Systems for Nuclear Auxiliary Power (SNAP). The SNAP program included both reactor and radioisotope heat sources, where the SNAP reactors were assigned even numbers and the radioisotope systems odd numbers. AI's first SNAP fuel development work was performed for radioisotope systems (Reference La51), because ^{235}U was initially considered too valuable a resource for space applications. However, with the increasing availability of fully enriched uranium, attention turned to the development of reactor systems for nuclear auxiliary power.

Atomics International, which has since merged with the Rocketdyne Division of what is now the Rockwell International Corporation, developed and operated six SNAP reactor systems under this program. The AI activities included fuel-element development and testing, fabrication of fuel elements for SNAP reactor cores, SNAP reactor operation, and the post-irradiation examination and evaluation of fuel elements and test components. The objective was to develop compact, lightweight nuclear reactor heat systems that would provide long-term auxiliary power for spacecraft. The SNAP fuel materials now stored at INEL were generated under this reactor development program.

A total of five different SNAP reactors, all typical of designs for spacecraft power systems, were built and ground-tested at AI. These five are as follows:

SER	SNAP-2 Experimental Reactor
S2DR (SDR)	SNAP-2 Developmental Reactor
10FS-3	SNAP-10A Ground Test Reactor
S8ER	SNAP-8 Experimental Reactor
S8DR	SNAP-8 Developmental Reactor

The first reactor of the SNAP design was the SNAP Experimental Reactor (SER), also called the SNAP 2 Experimental Reactor. There were many changes in program direction during SNAP development, and the SNAP 2 series was followed by the SNAP 10A series of reactors. The SNAP 8 Developmental Reactor (S8DR) was built *after* the 10A systems. One of the 10A systems, the SNAP 10A FS-3, was a complete reactor-based electric generating plant which operated for more than 10,000 hours at full power under simulated space-flight conditions. A duplicate system, the 10A FS-4, was launched in April 1965 into a nominal 700-mile polar orbit whose estimated lifetime is 3800 years. That reactor was started up by ground signal and operated entirely on passive control until it was automatically shut down, and is still in orbit. A third duplicate system, the 10A FS-5, was also fabricated but never operated, and its fuel elements were shipped to Oak Ridge National Laboratory for disposal. Other SNAP developmental reactor cores fabricated by AI included those used for the SNAPTRAN tests at INEL.

AI also built and operated a low-power research reactor, the Shield Test Reactor (STR), in support of the SNAP program. This facility utilized SNAP Experimental Reactor (SER)-type fuel rods and was used for testing reactor components. It pre-dated all but the SER and an earlier critical facility that used SNAP-type developmental fuel materials.

The SNAP program also included the conduct of many in-pile experiments in off-site research reactors to test fuel performance under controlled conditions. Those experiments were heavily instrumented, with performance measured and recorded in real time, at temperature, and under irradiation. A few of those irradiated fuel materials are also in storage at INEL, and come from the following four in-pile tests: NAA-82-1, NAA-115-2, NAA-117-1, and NAA-121. The NAA-117 fuel material has the highest burnup of the SNAP fuels stored at INEL.

All of the SNAP fuels were made by AI. This was initially a project carried out by research and development personnel, but fuel fabrication gradually evolved into a disciplined manufacturing activity. It operated with written procedures and specifications, and was monitored by a separate quality assurance (QA) organization. Thus the later fuel materials have significantly better QA documentation. Fuels from both the SNAP reactor cores and the in-pile experiments were examined in the Atomics International Hot Laboratory (AIHL) after irradiation. They were then packaged for shipment, stored temporarily at AI's Radioactive Materials Disposal Facility (RMDF), and shipped to INEL. Because the SNAP fuels were 93% enriched in ^{235}U, the intention was to reprocess them at ICPP and recover unburned ^{235}U.

This report reviews all of the Rockwell shipments of SNAP fuels to INEL, based on our review of the original shipping and accountability records retrieved from ETEC files. The report also references and reviews available documentation on the characterization of the fuels that has been retrieved from the Rocketdyne Technical Information Center, AI Technical Data Center, archived AIHL records, shipping files, and other archived personal and storage files. Coverage includes all six ground reactors and those in-pile experiments for which fuels were shipped to INEL. Some fuels were destructively examined in the AIHL, resulting in multiple fuel segments. Part of these were shipped to INEL, while others were shipped to different storage sites or disposed of as radioactive scrap.

1.2 SNAP NOMENCLATURE

Nomenclature used in AI publications on the SNAP reactor programs has not been consistent. This is due at least in part to the evolutionary nature of the programs. For example, the SNAP 2 program has also been referred to as SNAP II, and sometimes in the same document.

In a given reactor program, the chronological sequence went from "E" for experimental (such as the SNAP-2 Experimental Reactor, SER), to "D" for developmental (such as the SNAP-2 Developmental Reactor, S2DR), to "F" for flight. In the designation S2DR, the initial "S" is a contraction of SNAP and the "R" stands for Reactor. However, the designation S2DS, where the final "S" stands for System, has also been used to identify the same reactor. The total system, of course, also includes components for heat transfer, shielding, electrical power generation, instrumentation/control, and structural support. The only SNAP reactor that powered a flight system (FS) was a component of FS-4 of the SNAP 10A series. The FS-3 is the ground-tested duplicate of that reactor. Fuel elements of the ground-test reactor are called 10A FS-3 fuel, often written as 10FS-3 or S10FS-3. However, the latter substitution is not accurate, because there *was* a SNAP 10 (S10). Its design was radically different from the 10A reactors, and it was never built. A SNAP 4 reactor was also designed but not built.

The Shield Test Reactor (STR) is also designated the STF reactor, where "F" refers to the facility - the entire installation in which components were exposed to the irradiation field. This dual-designated reactor is different from the Shield Test and Irradiation Reactor (STIR), which operated at much higher power (1 MW versus 50 kW) and used MTR rather than SNAP-type fuel elements.

Although the SNAP ***reactor*** program nomenclature is not consistent, the in-pile fuel experiments are all designated by NAA-, followed by numerals. These are roughly in chronological order, although some of the experiments were irradiated in different reactors.

1.3 REFERENCES

La51 R. A. Laubenstein, "Preliminary Considerations on the Feasibility of an Auxiliary Power Plant to be Operated from Radioactive Fission Products," Atomics International Report NAA-SR-152 (October 1951) (22 pages)

2. SNAP FUELS - GENERAL BACKGROUND

All SNAP reactors utilize the same solid, homogeneous mixture of fuel (uranium, 93% ^{235}U) and moderator (hydrogen, as ZrH_x), designated $ZrUH_x$ or $UZrH_x$. The total uranium content ranges from 7 to 10.5 wt. %. There are several ways of designating the hydrogen content. N_H is the number of hydrogen atoms per cm^3 of the mixture, usually near 6 (understood to be multiplied by 10^{22}). Alternatively, the (H/Zr) ratio, H atoms to Zr atoms, is used. This is typically 1.6 to 1.7, with a corresponding hydride formula of $ZrH_{1.6}$ or $ZrH_{1.7}$, and represents about 1.6 wt.% hydrogen.

SNAP fuel elements consisted of an unsegmented, cylindrical fuel rod enclosed in a metal tube sealed by metal end caps. The STR elements operated in a water pool at 65 °F, and were adequately clad in bare (uncoated) Type 6061 aluminum alloy. All of the other SNAP reactors operated with a liquid metal (eutectic NaK, 78 wt.% potassium) coolant, flowing upward along the fuel element lengths, at temperatures of 1200 °F and above. Cladding for these reactors was Type 347 stainless steel or the nickel-based alloy Hastelloy N, where the latter was the most commonly used.

The chemical form of hydrogen in SNAP fuel is designated ZrH_x. For temperatures in the range of 1400 to 1500 °F, this material retains an N_H near that of cold water. Nevertheless, to minimize loss of hydrogen by permeation through the cladding over long operation at elevated temperatures, the entire inside cladding surface, including end caps, was coated with a glass-ceramic mixture. This composition was gradually refined so that a very low hydrogen loss rate was achieved, even under extreme conditions, provided only that the metal cladding remained intact.

It is ETEC's understanding that all of the SNAP fuels in storage at INEL have been declad. The following discussion thus deals primarily with the fuel rods themselves. Further, in this report we distinguish carefully between fuel <u>rods</u> and fuel <u>elements</u>. The fuel rods are the bare fuel material ($ZrUH_x$), and the fuel elements are the clad assemblies. Some of the post-irradiation examination work was performed on the clad assemblies, and (usually) more detailed examinations were subsequently performed on selected declad fuel materials.

SNAP reactor cores were "single use." No refueling was ever planned or performed. The entire reactor assembly, including the fuel, constituted an experiment, en route to a possible flight system. Thus the entire reactor system was disassembled following irradiation for detailed examination.

Both the fuel elements and the matching fuel rods were identified by serial numbers. The locations of those numbers on the rods and assemblies are shown on available engineering drawings, where one must take care to identify the latest design revisions.

2.1 THE FUEL ROD

2.1.1 Zirconium-Uranium Alloy

The initial zirconium-uranium alloy was produced by multiple consumable arc meltings and extrusions from reactor-grade enriched uranium metal and crystal bar zirconium. This process, as all aspects of SNAP fuel production, was rigidly controlled by written procedures and specifications, and the SNAP 10 and SNAP 8 fabrication were monitored by an independent QA organization. The Zr-U alloy composition formed is located in that part of the Zr-U equilibrium diagram that shows (at low temperatures) δ-phase UZr_2 in a matrix of Zr (Mo90).

2.1.2 Massive Hydriding

The zirconium-uranium alloy rod is converted to a hydride by diffusion of hydrogen gas at high temperatures (1600 °F peak). In this procedure the fuel volume increases by more than 15%, but dense, crack-free $ZrUH_x$ can be produced by close control of process variables, including temperature vs. time and hydrogen pressure. The resulting fuel rod retains its integrity (hence, "massive hydriding") (Reference Gi56). Because the massive hydriding is conducted isothermally, the as-built (H/Zr) ratio is essentially constant along the element length.

It was found during this work that the yield of acceptable fuel rods was greatly increased, with simultaneous improvement in microstructure, by adding 0.15 ± 0.05 weight-% of ZrC (called the "modifier") to the melt (Ra63). In AI literature, the quantity $(H/Zr)_{eff}$ is the ratio of H-to-Zr atoms calculated on the assumption that ZrC does not hydride (Ar74). However, a later study showed that the ZrC tends to form the compound $ZrC \cdot ZrH_y$, where y can approach 2 (Ta65). No correction was ever made for this contingency in calculating $(H/Zr)_{eff}$. The hydrogen dissociation pressures in fuel made with the ZrC addition were found to be lower than those in the initial composition (Jo64).

2.1.3 Fuel Microstructure (Ra63)

All SNAP fuels consist of a dispersion of uranium-rich particles and modifier particles in a matrix of substoichiometric ZrH_2, which is the terminal composition in the Zr-H equilibrium diagram (Mo68a). On an atomic basis, hydrogen constitutes 66-2/3 % of this compound. In the massive hydriding of ZrU alloy, the uranium is forced out of combination with Zr, while the hydrogen atoms constitute between 30% and 50% of the Zr-H mixture. The uranium does not hydride, but remains in micron-size particles in the ZrH_x matrix (Ra65, Pe66, Mo68a, Mo68b).

As the H/Zr ratio increases during the hydriding process, the resulting ZrH_x undergoes a series of phase changes, culminating in the face-centered tetragonal ε-phase with its characteristic banded microstructure. These phase changes are accompanied by volume changes, which must be taken into account in conducting the massive hydriding process. Also, they enter into the changes produced by reactor operation.

2.2 BARRIERS TO HYDROGEN PERMEATION (Ro61, Bu64)

Retention of hydrogen in the fuel rods was greatly improved by lining the inside of the cladding, including end caps, with a silicate composition. Like other constituents of the SNAP fuel, the composition was changed incrementally, based on testing, to minimize chemical interactions with the hydrogen and to reduce neutron absorption.

The barrier material was applied in separate layers. A burnable "poison," Sm_2O_3, was incorporated in the final layer. Ideally, the fuel rod would "see" only the Sm_2O_3-bearing layer and the gas atmosphere used during the element assembly procedure.

2.3 FUEL ELEMENT ASSEMBLY AND TESTING

The fuel element tubular cladding and one end closure were welded before the internal hydrogen barrier layers were applied. The fuel rod was then inserted, and the coating on the cladding was fused to the pre-coated closure at the other end prior to welding on the end cap to seal the element. One sealing procedure minimized disruption of the coating by using a two-piece end closure arrangement. A coated "cup plug" was inserted first, and "blended" at high temperature to the coating on the inside of the cladding. A separate uncoated end cap was then welded to the cladding, along an uncoated area. Appropriate temperature controls were used to minimize deleterious effects to the "blended" coating. This end of the fuel element was inserted into the lower grid of the reactor core tank, near the coolant entrance, where its temperature was lowest.

Every element was tested after fabrication to determine whether the hydrogen permeation rate met the specified permissible maximum. There were many additional acceptance tests, including exposure to the shock and vibration loads expected from the launch and orbital operation of the flight systems. The elements in general proved to be remarkably unaffected by any of these mechanical loads. Thermophysical properties of the unirradiated elements were measured to provide baseline values (Na67) for comparison with post-irradiation measurements.

2.4 SNAP FUEL QUALITY SYSTEMS

SNAP fuel was produced during the late 1950's through 1967. During this period, the sophisticated modern total *quality assurance* program documents, such as NQA-1, RDT F2-2, and the ISO 9000 series, were not in existence. Total quality assurance programs cover quality provisions for management, development, design, procurement, manufacture, test and inspection, construction, maintenance and modification, and operation. The most comprehensive quality system requirement available at the time of SNAP fuel production was defined in military specification MIL-Q-9858A. This document provided requirements for, and was concerned exclusively with, procurement, manufacture, and inspection of deliverable production hardware. Its approach can best be termed *quality control.* This specification was never a requirement of the SNAP reactor development contract between the AEC and Atomics International. However, AI did at this time have in place a quality system that met MIL-Q-9858A. Starting in 1962, this military-developed system was followed for the fabrication of SNAP 10A and SNAP 8 fuels by organizationally divorced fabrication and quality control personnel. Fuel for the SER, S2DR, and STF reactors, as well as irradiation experiments, were fabricated, tested, and inspected by Engineering Department development laboratory personnel, under a set of Engineering Management Procedures established by the Engineering Department vice president. All reactor operations and post-irradiation examination of SNAP fuels were controlled by these Engineering Management Procedures. Manufacturing, inspection, and test planning for SNAP 10A and SNAP 8 fuel from procurement through delivery for reactor use are given in various reports, as referenced in individual reactor sections of this report, along with detailed post-irradiation test and inspection results. Similar data in less detail are available for some reactor fuels manufactured prior to SNAP10A/8 production.

2.5 RADIATION EFFECTS

Although post-irradiation examination of the fuel elements from the lower-power reactors (particularly 10A FS-3) showed that they remained intact, the S8ER and S8DR fuel elements showed extensive failures (Pe66, Mi67a). These failures included cladding cracks, fuel rod deformation, and fuel rod cracks. Lessons learned (and applied) from S8ER were not adequate to eliminate failures in S8DR. Termination of the development program in 1976 prevented the testing of additional improvements.

The most extensive post-irradiation studies on core elements were for S8ER and S8DR, to identify the underlying causes of fuel element failure. A summary of irradiation phenomena in S8ER fuel (Pe66, Section VII; too lengthy to include here) describes the microstructure of the as-fabricated fuel, plus the processes at the atomistic level during reactor operation which could affect the overall element performance. Although specific to high-power S8ER operation, the analysis is generally applicable to all SNAP reactor fuels.

2.5.1 Reactor Operation

Because of the axial and radial temperature gradients in the operating reactors (particularly axially along the coolant flow direction), the initial uniform (H/Zr) profiles in the fuel elements must change. Each element, depending on its location in the core, is a closed, individual system. It is not, however, in equilibrium, because of the temperature gradient of the reactor and the fact that there can be only one steady-state hydrogen pressure inside the element. The (H/Zr) ratio along the fuel length, starting at the coolant entrance (lower grid), adjusts to provide a single, steady-state hydrogen partial pressure corresponding to the (H/Zr) ratio and to the temperature at the given location. These variations in (H/Zr) are reflected in changes in fuel rod density and volume. As an example, the temperature profile for the SER is indicated in Ref. Fe59. In the later SNAP reactors, both axial and diametrical clearance was provided to allow for fuel rod volume changes under irradiation.

The effect of reactor operation is illustrated by the changes in (H/Zr) in the 10A FS-3 fuel elements, after over 10,000 hours of operation with reactor outlet temperatures up to 1055 °F, as shown in the following table. As expected, the bottoms of the fuel rods, at the cooler location, had a higher $(H/Zr)_{eff}$ than the as-fabricated values. The tops had lower values. Overall, the 37 fuel rods in 10A FS-3 showed very good hydrogen retention.

SNAP 10A FS-3 Hydrogen Analysis Summary

Fuel Rod	As-Fabricated $(H/Zr)_{eff}$	Post-Irradiation $(H/Zr)_{eff}$ (Range from Core Top to Bottom)
422	1.840	1.754 to 1.862
331	1.830	1.760 to 1.860
529	1.770	1.680 to 1.800

The 10A FS-3 reactor operated at a fairly steady power level (37 to 38 kW). Other reactors experienced large, abrupt power level changes, deliberately induced to determine the reactor physics (i.e., the control characteristics). Such changes imposed rapid temperature transients which were often repeated, to the potential detriment of fuel element integrity.

The SNAP 2/10A reactors operated at a mean neutron fission energy near 0.18 eV (Go67c, Table 3.1), while S8DR operated at about 0.26 eV (Sw69). These are slightly epithermal. As a result, the radial neutron distribution does not resemble that of some water or graphite moderated cores, where the thermal flux peaks in the moderator and is depressed in the fuel. The flux is fairly flat across a fuel element, resulting in a rather flat radial burnup profile across each element. Radially across the whole core, the flux and burnup are highest at the center element, and lowest at the outside elements near the edge of the cylindrical core. The flux profile in the axial direction is not symmetrical. It is higher below midplane, because of the higher (H/Zr) near the bottom ends of the fuel.

2.5.2 Structural Effects of Fission

Microstructural effects of the fission products and neutrons appeared to be relatively slight. Although the fuel material is brittle, it is strong. The ultimate tensile strength increases with temperature, to a value for unirradiated Zr-10$UH_{1.74}$ near 10,000 psi at about 1200 °F (Lu64). Swelling due to accumulated fission products was observed (Mi67a), which showed the small (micron) size of the uranium particles.

The volume changes in uranium due to its phase changes over the SNAP reactor operational temperature range, starting at room temperature, did not appear significant. This may be because of the small size of the particles or the "strong box" effect provided by the ZrH_x, or both. Empirical relations were developed among fuel growth, temperature, and burnup, although a phenomenological basis was proposed (Li73).

Macro effects were a different story. Local overheating of fuel (by coolant flow starvation) resulted in drastic deformation, from the combined effects of dehydriding (volume/density changes) and the higher fission density in the adjacent higher (H/Zr) regions. Fuel rod bowing stressed the cladding, resulting in cladding cracking (Da70).

2.5.3 Burnup and Transmutation (Pe66)

Because the principal interest in the SNAP reactor program was on the overall system operation, relatively little emphasis was placed on achieving high burnup. It was rarely as high as 0.1 metal-atom %. The highest burnup (at least for fuels stored at INEL) was 0.77 metal-atom % for the NAA-117 in-pile experiment.

Profiles of gamma activity along the fuel rod length were determined by gamma scanning in the AIHL, and were then calibrated by measurements made on fuel samples cut from selected locations. The samples were analyzed in several ways, of which mass spectroscopy was deemed the most accurate (Go67b).

Neutron capture did occur in the uranium. However, with only about 7 wt.% ^{238}U, capture without fission was relatively slight. In S8DR the core average burnup and transmutation (plus data for each ring of fuel elements) were estimated from ring power calculations made using the computer code ANISN (Ro70). About 4.79 grams of ^{239}Pu were estimated to have been formed, from the 521 grams of ^{238}U initially present in the whole core. In these calculated results, the burnups of ^{236}U and ^{239}Pu were assumed negligible. The calculated ^{239}Pu formation is equivalent to 22.7 mg of ^{239}Pu per fuel element. No measurements were made to confirm these calculations.

2.6 CONSIDERATIONS OF ENVIRONMENTAL EFFECTS ON FUEL

The as-built SNAP fuel was in limited mechanical contact with the silicate-coated inside surface of the cladding, through an atmosphere which was variously helium-air, low-pressure helium, or vacuum (Li73). Experience (Go67b) showed that *if* the cladding remained intact, reactor operation would produce only modest fuel growth, so that the fuel would attain its design life. After shutdown, the reactor fuel would show normal decay of fuel element radioactivity. Figure 1 shows the estimated decay of the SNAP 10A FS-4 fuel elements with time (Sa66). FS-4, also designated SNAPSHOT-1, operated at 40.7 kW thermal power for 1032 hours before shutdown. The general character of the curve is valid for all SNAP fuel.

Cracked cladding allows the NaK coolant to come into direct contact with the interior of the fuel element. There is little (if any) direct effect of NaK on the fuel material. Loss of the coating at a cracked area would result in a locally higher loss rate of hydrogen. Chemical reactions did occur between the NaK and

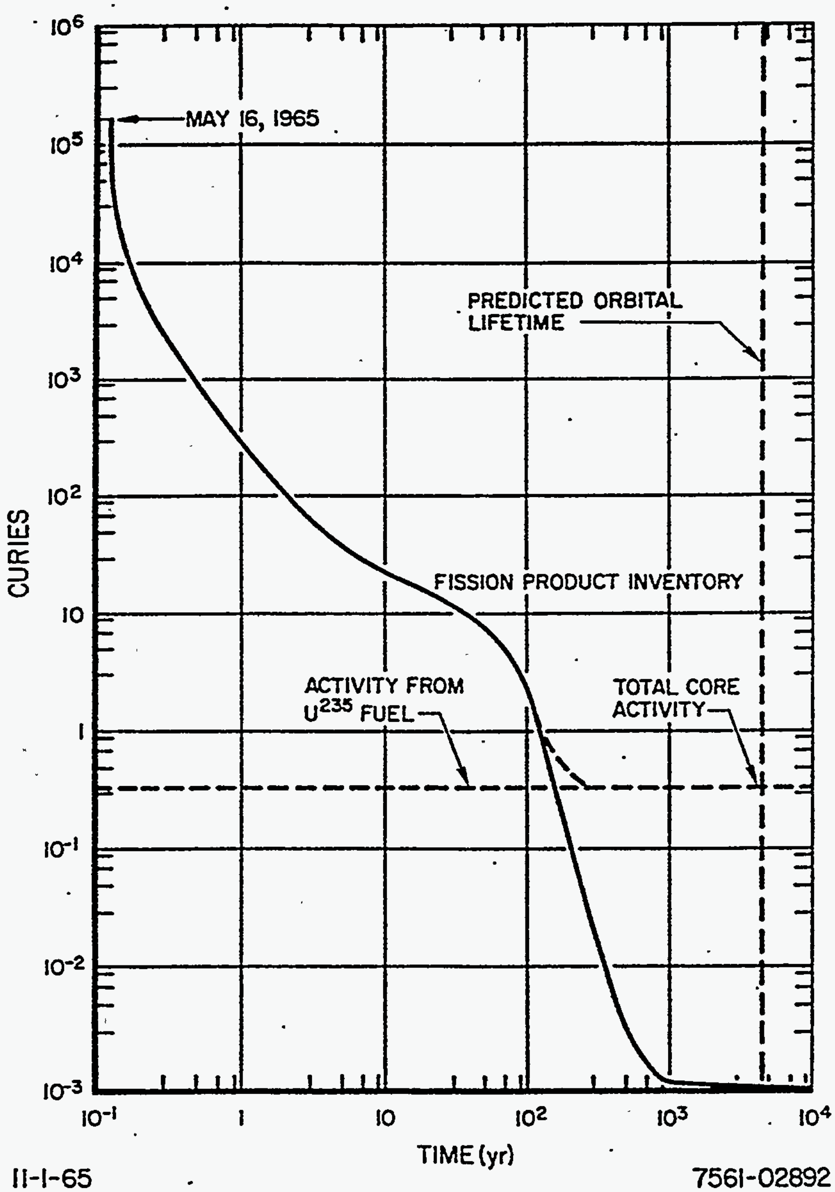

Figure 1. Calculated Fission Product Inventory Decay for SNAPSHOT-1 (from Sa66).

the silicate coating. Although this has not been explored systematically, it is evident from post-irradiation examinations that whole pieces of coating were removed from the cladding, and some pieces were stuck to the fuel (Mi66a). Whether the reaction products themselves attacked the $ZrUH_x$ fuel was not investigated.

The ground test reactors (excluding STR) were moved from the test buildings after they completed their tests to the AIHL for disassembly and examination. The initial AIHL atmosphere was nitrogen gas, containing less than 3% oxygen, to minimize chemical reactions with residual NaK. Fuel elements were cleaned in a butyl ether-alcohol liquid compound (Dowanol EB, Dow Chemical Company) to remove residual NaK. With cracked cladding, and sometimes cracked fuel, it could not be certain that all crevices were cleaned out. Nevertheless, fuel elements were handled, stored, and shipped in air following Dowanol cleaning. Communications between Aerojet Nuclear (Idaho) and Atomics International indicate that some elements retained chemically active NaK (Ad74).

Based on information received from INEL, it is our understanding that the declad fuel was (at least initially) stored in water containing 3.8 g/liter natural boron for criticality control (Am68). No studies were ever made at AI of the possible chemical interactions between $ZrUH_x$ and water or such a solution.

2.7 FUEL REFERENCES

Ad74 T. Adamson (Aerojet Nuclear), in *Atomics International Telephone Conversation Record* to V.J. Schaubert (AI) dated 28 January 1974

Am68 C. B. Amberson (Idaho Nuclear Corporation), "SNAP Fuels from A.I. in 1967-1968 Fai-9-68," Interoffice Correspondence

Ar74 J. R. Armstrong, "Uranium-Zirconium Fuel Alloy, Hydriding of," Atomics International Specification NA0611-001 (January 1974) (11 pages)

Bu64 G. F. Burdi, Ed., "SNAP Technology Handbook. Volume II. Hybrid Fuels and Claddings," Atomics International Report NAA-SR-8617, Volume II (November 1964) (129 pages)

Da70a W. H. Dauterman, "Bowing of S8DR Fuel," Atomics International Technical Information Document TI-568-24-042 (March 1970) (20 pages)

Fe59 L. D. Felten, "Axial Temperature Distribution of S.E.R. Central Fuel Element," Atomics International Technical Data Record NAA-SR-MEMO-5121 (October 1959) (8 pages)

Gi56 P. T. Gilbert, Jr., "Zirconium Hydride: A Compendium on the Systems Zirconium-Hydrogen and Hafnium-Hydrogen and Related Topics," Atomics International Report NAA-SR-1508 (October 1956) (834 pages)

Go67b T. A. Golding, "Post-Irradiation Examination of Selected S10FS-3 Fuel Elements," Atomics International Technical Data Record NAA-SR-MEMO-12341 (June 1967) (28 pages)

Go67c K. G. Golliher, et al., "SNAP Engineering Test Facility Safety Analysis Report: Addendum for Intrinsic Supercriticality Experiments," Atomics International Technical Data Record NAA-SR-MEMO-12614 (November 1967) (80 pages)

Jo64 H. E. Johnson, "Hydrogen Dissociation Pressures of Modified SNAP Fuels," Atomics International Report NAA-SR-9295 (March 1964) (47 pages)

Li73 A. F. Lillie, D. T. McClelland, W. J. Roberts, and J. H. Walter, "Zirconium Hydride Fuel Element Performance Characteristics," Atomics International Report AI-AEC-13084 (June 1973) (75 pages)

Lu64 L. B. Lundberg, "Parametric Studies on the Short-Term Tensile Mechanical Properties of Zr-10U Alloy Hydrides," Atomics International Report NAA-SR-9875 (July 1964) (42 pages)

Mi66a K. J. Miller, "Post-Irradiation Detailed Examination of Selected S8ER Fuel Elements," Atomics International Technical Data Record NAA-SR-MEMO-12165 (1966) (complete color copy) (304 pages)

Mi67a K. J. Miller, "Electron Microscopy Analysis of Irradiated SNAP 8 Fuel," Atomics International Report NAA-SR-12449 (August 1967) (66 pages)

Mo68a K. E. Moore, W. A. Young, M. M. Nakata, and C. A. Smith, "Phase Relations at High Hydrogen Contents in the SNAP Fuel System," Atomics International Report NAA-SR-12587 (February 1968) (55 pages)

Mo68b K. E. Moore and M. M. Nakata, "Phase Relationships in the ($\alpha + \delta$) Region of the Zr-H System," Atomics International Report AI-AEC-12703 (September 1968) (47 pages)

Mo90 W. G. Moffett, Ed., *The Handbook of Binary Phase Diagrams*, Vol. V (Genium Publishing Co., Schenectady, NY, 1990 update)

Na67 M. M. Nakata, C. J. Ambrose, and R. A. Finch, "Thermophysical Properties of SNAP Fuel Materials," Atomics International Report NAA-SR-11959 (September 1967) (93 pages)

Pe66 H. Pearlman, et al., "SNAP 8 Experimental Reactor Fuel Element Behavior: Atomics International Task Force Review," Atomics International Report NAA-SR-MEMO-12210 (November 1966) (73 pages)

Ra63 J. W. Raymond, "Development of a High Yield 90 Zr - 10 U Alloy Massive Hydride," Atomics International Report NAA-SR-7305 (January 1963) (22 pages)

Ra65 J. W. Raymond, "Phase Relationships and Microstructures in the Zirconium - 10 wt % Uranium Alloy Hydride System," Atomics International Report NAA-SR-10965 (November 1965) (36 pages)

Ro61* C. J. Romero and S. Elchyshyn, "Cladding Development for SER Fuel Elements," Atomics International Report NAA-SR-4831 (February 1961) (42 pages)

Ro70 H. Rood, "S8DR Fuel Burnup and Transmutation Data," Atomics International Technical Information Document TI-568-24-063 (July 1970) (5 pages)

Sa66 W. B. Sayer and R. S. Hart, "The Predicted Fission Product Decay of Snapshot-1," Atomics International Report NAA-SR-11642 (January 1966) (20 pages)

Sw69* L. D. Swenson, "SNAP 8 Development Reactor Nuclear Analysis," Atomics International Report AI-AEC-12864 (October 1969) (83 pages)

Ta65 H. Taketani, L. Silverman, and W. L. Korst, "Identification of the Modifier in SNAP Fuel," Atomics International Report NAA-SR-10174 (February 1965) (21 pages)

(Reports identified with an asterisk were referenced in the WINCO SNAP summary report WINCO-1222 by R. E. Lords.)

3. ISSUES FOR THE SAFE DISPOSAL OF STORED SNAP FUELS

SNAP fuels were never intended as waste forms for repository emplacement. The safety issues involved in space flight were studied extensively (De65, Cu65, Go67c). The emphasis was on intrinsic reactor safety, and on safe return of the reactor from orbit. Other than investigations of nuclear criticality when immersed in water (should the reactor land in the sea), only one study is known to have been made on the chemical reaction of water with SNAP fuel.

3.1 SNAP FUELS IN THE STORAGE ENVIRONMENT

As indicated in Ref. Am68 and more recently in discussions with INEL personnel, the irradiated declad fuel rods were stored at INEL in aluminum cans that have been breached by water. Although the radioactivity of the fuel has decayed substantially, there may still be a possibility of radiolysis of the water, which would produce hydrogen gas and solute species.

The choice of aluminum for the canister raises the possibility of galvanic action. A survey of potential electrode reactions involving aluminum and uranium when linked by an electrolyte shows that aluminum would be anodic to the uranium. It would thus corrode sacrificially, protecting the uranium (We85). Dissolution of aluminum would change the pH of the water, with effects on the uranium that have not been investigated.

In the event that the canisters go dry, fuel reactions with air are possible, especially if the fuel material is fragmented. The large surface-to-volume ratio favors a reaction, which can progress rapidly enough to cause a fire. The personal experience of one of us (hp) with a barrel of unalloyed zirconium turnings, initially covered with water, illustrated this vividly: corrosion of a barrel (not observed until after the fact) exposed high surface-to-volume-ratio zirconium turnings to air and produced a fire. Under some circumstances zirconium hydride powder can explode in air (Gi56).

These potential reactions also probably hold true for $ZrUH_x$, because the particulate uranium is not stable either in air or in water. All of the potential reactions with air and water could be slowed substantially by the presence of an oxide layer on the surface of the fuel piece (or particle). However, stability of such a layer over an indefinitely long period of time cannot be assured.

3.2 RELEVANT AI EXPERIENCE

Chemical effects of water or air on irradiated $ZrUH_x$ fuel were not systematically investigated in the AI program. As part of the fuel development program for the SNAP 4 system, a very limited investigation was made of corrosion effects on bare fuel, in contact with Type 304 and Type 446 steels. The SNAP 4 system was based on a boiling water reactor that used SNAP-type fuel clad in austenitic stainless steel, such as Type 304. In another program, the oxidation behavior of the bare fuel alloy was studied.

3.2.1 Compatibility Tests for SNAP 4 (St63)

Samples of unirradiated Zr-15U-H_x, "sandwiched" between flat plates of Type 304 and Type 446 steels, were exposed in water in both pyrex and Type 304 containers. They were held in an autoclave at 570 °F and 1250 psi for 400 hours. Cracking of the fuel was observed in the pyrex container samples. This was attributed to attack by water on the pyrex glass, resulting in increased pH from 7 to about 10. Fuel samples in the Type 304 container were unaffected. However, the Type 446 plates that formed part of the "sandwich" sample were attacked by the water. The pH of the water in that container decreased from 7 to about 5.2. There is no statement in Ref. St63 about the hydrogen content of the fuel sample, before or after exposure. However, Ref. Bu64 (Table 1.1.6) identifies the fuel composition as Zr-15U-$H_{1.6}$.

It is difficult to extrapolate these results to irradiated SNAP fuel stored in water under less aggressive conditions. There is some implication that water at a pH of 7 or below will not attack $ZrUH_x$, but the influence of other metals linked to the fuel material via an electrolytic solution is not clear.

3.2.2 Oxidation Kinetics in the Zr-U System

A study was made of the reaction of the unirradiated, unhydrided fuel alloy with air and with oxygen gas (Ha65) in order to provide basic data on the fate of SNAP fuel re-entering the atmosphere from earth orbit. Samples of Zr-10U (with and without modifier) and of pure zirconium metal were exposed over a temperature range from 570 °C to 1190 °C. Parabolic and cubic rate laws were found to fit the data for weight gain versus time, up to "breakaway." The samples were dense, 1-inch-diameter spheres.

At the high test temperatures, hydrided fuel would certainly lose hydrogen, leading to changes in density and sample cracking. The increased surface area (over the intact sphere) would accelerate oxidation. The effect of irradiation on the test results was not investigated.

3.2.3 Dehydriding

Samples of irradiated SNAP fuels were dehydrided in the process of studying fission gas retention (Ce64) and in experiments aimed solely at the dehydriding process (Le66). High temperatures (2000 °F to 3200 °F) and times from 3 to 8 hours were used to evaluate the release dependence on these variables (Ce64).

3.2.4 Conclusions

Data available from the limited AI experiments on the reactions between unirradiated fuel alloy or hydrided Zr-U are inadequate to predict irradiated fuel behavior for long-term storage in water or air. Further, it is not evident that dehydriding would be a worthwhile process to improve the storage characteristics of irradiated SNAP fuel. Alternative methods for treating the fuel are suggested in Section 5.

3.3 REFERENCES

Bu64 G. F. Burdi, Ed., "SNAP Technology Handbook. Volume II. Hybrid Fuels and Claddings," Atomics International Report NAA-SR-8617, Volume II (November 1964) (129 pages)

Ce64 W. M. Cegelski, "Fission Product Release from Irradiated Uranium-Zirconium Hydride SNAP Fuel," Atomics International Report NAA-SR-9673 (July 1964) (37 pages)

Cu65 G. E. Culley, "Analysis of SNAPTRAN-3 Destructive Experiment Debris," Atomics International Report NAA-SR-11584 (December 1965) (37 pages)

De65 R. L. Detterman, "Progress Report: SNAP Aerospace Safety Program April-June 1965," Atomics International Report NAA-SR-11497 (August 1965) (one of a series, 39 pages)

Gi56 P. T. Gilbert, Jr., "Zirconium Hydride: A Compendium on the Systems Zirconium-Hydrogen and Hafnium-Hydrogen and Related Topics," Atomics International Report NAA-SR-1508 (October 1956) (834 pages)

Go67c K. G. Golliher, et al., "SNAP Engineering Test Facility Safety Analysis Report: Addendum for Intrinsic Supercriticality Experiments," Atomics International Technical Data Record NAA-SR-MEMO-12614 (1967) (80 pages)

Ha65 S. D. Harkness, "The Oxidation Kinetics of SNAP Fuel Alloy: Modified 90-10 Zirconium-Uranium," Atomics International Technical Data Record NAA-SR-MEMO-10914 (January 1965) (33 pages)

Le66 S. K. Lee, "A Study of Dehydriding of S8ER Fuel," Atomics International Technical Data Record NAA-SR-MEMO-12197 (October 1966) (20 pages)

St63 D. H. Stone, "Corrosion Tests of Candidate SNAP 4 Fuel Element Materials," Atomics International Technical Data Record NAA-SR-TDR-8068 (February 1963) (6 pages)

We85 R. C. Weast, Ed., *CRC Handbook of Chemistry and Physics*, 66th Edition (CRC Press Inc., Boca Raton, FL, 1985)

4. DATA-GATHERING APPROACH AND REPORT CONTENTS

The approach used in this task was to identify and examine sources of SNAP information at ETEC and Rocketdyne, and to assemble and review the fuel-related information for its relevance to INEL needs. Those sources investigated included the Rocketdyne Technical Information Center (TIC, which incorporates the former Atomics International library resources), the ETEC library, Rocketdyne's Advanced Power Systems (formerly Atomics International) Technical Data Center, archived Atomics International shipping records, archived Atomics International Hot Laboratory data records, and other available archived personal and storage files.

The Rocketdyne TIC and ETEC library were the source of a large number of formal reports and technical data records, while the Technical Data Center was the source of technical information documents, engineering drawings, and specifications (typically on microfiche or microfilm). The archived shipping files provided detailed shipping information over the time period of SNAP fuel transfers to INEL and other disposal locations. The AIHL and miscellaneous personal files provided a large volume of source data (data sheets, strip chart recordings, internal letters, photographs, negatives, examination plans and proposals, etc.), particularly for 10FS-3, S8ER, and S8DR. We uncovered several file cabinet drawers full of unsorted photomicrograph negatives for the pre- and post-irradiation examination of fuel elements and fuel rods.

This report covers the SNAP documentation we have assembled, reviewed, and deem to have relevance to the characterization of the SNAP fuel rods now in storage at INEL. It is divided into separate sections according to reactor core and in-pile experiment, where each section provides the following information:

(1) A general description of each system.

(2) A listing of the documentation found for each item of interest, as outlined in the INEL Memo Purchase Order. This is presented as a reference number for each document, followed by a brief description of the document contents. Areas of INEL interest where no information was found were left blank, and serve to identify those areas.

(3) A complete reference list for the documentation assembled for that SNAP core or experiment. The reference code used in this report identifies each document by using the first two characters of the first author's last name plus the last two digits of the year of publication. Where there are duplications, the characters a,b,c etc. are added to distinguish documents. Those documents referenced in the WINCO report "SNAP and AI Fuel Summary Report" by R. E. Lords (WINCO-1222, August 1994) are identified in the bibliographies by an "*" following the reference code.

(4) A spreadsheet summarizing the shipping records for that fuel element type, correlated both with available INEL receiving records and with our source documentation on fuel fabrication.

In addition, Appendix 1 provides a combined listing of all the reference documents covered in this report. The formal documentation summarized in Appendix 1 totals well over 10,000 pages. It does not include several file cabinets full of post-irradiation examination source data and negatives which could not be readily indexed.

A large volume of additional SNAP-related information was also uncovered during this investigation that is not referenced in this report. That information includes a large number of development reports covering preliminary work that did not necessarily represent the final reactor systems, several qualification testing reports (including thermal cycling, simulation-launch vibration tests, etc.), as those tests were perfomed using natural uranium, reports on other (non-fuel) components of the reactor systems, and documentation on the SNAPTRAN cores and SNAP Critical Assembly (SCA), which are outside the scope of this investigation.

5. RECOMMENDATIONS FOR FUTURE WORK

The document search and review activities performed as part of the present work led to the acquisition of more source documentation than could be reviewed and evaluated as part of this program. Additionally, we have developed views on the ultimate disposition of the SNAP fuel materials now stored at INEL. Those fuel materials contain about 26.3 kg of uranium, of which about 24.4 kg is ^{235}U. We offer the following recommendations for consideration in future work leading to the final disposition of the fuels.

5.1 COMPLETION OF DOCUMENT REVIEW AND EVALUATION

The source documents relating to the SNAP fuels (AIHL photographs, log books, and other analyses of various kinds) are all, as of the date of this report, well over 20 years old. This material was found scattered in several storage locations, and provided more information than time and funding limitations permitted us to review and evaluate. In the interest of providing a more complete account of the fuel used in this now inactive program, we recommend an extension to the Memorandum Purchase Order effort of 4 - 6 man-months at ETEC/Rocketdyne to complete this evaluation.

5.2 COMPUTER CODE CALCULATIONS

Some of the key post-irradiation fuel information, such as transuranic and fission product inventories, were not measured at the time the fuels were examined, and computer codes for predicting this information were relatively primitive. (See, for example, Sa66.) Present-day, more sophisticated computer codes such as ORIGEN2 can be applied to the available data to provide this information. We recommend such calculations, and estimate that such an effort at ETEC would require about 3 man-months.

5.3 ASSESSMENT OF PRESENT CONDITION OF STORED FUEL

Because of the potential chemical interaction of the fuels with water, and mechanical shocks of handling, we strongly recommend examination of representative samples of the fuel. This should include the determination of microstructure, physical state (e.g., intact or crumbled), and overall chemical condition (adherent or flaky oxide, etc.). We recommend beginning with samples of fuel material with the highest burnup (0.77 metal-atom %, from Experiment NAA-117-1) and with wafers cut from S8DR fuel.

The main effort for such an examination must take place at INEL. However, ETEC/Rocketdyne can provide guidance for hot cell examination, based on our extensive documentation of past SNAP fuel analyses (e.g., Sh64, Ra65, Mc65, Ra65a, and the many detailed AI post-irradiation examinations, such as Mi66a and Mi67), and has personnel available who are familiar with in-cell metallography. We estimate a 12-man-month effort at INEL and a 2-man-month support effort at ETEC/Rocketdyne to perform this examination.

5.4 PROCESSES TO CONVERT SNAP FUELS TO ACCEPTABLE WASTE FORMS

In view of the present DOE-ordered shutdown of spent nuclear fuel reprocessing (Be94), there are only three alternatives for handling SNAP fuels. The first is simply to leave them where they are in INEL storage pools, the second is to place them in a repository in their present form, and the third is to perform some type of chemical conditioning before repository emplacement. The first alternative is obviously unacceptable because it does not provide a long-term storage solution.

Past AI determinations of the microstructure of irradiated SNAP fuels provide information that bears directly on the second alternative. These investigations have established that the uranium in the ZrH_x matrix is present in small (micron-size) particulate metallic form (Pe66, Mi67). The chemical reactivity of both uranium and zirconium hydride is potentially high, both in air and in water. The current status of the stored SNAP fuels suggests that surface layers, probably oxide, may be present that limit or suppress chemical reactions. Dependence on the integrity of these layers for the indefinite future poses a risk, and thus raises questions on repository disposal of the SNAP fuels in their present form.

The third alternative, chemical conditioning of the fuels before repository disposal, is the most viable option. Glass is the composition of choice for immobilizing high level waste (HLW) in France, and glass technology is further advanced than alternatives at present in the U.S. However, the feed for glass-forming in France (and in the U.S.) is the waste stream from chemical reprocessing of spent nuclear fuel (SNF) - not the SNF itself. More compact waste forms like Synroc or Tailored Ceramics have the potential to support a higher loading of HLW than glass. This would be required for the emplacement of SNF materials like SNAP and TRIGA fuel in a future repository.

These compact waste forms have been investigated in the past to "lock" radioactive elements into stable chemical structures for final disposal. Tailored Ceramics were studied intensely for several years at Rockwell International (Gr83, Mc83). This waste form was intended primarily for a Purex waste stream, but may be applicable to small-batch, special fuel compositions like SNAP. We recommend a study of the applicability of this type of chemical conditioning for the SNAP fuels, and have personnel available at ETEC/Rocketdyne who participated in the Tailored Ceramics program. We suggest a 9-man-month study at ETEC of applicable chemical conditioning processes to produce stable waste forms from SNAP fuels, with 2-man-months of support from INEL personnel.

A recent INEL listing of stored spent fuels (Fi94) includes both SNAP and TRIGA fuel types. Generically, TRIGA fuel is also hydrided Zr-U alloy, usually with a lower hydrogen content than SNAP fuel. (See, for example, Si61.) It is possible that TRIGA fuel would be amenable to chemical treatment analogous to SNAP fuel, providing an additional benefit from this proposed study.

5.5 REFERENCES

Be94 C. L. Bendixsen, in *Proc. 1994 International High Level Radioactive Waste Management Conference*, American Nuclear Society, 22-26 May 1994

Fi94 D. L. Fillmore, "Categorization of Department of Energy Spent Nuclear Fuel," in *Proc. 1994 International High Level Radioactive Waste Management Conference*, American Nuclear Society, 22-26 May 1994

Gr83 L. F. Grantham, et al., "Process Description and Plant Design for Preparing Ceramic High Level Waste Forms," Rockwell International Energy Systems Group Report ESG-DOE-13397 (February 1983) (71 pages)

Mc65 D. D. McAfee, "Hot Laboratory Metallography of SNAP 8 Fuel," Atomics International Technical Data Record NAA-SR-MEMO-11217 (March 1965) (12 pages)

Mc83 R. L. McKisson, et al., "Commercial High Level Waste Management, Options and Economics: A Comparative Analysis of the Ceramic and Glass Waste Forms," Rockwell International Energy Systems Group Report ESG-DOE-13391 (February 1983) (145 pages)

Mi66a K. J. Miller, "Post-Irradiation Detailed Examination of Selected S8ER Fuel Elements," Atomics International Technical Data Record NAA-SR-MEMO-12165 (1966) (complete color copy) (304 pages)

Mi67 K. J. Miller, "Electron Microscopy Analysis of Irradiated S8ER and NAA 115-1 Fuel," Atomics International Technical Data Record NAA-SR-MEMO-12368 (March 1967) (51 pages)

Pe66 H. Pearlman, et al., "SNAP 8 Experimental Reactor Fuel Element Behavior: Atomics International Task Force Review," Atomics International Report NAA-SR-MEMO-12210 (November 1966) (73 pages)

Ra65 J. W. Raymond, "Phase Relationships and Microstructures in the Zirconium - 10 wt % Uranium Alloy Hydride System," Atomics International Report NAA-SR-10965 (November 1965) (36 pages)

Ra65a J. W. Raymond and D. T. Shoop, "The Metallography of Zirconium-Base Alloy Hydrides," Atomics International Technical Data Record NAA-SR-TDR-10927 (February 1965) (9 pages)

Sa66 W. B. Sayer and R. S. Hart, "The Predicted Fission Product Decay of Snapshot-1," Atomics International Report NAA-SR-11642 (January 1966) (20 pages)

Sh64 D. T. Shoop, "Metallographic Preparation of Zirconium Hydride," Atomics International Technical Data Record NAA-SR-MEMO-10145 (June 1964) (7 pages)

Si61 M. T. Simnad and W. P. Wallace, "The Metallurgy of the TRIGA Fuel Elements," *Elecktrotechnik und Maschinenbau 78*, 581 (1961)

SNAP REACTOR FUELS

6. SNAP-2 EXPERIMENTAL REACTOR (SER)

Date went critical:	September 19, 1959 (Hu61, Ja73)
First power operation:	November 5, 1959 (Ja73)
Thermal power:	50 kWt
Thermal energy:	225,000 kWt-h
Time at power and temperature:	1900 h at 1200°F, 3300 h above 900°F
Operating location:	SSFL Building 010

6.1 SER GENERAL DESCRIPTION

The SER was the first of the series of reactors in the SNAP program (Di67). The reactor core was made up of 61 fuel-moderator elements, 1 inch in diameter, on a triangular matrix contained in a 9-inch-diameter core vessel. Heat was removed from the core by primary NaK coolant flow, transferred to a secondary NaK coolant loop by means of an intermediate heat exchanger, and dissipated to the atmosphere through an airblast heat exchanger. The fuel was composed of a hydride uranium-zirconium alloy that was formed into rods and canned in thin-walled 347 stainless steel tubes. The fuel rods contained 6.88 wt. % uranium, 93% enriched in ^{235}U, and 6×10^{22} atoms/cm^3 of H_2 in the form of ZrH_x. Analysis of the characteristics of the fuel alloy indicated that it could be considered as a mixture of zirconium hydride plus uranium metal with a density of 5.89 g/cm^3. Excess and reject fuel elements from the fabrication process were used in the Shield Test Facility (STF), covered in Section 11. General descriptions of the SER are given in references Eg59, Hu61, Be62, Di67, and Ja73.

6.2 REACTOR AND FUEL IDENTIFICATION

Reactor: SNAP Experimental Reactor (SER)
Also identified as: SNAP-2 Experimental Reactor (S2ER)
SNAP-II Experimental Reactor
SNAP-II

Core fuel identification: SER (same as reactor name)

Reactor designer, builder, operator: Atomics International
Reactor owner: Atomic Energy Commission (now DOE)

Fuel designer, builder: Atomics International
Fuel owner: Atomic Energy Commission (now DOE)

6.3 SER DESIGN REACTOR PHYSICS AND CORE DESCRIPTION

Wi58 Pre-loading nuclear analysis, to evaluate required fuel loading
Wi59 Pre-operation design summary
Hu61 Some reactor physics parameters
Be62 Operating parameters and reactor physics information
Ja73 Summary of SER parameters, including flux ratios (Table 8)

Fuel Elements/Assemblies: 61 used in core; 68 were fabricated (Ka59)
Design Neutron Flux: Flux ratios given in Ja73 (Table 8)
Design Neutron Energy: Mean neutron fission energy near 0.18 eV (Go67c)
Core Map and Element Locator:

Dr60, Hu61, Be62, Mi63
Core configuration (without element location identification)

6.4 SER FUEL FABRICATION INFORMATION

Fuel Specifications and Drawings:

- Tu58 Manufacturing planning document for fabrication
- Re59 Specification and engineering design drawing
- Bl59 Quality control planning document
- Ta63 Non-dimensional fuel element specifications (Table 1)
- Bu64 S2ER (SER) description (Table 5.2.2), providing nominal design data, and schematic drawing

- Drawing 7512-971801 "Fuel Rod Assembly - SER - Layout," showing dimensioned fuel rod as part of this assembly (05/59)
- Drawing 7512-71102 "Slug-S.E.R.-End Reflector," showing Be end reflectors (06/58)

Fuel Fabrication and Quality Control:

- Dr60 Typical spectrographic analyses of U, Zr starting materials (Tables I, II); melting and forming of fuel rods; schematic drawing of fuel rod; quality control description; sample photomicrographs of as-fabricated fuel (prior to hydriding)
- Ka59 Melting and casting of U-Zr alloy (how it was done)
- Ka59 Tables: fuel element identification and dimensions, pre- and post-hydriding dimensions, density, electrical resistivity, weight, hydrogen and uranium compositions; assembled element dimensions; hydrogen leak rates; "scintillation testing" (^{235}U gamma spectrum; L-77 reactor neutron scattering to measure hydrogen content - not effective)
- Eg59 Schematic drawing of fuel rod
- Mi63 Schematic drawing of fuel rod
- Ta63 Hydriding process details and hydriding yield (Table 7); general studies of hydrided fuel

Fuel Cladding, Inner Coatings, and Other Non-Fuel Materials:

- Ka59 347 SS cladding (three were Hastelloy B: R-7, U-10, X-10), with Solaramic S1435-A, Be end reflectors (those in Hastelloy B had Hastelloy B end caps), incl. end reflector composition data. Cladding tube dimensions and coating thickness
- Ro61 Cladding development for hydrogen retention, leading to identification of Hastelloy N and B as the most favorable cladding materials and Solaramic S-1435A as an effective hydrogen barrier
- Hu61 Coolant = eutectic NaK (78 wt% potassium)
- Bu64 Solaramic 14-35A coating general description (Table 4.1.1)

Potential Fuel Chemical Reactions:
Potential RCRA-Regulated Materials:

- Ta63 SER end reflectors were composed of Be (removed during INEL decladding)

Quantities of Materials:

See shipping spreadsheets (fuel rods, U, ^{235}U)

Fuel Assembly Identification Methods:

Ka59 Table II diameter data columns distinguish between "numbered end" and "opposite end"

6.5 SER OPERATIONAL HISTORY

Core Operating History:

- Hu61 Operations chronology (Table IV); total SER history (Fig. 21)
  Includes power & reactivity information, but no neutron data
- Be62 Operational report, incl. history (Fig. 16, Table IV)
- Mi63 Operations chronology summary
- Ja73 Summary of SER operation, operations chronology, analysis of reactor operations

Core Neutron Flux Profile:

Operation Abnormalities:

- Hu61 Scrams (Table V); describes maintenance problems
- Be62 Abnormality summary (Table V)
- Ja73 Scram summary (Table 10)

Core Shut-Down Date:

Hu61 Final shutdown 19 November 1960

Core Discharge Date:

Be62 SER disassembly started 19 May 1961, completed 13 June 1961

6.6 SER POST-IRRADIATION PROPERTIES AND EXAMINATION

Visual Observations:

Mi63 Visual examination - appeared to be no problems with fuel or cladding

Dimensions:

Mi63 Dimensional and density measurements

Metallurgical States of Materials:

Mi63 photomicrograph of hydrided fuel after irradiation

Cladding Defects:

Corrosion or Chemical Contamination:

Hydrogen Effects:

Mi63 hydrogen leak rate

Burnup:

Mi63 Burnup profile plots

Fission products:

Mi63 Fission gas release (none observed)

Transuranics:

(can be calculated from available reactor information)

Activation Products:
Decay Heat Generation:
Radiation Level Decay Curves:
Alpha Contamination:

(not relevant for the declad fuels)

6.7 SER ON-SITE STORAGE HISTORY

SNAP fuel elements were stored in air at the Radioactive Materials Disposal Facility (RMDF) in dry storage vaults, in preparation for shipment to INEL following Atomics International Hot Laboratory (AIHL) operations. The AIHL operations included the disassembly of the reactor and core, and the removal and evaluation of the fuel elements. Until cleanup of residual NaK was complete, all AIHL operations were performed in an atmosphere of nitrogen gas, containing no more than 3% oxygen. There were no wet-storage facilities available at Atomics International. No detailed storage history documentation is available, and storage information is based on personal recollections of individuals involved in the program.

6.8 SER SHIPPING RECORDS

The shipping records for SER fuel shipments to INEL were reviewed in detail. The original records are difficult to read, and were transcribed onto the SER spreadsheet that is reproduced at the end of this section. The spreadsheet lists the fuel rod (unclad), total uranium, and ^{235}U masses for each fuel element, plus the shipping document number, shipment date, Atomics International can number, and probable INEL aluminum can number. The latter is base on document copies received from INEL as part of the current review activity. 54 irradiated STF fuel elements were shipped to INEL, out of 68 original fuel elements. The other 14 include 7 irradiated/sectioned fuel elements and 7 unirradiated fuel elements that were disposed of by other means.

The mass totals shown on the spreadsheet were calculated from the individual recorded fuel element data. They were compared with an SER Fuel Data mass sheet dated 7/27/64 and filed with the shipping records. This comparison shows a slight discrepancy for shipment LAE-JWA-27 (AI Can No. 14-H), where the data sheet lists the total fuel rod mass as 4337 g, instead of 4339 g as obtained from the shipping records (with a corresponding total fuel mass for SER of 39023 g instead of 39025 g).

6.9 SER REFERENCES

Be62 J. P. Beall and M. W. Hulin, "SNAP-2 Experimental Reactor Operation and Test Program," Atomics International Report NAA-SR-7088 (April 1962) (76 pages)

Bl59 M. Bloomfield, "Tentative SER Fuel Element Inspection Specification," Atomics International Memorandum NAA-SR-MEMO-4063 (June 1959) (8 pages)

Bu64 G. F. Burdi, Ed., "SNAP Technology Handbook. Volume II. Hybrid Fuels and Claddings," Atomics International Report NAA-SR-8617, Volume II (November 1964) (129 pages)

Di67 H. M. Dieckamp, "Nuclear Space Power Systems," Atomics International (September 1967) (unpublished book, 388 pages)

Dr60 P. S. Drennan, "Melting and Forming of SER Fuel Rods," Atomics International Report NAA-SR-4762 (October 1960) (30 pages)

Eg59 R. R. Eggleston and G. L. Schmidt, Ed., "SNAP II Experimental Reactor Hazards Report," Atomics International Report NAA-SR-3465 (April 1959) (156 pages)

Go67c K. G. Golliher, et al., "SNAP Engineering Test Facility Safety Analysis Report: Addendum for Intrinsic Supercriticality Experiments," Atomics International Technical Data Record NAA-SR-MEMO-12614 (November 1967) (80 pages)

Hu61 M. W. Hulin and J. Beall, Ed., "Preliminary Results of the SNAP 2 Experimental Reactor," Atomics International Report NAA-SR-5991 (April 1961) (60 pages)

Ja73 A. A. Jarrett, "SNAP 2 Summary Report," Rockwell International, Atomics International Division Report AI-AEC-13068 (July 1973) (190 pages)

Ka59 N. H. Katz, "SER Data Report," Atomics International Memo NAA-SR-MEMO-4722 (December 1959) (25 pages)

Mi63* J. R. Miller, "Postirradiation Evaluation of SER Fuel Elements," Atomics International Report NAA-SR-8090 (May 1963) (29 pages)

Re59 E. L. Reed, "SER Fuel Element Specification," Atomics International Internal Letter dated 11 June 1959 (4 pages)

Ro61* C. J. Romero and S. Elchyshyn, "Cladding Development for SER Fuel Elements," Atomics International Report NAA-SR-4831 (February 1961) (42 pages)

Ta63 H. Taketani, "Hydriding of SNAP 2 Fuel Rods," Atomics International Report NAA-SR-5037 (June 1963) (37 pages)

Tu58 R. R. Turk, "Feasibility Report for Fabrication of SNAP II Fuel Elements," Atomics International Memo NAA-SR-MEMO-3083, Rev. II (October 1958) (32 pages)

Wi58 L. A. Wilson, "Evaluation of Required Fuel Loading in the SER," Atomics International Technical Data Record NAA-SR-MEMO-3406 (December 1958) (12 pages)

Wi59 L. A. Wilson, "SNAP II Experimental Reactor Physics Analysis," Atomics International Report NAA-SR-3607 (June 1959) (71 pages)

SER SHIPPING RECORD SUMMARY

Shipping Document	Shipment Date	Al Can No.	Fuel Identification Element	Rod	Location	Fuel Masses (grams) Unclad	total U	U-235	Notes	Prob. INEL Can No.
CAE-JZA-15	4/27/66	4	5R-9			4342	295	277		4
			9R-5							
			M-6							
			Y-6							
			Q-4							
			Q-6							
		5	Y-7			4337	298	280		4
			Z-1							
			T-8							
			11R-7							
			9R-10							
			5R-2							
		6	R-4			4338	296	278		4
			Q-3							
			Q-2							
			5R-3							
			10R-10							
			6R-4							
	Shipment totals:		18	elements		13017	889	835		
CAE-JZA-20	5/25/66	3	10R-7			4335	296	278		5
			6R-3							
			M-2							
			R-3							
			Z-7							
			Q-5							
		2	W-6			4337	295	277		5
			9R-9							
			11R-3							
			T-2							
			R-1							
			5R-8							
		1	X-7			4326	293	275		5
			W-2							
			Q-1							
			10R-9							
			9R-3							
			W-1							
	Shipment totals:		18	elements		12998	884	830		

Shipping Document	Shipment Date	AI Can No.	Fuel Identification Element	Rod	Location	Fuel Masses (grams) Unclad	total U	U-235	Notes	Prob. INEL Can No.
LAE-JWA-27	9/30/68	14-H	347 T-1			723	48	45		SN-20
			347 U-2			722	49	46		
			347 U-1			721	48	45		
			347 11R-4			724	50	47		
			347 R-2			724	49	46		
			347 M-5			725	51	48		
	Shipment totals:		6	elements		4339 (see notes below)	295	277		
LAE-JWA-28	10/14/68	15	347 W-4			724	49	46		SN-20
			347 W-10			724	48	45		
			347 U-2*			724	51	48	actually V-2*	
			347 U-8*			724	48	45	actually V-8*	
			347 U-10*			723	50	47	actually V-10*	
			347 9R-4			720	49	46		
			* listed as U's on AI shipping document, but backup data identifies as V's							
	Shipment totals:		6	elements		4339	295	277		
LAE-JWA-29	11/4/68	16	347 X-6			724	49	46		SN-20
			347 Y-8			722	49	46		
			347 S-4			720	49	46		
			347 10R-1			720	50	47		
			347 Y-2			722	48	45		
			347 R-5			724	50	47		
	Shipment totals:		6	elements		4332	295	277		
	1968 shipment totals:		18			13010	885	831		
SER reactor totals:			54	elements		39025	2658	2496		

Reference: Data transcribed from archived copies of original shipping records
Comparison made with SER Fuel Data sheet dated 7/27/64

Notes: 54 fuel elements received by INEL per BUC-41-66A
68 fuel elements were made (Ka59: NAA-SR-MEMO-4722 12/10/59);
documentation (Ka59) does not distinguish between a fuel rod identification number and a fuel element number
61 fuel elements were used in the core
7 core fuel elements were cut up for hot cell examination (Mi63: NAA-SR-8090 05/01/63)
The 7 unused and 7 irradiated/sectioned fuel elements were not shipped to INEL

For LAE-JWA-27, the original inventory sheet of October 1963 listed the total mass of material that went into Can 14-H as 4337 g instead of 4339 g, and a reactor total fuel mass of 39023 g. The shipping document, which tabulates individual elements, lists 4339 g. This spreadsheet total for Can 14-H (4339 g) is the sum of the tabulated individual elements.

c-ser.xls 8/30/94

7. SNAP-2 DEVELOPMENTAL REACTOR (S2DR)

Date went critical:	April 1961 (O165)
Date of shutdown:	December 1962 (O165)
Thermal power:	65 kWt
Thermal energy:	273,000 kWt-h
Time at power and temperature:	2800 h at 1200°F, 7700 h above 900°F
	2100 h at 1200°F, 6900 h above 900°F (Di67)
Operating location:	SSFL Building 024

7.1 S2DR GENERAL DESCRIPTION

The S2DR was the second-generation SNAP-2 design, which more nearly approximated the detailed requirements of a space reactor (Di67). The SDR core consisted of 37 fuel elements of 1¼-inch diameter containing 10 wt. % uranium and 6.5×10^{22} hydrogen atoms/cm^3. The core geometry was similar to that of the SER. A general description is given in Ja73.

7.2 S2DR REACTOR AND FUEL IDENTIFICATION

Reactor: SNAP-2 Developmental Reactor (S2DR)
Also identified as: SNAP Developmental Reactor (SDR)

Core fuel identification: S2DR (same as reactor name)

Reactor designer, builder, operator: Atomics International
Reactor owner: Atomic Energy Commission (now DOE)

Fuel designer, builder: Atomics International
Fuel owner: Atomic Energy Commission (now DOE)

7.3 S2DR DESIGN REACTOR PHYSICS AND CORE DESCRIPTION

Fuel Elements/Assemblies: 80 fuel elements fabricated (De62), 37 used in core (De62, Mi64)
Design Neutron Flux:
Design Neutron Energy: Mean neutron fission energy near 0.18 eV (Go67c)
Core Map and Element Locator:

- De62 Core map (Fig. 16) - prior to reactor assembly
- Mi64 Core map (Fig. 1)
- Fe64 Core map (Fig. 3)
- O165 Core map with identification of fuel elements for detailed analysis (Fig. 2)

7.4 S2DR FUEL FABRICATION INFORMATION

Fuel Specifications and Drawings:

De62 Schematic drawing of fuel element (Fig. 1)

Ta63 Non-dimensional fuel element specifications (Table 1)

Bu64 S2DR description (Table 5.2.3), providing nominal design data, plus fuel element schematic drawing

O165 Fuel element schematic drawing

Drawing 7512-22151 "S2DR Fuel Element," in which the fuel rod is Detail-5 (07/61)

Fuel Fabrication and Quality Control:

Dr61	Table I:	Assembly of fuel element components, including correlation of fuel element number with fuel rod number, top and bottom reflector numbers, tube number, and top and bottom end cap numbers for all 80 elements.
	Table II:	Pre-irradiation hydrogen leak rate data
	Table III:	Fuel element correlation with total weight and ^{235}U content
	Table IV:	Fuel element dimensions
	Table V:	Fuel rod N_H and density
	Table VI:	Fuel rod weight, resistivity, surface quality (Zyglo), internal soundness (ultrasonic category)
	Table VII:	Dimensions of finished machined fuel rods
	Table VIII:	Pre-hydride uranium content, hardness, weight, and density
	Table IX:	Pre-hydride fuel rod dimensions
	Table X:	Chemical impurity of pre-hydride fuel alloy
	Table XI:	Dimensions, weight, and density of BeO end reflectors
	Table XII:	Dimensions of "A" end caps (uncoated)
	Table XIII:	Dimensions of "A" end caps (coated)
	Table XIV:	Dimensions of "H" end caps (uncoated)
	Table XV:	Wall thickness and hardness of tubing (uncoated)
	Table XVI:	Coating thickness, weight, and Sm_2O_3 content
	Table XVII:	Sm_2O_3 concentration in coating
	Table XVIII:	AI analysis of Hastelloy-N cladding

De62 80 fuel elements were produced, of which 37 were used. An additional 9 were used in dry critical experiments and removed. (Thus 46 were irradiated.) Table II identifies all 80 elements, their corresponding fuel rod numbers, and their end use, including those not used in core. This table also gives N_H, hydrogen leak rate, and longitudinal bow (TIR, total indicator reading). Document gives fuel element general assembly process.

Ta63 Chemical analyses of U metal, Zr sponge, and fuel alloy; provides hydriding process details and hydriding yield (Table 7). General studies of hydrided fuel.

Fuel Cladding, Inner Coatings, and Other Non-Fuel Materials:

De62 Cladding identification, composition, tensile properties; coating identified as Solaramic S-1435-SM2

Bu64 Solaramic 14-35A coating general description (Table 4.1.1); S-1435-SM2 refers to addition of Sm_2O_3 poison

Potential Fuel Chemical Reactions:

Potential RCRA-Regulated Materials:

Ta63 S2DR end reflectors were composed of BeO (removed during INEL decladding)

Quantities of Materials:

See shipping spreadsheets (fuel rods, U, ^{235}U)

Dr61 Total element weights in Table III

Fuel Assembly Identification Methods:

Mi64 Element numbers scribed on end caps, as shown in Figs. 3 and 6

7.5 S2DR OPERATIONAL HISTORY

Core Operating History:

Fe64 Overall operations history, summarized in Table 4

O165 Total operating history timeline (Fig. 3); total energy release of 272,900 kWh; peak fuel temperature 1279 °F

Ja73 Summary of S2DR operation, operations chronology, analysis of reactor operations

Core Neutron Flux Profile:

Operation Abnormalities:

Fe64 Operation abnormalities (Table 5)

O165 Reactor shutdown for maintenance in December 1961, February 1962; air-blast heat exchanger failures in April and July 1962; shutdown for suspected NaK leak in September 1962 (none found)

Ja73 History of operation abnormalities

Core Shut-Down Date:

O165 End of reactor operation December 1962

Core Discharge Date:

O165 Reactor disassembled in the first half of calendar year 1963

7.6 S2DR POST-IRRADIATION PROPERTIES AND EXAMINATION

Reference O163 provides the hot cell examination requirements of S2DR fuel elements (planning document), and identifies (by element number) the six fuel elements selected for post-irradiation examination. Reference Mi64 is the detailed post-irradiation analysis report, and O165 is a final release version of Mi64. It contains visual observations and photographs, metallography, axial and radial gamma scans, fuel element dimensions, hydrogen leak rates, and fission gas release, for the fuel elements, rods, and cladding.

Visual Observations:

Mi64 Visual observations before fuel element disassembly, for 8 selected fuel rods; included photographs of typical findings.

Mi64 Fuel rod observations following decladding of selected elements (Fig. 17-20).

Dimensions:

Mi64 Post-irradiation fuel element dimensions for selected elements (Table I - V)

Mi64 Radial dimensional growth of fuel rod (Table VIII); fuel rod densities (Table IX)

Metallurgical States of Materials:

Mi64 Fuel rod and cladding photomicrographs (Fig. 23 - 39)

Cladding Defects:

Mi64 Photomicrographs (no obvious defects)

Corrosion or Chemical Contamination:

Mi64 Photographs (Figures 6-9) indicate chemical reactions at the external cladding surface of selected fuel elements.

Hydrogen Effects:

Mi64 Hydrogen leak rates of selected fuel elements (Table VII)

Mi64 Hydrogen analysis of selected fuel rods (Table X)

Burnup:

Mi64 Burnup analyses (Section B.2.), based on fission gas analysis (^{85}Kr activity).

O165 0.027 metal-atom % burnup

Fission products:

Mi64 Longitudinal (Fig. 11, 12) and diametrical (Fig. 22A, B) gamma scan traces and summary (Table VI); analyzer set to counting peak of ^{95}Zr and used to obtain relative fuel burnup.

Transuranics:

(can be calculated from available reactor information)

Activation Products:
Decay Heat Generation:
Radiation Level Decay Curves:
Alpha Contamination:

(not relevant for the declad fuels)

7.7 S2DR ON-SITE STORAGE HISTORY

The reactor was disassembled in the first half of calendar year 1963 (O165). Reference O163 addresses the 6 elements stored at the Radioactive Materials Disposal Facility (RMDF) prior to detailed post-irradiation examination at the Atomics International Hot Laboratory (AIHL). SNAP fuel elements were stored in air at the RMDF in dry storage vaults following removal from the AIHL in preparation for shipment to INEL. There were no wet-storage facilities available at Atomics International. No detailed storage history documentation is available. The only fuel disassembly that took place was for hot cell examination, which is covered in the post-irradiation examination section.

7.8 S2DR SHIPPING RECORDS

The shipping records for the S2DR fuel shipments to INEL were reviewed in detail. The original records are difficult to read, and were transcribed onto the S2DR spreadsheet that is reproduced at the end of this section. The spreadsheet lists the core location, fuel rod identification, unclad fuel rod mass, total uranium mass, and ^{235}U mass for each fuel element, plus the shipping document number, shipment date, Atomics International can number, and probable INEL aluminum can number. The latter is based on document copies received from INEL as part of the current review activity. The fuel rod identifications and fuel element core locations are from Reference De62. A total of 37 irradiated S2DR fuel elements were shipped to INEL.

The mass totals shown on the spreadsheet were calculated from the individual recorded fuel element data. They were compared with, and agree with, an S2ER Fuel Data mass sheet dated 7/27/64 and filed with the shipping records.

7.9 S2DR REFERENCES

Bu64 G. F. Burdi, Ed., "SNAP Technology Handbook. Volume II. Hybrid Fuels and Claddings," Atomics International Report NAA-SR-8617, Volume II (November 1964) (129 pages)

De62* W. F. Dennison and T. S. Jakobowski, "Assembly of S-2 DR Fuel Elements," Atomics International Report NAA-SR-7048 (December 1962) (26 pages)

Di67 H. M. Dieckamp, "Nuclear Space Power Systems," Atomics International (September 1967)

Dr61 P. S. Drennan and W. Sawicky, "Summary of Test Data for S-2-DR Fuel Elements," Atomics International Technical Data Record NAA-SR-MEMO-6745 (September 1961) (32 pages)

Fe64 L. D. Felten, et al., "Final Report on the SNAP 2 Development Reactor (S2DR) Test Program," Atomics International Report NAA-SR-8295 (April 1964) (156 pages)

Go67c K. G. Golliher, et al., "SNAP Engineering Test Facility Safety Analysis Report: Addendum for Intrinsic Supercriticality Experiments," Atomics International Technical Data Record NAA-SR-MEMO-12614 (November 1967) (80 pages)

Ja73 A. A. Jarrett, "SNAP 2 Summary Report," Rockwell International, Atomics International Division Report AI-AEC-13068 (July 1973) (190 pages)

Mi64 K. J. Miller, "Postirradiation Examination of the SNAP-2 Developmental Reactor Fuel Elements," Atomics International Technical Data Record NAA-SR-MEMO-9753 (April 1964) (74 pages)

Ol63 P. S. Olson, "Hot Cell Examination Requirements of S2DR Fuel Elements," Atomics International Technical Data Record NAA-SR-8716 [NAA-SR-TDR-8716] (July 1963) (8 pages)

Ol65* P. S. Olson, " Evaluation of Fuel Elements from SNAP 2 Developmental Reactor Core," Atomics International Report NAA-SR-9648 (January 1965) (35 pages)

Ta63 H. Taketani, "Hydriding of SNAP 2 Fuel Rods," Atomics International Report NAA-SR-5037 (June 1963) (37 pages)

S2DR SHIPPING RECORD SUMMARY

Shipping Document	Shipment Date	Al Can No.	Fuel Identification Element	Rod	Location	Fuel Masses (grams) Unclad	total U	U-235	Notes	Prob. INEL Can No.
CAE-JZA-13	4/15/66	2	E-22	515-3	IV-8	1142	113	105		1
			E-26	515-5	II-2	1141	113	105		
			E-30	517-3	II-5	1141	113	105		
			E-34	518-2	II-4	1140	113	105		
			E-38	518-1	III-6	1140	113	105		
			E-39	520-2	IV-6	1143	113	105		
	Shipment totals:		6	elements		6847	678	630		
CAE-JZA-14	4/20/66	7	E-51	520-7	IV-18	1141	113	105		3
			E-60	520-3	IV-1	1153	114	106		
			E-65	511-4	IV-16	1153	114	106		
			E-73	512-6	III-1	1137	113	105		
			E-79	511-3	IV-17	1139	113	105		
			E-76	526-1	III-12	1140	113	105		
	Shipment totals:		6	elements		6863	680	632		
CAE-JZA-16	5/2/66	5	E-15	515-1	IV-12	1143	113	105		2
			E-35	519-7	III-4	1142	113	105		
			E-21	518-6	IV-7	1142	113	105		
			E-32	517-1	IV-9	1142	113	105		
			E-67	516-3	III-5	1141	113	105		
			E-69	513-6	IV-10	1155	115	107		
	Shipment totals:		6	elements		6865	680	632		
CAE-JZA-17	5/5/66	6	E-74	510-4	III-3	1140	113	105		3
			E-72	509-3	III-9	1136	113	105		
			E-14	519-2	II-1	1142	113	105		
			E-31	517-5	III-2	1141	113	105		
			E-77	513-5	IV-3	1141	113	105		
			E-66	511-6	IV-4	1155	115	107		
	Shipment totals:		6	elements		6855	680	632		
CAE-JZA-18	5/9/66	1	E-37	537-3	III-7	1143	113	105		1
			E-8	515-4	III-8	1141	113	105		
			E-58	516-2	I-1	1141	113	105		
			E-33	520-5	IV-13	1142	113	105		
			E-49	527-1	IV-5	1142	113	105		
			E-18	517-6	IV-11	1141	113	105		
	Shipment totals:		6	elements		6850	678	630		
CAE-JZA-19	5/12/66	4	E-48	515-6	II-3	1028	102	95		2
			E-50	519-6	IV-15	1027	102	95		
	Shipment totals:		2	elements		2055	204	190		

Shipping Document	Shipment Date	Al Can No.	Fuel Identification Element	Rod	Location	Fuel Masses (grams) Unclad	total U	U-235	Notes	Prob. INEL Can No.
LAE-JWA-14	4/3/68	17	E-24	519-1	IV-2	1142	113	105		SN-17
		(originally	E-36	518-3	III-10	1141	113	105		
		can #3)	E-75	510-5	III-11	1140	113	105		
			E-55	519-4	II-6	1139	113	105		
			E-23	517-7	IV-14	1142	113	105		
		Shipment totals:	5	elements		5704	565	525		

S2DR reactor totals:	37	elements	42039	4165	3871

References: Data transcribed from archived copies of original shipping records
Mass sums compared with (and agree with) S2DR Fuel Data sheet dated 7/27/64
Fuel rod identifications from De62 (NAA-SR-7048, 12/15/62, Table II)

Notes: 37 fuel elements shipped to INEL

8. SNAP-10A GROUND TEST REACTOR (10FS-3)

Date went critical:	January 1965
Date of shutdown:	March 16 1966 (Go67a)
Thermal power:	40 kWt nominal
Thermal energy:	382,944 kWt-h
Electric power:	402 Watts
Electric energy:	4028 kW-h
Time at power and temperature:	10,005 h (417 d) above 900 °F
Operating location:	SSFL Building 024

8.1 10FS-3 GENERAL DESCRIPTION

The SNAP-10A reactor system utilized SNAP-2 reactor technology with minimum modifications. The reactor core consisted of 37 hydrided zirconium-uranium alloy fuel elements clad in the nickel-base alloy Hastelloy N. The fuel elements were positioned and constrained in the core by upper and lower grid plates. They were arrayed in a hexagonal pattern within a circular reactor vessel, and surrounded by a beryllium reflector structure.

The hydrided zirconium-uranium alloy fuel material alloy served both as fuel and moderator. The unhydrided alloy contained a nominal 10 wt. % uranium and was hydrided to a hydrogen concentration of 6.35×10^{22} atoms of hydrogen/cm^3, which is slightly less than the hydrogen concentration in cold water. The uranium was 93% enriched in ^{235}U. This material was formed into rods of 30.7 mm (1.21 in.) diameter and 31.1 cm (12.25 in.) length and was canned in Hastelloy-N cladding tubes. The cladding tubes were 31.75 mm OD and had a wall thickness of 0.38 mm (0.015 in.). Internal surfaces of the cladding tubes were coated with a 0.076-mm (0.003-in.) layer of ceramic glass material. This coating acted as a barrier to hydrogen leakage from the fuel. The ends of the fuel elements were sealed with end caps of Hastelloy-N material welded to the cladding tube. The end pins that positioned and held the elements between the grid plates were an integral part of the end caps. Each fuel element weighed 1.5 kg (3.4 lb) and had an overall length, including end pins, of 32.6 cm (12.82 in.).

Each unclad fuel rod contained approximately 128 g of ^{235}U, 11.8 g of ^{238}U, 24.6 g of hydrogen, and 1215 g of zirconium. Its total weight was approximately 1380 g. A small amount of carbon (approximately 0.15 wt.%) was added as a grain refiner to produce an optimum microstructure, and thus increase yield during hydriding. The fuel was a hard, semibrittle material with a machined surface and a metallic appearance. All edges of the fuel rod were rounded to prevent damage to the ceramic hydrogen barrier during assembly and handling of the fuel element.

The ceramic glass barrier, commercially known as Solaramic 1435-SM2, was composed primarily of oxides of aluminum, silicon, titanium, manganese, and barium with smaller amounts of sodium, lithium, and potassium. This ceramic coating was generally applied in three firings. In the last two firings, a small quantity of the burnable poison samarium oxide (Sm_2O_3) was incorporated into the coating. The nominal samarium oxide loading was 3 mg/cm. Depletion of this material inserted reactivity at the proper rate to provide more than one year of stable reactor operation without complex mechanical control rods.

One end cap of the fuel element was welded to a cladding tube before the ceramic barrier was applied. After the ceramic barrier had been fired, a fuel rod was inserted into the cladding tube. The cladding tube was then sealed against hydrogen loss by the insertion of a ceramic-coated blend cup. The hydrogen barrier was made complete by locally heating the cladding tube and blend cup to blend the ceramic surfaces together. Following the blending operation, an end cap was welded to the cladding tube, covering the blend cup. No play existed between the fuel rod and the cladding tube.

General descriptions of the SNAP 10A system are given in references Mi65, Di67, St67, AN76, and An83.

8.2 10FS-3 REACTOR AND FUEL IDENTIFICATION

Reactor: SNAP 10A Ground Test Reactor (10FS-3)
Also Identified As: SNAP 10A FS-3 Ground Test System
SNAP 10 Flight System 3 Reactor (S10FS-3)

Core fuel Identification: 10FS-3 (same as reactor name)

Note (Sa64d): fuel elements were originally fabricated for three SNAP 10A cores with core designations FS-1, FS-3, and FS-4. Data from FS-1 operation showed that SNAP 10A operating characteristics could be improved with the fabrication of some new low-N_H, high-Sm_2O_3 fuel elements and the redistribution of elements in the three cores. The revised cores were redesignated as the FS-3, FS-4, and FS-5 cores. Thus the core designations for the fuel elements in the original data packages do not correspond to the final distribution of fuel elements. See below under Fuel Fabrication for identification of those fuel elements incorporated in the final 10FS-3 core.

Reactor designer, builder, operator: Atomics International
Reactor owner: Atomic Energy Commission (now DOE)

Fuel designer, builder: Atomics International
Fuel owner: Atomic Energy Commission (now DOE)

8.3 10FS-3 DESIGN REACTOR PHYSICS AND CORE DESCRIPTION

Fuel elements/assemblies: 37 fuel elements (Mi65, Go67a)
Design neutron flux:
Design neutron energy: Mean neutron fission energy near 0.18 eV (Go67c)
Core map and element locator:

Mi65 Fuel loading table (Table 6) and core loading configuration (Fig. 7)

Go67, Go67a, Go67b:

Core maps showing locations of individual fuel elements (Fig. 1); Go67a includes table identifying fuel element numbers and core locations

8.4 10FS-3 FUEL FABRICATION INFORMATION

Fuel specifications and drawings:

Sa64, Sa64a, Sa64b, Sa64c, Sa64d:

Reference the engineering drawings and the engineering design specifications for the fuel elements, fuel rods, and raw materials (such as the cladding materials)

Go67 Sketch of fuel element with nominal design information (Fig. 2, Table 1)

Go67a, Go67b:

Schematic of the fuel element (Fig. 2); fuel element design criteria (Table 1; probably average of the 37 in-core elements)

Drawing 7580-18020	Drawing for fuel elements 136, 150, 190
Drawing 7573-18002	"Element Assembly - 10A Fuel" (04/63; covers other 34 fuel elements)
Drawing 7573-18004	"Rod - Fuel Element"

Fuel fabrication and quality control:

Bu64 General fuel element information and schematic diagram

Ki63 Melting, casting, and extrusion process procedures for SNAP 10A and 2 ZrU fuel materials

Sa63 Quality control planning document for fuel cladding, fuel rod, and fuel element testing and inspection. Contains examples of all source data forms prepared.

Ca63a, Ar63:

Sequential manufacturing planning sheets for fuel cladding, fuel rod, and fuel element fabrication. These documents complement Sa63. They reference engineering drawings and specifications for the SNAP 10A fuel.

Sa64, Sa64a, Sa64b, Sa64c, Sa64d:

Fabrication quality control (test and inspection) data packages for the SNAP 10A fuel elements used for the FS-1, FS-3, FS-4, and FS-5 cores. These data packages cover the raw materials through the finished fuel elements. Examples include dimension data, chemistry, hydrogen leak rate, raw materials traceability, visual and non-destructive testing data, and process traceability. Reports Sa64, Sa64a, Sa64b, and Sa64c contain fuel element pre-assembly core loading assignments for the FS-1, FS-3, and FS-4 cores that were later changed. For example, Sa64b covers the fuel elements for the originally planned FS-3 core, none of which were used in the final 10FS-3 core.

Sa64d:

Data obtained from the FS-1 System test showed that the long-term operating characteristics of the SNAP 10A reactors could be improved by increasing the Sm_2O_3 content and decreasing slightly the average N_H of the core. This required fabrication of several new low-N_H, high-Sm_2O_3 fuel elements (including 525, 528, and 529), and the redistribution of elements in the FS-1, -3, and -4 cores. These cores were then redesignated as the FS-3, -4, and -5 cores, respectively.

A summary of the fuel elements used in the 10FS-3 ground-test reactor (from Sa64d) and the appropriate reference for the data package for each is given in the table below:

Fuel cladding, inner coatings, and other non-fuel materials:

Sa64, Sa64a, Sa64b, Sa64c, Sa64d:

Full data packages on all fuel element materials

Bu64 Hastelloy-N cladding with chromized surfaces; Solaramic S14-35A ceramic coating on cladding inside surfaces

Potential fuel chemical reactions:

Potential RCRA-regulated materials:

Quantities of materials:

See shipping spreadsheets (fuel rods, U, ^{235}U)

Identifications and Data Package References for the 10FS-3 Fuel Elements

Core Position	Element No.	Reference	Core Position	Element No.	Reference	Core Position	Element No.	Reference
1	422	Sw64a	14	365	Sw64a	27	190	Sw64
2	486	Sw64a	15	421	Sw64a	28	150	Sw64
3	529	Sw64d	16	441	Sw64a	29	337	Sw64a
4	484	Sw64a	17	459	Sw64a	30	350	Sw64a
5	528	Sw64d	18	336	Sw64a	31	349	Sw64a
6	440	Sw64a	19	334	Sw64a	32	338	Sw64a
7	525	Sw64d	20	345	Sw64a	33	439	Sw64a
8	482	Sw64a	21	136	Sw64	34	442	Sw64a
9	331	Sw64a	22	333	Sw64a	35	348	Sw64a
10	332	Sw64a	23	479	Sw64a	36	364	Sw64a
11	456	Sw64a	24	419	Sw64a	37	352	Sw64a
12	435	Sw64a	25	390	Sw64a			
13	418	Sw64a	26	393	Sw64a			

Fuel assembly identification methods:

Go67 Fuel elements were scribed to show reactor orientation with respect to centerline before removal

Drawing 7573-18004 calls out identification by electric etching on the side of the rod

8.5 10FS-3 OPERATIONAL HISTORY

Core Operating History:

Mi65 Report on first 90 days of power operation; graph of core thermal power vs. time (Fig. 24), decreasing from ~41 kWt to ~37 kWt

Go67a 10,028 hours of continuous operation; 970 °F coolant outlet temperature

St67 System operations summary (Tables 18, 19)

Core neutron flux profile:

Operation abnormalities:

Core shut-down date:

Go67a March 16, 1966

Core discharge date:

Do67 Reactor core removed from facility and placed in shipping cask on May 21, 1966

8.6 10FS-3 POST-IRRADIATION PROPERTIES AND EXAMINATION

Available source data include extensive individual element-by-element fuel element data trace files for gamma scans, hydrogen retention, etc. These data are contained in individual file folders rather than in report format. Also included in these files are several original photographs and negatives.

References Do67 and Kr67 describe the disassembly and post-irradiation examination of the reactor components, including metallography, for those reactor components other than the fuel.

Go67 All 37 fuel element subjected to post-irradiation screening examination. On removal from the core, each fuel element was cleaned with Dowanol to remove residual NaK, in a nitrogen atmosphere. 6 elements were selected for detailed post-irradiation examination: 442, 331, 529, 440, 482, and 422; results given for 422, 331, and 529.

Go67a Post-irradiation screening examination results

Go67b Post-irradiation detailed (destructive) examination of six selected fuel elements; elements 422, 331, and 529 were declad

Visual observations:

Go67 Visual examination results, concluding that there were apparent cladding or weld defects.

Go67a Photographs of typical fuel element surfaces (Fig. 10 - 12); Fig. 12 shows identification of the fuel element number.

Go67b Photographs of typical fuel element surfaces

Dimensions:

Go67 Summary of fuel element length (Table 2) and diameter (Table 3) change data

Go67a Summary of fuel element length (Table 2) and diameter (Table 3) change data; detailed fuel element dimensional data (Appendix A)

Go67b Declad fuel rod dimension and density measurements (Figs. 4-6, Appendix A)

Metallurgical states of materials:

Go67b Fuel metallography for elements 331, 529, 422; some examples of minor microcracking (not considered significant)

Cladding defects:

Go67 No defects or unexpected results were observed, and the examination supported the conclusion that all the fuel elements completed reactor operation in good condition; descriptions of metallography for selected elements

Go67b Cladding metallography (Fig. 8); no visual signs of damage

Corrosion or chemical contamination:

Go67b Evidence of light contact of fuel-to-coating along the side of the fuel rods in some areas, as would be expected where rods and cladding tubes were not held straight throughout their lifetimes

Hydrogen effects:

Go67 Hydrogen analysis results for 3 detailed-examination elements

Go67b Hydrogen analysis results for 3 detailed-examination elements

Burnup:

Go67 Fuel burnup for fuel element 422 (Table 5), including mass spectrographic results (0.0177 to 0.0235 metal-atom %)

Go67b Fuel burnup for fuel element 422 (Table 3), based on gamma scan (Fig. 7)

Transuranics:

(can be calculated from available reactor information)

Fission products:

Go67 Fission gas sampling from fuel element 440; gamma scan data to count ^{95}Zr activity, summarized by core location (Fig. 13) and including typical gamma scan (Fig. 12)

Go67a Typical gamma scan (Fig. 13; same as Go67); core map showing gamma scan activity of each fuel element (Fig. 14)

Go67b Fission gas sampling from fuel element 440

Activation products:

Decay heat generation:

Radiation level decay curves:

Do67 Radiation levels of system components at time of disassembly; core radiation level approximately 200 mr/h 'general', 400 mr/h maximum 1 inch from bottom, and 50 mr/h maximum at 1 meter

Alpha contamination:

(not relevant for the declad fuels)

8.7 10FS-3 ON-SITE STORAGE HISTORY

The reactor core was removed from the SNAP SETF facility 2 months after shutdown and transferred to the AI Hot Laboratory for post-irradiation examination (Do67). A photograph of the intact 10FS-3 reactor assembly as received at the AI Hot Laboratory is reproduced in Ref. Go67a. The only fuel disassembly that took place was for hot cell examination, which is covered in the post-irradiation examination section. The SNAP fuel elements were stored in air at the RMDF in dry storage vaults following Hot Laboratory examination, in preparation for shipment to INEL. There were no wet-storage facilities available at Atomics International. No detailed storage history documentation is available.

8.8 10FS-3 SHIPPING RECORDS

The shipping records for the 10FS-3 fuel shipments to INEL were reviewed in detail. The original records are difficult to read, and were transcribed onto the 10FS-3 spreadsheet that is reproduced at the end of this section. The spreadsheet lists the core location, fuel rod identification, unclad fuel rod mass, total uranium mass, and ^{235}U mass for each fuel element, plus the shipping document number, shipment date, Atomics International can number, and probable INEL aluminum can number. The latter is base on document copies received from INEL as part of the current review activity. A total of 37 irradiated 10FS-3 fuel elements, of which 4 were declad and/or sectioned, were shipped to INEL.

The shipping documents have typographical errors for five fuel elements: E-332 (recorded as E-322), E-421 (recorded as E-42), E-338 (recorded as E-388), E-136 (recorded as E-316), and E-442 (recorded as E-440). Our corrected identification is based on fuel element number comparisons with documentation for fuel element core allocations, as tabulated in the final QA data package (Sa64d) and in the post-irradiation screening examination reports (Go67a). Fuel rod identification numbers were obtained from the fuel element QA data packages (Sa64, Sa64a, Sa64d).

The mass totals shown on the spreadsheet were calculated from the individual recorded fuel element data. They were compared with the 10FS-3 Fuel Data mass summary sheet that was filed with the shipping documents. This comparison showed one 1-g mass discrepancy, for ^{235}U in AI Can No. 13, which has not been resolved. The 10FS-3 Fuel Data summary sheet also contains a few identified recording errors: the total material available included 33 (not 34) complete elements, and ID #18 (AI Can No. 18, as identified on the 10FS-3 Fuel Data summary sheet) included element E529 (not element E524).

8.9 10FS-3 REFERENCES

AN76 *Anonymous*, "Zirconium Hydride (SNAP) Reactor Space Power Programs. Summary of Major Activities and Accomplishments," Rockwell International Energy Systems Group (June 1976)

An83 R. V. Anderson, et al., "Space Reactor Electric Systems. Subsystem Technology Assessment," Rockwell International Energy Systems Group Report ESG-DOE-13398 (March 1983)

Ar63 J. R. Armstrong, et al., "SNAP 10 A FS-3, -4, and -5 Fabrication Process Sheets and Auxiliary Forms," Atomics International Technical Data Record NAA-SR-MEMO-8809 (July 1963) (41 pages)

Bu64 G. F. Burdi, Ed., "SNAP Technology Handbook. Volume II. Hybrid Fuels and Claddings," Atomics International Report NAA-SR-8617, Volume II (November 1964) (129 pages)

Ca63a D. C. Campbell, et al., "SNAP 10A FS-1 Fabrication Process Sheets and Auxiliary Forms," Atomics International Technical Data Record NAA-SR-MEMO-8272 (February 1963) (37 pages)

Di67 H. M. Dieckamp, "Nuclear Space Power Systems," Atomics International (September 1967) (unpublished book, 388 pages)

Do67 J. P. Dooley and J. P. Beall, "The Disassembly and Postoperation Component Examination of the SNAP 10A FS-3," Atomics International Report NAA-SR-12504 (September 1967) (70 pages)

Go67* T. A. Golding, "Post-Operation Evaluation of Fuel Elements from the SNAP 10 Flight System 3 Reactor," Atomics International Report NAA-SR-12031 (September 1967) (41 pages)

Go67a T. A. Golding, "Post-Irradiation Screening Examination of the S10FS-3 Fuel Elements," Atomics International Technical Data Record NAA-SR-MEMO-12338 (February 1967) (29 pages)

Go67b T. A. Golding, "Post-Irradiation Examination of Selected S10FS-3 Fuel Elements," Atomics International Technical Data Record NAA-SR-MEMO-12341 (June 1967) (28 pages)

Go67c K. G. Golliher, et al., "SNAP Engineering Test Facility Safety Analysis Report: Addendum for Intrinsic Supercriticality Experiments," Atomics International Technical Data Record NAA-SR-MEMO-12614 (November 1967) (80 pages)

Ki63 T. S. Kirsch, "Specifications for Alloying and Forming SNAP 10A and 2 Fuel Materials," Atomics International Technical Data Record NAA-SR-MEMO-6645, Revised (August 1963) (15 pages)

Kr67 W. E. Krupp, "Post-Irradiation Metallographic Examination of Selected S10FS-3 Reactor Components," Atomics International Technical Data Record NAA-SR-TDR-12390 (March 1967) (10 pages)

Mi65 S. Minor, L. Bixon, and D. Brinkman, "Preliminary Test Results - SNAP 10A FS-3," Atomics International Report NAA-SR-11206 (September 1965) (44 pages)

Sa63 W. Sawicky, "Fabrication Process Test and Inspection Sheets for SNAP 10A Flight System Cores 3, 4, and 5 Fuel Elements," Atomics International Technical Data Record NAA-SR-MEMO-9367 (November 1963) (46 pages)

Sa64 W. Sawicky, "SNAP 10A FS-1 Core-Fuel Element Data Packages for First Delivery," Atomics International Technical Data Record NAA-SR-MEMO-9855 (April 1964) (414 pages)

Sa64a W. Sawicky, "SNAP 10A FS-1 Core-Fuel Element Data Packages for Second Delivery," Atomics International Technical Data Record NAA-SR-MEMO-9926 (May 1964) (192 pages)

Sa64b W. Sawicky, "SNAP 10A FS-3 Core-Fuel Element Data Packages," Atomics International Technical Data Record NAA-SR-MEMO-10033 (June 1964) (171 pages)

Sa64c W. Sawicky, "SNAP 10A FS-4 Core-Fuel Element Data Packages," Atomics International Technical Data Record NAA-SR-MEMO-10208 (July 1964) (171 pages)

Sa64d W. Sawicky, "Addendum to SNAP 10A FS-1, FS-3, and FS-4 Fuel Data Packages for FS-3, FS-4, and FS-5 Systems," Atomics International Technical Data Record NAA-SR-MEMO-10815 (December 1964) (84 pages)

St67* D. W. Staub, "SNAP 10A Summary Report," Atomics International Report NAA-SR-12073 (March 1967) (237 pages)

10FS-3 SHIPPING RECORD SUMMARY

Shipping Document	Shipment Date	Al Can No.	Fuel Identification Element	Rod	Location	Fuel Masses (grams) Unclad	total U	U-235	Notes	Prob. INEL Can No.
LAE-JWA-6	5/1/67	13	E-486	668-6	2	1355	139	129	sectioned	SN-11
			E-150	608-6	28	1359	138	129		
			E-337	659-4	29	1356	137	128		
			E-352	664-4	37	1354	135	126		
			E-350	664-3	30	1354	135	126		
			E-332	661-5	10	1357	139	129	*see note 1*	
	Shipment totals:		6	elements		8135	823	767		
	Transfer fuel data totals:		6	elements		8135	823	768 *see note 2*		
LAE-JWA-7	5/3/67	16	E-439	682-1	33	1357	138	129		SN-14
			E-419	677-1	24	1357	139	129		
			E-390	666-1	25	1356	138	129		
			E-364	661-2	36	1358	140	130		
			E-393	663-5	26	1356	138	129		
			E-484	674-2	4	1357	140	130		
	Shipment totals:		6	elements		8141	833	776		
	Transfer fuel data totals:		6	elements		8141	833	776		
LAE-JWA-8	5/5/67	15	E-336	660-2	18	1357	138	129		SN-13
			E-334	659-3	19	1356	137	128		
			E-456	641-1	11	1357	140	130		
			E-349	664-2	31	1354	135	126		
			E-421	678-2	15	1357	139	129	*see note 3*	
			E-345	658-5	20	1356	137	128		
	Shipment totals:		6	elements		8137	826	770		
	Transfer fuel data totals:		6	elements		8137	826	770		
LAE-JWA-9	5/9/67	14	E-338	659-5	32	1355	137	128	*see note 4*	SN-12
			E-365	661-1	14	1358	140	130		
			E-136	615-5	21	1357	136	127	*see note 5*	
			E-348	659-2	35	1355	137	128		
			E-333	661-6	22	1358	140	130		
			E-479	673-2	23	1355	139	129		
	Shipment totals:		6	elements		8138	829	772		
	Transfer fuel data totals:		6	elements		8138	829	772		
LAE-JWA-10	5/11/67	17	E-441	682-4	16	1357	138	129		SN-15
			E-440	682-2	6	1357	138	129		
			E-528	595-5	5	1364	137	128		
			E-435	678-6	12	1357	139	129		
			E-190	613-4	27	1358	137	128		
			E-442	683-6	34	1357	139	129	*see note 6*	
	Shipment totals:		6	elements		8150	828	772		
	Transfer fuel data totals:		6	elements		8150	828	772		

Shipping Document	Shipment Date	Al Can No.	Fuel Identification Element	Rod	Location	Fuel Masses (grams) Unclad	total U	U-235	Notes	Prob. INEL Can No.
LAE-JWA-11	5/13/67	19	E-529	583-1	3	98	10	9	pieces	SN-18
	(pieces of elements		E-422	678-3	1	84	9	8	pieces	
	from next group)		E-331	661-4	9	70	7	7	pieces	
	Shipment totals:		3	pieces		252	26	24		
	Transfer fuel data totals:		3	pieces		252	26	24		
LAE-JWA-12	5/16/67	18	E-529	583-1	3	955	95	89	declad/sect.	SN-18
	(see element pieces		E-331	661-4	9	954	98	91	declad/sect.	
	in shipment above)		E-422	678-3	1	885	90	84	declad/sect.	
			E-482	680-5	8	1344	137	128	sectioned	
			E-459	680-6	17	1358	139	129		
			E-525	610-5	7	1365	139	129		
			E-418	677-2	13	1357	139	129		
	Shipment totals:		7	elements		8218	837	779		
	Transfer fuel data totals:		7	elements		8218	837	779		
10FS-3 reactor totals:			37	elements		49171	5002	4660		
Transfer fuel data totals:			37	elements		49171	5002	4661 *see note 2*		

References: Data transcribed from archived copies of original shipping records
Mass totals and element IDs compared with 10FS-3 fuel data summary shipping sheet
Fuel element identifications correlated with post-irradiation examination document Go67a (NAA-SR-MEMO-12338) and final (addendum) data package Sa64d (NAA-SR-MEMO-10815); note post-irradiation reports provide best identification of in-core elements because of redistribution between 10A cores after first FS-1 tests.
Fuel rod identifications from QA data packages Sa64 (E-100's), Sa64a (E-300's and 400's), and Sa64d (E-500's).

Notes:

1. Shipping document incorrectly identifies as E-322; in-core elements identified by cross-comparison with Go67a
2. For Al Can No. 3, 1-g discrepancy between U-235 mass sum and fuel data summary sheet entry not resolved
3. Shipping document incorrectly identifies as E-42; in-core elements identified by cross-comparison with Go67a
4. Shipping document incorrectly identifies as E-388; in-core elements identified by cross-comparison with Go67a
5. Shipping document incorrectly identifies as E-316; in-core elements identified by cross-comparison with Go67a
6. Shipping document incorrectly identifies as E-440; in-core elements identified by cross-comparison with Go67a

10FS-3 Fuel Data transfer summary sheet has two additional typos:
Total material available includes 33 (not 34) complete elements
Element list for ID #18 should list E529 instead of E524

9. SNAP-8 EXPERIMENTAL REACTOR (S8ER)

Date went critical:	May 1963 (Di67)
Initiation of testing:	November 22, 1963 (O167)
Date of shutdown:	April 15, 1965 (O167)
Thermal power:	600 kWt (Di67)
Thermal energy:	5.1 x 10^6 kWt-h (Sw69, Di67)
Time at power and temperature:	365 d > 400 kWt & 100 d at 600 kWt, at 1300°F (Di67)
Operating location:	SSFL Building 010

9.1 S8ER GENERAL DESCRIPTION

The SNAP-8 reactor program represented an extension of zirconium hydride technology from the 50-kWt, 1200°F, one-year life design of the SNAP-2 reactor to a 600-kWt, 1300°, 10,000-hour life reactor. [Di67] In order to achieve the higher power and temperature, the core size was increased and the fuel diameter reduced. The S8ER reactor consisted of a homogeneously combined fuel and moderator (a hydrided zirconium-uranium alloy) with movable beryllium reflectors for control and liquid NaK as the heat transfer medium. The subsystem was a direct extension of the technology developed and tested for the lower powered SNAP-2 reactors. The reactor contained 211 fuel-moderator elements arranged in a triangular pattern on 14.5-mm (0.57-in.) centers. The assembly resulted in a hexagonal right cylinder, with slightly chopped corners, 20.3 cm (8 in.) across flats and 36.75 cm (14.47 in.) long.

Each fuel element consisted of an individual fuel rod, a hydrogen diffusion barrier containing burnable poison, exterior cladding, end caps, and grid plate indexing pins. The rods were composed of enriched (93.15% ^{235}U) fuel and hydrogen moderator in the form of a solid zirconium-uranium alloy (10% uranium) hydrided to a hydrogen density of 6 x 10^{22} atoms/cm^3. The rods were 13.5 mm (0.532 in.) in diameter and 0.356 m (14 in.) long. The hydrogen diffusion barrier was a 0.0776-mm- (0.003-in.-) thick ceramic coating applied on the inside surface of the cladding to prevent the loss of the hydrogen moderator. A Sm_2O_3 burnable poison (1.3 mg/cm) was added to the ceramic coating to compensate for the reactivity loss caused by the buildup of fission-product poison. The cladding consisted of 13.7-mm- (0.43-in.-) ID Hastelloy N (a nickel-base alloy) tubing with a 0.25-mm (0.01-in.) wall thickness. The cladding envelope was completed by Hastelloy N end caps seal-welded to the tubing. The manufacturing process ensured that a continuous diffusion barrier was provided on the inside surface of the cladding. The finished fuel element was 14.2 mm (0.56 in.) in diameter and 0.368 m (14.47 in.) long (excluding the positioning pins located at the ends of each element).

Post-test examination of the S8ER revealed that 79% of the 211 fuel elements had cladding cracks, even though the reactor had met or exceeded all test objectives. Detailed examination of selected fuel elements indicated that the cracks resulted primarily from over-temperature swelling of the UZrH fuel, coupled with the low irradiated ductility of the Hastelloy N alloy cladding material.

General descriptions of the S8ER are given in references Di67, AN76, and An83, the latter including a design data summary table and schematic drawing. Schematic drawings are also given in Mi66.

9.2 S8ER REACTOR AND FUEL IDENTIFICATION

Reactor: SNAP-8 Experimental Reactor (S8ER)

Core fuel Identification: S8ER (same as reactor name)

Reactor designer, builder, operator: Atomics International
Reactor owner: Atomic Energy Commission (now DOE)

Fuel designer, builder: Atomics International
Fuel owner: Atomic Energy Commission (now DOE)

9.3 S8ER DESIGN REACTOR PHYSICS AND CORE DESCRIPTION

Fuel elements/assemblies: 211
Hu64 211 fuel elements were in the core; 215 fuel elements were delivered

Design neutron flux:
Design neutron energy: Mean neutron fission energy near 0.26 eV (Sw69)
Core map and element locator:
O167 Core maps showing location of elements selected for detailed post-irradiation examination
An original (legible) vellum of the core loading was located in a separate source data file

9.4 S8ER FUEL FABRICATION INFORMATION

Fuel specifications and drawings:
Ca63 Page 4 tabulates the fuel element engineering design drawings
Hu64 Table 1 abstracts fuel element specification requirements
(most drawings and specs available through Rocketdyne Advanced Power Systems Technical Data)

Drawing 7570-18011 "Fuel Element Ass'y" (04/62)
Drawing 7570-18010 "Fuel Rod" (02/62)

Fuel fabrication and quality control:
Ca63 Manufacturing planning document: fabrication process flow sheet forms (Manufacturing Travelers). Completed flow sheets are not available.
Na62 Compilation of original certified inspection and test reports for fuel elements delivered for the S8ER core (215 elements delivered for the 211-element core). Contains dimensional, chemical, and hydrogen leak data; non-conformance information, manufacturing traceability, and nondestructive test results for all components of each fuel element. This includes cladding (end caps, coating, tubing) and fuel material.
Na63 Quality control program flow sheet forms used for fabrication of the S8ER fuel elements. The body of this report lists all engineering drawings and specifications used for the fuel element fabrication.
Hu64 Fuel alloy impurities
Ca64 Overall summary of S8ER fuel fabrication (including photographs of equipment and parts)

Fuel cladding, inner coatings, and other non-fuel materials:

Na63 Includes quality control for fuel cladding, etc. as well as fuel rods.

Hu64 Description of fuel element cladding, coating (AI 8763D), end caps. Properties of cladding raw materials.

Bu64 Qualitative composition of AI 8763D ceramic coating

Potential fuel chemical reactions:

Potential RCRA-regulated materials:

Quantities of materials:

See shipping spreadsheets (fuel rods, U, ^{235}U)

Hu64 Extended table correlates fuel element serial number with fuel rod serial number. Provides net weights, hydrogen leak rates, dimensions, uranium enrichment, carbon and hydrogen concentrations, etc.

Fuel assembly identification methods:

Fuel element and rod identification marking methods and locations are given on the engineering design drawings.

9.5 S8ER OPERATIONAL HISTORY

Core Operating History:

O167 Testing occurred from November 22, 1963 to April 15, 1965, for a total of 12,000 hours, with 5×10^6 kWh energy release, of which 6760 hours were between 400 and 600 kWt.

O167 S8ER operating history timeline (Fig. 1)

Core neutron flux profile:

Operation abnormalities:

O167 As of April 28, 1964, there were 37 reactor scrams. The scrams were associated with test support equipment and instrumentation not intended for use in an actual flight system.

Core shut-down date:

O167 April 15, 1965

Core discharge date:

Mi66 S8ER cask received at AI Hot Laboratory for post-irradiation examination July 28, 1965

9.6 S8ER POST-IRRADIATION PROPERTIES AND EXAMINATION

Pe66 S8ER fuel element post-irradiation evaluation review of the entire test program

Visual observations:

Mi66 Highly detailed screening examination data for all 211 core fuel elements, including visual, macro photographic, gamma scanning, mass, and some dimensional data

Mi66a Comprehensive detailed examination data for selected S8ER fuel elements and fuel material. Includes fuel element and fuel material weights, length and diameter measurements, gamma scans, longitudinal distortion, hydrogen leak testing, density measurements (fuel material), fission gas analyses, burnup analyses, fuel metallography for microstructure, coating adherence testing, cladding tensile testing and metallography

Mi66b Summary of AI66a book volume vs. core location for Kollmorgan and stereo photographs
AI66a Complete set of Kollmorgan and stereo examination photographs (over 2000) taken of all of the core elements
AI66b Source screening examination stereo data sheets for all fuel elements
O167 Summarizes, and provides analysis of, highly detailed information provided in Mi66 and Mi66a. Table 1 summarizes fuel element cracks by ring number (167 of 211 elements, 79%) had observable cracks. Locations summarized in core layout drawing in Fig. 5. The text provides an analysis of all of the data with extensive photomicrographs, visual observations, detailed data measurements, and methods of inspection and testing. Appendix I includes selected fabrication data. Appendix II contains photographs of the condition of every fuel element examined. Data summaries, including photomicrographs of all elements examined, are given in Appendix III; these data include diameter changes, density, hydrogen content, sample locations for burnup analysis etc., and relative gamma activity plots.

Dimensions:

Mi66 Some dimensional data
AI66c Source screening examination length measurement data sheets for all fuel elements
AI66e Source screening examination diameter measurement data sheets for all fuel elements
Mi66a Detailed dimensional data on selected de-clad fuel material
O167 Post-irradiation dimensional analyses
AI65 Source data sheets of fuel element post-irradiation dry weights

Metallurgical states of materials:

Le66 Studies of dehyriding and rehydriding of unirradiated fuel to provide a baseline for irradiated fuel studies (may be useful if it becomes necessary to dehydride the SNAP fuels)
Mi66a Fuel metallography for microstructure; includes color photomicrographs
Mi67 Photomicrographs and analysis of metallurgical structure shown
O167 Detailed photomicrographs

Cladding defects:

Mi66a Cladding tensile testing; cladding metallography
O167 Detailed examination of cladding cracks

Corrosion or chemical contamination:

Mi66a Coating adherence testing (to cladding)
O167 Ceramic coating adhesion to fuel rods, including contamination information (Table 5)

Hydrogen effects:

AI66f Source screening examination hydrogen analysis data sheets for all fuel elements
Mi66a Hydrogen leak testing and hydrogen analysis of fuel rod samples
O167 Average hydrogen loss (Table 11)

Burnup:

Mi66a Burnup analyses for selected elements (24 samples from 8 rods). Ranges from 0.04 to 0.22 metal-atom % by mass spectrometer analysis at INEL, and 0.099 to 0.234 metal-atom % by radiochemical analysis.
O167 Burnup analyses (Table 10 and Fig. 50)

Fission products:

Mi66a Fission gas analysis

O167 Fission product release fractions (Table 16), based on samples of the NaK coolant during three periods of S8ER operation

Mi66a Gamma scans across fuel elements of ^{95}Zr photopeak

AI66d Source screening examination gamma scanning data sheets for all fuel elements

Transuranics:

(can be calculated from available reactor information)

Activation products:
Decay heat generation:
Radiation level decay curves:
Alpha contamination:

(not relevant for the declad fuels)

9.7 S8ER ON-SITE STORAGE HISTORY

Post-irradiation examinations took place in the AI Hot Laboratory (AIHL) following reactor shut-down. Reactor power test operations terminated on April 15, 1965 and the S8ER cask was received at the AIHL on July 28, 1965 (Mi66). The only fuel disassembly that took place was for hot cell examination, which is covered in the post-irradiation examination section. SNAP fuel elements were subsequently stored in air at the RMDF in dry storage vaults in preparation for shipment to INEL. There were no wet-storage facilities available at Atomics International. No detailed storage history documentation is available.

9.8 S8ER SHIPPING RECORDS

The shipping records for the S8ER fuel shipments to INEL were reviewed in detail. The original records are difficult to read, and were transcribed onto the S8ER spreadsheet that is reproduced at the end of this section. The spreadsheet lists the core location, fuel rod identification, unclad fuel rod mass, total uranium mass, and ^{235}U mass for each fuel element, plus the shipping document number, shipment date, Atomics International can number, and probable INEL aluminum can number. The latter is base on document copies received from INEL as part of the current review activity. A total of 211 irradiated S8ER fuel elements were shipped to INEL.

The fuel element and fuel rod identifications were compared with QA document Hu64. Transcription errors were found for two fuel element numbers in the shipping documents and are noted in the spreadsheet. Fuel element locations in the core were obtained from O167 and an original velum of the core loading found in archived Atomics International Hot Laboratory files. The mass totals shown on the spreadsheet were calculated from the individual recorded fuel element data. No summary fuel data sheet is available for mass sum comparisons.

9.9 S8ER REFERENCES

AI65 Atomics International, "S8ER Screening Examination Dry Weight Data Sheets in Sequence from Ring I to Ring IX," Atomics International binder of source data sheets

AI66a Atomics International Hot Laboratory Staff, "[S8ER] Post Irradiation Screening Examination Photographs," Atomics International Technical Data Records NAA-SR-MEMO-12011, Volumes I to XIV (1966) *(volumes not labeled with NAA number)*

AI66b Atomics International Hot Laboratory Staff, "S8ER Screening Examination Stereo Data Sheets in Sequence from Ring I to Ring IX," Atomics International source data book (1966)

AI66c Atomics International Hot Laboratory Staff, "S8ER Screening Examination Length Measurement Data Sheet in Sequence from Ring I to Ring IX," Atomics International source data book (1966)

AI66d Atomics International Hot Laboratory Staff, "S8ER Screening Examination Gamma Scanning Data Sheets in Sequence from Ring I to Ring IV, from Ring V to Ring VII, *and* from Ring VIII to Ring IX," Atomics International source data books, 3 volumes (1966)

AI66e Atomics International Hot Laboratory Staff, "S8ER Screening Examination Diameter Measurements Data Sheets in Sequence from Ring I to Ring V, *and* from Ring VI to Ring IX," Atomics International source data books, 2 volumes (1966)

AI66f Atomics International Hot Laboratory Staff, "S8ER Screening Examination Hydrogen Analysis Data Sheets in Sequence from Ring I to Ring IX," Atomics International source data book (1966)

AN76 *Anonymous*, "Zirconium Hydride (SNAP) Reactor Space Power Programs. Summary of Major Activities and Accomplishments," Rockwell International Energy Systems Group (June 1976)

An83 R. V. Anderson, et al., "Space Reactor Electric Systems. Subsystem Technology Assessment," Rockwell International Energy Systems Group Report ESG-DOE-13398 (March 1983)

Bu64 G. F. Burdi, Ed., "SNAP Technology Handbook. Volume II. Hybrid Fuels and Claddings," Atomics International Report NAA-SR-8617, Volume II (November 1964) (129 pages)

Ca63 D. C. Campbell, et al., "SNAP 8ER Fabrication Process Sheets and Auxiliary Forms," Atomics International Technical Data Record NAA-SR-MEMO-8273 (February 1963) (41 pages)

Ca64 D. C. Campbell, et al., "SNAP 8 Experimental Reactor Fabrication and Engineering," Atomics International Report NAA-SR-8430 (March 1964) (55 pages)

Di67 H. M. Dieckamp, "Nuclear Space Power Systems," Atomics International (September 1967) (unpublished book, 388 pages)

Hu64 H. J. Hubbell and W. E. Nagel, "SNAP 8 Experimental Reactor Fuel Element Acceptance Testing," Atomics International Report NAA-SR-8589 (March 1964) (63 pages)

Le66 S. K. Lee, "A Study of Dehydriding of S8ER Fuel," Atomics International Technical Data Record NAA-SR-MEMO-12197 (October 1966) (20 pages)

Mi66 K. J. Miller, "Post-Irradiation Screening Examination of the S8ER Fuel Elements," Atomics International Technical Data Record NAA-SR-MEMO-11880 (March 1966) (299 pages)

Mi66a K. J. Miller, "Post-Irradiation Detailed Examination of Selected S8ER Fuel Elements," Atomics International Technical Data Record NAA-SR-MEMO-12165 (1966) (color copy, 304 pages)

Mi66b K. J. Miller, "S8ER Screening Examination Photograph Books," Atomics International Internal Letter to G. W. Meyers dated 17 June 1966

Mi67 K. J. Miller, "Electron Microscopy Analysis of Irradiated S8ER and NAA 115-1 Fuel," Atomics International Technical Data Record NAA-SR-MEMO-12368 (March 1967) (51 pages)

Na62 W. E. Nagel, "S8ER Fuel Element Data Package," Atomics International Technical Data Record NAA-SR-MEMO-7623 (October 1962) (1078 pages)

Na63 W. E. Nagel, "S8ER Fuel Element Quality Control - Upgrading Point Tests and Inspections," Atomics International Technical Data Record NAA-SR-8283 (February 1963) (44 pages)

Ol67* P. S. Olson, K. J. Miller, and E. J. Donovan, "Postoperation Evaluation of Fuel Elements from the SNAP 8 Experimental Reactor," Atomics International Report NAA-SR-12029 (September 1967) (223 pages)

Pe66 H. Pearlman, et al., "SNAP 8 Experimental Reactor Fuel Element Behavior: Atomics International Task Force Review," Atomics International Report NAA-SR-MEMO-12210 (November 1966)

Sw69* L. D. Swenson, "SNAP 8 Development Reactor Nuclear Analysis," Atomics International Report AI-AEC-12864 (October 1969) (75 pages)

Shipping Document	Shipment Date	Al Can No.	Fuel Identification Element	Rod	Location	Fuel Masses (grams) Unclad	total U	U-235	Notes	INEL Can No.
LAE-JWA-15	4/8/68	7	E-458	562D-1	VII-2	309	30	28		SN-15
			E-323	565B-2	VII-35	310	31	29		
			E-508	563C-2	VIII-24	312	31	29		
			E-506	555B-2	VII-32	310	31	29		
			E-421	536C-4	VI-13	308	30	28		
			E-353	569D-3	VI-14	310	31	29		
			E-530	557D-2	VI-15	310	31	29		
			E-481	582C-3	VI-16	309	30	28		
			E-429	568A-1	VI-17	310	30	28		
			E-501	538C-4	VI-18	308	31	29		
			E-422	556B-4	VI-19	309	31	29		
			E-526	556A-3	VI-20	309	30	28		
			E-348	562C-1	VI-21	305	30	28		
			E-477	558A-2	VI-22	310	30	28		
			E-532	582B-4	VI-23	308	30	28		
			E-456	565A-1	VI-26	310	30	28		
			E-400	595A-5	VI-28	311	31	29		
			E-531	540D-4	VI-30	308	30	28		
	Shipment totals:		18	elements		5566	548	512		
LAE-JWA-16	4/11/68	4	E-488	534D-4	VII-23	308	30	28		SN-13
			E-449	582A-3	VII-25	309	30	28		
			E-470	537A-2	VII-26	307	30	28		
			E-283	537A-4	VII-30	310	30	28		
			E-446	561D-1	VII-31	310	30	28		
			E-457	568A-4	VII-33	310	31	29		
			E-191	558D-2	IX-12	310	30	28		
			E-478	582B-3	VIII-5	309	30	28		
			E-274	557A-3	VIII-4	310	31	29		
			E-141	563D-2	VIII-22	311	31	29		
			E-389	564A-2	VII-3	310	30	28		
			E-440	549D-4	VII-9	311	30	28		
			E-463	563B-3	VII-11	310	31	29		
			E-516	562A-4	VII-16	310	30	28		
			E-310	565C-3	VII-18	310	30	28		
			E-384	565B-4	VII-20	310	30	28		
			E-437	560B-3	VII-21	310	31	29		
			E-367	560B-4	VII-22	310	30	28		
	Shipment totals:		18	elements		5575	545	509		
LAE-JWA-17	4/16/68	3	E-372	563A-2	VIII-21	310	31	29		SN-12
			E-402 *	563A-3	VIII-26	309	31	29		
			E-187	563C-3	VIII-28	310	31	29		
			E-439	567D-2	VIII-30	310	30	28		
			E-174	572C-3	VIII-32	310	31	29		
			E-433	562D-4	VIII-34	310	31	29		
			E-153	565B-1	VIII-35	310	30	28		
			E-150	564B-3	VIII-37	310	31	29		
			E-286	532D-2	VIII-39	309	30	28		
			E-154	567B-3	VIII-40	310	30	28		
			E-462	559D-1	VIII-41	305	30	28		
			E-247	571A-4	VIII-42	310	30	28		
			E-475	568B-4	VII-1	310	31	29		
			E-431	558C-4	VII-4	309	30	28		
			E-483	571D-2	VII-6	310	31	29		
			E-403	558A-1	VII-7	310	31	29		
			E-359	555C-4	VII-8	310	31	29		
			E-441	550C-2	VII-12	310	31	29		
	Shipment totals:		18	elements		5572	551	515		

* Shipping document, listing element number as E-412, is incorrect

Shipping Document	Shipment Date	Al Can No.	Fuel Identification Element	Rod	Location	Fuel Masses (grams) Unclad	total U	U-235	Notes	INEL Can No.
LAE-JWA-18	4/18/68	11A	E-412	558D-3	VIII-8	311	31	29		SN-17
			E-415	562C-4	VIII-9	310	31	29		
			E-430	562D-2	VIII-12	310	30	28		
			E-397	549D-2	VIII-13	310	31	29		
			E-479	559C-2	VIII-14	310	30	28		
			E-451	568B-3	VIII-15	309	31	29		
			E-272	561C-4	VIII-16	311	31	29		
			E-452	571A-2	VIII-17	310	31	29		
			E-453	582B-1	VIII-19	308	30	28		
			E-411	557B-1	VIII-20	309	30	28		
			E-428	567C-1	V-16	310	30	28		
			E-499	556C-3	V-17	309	31	29		
			E-417	562A-1	V-20	310	31	29		
			E-322	555D-2	VIII-7	310	31	29		
			E-529	570D-2	IV-8	310	31	29		
			E-445	549C-2	VI-9	309	31	29		
			E-360	582B-2	VII-14	309	30	28		
			E-349	570A-2	VI-11	310	31	29		
	Shipment totals:		18	elements		5575	552	516		
LAE-JWA-19	4/23/68	6	E-503	556-D1	IX-2	305	30	28		SN-14
			E-192	557-C3	IX-37	310	31	29		
			E-419	549-C1	VII-5	310	31	29		
			E-484	570-D1	VII-10	310	31	29		
			E-454	567-A3	VII-13	310	31	29		
			E-413	567-B1	VIII-31	310	30	28		
			E-523	555-A2	VII-15	310	30	28		
			E-504	561-C1	VII-24	312	31	29		
			E-466	567-C3	VII-34	310	30	28		
			E-343	555-D3	VI-1	310	31	29		
			E-513	582-C4	VI-2	302	31	29		
			E-505	555-A4	VI-4	310	31	29		
			E-177	558-A4	VI-5	310	31	29		
			E-369	561-A4	VI-6	310	31	29		
			E-536 *	557-A1	VI-7	310	30	28		
			E-357	555-A1	VI-8	310	31	29		
			E-517	559-B4	VI-10	311	31	29		
			E-448	549-D3	VI-12	311	30	28		
			* Shipping document appears to list as E-536; should be E-356							
	Shipment totals:		18	elements		5571	552	516		
LAE-JWA-20	4/25/68	1	E-249	568C-2	IX-8	310	30	28		SN-11
			E-212	535A-1	IX-13	308	30	28		
			E-471	556D-3	IX-16	310	31	29		
			E-281	530D-1	IX-18	308	30	28		
			E-146	570A-3	IX-19	310	30	28		
			E-164	569B-3	IX-20	308	31	29		
			E-149	570B-2	IX-26	310	31	29		
			E-245	558D-4	IX-27	310	30	28		
			E-237	570B-1	IX-32	309	30	28		
			E-383	571A-3	IX-35	310	30	28		
			E-291	555C-2	IX-40	310	31	29		
			E-137	569B-4	IX-43	309	30	28		
			E-292	568D-2	IX-44	305	31	29		
			E-280	556A-4	IX-45	309	30	28		
			E-311	564B-2	IX-46	310	31	29		
			E-313	550D-1	IX-47	306	31	29		
			E-350	561C-3	VIII-1	311	31	29		
			E-424	567A-2	VIII-3	310	31	29		
	Shipment totals:		18	elements		5563	549	513		

Shipping Document	Shipment Date	Al Can No.	Fuel Identification Element	Rod	Location	Fuel Masses (grams) Unclad	total U	U-235	Notes	INEL Can No.
LAE-JWA-21	5/6/68	8	E-330	555B-4	V-1	310	31	29		SN-16
			E-289	570A-4	V-2	310	31	29		
			E-315	560C-1	V-3	310	30	28		
			E-201	569D-2	V-4	310	31	29		
			E-443	530A-1	V-6	307	31	29		
			E-482	560C-4	V-7	310	31	29		
			E-361	571B-2	V-8	310	31	29		
			E-370	557D-1	V-9	309	30	28		
			E-518	572A-2	V-11	310	31	29		
			E-487	582A-2	V-12	309	30	28		
			E-472	561C-2	V-13	310	31	29		
			E-496	532C-2	V-14	308	31	29		
			E-461	560B-2	V-15	310	30	28		
			E-489	568C-1	V-18	310	30	28		
			E-427	568A-3	V-19	310	31	29		
			E-434	550D-3	V-22	310	30	28		
			E-392	559A-3	V-23	310	31	29		
			E-509	564A-1	V-24	310	31	29		
	Shipment totals:		18	elements		5573	552	516		
LAE-JWA-22	5/8/68	9	E-420	558C-3	IX-4	310	30	28		SN-16
			E-319	555C-3	IX-21	310	31	29		
			E-423	572D-1	VIII-11	310	31	29		
			E-490	572B-1	VIII-27	310	30	28		
			E-308	557D-4	IV-2	310	31	29		
			E-271	559C-4	IV-3	305	31	29		
			E-303	565A-4	IV-4	303	30	28		
			E-522	568A-2	IV-5	310	30	28		
			E-408	557C-4	IV-6	310	30	28		
			E-155	569C-4	IV-7	310	31	29		
			E-278	565A-2	IV-9	310	31	29		
			E-336	559A-2	IV-10	310	31	29		
			E-352	569D-4	IV-11	310	30	28		
			E-416	555D-1	IV-14	309	30	28		
			E-521	567D-3	IV-15	310	30	28		
			E-425	571D-3	IV-16	310	30	28		
			E-500	557B-5	IV-17	307	30	28		
			E-511	571D-1	IV-18	309	31	29		
	Shipment totals:		18	elements		5563	548	512		
LAE-JWA-23	7/29/68	10	E-464	569C-1	IX-14	310	31	29		SN-19
			E-231	571C-3	VIII-33	311	31	29		
			E-485	571B-4	VI-27	310	30	28		
			E-520	560A-3	VI-29	311	31	29		
			E-447	582A-4	VII-17	309	30	28		
			E-394	565B-3	VI-3	310	31	29		
			E-442	563B-2	VIII-23	310	31	29		
			E-347	567C-4	V-21	310	31	29		
			E-436	567A-4	IX-36	310	30	28		
			E-337	569B-2	III-1	310	31	29		
			E-426	567C-2	III-4	310	31	29		
			E-493	564A-3	III-6	311	31	29		
			E-524	555B-1	III-8	309	31	29		
			E-329	557A-2	III-10	309	30	28		
			E-287	531C-3	III-11	308	30	28		
			E-346	565A-3	III-12	310	30	28		
			E-379	569A-2	II-1	310	30	28		
			E-514	569D-1	II-3	309	30	28		
	Shipment totals:		18	elements		5577	550	514		

Shipping Document	Shipment Date	Al Can No.	Fuel Identification			Fuel Masses (grams)			Notes	INEL Can No.
			Element	Rod	Location	Unclad	total U	U-235		
LAE-JWA-24	8/21/68	13	E-304	571D-4	IX-31	310	30	28		SN-19
			E-176	562A-2	IX-24	310	31	29		
			E-244	572B-4	IX-39	310	30	28		
			E-527	564B-1	IX-38	309	31	29		
			E-288	571C-4	IX-23	311	30	28		
			E-197	550B-3	IX-22	310	30	28		
			E-510	556B-3	III-5	309	30	28		
			E-450	563C-1	V-10	311	30	28		
			E-268	572C-1	IX-30	310	31	29		
			E-318	567D-4	IX-3	310	30	28		
			E-358	565C-1	II-2	310	30	28		
			E-396	571B-1	VIII-10	310	31	29		
			E-109	550B-2	VIII-2	310	30	28		
			E-182	571C-2	IX-11	306	31	29		
			E-175	570C-4	VIII-38	310	31	29		
			E-307	559C-1	III-7	310	31	29		
			E-533	531D-4	IX-34	310	30	28		
			E-236	569C-3	IX-48	310	30	28	DDE	
	Shipment totals:		18	elements		5576	547	511		
LAE-JWA-26	9/18/68	14	E-474	572A-3	II-4	140	14	13	DDE	SN-20
			E-378	549B-1	IX-28	228	22	20		
			E-293	531C-4	IX-29	176	17	16		
			E-476	572B-2	VII-19	74	7	7	DDE	
			E-486	582C-2	III-3	276	27	25	DDE	
			E-309	559C-3	V-5	162	16	15	DDE	
			E-332	565C-2	VIII-25	129	13	12	DDE	
			E-206	572C-4	IX-15	239	24	22	DDE	
			E-188	532D-3	VIII-6	121	12	11	DDE	
			E-525	550A-1	VII-28	140	14	13	DDE	
			E-111	531C-1	IX-42	111	11	10	DDE	
			E-342	549B-2	I-1	250	25	23	DDE	
			E-519	564B-4	IV-1	173	17	16	DDE, IS	
			E-398	559B-1	IV-13	57	6	6		
			E-232	565C-5	VI-24	112	11	10		
			E-432	567B-2	VIII-29	167	16	15	DDE	
	irrad. cpsl		117-1	(NAA in-pile expt.)		515	51	48	pieces	SN-20
	irrad. cpsl		117-1	(NAA in-pile expt.)		86	9	8	pieces	SN-20
	irrad. cpsl		115-2	(NAA in-pile expt.)		117	12	11	pieces 3-6	SN-20
	irrad. cpsl		115-2	(NAA in-pile expt.)		155	15	14	pieces 1,2	SN-20
	irrad. cpsl		82-1	(NAA in-pile expt.)		1149	118	110		SN-20
	Shipment totals:		16	elements		4577	457	425		
			5	in-pile capsules		2022	205	191		
LAE-JWA-30	2/26/69	12	E-351	582A-1	IV-12	275	27	25	IS	SN-19
			E-491	567A-1	VII-29	288	28	26	TT	
			E-340	565D-3	IX-7	288	28	26	TT	
			E-335	559A-1	II-5	313	31	29	TT	
			E-316	571C-1	VIII-36	289	29	27	TT	
			E-341	565C-4	IX-10	266	26	24		
			E-302	572B-3	IX-5	299	29	27	DDE	
			E-388	565D-4	VI-25	299	29	27	IS	
			E-492	558D-1	III-2	265	26	24	IS	
			E-181	558E-4	II-6	282	28	26	TT	
	Shipment totals:		10	elements		2864	281	261		

Shipping Document	Shipment Date	Al Can No.	Fuel Identification Element	Rod	Location	Fuel Masses (grams) Unclad	total U	U-235	Notes	INEL Can No.
LAE-JSB-02	6/20/73	10 (S8DR)	E-339	562A-3	VII-36	310.00	30.50	28.41		Al-10
			E-410	561D-2	VII-27	299.00	29.60	27.57	IS	
	packaged		E-515	560A-2	VIII-18	312.00	30.48	28.39		
	with S8DR		E-528	531A-1	III-9	128.00	12.68	11.82		
	can #10		E-171	549C-4	IX-6	100.00	9.88	9.20		
	Partial shipment totals:		5	elements		1149.00	113.14	105.39		
S8ER reactor totals:			211	elements		64301.00	6345.14	5925.39		

References: Data transcribed from archived copies of original shipping records
Fuel element and fuel rod serial numbers cross-checked with QA document Hu64 (NAA-SR-8589, 03/01/64)

Notes: DDE = detailed destructive examination, TT = tensile tests, IS = interface studies
Fuel element identifications from shipping records compared with, and in a few cases corrected to, tabulation in QA document Hu64
No S8ER Fuel Data sheet available for mass sum comparisons

o-s8er.xls 8/30/94

10. SNAP-8 DEVELOPMENTAL REACTOR (S8DR)

Date went critical:	June 1968
Date of shutdown:	December 1969 (Li70, Li71)
Thermal power:	600-1000 kWt
Thermal energy:	4.3×10^6 kWt-h
Time at power and temperature:	6688 h at 600 kWt (5575 h at 1200°F, 1670 h at 1300°F), 429 h at 1000 kWt (1150°F)
Operating location:	SSFL Building 059

10.1 S8DR GENERAL DESCRIPTION

The SNAP-8 developmental reactor assembly included the fuel elements, the core vessel assembly, and the reflector and control components assembly. The S8DR core contained 211 fuel elements cooled by eutectic NaK flowing axially between them. Each fuel element consisted of a cylindrical rod of fuel moderator, clad in Hastelloy N tubing.

The fuel-moderator material was composed of a hydrided alloy of zirconium and fully enriched (93.15 ± 0.15 % ^{235}U). The uranium content constituted 10.5 ± 0.3 wt. % of the unhydrided alloy. This material also contained 0.15 ± 0.05 wt. % carbon, added in the form of zirconium carbide (ZrC) prior to the initial arc melting of the zirconium-uranium alloy. This carbon additive resulted in a refinement of the alloy grain structure and a consequent improvement in the production yield during the hydriding process.

The fuel rods were hydrided to hydrogen atom densities between 5.97 and 6.15×10^{22} atoms/cm^3, with an average for all 211 rods of 6.05×10^{22}. At the operating temperatures characteristic of S8DR, hydrogen pressures in the range to 0.1 or 0.2 MPa were produced in the annulus between the fuel rod and the cladding. To minimize hydrogen loss from the fuel element, the interior surface of the cladding was coated with a ceramic material. This coating reduced the leak rate of hydrogen through the cladding wall by a factor of about 1000. The ends of each cladding tube were enclosed by Hastelloy N end caps. These end closures were welded to the tubing. The ceramic coating on the final end closure was fused to that on the tubing to form a continuous hydrogen seal.

The active length of the S8DR fuel element (i.e., the length of the fuel rod) was 0.427 m (16.825 in.). An axial gap of 6.1 mm (0.24 in.) between the top surface of the fuel rod and the ceramic layer on the end cap was designed to accommodate axial fuel swelling and to minimize the heat transfer effect due to fission gas accumulation. The 13.4-mm- (0.529-in.-) diameter fuel rod was separated from the 0.05-mm- (0.002-in.-) thick ceramic layer on the cladding by a nominal hot diametral gap of 0.19 mm (0.007 in.) that served to accommodate fuel swelling. The Hastelloy N tubing had an ID of 13.7 mm (0.540 in.) and a nominal thickness of 0.287 mm (0.011 in.).

Changes in the fuel rod design from S8ER, including increased radial and axial clearances between fuel rods and cladding to accommodate fuel growth, and redesigned fuel element end caps, to alleviate stress cracking, were based on the post-irradiation observations of cladding cracks in the S8ER fuel elements. The S8ER post-irradiation evaluation also resulted in a change in the fuel element hydrogen permeation barrier to a ceramic coating more resistant to thermal cycling. One of the major differences between the S8ER and S8DR fuel elements was the closure end plug. For S8ER this was a two-piece assembly, while for S8DR it was a single piece.

Post-irradiation examination of S8DR showed 72 of the 211 fuel elements had small cladding cracks typical of stress-rupture. These were attributed to excessive cladding strain due to excessive fuel swelling, resulting from fuel temperatures an estimated 100°F higher than the design values in localized core regions. The higher temperatures were attributed to the agglomeration of entrained gas bubbles in the coolant and fuel element clustering. There was no damage in regions of nominal coolant flow.

General descriptions of the S8DR are given in AN76 and An83, with a schematic drawing and a summary table of reactor characteristics in An83.

10.2 S8DR REACTOR AND FUEL IDENTIFICATION

Reactor: SNAP-8 Developmental Reactor (S8DR)
Also identified as: S8DS (earlier designation, as noted in Sw67)

Core fuel Identification: S8DR (same as reactor name)

Reactor designer, builder, operator: Atomics International
Reactor owner: Atomic Energy Commission (now DOE)

Fuel designer, builder: Atomics International
Fuel owner: Atomic Energy Commission (now DOE)

10.3 S8DR DESIGN REACTOR PHYSICS AND CORE DESCRIPTION

Sw67 Reanalysis of the S8DR reactor to redesign core (after fuel manufacture but before cladding and core assembly) to incorporate information gained from S8ER test and new fuel irradiation test data

Fuel elements/assemblies: 211 fuel elements
Design neutron flux:
Design neutron energy: Mean neutron fission energy about 0.26 eV (Sw69)
Core map and element locator:
Li71 Core map (Fig. 26)
Li70 Core map showing locations of cracked fuel elements

10.4 S8DR FUEL FABRICATION INFORMATION

Fuel specifications and drawings:
Do68 Complete list of engineering drawings, specifications, test procedures, and manufacturing instructions for the S8DR fuel fabrication (Table 1, 46 documents)
Mo69 Fuel element drawings (provided as reference for post-irradiation analyses)
(Most are available from the Rocketdyne Advanced Power Systems Technical Data Center; they should be ordered by revision number)

Drawing S8DR-18001	"Fuel Element Assembly" (01/67)
Drawing S8DR-18008	"Fuel Rod-Fuel Element" (01/67)
Drawing S8DR-18011, Rev. A	Engineering release drawing - fuel element configuration

Fuel fabrication and quality control:

Ma64 Fabrication sequence and processes for vacuum arc melting and extruding the Zr-U alloy

La67 Fabrication sequence and procedures for fabricating the hydrogen barrier ceramic powder

De67 Specification for chemically etching the S8DR fuel to reduce its diameter

Mc67 Process flow sheets - S8DR final fuel element assembly (manufacturing planning sheets and procedures to be used)

Do68 List of dimensional discrepancies in accepted fuel rods (Table 2)

Do68 Appendix IX Fuel element assembly closure weld and element proof tests (Part 1), final acceptance tests and inspections (Part 2), and fuel rod inspection and traceability record (Part 3)

Appendix X Fuel element hydrogen leak rates

Appendix XI Completed fuel element release notices

Appendix XII Fuel element closure weld process and operator qualifications

Sw70 Film radiographic method for determining H/Zr in S8DR-type fuel (not sensitive enough for required application)

Fuel cladding, inner coatings, and other non-fuel materials:

Mc67a Process flow sheets - S8DR cladding tube fabrication (manufacturing planning sheets and procedures to be used)

Ma67 Process flow sheets - closure (cup) plug fabrication (manufacturing planning sheets and procedures to be used)

Do68 Appendix I Hastelloy N supplier end cap bar stock chemistry, mechanical properties, and internal soundness

Appendix II Hastelloy N supplier clad tubing chemistry, mechanical properties, and internal soundness

Appendix III Tube-to-first end cap inspection and raw material traceability data

Appendix IV Chromized cladding tube assembly inspection and traceability

Appendix V Ceramic-coated tube assembly test and inspection data

Appendix VI Ceramic coating and as-machined closure end plug test, inspection, and traceability records (incl. Sm_2O_3 and boron contents - other information was omitted due to classification)

Appendix VII Chromized closure end plug test, inspection, and traceability records

Appendix VIII Coated closure end plug test, inspection, and traceability records

Mc69 Fuel rod fabrication sequence and procedures from raw material through finish- machined and hydrided fuel alloy

Mc69a Summarizes the overall fuel element fabrication (information in the manufacturing process flow sheets, with yields, after fuel element fabrication)

Da70 Cr_2C embrittlement of Hastelloy-N cladding during chromizing operation (identifies certain elements having this problem, shown in core map)

Potential fuel chemical reactions:

Potential RCRA-regulated materials:

Quantities of materials:

See shipping spreadsheets (fuel rods, U, ^{235}U)

Mi70b, Sc71 Some fuel losses occurred during hot-laboratory cutting for post-irradiation examination, with the calculated waste assigned to scrap cans; calculated values given in Mi70b

Fuel assembly identification methods:

Ar77 Specification for marking parts and assemblies (dated 1977 but replaces a specification that may have been used for the fuel element components of several of the SNAP cores)

Drawing S8DR-18008 Calls out fuel rod identification by electric etching on the end of the small-diameter portion of the rod

10.5 S8DR OPERATIONAL HISTORY

Core Operating History:

La68, La68a, La68b S8DR Operations Manual (little relevant information)

Ha69 Selected operating parameters for 1-19-69 to 4-28-69, including reactor power and reactivity data

Li71 Summary of operating history (Fig. 6); power operating history (Fig. A-1); hydrogen loss history (Fig. A-2); inlet and outlet temperature history (Fig. A-3)

Core neutron flux profile:

Operation abnormalities:

Li71 Scrams included in operating history (Fig. 6)

Core shut-down date:

Li70 Shut down around the middle of December 1969, to investigate an observed reactivity loss rate and a fission product inventory in the NaK coolant

Core discharge date:

Li70 Allowed to cool for a few months after shut-down and delivered to the AI Hot Laboratory (AIHL) on March 22, 1970

10.6 S8DR POST-IRRADIATION PROPERTIES AND EXAMINATION

Available source data include extensive individual element-by-element fuel rod and fuel element dimensional data (length, diameter, ovality, length, bow), visual examination information, gamma scan data (strip charts), and crack data from pre- and post-irradiation examinations. These data are contained in individual file folders rather than in report format.

Tr69 Computerized selected pre-irradiation quality data from Do68 for use by post-irradiation analysts

Mi69 Preliminary planning and cost estimates for S8DR hot cell examination

Mi69a S8DR examination planning document for AI Hot Laboratory

Sc70 S8DR fuel element sample numbering system (for sectioned samples)

Li70 S8DR screening examination

Li70a S8DR screening examination weekly status reports (No. 1-16)

Mi70 Selection of S8DR fuel elements for detailed post-irradiation examination, with core map locations

Mi70a S8DR detailed examination progress reports (No. 1-10), including graphs, photos, scans

Li71 Fuel-growth data table with extensive pre- and post-test fuel properties

Visual observations:

Li70 Screening examination cell-window and in-cell photographs of fuel elements with tabulations of results

Dimensions:

Mo69 Pre-irradiation dimensional and density data for selected fuel elements
Li70 Screening examination measurements of length, diameter, and longitudinal bow of fuel elements
Mi70c Screening examination fuel rod dimensional data from neutron radiography
Li71 Fuel growth, density, dimensions

Metallurgical states of materials:

Li71 Summary of metallurgical phase information

Cladding defects:

Li70 Screening examination of clad cracking, including core map showing locations of cracked fuel elements
Li71 Summary of cladding condition
Bu70 Statistical evaluation of fuel-clad gap and crack length from S8DR fuel elements

Corrosion or chemical contamination:

Li71 Summary of locations where coating contacted fuel material
Va70 Initial report on metallic coating that developed on outside of the fuel cladding
Ei70 More detailed discussion of coating that developed on outside of the fuel cladding
Co71 Electron microprobe x-ray analysis of three irradiated S8DR fuel specimens, providing chemical analysis of glass-fuel interface

Hydrogen effects:

Li71 Fuel element hydrogen leak rates

Burnup:

Ro70 Fuel burnup calculations, core average for ^{235}U and ^{238}U

Fission products:

Li70 Screening examination gamma scans of fuel elements

Transuranics:

Ro70 Fuel burnup calculations, including estimates of ^{236}U and ^{239}Pu
(can be calculated from available reactor information)

Activation products:
Decay heat generation:
Radiation level decay curves:
Alpha contamination:
(not relevant for the declad fuels)

10.7 S8DR ON-SITE STORAGE HISTORY

The S8DR fuel elements were delivered from the reactor facility to the Atomics International Hot Laboratory (AIHL) on March 22, 1970 (Li70), about three months after shut-down. The only fuel disassembly that took place was for hot cell examination, which is covered in the post-irradiation examination section. The disassembled components were then stored at the AI's Radioactive Materials Disposal Facility (RMDF) in dry storage vaults following post-irradiation examination prior to shipment to INEL. They were stored in air at both facilities; there were no wet-storage facilities available at AI. No detailed storage history documentation is available.

Ke70 Criticality study for fuel element transfer from core to AIHL for post-irradiation analysis
Ke71 Criticality study for dry storage of S8DR fuel elements in RMDF vault no. 2

Mc71, Mc71a, Sc73:
Information on the batching, canning, and shipping of S8DR fuel from the AIHL to the RMDF, including a listing of all elements by core position and whether each was disassembled for examination

10.8 S8DR SHIPPING RECORDS

The shipping records for the S8DR fuel shipments to INEL were reviewed in detail. The original records are difficult to read, and were transcribed onto the S8DR spreadsheet that is reproduced at the end of this section. The spreadsheet lists the core location, fuel rod identification, unclad fuel rod mass, total uranium mass, and ^{235}U mass for each fuel element, plus the shipping document number, shipment date, Atomics International can number, and probable INEL aluminum can number. The latter is base on document copies received from INEL as part of the current review activity. A total of 211 irradiated and 2 archived (unirradiated) S8DR fuel elements were shipped to INEL.

The fuel element and fuel rod identifications were compared with QA traceability document Do68. Three transcription errors were found on the shipping documents and are noted in the spreadsheet (Notes 1,4,6).

The mass totals shown on the spreadsheet were calculated from the individual recorded fuel element data. They were compared with the transfer document mass summaries that were filed with the shipping documents. The transfer document mass summary was found to have two typographical errors (spreadsheet Notes 5,7) and two offsetting 2-g mass errors (spreadsheet Note 8). One remaining discrepancy is a 0.06-g difference in the AI Can No. 7 total fuel mass, where the sum of the individual fuel rod masses is 0.06 g lower than the transfer document summary value (spreadsheet Note 2). This difference is also reflected in the total S8DR fuel mass sum. This is likely to be due to an addition or transcription error in the transfer document sum, but we cannot rule out a 0.06-g recording error in an individual element mass.

10.9 S8DR REFERENCES

AN76 *Anonymous*, "Zirconium Hydride (SNAP) Reactor Space Power Programs. Summary of Major Activities and Accomplishments," Rockwell International Energy Systems Group (June 1976)

An83 R. V. Anderson, et al., "Space Reactor Electric Systems. Subsystem Technology Assessment," Rockwell International Energy Systems Group Report ESG-DOE-13398 (March 1983)

Ar77 D. V. Arnold, C. C. Woolsey, and W. H. Friske, "Specification for Marking," Atomics International Specification N001A0104001 (June 1977) (7 pages)

Bu70 D. S. Burgess, "Statistical Evaluation of the Fuel-Clad Gap and Crack Length from S8DR Fuel Elements," Atomics International Technical Information Document TI-568-24-058 (June 1970) (14 pages)

Co71 L. Cooper, "Results from Abbreviated EMP Examination of Three S8DR Irradiated Fuel Specimens," Atomics International Technical Information Document TI-653-240-003 (July 1971) (7 pages)

Da70 N. F. Davies, "S8DR Examination," Atomics International Technical Information Document TI-568-24-049 (April 1970) (3 pages)

De67 W. F. Dennison and P. J. Fessia, "Fuel Element Assembly, S8DR," Atomics International Specification ST0622NA0041 (May 1967) (14 pages)

Do68* C. Donnelly and W. Nagel, "S8DR Fuel Element Quality Assurance Record," Atomics International Technical Data Record AI-AEC-TDR-12702 (March 1968) (137 pages)

Ei70 R. L. Eichelberger, "Coating on S8DR Fuel Cladding," North American Rockwell Corporation Aerospace and Systems Group Internal Letter dated 3 August 1970 (6 pages)

Ha69 F. J. Halfen, "Summary of Selected S8DR Operating Data from 1-19-69 to 4-28-69," Atomics International Technical Information Document TI-696-13-033 (May 1969) (19 pages)

Ke70 N. Ketzlach, "Criticality Study - S8DR Core Disassembly," Atomics International Technical Information Document TI-568-24-041 (January 1970) (15 pages)

Ke71 N. Ketzlach, "Storage of S8DR Fuel at RMDF," Atomics International Technical Information Document TI-568-240-083 (May 1971) (4 pages)

La67 K. Langrod, "Process Flow Sheets - S8DR Hydrogen Barrier Material Fabrication," Atomics International Technical Data Record NAA-SR-MEMO-12469 (May 1967) (27 pages)

La68 E. M. Larson, "Radiological Safety and Hazardous Materials Handbook for Building 059," Atomics International Technical Data Record AI-AEC-12682 (April 1968) (part of S8DR Operations Manual)

La68a E. M. Larson, "S8DR Operations Manual," Atomics International Technical Data Record AI-AEC-12683 (May 1968) (part of S8DR Operations Manual)

La68b E. M. Larson, "SNAP Ground Prototype Test Facility (Building 059)," Atomics International Technical Data Record AI-AEC-12684 (April 1968) (part of S8DR Operations Manual)

Li70 A. F. Lillie, "The S8DR Screening Examination," Atomics International Technical Information Document TI-568-240-068 (October 1970) (112 pages)

Li70a A. F. Lillie and K. J. Miller, "S8DR Screening Examination Weekly Status Reports," Atomics International Technical Information Documents: #1 (TI-568-24-046, 20 March 1970, 5 pages), #2 (TI-568-24-048, 27 March 1970, 8 pages), #3 (TI-568-24-050, 3 April 1970, 8 pages), #4 (TI-568-24-051, 10 April 1970, 13 pages), #5 (TI-568-24-052, 17 April 1970, 27 pages), #6 (TI-568-24-053, 24 April 1970, 22 pages), #7 (PSR-568-24-001, 1 May 1970, 17 pages), #8 (PSR-568-24-002, 8 May 1970, 13 pages), #9 (PSR-568-24-003, 15 May 1970, 15 pages), #10 (PSR-568-24-004, 22 May 1970, 31 pages), #11 (PSR-568-24-005, 29 May 1970, 19 pages), #12 (PSR-568-24-006, 5 June 1970, 31 pages), #13 (PSR-568-24-007, 12 June 1970, 38 pages), #14 (PSR-568-24

-008, 19 June 1970, 29 pages), #15 (PSR-568-24-009, 26 June 1970, 26 pages), and #16 (PSR-568-24-010, 3 July 1970, 22 pages)

Li71* A. H. Lillie and V. L. Rooney, Jr., "The SNAP 8 Developmental Reactor (S8DR) Post-Test Examination," Atomics International Report AI-AEC-13003 (June 1971) (93 pages)

Ma64 C. A. Martin and W. H. Timmerman, "Process Flow Sheet for SNAP 8DS Melting and Extruding," Atomics International Technical Data Record NAA-SR-MEMO-9630 (March 1964) (17 pages)

Ma67 C. A. Martin, "Process Flow Sheets - S8DR Cup Plugs," Atomics International Technical Data Record NAA-SR-MEMO-12470 (May 1967) (18 pages)

Mc67 D. T. McClelland, "Process Flow Sheets - S8DR Fuel Element Fabrication," Atomics International Technical Data Record NAA-SR-MEMO-12473 (May 1967) (19 pages)

Mc67a D. T. McClelland, K. Langrod, and C. A. Martin, "Process Flow Sheets - S8DR Cladding Tube Fabrication," Atomics International Technical Data Record NAA-SR-MEMO-12475 (May 1967) (67 pages)

Mc69 D. T. McClelland, "S8DR Fuel Rod Fabrication," Atomics International Technical Data Record AI-AEC-TDR-12775 (January 1969) (43 pages)

Mc69a D. T. McClelland, "SNAP 8 Development Reactor Fuel Element Fabrication," Atomics International Report AI-AEC-12806 (June 1969) (55 pages)

Mc71 W. R. McCurnin, "S8DR Fuel Canning and Storage," North American Rockwell Internal Letter dated 22 June 1971 (5 pages)

Mc71a W. R. McCurnin, "FINAL S8DR Fuel Canning and Storage," North American Rockwell Internal Letter dated 30 June 1971 (5 pages)

Mi69 K. J. Miller, "S8DR Examination Planning Document for AIHL," Atomics International Test Plan TP-568-24-004 (November 1969) (51 pages)

Mi69a K. J. Miller, "S8DR Examination Planning Document for AIHL," Atomics International Technical Information Document TI-568-24-037 (December 1969) (42 pages)

Mi70 K. J. Miller, "Selection of S8DR Fuel Elements for the Detailed Postirradiation Examination," Atomics International Technical Information Document TI-568-24-061 (June 1970) (11 pages)

Mi70a K. J. Miller, et al., "S8DR Detailed Examination Progress Report," Atomics International Program Status Reports: Number 1 (PSR-568-240-011, September 1970, 36 pages), Number 2 (PSR-568-240-012, October 1970, 62 pages), Number 3 (PSR-568-240-013, November 1970, 54 pages), Number 4 (PSR-568-240-014, December 1970, 11 pages), Number 5 (PSR-568-240-015, January 1971, 15 pages), Number 6 (PSR-568-240-016, February 1971, 44 pages), Number 7 (PSR-568-240-017, March 1971, 26 pages), Number 8 (PSR-568-240-018, May 1971, 23 pages), Number 9 (PSR-653-240-001, July 1971, 92 pages), and Number 10 (PSR-653-240-002, July 1971, 29 pages)

Mi70b K. J. Miller, "S8DR Fuel Accountability Losses at AIHL," North American Rockwell Corporation Aerospace and Systems Group Internal Letter dated 21 December 1970 (2 pages)

Mi70c M. Mike, "Fuel Rod Dimensional Data/Plots From N-Ray Results - S8DR Screening Examination," Atomics International Technical Information Document TI-568-24-066 (July 1970) (213 pages)

Mo68 L. Morrison and E. L. Babbe, "S8ER Fuel Rod and Fuel Element Pre-Irradiation Measurements," Atomics International Technical Information Document TI-S8DR-28-012 (May 1968) (16 pages)

Ro70 H. Rood, "S8DR Fuel Burnup and Transmutation Data," Atomics International Technical Information Document TI-568-24-063 (July 1970) (5 pages)

Sc70 F. G. Schmidt, "S8DR Fuel Element Sample Numbering System," North American Rockwell Corporation Aerospace and Systems Group Internal Letter dated 13 July 1970 (2 pages)

Sc71 F. G. Schmidt, "S8DR Fuel Cutting Losses Assigned to Scrap at AIHL," North American Rockwell Corporation Aerospace and Systems Group Internal Letter dated 1 July 1971 (1 page)

Sc73 F. G. Schmidt, "Shipment of SNAP Fuel to RMDF," North American Rockwell Internal Letter dated 16 March 1973 (2 pages)

Sw67 L. D. Swenson, "S8DR Core Performance Evaluation," Atomics International Report NAA-SR-12482 (August 1967) (75 pages)

Sw69* L. D. Swenson, "SNAP 8 Development Reactor Nuclear Analysis," Atomics International Report AI-AEC-12864 (October 1969) (75 pages)

Sw70 V. A. Swanson, "Neutron Radiography as a Means for Determining H/Zr in SNAP-Type Fuel Elements," Atomics International Technical Information Document TI-568-24-056 (May 1970) (16 pages)

Tr69 J. Tribfelner, "S8DR Pre-Irradiation Data File and Corresponding Computer Printout," Atomics International Technical Information Document TI-568-24-035 (November 1969) (46 pages)

Va70 R. Van Houten, "First Status Report (to 5:00 p.m., April 3, 1970) on the Coatings on S8DR Fuel Elements," North American Rockwell Corporation Aerospace and Systems Group Internal Letter dated 7 April 1970 (3 pages)

S8DR SHIPPING RECORD SUMMARY

Shipping Document	Shipment Date	Al Can No.	Fuel Identification Element	Rod	Location	Fuel Masses (grams) Unclad	total U	U-235	Notes	Prob. INEL Can No.
LAE-JSB-01	6/11/73	1	E720	706B-2	III-2	364.98	36.94	33.98		Al-3/Al-4
			E116	698D-3	III-3	364.38	36.69	33.66		
			E579	705B-4	IV-2	364.18	36.64	33.81		
			E650	701A-2	IV-3	363.90	36.68	33.88		
			E730	700A-3	IV-10	364.40	36.73	33.90		
			E101	698D-2	IV-15	364.72	36.76	33.75		
			E655	693B-3	V-2	364.97	36.46	33.64		
			E710	702A-1	V-3	364.52	36.74	34.00		
			E146	698A-4	V-4	365.05	37.09	33.77		
			E471	700C-2	V-8	364.38	37.20	34.24		
			E168	705A-4	V-14	364.63	36.75	33.79		
			E108	703B-1	V-20	364.56	37.22	34.29		
			E402	695C-3	V-22	364.70	36.87	33.84		
			E408	696A-3	V-23	364.67	36.10	33.22		
			E419	695D-2	V-24	364.70	37.05	34.16		
			E403	699C-1	VI-5	364.63	36.79	33.82		
			E557	703A-1	VI-8	364.15	37.14	34.31		
			E409	696A-2	VI-9	364.56	36.16	33.31		
			E252	699D-1	VI-16	365.72	36.90	33.78		
			E702	701C-1	VI-19	364.67	37.34	34.48		
			E149	704D-3	VI-20	364.53	36.93	33.96		
			E705	708A-2	VI-22	364.71	36.40	33.64		
			E708	701C-2	VI-24	364.49	37.32	34.53		
			E603	713C-4	VI-25	364.94	37.30	34.54		
Shipping Can 1 totals:			24	elements		8751.14	884.20	814.30		
Transfer document summary:			24	elements		8751.14	884.20	814.30		
LAE-JSB-01	6/11/73	2	E675	703C-4	VI-30	364.43	37.10	34.31		Al-8/Al-11
			E293	695D-4	VII-1	364.67	37.20	34.29		
			E509	693D-4	VII-3	364.19	36.78	33.62		
			E428	712D-1	VII-7	364.71	36.91	33.92		
			E652	704B-4	VII-10	365.12	36.84	33.89		
			E435	694D-3	VII-21	364.19	36.35	33.51		
			E164	708B-3	VII-22	364.77	36.91	34.32		
			E175	698C-2	VII-27	364.87	37.18	34.30		
			E584	700A-1	VII-29	364.40	36.95	34.33		
			E326	693B-2	VII-32	364.98	36.61	33.59		
			E319	707A-3	VII-36	365.00	36.54	33.66		
			E525	707C-1	VIII-6	364.73	36.98	33.91		
			E484	700C-3	VIII-9	364.62	37.45	34.63		
			E570	708A-1	VIII-25	364.38	36.51	33.37		
			E376	704C-3	VIII-36	365.43	37.05	34.01		
			E234	701D-3	VIII-42	364.46	37.17	34.36		
			E324	706D-1	IX-8	364.93	37.70	34.76		
Shipping Can 2 totals:			17	elements		6199.88	628.23	578.78		
Transfer document summary:			17	elements		6199.88	628.23	578.78		
LAE-JSB-01	6/11/73	3	E137	695B-2	II-1	364.51	36.31	33.42		Al-9
			E414	701B-3	IV-14	364.44	37.03	34.20		
			E394	695A-1	V-12	364.99	36.83	33.88		
			E457	708A-3	VI-12	365.40	36.47	33.78		
			E697	697A-1	VI-13	364.19	36.60	33.70		
			E700	694B-3	VI-17	364.99	36.54	33.74		

Shipping Document	Shipment Date	Al Can No.	Fuel Identification Element	Rod	Location	Fuel Masses (grams) Unclad	total U	U-235	Notes	Prob. INEL Can No.
LAE-JSB-01 (con.)	6/11/73	3 (con.)	E703	700B-3	VI-18	364.47	37.18	34.25		
			E493	702C-4	VII-11	364.40	36.91	34.00		
			E138	694B-2	VII-12	364.51	36.56	33.79		
			E572	685D-3	VII-15	364.55	37.18	34.34		
			E366	713A-2	VII-19	364.83	36.77	33.78		
			E467	705D-1	VII-20	365.08	37.42	34.50		
			E494	702C-2	VII-24	364.59	36.93	34.16		
			E265	701A-4	VIII-1	365.73	37.16	34.23		
			E244	699B-4	VIII-8	365.61	36.85	33.92		
			E224	701D-2	VIII-14	364.62	37.19	34.34		
			E197	699A-3	VIII-16	365.08	37.06	34.17		
			E517	712B-3	VIII-17	364.98	37.23	34.34		
			E316	713C-2	II-5	365.45	37.50	34.71	*see note 1*	
			E213	705A-2	VIII-20	364.93	37.04	34.26		
			E333	706A-2	VIII-21	365.20	36.92	34.03		
Shipping Can 3 totals:			21	elements		7662.55	775.68	715.54		
Transfer document summary:			21	elements		7662.55	775.68	715.54		
LAE-JSB-01	6/11/73	4	E314	697C-1	IX-31	364.52	36.45	33.55		Al-1/Al-3
			E251	696C-1	IX-32	365.66	37.11	34.35		
			E205	698A-1	IX-34	365.46	37.42	34.45		
			E294	706C-4	IX-35	364.93	37.30	34.40		
			E563	702B-3	IX-37	364.56	37.11	34.21		
			E355	700D-1	IX-42	364.52	37.29	34.39		
			E196	698A-3	IX-44	365.20	37.40	34.55		
			E532	705C-2	IX-45	365.21	37.03	34.20		
			E315	708B-2	IX-46	364.96	37.08	34.25		
			E296	697B-4	IX-47	365.00	36.24	33.55		
			E372	704C-1	VIII-30	365.39	37.05	34.26		
			E561	703D-1	VIII-33	365.45	37.93	34.93		
			E345	701B-1	VIII-34	365.00	37.38	34.49		
			E539	700A-4	VIII-35	364.71	37.05	34.34		
			E237	704D-4	VIII-37	365.18	37.14	34.28		
			E420	696D-2	V-16	365.00	36.76	33.83		
			E560	711B-4	IX-3	364.55	36.97	34.11		
			E374	704C-2	VIII-22	365.53	37.06	34.24		
			E228	701B-4	VIII-27	364.81	37.36	34.47		
			E489	702C-1	VIII-31	364.47	36.99	34.10		
			E363	700D-3	VIII-2	364.62	37.23	34.31		
			E219	705B-1	VIII-11	364.71	36.98	34.13		
			E511	693C-4	VIII-38	364.44	37.06	34.21		
			E305	705A-1	VIII-40	365.07	37.05	34.16		
Shipping Can 4 totals:			24	elements		8758.95	890.44	821.76		
Transfer document summary:			24	elements		8758.95	890.44	821.76		
LAE-JSB-01	6/11/73	5	E706	713B-2	III-4	365.22	36.96	33.96		Al-7/Al-8
			E165	698C-3	IV-7	364.17	36.98	33.99		
			E120	698C-4	IV-16	364.37	36.91	33.97		
			E413	708D-3	IV-17	364.75	36.99	34.23		
			E449	709A-2	IV-18	364.89	36.27	33.40		
			E424	709C-2	V-10	364.55	36.60	33.71		
			E401	709D-2	V-11	367.13	36.90	33.92		
			E425	709B-3	VI-3	364.77	36.44	33.58		

Shipping Document	Shipment Date	Al Can No.	Fuel Identification Element	Rod	Location	Fuel Masses (grams) Unclad	total U	U-235	Notes	Prob. INEL Can No.
LAE-JSB-01 (con.)	6/11/73	5 (con.)	E254	694C-3	VI-4	364.52	36.49	33.59		
			E678	709A-1	VI-28	364.58	36.35	33.45		
			E190	708C-3	VII-9	364.61	36.83	33.95		
			E512	710D-3	VII-23	364.93	37.33	34.45		
			E427	709B-1	VII-26	364.90	36.53	33.58		
			E513	710D-2	VIII-13	364.66	37.38	34.60		
			E152	710A-1	IX-39	364.63	36.50	33.60		
			E490	702D-4	IX-48	364.77	37.39	34.56		
			E387	701D-1	III-10	364.37	36.84	33.67		
			E447	708B-1	III-11	365.32	36.71	33.57		
			E530	704B-1	III-12	366.08	36.68	33.79		
			E649	704D-1	IV-11	365.11	36.84	33.79		
			E574	702A-4	VI-11	364.51	36.82	33.64		
			E264	699C-4	VII-13	365.91	36.99	34.10		
			E232	704D-2	VII-14	365.07	37.05	34.13		
			E126	707B-1	VII-35	364.31	36.58	33.35		
		Shipping Can 5 totals:	24	elements		8758.13	883.36	812.58		
		Transfer document summary:	24	elements		8758.13	883.36	812.58		
LAE-JSB-01	6/11/73	6	E505	698D-4	IX-27	364.39	37.09	34.23		Al-4/Al-5
			E404	695B-1	IX-28	364.31	36.76	33.97		
			E577	714A-3	IX-29	364.64	37.41	34.47		
			E434	705B-2	IX-30	364.52	37.04	34.14		
			E327	693B-4	IX-38	364.44	36.74	34.16		
			E411	713D-3	V-13	365.41	37.67	34.67		
			E430	714C-3	VI-10	364.38	36.98	34.22		
			E354	697D-3	VIII-4	364.48	37.07	34.18		
			E483	705D-4	VIII-10	364.58	37.44	34.48		
			E464	712C-1	VIII-26	364.80	37.10	34.25		
			E379	713A-3	III-8	364.43	36.48	33.38		
			E538	700A-2	VII-5	364.56	36.97	34.08		
			E338	706C-1	IX-2	365.37	37.34	34.33		
			E018	704B-2	IX-6	365.04	37.02	34.26		
			E422	712D-3	IX-7	365.30	37.11	34.24		
			E439	712C-2	IX-11	365.18	37.21	34.63		
			E010	697D-2	IX-12	364.38	37.13	34.27		
			E322	693A-4	IX-14	364.96	36.42	33.70		
			E256	697B-3	IX-15	365.02	36.25	33.46		
			E378	713A-4	IX-16	364.73	36.91	34.05		
			E371	706C-2	IX-18	365.61	37.40	34.46		
			E157	695D-1	IX-19	365.16	37.39	34.45		
			E250	695C-1	IX-23	365.38	37.23	34.47		
			E320	695A-2	IX-26	364.27	37.05	34.19		
		Shipping Can 6 totals:	24	elements		8755.34	889.21	820.74		
		Transfer document summary:	24	elements		8755.34	889.21	820.74		
		LAE-JSB-01 shipment totals:	134	elements (S8DR)		48885.99	4951.12	4563.70		
		Transfer document:	134	elements		48885.99	4951.12	4563.70		

Shipping Document	Shipment Date	Al Can No.	Fuel Identification Element	Rod	Location	Fuel Masses (grams) Unclad	total U	U-235	Notes	Prob. INEL Can No.
LAE-JSB-02	6/20/73	7	E133	712D-4	V1-26	297.50	30.05	27.69	14 pcs	Al-6/Al-7
			E128	693D-1	VI-29	364.48	36.74	33.84	3 pcs	
			E110	696B-1	IV-4	363.37	36.45	33.41	5 pcs	
			E631	703C-2	V-7	362.63	36.81	33.90	6 pcs	
			E302	699A-2	V-17	362.34	36.56	33.62	1 pc	
			E391	695A-4	V-18	363.02	36.63	33.40	4 pcs	
			E383	713B-3	V-21	363.33	36.88	34.04	3 pcs	
			E463	703D-3	VI-14	363.27	37.53	34.95	3 pcs	
			E267	699C-3	VII-6	364.36	36.84	33.97	3 pcs	
			E711	712B-4	VII-17	362.86	36.94	34.04	2 pcs	
			E199	699A-1	VII-18	363.55	36.83	34.00	1 pc	
			E230	701C-4	VIII-15	363.16	37.33	34.24	1 pc	
			E526	701A-1	VIII-19	363.70	36.95	34.03	1 pc	
			E365	700D-4	VIII-23	363.17	37.08	34.10	3 pcs	
			E308	707B-3	VIII-29	362.85	36.50	33.84	1 pc	
			E571	706A-1	VIII-32	362.76	36.68	34.02	5 pcs	
			E233	701D-4	VIII-28	364.56	37.19	34.30	2 pcs	
			E321	706C-3	IX-24	365.06	37.31	34.37		
			E274	699B-1	VIII-41	364.34	36.73	33.92	2 pcs	
			E454	712A-1	IV-12	360.05	36.55	33.79	17 pcs	
	Shipping Can 7 totals:		20	elements		7200.36	730.58	673.47	*see note 2*	
	Transfer document summary:		20	elements.		7200.42	730.58	673.47	19 sectioned	
LAE-JSB-02	6/20/73	8	E386	701B-2	VI-1	223.91	22.84	21.12	19 pcs	Al-2
			E131	693D-3	IV-13	325.84	32.71	30.40	14 pcs	
			E657	697C-4	VI-23	267.77	26.59	24.46	16 pcs	
			E396	712B-2	III-1	356.90	36.08	33.15	5 pcs	
			E412	693D-2	IV-5	359.46	36.09	33.47	5 pcs	
			E673	705C-3	V-6	359.86	36.09	33.14	8 pcs	
			E472	706A-4	V-19	362.44	36.57	33.54	10 pcs	
			E531	703D-4	VI-6	363.91	37.59	34.63	9 pcs	
			E699	694A-4	VI-15	361.70	36.03	33.26	2 pcs	
			E140	706D-3	VII-2	301.27	31.00	28.65	19 pcs	
			E061	698B-4	VIII-5	326.67	33.16	30.48	*see note 3*	
			E738	702A-3	VIII-39	361.73	36.68	33.92	2 pcs	
			E151	696D-1	IX-10	362.20	36.80	33.92	2 pcs	
			E334	695C-2	IX-43	362.45	36.93	34.26	2 pcs	
			E545	704A-2	IX-20	304.21	30.94	28.49	18 pcs	
			E328	696B-3	IX-40	301.95	30.59	28.22	18 pcs	
			E723	690B-4	VIII-3	253.27	25.61	23.65	14 pcs	
			E271	699A-4	VII-8	364.22	36.90	33.96	11 pcs	
			E111	698D-1	VI-2	327.46	33.14	30.57	14 pcs	
			E127	695B-3	I-1	362.98	36.15	33.30	2 pcs	
	Shipping Can 8 totals:		20	elements		6610.20	668.49	616.59		
	Transfer document summary:		20	elements		6610.20	668.49	616.59	20 sectioned	
LAE-JSB-02	6/20/73	9	E132	695C-4	VII-25	363.31	36.88	33.87	2 pieces	Al-10/Al-12
			E426	709B-2	VII-33	362.93	36.33	33.50	1 pc	
			E185	714B-1	VII-34	325.78	33.07	30.46	7 pcs	
			E335	693B-1	archive	236.39	24.82	23.12	8 pcs	
			E564	703A-3	archive	265.05	28.25	26.32	10 pcs	
			E144	710D-1	V-5	254.79	25.96	23.88	20 pcs + 6 wafers	
			E416	708D-4	IV-8	291.17	29.52	27.29	23 pcs + 3 wafers	

Shipping Document	Shipment Date	Al Can No.	Fuel Identification Element	Rod	Location	Fuel Masses (grams) Unclad	total U	U-235	Notes	Prob. INEL Can No.
LAE-JSB-02 (con.)	6/20/73	9 (con.)	E061	698B-4	VIII-5	12.03	1.22	1.12	*see note 3*	
			E529	693C-3	IX-36	364.13	37.10	34.35	intact	
			E573	693C-2	IX-13	364.42	37.13	34.42	intact	
			E562	713B-1	IX-5	364.79	37.32	33.95	*see note 4*	
			E486	704B-3	IX-4	365.01	37.01	34.18	intact	
			E536	697A-4	III-9	178.02	17.80	16.37	21 pcs + 8 wafers	
			E268	699D-3	VII-31	170.80	17.27	15.94	18 pcs + 15 wafers	
			E452	709A-3	V-15	363.14	36.13	33.36	5 pcs	
			E389	711B-1	II-6	266.98	26.75	24.61	13 pcs + 3 wafers	
			E398	712B-1	III-7	362.97	36.70	33.64	1 pc	
			E141	707A-2	VIII-24	337.09	33.81	31.41	7 pcs	
			E337	702D-3	IX-22	338.25	34.67	31.98	6 pcs	
			E226	697C-3	IX-21	337.79	33.78	31.09	6 pcs	
			E195	700B-4	VII-16	248.23	25.37	23.31	11 pcs	
			E397	710A-3	III-6	363.54	35.95	33.08	1 pc	
			E103	708C-4	III-5	362.52	36.36	33.67	1 pc	
			E156	704C-4	VIII-12	363.09	36.82	33.98	3 pcs	
Shipping Can 9 subtotals:			21	elements		6748.75	681.73	628.34		
			1	segment (E061)		12.03	1.22	1.12	*see note 3*	
			2	archives		501.44	53.07	49.44		
Transfer document subtotals:			18	elements (cut)		5302.43	534.39	492.56		
			4	elements (intact)		1458.35	148.56	136.96	*see note 5*	
			2	archives		501.44	53.07	49.44		
Shipping Can 9 totals:			24	items		7262.22	736.02	678.90		
Transfer document summary:			24	items		7262.22	736.02	678.90		
						see note 7		*see note 5*		
LAE-JSB-02	6/20/73	10	117-1	*		63.00	6.29	5.86	5 pcs	Al-10/Al-12
			E236	702D-1	VIII-7	364.78	37.32	34.39	intact	
			E632	706D-2	VI-27	365.18	37.50	34.56	intact	
			E134	694D-2	VI-21	365.37	36.39	33.52	intact	
			E466	707D-4	V-9	364.47	36.59	33.81	intact	
			E433	709C-3	VI-7	364.64	36.68	33.89	intact	
			E104	703B-2	IV-9	364.53	37.15	34.13	intact	
			E704	699D-2	IV-6	364.46	36.63	33.47	intact	
			E568	713C-2	VIII-18	365.05	37.09	34.05	*see note 6*	
			E395	695A-3	II-4	364.51	36.63	33.56	intact	
			E453	708A-4	II-3	365.49	36.26	33.58	intact	
			E707	696C-3	II-2	364.84	36.56	33.52	intact	
			E492	713A-1	VII-4	351.67	35.45	32.71	2 pcs, clad	
			E653	702D-2	V-1	307.49	31.27	28.71	15 pcs	
			E142	698B-2	VII-28	245.53	24.87	23.08	12 pcs	
			E368	712D-2	VII-30	304.12	30.78	28.12	17 pcs	
			E701	694A-1	IV-1	338.46	33.61	30.71	24 pcs	
			NAA-121	Exp#1 694B-1 E 4570		178.88	18.75	17.46	10 pcs	
			NAA-121	Exp#2 694B-4 E 5204		178.88	18.60	17.33	10 pcs	
			NAA-121	Exp#3 694C-4		167.88	17.46	16.26	10 pcs	
			NAA-121	Exp#4 644A-3		191.20	19.88	18.52	10 pcs	

* pieces of 623D-7,626D-7,624B-5,624B-1

Shipping Document	Shipment Date	Al Can No.	Fuel Identification			Fuel Masses (grams)			Notes	Prob. INEL Can No.
			Element	Rod	Location	Unclad	total U	U-235		
Shipping Can 10 subtotals:			16	elements		5560.59	560.78	515.81	*see note 8*	
			1	NAA-117-1 pieces		63.00	6.29	5.86		
			4	NAA-121 remnants		716.84	74.69	69.57	*see note 8*	
Plus S8ER elements:			5	S8ER elements		1149.00	113.14	105.39	*see note 9*	
Transfer document subtotals:			16	S8DR elements		5560.59	560.78	517.81	*see note 8*	
			1	NAA-117-1 pieces		63.00	6.29	5.86		
			4	NAA-121 remnants		716.84	74.69	67.57	*see note 8*	
			5	S8ER elements		1149.00	113.14	105.39	*see note 9*	
Shipping Can 10 totals:			26	items		7489.43	754.90	696.63		
Transfer document summary:			26	items		7489.43	754.90	696.63		
LAE-JSB-02 Shipment Totals:										
S8DR-irradiated:			77	elements, segments		26119.9	2641.58	2435.33		
S8DR archives:			2	archives		501.44	53.07	49.44		
Irradiated in-pile specimens:			6	in-pile segments		791.87	82.2	75.43		
S8ER elements:			5	elements		1149.00	113.14	105.39		
LAE-JSB-02 shipment totals:			90	items		28562.21	2889.99	2665.59		
Transfer document:			90	items		28562.27	2889.99	2665.59		
						see note 2				
S8DR reactor totals:			211	irradiated elements		75005.89	7592.7	6999.03		
			2	archived elements		501.44	53.07	49.44		

References:
- Data transcribed from archived copies of original shipping records
- Mass sums compared with S8DR transfer document summary sheets for the two shipments
- Fuel element and fuel rod identifications compared with QA traceability document Do68 (AI-AEC-TDR-12702, 3/21/68)
- Disassembly status of elements in Cans 7-10 from North American Rockwell Internal Letters Mc71a, Sc73

Notes:
1. Element E316 in Al Can No. 3 was in core location II-5; location VIII-18 listed in shipping document is incorrect.
2. Unclad fuel total mass for Al Can No. 7 is 0.06 g lower for the spreadsheet sum than the transfer document sum. This could be an addition error in the transfer document, a transcription error in a transfer document sum (recording 5385.36 g for class 38 instead of 5385.30 g), or an element mass error. This difference is also the 0.06-g difference in the spreadsheet vs. transfer document total masses for shipment LAE-JSB-02.
3. One small segment of E061 was shipped in Al Can No. 9; most (in 2 pcs) was in Al Can No. 8.
4. Shipping records for Al Can No. 9 incorrectly list E562 core location as IV-5; IX-5 is correct. Element shipped intact.
5. Transfer document summary for Al Can No. 9 Ux total for 4 intact elements has a typo: should be 136.90 g instead of 136.96 g. Transfer document sum (678.90 g) is correct.
6. Element E568 in Al Can No. 10 was in core location VIII-18; location II-5 listed in shipping document and Hot Cell records is incorrect. Element shipped intact.
7. Transfer document summary for Al Can No. 10 lists Net $UZrH_2$ as 1262.22 g instead of 7262.22 g, a typo.
8. Transfer document summary has two offsetting 2-g errors: total for 16 S8DR elements should be 517.81 g, and total for NAA-121 remnants should be 69.57 g.
9. S8ER fuel elements are documented in S8ER Shipping Record Summary spreadsheet.

c-s8dr.xls 9/15/94

11. SHIELD TEST FACILITY (STF)

Initial criticality	October 9, 1962 (To62)
Date of shutdown:	July 1964 (then rebuilt as STIR with MTR-type fuel elements)
Thermal power:	50 kWt
Operating location:	SSFL Building 028

11.1 STF GENERAL DESCRIPTION

The Shield Test Facility (STF) was a pool-type reactor used primarily to generate radiation fields for shielding tests of SNAP reactor components. It initially had a 50-kWt core fueled with SNAP reactor fuel elements, and was operated at this power level between 1961 and 1964. In 1964 the reactor was modified to raise the power rating to 1000 kWt, by installing a new core type fueled with Materials Test Reactor (MTR) type fuel elements. This rebuilt configuration was renamed the Shield Test and Irradiation Reactor (STIR), which operated between 1964 and 1972. The fuel elements of interest for the present investigation are those from the original STF 50-kWt SNAP-type core, which had hydrided zirconium-uranium fuel rods. The uranium was 7 wt.%, with a nominal ^{235}U content of 93^+ wt.%.

There were two core loadings using SNAP fuel (To62). The initial loading consisted of 60 fuel rods. It was not effective for driving the facility's fission plate, the purpose for which the reactor was constructed, and thus the reactor core loading was reconfigured. The second loading consisted of 56 fuel rods and several BeO reflector elements. This is the INEL-stored SNAP reactor fuel for which the least information was obtained. However, since the facility was fueled with reject SER fuel rods, a description of the fuel rods may be taken from the SER documentation.

An overview description of the facility is given in O191, and a pre-operation description in Jo61.

11.2 STF REACTOR AND FUEL IDENTIFICATION

Reactor:	Shield Test Facility (STF)
Also identified as:	Shield Test Reactor (STR)
	SNAP Shield Test Experiment (STE)

Core fuel Identification: STF (same as reactor name)

Reactor designer, builder, operator: Atomics International
Reactor owner: Atomic Energy Commission (now DOE)

Fuel designer, builder: Atomics International
Fuel owner: Atomic Energy Commission (now DOE)

11.3 STF DESIGN REACTOR PHYSICS AND CORE DESCRIPTION

To62 There were two SNAP-type core loadings. The first core used 61 reject SER fuel rods, clad in Type 6061 aluminum alloy. No hydrogen barrier was used. The second core loading used 56 of the reject SER fuel rods, and included several BeO reflector elements.

Jo61 Pre-operation reactor physics parameters (Table III), for 45-element core
To61 Pre-operation reactor physics parameters (Table VI), for 45-element core

Fuel elements/assemblies:
To62 61 in first core loading, 56 in second

Design neutron flux:
Design neutron energy:
Core map and element locator:
To62 Initial core map (Fig. 1, 60 fuel rods) and second core map (Fig. 2, 56 fuel rods)
(Note: we have not correlated the two core maps to verify that the same fuel elements were used in both core loadings.)

11.4 STF FUEL FABRICATION INFORMATION

Fuel specifications and drawings:
Jo61 Fuel element drawing and some composition information (B.1.a)
To61 Brief description of fuel element chemistry and design

Fuel fabrication and quality control:
Fuel cladding, inner coatings, and other non-fuel materials:
Cladding was Type 6061 aluminum alloy, with no inner coating

Potential fuel chemical reactions:
Potential RCRA-regulated materials:
Quantities of materials:
See shipping spreadsheets (fuel rods, U, ^{235}U)

Fuel assembly identification methods:

11.5 STF OPERATIONAL HISTORY

Core Operating History:
Core neutron flux profile:
Operation abnormalities:
Core shut-down date:
Core discharge date:

11.6 STF POST-IRRADIATION PROPERTIES AND EXAMINATION

(There was no post-irradiation examination of these fuel elements.)

Visual observations:
Dimensions:
Metallurgical states of materials:
Cladding defects:
Corrosion or chemical contamination:

Hydrogen effects:
Burnup:
Fission products:
Transuranics:
Activation products:
Decay heat generation:
Radiation level decay curves:
Alpha contamination:

(not relevant for the declad fuels)

11.7 STF ON-SITE STORAGE HISTORY

SNAP fuel elements were stored in air at the RMDF in dry storage vaults in preparation for shipment to INEL. There were no wet-storage facilities available at Atomics International. No detailed storage history documentation is available.

11.8 STF SHIPPING RECORDS

The STF fuel shipping records for shipments to INEL were reviewed in detail. The original records are difficult to read, and were transcribed onto the STF spreadsheet that is reproduced at the end of this section. The spreadsheet lists the fuel rod (unclad), total uranium, and ^{235}U masses for each fuel element, plus the shipping document number, shipment date, Atomics International can number, and probable INEL aluminum can number. The latter is base on document copies received from INEL as part of the current review activity. 61 STF fuel elements were shipped to INEL.

The mass totals shown on the spreadsheet were calculated from the individual recorded fuel element data. They were compared with a Shield Test Fuel Data mass sheet dated 7/27/64 and filed with the shipping records. This comparison shows a slight discrepancy for AI Can No. 3, where the data sheet lists the total uranium mass as 275 g, instead of 269 g as obtained from the shipping records (with a corresponding total uranium for STF of 2883 g instead of 2877 g). The fuel data sheet is probably in error, but we cannot verify.

11.9 STF REFERENCES

Jo61* R. P. Johnson, "SNAP Shield Test Experiment Operations Manual," Atomics International Report NAA-SR-5897 (June 1961) (196 pages)

O191 R. D. Oldenkamp and J. C. Mills, "Nuclear Operations at Rockwell's Santa Susana Field Laboratory - A Factual Perspective," Rockwell International, Rocketdyne Division Supporting Document N001ER000017 (September 1991) (95 pages)

To61* R. L. Tomlinson, "SNAP Shield Test Experiment Final Hazards Summary," Atomics International Report NAA-SR-5896 (March 1961) (173 pages)

To62* R. L. Tomlinson, R. P. Johnson, and S. G. Wogulis, "SNAP Shield Test Experiment Reactor Physics Tests," Atomics International Report NAA-SR-7368 (July 1962)

STF SHIPPING RECORD SUMMARY

Shipping Document	Shipment Date	Al Can No.	Fuel Identification Element	Rod	Location	Fuel Masses (grams) Unclad	total U	U-235	Notes	Prob. INEL Can No.
CAE-JZA-21	5/20/66	3	A-51			648	45	42		7
			A-31			644	45	42		
			A-19			641	45	42		
			A-52			644	44	41		
			A-45			649	45	42		
			A-37			647	45	42		
	Shipment totals:		6	elements		3873	269	251		
CAE-JZA-22	5/27/66	1	A-1T			644	44	41		7
			A-16			672	47	44		
			A-2T			644	44	41		
			A-7			647	45	42		
			A-13			651	46	43		
			A-18			615	43	40		
	Shipment totals:		6	elements		3873	269	251		
CAE-JZA-23	6/4/66	5	A-50			651	45	42		8
			A-24			644	45	42		
			A-23			644	45	42		
			A-30			646	45	42		
			A-8			615	43	40		
			A-17			648	45	42		
	Shipment totals:		6	elements		3848	268	250		
CAE-JZA-24C	6/7/66	4	C-1			699	69	64		8
			B-3			629	43	40		
			A-38			649	45	42		
			A-9			644	45	42		
			A-21			647	45	42		
			A-10			648	45	42		
	Shipment totals:		6	elements		3916	292	272		
CAE-JZA-25	6/10/66	6	A-3T			639	44	41		8
			B-5			618	42	39		
			A-36			678	47	44		
			A-5			643	45	42		
			A-41			640	45	42		
			C-5			704	70	65		
	Shipment totals:		6	elements		3922	293	273		
CAE-JZA-26	6/14/66	2	A-20			650	46	43		7
			A-33			645	45	42		
			A-34			651	46	43		
			A-35			652	46	43		
			A-39			649	45	42		
			A-44			648	45	42		
	Shipment totals:		6	elements		3895	273	255		

Shipping Document	Shipment Date	Al Can No.	Fuel Identification Element	Rod	Location	Fuel Masses (grams) Unclad	total U	U-235	Notes	Prob. INEL Can No.
CAE-JZA-27	6/16/66	8	A-15			648	45	42		9
			B-4			635	43	40		
			A-48			669	47	44		
			A-26			640	45	42		
			A-11			648	45	42		
			C-4			762	75	70		
	Shipment totals:		6	elements		4002	300	280		
CAE-JZA-28	6/20/66	7	C-2			702	70	65		9
			A-32			648	45	42		
			A-27			652	46	43		
			A-47			644	45	42		
			A-40			641	45	42		
			A-12			649	45	42		
	Shipment totals:		6	elements		3936	296	276		
CAE-JZA-29	6/22/66	9	A-46			645	45	42		10
			A-25			646	45	42		
			A-4			669	47	44		
			B-2			643	44	41		
			A-42			644	45	42		
			C-3			708	70	65		
	Shipment totals:		6	elements		3955	296	276		
CAE-JZA-30	6/24/66	10	A-22			645	45	42		10
			A-28			643	45	42		
			A-43			647	45	42		
			A-14			644	45	42		
			A-29			645	45	42		
			A-6			679	48	45		
	Shipment totals:		6	elements		3903	273	255		
CAE-JZA-31	6/28/66	11	A-49			685	48	45		10
	Shipment totals:		1	elements		685	48	45		
STF reactor totals:			61	elements		39808	2877	2684		

Reference: Data transcribed from archived copies of original shipping records
Comparison made with Shield Test Fuel Data sheet dated 7/27/64

Notes: 61 fuel elements received by INEL per BUC-41-66A
Five fuel elements (C -1,2,3,4,5) have higher uranium concentrations (10 wt. % instead of 7 wt. %)

For Al Can No. 3, original shield test fuel data sheet lists total U mass as 275 g (instead of 269 g) and total U for STF as 2883 g (instead of 2877 g). The individual rods total 269 g (shown on spreadsheet). The fuel data sheet is probably in error, but cannot verify.

c-stf.xls 8/30/94

IN-PILE FUEL EXPERIMENTS

12. NAA-82-1 IN-PILE FUEL EXPERIMENT

Placed in reactor:	April 10, 1964 (Kr66)
Removed from reactor:	August 16, 1965 (Bu67)
Thermal power:	Not Available
Irradiation location:	NRTS (INEL) MTR

12.1 NAA-82 GENERAL DESCRIPTION

The objective of the NAA-82-1 experiment was to determine the irradiation behavior of a prototype zirconium-uranium hydride SNAP 2 fuel element at peak temperature and burnup conditions for 125 kWt operation of a SNAP 2 core (Kr66). The single fuel element was irradiated in the Materials Test Reactor (MTR) for 8760 hours at an estimated peak fuel temperature of 1490 °F to a burnup of 0.09 metal-atom %.

12.2 NAA-82 FUEL IDENTIFICATION

Reactor type: SNAP 2
Fuel identification: NAA-82

Fuel Specimen Identification: Single fuel element E2004 (Kr66)

Experiment designer, builder: Atomics International
Reactor owner: Atomic Energy Commission (now DOE)

Fuel designer, builder: Atomics International
Fuel owner: Atomic Energy Commission (now DOE)

12.3 NAA-82 REACTOR EXPERIMENT DESCRIPTION

Bu67 Irradiation capsule schematic (Fig. 2)

Experiment design parameters:
Design neutron flux:
Bu67 MTR calculated position-dependent relative thermal and fast fluxes (Fig. 8)

Design neutron energy:
Core element locator:
Bu67 MTR cross section mockup for nuclear analysis with capsule location noted (Fig. 7)

12.4 NAA-82 FUEL FABRICATION INFORMATION

Fuel specifications and drawings:
Kr66 Experiment assembly drawing (Fig. 1)
Bu67 Fuel rod and fuel element assembly schematic drawings (Fig. 1)

Fuel fabrication and quality control:

Sc65 Reference SNAP 2 fuel element

Kr66 Fuel element dimensions (page 5); fuel rod information (page 6), including dimensions, U, Zr, H, C, and total impurity content, density, N_H, weight, ^{235}U weight

Bu67 Fuel rod Zr-10 wt.%U alloy; fuel element and fuel rod data in Table 1

Fuel cladding, inner coatings, and other non-fuel materials:

Kr66 Hastelloy-N cladding and AI-8763D hydrogen barrier coating (page 5)

Bu67 Chemical composition of Hastelloy-N cladding (Table 2); NaK surrounded fuel element

Potential fuel chemical reactions:

Potential RCRA-regulated materials:

Quantities of materials:

See shipping spreadsheets (fuel rods, U, ^{235}U)

Fuel assembly identification methods:

Kr66 Identification markings on lower end of fuel rod shown in Fig. 6b

Bu67 Identification markings on lower end of fuel rod shown in Fig. 23a

12.5 NAA-82 IRRADIATION HISTORY

Experiment Operating History:

Sc65 Irradiated to 125 kWt SNAP 2 conditions

Kr66 Irradiated for 8,760 hours at peak fuel temperature of 1490 °F

Bu67 Irradiated for 23 reactor cycles in the MTR, starting April 9, 1964 (cycle 207) and discharged August 16, 1965 (end of cycle 229); design peak cladding maintained at 1300 ± 15 °F for 93.4% (8479 h) of full-power operation

Neutron flux profile:

Operation abnormalities:

12.6 NAA-82 POST-IRRADIATION PROPERTIES AND EXAMINATION

Sc65 Diagram of planned sectioning for future post-irradiation examination

Kr66 Fuel rod sectioning diagram for post-irradiation examination (Fig. 9)

Bu67 Fuel rod sampling plan (Fig. 25)

Visual observations:

Sc65 Element exterior appeared to be in excellent condition with no observable defects

Kr66 Photographs of fuel element (after NaK removal; Figs. 4A, 4B), fuel rod (Fig. 5, 6), inside surface of cladding (Fig. 7, 8)

Bu67 Photograph of fuel element (Fig. 20, no defects observed), fuel rod (Fig. 22, no damage)

Dimensions:

Kr66 Post-irradiation fuel element diameters (Table I), fuel element length measurements (page 15), fuel rod diameters (Table III), fuel rod length measurements (page 24), density measurements (Table VIII and Appendix III), fuel element dimensions (Appendix I), fuel rod dimensions (Appendix II)

Bi67 Percent volume change of 3 samples (Table 6); diametrical fuel growth (Fig. 3, 5-11); volumetric fuel growth (Fig. 13, 14)
Bu67 Fuel element diameters (Table 8) and length measurements; fuel rod diameters (Table 10), densities (Table 11), and dimensional changes (Table 13)

Metallurgical states of materials:
Kr66 Photomicrographs of fuel alloy (Fig. 15), and cladding plus ceramic coating (Fig. 16) and metallography analysis (Appendix IV)
Bu67 Photomicrographs of cladding (Fig. 24) and unirradiated plus irradiated fuel rod material (Fig. 29)

Cladding defects:
Kr66 No cladding defects found

Corrosion or chemical contamination:
Kr66 Photograph of ceramic adhering to fuel rod (Fig. 6b)
Bu67 Some transfer of ceramic coating from inside of end caps to ends of fuel rod

Hydrogen effects:
Kr66 Hydrogen analysis results (Table IX)
Bi67 Initial and final hydrogen content for 8 samples (Table 6); H/Zr ratios (Fig. 20)
Bu67 Fuel hydrogen analysis (Table 12) and distribution (Fig. 26), and variation of zirconium hydride density with hydrogen content (Fig. 27)

Burnup:
Sc65 0.08 metal-atom % burnup at peak fuel temperature in excess of 1500 °F
Kr66 Calculated burnup of 0.08 metal-atom %; mass spectrographic burnup results (based on analyses at Phillips Petroleum), giving an average of 0.09 metal-atom %, and compares with calculated value of 0.08 m-a% (Table VII)
Bi67 Burnup for 3 samples (Table 6); diametric fuel growth vs. burnup (Fig. 3, 6)
Bu67 Gamma scans & autoradiographs of burnup data (Fig. 30); fuel burnup analysis (Table 14)

Fission products:
Kr66 Fission gas release data (Table II); Gamma scan data (Tables V, VI and Fig. 10, 11)
Bu67 Fission gas release measurements

Transuranics:
Activation products:
Decay heat generation:
Radiation level decay curves:
Alpha contamination:
(not relevant for the declad fuels)

12.7 NAA-82 ON-SITE STORAGE HISTORY

The NAA 82-1 experiment was received at the Atomics International Hot Laboratory (AIHL) on September 28, 1965 (Kr66). It was shipped to AI full of NaK, and declad in the AIHL in an inert (nitrogen) atmosphere to prevent NaK fires. The fuel element was stored at the AIHL while post-irradiation examinations were proceeding, and then shipped to AI's Radioactive Materials

Disposal Facility (RMDF) in a shipping cask for temporary storage before shipment to INEL. It was always stored in air at both the RMDF and the AIHL.

12.8 NAA-82 SHIPPING RECORDS

The NAA-82-1 fuel rod material (from one irradiation capsule) shipped to INEL is listed in the S8ER Shipping Record Summary, where it is included under shipment LAE-JWA-26.

12.9 REFERENCES

Bi67* K. R. Birney, "An Empirical Study of SNAP Reactor Fuel Irradiation Behavior," Atomics International Report NAA-SR-12284 (August 1967)

Bu67* J. S. Buck, W. E. Krupp, and T. A. Park, "Irradiation Performance of a Full-Scale Prototype SNAP 2 Reactor Fuel Element," Atomics International Report NAA-SR-12030 (August 1967) (82 pages)

Kr66 W. E. Krupp, "Post-Irradiation Examination of the NAA 82-1 Experiment," Atomics International Technical Data Record NAA-SR-MEMO-11891 (March 1966) (51 pages)

Sc65 A. R. Schmitt, "Post-Irradiation Examination of NAA-82-1," Atomics International Internal Letter dated 27 October 1965 (4 pages)

13. NAA-115-2 IN-PILE FUEL EXPERIMENT

Irradiation start date:	Not available
Irradiation end date:	Not available
Thermal power:	Not available
Irradiation location:	Hanford K-East Reactor

13.1 NAA-115 GENERAL DESCRIPTION

NAA-115-2 was one of three similar irradiation experiments (NAA-115-1, NAA-115-2, and NAA-117-1) that were statistically designed to test and evaluate sub-length SNAP 8 fuel elements. NAA-115-1 and NAA-115-2 were nearly identical in design with the exception that NAA-115-2 contained a newer hydrogen barrier (SCB-1) which was to be used for S8DR fuel elements, and the closure cup plug component was changed to the S8DR design (Fo67b). Both NAA-115 experiments contained 12 sub-length fuel elements with target burnups ranging from 0.3 to 0.4 metal-atom %. The NAA-115-2 elements (for which fuel material is in storage at INEL) were formed into six separate experimental capsules which were connected in series to form a 12-foot-long test assembly.

Additional information is available for the separate experiment NAA-115-1, including a report compilation of NAA-115-1 photographs. A summary of several of the in-pile experiments is provided in Bi67.

13.2 NAA-115 FUEL IDENTIFICATION

Reactor type: S8DR
Fuel identification: NAA-115

Fuel Specimen Identification:

- Fo67b Sublength fuel elements (Fig. 2); identification of fuel elements in each capsule (Fig. 1); fuel elements identified as IS 21 through IS 32
- Fo68 Fuel element identification/fuel rod number identification (with compositions)

Experiment designer, builder: Atomics International
Reactor owner: Atomic Energy Commission (now DOE)

Fuel designer, builder: Atomics International
Fuel owner: Atomic Energy Commission (now DOE)

13.3 NAA-115 REACTOR EXPERIMENT DESCRIPTION

Experiment design parameters:

- Fo67b 12 test fuel elements, contained in 6 capsules (Fig. 1)
- Fo68 Reports intentionally do not disclose information that might disclose Hanford production information

Design neutron flux:
Design neutron energy:

Core element locator:

Fo67b 12 test fuel elements incorporated in experiment

13.4 NAA-115 FUEL FABRICATION INFORMATION

Fuel specifications and drawings:

Fo68 Engineering drawing of a four-fuel-element capsule (Fig. 8); engineering drawing of a single-fuel-element capsule (Fig. 9)

Fuel fabrication and quality control:

Fo67b Schematic cross section of a typical fuel element (Fig. 1); schematics of instrumented and uninstrumented fuel elements (Fig. 2); fuel rod composition (U, ZrC, H, N_H, density; Table I); fuel rod dimensions (O.D., length; Table II)

Fo68 S8DR sublength fuel element; schematic comparing fuel element with full length S8DR fuel element design (Fig. 3); 15 were fabricated and 12 irradiated, where the 3 unirradiated elements were used in the hot cell to develop disassembly procedures. *(This suggests that some unirradiated pieces might also have been shipped.)* Design materials and dimensions (Table 1); chemical composition of hydrided Zr-U alloy material (Table 6); photographs of pre-irradiation capsule parts, including fuel elements and fuel rods (Appendix I)

Fuel cladding, inner coatings, and other non-fuel materials:

Fo67b Cladding was Hastelloy-N (Fig. 2); coating was SCB-1

Fo68 Chemical composition and mechanical properties of Hastelloy-N used in the experiment (Tables 2, 3); composition of SCB-1 hydrogen barrier (Table 4); chromizing and hydrogen barrier coating thicknesses (Table 5)

Potential fuel chemical reactions:

Potential RCRA-regulated materials:

Quantities of materials:

See shipping spreadsheets (fuel rods, U, ^{235}U)

Fuel assembly identification methods:

Fo67b Photograph showing fuel element ID scribed on end (Fig. 7)

13.5 NAA-115 IRRADIATION HISTORY

Experiment Operating History:

Fo67b Approximate 10,000-hour irradiation time; cladding temperatures varied from 970 °F to 1335 °F, with calculated fuel centerline temperatures ranging from 1290 °F to 1620 °F

Fo68 Detailed cladding thermal history data (Table 16)

Neutron flux profile:

Operation abnormalities:

Fo67b Experiment was discharged from the Hanford K-East reactor after it experienced a full reactor scram at design temperatures. It was cooled from design temperatures (1300 °F to 1600 °F fuel centerline) to coolant temperature (~250 °F) in 80 seconds.

13.6 NAA-115 POST-IRRADIATION PROPERTIES AND EXAMINATION

Fo67b Detailed post-irradiation examination report

Fo68 Summary and detailed analysis of information from Fo67b, plus additional design and instrumentation (including dosimetry) details; comparisons with NAA-115-1 and NAA-117-1 experiment results; post-irradiation photographs of each specimen with data summaries, including locations from which samples were taken (Appendix II)

Visual observations:

Fo67a A large volume of post-irradiation photographs

Fo67b Visual and photographic results of capsules and the fuel elements they contained; visual and photographic results of the fuel rods after disassembly

Fo68 Representative photographs of post-irradiation fuel elements and fuel rods

Dimensions:

Fo67b Cladding & fuel rod dimensional measurements, fuel rod densities

Metallurgical states of materials:

Fo67b Photomicrographs of the fuel and cladding (including color photomicrographs and identification of fuel material phases); fuel and cladding hardness measurements; cladding tensile strengths

Fo68 Photomicrographs and hardness data for fuel and cladding

Cladding defects:

Fo67b Photographs of cladding cracks in fuel elements IS 22, 29, 30 (Fig. 4)

Corrosion or chemical contamination:

Fo67b Photograph of coating adhering to fuel (Fig. 13)

Hydrogen effects:

Fo67b Hydrogen analysis

Burnup:

Fo67b Calculated burnup based on thermal history 0.33 to 0.37 metal-atom %; burnup analysis provided; fuel rod autoradiograph profiles for burnup analysis

Fission products:

Fo67b Fission gas analysis

Transuranics:

Activation products:

Fo67b Fuel element gamma scan results

Decay heat generation:

Radiation level decay curves:

Alpha contamination:

(not relevant for the declad fuel)

13.7 NAA-115 ON-SITE STORAGE HISTORY

The NAA-115-2 test assembly was sectioned at Hanford and the six capsules were shipped to the Atomics International Hot Laboratory (AIHL). They were received at the AIHL in two different cask shipments in December 1966 (Fo67b). The fuel materials were stored at the AIHL while post-irradiation examinations were proceeding, and then shipped to AI's Radioactive Materials Disposal Facility (RMDF) for temporary storage before shipment to INEL. They were always stored in air at both the RMDF and the AIHL.

13.8 NAA-115 SHIPPING RECORDS

The NAA-115-2 fuel rod material (from two irradiation capsules) shipped to INEL is listed in the S8ER Shipping Record Summary, where it is included under shipment LAE-JWA-26.

13.9 REFERENCES

Bi67* K. R. Birney, "An Empirical Study of SNAP Reactor Fuel Irradiation Behavior," Atomics International Report NAA-SR-12284 (August 1967)

Fo67a R. E. Forrester, "Post-Irradiation Photographs for the NAA 115-2 Experiment," Atomics International Technical Data Record NAA-SR-MEMO-12430 (April 1967) (366 pages)

Fo67b R. E. Forrester, "Post-Irradiation Examination of the NAA-115-2 Irradiation Experiment," NAA-SR-MEMO-12588, Atomics International Technical Data Record (November 1967) (136 pages)

Fo68 R. E. Forrester and W. J. Roberts, "In-Pile Behavior of SNAP 8 Experimental Reactor Type Sublength Fuel Elements (NAA-115-2 Experiment)," Atomics International Report NAA-SR-12625 (June 1968) (190 pages)

14. NAA-117-1 IN-PILE FUEL EXPERIMENT

Irradiation start date:	Not available
Irradiation end date:	Not available
Thermal power:	Not available
Irradiation location:	Hanford K-West Reactor

14.1 NAA-117 GENERAL DESCRIPTION

NAA-117-1 was one of three similar irradiation experiments (NAA-115-1, NAA-115-2, and NAA-117-1) that were statistically designed to test and evaluate SNAP 8 fuel elements. NAA-117-1 differed from the NAA-115 experiments in that it contained 16 sub-length fuel elements with target burnups ranging from 0.3 to 0.8 metal-atom % (Fo67). The measured burnup (0.37 to 0.77 metal-atom %) is the highest of any of the SNAP fuels. The NAA-117 fuel element pieces were shorter rods taken from the S8ER production inventory and irradiated at the Hanford K-West reactor. Fabrication data, compositions, etc. would thus be as described for the S8ER fuel rods. The fuel elements were formed into 6 separate experimental capsules which were connected in series to form a 15-foot-long test assembly.

A good general description of the experiment, its objectives, and the results is given in Co67, and a summary of several in-pile experiments is provided in Bi67.

14.2 NAA-117 FUEL IDENTIFICATION

Reactor type: S8ER
Fuel identification: NAA-117

Fuel Specimen Identification: M-1 through M-18 (Co67)

Experiment designer, builder: Atomics International
Reactor owner: Atomic Energy Commission (now DOE)

Fuel designer, builder: Atomics International
Fuel owner: Atomic Energy Commission (now DOE)

14.3 NAA-117 REACTOR EXPERIMENT DESCRIPTION

Experiment design parameters:

Co67 List of parameters (Table 8) and experiment description; specimens enclosed in six capsules, arranged in series.

Fo67 16 fuel specimens in 6 capsules, schematic layout in Fig. 1

Design neutron flux:
Design neutron energy:
Core element locator:

14.4 NAA-117 FUEL FABRICATION INFORMATION

Fuel specifications and drawings:

- Co67 Design of fuel element specimens (Fig. 2B, 3)
- Fo67 Design of fuel element specimens (Fig. 2)
- Co67 Dimensional and compositional data (Table 1), compared with S8ER fuel elements
- Fo67 Fuel composition (Table 1)

Fuel fabrication and quality control:

- Co67 Fabrication description in Section II
- Co67 Listing of fuel element specimen numbers (Table 5), along with chem analyses for C,U,H
- Co67 Dimensional data and hydrogen leak rates (Tables 6, 7)

Fuel cladding, inner coatings, and other non-fuel materials:

- Co67 Fabrication description of cladding, end caps, and plugs in Section II: Hastelloy N; chemical analyses and mechanical properties in Table 2
- Co67 Cladding inner surface chromized and oxidized for adherence of inner coating to the cladding
- Co67 Coating = AI-8763D, composition given (Table 3); Coating thicknesses in Table 4

Potential fuel chemical reactions:

- Co67 NaK in contact with cladding outside surface

Potential RCRA-regulated materials:
Quantities of materials:

See shipping spreadsheets (fuel rods, U, ^{235}U)

Fuel assembly identification methods:

14.5 NAA-117 IRRADIATION HISTORY

Experiment Operating History:

- Fo67 Total irradiation time about 10,000 hours. Centerline temperatures varied from 1300 °F to 1500 °F. Cladding temperatures varied from 895 to 1315 °F.
- Co67 Vague irradiation data (originally classified); thermal data in Fig. 50 - 65

Neutron flux profile:
Operation abnormalities:

14.6 NAA-117 POST-IRRADIATION PROPERTIES AND EXAMINATION

- Fo67 A detailed description of the post-irradiation examination results
- Fo66 A two-volume collection of the photographs from the post-irradiation examination
- AI66 8 laboratory notebooks covering the post-irradiation hot-cell examination; Book 6 includes color photomicrographs
- Co67 Post-irradiation examination analysis, with results excerpted from Fo67. Includes summary figures of post-irradiation data and photographs of each specimen (Fig. 50 - Fig. 65), except for specimens M-4, M-8

Visual observations:

Co67 Visual notations; photographs of heavy spall
Fo66 Books of post-irradiation examination photographs (2 volumes, 774 pages)
Fo67 Visual and photographic results, from all states of disassembly; a "frosty" surface was observed on the outside of the as-received fuel elements, thought to be due to water residue on the hot surface of the experiments after they were moved (at Hanford) from the reactor core to the water pit

Dimensions:

Co67 Comparison pre- and post-irradiation lengths, diameters; pre- and post-irradiation density measurements
Fo67 Dimensional measurements

Metallurgical states of materials:

Co67 Discussion of fuel material after irradiation, showing spalling of some specimens; extensive fuel metallography (including photomicrographs)
Fo67 Metallographic analysis, including numerous photographs; hardness measurements

Cladding defects:

Co67 Cladding photomicrographs showing cladding failures and poor adherence of hydrogen barrier coating in some instances. Cladding metallography, hardness testing, and grain size.

Corrosion or chemical contamination:

Co67 Photographs of contamination of fuel material by cladding; plus visual descriptions of contamination from NaK and cladding by specimen number (M-1 through M-18).

Hydrogen effects:

Co67 Pre- and post-irradiation hydrogen analysis; tabulation of H/Zr effective.
Fo67 Hydrogen analysis

Burnup:

Fo67 Burnup analysis; calculated burnups were 0.37 to 0.77 metal-atom %
Co67 Burnup analysis from dosimetry and mass spectrometry; ^{95}Zr gamma activity

Fission products:

Fo67 Fission gas analysis; autoradiographs; gamma scan profiles
Co67 Fission gas analysis (^{85}Kr)

Transuranics:
Activation products:
Decay heat generation:
Radiation level decay curves:
Alpha contamination:

(not relevant for the declad fuels)

14.7 NAA-117 ON-SITE STORAGE HISTORY

The NAA 117-1 test assembly was sectioned at Hanford between each of the six capsules and shipped to the Atomics International Hot Laboratory (AIHL) for destructive post-irradiation analysis. They were received at the AIHL in two different cask shipments in the early part of July 1966 (Fo67). The fuel materials were stored at the AIHL while post-irradiation examinations were proceeding, and then shipped to AI's Radioactive Materials Disposal Facility (RMDF) for temporary storage before shipment to INEL. They were always stored in air at both the RMDF and the AIHL. Reference Fo67 indicates some underwater storage of the irradiated test assembly at Hanford on removal from the core and prior to shipment to AI.

14.8 NAA-117 SHIPPING RECORDS

The NAA-117-1 fuel rod material from two irradiation capsules, shipped to INEL with the S8ER fuel elements under shipment LAE-JWA-26, is listed in the S8ER Shipping Record Summary. Five pieces were shipped to INEL with the S8DR fuel elements in AI Can No. 10, shipment LAE-JSB-02, and are included in the S8DR Shipping Record Summary.

14.9 REFERENCES

AI66 Atomics International Laboratory Notebooks "M NAA 117-1," covering NAA-117-1 hot cell examinations: Book 1 (B 326601, July 1966), Book 2 (B 335101, August 1966), Book 3 (B335051, September 1966), Book 4 (B 326801, September 1966), Book 5 (B 335251, October 1966), Book 6 (B 345401, December 1966), Book 7 (B 345601, January 1967, and Book 8 (B345551, January 1967)

Bi67* K. R. Birney, "An Empirical Study of SNAP Reactor Fuel Irradiation Behavior," Atomics International Report NAA-SR-12284 (August 1967) (51 pages)

Co67 R. C. Courtright and R. E. Forrester, "In-Pile Behavior of SNAP 8 Experimental Reactor Type Sublength Fuel Elements (NAA 117 Experiment)," Atomics International Report NAA-SR-12032 (November 1967) (140 pages)

Fo66 R. E. Forrester, "Postirradiation Photographs for the NAA 117-1 Experiment," Atomics International Technical Data Record NAA-SR-MEMO-12212, Volumes 1 and 2 (November 1966) (674 pages)

Fo67 R. E. Forrester, "Post Irradiation Examination of the NAA-117-1 Experiment," Atomics International Technical Data Record NAA-SR-MEMO-12365 (March 1967) (144 pages)

15. NAA-121 IN-PILE FUEL EXPERIMENT

Irradiation start date:	October 15, 1967 (Le70)
Discharged:	July 4, 1969 (Le70)
Thermal power:	Not available
Irradiation location:	Hanford K-East Reactor

15.1 NAA-121 GENERAL DESCRIPTION

The NAA-121 experiment was a reactor fuel element performance test of eight uranium zirconium hydride fuel rods clad with Hastelloy N and coated with an SCB-1 glass hydrogen barrier. Four of the elements were irradiated in the Hanford K-East reactor (Le71). The primary objective of the in-pile experiment was to evaluate the performance of three full-length S8DR-type fuel elements under approximated S8DR reactor operating conditions. A fourth test element was included in the irradiation for in-pile phase transformation studies (Fo70). The uranium concentration in the irradiated fuel rods ranged from 10.32 to 10.58 wt. % (Le71). The four out-of-pile fuel elements were the same as the in-pile elements except that they contained natural uranium.

The NAA-121 in-pile experimental assembly included four individual experiments, denoted -1, -2, -3, and -4, designed to operate at three different power levels. In some reports these experiments are identified separately, such as NAA-121-1, and were called capsules (instead of experiments) during post-irradiation examination. Report Le71 identifies the overall experiment as NAA-121-1, but covers all four irradiated capsules.

15.2 NAA-121 FUEL IDENTIFICATION

Reactor type: S8DR
Fuel identification: S8DR-type, four full-length elements (Le71)

Fuel Specimen Identification: NAA-121-1, -2, -3, -4

Experiment designer, builder: Atomics International
Reactor owner: Atomic Energy Commission (now DOE)

Fuel designer, builder: Atomics International
Fuel owner: Atomic Energy Commission (now DOE)

15.3 NAA-121 REACTOR EXPERIMENT DESCRIPTION

Experiment design parameters:

Le70 As-built fuel element data and reference heat generation rates
Le71 In-pile neutron irradiation experiment

Design neutron flux:
Design neutron energy:
Core element locator:

15.4 NAA-121 FUEL FABRICATION INFORMATION

Fuel specifications and drawings:

Le70 Sketches of fuel element (Fig. 1) and capsule (Fig. 2) design

Le71 Schematic drawing of the fuel element (Fig. 1a)
Drawings available from loose files

Fuel fabrication and quality control:

Le71 Fuel element as-built dimensions and uranium content (Table 1); fuel alloy chemical composition, including U range of 10.32 to 10.58 wt.% (Table 2)

Fuel cladding, inner coatings, and other non-fuel materials:

Le71 Hastelloy-N cladding, SCB-1 coating; see S8DR

Potential fuel chemical reactions:
Potential RCRA-regulated materials:
Quantities of materials:

See shipping spreadsheets (fuel rods, U, ^{235}U)

Fuel assembly identification methods:

15.5 NAA-121 IRRADIATION HISTORY

Experiment Operating History:

Fo70 Detailed information on the operational history not reported because of its security classification (see Le70)

Le70 Operations summary (Table 2), including 15,000 hours in-pile time and 11,457 hours time at power; charts of time-temperature history for each thermocouple (Appendix 1); power graphs showing complete time power history of the experiment (Appendix 4)

Le71 Experiment ran for 12,000 hours at cladding temperatures up to 1300 °F and fuel temperatures up to 1450 °F at the Hanford K-East reactor. In-pile operations summary (Table 4).

Neutron flux profile:
Operation abnormalities:

15.4 NAA-121 POST-IRRADIATION PROPERTIES AND EXAMINATION

Le71 Schematic (Fig. 16) of the sectioning of the fuel rods for post-irradiation examination (hydrogen, metallurgical, and burnup), plus the identification system

Visual observations:

Fo70 Post-irradiation photograph of Elements 1-4 (Fig. 5 - 8)

Le71 Fuel element and fuel material sample photographs

Dimensions:

Fo70 Fuel element dimensional profile scans (Fig. 11); density measurements of sections from the four fuel rods (Table VI)

Le71 Density summary (Table 6)

Metallurgical states of materials:

Le69 Discussion of possible fuel phase changes in Element 4 based on early temperature profile information

Fo70 Neutron radiographs showing fuel-rod cracks and distortions within the cladding (Fig. 10); photographs of fuel rods from Elements 1,2,3 after cladding removal (Fig. 12-14); Element 4 cross section photographs (Fig. 15, approx. 19 pieces); photomicrographs of fuel rods 1,2,3,4 (Fig. 22 - 25)

Cladding defects:

Fo70 Element 4 cracked and shows evidence of NaK contamination (Fig. 8)

Corrosion or chemical contamination:

Fo70 Element 4 cracked and shows evidence of NaK contamination (Fig. 8)

Hydrogen effects:

Le70 Pre- and post-irradiation fuel rod hydrogen content

Fo70 Hydrogen analysis from Elements 1,2,3 (Table VIII)

Mi71 Hydrogen analysis (Table I)

Le71 Hydrogen retention summary (Table 6); barrier performance (hydrogen loss; Table 7)

Burnup:

Fo70 Calculated burnups based on temperature history (Table II)

Mi71 Fuel burnup analysis (Table II); source analytical chemistry data sheets by mass spectrometry are available

Le71 Peak burnup of 0.32 metal-atom %; burnup summary (Table 6)

Fission products:

Sw69a Preliminary fission gas release computation for Elements 1,2,3

Le71 Fission gas release summary (Table 8)

Fo70 Gamma scan profiles of the 4 elements (Fig. 9)

Transuranics:
Activation products:
Decay heat generation:
Radiation level decay curves:
Alpha contamination:

(not relevant for the declad fuels)

15.7 NAA-121 ON-SITE STORAGE HISTORY

The NAA-121 test assembly was sectioned at Hanford between each of the four experiments and the shroud tubes were removed. The experiments (capsules) were then shipped to the Atomics International Hot Laboratory (AIHL) for destructive post-irradiation analysis (Fo70). This analysis began in August 1969. The fuel materials were stored at the AIHL while post-irradiation examinations were proceeding, and then shipped to AI's Radioactive Materials Disposal Facility (RMDF) for temporary storage before shipment to INEL. They were always stored in air at both the RMDF and the AIHL.

15.8 NAA-121 SHIPPING RECORDS

The NAA-121 fuel rod material (from Experiments 1, 2, 3, and 4) shipped to INEL is listed in the S8DR Shipping Record Summary, where it is included under shipment LAE-JSB-02.

15.9 REFERENCES

Fo70 R. E. Forrester, "Post-Irradiation Examination of the NAA-121 Irradiation Experiment," Atomics International Technical Data Record AI-AEC-MEMO-12940 (March 1970) (53 pages)

Le69 J. G. LeBlanc and J. R. Miller, "NAA 121-1-4 Cladding Temperature Profiles," Atomics International Technical Information Document TI-696-24-048 (December 1969) (7 pages)

Le70 J. G. LeBlanc and H. J. Tribfelner, "NAA 121-1 Irradiation Experiment Operations Report," Atomics International Technical Data Record AI-AEC-MEMO-12930 (February 1970) (218 pages)

Le71 J. G. LeBlanc, "Prototype S8DR Fuel Element Performance Test (NAA 121-1 Experiment)," Atomics International Report AI-AEC-13002 (July 1971) (61 pages)

Mi71 K. J. Miller, "NAA-121 Fuel Burnup and Hydrogen Analysis," Atomics International Technical Information Document TI-696-240-083 (March 1971) (7 pages)

Sw69a L. D. Swensen, "Preliminary Computed Fission Gas Release Fraction from NAA-121-1 Fuel Elements," Atomics International Technical Information Document TI-568-24-032 (November 1969) (4 pages)

16. NAA-67 IN-PILE FUEL EXPERIMENT

Rebecca Lords of WINCO provided the information during her visit to ETEC on 22-23 September 1994 that two SNAP NAA-67 fuel pieces were also in storage at INEL. We have not located any documented evidence of this experiment, and have no descriptive information on it at this time. However, based on Ms. Lords' input, we have reviewed our archived shipping records and have identified shipments of NAA-67 irradiation capsules both to and from INEL (Phillips Petroleum Company). This indicates that some NAA-67 materials were sent to INEL for one or more irradiation experiments and subsequently returned to AI for analysis. These shipments are summarized as follows:

Shipments to INEL:

Date	Sample Identification		Uranium	^{235}U
04/26/62	NAA-67-1,2	No. 1,2,3,4,7,8,9,10	203 g	190 g
05/07/62	NAA-67-1,2	14 rod pieces	457 g	425 g
Totals:	*22 samples*		*660 g*	*615 g*

Shipments from INEL:

Date	Sample Identification	Uranium	^{235}U
10/02/62	NAA-67-1	46.0 g	43.0 g
11/19/62	NAA-67-3	46.0 g	43.0 g
09/17/62	NAA-67-4	42.0 g	39.0 g
11/23/62	NAA-67-5	46.3 g	43.0 g
08/10/62	NAA-67-6	46.3 g	43.0 g
03/15/63	NAA-67-7	5.8 g	5.5 g
03/15/63	NAA-67-8	5.8 g	5.5 g
03/15/63	NAA-67-9	5.7 g	5.5 g
03/15/63	NAA-67-10	5.7 g	5.5 g
08/10/62	NAA-67-11	46.2 g	43.0 g
11/23/62	NAA-67-12	46.2 g	43.0 g
10/26/62	NAA-67-13	41.5 g	38.7 g
11/27/62	NAA-67-14	41.5 g	38.7 g
03/15/63	NAA-67-15	41.5 g	38.7 g
03/15/63	NAA-67-16	41.5 g	38.7 g
03/15/63	NAA-67-17	41.5 g	38.6 g
03/15/63	NAA-67-18	41.5 g	38.6 g
11/19/62	NAA-67-19	5.8 g	5.3 g
11/08/62	NAA-67-21	5.7 g	5.2 g
09/17/62	NAA-67-22	5.7 g	5.2 g
Totals:	*20 samples*	*608.2 g*	*566.7 g*

The return shipping documents account for 20 of the 22 samples shipped to INEL. The two samples that were apparently not returned to AI, based on the numbering sequence of the samples that were returned, were samples NAA-67-2 and NAA-67-20. The material mass not returned, based on the differences between the two sets of totals, was 51.8 g of uranium, of which 48.3 g was ^{235}U (neglecting differences in round-off conventions). Based on the masses of samples with neighboring identification numbers, one would guess that NAA-67-2 weighed 46 g (43 g of ^{235}U) and NAA-67-20 weighed 5.7 g (5.2 g of ^{235}U). Their sum accounts for the difference between the AI-shipped and AI-received total NAA-67 uranium masses, within round-off differences.

APPENDICES

APPENDIX 1
COMBINED LISTING OF REFERENCE DOCUMENTS

Ad74 T. Adamson (Aerojet Nuclear), in *Atomics International Telephone Conversation Record* to V.J. Schaubert (AI) dated 28 January 1974

AI65 Atomics International, "S8ER- Screening Examination Dry Weight Data Sheets in Sequence from Ring I to Ring IX," Atomics International binder of source data sheets

AI66 Atomics International Laboratory Notebooks "M NAA 117-1," covering NAA-117-1 hot cell examinations: Book 1 (B 326601, July 1966), Book 2 (B 335101, August 1966), Book 3 (B335051, September 1966), Book 4 (B 326801, September 1966), Book 5 (B 335251, October 1966), Book 6 (B 345401, December 1966), Book 7 (B 345601, January 1967, and Book 8 (B345551, January 1967)

AI66a Atomics International Hot Laboratory Staff, "[S8ER] Post Irradiation Screening Examination Photographs," Atomics International Technical Data Records NAA-SR-MEMO-12011, Volumes I to XIV (1966) *(volumes not labeled with NAA number)*

AI66b Atomics International Hot Laboratory Staff, "S8ER Screening Examination Stereo Data Sheets in Sequence from Ring I to Ring IX," Atomics International source data book (1966)

AI66c Atomics International Hot Laboratory Staff, "S8ER Screening Examination Length Measurement Data Sheet in Sequence from Ring I to Ring IX," Atomics International source data book (1966)

AI66d Atomics International Hot Laboratory Staff, "S8ER Screening Examination Gamma Scanning Data Sheets in Sequence from Ring I to Ring IV, from Ring V to Ring VII, *and* from Ring VIII to Ring IX," Atomics International source data books, 3 volumes (1966)

AI66e Atomics International Hot Laboratory Staff, "S8ER Screening Examination Diameter Measurements Data Sheets in Sequence from Ring I to Ring V, *and* from Ring VI to Ring IX," Atomics International source data books, 2 volumes (1966)

AI66f Atomics International Hot Laboratory Staff, "S8ER Screening Examination Hydrogen Analysis Data Sheets in Sequence from Ring I to Ring IX," Atomics International source data book (1966)

Am68 C. B. Amberson (Idaho Nuclear Corporation), "SNAP Fuels from A.I. in 1967-1968 Fai-9-68," Interoffice Correspondence

AN76 *Anonymous*, "Zirconium Hydride (SNAP) Reactor Space Power Programs. Summary of Major Activities and Accomplishments," Rockwell International Energy Systems Group (June 1976)

An83 R. V. Anderson, et al., "Space Reactor Electric Systems. Subsystem Technology Assessment," Rockwell International Energy Systems Group Report ESG-DOE-13398 (March 1983)

Ar63 J. R. Armstrong, et al., "SNAP 10 A FS-3, -4, and -5 Fabrication Process Sheets and Auxiliary Forms," Atomics International Technical Data Record NAA-SR-MEMO-8809 (July 1963) (41 pages)

Ar74 J. R. Armstrong, "Uranium-Zirconium Fuel Alloy, Hydriding of," Atomics International Specification NA0611-001 (January 1974) (11 pages)

Ar77 D. V. Arnold, C. C. Woolsey, and W. H. Friske, "Specification for Marking," Atomics International Specification N001A0104001 (June 1977) (7 pages)

Be62 J. P. Beall and M. W. Hulin, "SNAP-2 Experimental Reactor Operation and Test Program," Atomics International Report NAA-SR-7088 (April 1962) (76 pages)

Be94 C. L. Bendixsen, in *Proc. 1994 International High Level Radioactive Waste Management Conference*, American Nuclear Society, 22-26 May 1994

Bi67* K. R. Birney, "An Empirical Study of SNAP Reactor Fuel Irradiation Behavior," Atomics International Report NAA-SR-12284 (August 1967) (51 pages)

Bl59 M. Bloomfield, "Tentative SER Fuel Element Inspection Specification," Atomics International Memorandum NAA-SR-MEMO-4063 (June 1959) (8 pages)

Bu64 G. F. Burdi, Ed., "SNAP Technology Handbook. Volume II. Hybrid Fuels and Claddings," Atomics International Report NAA-SR-8617, Volume II (November 1964) (129 pages)

Bu67* J. S. Buck, W. E. Krupp, and T. A. Park, "Irradiation Performance of a Full-Scale Prototype SNAP 2 Reactor Fuel Element," Atomics International Report NAA-SR-12030 (August 1967) (82 pages)

Bu70 D. S. Burgess, "Statistical Evaluation of the Fuel-Clad Gap and Crack Length from S8DR Fuel Elements," Atomics International Technical Information Document TI-568-24-058 (June 1970) (14 pages)

Ca63 D. C. Campbell, et al., "SNAP 8ER Fabrication Process Sheets and Auxiliary Forms," Atomics International Technical Data Record NAA-SR-MEMO-8273 (February 1963) (41 pages)

Ca63a D. C. Campbell, et al., "SNAP 10A FS-1 Fabrication Process Sheets and Auxiliary Forms," Atomics International Technical Data Record NAA-SR-MEMO-8272 (February 1963) (37 pages)

Ca64 D. C. Campbell, et al., "SNAP 8 Experimental Reactor Fabrication and Engineering," Atomics International Report NAA-SR-8430 (March 1964) (55 pages)

Ce64 W. M. Cegelski, "Fission Product Release from Irradiated Uranium-Zirconium Hydride SNAP Fuel," Atomics International Report NAA-SR-9673 (July 1964) (37 pages)

Co67 R. C. Courtright and R. E. Forrester, "In-Pile Behavior of SNAP 8 Experimental Reactor Type Sublength Fuel Elements (NAA 117 Experiment)," Atomics International Report NAA-SR-12032 (November 1967) (140 pages)

Co71 L. Cooper, "Results from Abbreviated EMP Examination of Three S8DR Irradiated Fuel Specimens," Atomics International Technical Information Document TI-653-240-003 (July 1971) (7 pages)

Cu65 G. E. Culley, "Analysis of SNAPTRAN-3 Destructive Experiment Debris," Atomics International Report NAA-SR-11584 (December 1965) (37 pages)

Da70 N. F. Davies, "S8DR Examination," Atomics International Technical Information Document TI-568-24-049 (April 1970) (3 pages)

Da70a W. H. Dauterman, "Bowing of S8DR Fuel," Atomics International Technical Information Document TI-568-24-042 (March 1970) (20 pages)

De62* W. F. Dennison and T. S. Jakobowski, "Assembly of S-2 DR Fuel Elements," Atomics International Report NAA-SR-7048 (December 1962) (26 pages)

De65 R. L. Detterman, "Progress Report: SNAP Aerospace Safety Program April-June 1965," Atomics International Report NAA-SR-11497 (August 1965) (one of a series, 39 pages)

De67 W. F. Dennison and P. J. Fessia, "Fuel Element Assembly, S8DR," Atomics International Specification ST0622NA0041 (May 1967) (14 pages)

Di67 H. M. Dieckamp, "Nuclear Space Power Systems," Atomics International (September 1967) (unpublished book, 388 pages)

Do67 J. P. Dooley and J. P. Beall, "The Disassembly and Postoperation Component Examination of the SNAP 10A FS-3," Atomics International Report NAA-SR-12504 (September 1967) (70 pages)

Do68* C. Donnelly and W. Nagel, "S8DR Fuel Element Quality Assurance Record," Atomics International Technical Data Record AI-AEC-TDR-12702 (March 1968) (137 pages)

Dr60 P. S. Drennan, "Melting and Forming of SER Fuel Rods," Atomics International Report NAA-SR-4762 (October 1960) (30 pages)

Dr61 P. S. Drennan and W. Sawicky, "Summary of Test Data for S-2-DR Fuel Elements," Atomics International Technical Data Record NAA-SR-MEMO-6745 (September 1961) (32 pages)

Eg59 R. R. Eggleston and G. L. Schmidt, Ed., "SNAP II Experimental Reactor Hazards Report," Atomics International Report NAA-SR-3465 (April 1959) (156 pages)

Ei70 R. L. Eichelberger, "Coating on S8DR Fuel Cladding," North American Rockwell Corporation Aerospace and Systems Group Internal Letter dated 3 August 1970 (6 pages)

Fe59 L. D. Felten, "Axial Temperature Distribution of S.E.R. Central Fuel Element," Atomics International Technical Data Record NAA-SR-MEMO-5121 (October 1959) (8 pages)

Fe64 L. D. Felten, et al., "Final Report on the SNAP 2 Development Reactor (S2DR) Test Program," Atomics International Report NAA-SR-8295 (April 1964) (156 pages)

Fi94 D. L. Fillmore, "Categorization of Department of Energy Spent Nuclear Fuel," in *Proc. 1994 International High Level Radioactive Waste Management Conference*, American Nuclear Society, 22-26 May 1994

Fo66 R. E. Forrester, "Postirradiation Photographs for the NAA 117-1 Experiment," Atomics International Technical Data Record NAA-SR-MEMO-12212, Volumes 1 and 2 (November 1966) (674 pages)

Fo67 R. E. Forrester, "Post Irradiation Examination of the NAA-117-1 Experiment," Atomics International Technical Data Record NAA-SR-MEMO-12365 (March 1967) (144 pages)

Fo67a R. E. Forrester, "Post-Irradiation Photographs for the NAA 115-2 Experiment," Atomics International Technical Data Record NAA-SR-MEMO-12430 (April 1967) (366 pages)

Fo67b R. E. Forrester, "Post-Irradiation Examination of the NAA-115-2 Irradiation Experiment," NAA-SR-MEMO-12588, Atomics International Technical Data Record (November 1967) (136 pages)

Fo68 R. E. Forrester and W. J. Roberts, "In-Pile Behavior of SNAP 8 Experimental Reactor Type Sublength Fuel Elements (NAA-115-2 Experiment)," Atomics International Report NAA-SR-12625 (June 1968) (190 pages)

Fo70 R. E. Forrester, "Post-Irradiation Examination of the NAA-121 Irradiation Experiment," Atomics International Technical Data Record AI-AEC-MEMO-12940 (March 1970) (53 pages)

Gi56 P. T. Gilbert, Jr., "Zirconium Hydride: A Compendium on the Systems Zirconium-Hydrogen and Hafnium-Hydrogen and Related Topics," Atomics International Report NAA-SR-1508 (October 1956) (834 pages)

Go67* T. A. Golding, "Post-Operation Evaluation of Fuel Elements from the SNAP 10 Flight System 3 Reactor," Atomics International Report NAA-SR-12031 (September 1967) (41 pages)

Go67a T. A. Golding, "Post-Irradiation Screening Examination of the S10FS-3 Fuel Elements," Atomics International Technical Data Record NAA-SR-MEMO-12338 (February 1967) (29 pages)

Go67b T. A. Golding, "Post-Irradiation Examination of Selected S10FS-3 Fuel Elements," Atomics International Technical Data Record NAA-SR-MEMO-12341 (June 1967) (28 pages)

Go67c K. G. Golliher, et al., "SNAP Engineering Test Facility Safety Analysis Report: Addendum for Intrinsic Supercriticality Experiments," Atomics International Technical Data Record NAA-SR-MEMO-12614 (November 1967) (80 pages)

Gr83 L. F. Grantham, et al., "Process Description and Plant Design for Preparing Ceramic High Level Waste Forms," Rockwell International Energy Systems Group Report ESG-DOE-13397 (February 1983) (71 pages)

Ha65 S. D. Harkness, "The Oxidation Kinetics of SNAP Fuel Alloy: Modified 90-10 Zirconium-Uranium," Atomics International Technical Data Record NAA-SR-MEMO-10914 (January 1965) (33 pages)

Ha69 F. J. Halfen, "Summary of Selected S8DR Operating Data from 1-19-69 to 4-28-69," Atomics International Technical Information Document TI-696-13-033 (May 1969) (19 pages)

Hu61 M. W. Hulin and J. Beall, Ed., "Preliminary Results of the SNAP 2 Experimental Reactor," Atomics International Report NAA-SR-5991 (April 1961) (60 pages)

Hu64 H. J. Hubbell and W. E. Nagel, "SNAP 8 Experimental Reactor Fuel Element Acceptance Testing," Atomics International Report NAA-SR-8589 (March 1964) (63 pages)

Ja73 A. A. Jarrett, "SNAP 2 Summary Report," Rockwell International, Atomics International Division Report AI-AEC-13068 (July 1973) (190 pages)

Jo61* R. P. Johnson, "SNAP Shield Test Experiment Operations Manual," Atomics International Report NAA-SR-5897 (June 1961) (196 pages)

Jo64 H. E. Johnson, "Hydrogen Dissociation Pressures of Modified SNAP Fuels," Atomics International Report NAA-SR-9295 (March 1964) (47 pages)

Ka59 N. H. Katz, "SER Data Report," Atomics International Memo NAA-SR-MEMO-4722 (December 1959) (25 pages)

Ke70 N. Ketzlach, "Criticality Study - S8DR Core Disassembly," Atomics International Technical Information Document TI-568-24-041 (January 1970) (15 pages)

Ke71 N. Ketzlach, "Storage of S8DR Fuel at RMDF," Atomics International Technical Information Document TI-568-240-083 (May 1971) (4 pages)

Ki63 T. S. Kirsch, "Specifications for Alloying and Forming SNAP 10A and 2 Fuel Materials," Atomics International Technical Data Record NAA-SR-MEMO-6645, Revised (August 1963) (15 pages)

Kr66 W. E. Krupp, "Post-Irradiation Examination of the NAA 82-1 Experiment," Atomics International Technical Data Record NAA-SR-MEMO-11891 (March 1966) (51 pages)

Kr67 W. E. Krupp, "Post-Irradiation Metallographic Examination of Selected S10FS-3 Reactor Components," Atomics International Technical Data Record NAA-SR-TDR-12390 (March 1967) (10 pages)

La51 R. A. Laubenstein, "Preliminary Considerations on the Feasibility of an Auxiliary Power Plant to be Operated from Radioactive Fission Products," Atomics International Report NAA-SR-152 (October 1951) (22 pages)

La67 K. Langrod, "Process Flow Sheets - S8DR Hydrogen Barrier Material Fabrication," Atomics International Technical Data Record NAA-SR-MEMO-12469 (May 1967) (27 pages)

La68 E. M. Larson, "Radiological Safety and Hazardous Materials Handbook for Building 059," Atomics International Technical Data Record AI-AEC-12682 (April 1968) *(part of S8DR Operations Manual)*

La68a E. M. Larson, "S8DR Operations Manual," Atomics International Technical Data Record AI-AEC-12683 (May 1968) *(part of S8DR Operations Manual)*

La68b E. M. Larson, "SNAP Ground Prototype Test Facility (Building 059)," Atomics International Technical Data Record AI-AEC-12684 (April 1968) *(part of S8DR Operations Manual)*

Le66 S. K. Lee, "A Study of Dehydriding of S8ER Fuel," Atomics International Technical Data Record NAA-SR-MEMO-12197 (October 1966) (20 pages)

Le69 J. G. LeBlanc and J. R. Miller, "NAA 121-1-4 Cladding Temperature Profiles," Atomics International Technical Information Document TI-696-24-048 (December 1969) (7 pages)

Le70 J. G. LeBlanc and H. J. Tribfelner, "NAA 121-1 Irradiation Experiment Operations Report," Atomics International Technical Data Record AI-AEC-MEMO-12930 (February 1970) (218 pages)

Le71 J. G. LeBlanc, "Prototype S8DR Fuel Element Performance Test (NAA 121-1 Experiment)," Atomics International Report AI-AEC-13002 (July 1971) (61 pages)

Li70 A. F. Lillie, "The S8DR Screening Examination," Atomics International Technical Information Document TI-568-240-068 (October 1970) (112 pages)

Li70a A. F. Lillie and K. J. Miller, "S8DR Screening Examination Weekly Status Reports," Atomics International Technical Information Documents: #1 (TI-568-24-046, 20 March 1970, 5 pages), #2 (TI-568-24-048, 27 March 1970, 8 pages), #3 (TI-568-24-050, 3 April 1970, 8 pages), #4 (TI-568-24-051, 10 April 1970, 13 pages), #5 (TI-568-24-052, 17 April 1970, 27 pages), #6 (TI-568-24-053, 24 April 1970, 22 pages), #7 (PSR-568-24-001, 1 May 1970, 17 pages), #8 (PSR-568-24-002, 8 May 1970, 13 pages), #9 (PSR-568-24-003, 15 May 1970, 15 pages), #10 (PSR-568-24-004, 22 May 1970, 31 pages), #11 (PSR-568-24-005, 29 May 1970, 19 pages), #12 (PSR-568-24-006, 5 June 1970, 31 pages), #13 (PSR-568-24-007, 12 June 1970, 38 pages), #14 (PSR-568-24-008, 19 June 1970, 29 pages), #15 (PSR-568-24-009, 26 June 1970, 26 pages), and #16 (PSR-568-24-010, 3 July 1970, 22 pages)

Li71* A. H. Lillie and V. L. Rooney, Jr., "The SNAP 8 Developmental Reactor (S8DR) Post-Test Examination," Atomics International Report AI-AEC-13003 (June 1971) (93 pages)

Li73 A. F. Lillie, D. T. McClelland, W. J. Roberts, and J. H. Walter, "Zirconium Hydride Fuel Element Performance Characteristics," Atomics International Report AI-AEC-13084 (June 1973) (75 pages)

Lu64 L. B. Lundberg, "Parametric Studies on the Short-Term Tensile Mechanical Properties of Zr-10U Alloy Hydrides," Atomics International Report NAA-SR-9875 (July 1964) (42 pages)

Ma64 C. A. Martin and W. H. Timmerman, "Process Flow Sheet for SNAP 8DS Melting and Extruding," Atomics International Technical Data Record NAA-SR-MEMO-9630 (March 1964) (17 pages)

Ma67 C. A. Martin, "Process Flow Sheets - S8DR Cup Plugs," Atomics International Technical Data Record NAA-SR-MEMO-12470 (May 1967) (18 pages)

Mc65 D. D. McAfee, "Hot Laboratory Metallography of SNAP 8 Fuel," Atomics International Technical Data Record NAA-SR-MEMO-11217 (March 1965) (12 pages)

Mc67 D. T. McClelland, "Process Flow Sheets - S8DR Fuel Element Fabrication," Atomics International Technical Data Record NAA-SR-MEMO-12473 (May 1967) (19 pages)

Mc67a D. T. McClelland, K. Langrod, and C. A. Martin, "Process Flow Sheets - S8DR Cladding Tube Fabrication," Atomics International Technical Data Record NAA-SR-MEMO-12475 (May 1967) (67 pages)

Mc69 D. T. McClelland, "S8DR Fuel Rod Fabrication," Atomics International Technical Data Record AI-AEC-TDR-12775 (January 1969) (43 pages)

Mc69a D. T. McClelland, "SNAP 8 Development Reactor Fuel Element Fabrication," Atomics International Report AI-AEC-12806 (June 1969) (55 pages)

Mc71 W. R. McCurnin, "S8DR Fuel Canning and Storage," North American Rockwell Internal Letter dated 22 June 1971 (5 pages)

Mc71a W. R. McCurnin, "FINAL S8DR Fuel Canning and Storage," North American Rockwell Internal Letter dated 30 June 1971 (5 pages)

Mc83 R. L. McKisson, et al., "Commercial High Level Waste Management, Options and Economics: A Comparative Analysis of the Ceramic and Glass Waste Forms," Rockwell International Energy Systems Group Report ESG-DOE-13391 (February 1983) (145 pages)

Mi63* J. R. Miller, "Postirradiation Evaluation of SER Fuel Elements," Atomics International Report NAA-SR-8090 (May 1963) (29 pages)

Mi64 K. J. Miller, "Postirradiation Examination of the SNAP-2 Developmental Reactor Fuel Elements," Atomics International Technical Data Record NAA-SR-MEMO-9753 (April 1964) (74 pages)

Mi65 S. Minor, L. Bixon, and D. Brinkman, "Preliminary Test Results - SNAP 10A FS-3," Atomics International Report NAA-SR-11206 (September 1965) (44 pages)

Mi66 K. J. Miller, "Post-Irradiation Screening Examination of the S8ER Fuel Elements," Atomics International Technical Data Record NAA-SR-MEMO-11880 (March 1966) (299 pages)

Mi66a K. J. Miller, "Post-Irradiation Detailed Examination of Selected S8ER Fuel Elements," Atomics International Technical Data Record NAA-SR-MEMO-12165 (1966) (complete color copy) (304 pages)

Mi66b K. J. Miller, "S8ER Screening Examination Photograph Books," Atomics International Internal Letter to G. W. Meyers dated 17 June 1966

Mi67 K. J. Miller, "Electron Microscopy Analysis of Irradiated S8ER and NAA 115-1 Fuel," Atomics International Technical Data Record NAA-SR-MEMO-12368 (March 1967) (51 pages)

Mi67a K. J. Miller, "Electron Microscopy Analysis of Irradiated SNAP 8 Fuel," Atomics International Report NAA-SR-12449 (August 1967) (66 pages)

Mi69 K. J. Miller, "S8DR Examination Planning Document for AIHL," Atomics International Test Plan TP-568-24-004 (November 1969) (51 pages)

Mi69a K. J. Miller, "S8DR Examination Planning Document for AIHL," Atomics International Technical Information Document TI-568-24-037 (December 1969) (42 pages)

Mi70 K. J. Miller, "Selection of S8DR Fuel Elements for the Detailed Postirradiation Examination," Atomics International Technical Information Document TI-568-24-061 (June 1970) (11 pages)

Mi70a K. J. Miller, et al., "S8DR Detailed Examination Progress Report," Atomics International Program Status Reports: Number 1 (PSR-568-240-011, September 1970, 36 pages), Number 2 (PSR-568-240-012, October 1970, 62 pages), Number 3 (PSR-568-240-013, November 1970, 54 pages), Number 4 (PSR-568-240-014, December 1970, 11 pages), Number 5 (PSR-568-240-015, January 1971, 15 pages), Number 6 (PSR-568-240-016, February 1971, 44 pages), Number 7 (PSR-568-240-017, March 1971, 26 pages), Number 8 (PSR-568-240-018, May 1971, 23 pages), Number 9 (PSR-653-240-001, July 1971, 92 pages), and Number 10 (PSR-653-240-002, July 1971, 29 pages)

Mi70b K. J. Miller, "S8DR Fuel Accountability Losses at AIHL," North American Rockwell Corporation Aerospace and Systems Group Internal Letter dated 21 December 1970 (2 pages)

Mi70c M. Mike, "Fuel Rod Dimensional Data/Plots From N-Ray Results - S8DR Screening Examination," Atomics International Technical Information Document TI-568-24-066 (July 1970) (213 pages)

Mi71 K. J. Miller, "NAA-121 Fuel Burnup and Hydrogen Analysis," Atomics International Technical Information Document TI-696-240-083 (March 1971) (7 pages)

Mo68 L. Morrison and E. L. Babbe, "S8ER Fuel Rod and Fuel Element Pre-Irradiation Measurements," Atomics International Technical Information Document TI-S8DR-28-012 (May 1968) (16 pages)

Mo68a K. E. Moore, W. A. Young, M. M. Nakata, and C. A. Smith, "Phase Relations at High Hydrogen Contents in the SNAP Fuel System," Atomics International Report NAA-SR-12587 (February 1968) (55 pages)

Mo68b K. E. Moore and M. M. Nakata, "Phase Relationships in the ($\alpha + \delta$) Region of the Zr-H System," Atomics International Report AI-AEC-12703 (September 1968) (47 pages)

Mo90 W. G. Moffett, Ed., *The Handbook of Binary Phase Diagrams*, Vol. V (Genium Publishing Co., Schenectady, NY, 1990 update)

Na62 W. E. Nagel, "S8ER Fuel Element Data Package," Atomics International Technical Data Record NAA-SR-MEMO-7623 (October 1962) (1078 pages)

Na63 W. E. Nagel, "S8ER Fuel Element Quality Control - Upgrading Point Tests and Inspections," Atomics International Technical Data Record NAA-SR-8283 (February 1963) (44 pages)

Na67 M. M. Nakata, C. J. Ambrose, and R. A. Finch, "Thermophysical Properties of SNAP Fuel Materials," Atomics International Report NAA-SR-11959 (September 1967) (93 pages)

Ol63 P. S. Olson, "Hot Cell Examination Requirements of S2DR Fuel Elements," Atomics International Technical Data Record NAA-SR-8716 (July 1963) (8 pages)

Ol65* P. S. Olson, " Evaluation of Fuel Elements from SNAP 2 Developmental Reactor Core," Atomics International Report NAA-SR-9648 (January 1965) (35 pages)

Ol67* P. S. Olson, K. J. Miller, and E. J. Donovan, "Postoperation Evaluation of Fuel Elements from the SNAP 8 Experimental Reactor," Atomics International Report NAA-SR-12029 (September 1967) (223 pages)

Ol91 R. D. Oldenkamp and J. C. Mills, "Nuclear Operations at Rockwell's Santa Susana Field Laboratory - A Factual Perspective," Rockwell International, Rocketdyne Division Supporting Document N001ER000017 (September 1991) (95 pages)

Pe66 H. Pearlman, et al., "SNAP 8 Experimental Reactor Fuel Element Behavior: Atomics International Task Force Review," Atomics International Report NAA-SR-MEMO-12210 (November 1966) (73 pages)

Ra63 J. W. Raymond, "Development of a High Yield 90 Zr - 10 U Alloy Massive Hydride," Atomics International Report NAA-SR-7305 (January 1963) (22 pages)

Ra65 J. W. Raymond, "Phase Relationships and Microstructures in the Zirconium - 10 wt % Uranium Alloy Hydride System," Atomics International Report NAA-SR-10965 (November 1965) (36 pages)

Ra65a J. W. Raymond and D. T. Shoop, "The Metallography of Zirconium-Base Alloy Hydrides," Atomics International Technical Data Record NAA-SR-TDR-10927 (February 1965) (9 pages)

Re59 E. L. Reed, "SER Fuel Element Specification," Atomics International Internal Letter dated 11 June 1959 (4 pages)

Ro61* C. J. Romero and S. Elchyshyn, "Cladding Development for SER Fuel Elements," Atomics International Report NAA-SR-4831 (February 1961) (42 pages)

Ro70 H. Rood, "S8DR Fuel Burnup and Transmutation Data," Atomics International Technical Information Document TI-568-24-063 (July 1970) (5 pages)

Sa63 W. Sawicky, "Fabrication Process Test and Inspection Sheets for SNAP 10A Flight System Cores 3, 4, and 5 Fuel Elements," Atomics International Technical Data Record NAA-SR-MEMO-9367 (November 1963) (46 pages)

Sa64 W. Sawicky, "SNAP 10A FS-1 Core-Fuel Element Data Packages for First Delivery," Atomics International Technical Data Record NAA-SR-MEMO-9855 (April 1964) (414 pages)

Sa64a W. Sawicky, "SNAP 10A FS-1 Core-Fuel Element Data Packages for Second Delivery," Atomics International Technical Data Record NAA-SR-MEMO-9926 (May 1964) (192 pages)

Sa64b W. Sawicky, "SNAP 10A FS-3 Core-Fuel Element Data Packages," Atomics International Technical Data Record NAA-SR-MEMO-10033 (June 1964) (171 pages)

Sa64c W. Sawicky, "SNAP 10A FS-4 Core-Fuel Element Data Packages," Atomics International Technical Data Record NAA-SR-MEMO-10208 (July 1964) (171 pages)

Sa64d W. Sawicky, "Addendum to SNAP 10A FS-1, FS-3, and FS-4 Fuel Data Packages for FS-3, FS-4, and FS-5 Systems," Atomics International Technical Data Record NAA-SR-MEMO-10815 (December 1964) (84 pages)

Sa66 W. B. Sayer and R. S. Hart, "The Predicted Fission Product Decay of Snapshot-1," Atomics International Report NAA-SR-11642 (January 1966) (20 pages)

Sc65 A. R. Schmitt, "Post-Irradiation Examination of NAA-82-1," Atomics International Internal Letter dated 27 October 1965 (4 pages)

Sc70 F. G. Schmidt, "S8DR Fuel Element Sample Numbering System," North American Rockwell Corporation Aerospace and Systems Group Internal Letter dated 13 July 1970 (2 pages)

Sc71 F. G. Schmidt, "S8DR Fuel Cutting Losses Assigned to Scrap at AIHL," North American Rockwell Corporation Aerospace and Systems Group Internal Letter dated 1 July 1971 (1 page)

Sc73 F. G. Schmidt, "Shipment of SNAP Fuel to RMDF," North American Rockwell Internal Letter dated 16 March 1973 (2 pages)

Sh64 D. T. Shoop, "Metallographic Preparation of Zirconium Hydride," Atomics International Technical Data Record NAA-SR-MEMO-10145 (June 1964) (7 pages)

Si61 M. T. Simnad and W. P. Wallace, "The Metallurgy of the TRIGA Fuel Elements," *Elecktrotechnik und Maschinenbau 78*, 581 (1961)

St63 D. H. Stone, "Corrosion Tests of Candidate SNAP 4 Fuel Element Materials," Atomics International Technical Data Record NAA-SR-TDR-8068 (February 1963) (6 pages)

St67* D. W. Staub, "SNAP 10A Summary Report," Atomics International Report NAA-SR-12073 (March 1967) (237 pages)

Sw67 L. D. Swenson, "S8DR Core Performance Evaluation," Atomics International Report NAA-SR-12482 (August 1967) (75 pages)

Sw69* L. D. Swenson, "SNAP 8 Development Reactor Nuclear Analysis," Atomics International Report AI-AEC-12864 (October 1969) (83 pages)

Sw69a L. D. Swensen, "Preliminary Computed Fission Gas Release Fraction from NAA-121-1 Fuel Elements," Atomics International Technical Information Document TI-568-24-032 (November 1969) (4 pages)

Sw70 V. A. Swanson, "Neutron Radiography as a Means for Determining H/Zr in SNAP-Type Fuel Elements," Atomics International Technical Information Document TI-568-24-056 (May 1970) (16 pages)

Ta63 H. Taketani, "Hydriding of SNAP 2 Fuel Rods," Atomics International Report NAA-SR-5037 (June 1963) (37 pages)

Ta65 H. Taketani, L. Silverman, and W. L. Korst, "Identification of the Modifier in SNAP Fuel," Atomics International Report NAA-SR-10174 (February 1965) (21 pages)

To61* R. L. Tomlinson, "SNAP Shield Test Experiment Final Hazards Summary," Atomics International Report NAA-SR-5896 (March 1961) (173 pages)

To62* R. L. Tomlinson, R. P. Johnson, and S. G. Wogulis, "SNAP Shield Test Experiment Reactor Physics Tests," Atomics International Report NAA-SR-7368 (July 1962) (35 pages)

Tr69 J. Tribfelner, "S8DR Pre-Irradiation Data File and Corresponding Computer Printout," Atomics International Technical Information Document TI-568-24-035 (November 1969) (46 pages)

Tu58 R. R. Turk, "Feasibility Report for Fabrication of SNAP II Fuel Elements," Atomics International Memo NAA-SR-MEMO-3083, Rev. II (October 1958) (32 pages)

Va70 R. Van Houten, "First Status Report (to 5:00 p.m., April 3, 1970) on the Coatings on S8DR Fuel Elements," North American Rockwell Corporation Aerospace and Systems Group Internal Letter dated 7 April 1970 (3 pages)

We85 R. C. Weast, Ed., *CRC Handbook of Chemistry and Physics*, 66th Edition (CRC Press Inc., Boca Raton, FL, 1985)

Wi58 L. A. Wilson, "Evaluation of Required Fuel Loading in the SER," Atomics International Technical Data Record NAA-SR-MEMO-3406 (December 1958) (12 pages)

Wi59 L. A. Wilson, "SNAP II Experimental Reactor Physics Analysis," Atomics International Report NAA-SR-3607 (June 1959) (71 pages)

APPENDIX 2
REPORT CONTRIBUTORS

D. W. Kneff

Dr. Kneff is a senior member of the Rocketdyne (and former Atomics International) technical staff who has been at Rockwell for 19 years. He has a background in nuclear physics and has conducted many nuclear measurement and radiation dosimetry experiments in reactors and particle accelerators. He was a contributor to the investigation of self-radiation effects in proposed crystalline high-level waste forms for repository emplacement. Dr. Kneff is a company expert on space power system survivability, and current activities include lethality analyses for theater missile defense and technical support to ETEC's decontamination and decommissioning development programs.

W. E. Nagel

Mr. Nagel was hired by Atomics International in July 1961 for the specific purpose of establishing a quality engineering and test function for SNAP fuel fabrication. He was made supervisor of the SNAP Fuel Quality Test Unit in January 1962, which he managed until he was assigned as Quality Engineering Manager for all of AI's fuel fabrication in June 1963. In late 1965 Mr. Nagel was reassigned as a research engineer responsible for developing nondestructive test methods for advanced nuclear fuels and in-core components. He was Fuels Quality Engineering Manager again from mid-1967 until 1970, when the S8DR fuel fabrication was complete and he became manager of quality engineering for reactor systems. Mr. Nagel retired in 1991 and was recalled for the present work.

H. Pearlman

Dr. Pearlman was Atomics International's Group Leader of Chemistry during the initial SNAP development work, and was in charge of the early work on zirconium hydride. He became Manager of Component Development and (later) the AI Fuels and Materials Division, where he was responsible for fabrication of the first SNAP fuel elements. He retained close association with the SNAP fuel element development and performance activities throughout the active phase of the reactor operation program. Dr. Pearlman also managed the later AI radiation effects investigations on proposed crystalline high-level waste forms. He was recalled from retirement to contribute to the present work.

V. J. Schaubert

Mr. Schaubert was part of Atomics International's Nuclear Materials Management Unit during SNAP development, which had primary responsibility for nuclear material control, including procurement, receiving, shipment, internal transfers, product storage, and accounting records. He was Manager of the unit beginning in 1968, which included management of the Radioactive Materials Disposal Facility (RMDF). Mr. Schaubert was responsible for shipping all of the spent SNAP fuel to INEL. He is presently retired, and was recalled for participation in the present work.