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OCTOBER 1965-JANUARY 1966
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SNAP 8 PROGRESS REPORT

OCTOBER 1965-JANUARY 1966

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D. G. MASON

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The previous SNAP 8 progress report was "SNAP 8 Progress Report, August-October 1965," D. G. Mason, NAA-SR-11692 (CRD), issued December 29, 1965.

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I. PROGRAM OBJECTIVES

The objectives of the SNAP 8 reactor program are:

- 1) Examine and analyze the post-operational data from the S8ER,
- 2) Assemble and test at both parametric and design endurance conditions the 600 kwt, 1300°F SNAP 8 developmental reactor to substantiate the capability of the SNAP 8 reactor for power and temperature growth while demonstrating competitive endurance capability,
- 3) Plan for long-term nuclear electric power system testing, and
- 4) Continue the development of technology to provide reactor capability for operation at 1 Mwt at 1300°F and at lower power at 1400°F.

The SNAP 8 reactor program is being performed for the Atomic Energy Commission. The performance objectives of the reference design SNAP 8 reactor are summarized in Table 1.

TABLE 1
SNAP 8 REACTOR PERFORMANCE
OBJECTIVES

Thermal Power	600 kwt
NaK Outlet Temperature	1300°F
Reactor Operating Period	10,000 hours
Reliability	96 to 99%

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II. PROGRESS SUMMARY

The primary effort on the SNAP 8 program was the detailed analysis and evaluation of the S8ER fuel elements. The analytical and fuel process development activities, in particular, were aimed at achieving an understanding of the S8ER fuel performance prior to establishing the final S8DS fuel element design and fabrication processes. S8DS fuel production has been delayed until this understanding is achieved.

A. FUEL DEVELOPMENT

The NAA-115-1 fuel element irradiation experiment was discharged from the reactor at Hanford and the screening examination and part of the detailed examination of this experiment were completed. The amount of fuel swelling measured in the NAA-115-1 experiment agrees well with the prediction and is considerably less than that observed in the S8ER.

A better understanding of the relation between process parameters on the ceramic hydrogen barrier adherence was achieved. Testing was continued on seven developmental S8DS fuel elements and the sublength SNAP 8 fuel elements at Hanford in the NAA-115-2 and NAA-117-1 experiments.

B. COMPONENT DEVELOPMENT

Evaluation tests of S8DS design components were continued to provide the basis for reactor performance and reliability estimates. Two each of the control-drive actuators, position sensors, and limit switches completed over 2300 hours of operation under a combined thermal-vacuum-irradiation environment with no measured loss of performance. Endurance and thermal cycle tests of other components, including drive and scram mechanism assemblies, continued to show successful performance at the S8DS design conditions.

The design of the temperature sensor was modified to facilitate fabrication and to improve reliability. The first units of the modified design are undergoing design evaluation testing.

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C. SYSTEM DEVELOPMENT

1. SNAP 8 Experimental Reactor (S8ER)

Disassembly of the S8ER and cleanup of the test facility were completed. Examination of the piping and components in the primary NaK system revealed no significant deposits, erosion, leaching, or mass transfer.

Analysis of the data from the S8ER fuel element screening examination was continued along with the detailed examination of selected elements. Of the various hypotheses to explain the cladding cracking, the two most plausible are for greater-than-predicted fuel swelling due to (1) phase transformation of the fuel after excessive hydrogen loss or (2) higher-than-calculated fuel temperatures. The evaluation is continuing.

2. SNAP 8 Ground Prototype (S8DS)

An analysis of the S8DS reference design indicated that the reactor lifetime will be controlled by the time it takes the fuel to undergo a phase change rather than by the loss of reactivity due to hydrogen leakage. Preliminary investigations showed that this was also the case on three possible core design modifications to accommodate an increase in cladding thickness from 10 mils to 15 mils. Further comparison of these three modifications to the reference design showed that reactivity penalties associated with them would be acceptable and that variations in fuel temperature would be minor.

The acceptance testing of the S8DS reactor components is nearing completion.

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III. REFERENCE DESIGN

The SNAP 8 nuclear system for instrument-rated applications consists of a shielded compact nuclear reactor moderated by zirconium hydride and cooled by liquid NaK-78. The reactor is controlled by movable beryllium reflectors which, by their position, control the amount of leakage flux. Each major assembly of the instrument-rated SNAP 8 nuclear system is described in the following paragraphs. The design reliability goal for the instrument-rated nuclear system has been tentatively established at >96%.

The SNAP 8 reactor core contains 211 fuel elements made of Zr - 10.5 wt % U alloy hydrided to a hydrogen content of 6×10^{22} hydrogen atoms/cc of $(U-Zr)H_x$. The uranium is 93% enriched. The elements are clad in 10-mil-thick Hastelloy-N tubing, coated internally with a hydrogen permeation barrier. The active length of the fuel element is 16.825 in.

The core vessel is 0.105-in.-thick stainless steel. The core vessel contains an upper grid plate of 316 SS, a lower grid plate of Hastelloy-C, a coolant baffle plate of 316 SS, internal BeO reflectors clad in 316 SS, and the 211 fuel elements previously described.

The beryllium control reflector assembly is made in two halves so that it forms a hollow right-circular cylinder completely surrounding the active core length. The beryllium reflector is a nominal 3 inches thick, with the ability to shim both positively and negatively 1 in. Each reflector drum is driven by its own drive mechanism. Three reflector drums are used as startup drums and are driven by long-term bidirectional control-drum actuators. A startup programmer and controller are provided to achieve proper operational sequence of the drums. A platinum resistance temperature sensor is used to measure the coolant outlet temperature.

A typical neutron shadow shield designed to limit the dose at the payload is in the shape of a truncated cone whose cone angle completely hides the payload from the reactor assembly. The neutron shield is made of lithium hydride (LiH) cast in a stainless-steel container. Between the neutron shield and the reactor assembly is a circular plate of tungsten designed to limit the gamma dose at the payload.

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The safety system includes a reentry actuator, reflector ejection springs, an end-of-life scram actuator, a telemetry scram actuator, a reflector unlocking actuator, and associated circuitry. In addition, void filler blocks and a drum lockout system are furnished as part of the ground handling safety equipment.

The reentry actuator is designed to hold the reflector assembly in place at the top of the core vessel. The actuator is designed and located so that it will be melted by reentry heat. Heating of this actuator allows the reflector ejection springs to remove the reflector assembly from the vicinity of the core vessel.

The reflector unlocking actuator is designed to hold the reflector assembly to the core vessel during launch. Thus, it is not necessary for the reentry actuator assembly to hold the reflector assembly under launch conditions.

The void filler blocks are neutron poison blocks designed to be slipped into the cavities in the stationary beryllium reflector normally occupied by the reflector drums when in their full-in position. The drum lockout system is a key-operated locking system designed to prevent reflector drum motion.

An environmental cover is supplied to protect the nuclear system from the natural environments such as rain, sand, and dust. The cover, using a side ejection concept, is supported from the base ring of the shield. View ports are provided for access to the pinpullers and the key locks.

IV. FUEL ELEMENT DEVELOPMENT

A. INTRODUCTION AND SUMMARY

The objective of the fuel element program is to develop fuel elements for the 600-kwt SNAP 8 reactor that meet the following performance requirements.

- 1) Peak power density of ~ 5.0 kw/ft of fuel rod at peak central temperature and peak fuel burnup of 1550°F and 0.3 met. at. %, respectively, in 10,000 hours.
- 2) Hydrogen loss from the fuel elements of less than 5% of that in the fuel alloy during 10,000 hours of reactor operation. The hydrogen barrier coating and closure seal must withstand thermal and mechanical checkout of the reactor, thermal transients during startup, and a peak cladding operation temperature of 1450°F . The fuel element cladding also must resist corrosion in the NaK environment and maintain dimensional integrity over the life of the core.
- 3) A fuel element with 99% reliability at the 50% confidence level of meeting these performance goals. Redundant fuel elements are provided in the core so that failure of the hydrogen barrier in a few elements will still permit successful operation of the reactor.

Primary efforts were concentrated on the detailed examination of the S8ER core, the examination of the NAA-115-1 irradiation experiment, S8DS process improvement development, and the fuel element irradiation tests at Hanford.

The analysis of the data from the screening examination of the S8ER fuel elements continued along with the detailed examination of selected elements. The results of the analyses and examinations are presented in Section VI-D of this report.

The NAA-115-1 fuel element engineering irradiation specimen was discharged from the reactor at Hanford during this reporting period. The screening examination and part of the detailed examination were completed. The amount of fuel growth and swelling measured from this experiment agrees well with predicted data and is considerably less than that observed in the S8ER.

Additional understanding of the relation between chromizing and sandblasting parameters on the ceramic hydrogen barrier adherence was achieved. The amount of chrome diffusion and chrome thickness may not be as predominant a variable of the adherence of the coating as previously determined. A more important variable appears to be the extent of weak sections in the chromium rich layer which are removed during sandblasting. However, a chrome diffusion zone must exist under this layer. Therefore, effort was concentrated on finer control of the sandblasting operation to obtain more uniform results.

Seven developmental S8DS fuel elements* reported last quarter were subjected to thermal shock and endurance testing. Although the data show a slight increase in permeation rate, the rate of change is significantly less than that observed on S8ER type elements.

The NAA-115-2 and NAA-117-1 tests of sublength SNAP 8 fuel elements continued irradiation at Hanford. Both experiments continued to operate near design conditions for all subcapsules.

B. FUEL ELEMENT DEVELOPMENT

1. Coating Process

Primary effort was directed toward improving the uniformity of adherence of the SCB-1 coating to the chromized Hastelloy-N tubes. Screening experiments were conducted over a range of chromizing and sandblasting parameters. The results of these experiments showed that excellent, uniform, coating adherence was attainable and major emphasis was placed on refinement and optimization of the chromizing and sandblasting processes.

a. Sandblasting Studies

The basic feasibility of utilizing a new internal sandblasting mechanism to prepare chromized tubes for coating was demonstrated in the previous reporting period. It was further shown that this process contributed significantly to the level and uniformity of ceramic coating adherence to the metal substrate. Data from recent experiments show a lack of uniformity of the chrome layer removal under apparently constant sandblasting parameters. A sustained effort

*D. G. Mason, "SNAP 8 Progress Report, August-October 1965," NAA-SR-11692 (CRD), December 29, 1965

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was initiated to achieve a uniform and reproducible abrasive action. The following equipment and process refinements were incorporated (refer to Figure 1 for schematic of sandblasting equipment): (1) The air pressure was controlled, using a regulator, at a constant 60 psi in the high pressure line and at 5 psi in the grit transport line. (2) A driving arrangement operated by an electric motor was added to assure uniform movement of the tube during sandblasting. Although the speed could be adjusted, it was held constant at 2 in./min. (3) A valve governing the flow of grit was replaced by a fixed orifice. (4) The range of mesh size of the grit, tubular alumina, was narrowed from -28 to +60 mesh to -28 to +35 mesh. (5) The grit was re-sieved after each cycle. (6) The grit container was enlarged to a 2800-gm capacity to assure an adequate supply of abrasive material. (7) The amount of grit used per tube was determined and controlled within $\pm 5\%$ of the nominal quantity, currently 2000 gm over a 20-in. tube.

Measurement of sandblasting tubes showed an increase in the TIR and inside diameter which required the addition of a re-sizing operation after sandblasting. This results in a second straightening process which is undesirable from a process standpoint.

Refinements were also investigated in the sandblasting process for chromized cup-plugs. The process consists of blasting the outside surface as it rotates inside a glove box at approximately 1700 rpm. The grit is directed through a nozzle set at a 36° angle to the cup-plug; current parameters are 20 psi air pressure for 15 sec. The process for blasting the inside of the cup-plug is under development. Present experiments are utilizing the tube blasting equipment under varying conditions and evaluating nozzle and deflector modifications to attain adequate, but not excessive, abrasive action. As was the case for tubing, the quality of the coating on cup-plugs is significantly affected by the sandblasting process.

b. Chromizing Studies

Chromizing parameter studies were continued with the objective of achieving uniformly good coating adherence along the length of tubes on a reproducible basis. The 2135°F , 12 min cycle, discussed in the last report, was tentatively rejected due to evidence of excessive alumina contamination in the chrome-rich layer apparently causing variable coating adherence. Data in the previous report

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Figure 1. Schematic of Sandblasting Equipment

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showed the relationship of increasing alumina concentrations to decreasing coating adherence values; however, the data also indicated that excellent adherence can be attained in spite of the presence of some alumina particles. Experimental runs at temperatures from 1800 to 2000°F showed a substantial reduction in the amount of alumina contamination at these lower temperatures. A second incentive for lowering the chromizing temperature was the probability of minimizing grain growth across the thin wall tubing.

With one exception, attempts to correlate the lower chromizing temperatures with coating adherence were inconclusive due to the lack of consistency in the data. It is believed that irregularities in the sandblasting process contributed significantly to variable coating adherence. The exception was a group of tubes chromized in four separate runs at 2000°F for 2 hours. End samples from all 17 tubes showed excellent adherence, retaining more than 50% of the ceramic coating after undergoing 23 flexes. Four tubes from the group were sectioned to determine the uniformity of adherence down the length of the tubes. The data, summarized in Table 2, indicate a significant improvement in achieving the goal of uniform coating adherence. The remaining tubes from this group were processed into fuel elements for developmental testing and are discussed in a later section.

During the evaluation of the 1800 to 2000°F chromizing experiments, two additional areas were investigated: (1) the thickness of the diffusion zone of the chrome layer, and (2) the amount of the chrome rich layer apparently missing. The chromized layer consists of a clearly distinguishable chrome-rich outer layer and a zone of diffusion into the base metal. The reporting of chromized layer thickness data has been confined to the depth of the chrome-rich outer layer only. Detailed metallographic examination revealed that local areas of the chrome rich layer were often missing. It appears that the missing areas of chrome are real and not caused by preparation of the specimens for examination. In view of these findings, procedures were initiated to record the depth of the diffusion zone and estimate the amount of chrome-rich layer missing in addition to the thickness of the chrome-rich layer. Typical data, before and after sandblasting, are shown for the tubes listed in Table 2.

Referring to Table 2, it was noted that coating adherence remained about the same although an estimated 10 to 15% of the chrome-rich layer was frequently

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TABLE 2

COATING ADHERENCE DATA FOR CLADDING TUBES CHROMIZED
AT 2000° F FOR 2 HOURS

Chromizing Run Number	Tube Number	Chromizing Thickness (in. x 10 ⁻³)				Coating Retained After 23 Flexes (%)	Estimated Amount of Chrome-Rich Layer Missing (%)	
		Before Sandblast		After Sandblast			Before Sandblast	After Sandblast
		Cr Rich	Diffusion	Cr Rich	Diffusion			
15	4389	0.36	-	0.36	0.54	80	2	5
	4391	0.32	-	0.30	0.53	65	3	10
	4392	0.37	-	0.28	0.49	80	2	10
	4393	0.30	-	0.29	0.53	97	5	3
	4395	0.39	-	0.34	0.53	65	5	10
	4395-1					70		
	4395-2					55		
	4395-3					40		
	4395-4					60		
	4395-5					60		
	4395-6					50		
	4395-7					50		
	4395-8					70		
16	4396	0.35	0.50	0.23	0.43	60	10	5
	4397	0.42	0.50	0.26	0.49	85	15	10
	4398	0.35	0.54	0.24	0.47	90	5	10
17	4399	0.38	0.44	0.33	0.58	98		
	4400	0.39	0.47	0.34	0.61	80		
	4401	0.37	0.45	0.34	0.66	85		
	4402	0.41	0.43	0.38	0.71	85		
	4403	0.32	0.42	0.32	0.49	90		
	4403-1					60		
	4403-2					60		
18	4403-3					70		
	4404	0.44	0.44	0.19	0.55	99		
	4404-1					95		
	4404-2					80		
	4404-3					70		
	4404-4					75		
	4404-5					85		
	4404-6					65		
	4404-7					70		
	4404-8					75		
	4405	0.37	0.48	-	-	75		
	4405-1					75		
	4405-2					75		
	4405-3					85		
	4405-4					70		
	4405-5					70		
	4405-6					60		
	4406	0.40	0.51	0.30	0.57	75		
	4407	0.41	0.53	0.27	0.54	95		

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missing after sandblasting. The tentative conclusion was that good ceramic coating adherence can be achieved either with the chrome-rich layer or the diffusion chrome zone. To gain further information, a tube was sandblasted five times to remove most of the chrome-rich layer. The tube was coated, sectioned, and tested for adherence which proved to be excellent and uniform.

It appears the major contributions of the sandblasting process are to remove surface contamination and to remove areas of the chrome-rich layer probably weakened by imbedded alumina particles. Adherence test samples with areas of good and poor adherence were subjected to detailed metallographic examination. Remnants of the ceramic coating were observed in areas of good adherence indicating that failure occurred within the glass layer. In areas of poor adherence it was apparent that failure occurred within the chrome-rich layer, not at the ceramic-chrome rich layer interface. Limited observations support the premise that proper sandblasting parameters can remove these potentially defective areas leaving a sound substrate to which the ceramic can adhere. Therefore, major emphasis was devoted to upgrading the sandblasting operation to a uniform reproducible process to make valid correlations of adherence as a function of varying chromizing parameters.

2. Closures

Two groups of normal fuel elements were assembled during the reporting period: (1) elements for welding evaluation, and (2) elements for developmental testing.

a. Weld Development Elements

A group of 28 elements was processed to evaluate a proposed change in the method of preparing coated tubes for element assembly. The present procedure is to machine the inside of the tube wall for 1/8 in. at the open end of the tube to remove the ceramic coating and some of the chrome layer. Skillful precision machining is required to maintain the close tolerances necessary on the inside diameter and wall thickness dimensions, and the operation is made considerably more difficult because of the oval-shaped tubing. The proposed method utilizes a sandblasting mechanism internally to remove the ceramic coating from the open end of the tube, a technique which requires less operator skill, yet maintains uniform wall thickness, regardless of tube ovality.

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TABLE 3
PERMEATION RATES OF DEVELOPMENTAL
TEST ELEMENTS

Element Number	Permeation Rate [cc(STP)/hr]	
	As Measured, 1400° F	Normalized, 1400° F, 2 atm
4389	2.22	1.80
4391	0.52	0.42
4392	0.49	0.38
4393	0.66	0.52
4397	0.48	0.43
4398	0.62	0.57
4399	0.35	0.36
4400	0.29	0.31
4402	0.50	0.57
4406	0.46	0.38
4407	0.39	0.43
Averages*	0.48	0.44

*Excluding number 4389

Figure 2. Effect of Temperature
on Barrier Quality

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Tubes prepared by the present and the proposed procedures were randomly processed through the standard element assembly, welding and blending operations. No apparent differences in behavior were observed during fabrication. All elements were submitted to permeation testing to ensure that the change in process would have no gross effect on permeation rate. As expected, there was no significant difference between the permeation rates of the two groups of elements. The average and range of permeation rates for each group were essentially identical. The closure ends of the elements have been submitted for metallographic examination and a subsequent statistical evaluation with regard to weld quality.

b. Developmental Test Elements

Eleven elements, using coated tubes from the coating process development effort, were assembled and submitted for developmental testing. The tubes were representative of a group which exhibited extremely high, and apparently uniform, coating adherence (reference Table 2). The elements have completed initial permeation testing and the measured and normalized permeation rate data are presented in Table 3.

3. Fuel Element Evaluation Testing

a. S8DS Evaluation

During the previous report period, endurance testing was terminated on 72 developmental fuel elements (60 coated with SCB ceramic and 12 coated with AI-8763D ceramic, all vendor-chromized and with original sandblast). Prior to termination of the tests, these fuel elements had been tested at various temperatures in order to evaluate variations in barrier quality (K) with temperature. During this report period the data have been evaluated. The data, shown in Figure 2, indicate a significant increase in K (decreased quality) with temperature. The data also indicate better performance of the SCB-coated fuel elements compared to those with AI-8763D coating. However, although the data indicate a variation of K with temperature, derivation of an equation which relates the temperature effects is not justified for the following reasons:

- 1) The effect of fuel-ceramic reaction is not included in the present permeation equations. A portion of the increases shown is probably due to this reaction.

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2) The present permeation equation will not allow the use of an equation which represents the average data shown on Figure 2 because of the scatter of the data; i. e., a fuel element on the low end of the scatter band would have a negative K, a theoretically impossible condition.

The seven developmental S8DS fuel elements (AI chromized at 2135°F for 12 min, new sandblast method) are continuing under test at 1400°F. The trend of the permeation data is upward, but these fuel elements are still performing significantly better than any prior SNAP 8 normal fuel elements. These elements are also being subjected to a more severe test program. The thermal shock rates are higher and thermal shock tests are being conducted at 1000-hour intervals during the 10,000-hour endurance test program. Average data for these elements are compared in Figure 3 and shown in Table 4.

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Figure 3. Comparison of Hydrogen Permeation Rates of 11 S8ER Type Elements vs 7 S8DS Type Elements

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TABLE 4
DATA OF SEVEN S8DS DEVELOPMENTAL
FUEL ELEMENTS

Test	Average Hydrogen Permeation at 1400° F [cc(STP)/hr]
Initial	0.33
Post-Vibration	0.37
Post 5 Ramp Heats	0.38
Post 20 Ramp Heats (25 total)	0.41
Endurance (accumulative hours)	
24	0.40
96	0.43
192	0.42
240	0.42
408	0.44
594	0.51
762	0.50
882	0.48
1026	0.48
Post Endurance	0.47
Post 5 Ramp Heat-Thermal Cycle Inputs	0.53
Endurance (accumulative hours)	
1218	0.65
1242	0.66
1386	0.69
1626	0.69
1794	0.75

Hydrogen permeation rates at 1400°F were measured on 27 S8DS fuel elements (AI chromized at 2135°F for 12 min, original sandblast) which were assembled for weld evaluation. Ten of the 27 fuel elements (37%) have permeation rates ≤ 0.5 cc(STP)/hr. The measured hydrogen permeation rates for the 27 fuel elements are shown in Table 5.

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TABLE 5
PERMEATION DATA OF S8DS FUEL ELEMENTS ASSEMBLED
FOR WELD EVALUATION

Element Number	Hydrogen Permeation at 1400° F [cc(STP)/hr]	Element Number	Hydrogen Permeation at 1400° F [cc(STP)/hr]
N-4135	0.40	N-4325	6.9
N-4136	0.41	N-4326	0.50
N-4137	0.38	N-4327	0.50
N-4138	1.2	N-4328	4.8
N-4140	7.4	N-4329	0.36
N-4238	0.63	N-4332	7.4
N-4240	0.47	N-4333	0.62
N-4255	1.4	N-4348	4.9
N-4316	3.4	N-4359	0.37
N-4317	4.8	N-4361	0.32
N-4319	0.79	N-4362	0.41
N-4320	0.63	N-4363	1.0
N-4321	1.1	N-4394	6.7
N-4324	4.3		

Tests were terminated on 10 early S8DS developmental fuel elements (vendor-chromized and original sandblast). The fuel elements completed pre-endurance inputs, 6000 hours of thermal endurance at 1400° F, and post-endurance inputs of 10 cycles of ramp heating and 140 thermal cycles. Final hydrogen permeation rates from endurance and the subsequent post-endurance permeation rates for the 10 fuel elements, shown in Table 6, indicate higher permeation rates after endurance.

b. Special Studies

Hydrogen dissociation pressure measurements have been made on 12 S8DS type fuel rods. Table 7 shows the average values of the effective H/Zr* derived from pressure measurements at three temperatures and comparative values

*Atomic ratio of hydrogen to zirconium

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TABLE 6
 TERMINAL DATA OF TEN EARLY S8DS
 TYPE FUEL ELEMENTS

Fuel Element Number	Hydrogen Permeation at 1400° F [cc(STP)/hr]		
	6000-Hour Endurance	Post Endurance	Post Ramp Heat-Thermal Cycle
2679	0.36	0.68	0.87
2807	0.49	0.64	1.1
2815	0.30	0.65	0.66
2827	0.25	0.49	0.57
2831	0.45	0.83	0.90
2833	0.32	0.63	0.63
2835	0.33	0.56	0.63
2914	0.31	0.74	0.66
2918	0.48	0.81	0.54
2939	0.44	0.65	0.72
Average	0.37	0.67	0.73

derived from weight gain measurements during hydriding. The table also shows the corresponding hydrogen dissociation pressures at 1400°F. The data show that the derivation of hydrogen content by weight gain results in values on the average of 0.4% higher than those based on hydrogen dissociation pressure measurements. This bias can be rationalized by considering:

1) Contamination — some of the weight gain during hydriding may be due to oxidation, nitriding, moisture, etc.

2) Thermal gradient — After SNAP 10A fuel rods were hydrided, a radial hydrogen gradient existed with the outside of the fuel rod (which was machined off before the rod was used) having the highest hydrogen concentration. This extreme condition has not been found with SNAP 8 fuel rods; however, smaller gradients outside the error of measurement could exist which contribute to the apparently higher hydrogen concentration.

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TABLE 7
S8DS NORMAL FUEL ROD DATA COMPARISON

Fuel Rod Number	$(H/Zr)_{eff}^*$	Pressure at 1400° F (atm)	$(H/Zr)_{eff}^\dagger$	Pressure at 1400° F (atm) [§]
389A1	1.767	2.90	1.771	3.19
389A2	1.763	2.80	1.770	3.15
389A4	1.766	2.90	1.777	3.45
389B1	1.766	2.90	1.774	3.31
389B2	1.760	2.70	1.771	3.20
389B3	1.751	2.30	1.771	3.21
389B4	1.751	2.30	1.765	2.85
389C1	1.739	2.00	1.738	1.95
389C2	1.733	1.80	1.736	1.91
389C3	1.729	1.75	1.731	1.80
389C4	1.725	1.65	1.731	1.79
389D1	1.726	1.70	1.733	1.85
Average	1.748	2.31	1.756	2.64

*Calculated from pressure measurement.

†Determined by weight gain adjusted for weight loss during pressure measurements.

§Calculated from H/Zr.

The disagreement in dissociation pressure at 1400° F is significantly more than the disagreement in H/Zr. In this case, the average dissociation pressure differs by 14%.

4. Irradiation Tests

a. Low-Temperature Fuel Materials Irradiation, NAA-116 Series

The objective of the low-temperature fuel materials irradiation program was to provide irradiated fuel samples for use in the correlation of post-irradiation annealing studies with high-temperature inpile performance of the fuel. The purpose of the annealing tests was to study the swelling behavior of the fuel as a function of the metallurgical parameters of the fuel.

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As previously reported, the NAA-116-1 and -2 capsules were irradiated in a Hanford reactor to nominal calculated burnups of about 0.15 and 0.7 met. at. % respectively. Much of the work was completed in GFY 65 and has been reported in the last five quarterly progress reports. In summary, the results indicated a volume change during irradiation which annealed out during subsequent post-irradiation annealing tests. Also, the isotopic uranium analysis was inconsistent so additional samples were re-submitted for analysis.

The re-run of the isotopic uranium analysis for burnup determinations was completed during this report period for the NAA-116-2 experiment. Though extra precautions were taken, anomalous results were again obtained (see Table 8). The calculated burnup values from dosimetry and flux profiles appear to be the best values.

TABLE 8
BURNUP DETERMINATIONS FOR
NAA-116 EXPERIMENT

Irradiation Capsule	Calculated Burnup (met. at. %)		
	Dosimetry	Flux	Isotopic Uranium
NAA-116-1	0.14	0.15	0.188 0.137 0.184
NAA-116-2	0.65	0.65	2.39 0.775 0.122* 0.181*

*Results from re-submitted samples analyzed this quarter

The thermal cycling tests of nonirradiated zirconium hydride continued during this reporting period. Both fueled and unfueled specimens were cycled for a total of 350 cycles. The data, shown in Table 9, are contrary to results reported last quarter in that no change in density has been observed.

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TABLE 9
RESULTS OF SECOND THERMAL CYCLING TESTS*

Sample Number	Uranium (wt %)	Density (gms/cm ³)				
		Precycled	50 Cycles	150 Cycles	250 Cycles	350 Cycles
101-A7A	None	5.64	5.64	5.64	5.64	5.64
101-A7C	None	5.64	5.64	5.64	5.64	5.68
261-D4B	10	6.08	6.08	6.07	6.08	6.08
261-D4C	10	6.08	6.08	6.08	6.08	6.08
261-D4D	10	6.08	6.08	6.08	6.08	6.08

*The thermal cycle was as follows: (1) heat from 50 to 590°C at 17°C/min, (2) hold for 20 minutes at 590°C, (3) quench to 50°C (~100°C/min for encapsulated fuel material), (4) hold for 2 minutes at 50°C, and (5) repeat.

The original samples were solid cylinders from archive specimens from the NAA-116-1 and -2 experiment. The recent specimens have a small thermocouple well machined in the center of the specimens to provide more accurate temperature control during the cycling. The difference in geometry, however, does not appear to account for the difference in results.

b. Engineering Irradiations

(1) NAA-115-1

The NAA-115-1 fuel element engineering irradiation was discharged from the reactor at Hanford. This test consisted of six experiments assembled in a train approximately 12 ft long exclusive of its instrumentation lead tube. Two of the experiments in this train contained four fuel elements each with the remaining experiments containing one fuel element each. The 12 sublength (4-in. fuel rod) fuel elements in this experiment were of the same design as that used for the S8ER fuel elements. The cladding material was from the same heat of material and the fuel composition spanned the range of composition used in the S8ER. The design center temperatures also spanned the range of the center temperatures of the S8ER fuel.

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Following the discharge, the six experiments were separated from their train. The flux depressors or shroud tubes were removed and the six experiments were shipped to the Atomics International Hot Laboratory (AIHL).

At the AIHL the experiment hardware surrounding the fuel elements was removed and the fuel elements were screened. The screening examination of the fuel elements included: visual and photographic coverage, detailed stereomicroscope examination, gamma activity measurements, eddy-current inspection for hydrogen concentrations in the fuel, and measurement of fuel specimen diameters and lengths. These data are presented in Table 10.

The screening examination revealed that the cladding of eleven of the twelve fuel elements was in excellent visual condition.

Fuel element, IS-10, from Experiment 4, had a 2-in. longitudinal crack in the middle of the cladding tube. The cladding was raised adjacent to the crack. A minor hairline crack was located at the base of the raised portion. The local rise in the cladding is attributed to high internal H_2 pressures for the following reasons. Experiment 4 was operated in the helium mode after approximately 6700 hours when thermocouple anomalies were noted. At the time of the thermocouple malfunction, attempts were made to raise the temperature of the fuel to design temperature. With the wide temperature margin provided by the control system, high fuel temperatures of about $2000^\circ F$, and subsequent high internal gas pressures, as well as high NaK pressures external to the fuel element, are believed to have occurred.

The stereomicroscope examination revealed a hairline crack at the weld fillet of the in-fuel thermocouple on the end plug of Fuel Element IS-1A. This minute crack allowed NaK to enter the fuel element. Although this condition existed, no adverse visual effect was noted on the outside of the cladding on this element.

Photographs of the twelve fuel element specimens immediately after disassembly and cleaning are shown in Figure 4. An enlarged view of the Fuel Element IS-10, from Experiment 4 is contrasted with Fuel Element IS-8 from

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TABLE 10
FUEL ELEMENT DESIGN DATA, NAA-115-1 IRRADIATION EXPERIMENT

Experiment and Element IS Number	Design Composition			Design Temperature (°F)		Design Burnup (met. at. %)	Hydrogen Permeation Rate [cc(STP)/hr]				Internal Clearance (in.)**	
	Additive (wt %)	H/Zr Atom Ratio	Uranium (wt %)	Fuel Centerline	Cladding		Pre-Irradiation		Post-Irradiation		Pre-Irradiation	
							1200°F	1400°F	1200°F	1400°F	Axial	Radial
1,1A*	0.1	1.7	10.0	1550	1220	0.4	0.29	3.82	-	-	0.110	0.0030
							0.03	0.33	-	-	0.126	0.0035
1,2	0.1	1.6	10.0	1550	1220	0.4	0.01	0.05	-	-	0.121	0.0031
1,3A	0.02	1.7	10.0	1550	1220	0.4	0.02	0.24	-	-	0.110	0.0034
1,4	0.4	1.7	10.0	1550	1220	0.4	0.02	0.10	0.04	0.52	0.112	0.0035
2,5A	0.02	1.7	10.0	1500	1170	0.4	0.04	0.45	-	-	0.124	0.0009
2,6	0.1	1.7	10.0	1500	1270	0.4	0.05	0.50	-	-	0.120	0.0037
2,7	0.1	1.85	10.0	1500	1170	0.4	0.11	1.18	-	-	0.118	0.0005
2,8	0.1	1.85	10.0	1500	1270	0.5	0.13	1.35	-	-	0.105	0.0038
3,9†	0.1	1.7	10.0	1600	1270	0.4	0.00	0.06	-	-	0.040	0.0039
4,10§	0.1	1.7	10.0	1500	1170	0.4	0.02	0.12	-	-	0.036	0.0041
5,11	0.1	1.7	10.0	1400	1065	0.4	0.01	0.09	-	-	0.028	0.0037
6,12	0.1	1.7	10.0	1300	970	0.4						

*Element contained infuel thermocouples.

†Operated at lower than design temperatures after 6700 hours.

§Operated at lower than design temperatures after 5600 hours.

**Determine by difference between measured values of fuel and cladding.

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Figure 4. NAA-115-1 Fuel Specimens Immediately after Disassembly

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← END FACING LEAD TUBE FUEL SPECIMEN IS-10 END FACING NaK BELLOWS →

FUEL SPECIMEN IS-8

4.3 X REDUCED TO 50%

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Figure 5. NAA-115-1 Fuel Specimens IS-10 and IS-8

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Experiment 2 in Figure 5. Fuel element IS-8 is typical of the condition of the ten elements with intact cladding.

Gamma scanning measurements showed essentially the same activity level in all elements. The activity levels were normalized to Element IS-3A in Experiment 1. This element had an activity level corresponding to the mean of all elements in the 12 feet long train. On this basis there is only a 9% deviation from the normalized mean activity for all elements. This result is in agreement with earlier design predictions.

Diameter and length measurements of the fuel elements revealed minor changes. Two elements, IS-6 and IS-8, are the only elements that show any positive increase in cladding diameter. As shown in Table 10 the radial gap of these two elements was 0.0009 and 0.0005 inch, respectively, before irradiation. Detailed analysis of pre- and post-irradiation dimensions show the diameter change to be 0.001 inch for these elements. Diameter measurements of Element IS-10 from Experiment 4 indicated that the cladding had formed around the fuel except in the area of the crack.

Length changes of the twelve elements ranged from -0.003 to +0.004 inch. The length measurements were taken over the thin disc which formed the vacuum barrier at both ends of the elements. Because this disc deformed slightly during operation, these small changes are not considered to be significant. See Figure 6 for the element design.

The results from the fuel element screening examination are given in Table 11. Included are fission gas release data on elements IS-1A, IS-4, IS-5A, IS-7, IS-9 and IS-12, hydrogen permeation measurement on IS-4, and calculated post-irradiation internal clearance between the fuel rod and cladding on all elements except IS-1A in which the fuel cracked.

Inspection of the basic components and their welds was included in the screening examination. This inspection disclosed that damage existed at the bulkhead on Experiments 3, 4, 5, and 6 at the location of the braze closure with the instrumentation and gas control lines. An enlarged view of a typical bulkhead showing the damaged lines is presented in Figure 7.

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Figure 6. Typical Fuel Element Schematics, NAA-115-1 Irradiation Experiment

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Figure 7. NAA-115-1 Thermocouple Bulkhead from Subcapsule Number 4

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TABLE 11
NAA-115-1 HOT LABORATORY EXAMINATION
a. Screening Results

Experiment and Element IS Number	Pre-Irradiation Chemical Analysis		Post-Irradiation Fuel Specimen Condition	Post-Irradiation Specimen Condition	Fuel Specimen Dimensions (in.)				Dimension Change (in.)		Density (gm/cc)		Change in Density (%)	Hydrogen Pre-Irradiation		Calculated Burnup (met. at. %)
					Pre-Irradiation		Post-Irradiation									
	Uranium (wt %)	Additive (wt %)			Length	Diameter	Length	Diameter	Δ Length	Δ Diameter	Pre-Irradiation	Post-Irradiation		(wt %)	(H/Zr) _{eff}	
1,1A	9.89	0.08	Broke at bottom of in-fuel TC wells	Intact, defective TC weld	4.007	0.562	4.004	0.562	-0.003	§	6.050	5.9532	-1.60	1.64	1.68	0.37
1,2	10.11	0.09	Intact-small cracks at both ends of fuel rod	Intact	3.997	0.562	3.999	0.562	+0.002	§	6.099	5.9705	-2.10	1.55	1.59	0.38
1,3A	10.02	0.04	INTACT	Intact	3.998	0.562	3.999	0.562	+0.001	§	6.061	5.8681	-3.18	1.62	1.66	0.38
1,4	9.98	0.40	INTACT	Intact	3.998	0.562	3.998	0.562	§	§	6.100	5.8604	-3.92	1.61	1.71	0.38
2,5A	10.07	0.04	INTACT	Intact	3.990	0.562	3.990	0.562	§	§	6.061	5.8355	-3.72	1.60	1.64	0.39
2,6	9.99	0.10	Intact-small cracks at end of fuel rod	Intact	3.997	0.562	4.001	0.563	+0.004	0.001	6.071	5.9443	-2.08	1.64	1.69	0.40
2,7	10.05	0.11	INTACT	Intact	3.997	0.563	3.996	0.563	-0.001	§	6.053	5.8748	-2.94	1.78	1.84	0.39
2,8	10.05	0.11	Intact-small cracks at end of fuel rod	Intact	3.997	0.562	3.996	0.563	-0.001	0.001	6.062	5.9474	-1.89	1.77	1.82	0.40
3,9*	9.89	0.08	INTACT	Intact	4.000	0.562	4.000	0.562	§	§	6.068	5.9741	-1.54	1.64	1.68	0.40
4,10†	10.04	0.12	INTACT	Cladding cracked	3.997	0.562	3.997	0.562	§	§	6.073	5.9921	-1.33	1.64	1.69	0.40
5,11	10.04	0.12	INTACT	Intact	3.998	0.562	3.997	0.562	-0.001	§	6.073	5.9601	-1.85	1.62	1.67	0.36
6,12	10.04	0.12	INTACT	Intact	3.999	0.561	3.999	0.561	§	§	6.072	5.9343	-2.26	1.64	1.69	0.35

*Operated at lower than design temperature after 6700 hours.

†Operated at lower than design temperature after 5600 hours.

§No significant change between pre- and post-irradiation measurements.

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TABLE 11
NAA-115-1 HOT LABORATORY EXAMINATION
b. Macroscopic Results

Experiment and Element IS Number	Dimension Change (in.)			$\frac{\Delta L}{L}$	$\frac{\Delta D}{D}$	Fuel Rod Dimensions (in.)						Normalized Gamma Activity	Internal Clearance (in.)††		Fission Gas Release (%)
						Pre-Irradiation			Post-Irradiation				Post-Irradiation		
	Δ Length	Δ Diameter				Length	Diameter		Length	Diameter			Axial	Radial	
		Min.	Max.				Min.	Max.		Min.	Max.				
1,1A	§	0.0033	0.0035	§	0.68	3.386	0.5274	0.5277	§	0.5307	0.5312	0.96	-	0.0012	0.58
1,2	0.0454	0.0021	0.0029	1.34	0.43	3.385	0.5274	0.5276	3.4304	0.5295	0.5305	0.99	0.081	0.0020	-
1,3A	0.0333	0.0044	0.0048	0.90	0.85	3.387	0.5274	0.5275	3.4203	0.5317	0.5327	1.00 (norm)	0.088	0.0007	-
1,4	0.0481	0.0044	0.0059	1.42	1.02	3.386	0.5273	0.5275	3.4342	0.5315	0.5338	0.99	0.062	0.0005	0.017
2,5A	0.0386	0.0049	0.0060	1.13	1.02	3.387	0.5275	0.5277	3.4251	0.5324	0.5337	1.02	0.073	0.0005	-
2,6	0.0412**	0.0020	0.0035	1.21	0.54	3.388	0.5336	0.5336	3.4292**	0.5356	0.5371	1.04	0.083	0	0.19
2,7	0.0071	0.0038	0.0059	0.20	0.83	3.386	0.5273	0.5274	3.3927	0.5304	0.5344	1.01	0.113	0.0008	0.053
2,8	0.0343	-0.0003	-0.0023	1.02	-0.24	3.386	0.5337	0.5338	3.4203	0.5307	0.5337	1.04	0.084	0.0016	-
3,9*	0.009	0.0027	0.0034	0.26	0.59	3.387	0.5273	0.5275	3.396	0.5300	0.5309	1.03	0.096	0.0020	0.014
4,10†	0.009	0.0020	0.0027	0.27	0.43	3.471	0.5275	0.5275	3.480	0.5295	0.5302	1.04	0.031	0.0025	-
5,11	0.0105	0.0017	0.0037	0.30	0.43	3.470	0.5272	0.7274	3.4805	0.5289	0.5311	0.94	0.025	0.0022	-
6,12	0.0080	0.0016	0.0038	0.23	0.49	3.469	0.5275	0.5276	3.4770	0.5291	0.5314	0.91	0.020	0.0018	0.007

*Operated at lower than design temperatures after 6700 hours.

†Operated at lower than design temperatures after 5600 hours.

§Fuel rod broken

**Dimension may not be significant due to local cracking on the ends of the element.

††Determine by difference between measured values of fuel and cladding.

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1 IN EXPERIMENTS 1 AND 2, 23 THERMOCOUPLES OPERATED 100% OF IRRADIATION TIME. TC NO. 17 WAS NOT FUNCTIONING AT START OF IRRADIATION, BUT OPERATED DURING THE LAST 50%.

2 IN-FUEL THERMOCOUPLES DEGRADED AFTER 80-hr OPERATION AT DESIGN TEMPERATURE.

Figure 8. Percentage of Time Thermocouples Operated in NAA-115-1 Experiment

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Inspection of the bulkheads of Experiments 3 and 5 indicated a fracture of Thermocouples 25, 26, 27, 28, 29, 31, 33, and 34, each of which failed during irradiation (Figure 8 for thermocouple reference). Figure 8 indicates the percentage of time the thermocouples operated during the inpile period. The thermocouples are grouped by experiment. Thermocouple 17 in Experiment 2 was inoperable at the start of the irradiation but was fixed during operation.

During and after the assembly of the NAA-115-1 experiment, studies were being conducted on methods of brazing, brazing alloys, and bulkhead configuration simultaneously with thermocouple studies. As a result of these studies, changes were made in the bulkhead design of the NAA-117-1 and NAA-115-2 experiments. The thermocouples in these experiments have operated very well to date which tends to indicate that the changes have solved this problem.

All other components from the NAA-115-1 experiments were considered to be in satisfactory condition.

Following the fuel element screening examination, each of the elements was subjected to a detailed examination. The detailed examination of the fuel specimens included: decladding of the elements; visual and photographic coverage; detailed stereomicroscope examination; measurement of density; and measurement of fuel diameters and lengths. These data were shown in Table 11.

NaK was present inside the cladding of IS-1A and IS-10 as discussed earlier. The presence of NaK in these elements caused a reaction with the barrier material resulting in poor adherence to the cladding and limited ceramic adherence to the fuel specimen. The fuel was measured with the ceramic present and after the ceramic had been removed. The difference in measurements agrees well with the estimated thickness of the ceramic.

The fuel rods from eleven of the twelve elements were intact. The fuel specimen from IS-1A, the fuel element with the infuel thermocouples, was broken at the bottom of the thermocouple well. This was expected because of the notch sensitivity of the fuel material.

The fuel specimens from the remaining eleven specimens appear to be in good condition with the exception of slight end cracking at the blind end of fuel specimens IS-2, IS-6, and IS-8. Analyses of the fuel temperature, temperature gradient, and flux profiles of these fuel rods are in progress.

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Figure 9. Volume Change of NAA-115-1 Fuel Rods Superimposed on Fuel Swelling Relationship for NAA-53 Experiment Data

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The post-irradiation hydrogen permeation rate for Element IS-4 was measured (Table 10). The post-irradiation value is approximately twice the pre-irradiation value. The cladding diameter increased by about 0.0015 inch in the center of the element during this test. Subsequent dimensional measurements of the fuel after decladding showed this change was not caused by the fuel.

The most important results of the detailed examination have been the density measurements which show the swelling in the NAA-115-1 to be lower than predicted. This is contrary to the S8ER where swelling was generally higher than predicted, especially in the cracked elements. The NAA-115-1 data also show that swelling up to a peak centerline temperature of 1600°F is not excessive (Figure 9). As the maximum centerline temperature that can be calculated for S8ER is about 1525°F it does not appear that the higher-than-predicted temperatures in the S8ER can be the primary cause of greater-than-predicted swelling.

The fuel rod diameter and length measurements have been recorded and also appear in Table 11. Preliminary analysis of the density and dimensional data shows a larger amount of swelling in the three specimens having low and high carbon additive, 0.02 and 0.40 wt %. Although a correlation appears to exist between swelling and the amount of additive, the magnitude of the swelling observed in the high-carbon fuel from NAA-115-1 is much less than the swelling of the S8ER fuel which was generally in this same range of carbon content. The carbon content of the S8DS fuel elements is 0.15 wt %, which is in the range of composition which showed the best performance in NAA-115-1. In addition, previous out-of-pile data show the strength of the unirradiated fuel material to be strongly dependent upon the carbon content with a peak strength at about 0.15 wt % carbon at a temperature of 1450°F.

A review of the inpile thermal cycles experienced by the NAA-115-1 irradiation experiment was conducted and compared to the thermal cycles experienced in the S8ER. The total number and severity of the thermal cycles seen by the NAA-115-1 irradiation experiment was about the same as the S8ER received prior to June 1964. This does not include the thermal cycles seen by the S8ER during the power coefficient tests of May 1964. Up until June 1964, the hydrogen permeation rate from the S8ER showed little change. After June of 1964, the hydrogen permeation rate increased significantly. Nonnuclear testing of the

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S8ER fuel elements had shown that the AI-8763D ceramic hydrogen barrier deteriorated significantly when subjected to thermal cycles. The data from the NAA-115-1 and the S8ER indicate that this deterioration may be cycle dependent in that the S8ER permeation rate for the same number and severity of thermal cycles seen by the NAA-115-1 experiment did not change. In addition, since the out-of-pile testing shows an increase in permeation rate after a limited number of cycles, cycle severity must also be considered. In particular, the glass closure end, which is relatively cool (1100°F) during normal operation but quite hot (1400°F) during out-of-pile testing, must be considered before any specific statements can be made regarding cycle dependency and/or severity.

(2) NAA-115-2

The NAA-115-2 irradiation test plan encompasses the same parameters as the NAA-115-1 except that the NAA-115-2 irradiation assembly contains twelve S8DS type sublength fuel elements.

Irradiation continued at Hanford and all operating data indicate that the experiment is functioning as designed. Design temperatures have been achieved during 78% of the available full power reactor operation time.

The 43 chromel-alumel thermocouples located among the 12 fuel specimens on this test continue to record cladding temperatures. The two Geminol thermocouples placed within one of the fuel specimens are also operating satisfactorily.

(3) NAA-117-1

The NAA-117-1 irradiation test is designed to evaluate the effect of increased power density on fuel element performance over a range of temperature and fuel compositional parameters. The system sublength fuel elements will be irradiated to a burnup equivalent to 1-1/2 years of SNAP 8 reference design operation at 1.2 Mwt.

Irradiation at Hanford continued during the reporting period and operating data indicate that the experiment is functioning as designed. Design temperatures have been achieved during 76% of the available full power reactor operation time. The first thermocouple failure was experienced in late December. The alumel leg of Thermocouple 71 located in Capsule 3 opened up during steady-state operation of the experiment. Attempts to rejoin the thermocouple by electrical discharge techniques will be made.

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In support of the NAA-115-1 and S8ER evaluations, data were obtained to determine the temperature response following a reactor scram from full power. Prior to the scram, Capsules 1, 2, 3, 4, and 6 were programmed to operate in the pure helium operating mode. Capsule 5, which contains two fuel specimens, one having an infuel thermocouple, remained at design temperatures. The data acquisition system was programmed so that only the temperature associated with Capsule 5 would be recorded, thereby providing about a six-fold decrease in the time required to complete one scanning and recording cycle. The results of this test are shown in Figure 10, in which the temperatures of the infuel thermocouple and one of the cladding thermocouples are shown as functions of time. Since the infuel thermocouple is known to have suffered degradation because of hydriding, corrections were applied to compensate for this condition. The calculated temperature response of this thermocouple is also shown in Figure 10. During the first minute, the temperature decreases of the cladding thermocouple and the infuel thermocouple, as corrected, were 585 and 940°F, respectively. The maximum rates of change were 21 and 37°F/sec, respectively.

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Figure 10. NAA-117-1 (Experiment 5)
Temperature Response Following
Reactor Scram

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V. COMPONENT DEVELOPMENT

The component development effort is concentrated on the evaluation testing of S8DS design components. Tests include performance mapping at the operational temperature and vacuum environment and evaluating the effects of multiple thermal cycles at the maximum operational rate and combined operational thermal-vacuum-irradiation environment.

Two each of the control-drive actuators, position sensors, and limit switches completed over 2300 hours of operation under combined thermal-vacuum irradiation environment with no measured loss of performance. The hot-cell examination of these components was initiated. Two control drum actuators have completed over 7200 hours of thermal-vacuum endurance. Two other actuators have completed 150 and 100 startup and thermal cycles. The first developmental control-drum-drive and scram-mechanism test assembly, which includes the drive components such as the gear, actuator, and bearings, has completed 6000 hours of endurance testing. During this testing, 650 drive-in and scram cycles and 22 thermal cycles were completed. The second unit has accumulated ~3000 hours of endurance. A delivery delay has limited evaluation testing of the control-drum bearing to one endurance test which has continued satisfactorily for 2300 hours. The additional prototype bearings of similar dimensions and the same material are performing without difficulty after 6000 hours. The design of the temperature sensor was modified to facilitate fabrication and to improve reliability. The first units of the modified design are undergoing the first design evaluation testing.

A. SCRAM MECHANISM

Safe operation of the SNAP 8 reactor system during ground testing requires means for shutting the reaction down quickly by withdrawal of the control drums. A control-drum scram drive mechanism has been designed which is capable of operating under the S8DS environmental requirements. Details of the design and operation of the drive mechanism were reported previously.*

*C. E. Johnson, "SNAP 8 Quarterly Progress Report, May through July 1964," NAA-SR-9992, September 22, 1964

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1. Scram Drive Assemblies

Two scram drive assemblies of this design have been under continuous test in the S8DS environment of temperature and vacuum.

The test environmental pressure is 10^{-5} torr, and the maximum temperature of the drive system is 1000°F with temperature distributions simulating those calculated for the S8DS reactor system.

The two scram drives are operated to drive their drum mass inward, scram the drums, and rewind the scram spring. Six such cycles are performed daily by each scram drive. A 16-hour dwell period at vacuum and operating temperature takes place during the night. Thermal cycling to room temperature is performed at the end of the week.

The total accumulated operation for the two scram drive units is as follows:

Scram Drive Unit No. 1

Hours of Operation	6086 hours
Scram/Rewind Cycles	650
Thermal Cycles	22

Scram Drive Unit No. 2

Hours of Operation	2902 hours
Scram/Rewind Cycles	420
Thermal Cycles	21

After 5000 hours of operation, the drum mass moment of inertia of scram drive Unit No. 1 was increased from 0.66 to 0.72 in.-lb-sec². The revised moment and mass correspond to the latest S8DS drum configuration with a 3-in. shim. The longest scram time recorded for 31° rotation from "Full In" with the increased drum weight is 290 msec (measured from the time the scram clutch is deenergized). The longest scram time recorded for 109° rotation (Full Out) for this unit is 533 msec. The maximum allowable scram times are 500 msec for 31° and 2 sec for 109°.

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2. Scram Spring Tests

a. Torque Check

During the assembly and checkout of the S8DS scram mechanism, it was reported that the measured friction torque between the individual scram drive units varied from 1.0 to 7.5 in.-lb. The maximum expected friction torque due to bearing friction is about 3.5 in.-lb. Experience with the SNAP 8 "proof of principle" scram drive had shown that considerable friction torque can be generated in the torsional scram spring due to contact between adjacent spring coils.

Torsional springs of this type have a tendency to "bunch" eccentrically toward one side of the housing. This eccentric motion of the coils is caused by the tangential pull of the outer spring coil toward the outer-spring attaching point at the spring housing. In order to minimize this condition, the S8DS scram spring utilizes a cantilevered spring attaching mode with a centering leaf.

Twelve S8DS springs were tested in a special fixture to determine the amount of eccentricity of the coils and the resulting friction torque. Most of these springs made very light contact at one point of the outer coil. The amount of eccentricity, however, was very small and the maximum friction was less than 0.125 in.-lb. The possible failure mode would be self-welding of the outer coil at the point of contact. This would reduce the effective spring length and failure would increase the spring rate by ~8%. The resulting increase in scram time would not exceed the maximum allowable scram time for the S8DS, however, this is unlikely because the scram spring surfaces are coated with a dry-film lubricant. Figures 11 and 12 show springs in the test fixture with 0° and 200° windup, respectively. The eccentricity of the spring is not enough to cause rubbing contact between the coils.

Subsequent reevaluation of the spring measurement technique indicated that up to 4.0 in.-lb of recorded friction torque could have been due to improper use of the spring torque measuring device. It was concluded that the high values reported were measurement error because scram times indicate that the bearing friction torque did not differ grossly between scram drive units.

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Figure 11. Spring in Test Fixture With 0° Windup

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Figure 12. Spring in Test Fixture With 200° Windup

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Figure 13. Static and Dynamic Torque vs Field Current on 28 Assembled Actuators

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b. Load Performance

A scram spring load performance test was started. The first test unit has shown no torque loss after 450 hours at 800°F. This temperature represents the maximum expected spring temperature for the S8DS test. No load loss has been detected so far during this test.

A helical spring of the same Rene' 41 material had performed satisfactorily at 1050°F in air.* In terms of yield strength, the tensile stress of the scram spring is much lower than the shear stress of the tested helical spring.

Therefore, the initial scram spring test results coupled with the low design stress provide confidence that scram torque loss will be minimal.

B. CONTROL DRUM ACTUATORS

The SNAP 8 reactor is controlled by rotating external beryllium control drums to increase or decrease the fraction of leakage neutrons. The control drum drive is mechanically connected to an electromagnetic actuator which rotates and positions the reflector drums in 0.26° steps through a gear train upon command from a controller. The actuator incorporates a built-in mechanical brake which locks the control drum in place during the nonoperational period. The release of the brake and actuator operation are sequenced to prevent free wheeling of the control drum.

The S8DS actuator is a direct-current, variable-reluctance, stepping motor. The motor is energized with d-c voltages in discrete pulses which cause the rotor to turn in steps of 1.8°. The output rotation can be in either direction as required. The actuator is energized by the controller to produce two actuator steps of 1.8° or a total of 3.6° rotation for each cycle. The geared drive reduces the 3.6° actuator output to 0.26° control drum rotation.

1. Bench Test and Assembly

A total of 15 developmental and 13 S8DS actuators have been successfully assembled and bench tested. The output torque test data in Figure 13 indicate that the performance of every unit which has been assembled is very similar. For comparison, the calculated points as well as the data of a developmental

*C. E. Johnson, "SNAP 8 Progress Report, February-April 1965,"
NAA-SR-11092 (CRD), June 15, 1965

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unit which was tested beyond normal operating limits are shown. Figure 13 may be considered as reference performance of the S8DS control-drum actuators.

2. High Temperature and Vacuum Tests

S8DS test actuators undergoing design evaluation tests under S8DS design temperature and vacuum environment are listed in Table 12. Three units have successfully passed the 5000-hour milestone with no significant change in performance.

After 7000 hours, the S/N-001 actuator was partially disassembled and examined for evidence of physical wear or damage. There were no observable changes or degradation in any of the actuator components. The molybdenum disulfide film on the brake surfaces appeared to be in excellent condition, showing only minute areas of wear.

A slight increase in end play was measured (0.008 vs the initial 0.006). This increase in end play is due to the flattening of spacer shims which were slightly

TABLE 12

S8DS DEVELOPMENTAL CONTROL DRUM ACTUATOR PERFORMANCE

S8DS Developmental Control Drum Actuator				Brake Operation Current to "Pull In" (amp)		Dynamic Load Operation		
Serial Number	Application or Test	Thermal Cycle	Time at 1025°F (hr)	Initial at 1025°F	Last Reading at 1025°F	Load Stepped (in. -oz)	Minimum Current to Step Load (amp)	
							Initial at 1025°F	Last Reading at 1025°F
001	Endurance	7	7226*	0.55	0.55	40	1.15	0.95
002	Endurance	6	7275*	0.6	0.58	40	1.18	0.95
003	Irradiation	6	3026 (2313 hr inpile)	0.49†	0.48†	20	1.03	Not Available
004	Irradiation	6	3026 (2313 hr inpile)	0.60†	0.62†	20	1.08	Not Available
006	Endurance Evaluation "Oil Contamination"	5	5725*	0.5	0.48	120	1.7	1.4
010	Startup and thermal cycle	156	724*	0.55	0.30	84	1.3	1.25
011	Startup and thermal cycle	104	900*	0.55	0.54	40	0.95	0.95
015	BMI out-of-pile UHV correlation	7	2586*	0.5	0.48	40	0.95	0.9

Design requirement: Torque - 40 in. -oz minimum
Input at 1025°F with 28 v
Phase current: 1.6 amp maximum for 40 in. -oz torque
Brake current: 0.9 amp maximum

*Test continuing

†Data at start and finish of irradiation (1×10^{19} nvt)

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bowed when installed and does not affect performance. Figures 14 and 15 show this actuator partially disassembled. The actuator has been reassembled and the test which is scheduled for 10,000 hours has been resumed.

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Figure 14. Actuator S/N-001 After 7000-Hour Test
(Detail Showing Rotor Toothed Disk)

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Figure 15. Interior View of Actuator S/N-001 After
7000 Hours at Temperature and Vacuum

a. Performance (Dynamic Stepping)

Each actuator is performance checked at test conditions by energizing and stepping the actuator through a minimum of 360° under load. There have been no significant changes in the performance of any of the units in test to date under normal operating conditions.

b. Nonoperation or Dwell

The effect of long term dwell is investigated as a means of determining whether or not an actuator subjected to extended periods of inactivity at the high

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temperature and vacuum of the S8DS environment could be reactivated and utilized for active control purposes. The results of three actuators subjected to this test were described previously.* Tests of the actuator which had been updated to the S8DS system design after it had exhibited tendencies for brake tooth adhesion were continued for a total of 2175 dwell hours. The actuator operational performance data after this dwell period duplicated that taken prior to the start of the dwell. There was no tendency or indication of sticking or adhesion in the brake operation. This was in contrast to the first post-dwell performance check at which time the brake did not release with the initial application of 28 volts to the brake, as reported previously.* The results of this second test demonstrated that proper fabrication of the brake teeth and the use of the S8DS design anti-adhesion coating should prevent brake tooth adhesion under S8DS design conditions.

c. Coil and Insulation Resistance

The coil resistance of all test actuators continues to remain constant. This indicates that the coil conductors are relatively stable under these test conditions.

Except for changes caused by the removal and reinstallation of the actuators in their test chambers, the coil insulation resistance remains fairly constant and has apparently reached equilibrium as shown in Figure 16. Actuator S/N-015 which is being tested in ultrahigh vacuum (an ion-pumped system), has not exhibited the steep insulation resistance degradation characteristics of the actuators being tested in a diffusion pumped vacuum. An interesting change in resistance level is shown in the data for actuator S/N-006 (which had been contaminated with oil vapor prior to the start of the test). The D-phase resistance-to-ground increased from 1 to 10 megohms. This vividly demonstrates that the arrangement of the leads and specifically physical distance-to-ground, can appreciably alter insulation resistance. The coil leads of actuator S/N-006 do not terminate in a connector but are brought out as "pig-tails." This configuration makes the lead routing more subject to shifting during handling.

3. Consecutive Startup and Thermal Cycle Test in Vacuum

In reactor operation, the startup cycle is the most severe condition under which the actuator must perform. The actuator must step at a rate of 1 cycle

*D. G. Mason, "SNAP 8 Progress Report, August-October 1965,"
NAA-SR-11692 (CRD), December 29, 1965

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Figure 16. Insulation Resistance Data for S8DS Actuators

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(2 steps of 1.8° each)/7.5 sec for 1863° of shaft rotation to move the control drum into the operational position. The extremely high current density in the wire as well as the rapid internal temperature increase is conducive to the formation of internal hot spots and wire burnouts. After the startup stepping cycle, the actuator windings are further subjected to both internally and externally induced temperature increases due to nuclear irradiation and heat transfer from the reactor.

The startup and thermal cycle test is designed to simulate the actuator thermal history during reactor startup. Beginning at room temperature (78 to 85°F) the actuator is stepped at a rate of 1 cycle (2 steps)/3.3 sec for 12 minutes (786° of actuator rotation). The temperature of the stator and brake coils at the end of the 12 minutes, as shown in Figure 17, is equivalent to that previously measured and reported* after actuator operation for simulated reactor startup (1 cycle/7.5 sec). The coil temperature rise rate of this test is somewhat accelerated. At the end of actuator operation the actuator is subjected to external heat which increases the housing temperature to 1025°F at a rate of $10^\circ\text{F}/\text{min}$. Two actuators have completed 100 and 150 startup and thermal cycles without any significant changes in performance. Additional evaluations have been made to determine the temperature rise of coils above room temperature as a function of stepping rate and applied power both in vacuum and air environment. The data shown in Figures 17, 18, and 19 indicate that the actuator can be operated in room temperature environment over a prolonged period in vacuum at a stepping rate as fast as 1 cycle/3.3 sec or continuously in air without the coils exceeding the S8DS design temperature.

4. Dynamic Performance

Series of tests were conducted on an actuator using a variable input controller to evaluate dynamic performance under various conditions of overlap, pulse width, and stepping rates. The results of these tests are shown in Figures 20, 21, and 22. A review of the data indicates the following:

- 1) The phase overlap may be set as low as $2\text{-}1/2$ msec with no appreciable loss in dynamic or stepping torque. Without overlap, the output torque will fall off as much as 50%.

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2) A pulse duration of approximately 300 msec is required to stabilize the output torque. A minimum pulse duration of 20 to 30 msec is required to produce any appreciable torque; however, with a phase overlap of 5 msec, a pulse duration as short as 58 msec will produce the design output torque. The S8DS system controller provides pulse lengths of 400 msec and a phase overlap of 4-1/2 msec.

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Figure 17. S8DS Control Drum Actuator Temperature Rise During Startup, 1 Cycle/3.3 Seconds

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Figure 18. S8DS Control Drum Actuator Temperature Rise During Startup, 1 Cycle/6.5 Seconds

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Figure 19. S8DS Control Drum Actuator Temperature Rise Above Ambient in Air vs Power Input

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Figure 20. S8DS Design Control Drum Actuator
Variation of Dynamic Torque With
Phase Overlap

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Figure 21. S8DS Design Control Drum Actuator Variation of Dynamic Torque With Phase Pulse Duration for Various Current Inputs

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Figure 22. S8DS Design Control Drum Actuator Variation of Dynamic Torque With Phase Current Durations

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C. BEARINGS

The SNAP 8 control-drum bearings are being tested at the S8DS operational environment of 10^{-5} torr and also at ultrahigh vacuum. The sets under test are pairs of self-aligning spherical bearings with a journal rotating through the center of the monoballs. The configuration and the material combinations were described in the previous progress report. The summary of data on bearings currently on test is shown on Table 13. Each bearing set is loaded to simulate the force of an S8DS beryllium control drum under 1 g (vertical) acceleration. The thrust load is carried by one of the bearings while the radial load is absorbed on both bearings.

TABLE 13
SNAP 8 REFLECTOR DRUM BEARING TEST DATA

Bearing Type	Pressure (torr)	Temperature (°F)	Time at 1150°F (hr)	Actuation Cycles	Range of Torque at 1150°F† (in. -lb)
S8DRM-1*	10^{-5}	1150	5956	960	0.8 - 2.25
S8DS (No Dry Film Lubricant)	10^{-5}	1150	695	410	0.9 - 2.4
S8DS*	10^{-5}	1150	2330	1105	0.4 - 2.4
S8DRM-1*	4×10^{-9}	1150	6278	911	0.2 - 1.8

*The material combinations of S8DS and S8DRM-1 design are Al_2O_3 vs graphite at primary radial and thrust surfaces and Al_2O_3 vs Al_2O_3 + sodium-silicate-bonded molydisulfide - graphite dry-film lubricant.

†The design torque is 3.5 in. -lb max.

1. S8DRM-1 Bearing Test at 10^{-5} Torr, 1150°F

Performance record tests of varying thrust and varying temperature were performed at the 5000-hour milestone of the S8DRM-1 bearing test. All performances were satisfactory. Figures 23 and 24 show the variation of the torque with thrust load and also with temperature. The bearings shown in Figure 25 exhibit the transfer of a thin film of carbon graphite from the bushings to the Al_2O_3 on the shafts and also the transfer of dry-film lubricant to the socket surface. The relatively low friction is attributed to the thin transfer films. The area of the bearings enclosed by the socket configuration shows little loss of

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Figure 23. S8DRM Bearing Torque vs Thrust Load After 5000 Hours at 1150°F and 10^{-5} Torr

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Figure 24. S8DRM Bearing Torque vs Temperature After 5000 Hours at 1150°F and 10^{-5} Torr

dry-film lubricant. The areas of the balls not shielded by the sockets have lost most of their dry film. Evidence from previous tests explains this loss as a combination of sublimation and oxidation of the dry film. The pattern of the dry-film loss is better seen in Figure 26. The whitish powder which formed on the carbon graphite exposed to the environment is a product of oxidation.

The test was resumed and continued to 5548 hours when a vacuum system failure exposed the bearings to ambient air pressure at 950°F for 25 hours. The performance of the bearings did not change significantly as a result of this air exposure. The dry-film lubricant was nearly completely oxidized, as noted in Figure 27, but did not leave a harmful residue. The exposed graphite also oxidized, but did not cause a functional failure. The graphite oxidized in the journal throat area as a function of the distance from the annulus end, as can be determined from the bore dimensions indicated on Figure 28. The test was resumed and the bearing has continued to perform satisfactorily. These results indicate that limited exposure to oxidizing environment can be tolerated.

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Figure 25. S8DRM-1 Bearing Set With Dry-Film Lubricant X-15
After 5000 Hours at 1150°F and 10⁻⁵ Torr

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BALL AREA EXPOSED FROM
SOCKET DURING TEST SHOWS
DRY FILM LOSS

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Figure 26. Thrust Bearing Monoball After 5000 Hours
at 10^{-5} Torr and 1150°F

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Figure 27. S8DRM-1 Bearing After 5523 Hours at 1150°F
and 10^{-5} Torr Plus 25 Hours at 950°F in Air

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Figure 28. Radial Bearing Bore
Oxidation Pattern After 25 Hours
at 950°F

2. S8DS Bearing Test (Without Dry-Film Lubricant) at 10^{-5} Torr and 1150°F

This test is being run to determine bearing performance without dry-film lubricant between the Al_2O_3 surfaces of the balls and sockets. The purpose of this test is to ascertain the need for the lubricant or the consequences of inadvertent loss of dry film during the S8DS reactor tests. Test results thus far demonstrate that the bearings will perform satisfactorily within the allowable torque of 3.5 in.-lb in 10^{-5} torr at 1150°F.

3. S8DS Bearing Test at 10^{-5} Torr and 1150°F

Over 2000 hours of operation on the S8DS reference design bearings were completed with satisfactory performance. These bearings include the sodium-silicate-bonded MoS_2 with graphite dry-film lubricant between the balls and sockets. The variation of torque with thrust load and temperature after 2000 hours of test are illustrated on Figures 29 and 30.

Figure 29. S8DS Bearing Torque vs Thrust Load After 2000 Hours at 1150°F and 10^{-5} Torr

Figure 30. S8DS Bearing Torque vs Temperature After 2000 Hours at 1150°F and 10^{-5} Torr

4. S8DRM-1 Bearing Test at Ultrahigh Vacuum and 1150°F

The S8DRM-1 bearings have performed satisfactorily for 6278 hours in an ultrahigh-vacuum ion pump system at 4×10^{-9} torr and 1150°F. The 5000-hour performance record test data of torque vs thrust load and temperature are shown in Figures 31 and 32. These data are very comparable to those of similar bearings after 5000 hours in 10^{-5} torr vacuum. Thus it appears that the SNAP 8 design bearings perform equally well at 1150°F in either the S8DS environment of 10^{-5} torr or ultrahigh clean vacuum.

Figure 31. S8DRM Bearing Torque vs Thrust Load After 5000 Hours at 1150°F and 4×10^{-9} Torr

Figure 32. S8DRM Bearing Torque vs Temperature After 5000 Hours at 1150°F and 4×10^{-9} Torr

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Figure 33. S/N-004 Sensor Output vs Time (continuous operation at 800°F)

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Figure 34. S/N-004 Sensor Coil Resistance vs Time (continuous operation at 800°F)

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D. DRUM-POSITION SENSING SYSTEM

The drum-position sensing system measures the reactor control-drum angular position and provides an output signal for diagnostic purposes. The desired system accuracy is $\pm 1\%$ of full scale. Since the sensor is mounted on the reflector assembly, it must be capable of operating in the reactor environment throughout the 10,000-hour life of the nuclear system.

1. Transmitter

A rotary variable-reluctance (differential transformer) type of sensor was developed for this application. The stator contains one primary and two secondary windings. The magnetic rotor is shaped so that the coupling to one secondary winding, and hence its output voltage, is at a maximum when the output voltage from the other secondary winding is at a minimum. The demodulator is designed to add output voltages in series-bucking, changing polarity at the center of the range of rotation.

Eight units are in test or have completed testing. S/N 006 and S/N 008 have completed the combined irradiation-vacuum-high-temperature environment test described in Section V-G of this report. S/N 001 and S/N 002 are being tested as part of the scram drive assemblies reported in Section V-A.

S/N 003 and S/N 004, undergoing a 10,000-hour thermal (800°F) vacuum (10^{-6} torr) endurance test, have accumulated approximately 1300 hours. No degradation in performance has been observed. Repeatability of output for a given shaft angle setting is within 2% of output span. This typical characteristic is shown in Figure 33. The coil resistance has varied approximately 0.3%, and appears to be leveling off as shown in Figure 34. Minimum insulation resistance of both units remains above 150 megohms.

S/N 005 and S/N 007 have completed 17 of the scheduled 200 thermal cycles between 100 and 800° in vacuum. Outputs as functions of shaft position, coil resistance, and insulation resistance are taken at both the low and high temperature ends of the cycle. No significant change in performance has been observed on either unit.

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2. Position Sensor Demodulator

The position sensor demodulator provides excitation for the position sensor primary winding and converts the output signals from the sensor secondary windings into a d-c output signal proportional to drum position.

Developmental tests were conducted on four demodulators. Long-term performance tests were initiated on two units. Two other units are being subjected to various inputs, thermal cycles, and operational temperatures, to demonstrate performance within the design requirements.

E. LIMIT SWITCH

Limit switches are required on the S8DS reactor to signal the controller that the control drum has reached various predetermined positions during startup. They are also used on the scram kit to indicate snubber zone and scram spring rewind. The S8DS limit switch is a shorting bar design and was described previously.*

The operational tests of two S8DS design switches in an irradiation-vacuum-high-temperature environment were completed as described in Section V-G. A 10,000-hour developmental test is being conducted at 1000°F and 10^{-7} torr on seven other switches. Four of these switches are cam-actuated once every hour and thermal cycled twice per week. These switches have been at temperature for 2300 hours with a total of 70 thermal cycles. Performance is satisfactory, with contact resistance below 0.2 ohm. The other three switches are being tested in the closed position. These switches have operated at temperature for 770 hours. The contact resistance has remained unchanged from the initial value of 1.5 ohm.

F. TEMPERATURE DETECTOR AND SWITCH

The temperature detector senses the reactor outlet primary coolant temperature. The detector provides an input signal, which is proportional to the reactor coolant outlet temperature to the temperature switch. The switch produces a low temperature signal whenever the coolant temperature is below the lower limit of the control dead-band and a high temperature signal whenever the temperature

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

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exceeds the upper limit. These switch output signals are utilized as input commands to the reactor controller to initiate reactor control-drum actuation as required to maintain reactor coolant outlet temperature within the deadband limits.

1. Temperature Detector

The reference design temperature detector (Figure 35) is an immersion type resistance temperature detector (RTD). The detector design utilizes platinum as the resistance element. The element is supported by alumina insulating tubes contained within a cast gold cylindrical housing. The resistance element is connected to platinum lead wires supported by swaged MgO insulation. The lead wires are terminated at a hermetically sealed high-temperature connector.

The platinum resistance element assembly and the lead wire assembly are completely enclosed in a platinum sheath which provides a protective shield between the resistance element and sources of potential contamination. An outer sheath of 316 stainless steel provides rugged mechanical support for the element and inner sheath, and permits direct immersion of the detector in the primary NaK coolant. The outer sheath of the detector is provided with a weld flange for attachment to the primary coolant pipe wall. The final configuration of the detector is in the form of an "L" with a 1-in. -radius bend shortly beyond the weld flange so that the detector penetrates the primary loop normal to the pipe wall with the major portion of the detector oriented parallel to the pipe wall. The detector diameter is 1/4 in. and the detector is immersed ~5-1/2 inches into the coolant.

The overall sensor design was reviewed and several significant improvements were made recently. The original gold tube of the swaged MgO-Pt lead-wire tube assembly was replaced with a platinum lined stainless-steel tube. The greater mechanical strength of the stainless-steel tubing provides better retention of the swaged MgO insulation around the platinum lead wire. The platinum liner enables the platinum sensing elements and the lead wires to be sealed hermetically to minimize contamination. The joint between the sensor element sheath and the lead wire tube assembly accommodates the differential expansion between the platinum and the stainless steel. The attachment of the lead wire

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Figure 35. S8DS Temperature Sensor

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to the sensing element was simplified and improved to eliminate the failure experienced within the first 1500 hours of testing on the first three prototypes of the original design.

The first five S8DS temperature sensors of this improved design were fabricated and are now undergoing design verification testing. The performance after ~500 hours of stability testing has been satisfactory.

To evaluate existing RTD designs, drift testing at 1300°F of alternate designs of the RTD was continued. The detectors are soaked at 1300°F in air to simulate long-term operating conditions. They are checked for drift every 500 hours by calibration on a triple-point-of-water cell (0.01°C) utilizing precision laboratory resistance measurement techniques. Drift in resistance at the calibration point is converted into an equivalent setpoint drift at operating temperature. Four of the original 25 detectors have exhibited an equivalent setpoint temperature shift of $< +2^{\circ}\text{F}$, -4°F over a 9500 hour post-stabilization operating period, as shown in Figure 36. All data is referenced to zero after a 1000-hour stabilizing period. The performance of the four detectors indicates that, when properly designed and constructed, platinum resistance temperature detectors can

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Figure 36. Resistance Temperature Detector Drift Characteristics at 1300°F in Air

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be utilized at a temperature of 1300°F for periods of approximately 10,000 hours without excessive resistance drift.

2. Temperature Switch

The temperature switch receives the output of the resistance temperature detector and provides signals to the reactor controller whenever the reactor outlet coolant temperature is outside of the deadband limits established by the temperature switch.

The original design S8DS switches* utilized the RTD as an element of a single resistance bridge. The output from this bridge was amplified by a linear magnetic amplifier to a higher voltage level. The linear magnetic amplifier output then supplied control current to two bistable magnetic amplifiers. One of these amplifiers was used to drive the high output relay and the other to drive the low output relay. The bridge was nulled at the midrange of the deadband so that as the upper deadband limit was exceeded, the amplifier bridge output polarity operated to turn the high output bistable magnetic amplifier to the ON position while maintaining the low output in the OFF position. Below the null point, the operation was reversed with the reverse polarity amplified bridge output activating the low output and keeping the high output turned off. Setpoint adjustment was accomplished by means of control of bias winding current. The performance of this design was marginal as reported in the previous progress report.

A revised design which utilizes the resistance temperature detector in a dual bridge was developed to produce more stable output. Setpoint adjustment is accomplished by changes in bridge parameters. The output of each bridge is amplified by high gain linear magnetic amplifiers. Bistable magnetic amplifiers controlled by the output of the linear amplifiers are used to drive the output relays. Switching is initiated in the revised circuit at the null points of the resistance bridges. The schematic of the revised design is shown on Figure 37. A prototype was fabricated and subjected to evaluation tests. The performance of the revised design switch was excellent and within the design requirements.

*C. E. Johnson, "Progress Report, SNAP 8, August-October 1964,"
NAA-SR-10492 (SRD), December 23, 1964

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Figure 37. S8DS Temperature Switch Block Diagram

G. IRRADIATION TEST

High flux irradiation testing of two S8DS actuator-position sensor assemblies and two shorting bar limit switches was conducted. Details of the design and operation of the test assembly were reported in a previous progress report.*

The test assembly was operated for 2313 hours in the combined radiation, vacuum, and high-temperature environment. Component operating temperatures and capsule vacuum levels were maintained between reactor operating cycles thereby accumulating 3026 total hours at temperature. The vacuum level in the ion-pumped test chamber was relatively constant throughout the test at about 1×10^{-6} mm Hg.

The total radiation doses to the actuators and position sensors ranged from 9.1×10^{18} to 1.16×10^{19} nvt ($E > 0.1$ Mev) and 2.8×10^{10} to 3.9×10^{10} rads, gamma. Exact radiation exposures will be determined after hot-cell removal and an analysis of the dosimeters attached to the test components has been completed.

*D. G. Mason, "SNAP 8 Progress Report, August-October 1965," NAA-SR-11692 (CRD), December 29, 1965

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The operating temperatures during the radiation test were as follows:

Actuators	1000°F
Brakes	1000°F
Upper Position Sensor	800°F
Lower Position Sensor	860°F
Limit Switches	900°F
Upper Magnetic Pickup	800°F
Lower Magnetic Pickup	860°F

The insulation resistances (in ohms) are shown in Table 14 at the operating temperatures.

TABLE 14
INSULATION RESISTANCE MEASUREMENTS

	Insulation Resistance (ohms)		
	Initial Pre-Irradiation 7/31/65	Initial Irradiation 8/19/65	End of Irradiation 12/6/65
<u>Upper Assembly</u>			
Actuator	2.5×10^4	2.0×10^3	1.1×10^3
Brake	2.4×10^6	1.5×10^5	3.9×10^5
Position sensor	5.4×10^7	7.7×10^7	2.2×10^7
Magnetic pickup	1.4×10^6	1.0×10^5	4.0×10^4
Dummy cable	3.0×10^7	5.0×10^8	8.0×10^8
<u>Lower Assembly</u>			
Actuator	2.0×10^3	700	240
Brake	1.0×10^7	5.0×10^6	7.8×10^5
Position sensor	7.2×10^6	3.5×10^5	2.9×10^4
Magnetic pickup	1.0×10^5	2.2×10^5	1.5×10^5
Dummy cable	2.0×10^7	1.0×10^8	1.2×10^8
Limit switch No. 1	1.1×10^7	2.9×10^7	2.3×10^7
Limit switch No. 2	1.1×10^7	1.9×10^7	2.3×10^7

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All components operated and performed satisfactorily throughout the test, although the insulation resistance of the lower actuator became very low. Each actuator-position sensor assembly successfully completed 61 functional tests at operating temperature. The upper actuator was operated in four step-sequences for 80 steps in each direction during each test to rotate the position sensor through a 144° arc. The lower actuator was operated in a similar manner through 128 steps in each direction. The additional movement was needed to actuate the limit switches. The actuators produced a minimum of 20 in.-oz torque during each test. Contact resistance of the limit switches was fairly constant throughout the test at between 0.10 and 0.20 ohm. More complete data analysis, which is underway, is necessary before a precise evaluation of component performance can be made.

After the test assembly had been moved out of its in-core location, air was bled into the test chamber while specimen temperatures were maintained. After two hours, the insulation resistance of the lower actuator had increased to 1.8×10^7 ohm while that of the upper actuator increased to 2.6×10^7 ohm. This large increase in insulation resistance indicates that these components may have been contaminated with organic materials either during fabrication or pre-irradiation testing at AI.

Preparations for the hot-cell examination and final dosimetry are underway.

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VI. SNAP 8 EXPERIMENTAL REACTOR (S8ER)

A. INTRODUCTION AND SUMMARY

The objective of the S8ER was to demonstrate reactor operation and to determine reactor performance characteristics at both 450 and 600 kw. The operational phase of the S8ER test program has been successfully accomplished and all experimental goals were achieved. The reactor and test system are presently undergoing an extensive post-operation examination to determine the effects of power operation on component behavior.

Disassembly of the S8ER has been completed and all components have been sent to the AI Hot Laboratory for examination. Cleanup of the test facility has been completed. The test facility has been released by the SNAP 8 project.

Examination of the piping and components associated with the primary NaK system has revealed no significant changes in the materials. The piping was free of deposits, and no obvious erosion, leaching, or mass transfer were observed.

Detailed examination of the S8ER fuel elements continued throughout the report period. Evaluation and analysis of the large amount of data generated as a result of the fuel examination is continuing. Of the various hypotheses to explain the cladding cracking, the two most plausible are for greater-than-predicted fuel swelling due to (1) phase transformation of the fuel after excessive hydrogen loss or (2) higher-than-calculated fuel temperatures. A complete evaluation is being made to establish the primary cause and identify any secondary modes of failure.

B. DISASSEMBLY

Reactor components were disassembled from the shutdown shield within a temporary hot cell (shown in Figures 38 and 39). Viewing of the operation was performed by means of television and periscope. Disassembly was accomplished with abrasive grinding tools and a bridge-mounted manipulator. Components were disassembled from the shutdown shield within the temporary hot cell in the S8ER Test Facility reactor room then loaded into casks and transported to the AI Hot Laboratory where a detailed examination using permanently installed equipment will be performed.

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Figure 38. Temporary Hot Cell for Shutdown Shield Disassembly

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Figure 39. Shutdown Shield Disassembly Equipment

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TABLE 15
INTERNAL REFLECTOR EXAMINATION

Reflector Type	Number	Visually Discernible Defects	Average % ΔL	Average % Δt	Average % Δw
Airfoil-Shaped	12	$\left\{ \begin{array}{l} 6 \text{ had both bottom pins missing} \\ 2 \text{ had one bottom pin missing} \\ 6 \text{ were cracked} \\ 2 \text{ were bowed} \end{array} \right\}$	+0.1	-3.4	-1.6
D-Shaped	6	1 had bottom pin missing	+0.4	-3.1	-1.1

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Following component disassembly, the shutdown shield was returned to the containment vessel and the temporary hot cell dismantled. Tools were decontaminated, or discarded if more economical. The reactor room of the S8ER Test Facility was decontaminated to less than 300 dpm. Equipment and materials were then reassigned to other programs to complete the disassembly activity.

C. COMPONENT AND PIPING EXAMINATION

Visual examination of the S8ER internal reflectors in the AI Hot Laboratory revealed that six were cracked, two were visibly bowed, and nine had pins missing. It is possible that missing pins could have been caused by the disassembly operations. Average dimensional changes observed between pre- and post-irradiation measurements were: 0.2% increase in length, 1.4% decrease in width, and 3.3% decrease in thickness. Table 15 summarizes results of the internal reflector examination to date.

Inspection of the piping during system disassembly and subsequent metallurgical examination revealed no changes in the piping materials which would be of process significance.

As the primary system was disassembled, it was noted that the NaK wetted surfaces were quite clean. The piping was free of deposits, and no obvious erosion, leaching, or mass transfer were observed. Discoloration and staining were minimal. Components and piping in general, displayed the satiny silver appearance characteristic of austenitic stainless steel which has been exposed for a long period to clean NaK. No wear was observed on the outlet end caps of the EM pumps; the flow deflectors of the new end caps successfully solved the erosion problem which had developed in the original end caps.

The surface of NaK wetted piping from the primary system differed little in appearance from the surface of unexposed reference piping samples taken from the same pipe runs used for fabricating the S8ER primary system.

It was concluded that 316 stainless steel had been a good choice for the NaK-containing portions of the S8ER. The system probably could have operated many more months without difficulty.

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SYMBOL

DESIGNATION

CLADDING CRACK LONGER THAN 5 in.

CLADDING CRACKS 0 TO 5 in. long

TOP END CAP FAILURE

ELEMENT WAS BROKEN INTO 3 PIECES
(POSITION III-2 ONLY)

SPIRALED CRACK IN CLADDING
(POSITION VII-36 ONLY)

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Figure 40. S8ER Core Configuration Showing the Location and Orientation of Cladding Cracks

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D. FUEL EVALUATION

The detailed examination of selected S8ER fuel elements continued during this reporting period. Evaluation of data obtained from both the screening and detailed examinations continued.

1. Analysis of Screening Examination

During the period considerable effort was devoted to the study and correlation of the screening examination data, which was completed near the end of the previous reporting period. Meticulous examination of the photographic record led to minor revisions to the tabulation of intact and cracked elements. The revised condition of fuel elements by ring is shown in Table 16. Similarly,

TABLE 16
S8ER FUEL ELEMENT CONDITION BY RING

Ring Number	Total Number of Elements	Number Cracked	Number Intact
I	1	1	0
II	6	2	4
III	12	11	1
IV	18	18	0
V	24	24	0
VI	30	27	3
VII	36	30	6
VIII	42	32	10
IX	42	22	20
	<u>211</u>	<u>167</u>	<u>44</u>

The pictorial presentation of the core showing the location of various types of cladding cracks was revised and is shown in Figure 40.

Three general categories of cladding cracks were observed: (1) circumferential cracks in the transition area where the top end cap connects to the cladding, (2) small longitudinal cracks in the cladding area from the bottom end of the fuel rod to the midpoint of the bottom end closure cup and plug, and (3) the most common longitudinal cladding cracks of a range of sizes.

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Figure 41.

Typical Circumferential Crack in the
Transition Area Between the Top End
Cap and the Cladding Element 511
(4.3X)

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Figure 42.

Typical Hairline Crack Over
Closure Plug - Position
VII-26, Element 470
(4.3X)

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Circumferential cracking in the vicinity of the top end cap weld occurred in 6 elements, all in Core Rings III and IV. This type of crack is illustrated in Figure 41.

Figure 42 shows a typical hairline crack over the bottom end closure plug. This type of cladding crack was observed in 34 elements, all of which exhibited other additional cladding cracks over the fuel. There was a high frequency of these cracks in Ring VI and to a lesser extent in Ring VII, with a low frequency in Rings I, II, III, and IX.

Longitudinal cladding cracks ranged in length from 1/8 inch up to 12 inches. The average open crack started 3 inches from the top of the element and was 4.3 inches long. Some of the elements displayed variations in the longitudinal crack pattern in the form of spiral, forked, or short circumferential cracks. These variations in the mode of longitudinal cracking suggest that bending moments in addition to other forces were imposed on these elements. The greatest incidence of this type of cracking was in Core Ring VI where six of the 27 cracked elements displayed this behavior.

Studies were made of fuel element diameter and length changes as determined by comparison of pre- and post-irradiation measurements.

The profilometer traces reveal considerable diametral changes along the length of the elements. The maximum local cladding diameter increases of intact elements were 4 to 5 mils or approximately an 0.8% diameter increase. However, some elements show local diameter decreases of as much as 4 mils. The highest frequency of peak diameter distortion of all intact elements occurred from 4 to 7 inches from the top. As shown in Figure 43, this also is where the average longitudinal crack occurred in the cracked fuel elements. Figure 43 also shows another high frequency of peak distortion near the top end of the elements and a third group peaking at 12-1/2 inches from the top. The local distortion which occurred 4 to 7 inches from the top of the elements was by far of greatest magnitude.

Post-irradiation measurement of element lengths revealed changes ranging from -0.015 to +0.077 inch. The average change for both intact and cracked elements is presented by core ring in Figure 44. The plot shows that the

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Figure 43. Frequency of Peak Diameter Change as a Function of Distance From Top of Element

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largest length increases occurred in Core Rings VI, VII, and VIII. Tests at the 95% confidence level indicate a significant difference between the length changes of intact vs cracked elements in Rings IX, VIII, VII, and II.

As previously reported, measurements taken of the upper and lower grid plates revealed that they were dish-shaped outward from the fuel element cluster. Maximum deformations of the top and bottom grid plates were +0.023 and +0.013 inch, respectively. The clearance between the grid plates at assembly of the core was 14.52 inches at room temperature. This provided approximately 0.05-inch clearance between the elements and grid plates in the cold condition. Post-irradiation measurements showed that some of the clad fuel elements were more than 0.020 inch longer than the nominal element-to-grid plate clearance. Additionally, the six elements which experienced circumferential cracking in the top end of the fuel element (reactor outlet) at the end cap are located in Rings III and IV and correspond to positions of near maximum grid plate deformations.

The gamma scan data were taken by counting the Zr^{95} photo peak as a function of length. Because Zr^{95} has a half life of 65 days, the gamma scans are only representative of the power distribution over the last 4 to 6 months of reactor operation. However, many useful observations during this period can be made from the data.

Analysis of the gamma scan data showed that, as expected, the axial distributions of fuel rod gamma activity were skewed slightly toward the bottom. Based on the activity scan data, approximately 54% of the burnup occurred in the lower half of the average fuel rods and 46% occurred in the top half. This shift in burnup was expected because of hydrogen redistribution toward the cooler region of the fuel elements.

The data also showed that several elements and groups of elements exhibited significant deviations from the average gamma activities of adjacent fuel elements. Deviations appeared in some elements from Rings VII, VIII, and IX, and could therefore be a result of reflector drum operation. Another group of elements located in the center of the core (Rings I, II, and most of III) indicated higher average activity levels than the surrounding elements. A third group of

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*MEANS ARE SIGNIFICANTLY DIFFERENT
95% CONFIDENCE LEVEL

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Figure 44. Average Change in Element Length With ± 1 Standard Deviation
(68%) Confidence Limits on Individual Elements

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elements, generally following the Ring VI elements, exhibited significantly lower gamma activity levels than did the adjacent elements. This could be a reflection of high hydrogen loss in these elements.

Figure 45 presents an overall tabulation of the deviations between observed and predicted average gamma activities based on normalization of activity data to Core Ring I. This chart shows that 61 of the elements had average gamma activities at least 5% lower than predicted. Of these 61 elements only 6.6% were intact, compared to 23.9% intact of those in the range of $\pm 5\%$ of predicted activity, and 37.5% intact of the elements which showed average gamma activities at least 5% above that predicted. On a total core basis of 211 elements, 21% of the elements were intact.

A comparison of observed average activities with the predicted activities for the elements in the sector of Drum No. 2 is shown in Figure 46. These curves show that the average gamma activity levels for Rings V through VI, in that sector, fall significantly below the calculated end-of-life prediction. This indicates that the elements in these rings, on the average, did not achieve predicted burnup levels.

Assuming that the end-of-life power profile represents the power delivered by the core, the total difference between the observed and predicted average gamma scan data should be zero. A plot of the predicted burnup profile based upon a burnup of 0.25 met. at. % for the center fuel element and a plot of the observed burnup profile based upon ring averages of element average gamma activity are shown in Figure 47. The observed profile was vertically positioned in such a manner that the areas under the two curves are equal. The curves show that the predicted and observed average burnups are equal in Ring III. The average observed activity for Ring III was, therefore, selected as the basis for evaluation of the level of activity of each element in the core.

Figure 48 is a core plan showing the deviations between predicted and observed average gamma activities for each element determined as outlined above. The plot indicates that all the elements in Ring IV, 75% of those in Ring V, and 47% of the elements in Ring VI displayed average gamma activity levels significantly below those predicted. Further, the plot shows that five elements in the center of the core and many in Rings VII, VIII, and IX, exhibited average gamma

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Figure 45. Average Gamma Scan Deviations From Predicted Power Levels

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Figure 46. Comparison Between Observed Gamma Activity Levels for Sector 2 of the S8ER and Predicted Power Profiles

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Figure 47. Ring Average Element Burnup Profiles in S8ER Core

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GAMMA ACTIVITY NORMALIZED
TO AVERAGE OF ALL ELEMENTS
FROM RING III AND COMPARED BY
ELEMENT FROM PREDICTED
LEVELS

DRUM 4
FINE CONTROL

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Figure 48. Deviations From End-of-Life Calculated Power Levels

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activity levels significantly above those predicted. These deviations illustrate that the elements in the center of the core and in the outer three rings operated at increased power to compensate for the reduced power level of those in Rings IV, V, and VI.

2. Detailed Examination

Fourteen intact and ten cracked fuel elements were subjected to various detailed examinations. Although elements were selected from each ring, they were chiefly selected from the sector of the core next to Drum No. 2. This is because the lowest frequency of cracks occurred in this sector. A core map of the elements selected is shown in Figure 49.

Five fuel elements with intact cladding were sectioned to obtain 10-inch-long cladding tube samples for tensile testing. The fuel was dehydrided to facilitate its removal from the cladding. These five tensile tests were run at 1400°F with a cross head speed of 0.002 in./min on the Instron machine. In addition, 10-inch tube lengths from the same heat material as the S8ER tubing were tested (1) in the as-received condition, (2) in the as-processed condition (i.e., as-chromized, straightened, and coated), and (3) from fuel elements which had been on test for 10,000 hours at 1300 and 1400°F. Results of all tests are summarized in Table 17 and compared to other data in Figure 50. There appears to be an increase in the tensile strength and a loss of ductility due to processing. Within the scatter of the data, there does not appear to be an appreciable change due to 10,000 hours at 1300 or 1400°F. A loss in both ductility and ultimate tensile strength is apparent due to time at temperature plus irradiation. As the material in all cases had less than 15 to 20 grains across the cladding, comparison to properties obtained from bulk material may not be valid.

Data analysis of the burst tests (biaxial strain) on seven 10,000-hour-aged but unirradiated S8ER cladding assemblies was also completed. Ultimate strength data of these assemblies show wide variation (from 27,000 to 42,500 psi) and are generally inconclusive in predicting ultimate strength of other Hastelloy-N material. This is because most of the cladding ruptures were in the form of pinholes or intergranular (brittle failure) hairline cracks which developed within 2 inches of either end of the assemblies. Two of the assemblies had gross ruptures near the center of the cladding (41,000 psi

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Figure 49. Location of Fuel Elements Selected for Detailed Examination

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SYMBOL	LAB	RADIATION CONDITIONS TEMPERATURE (°F) nvt	STRESS STATE	THERMAL HISTORY	STRAIN RATE (%/min)
△	ORNL	NONE	UNIAXIAL	ANNEALED	0.2
○	ORNL	7 X 10 ²⁰ (E>1 mev) 9 X 10 ²⁰ (thermal)	UNIAXIAL	ANNEALED	0.2
□	BNEL	NONE	UNIAXIAL	ANNEALED	0.5
■	BNWL	1364 1.91 X 10 ²⁰ (E>1 mev)	UNIAXIAL	ANNEALED	0.5
●	AI	NONE	UNIAXIAL	AS-RECEIVED	0.2
◆	AI	NONE	UNIAXIAL	COATED	0.2
▲	AI	NONE	UNIAXIAL	COATED PLUS 10,000 hr AT 1300°F	0.2
◆	AI	NONE	UNIAXIAL	COATED PLUS 10,000 hr AT 1400°F	0.2
◇	AI	S8ER 5-7 x 10 ²⁰ (E>1 Mev)	UNIAXIAL	S8ER	0.2
●	AI	NONE	BIAXIAL	10,000 hr AT 1300°F	SLOW
◆	AI	NONE	BIAXIAL	10,000 hr AT 1400°F	SLOW

Figure 50. Hastelloy-N Short-Term Mechanical Properties

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TABLE 17
TENSILE TEST DATA* ON S8ER CLADDING SAMPLES

Condition of Specimens	S8ER Core Position	Element Number	Test Temperature (°F)	Ultimate Load (lb)	Strain-to-Rupture [†]	
					Reference Position [§]	Average**
As-Received, Heat 58304	-	-	1400	395	5.6	6.9
	-	-	1400	400	6.0	7.6
	-	-	1400	320	3.9	5.8 } 6.8
As-Processed	-	980	1400	765	5.6	4.5
	-	885-N	1400	835	3.1	3.2 } 3.9
Aged, As-Processed						
10,000 hr at 1400° F	-	858	1400	590	4.9	6.1
10,000 hr at 1300° F	-	1052	1400	885	4.9	4.5
10,000 hr at 1300° F	-	1001	1400	885	5.9	6.1
Irradiated	II-5	335	1400	670	0.7	0.7
	II-6	181	1400	550	1.0	0.8
	VII-29	491	1454	470	1.0	1.3
	VIII-36	316	1400	600	0.7	0.8
	IX-7	340	1400	620	1.5	1.2

*Data not corrected for load cell compliance or test specimen low temperature elasticity.

[†]Elongation = 1 in. gauge length.

[§]Measured along the vertical line where calibration thermocouples were located.

**Average of three readings at 120°.

stress) and two of the assemblies failed at the closure end-cap weld. Three of the assemblies were aged at 1300°F and four were aged at 1400°F before the burst test, but no correlation with aging temperature was found. The burst tests were conducted at 1400°F. Individual data from the burst tests are listed in Table 18. These data were also shown in Figure 50 and are in the same range as the uniaxial data on aged S8ER tubing.

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TABLE 18
BURST TEST DATA OF THERMALLY AGED S8ER FUEL CLADDING

Cladding Number	Aging Temperature* (°F)	Maximum Pressure (psig)	Approximate Yield (0.2% offset) (psi)	Ultimate Strength (psi)	Type of Failure
N-1019	1300	1000	Not attained	27,500	Closure weld crack
N-804	1400	1300	28,000	36,000	Pinhole near probe cap
N-1029	1300	1300	26,000	36,000	Closure weld crack
N-1032	1300	1450	33,000	40,500	Pinhole near probe cap
N-907	1300	1500	31,000	42,500	Gross cladding
N-972	1400	1400	31,000	39,500	Gross cladding
N-992	1400	1500	32,000	42,000	Longitudinal crack 2 in. from probe end

*10,000 hr

Metallographic examination of the above assemblies was completed. All fractures observed were intergranular. Failures near the probe cap end appear to be due to thinning which resulted when the surfaces were machined in preparation for welding prior to performing the burst test. Four tubes, representing the two aging temperatures, were evaluated to determine grain size and microhardness of the aged material. The different aging temperatures (1300 or 1400°F) made no significant difference in the microhardness or the grain size. The results are shown in Table 19.

Nine intact fuel elements were punctured for quantitative fission gas analysis. These data are presented in Table 20 for the percentage of Kr⁸⁵ released from the inventory in the individual fuel rods. This fractional release is slightly less than expected. On the basis of the fractional Kr⁸⁵ release, the total calculated volume of all significant gaseous fission products released is only about 0.005 cc(STP) of gas. Therefore, the cladding cracks could not have been caused by high internal pressure due to excessive fission gas release.

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TABLE 19
MICROHARDNESS AND GRAIN SIZE OF
THERMALLY AGED S8ER FUEL CLADDING

Tube Number	Aging Temperature * (°F)	KHN Microhardness	Grain Size
1029	1300	140 154 170 Average 156	7
907	1300	191 198 227 Average 205	8
972	1400	191 205 219 Average 205	8
992	1400	164 164 185 Average 171	8

*10,000 hr

TABLE 20
FISSION GAS ANALYSES

Ring Position	Element Number	Kr ⁸⁵ Released (%)
II-2	358	0.017
II-4	474	0.016
VI-24	232	0.017
VI-25*	388	0.010
VII-28	525	0.010
VIII-6	188	0.016
VIII-38	175	0.017
IX-15	206	0.013
IX-29	293	0.014

*Two punctures were made on this element, one at the top and one at the bottom.

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Fourteen fuel elements were declad by longitudinal Elox cuts and end cap removal to obtain two intact halves (clam shells) of the cladding for hydrogen barrier evaluation. The fuel rods from these elements are being selectively examined for fuel swelling, hydrogen retention, microstructure, and burnup.

Five elements were cut to obtain intact cross sections of the cladding, hydrogen barrier coating and fuel. These cross sections are being metallographically examined for interface effects, cladding thickness, and material microstructures.

Those elements which were declad were examined visually with the Kollmorgen periscope and stereomicroscope to detect any abnormalities. The surfaces of the ceramic coating and fuel rods were closely examined for any reaction effects between the fuel, ceramic, and cladding. Also the effects of NaK on the components of the cracked elements were investigated. Figure 51 shows Element 432 (Core Position VIII-29), an intact element, after it was declad showing internal condition. The ceramic coating was in relatively good condition, as was the fuel rod. A deposit which increases in thickness in the higher temperature regions of the cladding is typical of that observed on out-of-pile tests of S8ER fuel elements tested for 10,000 hours at 1400°F. Figure 52 shows Element 111 (Core Position IX-42), a cracked element, after being opened. NaK had entered the element and interacted with the ceramic and much of the ceramic had loosened from the cladding. Figure 52 shows the poor condition of the ceramic coating near the top end of the element. The white areas are NaK oxide deposits. The inside surface of one of the cladding cracks in Element 111 is also shown in Figure 52. The edges next to the crack are rounded and eroded away. The typical condition of the ceramic coating not affected by NaK was shown in Figure 49. Some pitting of the ceramic can be seen but this is not an unusual condition. Elements 432 and 111 are representative of the elements selected for examination.

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Figure 51. Typical Condition of an Intact Element:
Element 432 after Disassembly

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Figure 52. Typical Condition of a Cracked Element:
Element 111 after Disassembly

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TABLE 21
MAXIMUM MEASURED DIAMETER CHANGES AND LOCATION
FOR 14 FUEL RODS

Element Number	Fuel Rod Orientation 0° Diameter Change		Distance From Top End (in.)	Fuel Rod Orientation 90° Diameter Change		Distance From Top End (in.)
	(in.)	(%)		(in.)	(%)	
332	0.0025	0.47	2	0.0065	1.22	7
432	0.0070	1.32	3-4	0.0074	1.39	3-4
476	0.0028	0.53	9-10	0.0028	0.53	9
111	0.0061	1.15	12	0.0062	1.17	12
378	0.0106	1.99	3	0.0106	1.99	3
342	0.0090	1.69	5	0.0090	1.69	5-8
474	0.0042	0.79	2-5	0.0044	0.83	3-4
188	0.0028	0.53	12	0.0034	0.64	11
525	0.0054	1.01	8-9	0.0048	0.90	8-9
206	0.0023	0.43	11	0.0023	0.43	11
293	0.0031	0.58	13	0.0030	0.57	13
309*	0.0140*	2.63*	3	0.0140*	2.63*	5
398	0.010	1.88	8	0.009	1.68	7
232	0.0063	1.18	7	0.0053	1.00	5

*Ceramic adhered to fuel rod surface resulting in abnormally large diameter readings.

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Diametral measurements were taken of all the disassembled elements. The profilometer measuring device was used for intact fuel rods and a dial indicator measuring device was used on broken fuel rods. Six of the 14 disassembled elements had intact fuel rods. The rest were in two or more pieces. The maximum diameter changes are presented in Table 21 for 0° and 90° element orientations. These data have not been corrected for changes due to hydrogen content.

Length measurements were taken of all the intact fuel rods using a dial indicator measuring device which was accurate to within 1 mil. The length data on the 6 intact rods are listed in Table 22.

TABLE 22
LENGTH CHANGES OF INTACT
FUEL RODS

Element Number	Length Increase (in.)	Length Change (%)
432	0.061	0.43
111	0.014	0.10
342	0.036	0.36
474	0.047	0.34
525	0.032	0.23
206	0.023	0.16

Adherence tests were performed on 31 cladding samples to determine the quality of the ceramic-to-metal bond. The samples were taken from six fuel elements (111, 188, 474, 342, 332, and 293) and were distributed along the length of each. The tests were accomplished by gripping two ends of a cladding section and bending the cladding back and forth until the ceramic spalled off. The tests indicated poor adherence of the ceramic as it spalled off easily and in most cases, during the first bend.

In addition to determining a rough adherence index, a qualitative measure of the room temperature ductility was obtained by bending each cladding sample to failure. The samples were bent 90°, then 180° in the opposite direction and back down through 180° to the first 90° position. This cycle constituted one complete "bend." Four to six samples each of the cladding of six elements were tested in

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this manner. Failure varied from 6 to 14 bends, with samples for one element averaging 29 bends prior to failure. In view of the relatively low value of ductility at high temperature and the high number of bends required for failure at room temperature, this test indicates a significant improvement in cladding ductility going from 1400°F to room temperature.

Selected density samples were cut from each of the fuel rods based on previous information obtained about the rods. The measurements were taken by the liquid displacement method. The data have not been analyzed in detail as yet with respect to dimensional changes, however, the percent density changes of all the samples varied from an increase in one sample from Element 332 of 6.18% (this sample indicated a marked hydrogen loss when visually examined) to density decreases as high as 4.36% measured on a fuel sample from Element 378.

Fuel samples have been cut and are being prepared for burnup analysis, vacuum extraction hydrogen analysis, and metallography. Also cladding metallography samples have been prepared for examination.

Hydrogen analyses were made of samples from Rods 432 (intact), 519 (cracked), 378 (cracked), 309 (cracked), and 111 (cracked). Some preliminary conclusions which can be drawn from the analyses completed to date are:

- 1) The axial hydrogen distribution shows the highest hydrogen content to be at the bottom (cold end) of the fuel rods, as would be expected.

- 2) The axial hydrogen distribution is consistently lower at the top of the fuel rods. Available data are insufficient to map the axial hydrogen distribution in detail. Hence, it cannot be determined at this time whether the expected minimum occurred in the region 4 to 6 inches from the top, as predicted.

- 3) The fuel rods from cracked elements near the edge of the core (378 and 111) have more hydrogen remaining than those from the central section (519 and 309).

- 4) The hydrogen content of the center specimen in each of the sampling wafers is within ± 0.02 wt % of the edge specimens in all except two cases.

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The overall accuracy of hydrogen content determination is ± 0.02 wt %. For a radial temperature difference of 50 to 100°F, this difference in hydrogen content is calculated to be between 0.05 and 0.10 wt %. It therefore appears that the attempt to "freeze in" the radial hydrogen distribution upon shutdown was unsuccessful.

The cladding metallography has been completed on six elements: 332, 342, 474, 188, and 293. Some preliminary conclusions which can be drawn from the cladding metallography completed to date are:

1) One side of the top end cap weld of Element 332 separated during operation. This element has shown a large loss of hydrogen in this region. The weld area located 180° from the cracked region appears normal. As this element increased in length by 0.030 inch, it appears the crack was caused by high local stresses in addition to a local thinning in the weld zone.

2) Examination of samples from cracked cladding indicated all fractures were intergranular in nature. In the cladding of some of the cracked elements there was heavy grain boundary precipitation. Profuse cracks appeared to occur along these grain boundaries. Because the grain boundary precipitates respond to the same rate of etching as the chromized surfaces of the cladding, it appears they may be occupied by chromium rich intermetallic compounds.

3) Based on cladding sections taken over the length of one intact element, these precipitates were more prevalent toward the bottom or cooler regions of the cladding.

4) There has been no evidence of gross corrosion of the cladding by the NaK on the cladding of the elements examined to date.

The fuel metallography which has been completed shows:

1) The phases of the material agree well with the hydrogen content as determined by vacuum extraction hydrogen analysis from small sections next to the metallography wafers.

2) Phase change has occurred in all cracked elements except 111, which was at the edge of the core and may have cracked late in life.

Evaluation of the large amount of data from the S8ER examination and the NAA-115-1 experiment is in process. Various explanations for the cracked

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cladding have been considered. Of these, the two most plausible hypotheses for greater-than-predicted fuel swelling are (1) phase transformation of the fuel after excessive hydrogen loss and (2) higher-than-calculated fuel temperatures. A complete evaluation is being made to establish the primary cause and identify any secondary modes of failure.

3. Performance Analysis

In support of the S8ER fuel evaluation, analyses of localized core conditions which may have contributed to the observed fuel behavior have been in progress. These analyses include an evaluation of stress levels arising from the following sources: circumferential temperature variations around the fuel element cladding, temperature gradients across the cladding wall, interference between fuel elements and grid plates, and the effect of the thermal transients on cladding stresses. The effect of radial gap changes between fuel and cladding on fuel peak temperatures as a function of core lifetime has also been determined.

The displacement and thermal stresses in the cladding of a center element were analyzed for two 450-kw, steady-state, temperature conditions. The first was a circumferential temperature variation and the second was the same circumferential variation with the addition of a thermal gradient across the cladding wall. In this analysis all pertinent variables, such as fuel element spacing, were assigned nominal values. For the first case, the temperature distribution is given by $T = 1233 + 37 \cos 6\theta$ ($^{\circ}\text{F}$); for the second case the expressions are $T_{\text{outer wall}} = 1233 + 37 \cos 6\theta$ ($^{\circ}\text{F}$) and $T_{\text{inner wall}} = 1235 + 35 \cos 6\theta$ ($^{\circ}\text{F}$). For Case 1, the calculated minimum radial expansion is 0.002560 in. and the maximum is 0.002565. The resulting circumferential stress is zero, and the maximum axial stress is 7500 psi. In Case 2, the calculated minimum radial expansion is 0.002563 in. and the maximum is 0.002565 in. The associated circumferential stress is 202 psi and the maximum axial stress is 7740 psi. These results indicate that the circumferential temperature variation during steady-state reactor operation has a negligible effect on the circumferential stresses and radial displacement of the fuel cladding.

Thermal analysis of the effect of eccentrically positioned fuel under steady-state and transient conditions was conducted. The results are to be used as input to cladding stress analyses. A digital computer simulation of the transient

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temperature behavior in the S8ER core during Scram 34 was made. This scram occurred on December 14, 1963, and was initiated by failure of a fuse in the pump circuit.

Prior to the scram, the reactor was assumed to be at 600 kwt with an outlet temperature of 1300°F. It was assumed that a channel between Rings IV and V was surrounded by three elements containing eccentrically located fuel which touched the cladding adjacent to the channel. These three elements showed the largest circumferential temperature gradient in the cladding, 80°F/inch at the axial midplane. This gradient arises only from the eccentricity of the fuel (fuel/clad gap) and does not include the effect of non-uniform heat transfer if the cladding touches an adjacent element. The resulting circumferential temperature gradient has been estimated to be more than 500°F/inch.

During the scram transient, the circumferential cladding temperature gradient decreased. The maximum instantaneous rate of change of local cladding temperature was 187°F/sec. This occurred on the center element at the end of the flow transient, prior to the start of the power transient. The analysis resulted in the conclusion that the maximum circumferential temperature gradient in the cladding exists during steady-state operation at power, the gradient during the transient being appreciably smaller.

The effects of insufficient clearances between the grid plates and the fuel elements were considered to determine whether the small clearances, coupled with thermal length changes, could be a cause of cladding failure. The calculated maximum tensile force that can be generated by the tie rods is 4870 lb. If this maximum load were distributed to one fuel element, it would tend to bow or relax and the load would be distributed to other elements. If failure occurred as a result of bowing, the cracks would occur in a circumferential direction.

As noted in the discussion under the screening examination, a total of six elements, located in Rings III and IV, actually exhibited this type of failure. This provides evidence that fuel element constraint by the grid plates may have contributed to this small group of failures.

Thermal transient analyses were also conducted to establish the maximum temperature gradients which could have occurred in the upper and lower end caps during S8ER operation. Both pre-startup and scram transients were

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Figure 53. S8DS Maximum Fuel Temperature

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Figure 54. S8ER Hot Element Temperature Profile
600 kw, 200°F ΔT

analyzed. The most severe temperature gradient occurred in the lower end cap during an EM pump restart transient which occurred on August 1, 1963. The results are being utilized to determine the resulting stresses.

The effect of fuel swelling upon S8DS and S8ER maximum fuel temperature history was determined. Isotropic swelling was assumed using the Miller correlation.*

$$\frac{\Delta D}{D_{\text{initial}}}(\%) = \frac{1}{3} \left[(1.2 \times 10^6) b^2 \exp\left(-\frac{22,600}{T_{\text{max}}}\right) + 1.0 b \right]$$

where D is the fuel diameter, b is the burnup in met. at. % and T is in °R. The maximum fuel temperature in the S8DS reactor decreases during operation from an initial value of 1450 to 1380°F when the fuel has swelled sufficiently to close the gas gap and contact the cladding (Figure 53). With 600-kwt power level and a nominal swelling rate, the gap is closed at 34-1/2 months; with three times nominal swelling rate, the gap is closed at 18-1/2 months.

It should be noted that this analysis does not consider the accumulation of fission gas products, principally xenon, in the radial gas gap. While this effect is negligible for S8DS elements in which large axial clearance is provided between fuel rod and upper end cap, fission gas buildup has as important a thermal effect as fuel swelling on a S8ER element assembled with minimal fuel rod-end cap clearance. In the latter case, the thermal conductivity of the xenon is sufficiently low to offset the increase in conductance due to fuel swelling and gap reduction. Consequently, the maximum S8ER fuel temperature (at 450 kwt) increases during lifetime from 1395°F to approximately 1410°F.

The radial temperature distribution in a fuel element during 600-kwt operation with no allowance for swelling is shown in Figure 54. The rod has been assumed to be positioned symmetrically within the cladding. This is the condition which gives the highest fuel temperature although cladding and ceramic barrier temperatures predicted by this model are somewhat lower than those obtained when the element touches the ceramic barrier. Also shown is the temperature distribution which results when conservative hot channel effects are included.

*G. Burdi, "SNAP Technology Handbook, Vol. II: Hydride Fuels and Cladding," NAA-SR-8617, Vol II (SRD), November 15, 1964

4. Analysis of Radiation Measurements

Analysis and evaluation of radioactivity measurements made during S8ER operations, after shutdown, and during post-mortem examinations were continued. Investigations of sample results and radiation measurements were made and are continuing in the following areas: (1) primary coolant activity (NaK samples), (2) cover gas activity (gas samples from surge tank), (3) vent gas activity (gas samples from holdup tanks), (4) direct radiation from primary system (surge tank), and (5) fuel element gases (gases released from intact elements which are punctured during post-mortem examinations). Results of these studies are also being used for improving and defining the sampling needs for the S8DS.

Results of the seven NaK samples which were collected during the program were reported in the previous progress report. Two of the samples were collected during the operating period, one during the final shutdown (three hours after reactor shutdown), and four during subsequent disassembly operations.

Calculations based on the cesium-137 activity in the sample taken from the coolant stream during final shutdown on April 15, 1965 indicate a predicted number of fuel failures which is in close agreement with the number actually observed during the subsequent core examination (168 failures predicted and 167 observed). The calculated value is based on the assumption that 0.17% of the Cs¹³⁷ inventory from a failed element would be released to the coolant (0.17% corresponds to the recoil activity from a SNAP 8 fuel rod)*. This sample, therefore, indicates that the fuel ruptures occurred before the sample was taken and probably sometime during reactor operation.

Similar calculations based on the results of the NaK sample which was taken on December 31, 1963, about a month after startup would indicate that there were no failed elements in the core at this time (about 13% of the activity expected from recoil considerations from one rod). However, the confidence in the results of this sample is less than for the April 15, 1965 sample. The December 31, 1963 sample was taken after two short operating periods (<150 Mw-hr of generated energy) and from relatively cold NaK which had been transferred from the primary system to the primary drain tank following a scram which had occurred on December 14, 1963.

*A. R. Piccot, Editor, "SNAP 8 Experimental Reactor (S8ER) Final Safeguards Summary Report - with Addendum II," NAA-SR-6958, February 28, 1962

The NaK sample which was taken on April 14, 1964, was not analyzed for fission products and is therefore not useful here. The remaining samples which were taken during system disassembly operations provided support and confirmation of the results and conclusions derived from the April 15, 1965 shutdown sample.

In summary, although the NaK samples indicate that fuel failures occurred sometime during reactor operation, the actual time of occurrence cannot be determined from this data.

Gamma spectrometer analysis of cover gas samples, for Xe^{135} activity, were also discussed in the previous quarterly. Calculations based on quantities of Xe^{135} in the samples led to the conclusion that no fuel failures had occurred prior to June 1964, and that only a few were present in the core thereafter. This calculation was again based on the assumption that 0.17% of the Xe^{135} inventory would be released from a failed element. However, the resulting conclusion is not consistent with the results of the NaK sample taken at shutdown or observations made during post-mortem examinations.

The discrepancy has been attributed, in part, to solubility of fission gases in hot NaK which would prevent representative sampling of the released activity. It is also possible that gas holdup in remote pockets in the primary system (such as the shield plug annulus) could have contributed to the failure to detect more activity by this sampling and analysis method.

Nevertheless, consideration of the relative activity values of the surge tank samples (coupled with the results of the NaK sample taken at shutdown) has lead to the conclusion that most, if not all, of the fuel failures probably occurred between June and September 1964, when a factor of 100 to 1000 increase in the Xe^{135} activity was observed. The conclusion is obtained by normalizing the gas sample results to the NaK sample results obtained on April 15, 1965.

Efforts to utilize the Health Physics vent gas sample results as indicators of fuel failures in S8ER are complicated by the presence of Ar^{39} activity. This isotope is a pure beta emitter and was not expected or specifically identified when the samples were taken and analyzed. However, the Ar^{39} generation in NaK, $\text{K}^{39} (n, p) \text{Ar}^{39}$ and corresponding concentrations in the cover gas as predicted in NAA-SR-11303* could account for much of the observed activity

*S. L. Rosen, "Argon Generation in NaK-Cooled Reactors," NAA-SR-11303, November 10, 1965

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reported as fission products in the early vent gas samples. This factor explains many of the apparent discrepancies in the early data. Argon-39 is a long-lived ($T_{1/2} = 260$ yr) emitter of moderately high energy beta ($E_{\max} = 0.56$ Mev) with no gamma radiation. The absence of γ_{rad} prevented detection by the gamma spectrometer methods. The beta counting system used is more sensitive to Ar^{39} than it is to the expected Xe^{133} . The sensitivity factor (about 8) combined with an improper decay "correction" led to high estimates of Xe^{133} concentrations and indication of possible cladding failures. Interpretation of the data, taking Ar^{39} into account, indicates no failed elements prior to June 1964. The presence of failed elements is indicated in subsequent months. It is concluded that the results of the vent gas samples support the results of the surge tank samples which indicate that fuel failures occurred in the June to September 1964 period.

In April 1964, an ionization chamber was installed above the primary surge tank in the vault to assist in monitoring power levels. During the periods June-September 1964 and January-April 1965, this monitor indicated an increase in activity level which could not be explained by changes in operating conditions. Calculations indicate that these increases could be explained by fission gases which would collect in the surge tank as a result of operation with failed elements in the core; therefore, these measurements would also support the previous conclusion that fuel failures occurred after June 1964.

Gamma spectrometer analyses of fission gases released from intact fuel elements which were punctured during post-mortem examinations have been performed for nine of the S8ER fuel elements. Results of these analyses were shown in Table 20.

These results indicate that the percent release is about a factor of ten less than one would expect from recoil or observations from previous fuel capsule experiments. The reasons for these discrepancies are not fully understood but since the measurements indicate that actual releases did not exceed the prediction, the number of fuel failures would tend to be underestimated rather than overestimated.

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E. SYSTEM ANALYSIS

Hydrogen worth significantly varies in SNAP type reactors with the concentration of hydrogen and the position of the control drums. The worth of hydrogen increases either from the drums turning out or from a decrease in the hydrogen density. Alternately, the worth decreases as the drums are turned in. A correlation of these relationships with the shape of the curve in Figure 55 requires a review of the operating history of the S8ER. The initial rise of hydrogen worth in Figure 1 can be attributed to the samarium burning out which turns the drums back out. The flattening out and decrease of hydrogen worth can be attributed to the reactivity losses which turns the drums in. The large hydrogen loss toward the end-of-life overrides the effect of the drums being turned in and the hydrogen worth increases again.

The hydrogen worth as a function of S8ER operating time is almost constant over the life of the reactor. Within the accuracy of the calculation the hydrogen worth in the S8ER could be considered a constant at $\$1.00/0.1 N_H$.

The redistribution of hydrogen during the shutdown sequence has been analyzed relative to its effect on reactivity. The rehydrating of the low concentration zones represents a reactivity gain of less than 0.1%.

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Figure 55. S8ER Hydrogen Worth

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VII. SNAP 8 GROUND PROTOTYPE SYSTEM (S8DS)

A. INTRODUCTION AND SUMMARY

The objectives of the S8DS program are to design, fabricate and test the ground test model of the SNAP 8 reactor. The ground tests of the reactor will demonstrate startup, steady state and transient performance of the reactor in a space simulated environment both coupled and uncoupled to a representative SNAP 8 power conversion system.

Analysis of the reference design indicates that the reactor lifetime will be controlled more by the time it takes the fuel to undergo a phase change than by the loss of reactivity due to hydrogen leakage. On this basis, the 3-inch one-piece control-drum shim may be undesirable due to decreased shutdown margin of the reactor early in life soon after the burnout of the prepoison.

Preliminary investigations were conducted on three possible core design modifications to accommodate an increased (15 mil in place of the current 10 mil) fuel element cladding thickness within the current core envelope. All three modifications result in reactivity penalties which are acceptable and only minor variations in fuel temperature. As in the case of the reference design, the lifetime limitation will be the time for the fuel to undergo phase change.

The acceptance testing of the S8DS reactor components is nearing completion. The rejection of the control-drum bearings due to nonconformance is pacing the assembly of both the S8DS reactor and reflector system design verification test assembly.

B. SYSTEM ANALYSIS

1. Reactor Lifetime

Excess reactivity as a function of time for the reference design fully shimmed S8DS reactor has been calculated for various samarium-149 prepoison loadings. The results are shown on Figure 56 for the S8DS reactor operating at 600 kw with nominal reactivity losses.

The currently specified samarium-149 loading of \$3.00 was originally specified for a much less reactive system. Consequently, the current reference

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Figure 56. S8DS Reactivity vs Time With Various Samarium Loadings

design will not have sufficient shutdown margin if the maximum credible accident is considered to be that in which two control drums stick in the most reactive (full-in) position.

One method of obtaining additional shutdown margin is to increase the prepoison loading. Although this reduces the excess reactivity (and increases the control margin) during the initial full power operation, the amount of added control is decreased as the prepoison is burned out. Increasing the samarium-149 loading from \$3.00 to \$12.00 only allows an overall improvement of \$1.40 in control worth at the maximum hot operating conditions. It would be necessary to employ a different prepoison which burns out at a slower rate if added control is required beyond the first few months operation.

Another possible limiting factor on reactor lifetime is the delta-beta phase change of the fuel. The expected operating time to point of delta-beta phase change has been calculated for various hydrogen loss rates for S8DS operation at 600 kw. The results are shown in Table 23.

TABLE 23

S8DS PHASE CHANGE LIMITS
(600 kw, 1100° F inlet, and 200° F ΔT)

Hydrogen Loss Rate	Phase Change Time (months)
Nominal	60
Three times nominal	36
Six times nominal	18
Ten times nominal	11

2. Drum Shimming

Additional control may be supplied to the S8DS reactor by decreasing the number of shims to two per drum from the current reference design of three shims per drum. Sufficient shutdown margin would then be provided to safely shut the reactor down in the case of the maximum credible accident. Decreasing the number of shims per drum from three to two will decrease the excess reactivity of the system while essentially

leaving the worth of an individual drum unchanged.*

Sufficient reactivity will exist with two shims for reactor operation to continue for approximately 26 months at full power with ten times the nominal hydrogen leakage rate.

Under this condition, reactor life would be limited by the fuel phase change (Table 23). Obtaining the desired additional control through the reduction of drum shims thus appears quite feasible. The samarium-149 prepoison loading of \$3.00 would be maintained.

The effect of using 5-1/8-in.-thick beryllium control drums (3 shims) on drum step size and stepping rate was investigated. Maximum step size is a function of deadband width and minimum step size is determined by PCS startup and long-time control requirements. Maximum allowable drum insertion rate is proportional to peak differential worth and drum step size. For the present peak differential worth and a step size of 0.26°, the minimum allowable time interval between steps is 1.4 sec.

3. Fuel Elements

Steady-state temperature distributions have been calculated for the S8DS core and are shown as isothermal plots in Figure 57. For comparison, similar

*D. G. Mason, "SNAP 8 Progress Report, May-July 1965," September 30, 1965, NAA-SR-11492 (CRD), pp 26-27

Figure 57. S8DS Maximum Fuel Temperature Isotherms - 600 kw, 200°F ΔT

Figure 58. S8ER Maximum Fuel Temperature Isotherms - 600 kw, 200°F ΔT

isotherms for S8ER are shown on Figure 58. The S8DS maximum temperature is somewhat higher than that of the S8ER primarily due to the larger hydrogen gas gap in the fuel elements. The slightly higher power peaking in the S8DS core is not evident in fuel temperatures because a more highly peaked flow distribution was also used. Both of these figures are for the initial hot dimensions of the element operating at 1100°F inlet temperature, 600 kw, and 200°F temperature rise. As time progresses and the gap tends to close due to fuel swelling, the fuel temperatures will be lowered.

4. Design Modification Studies

As one step in increasing fuel element reliability, consideration was given to the use of 15-mil cladding in place of the presently used 10-mil cladding on the fuel elements. Since the additional 5 mils could be added to either the OD or the ID of the cladding, an investigation was made of the effect of both approaches on mechanical aspects of the system. No changes to the core components other than the regrinding of the fuel slugs to a smaller diameter would be required if the additional 5 mils were added to the cladding ID. The alternate solution required a reduction in the number of fuel elements which, in turn, required new grid plates, internal reflectors, and flow baffle plate. A summary of design alternates is given in Table 24.

The reactivity penalty relative to the S8DS reference design is nearly equal for all three modifications considered as noted in Table 25. The total reactivity loss at various hydrogen permeation rates is shown as a function of operating time at rated power and temperature in Figure 59. The available excess

TABLE 24
DESIGN MODIFICATION FOR CLADDING THICKNESS STUDY

Modification	Number of Fuel Elements	Fuel Rod Diameter (in.)	Cladding Thickness (in.)	Fuel Element Spacing (in.)	Fuel Element Pitch (in.)
S8DS Reference	211	0.530	0.010	0.010	0.570
I	211	0.520	0.015	0.010	0.570
II	199	0.530	0.015	0.010	0.580
III	199	0.530	0.015	0.015	0.585

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Figure 59. Total Reactivity Loss for S8DS at Various Hydrogen Permeation Rates

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TABLE 25
REACTIVITY PENALTY ASSOCIATED WITH
INCREASED CLADDING THICKNESS

Modification	Number of Fuel Elements	Fuel Element Pitch (in.)	Reactivity Penalty (\$)
I	211	0.570	-4.30
II	199	0.580	-4.25
III	199	0.585	-4.50

reactivity is also shown for the fully shimmed S8DS reference design, the nominal S8DS with 2 shims per control drum (4-1/8 in. reflector), and the three alternate designs being considered. The lifetime at various hydrogen permeation rates for a given excess reactivity is thus the point at which the reactivity loss line crosses the appropriate available excess reactivity line. At higher permeation rates, lifetime is limited by the fuel phase change.

Calculations show that any of the three proposed design modifications would have a minimal effect on the fuel and cladding temperatures. The temperature changes relative to the S8DS reference design are shown in Table 26. While

TABLE 26
CHANGES IN MAXIMUM FUEL AND CLADDING TEMPERATURE
WITH 0.015-in. CLADDING THICKNESS

Modification	Number of Fuel Elements	Fuel Element Pitch (in.)	Maximum Fuel Temperature Change* (° F)	Maximum Clad Temperature Change* (° F)
I	211	0.570	+3	+3
II	199	0.580	+12	+4
III	199	0.585	+8	-2

*Relative to S8DS reference design (211 element, 0.010-in. clad)

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changes in maximum clad temperatures are small, change in fuel element pitch (Case III) may significantly lower peak cladding temperatures by providing increased area for crossflow and by elimination of touching elements.

The S8DS flow baffle was designed specifically for the 211 element, 0.570-in. pitch core. Thus, design modifications II and III will result in altered flow patterns. The element spacing for modification II is not changed from the reference design and although the channel area is increased about 3%, flow profiles should not change markedly. Any changes would be minimized if the flow bypass area in the outer annulus is not changed.

The increased element spacing in Design III will have considerable effect on the flow profile. With the larger crossflow area, the flow will experience increased flattening as it progresses through the core. The S8DS design baffle produces a maximum fuel temperature (fuel centerline) radial profile which is slightly lower in the center of the core. Increased flow flattening will tend to compensate this temperature profile and move the region of high fuel temperatures to the center of the core.

Based on the preliminary analysis, any of the three design modifications should not require redesign of the flow baffle.

C. REFLECTOR AND DRIVES

1. Control Drum Bearings

The first set of control drum bearings delivered did not meet the design requirements and was considered unacceptable for installation on either the half-reflector test assembly or the S8DS reactor assembly. These bearings had been subjected to various rework to overcome the difficulties encountered during fabrication. It was concluded from re-evaluation of the design requirement that acceptable bearings can be fabricated by the careful application of the developed techniques. A second set of the control drum bearings is now being procured.

2. Control-Drum Shims

The set of 3-in.-thick control-drum shims was anodized utilizing the same procedures as were used on the reflector and the control drum. From a recent evaluation of alternate methods of providing high surface emittance to beryllium

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it was concluded that only the anodized surface can withstand the thermal vacuum environment.* The previously reported reduction of surface emittance of the anodized beryllium as a function of exposure time† will increase the reflector temperatures moderately. The maximum expected temperature is acceptable.

D. CONTROLS AND INSTRUMENTATION

The fabrication and the assembly of the precision drum position indicator§ and the automatic controller** were completed. Both components performed adequately during the functional checkout. These units are ready for the final acceptance tests.

The design of the temperature switch was revised to incorporate the double bridge concept described in Section V-F of this report. The assembly of the temperature switches was initiated.

E. S8DS ASSEMBLY AND ACCEPTANCE TESTING

The S8DS components are subjected to formal acceptance tests to ensure adequate performance. As described in the previous progress report, these tests are conducted at S8DS operational thermal vacuum environments as well as at room ambients. Complete documentation of the test procedures and test results are maintained.

The assembly of the reactor system is also conducted in accordance with documented procedures. All in-process measurements are recorded. The acceptance checkout and test of the reactor system is also conducted under S8DS simulated thermal vacuum environments.

1. Component Acceptance Testing

The acceptance testing of the components is progressing satisfactorily. As noted in Table 27 enough components, with the exception of the position sensor,

*D. G. Mason, "SNAP 8 Progress Report, August-October 1965," NAA-SR-11692 (CRD), December 29, 1965

†D. G. Mason, "SNAP 8 Progress Report, May-July 1965," NAA-SR-11492 (CRD), September 30, 1965

§C. E. Johnson, "SNAP 8 Progress Report, November 1964-January 1965," NAA-SR-10792 (CRD), March 15, 1965

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

TABLE 27
S8DS COMPONENT REQUIREMENTS

Component	S8DS System	Reflector System Design Verification Tests*	Planned Spares	Completed Acceptance Test
Snubber	6	3	3	12
Limit Switch	32	16	22	64
Actuator	6	3	5	13
Scram Mechanism	6	3	3	7
Position Sensor	6	3	3	0
Demodulator	6	3	0	0

*Components to be retained as spares after testing

have been accepted to meet the need for the half-reflector system design verification test and the S8DS reactor.

Of the 13 actuators accepted, four are being restricted to use as spares due to minor discrepancies. The last actuator was not assembled and thus will not be tested. No problems have been encountered during the acceptance of the scram mechanism. Three of the unaccepted limit switches will require rework or some other disposition due to the presence of hairline cracks in the insulation. The other three are awaiting acceptance tests.

The position sensor and its demodulator are acceptance tested first as individual units and then as a combined system. Since there are twelve position sensors, but only nine demodulators, the plan is to test six units for the S8DS system as matched sets. The remaining six position sensors will be tested in sets of three with the three demodulators. The last set along with the demodulator will be used for the half-reflector system design verification testing. The other three position sensors will be retained as unused spares and can be matched to any demodulator when required. The acceptance testing of the position sensor and demodulator is now in progress.

2. Reactor Control System Design Verification Test

An assembly of half of the reflector system is being prepared for a design verification testing of the reactor control and scram system. The three control drums and their drive components will be the duplicate of the S8DS system. These components include the stationary beryllium reflector half, beryllium control drums with shims, control-drum bearings, scram mechanism, actuator, limit switches, position sensor, and the cable harness. This assembly will be fitted around a simulated core vessel which is equipped with heaters to simulate the reactor core temperature. A nonoperating stationary reflector half and control drum will also be installed to simulate the remainder of the reactor.

This test assembly will be installed in the vacuum chamber and electrical connections to the controller, precision position sensor, and the position sensor demodulator as well as to data acquisition system and test controls will be completed. Complete operational checkout of the control system will be made in vacuum both at room ambient and simulated S8DS design operational temperatures to demonstrate system performance and to establish reference. Subsequently, the control system will be operated under various operational modes for approximately 1000 hours. Tests will include repeated simulation of reactor startup, normal operation, and shutdowns. Operational checkout similar to the initial one will be conducted at the end of the test period. The results of this design verification test will establish the adequacy of the S8DS reactor control system.

The progress of this test is being paced by the control drum bearings. The assembly procedure has been revised to permit maximum subassembly, pre-assembly, and prefit to reduce the final assembly time span.

The assembly of the vacuum facility and the test control and data acquisition systems are continuing and should be completed well ahead of the requirement data.

3. S8DS Reactor System Assembly and Acceptance

The progress of the S8DS reactor assembly is also paced by the control drum bearings. Replanning similar to that described for the half-reflector assembly was initiated.

F. TEST SUPPORT EQUIPMENT

The NaK pressure transducers and the immersion thermowells have been installed in the reactor heat transfer system at the ground prototype test facility. Final x-ray and leak testing is being completed. Checkout of instrumentation and process system controls has been initiated.

G. FACILITY

The fabrication of the uninterruptable power supply was complete and the unit is presently undergoing acceptance testing. The plans for the power supply building and installation were completed.