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SUMMARY OF SNAP 8 DEVELOPMENTAL
REACTOR (S8DR) OPERATIONS

AEC Research and Development Report

Atomics International Division
Rockwell International

P.O. Box 309
Canoga Park, California 91304

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REACTOR (S8DR) OPERATIONS

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H. D. MAY

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ABSTRACT

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

I. INTRODUCTION AND SUMMARY

The Systems for Nuclear Auxiliary Power (SNAP) reactor program is directed toward the development of lightweight, highly reliable nuclear reactors which will serve as heat sources for the production of electrical energy in the environment of outer space. Other potential uses include remote terrestrial and undersea bases, and other applications where it is desirable and/or necessary to provide a highly reliable system which has the capability of operating for long lifetimes ($>10,000$ hr) without refueling.

The SNAP 8 Developmental Reactor (S8DR) was the second generation SNAP 8 reactor design to be constructed and tested. The first generation SNAP 8 Experimental Reactor (S8ER) operated for over 10,000 hr at high temperature in an inert atmosphere. Several other SNAP reactor designs also were tested. The SNAP 10A reactors were tested in a vacuum environment; S10FS-3 operated for 10,000 hr in a ground test, and S10FS-4 operated in space for 43 days. In addition, the SNAP 2 Experimental Reactor (SER) and the SNAP 2 Developmental Reactor (S2DR) were extensively tested at high temperature in an inert atmosphere, but at lower powers than the SNAP 8 tests. This broad experience served as the technological base for the development of the S8DR design.

These reactors were fueled with a hydrided, zirconium-enriched uranium alloy, and cooled with eutectic sodium-potassium alloy (NaK-78). All were controlled by movable beryllium neutron reflectors which regulate the neutron leakage from the core. The trend of the development work has been toward the production of higher power reactors which will operate at higher coolant-outlet temperatures to provide better power-conversion efficiency. The SNAP 8 family of reactors is designed to operate at powers in excess of 300 kwt with average coolant temperatures of 1200°F . By contrast, the SNAP 10A family of reactors operated at power levels of less than 50 kwt and average coolant temperatures of about 1000°F .

This report describes the S8DR testing program and summarizes the test results. An operating summary is shown in Table 1; a summary of experimental results is shown in Table 2. The results of the post-test examination are reported in Reference 1.

TABLE 1

S8DR OPERATING SUMMARY

Time at critical (hr)	7533
Time at power, ≥ 300 kwt (hr)	7023
Time at 1000 kwt (hr)	429
Time at nominal 600 kwt (hr)	6688
at 1300° F	1670
at $\geq 1200^\circ$ F	5575
at $\geq 1100^\circ$ F	6688
Total Energy release (kwht)	4.35×10^6

TABLE 2
SUMMARY OF EXPERIMENTAL RESULTS

Parameter	Design ⁽²⁾ Prediction	Experimental Results	Post-Mortem Calculation
Total isothermal temperature coefficient of reactivity (¢/° F)			
at 100° F	-0.21	-0.17	-
at 1200° F	-0.23	-0.19	-
Power coefficient of reactivity for constant coolant tem- perature (¢/kwt)	-0.03	-0.02	-
Reactivity to 600-kw power and 1200° F core average temperature (\$)	-3.00	-2.10	-
NaK reactivity worth (\$)	+0.25	+0.18	-
Clean wet excess reactivity, cold (\$)	14.40	13.35	-
Total control drum worth (\$)	21.06	18.61	-
Reactivity effects at 600 kw			
Hydrogen redistribution (\$)	-0.29	*	-
Xenon buildup (\$)	-1.05	†	-1.00
Initial samarium burn- out (¢/day)	1.14	*	1.38
Fission product buildup and uranium depletion (¢/day)	-0.34	*	-0.69

*Not measured; calculated values are required to correct reactivity data.

†Consistent; calculated value was successful in correcting reactivity data for xenon transients.

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Figure 1. Partial Fuel Array in Reactor Core Vessel

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II. TEST SYSTEM DESCRIPTION

A. REACTOR

SNAP 8 is a highly enriched, uranium-fueled, hydrogen-moderated, beryllium-reflected reactor. The S8DR reactor design was an extension of the basic SNAP 8 technology developed during the production and testing of the S8ER.⁽³⁾

The reactor core contained 211 fuel elements supported and held laterally by grid plates at the top and bottom. Fuel elements had a diameter of 0.529 in. and an active fuel length of 16.825 in. Spaces between the flats of the hexagonal fuel array were filled by 30, Type 316 SS clad BeO internal reflectors, and 12, solid Type 316 SS filler rods. Fuel element end pins and side reflector end pins engaged holes in the grid plates, and the array was contained within a 0.091-in. - thick, 9.244-in. -diameter Type 316 SS core vessel (Figure 1).

A flow distributor and a flow baffle plate were located below the lower grid plate. The flow distributor was a basket-shaped piece with a wedge facing the coolant nozzle and 64 holes around the side to distribute the flow evenly to the baffle plate. The baffle plate was a disk with 73 holes sized and shaped to provide the desired flow distribution to the core.

Coolant entered the lower plenum, flowed through the holes in the flow distributor, on through the baffle plate holes and into the 420 tricuspid coolant channels between the fuel elements. Coolant then flowed upward through the coolant channels and exited through holes in the upper grid plate into the upper plenum. In-core thermocouples monitored 27 selected upper grid plate flow channels; these thermocouples were held in place by a bridge which positioned the junctions at the centers of the coolant passages just over the top surface of the upper grid plate. The coolant then mixed in the upper plenum and exited through the reactor outlet pipe.

Neutron leakage from the core was reduced by the external beryllium reflector, which was a hollow cylinder 18.5 in. high and nominally 5-1/8 in. thick (Figure 2). The external reflector was divided axially into two halves; each half consisted of a stationary reflector and three control drums with drives, bearings and support brackets. The reflector assemblies were attached to and

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SIDE VIEW

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Figure 2. Reflector and Control Drums for SNAP 8 Developmental Reactor

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supported by brackets which were an integral part of the reactor core vessel. Latching mechanisms were provided at the upper end of the assembly to lock the reflector halves together. Reactivity control was achieved by rotating the six control drums over the range from 0 degrees (full in) and 110 degrees (defined as full out). Each drum was driven in and out by stepper motors. Scram springs were initially wound and "cocked" by energizing an electromagnetic clutch. A scram signal released all six clutches and allowed the drums to snap to the full-out position and shut down the reactor. The installed reflector assembly is shown in Figure 3.

Long-term reactivity losses were compensated in part by burnable samarium prepoison, in the form of Sm_2O_3 , blended in the ceramic H_2 diffusion barrier on the inner surface of the fuel cladding.

B. TEST SYSTEMS

Auxiliary systems used in ground testing S8DR included the primary heat transfer loop, intermediate heat exchanger (IHX), secondary heat transfer loop, and neutron shield test system.

1. Primary Loop

The primary loop consisted of the NaK coolant, the reactor core vessel, the primary surge tank, the IHX, the primary system pump, a parasitic load heater, and associated instrumentation and piping (Figure 4). NaK was pumped through the coolant passages in the core, where it absorbed thermal energy. The hot NaK then flowed through the surge tank to the IHX where the thermal energy was transferred to the coolant in the secondary heat transfer loop. The pump then received the cooled NaK and pumped it back to the reactor.

The primary surge tank was mounted at the high point of the primary system. NaK entered the surge tank through a normally submerged side nozzle and exited through a bottom nozzle fitted with a vortex breaker. An Inconel "J" tube provided remote level measurement. The primary pressure relief system consisted of a rupture disk and pressure relief valve acting through a locked-open gate valve to a knockout drum (with a demister to separate entrained NaK from vent gas) and holdup tank. Overpressurization through the helium supply station was prevented by automatic closure of the helium valve on high pressure.

Figure 3. Installed Reflector Assembly

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Figure 4. P&I Diagram - Nuclear Shakedown Heat Transfer System

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The primary cover gas sampling system was an auxiliary gas loop which allowed sampling for Xe^{135} , Xe^{133} and other fission gases. The surge tank helium cover gas was drawn from the surge tank through a NaK trap, gas blower and sample chamber, then returned to the surge tank. Cover gas samples were taken periodically by isolating, removing and replacing the sample chamber. The gas was also analyzed periodically with a multichannel gamma spectrometer while the gas was circulating. Results of the gamma spectrometric analyses and chemical analyses of the gas samples are summarized in Section IV. F.

The IHX was a counter-flow, spiral tube-and-shell unit with a design rating of 600 kw at a coolant flow of 49,000 lb/hr on each side. Primary NaK flowed through the tube side and secondary NaK flowed through the shell side. The heat exchanger was Type 316 SS all-welded construction, designed for 1450°F at 100 psig.

The primary NaK pump was a three-phase linear induction electromagnetic pump with a coolant flow capacity of 140 gpm with a 40-psi head. The pump was vertically mounted, with coolant flow in the upward direction. Flow was controlled by adjusting power input to the stator coils through an auto-transformer. Stator coils were cooled by a blower which circulated ambient vault atmosphere over the coils.

The parasitic load heater was a clamshell design with a nichrome wire heating element of 30-kwt capacity. The heater was used to supplement the secondary heater during nonnuclear thermal testing, and to make small adjustments for fine control of reactor inlet temperature.

2. Secondary Loop

The secondary loop consisted of the secondary NaK coolant, the IHX, the airblast heat exchanger (ABHX), the secondary heater, the secondary surge tank, the secondary system pump, and associated instrumentation and piping (see Figure 4). NaK was pumped through the shell of the IHX, where it absorbed thermal energy from the primary NaK. The hot NaK then flowed through the coils of the ABHX, where excess heat was dumped to the environment. Then the cooled NaK passed through the secondary surge tank and returned to the pump to complete the cycle. When the reactor was not supplying heat, as for

example after a reactor scram, the NaK loops could be heated with the secondary heater by diverting secondary flow from the ABHX to the heater. Flow could be divided between the ABHX and heater in any proportion, but a small "trickle" flow was normally maintained at all times, to keep the unused component hot and to prevent cold trapping of impurities in the coolant.

The ABHX consisted of two units connected in parallel, only one of which was required for operations up to 600 kwt. The units were slightly different in design, but each was equipped with a variable speed blower which blew outside air up through a tube bundle through which the hot secondary NaK was flowing. Manually adjusted louvers over the top of the tube bundle provided additional control flexibility.

The secondary heater was an in-line immersion type unit designed to deliver 150 kwt with 140-gpm flow. Power was supplied through a variable output transformer which was interlocked with the heater inlet valve and secondary NaK flowmeter to prevent operation without sufficient flow.

The secondary surge tank was slightly more than twice as large as the primary surge tank. Separate nozzles were provided for NaK inlet from the ABHX and from the secondary heater. NaK exited from the tank through a bottom nozzle fitted with a vortex breaker. An Inconel "J" tube provided remote level measurement. Over-pressure protection was provided by a rupture disk which relieved to the building exhaust stack.

The secondary NaK pump was identical in operation and design to the primary pump. However, pumping direction was down instead of up.

3. Neutron Shield Test

A prototype SNAP 8 instrument-rated shadow shield was mounted directly underneath the reactor. The purpose of the shield was to reduce the neutron and gamma flux at flight system instrumentation and sensitive component locations.

The shield used for the S8DR ground test was modified to accommodate test conditions. Lithium enriched in the Li-7 isotope was used in the shield to reduce the heat generation from the (n, α) reaction with Li-6. To further offset the added heat load due to ground test conditions, the distance between reactor and shield was increased slightly.

The shield case was modified by connecting a pipe to be used for measuring and adjusting shield pressure, and for routing ground test instrumentation leads. The entire shield was encased in a confinement vessel for secondary containment in the event of a leak.

The objective of the shield test was to provide sufficient instrumentation to determine the behavior of the shield during the test, and to obtain operating data for use in the design of future shields.

C. FACILITY

The facility in which S8DR was tested is located in the Burro Flats area of the Rockwell International Nuclear Field Laboratory at Santa Susana, California (Figure 5). The facility contained equipment to control test vault temperature, pressure, and oxygen concentration within allowable limits, and supply the necessary shielding, containment, industrial services and auxiliary functions in support of the reactor and test systems.

The reactor was installed within a vacuum chamber in a shielded test pit below the test vault floor level (Figure 6). A vacuum of 10^{-5} torr, or better, was established and maintained within the chamber by two mechanical fore pumps and two oil diffusion pumps located in an equipment room adjacent to the test vault. The vacuum chamber was pumped through a 5-ft-diameter, 63-ft-long pumping duct, which penetrated the vault west shield wall, made four right angle turns and rose through a shielded "chimney" in the vacuum equipment room. The room beneath the vacuum equipment room was filled with sand to provide the necessary radiation shielding.

All radioactive primary NaK piping and equipment was located within the sealed test vault. All nonradioactive secondary coolant loop piping and components, except for the secondary drain system and piping from the IHX, were located in accessible areas at grade level.

Normal electrical equipment received 480-v service directly from the electrical substation. Emergency equipment, which cannot be left without power for more than a few minutes, was automatically supplied from a 200-kw, natural-gas-powered emergency generator. On loss of electrical power, the emergency generator was to start and pick up the load through an automatic transfer switch within 13 sec.

Figure 5. S8DR Test Facility

Figure 6. SNAP 8 Developmental Reactor in Test Vault

Critical test and facility equipment, which could not be lost even momentarily without causing a scram or other test interruption, was supplied through an uninterruptible power supply (UPS). A 600-hp synchronous motor drove the 400-kw UPS generator, which was also coupled to a diesel generator through an electromagnetic clutch. On loss of electrical power to drive the motor, a 9-1/2-ton flywheel at 1750 rpm maintained the motor-generator at 1200-rpm synchronous speed through an eddy-current coupling, while the diesel generator started and came up to speed. When the diesel engine reached synchronous speed, the electromagnetic clutch automatically energized and picked up the motor-generator load, so that UPS buses would never lose power. After site power returned, the UPS was returned to the standby mode manually.

III. OPERATION HISTORY

A. PREOPERATIONAL TESTING

Significant reactor historical events are shown in Figure 7. Fuel loading was initiated June 18, 1968, with the core vessel and reflector assembly mounted on a support stand on the floor of the test vault. The core was initially loaded with a full complement of stainless steel dummy elements. A loading step consisted of tilting away the reflector halves, removing the 5-Ci neutron source and reactor top lid, and replacing a specified group of dummy elements with fuel elements. The lid and source were then reinstalled, reflectors returned, and k_{eff} was measured with all control drums full in.

Criticality was achieved on June 22, 1968, with 169 fuel elements loaded and 42 stainless steel dummies. Fuel loading continued until 196 elements were loaded, then loading was interrupted while control drums were calibrated. Fuel loading was then continued until the full complement of 211 elements was loaded. Then drum calibrations were repeated (Section IV. A).

Operations were delayed several months to find and repair leaks in the vault liner, and to await delivery of replacement NaK pump throats for primary and secondary systems.

In late October, the reactor was welded into the primary system, the heat transfer systems were checked out, and in late November the reflector assembly was installed and dry critical testing in the vacuum chamber was completed. The primary system was loaded with reactor grade NaK, wet critical testing was completed, and in mid-December, half-rate automatic reactor startup simulations were initiated. Testing was interrupted to perform routine maintenance and repairs on auxiliary equipment. Then in early January 1969, full rate automatic reactor startup simulations were performed.

B. POWER OPERATION

The initial approach to full 600-kwt design power was initiated on January 17, 1969, following the automatic startup tests. Reactor power was brought to 150 kwt (one-fourth of design power), with primary and secondary NaK loops circulating at full design flow. Flows were then reduced gradually to approximately one-fourth of design flow, to achieve 1100° F inlet, 1300° F outlet reactor

Figure 7. S8DR Operations and Events History

design temperatures. All system and facility process instrumentation was monitored for off-normal indications, and flow and reactivity oscillation tests were conducted (Section IV. D).

After > 8 hours, flows were gradually increased to approximately one-half of design while reactor power was programmed to 300 kwt to maintain design temperatures. System and facility instrumentation was again monitored for off-normal indications, and flow and reactivity oscillations were repeated.

After another > 8 hr period, flows were gradually increased to approximately three-fourths of design while reactor power was raised to 450 kwt and instruments were monitored. Flow and reactivity oscillation tests were conducted as before.

Following a > 8-hr hold, flows were gradually increased to full design 48,800 lb/hr and reactor power was raised to full 600-kwt design power. Monitoring of instrumentation, and flow and reactivity oscillation tests were repeated. Samples of primary NaK and primary surge tank cover gas were obtained (Section IV. F).

Again, after > 8 hr, reactor power was slowly reduced to 150 kwt while holding the inlet temperature at 1100° F with full flow. On January 19 and 20, perturbations were put into the system, with the reactor level at various steady-state powers, and power and temperature were allowed to seek equilibrium. The purpose of these tests was to investigate the response of the reactor to abrupt changes in secondary system heat removal, and to assist in planning mercury injection simulation tests (Section IV. B).

On January 23, a series of mercury injection simulations was initiated and, at the conclusion of the final simulation on January 25, the 500-hr endurance test was initiated at 600 kwt, 1300° F reactor outlet, and 200° F reactor ΔT . The endurance test was completed on February 24 with two short interruptions due to scrams (see Appendix A).

On March 20, the 9000-hr endurance run was initiated at 600 kwt, 1300° F reactor outlet, and 200° F reactor ΔT . The run continued through April 10 when reactor power was increased to the alternate 1-Mwt design power, with 1150° F outlet temperature and 270° F reactor ΔT .

On April 28, the reactor was returned to the 600 kwt, 1300° F outlet, 200° F ΔT condition and the 9000-hr endurance run was resumed. On June 1, the endurance run was interrupted by a loss-of-flow scram (see Appendix A). Because of the high reactivity loss rate, it was decided to continue the endurance run at a lower temperature. The run was continued on June 5 at 600 kwt, 1200° F outlet, and 200° F ΔT . This endurance run was terminated prematurely because of higher than predicted reactivity loss rate resulting from failed hydrogen barrier and fuel cladding. The endurance runs are described in Section IV. C, and results of monitoring for fission products are given in Section IV. F. Shutdown from the final endurance run was initiated on November 20, when reactor power was reduced to 300 kwt while holding at 1200° F outlet and 200° F ΔT . After 11 days, outlet temperature was reduced to 1140° F, power was returned to 600 kwt while holding 200° F ΔT , and these conditions were maintained for 7 days. Reactor outlet temperature was then increased to 1200° F for 7 days, and final reactor shutdown was then initiated.

Shutdown was accomplished by reducing outlet temperature to 1000° F while holding at 600 kwt and 200° F ΔT , then reducing power to collapse outlet temperature to 800° F while holding inlet temperature at 800° F. The reactor was held critical at 800° F to follow the xenon transient, and on December 21, temperatures were reduced to 440° F to monitor reactor decay power level.

On January 6, system temperatures were raised to 750° F and on January 9 the primary system was drained. Reactor in-core thermocouple temperature dropped steadily and leveled at 180 to 200° F.

IV. EXPERIMENTAL RESULTS

A. FUEL LOADING AND CRITICAL EXPERIMENTS

1. Reactor Fuel Loading

Fuel loading was performed with the reactor located in the test vault. The reactor vessel rested in a structure which located the bottom of the vessel 8 in. from the vault floor. The entire core was initially filled with stainless steel dummy elements, which were replaced with the real fuel elements as loading proceeded. Fuel loading of the reactor was begun on June 18, 1968, and was completed on June 28, 1968.

Reactor criticality was achieved at 0126 hours on June 22, 1968, with 169 fuel elements and 42 stainless steel dummies in the core. Extrapolation of the inverse countrate ($1/C$) plots from the 164-element loading gave a critical fuel loading of 164.5 fuel elements. The fully loaded reactor was critical with control drums Nos. 1, 3, and 5 full out, Nos. 2 and 4 full in, and No. 6 at 56.44 degrees. Ambient temperature was 92° F. The measurement was made with a 1 Ci Pu-Be neutron source located 2 ft above the top of the reactor core. From the last several fuel loading step critical positions, the average worth of an element in the outside ring was estimated to be 19¢.

2. Control Drum Calibrations

Typical control drum calibration curves are shown in Figure 8 to 11. Worth curves for control drums Nos. 1 and 6 had a slightly different shape from that of the other four worth curves. Drums Nos. 1 and 6 were adjacent to each other on the side of the reactor where the control system electrical harness connects with the reactor. Figure 12 is a graph of total drum worths as a function of neighbor drum position. The total excess reactivity was too great to allow complete calibration of a drum with neighbors full in, so the total values were extrapolated from partial calibrations according to the curve shape from calibrations with neighbor drums in intermediate positions. The average drum worth with neighbors full in was then determined to be \$4.43. The worth of a drum facing a row of steel dummy elements was found to be 62% of the worth of a drum facing fuel elements.

Figure 8. Control Drum No. 1 Worth Curve

Figure 9. Control Drum No. 1 Differential Worth

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Figure 10. Control Drum No. 5
Worth Curve

6531-6105

Figure 11. Drum No. 5
Differential Worth

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Figure 12. Drum Worths as a Function of Neighbor Drum Positions
(Dry Critical Tests)

After the reactor was put into place in the vacuum vessel and the NaK coolant was loaded, another series of control drum calibrations was performed. The resulting worths were 12 to 14% lower than those from the dry critical calibrations which had been done on the test vault floor. Calibrations were performed for startup control drums Nos. 2, 4, and 6 with neighbor drums out and for control drums Nos. 1, 3 and 5 with neighbor drums at 68.14 degrees. A later series of calibrations was performed, which included calibrations with neighbor drums at 35.12 degrees. This later series was somewhat unsatisfactory because of poor neutron chamber position and because of photo-neutron source effect from the automatic startup tests to power. Calibration curves for drums Nos. 1, 3 and 5, with neighbor drums at 35.12 degrees were obtained from the last series by normalizing the worth results to the data of the test series performed prior to the automatic startup tests. The drum worth data was extrapolated to obtain an average worth of \$3.80 for a drum with neighbor drums in.

A final series of drum calibrations was performed after final reactor shutdown. The results had not changed significantly from the beginning of life results. Drum worths for a variety of operating configurations are given in Table 3.

TABLE 3
CONTROL DRUM WORTHS
(NaK in Core)

Control Drum No.	Neighbor Positions (deg)	BOL Worth Result (¢)	EOL Worth Result (¢)
1	68.14	299	303
3	68.14	290	287
5	68.14	321	322
1	35.12	353	358
3	35.12	347	349
5	35.12	376	372
1	out	245	
2	out	248	
4	out	234	
6	out	240

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Figure 13. S8DR Isothermal Temperature Coefficient of Reactivity

3. Critical Tests

The reactor, as installed in the vacuum vessel and with coolant drained, was taken critical at 0442 hours on November 29, 1968. The critical position was with control drums Nos. 1, 3 and 5 full out, control drums Nos. 2 and 4 full in, and control drum No. 6, at 67.36 degrees. Reactor temperature was 62° F. Shutdown margin for this condition was \$5.76. Excess reactivity available was \$12.85. Because of the required extrapolation to obtain a value of drum worths with neighbors full in, the excess reactivity value is assigned a possible error of $\pm \$1.00$.

Comparison of critical configurations for the reactor in the vacuum vessel and out on the test vault floor show a reactivity worth contribution of 32¢ for the vacuum vessel pit. NaK coolant worth was found to be 18¢.

Data for the reactor temperature coefficient and the operating temperature reactivity defect are shown in Figures 13 and 14. The temperature coefficient ranged from $-0.17\text{¢}/^{\circ}\text{F}$ at 200° F, to $-0.19\text{¢}/^{\circ}\text{F}$ at 1300° F.

B. AUTOMATIC STARTUP TESTS

1. Half-Rate Startups (Pre-Mercury Injection)

The first automatic startup test, denoted by Run A.1 in Table 4, was initiated on December 13, 1968, with system temperature of 200° F and 20% coolant flow. Reactivity insertion rate was 0.00255 \$/min, or approximately one-half of the analytically determined desired startup rate.

Results of this first run were obscured by malfunctioning of the automatic controller. However, the first power peak occurred much lower than the predicted 46 kwt, based on the predicted fuel-to-coolant heat transfer coefficient (UA) shown in Figure 15. Run A.1.a was then performed to verify the low value of UA implied by Run A.1. Sufficient reactivity was inserted to put the reactor on a period of approximately 100 sec, then temperature feedback was allowed to turn the power rise over. Power was then held level at one-half the peak power, or 3 kwt, while reactivity feedback from the fuel temperature rise was compensated by drum insertions. The time constant measured during this test was approximately 27 min, and this long-time constant was another indication of

Figure 14. S8DR Temperature Defect of Reactivity

TABLE 4

PRE-MERCURY INJECTION STARTUPS

Conditions	Half-Rate Startups								
	A.1	A.1.a	A.2	A.3.a	A.3.b	A.3.c	A.3.d	A.4	A.5
Date performed	12-13-68	12-13-68	12-17-68	12-18-68	12-19-68	12-19-68	12-19-68	12-20-68	1-8-69
Time at 50¢ subcritical	0612:35	1307	2006	-	-	-	-	-	1130
Control under which insertions made	Auto	Manual	Manual	Manual	Manual	Manual	Manual	-	Manual
Initial temperature (°F)	200	209	200	907	950	1050	1200	400	200
Initial insertion rate (sec/step)	240	-	120	-	-	-	-	-	120
Final insertion rate (sec/step)	-	-	240	-	-	-	-	-	240, 180
Remarks		Objective was a period of ~100 sec						Primary flow was oscillated with reactor power at 20 kw	Reached 1300°F first dead-band control demonstration

Conditions	Full-Rate Startups				
	B.1	B.2.a	B.3	B.4.a	B.5
Date performed	1-10-69	1-11-69	1-13-69	1-14-69	1-15-69
Time at 50¢ subcritical	1951.6	1918.2	0034	1133	1101
Control under which insertions made	Auto	Auto	Auto	Auto	Auto
Initial temperature (°F)	200	200	200	200	200
Initial insertion rate (sec/step)	130	130	110	110	110
Final insertion rate (sec/step)					
Remarks	Secondary flow ramped from 20% to 100% when power reached 30 kw.

Figure 15. S8DR Core Fuel-to-Coolant
Heat Transfer Coefficient, UA

low fuel UA. The "predicted" curve of Figure 15 was derived from the "theoretical" curve, coupled with the anticipation of a residual hydrogen pressure in the gas gap between fuel and cladding at low temperature.

In an attempt to learn more about UA, additional transients and automatic* startups were conducted over a temperature range of 400° to 1200° F. These were Runs A.3.a, A.3.b, A.3.c, A.3.d, and A.4, shown in Table 4. Analysis of the data from these runs resulted in the "estimated" curve shown in Figure 15. The close agreement between theoretical and estimated is an indication that essentially all gaseous hydrogen is lost from the gas gap at lower temperatures.

*Due to problems with the automatic controller, automatic drum insertions were simulated manually by the reactor operators.

2. Full-Rate Startups (Pre-Mercury Injection)

Five automatic full-rate startups were performed during the period January 10 to January 15, 1969. Initial conditions for each startup are listed below:

- 1) Primary and secondary loop coolant flow at 20% rated (9,760 lb/hr) and 200° F isothermal temperature.
- 2) Trickle flow through the ABHX; secondary loop heater off.
- 3) Initial reactor power over the range 0.1 to 1.5 w.

All startups were initiated by ramping drums alternately (first the startup drums then the control drums) at a maximum drive rate of 2-1/2 sec/step or 2.08 degrees/min/drum, until startup drums were banked at 68 degrees and control drums were banked at approximately the 50¢ subcritical position. From this configuration the three control drums were stepped in one at a time at a reduced rate. For the first two startups, the automatic controller was set to step a drum 1 step/130 sec, and drum insertions continued until the 1300° F (+20°, -10°) deadband was reached. The last three startups were the same, except that the automatic controller stepping interval was 110 sec from the 50¢ subcritical condition.

These full-rate startups verified that the SNAP 8 reactor can be automatically brought to power over an 8-decade power range from shutdown, in a well controlled manner, without excessive power overshoot on entry into the useful power range, and without excessive reactor fuel or coolant temperature transients.

A detailed description and analysis of these tests is presented in Reference 4.

3. Mercury Injection Simulations

The SNAP 8 Power Conversion System (PCS) startup sequence is shown in Figure 16. The initial part of the sequence up to time zero is the nuclear system startup, which was simulated with full-rate startups and is discussed above.

In late January 1969, a series of mercury injection simulations was performed, as shown in Table 5. The "B" and "C" series were limited simulations for the purpose of verifying system response. The final five simulations, the

Figure 16. SNAP 8 PCS Startup Sequence

"D" series, were full simulations performed to duplicate the proposed mercury injection phase of the SNAP 8 startup sequence. A detailed description and analysis of the full simulations is presented in Reference 4.

The mercury injection simulations performed with S8DR verified the capability of the reactor to meet requirements for the proposed automatic startup of the electrical generating system. The tests were restricted by ground test system component stress considerations to approximations of the desired mercury injection simulation. However, these approximations were close enough to predicted flight conditions to indicate conclusively that the SNAP 8 reactor would respond as required in the flight system.

C. REACTIVITY LOSS HISTORY

The reactivity history of the reactor, as determined from the control drum position records of the data logger, is shown in Figure 17. This can be compared to the long-term power and temperature history, shown in Figures 18 and 19. The measured excess reactivity, Figure 17, was corrected for temperature

TABLE 5
MERCURY INJECTION SIMULATIONS

Parameters	B.1	B.1.a	B.1.b	B.2	B.2.a	B.3	B.3.a	C.1	C.2	C.2.a	C.3	D.1, D.2, D.3, D.4, and D.5
Date performed (1969)	1-23	1-23	1-23	1-24	1-24	1-24	1-24	1-25	1-25	1-25	1-25	1-25
Primary flow (lb/hr)												
Initial	24,400	24,400	24,400	24,400	24,400	24,400	24,400	24,400	24,400	24,400	24,400	24,400
Final	24,400	24,400	24,400	24,400	24,400	24,400	24,400	42,200	48,800	48,800	48,800	48,800
Secondary flow (lb/hr)												
Initial	9,760	9,760	9,760	9,760	9,760	9,760	9,760	9,760	9,760	9,760	9,760	9,760
Final	9,760	9,760	9,760	9,760	9,760	9,760	9,760	29,500	47,000	48,800	29,300	48,800
Initial outlet temperature (°F)	1300	1288	1285	1284	1267	1279	1280	1256	1252	1240	1248	1245 to 1275
Initial reactor power (kwt)	170	157	157	166	148	162	165	180	200	190	200	175 to 190
Initial ABHX slug (kwt-hr)	6.2	3.6	6.2	8.9	11.6	10.7	16.0	10.0	10.0	Special run to evalu- ate ABHX effects; not a simula- tion	12.5	10.0
Louver position afterward	Open	Closed	Closed	Closed	Closed	Closed	Closed	Closed	Closed		Closed	Open
Second ABHX heat load setting (kwt)	0	0	0	0	0	0	0	400	40		500	500

† Based on heat removal being proportional to ABHX fan speed; actual cold slug would have been less.

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Figure 17. S8DR Operating History -
Excess Reactivity

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Figure 18. S8DR Operating History - Power

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Figure 19. S8DR Operating History -
Inlet and Outlet Temperature

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and power variations, xenon transients, samarium burnout, fission product buildup, and uranium depletion. This corrected reactivity drift was then attributed to hydrogen loss from the fuel elements. The integral loss is shown in Figure 20. Hydrogen loss rate reactivities are shown in Figure 21. Note the rise in hydrogen loss rate in June and July (Figure 21). This indicates a degradation of the hydrogen barrier on the cladding. A comparison with the fission product measurements in the NaK coolant and in the NaK cover gas shows that the fission product release from the fuel (Section F) began to increase during this same time period. An extrapolation of the excess reactivity to end-of-life is shown in Figure 22.

The temperature coefficients used to correct the reactivity data were chosen so as to give the smoothest correction. The temperature coefficient was broken down into inlet plenum, outlet plenum, fuel, and delayed reflector coefficients. As in the case of the startup simulations (Section B), the outlet plenum coefficient was found to be 0.0. This can be explained as the effect of the coolant bypass at the core edge which kept the core vessel wall close to the inlet temperature clear up to the outlet plenum. As the core vessel temperature expansion controlled the reflector distance from the core, this distance was affected mostly by the inlet temperature. The inlet plenum coefficient which best fit the data was -0.06 to $-0.08\text{¢}/^{\circ}\text{F}$. The fuel temperature coefficient required was -0.07 to $-0.09\text{¢}/^{\circ}\text{F}$. (The fuel temperature coefficient derived from the startup transient was $-0.10\text{¢}/^{\circ}\text{F}$.)⁽⁴⁾ Total fuel and inlet temperature coefficient was -0.14 to $-0.15\text{¢}/^{\circ}\text{F}$. The reflector coefficient was estimated by following the temperature of one reflector thermocouple. The best fit value for the coefficient changed when the control drum configuration changed, so the values were not considered reliable. The values ranged from -0.05 to $+0.10\text{¢}/^{\circ}\text{F}$. A negative value is required to make up the total temperature coefficient measured.

The overall fuel-to-coolant heat transfer coefficient was derived from the power perturbation after each control drum step. The equation used was:

$$\frac{f}{UA} = \frac{-\Delta\rho}{\alpha_f \Delta P} - \frac{1}{2 w C_p}$$

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Figure 20. S8DR Operating History -
Hydrogen Loss

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Figure 21. Hydrogen Loss Rate

70-J13-003-38A

70-J6-003-8A

Figure 22. S8DR Excess Reactivity

where

UA = fuel heat transfer coefficient

f = importance weighting ratio (~ 1.2)

$\Delta\rho$ = reactivity from drum step

α_f = fuel temperature coefficient ($-0.10\text{¢}/^\circ\text{F}$)

ΔP = power change

wC_p = coolant heat capacity ($\text{kw}/^\circ\text{F}$)

The fuel temperature coefficient required for this analysis was taken from the startup simulation results.⁽⁴⁾ The heat transfer coefficient obtained averaged between 4 and 5 $\text{kw}/^\circ\text{F}$ for nominal operating conditions. For the 1000 kw operation, it tended to be higher, about 5 to 6 $\text{kw}/^\circ\text{F}$.

D. NOISE ANALYSIS

The objectives of the flow and reactivity oscillation tests were to measure the reactor transient response and to determine the reactor partial temperature coefficients and the average fuel-to-coolant heat transfer coefficient.

The flow oscillation tests and the reactivity oscillation tests were performed at each power step of the first approach to 600 kwt. These tests were repeated during the extended runs at 600 kwt and 1 Mwt. For each condition, several reactor system parameters such as power, flow, inlet temperature, outlet temperature, and in-core temperatures, were recorded on FM magnetic tape and on a Brush strip chart recorder for further analysis. The system was perturbed during the recordings by: (1) a primary flow driving function, and (2) a reactivity driving function. The perturbations were generated at fixed frequencies and the resulting responses analyzed by time-averaging analysis which determined the average response of a variable over many cycles.

The flow oscillation system was connected through a transformer across one phase of the primary pump. An adjustable load level bank was used to control the magnitude of the flow perturbations and a signal generator switched the load on and off at the desired frequency. The magnitude of the perturbations was held to a maximum of 3% of indicated flow.

The reactivity oscillations were performed by driving a control drum through the automatic controller inputs to the drum actuator. This allowed the drum to be oscillated when the control mode switch was in "automatic" and also allowed the operator to assume manual control merely by switching to "manual." A dual counter was connected to the "drive-in" and "drive-out" relays of drum No. 1 and set to operate at the desired frequency.

The signal-averaged results for the flow oscillation tests at the 150, 300 and 450 kwt approach to power steps were inconclusive. The small number of cycles collected for the lowest oscillation frequency (0.002 Hz) did not allow a good statistically averaged signal to be obtained for this test. Also, the reactor conditions (i. e. , xenon buildup and hydrogen redistribution effects) did not allow these low frequency tests to proceed without making intermediate control drum corrections. During the 0.002-Hz flow oscillation test, the reactor operators would regularly compensate for the effects of a flow step by stepping a control drum a few minutes later. These events made the low frequency data more complex than was intended for the planned analysis.

The intermediate frequency (0.03 Hz) oscillations were simulated with an analog computer model of the reactor. The model included six delayed neutron groups for reactor kinetics and five axial nodes for fuel and for coolant heat transfer. A fuel temperature coefficient of -0.10 to -0.11% / $^{\circ}\text{F}$, and a UA of $4 \text{ Btu}/^{\circ}\text{F}$ were used in the attempt to match the experimental traces. Of particular interest was an apparent rise in power at the decreasing flow step for the 300 and 450 kwt intermediate flow oscillation tests. An attempt to simulate this particular effect with a positive upper plenum temperature coefficient was unsuccessful. Because of the anomalous reactivity events that were occurring during this time, and the reactor operators making manual drum corrections, the time-averaged signals had phase errors and amplitude distortion that would mask any positive effect on power. However, a sustained flow coefficient between steady-state conditions has not been observed.

The power/flow transfer function for the approach-to-power tests is shown in Figure 23 along with values from flow oscillations at 600 kwt.

Additional drum oscillation tests were performed with the reactor at 730 kwt. The upper curve in Figure 24 shows the transfer function obtained in these tests.

6531-6110

Figure 23. Reactor Transfer Function

6531-6109

Figure 24. Results of Drum Oscillation Test at 730 kw

Figure 25. Core Coolant Inlet Profile

Figure 26. Reactor Radial Power Profile

The power transfer function was divided by the zero-power transfer function to find the fuel time constant break point. The results of this manipulation are shown as the lower set of points in Figure 24. The break point occurs at 0.029 Hz, which gives a fuel time constant of 5.5 sec. The fuel time constant equals fuel heat capacity, MC, divided by the fuel heat transfer coefficient, UA. We now divide the fuel heat capacity value of 32 kw-sec/° F by the derived time constant of 5.5 sec to obtain a value of 5.9 kw/° F for UA at the time of the test. The reactor had generated 683 Mwht at the time of this test.

E. CORE TEMPERATURE PROFILE

The radial coolant flow profile in this core was shaped to control the fuel element temperatures. The target profile was selected⁽⁵⁾ to obtain the maximum reactor life before any fuel material would change phase from δ to β because of hydrogen loss. This profile (Figure 25) has a peak-to-average of 1.33. It is between the flow shape to obtain uniform maximum fuel temperatures, which would be a more peaked shape, and the one to provide uniform coolant outlet temperatures, which would be less peaked. The radial power profile has a peak-to-average of 1.28 (Figure 26). Because of cross flow between coolant channels, the initial flow shape at the core inlet tends toward a uniform shape as the coolant flows through the core. Figure 25 shows the experimental flow shape toward mid-core. It has a peak slightly higher than the target flow profile.

During the reactor test, temperatures were to be measured at the outlets of 27 coolant channels (Figure 27). Thermocouple No. 120 was inoperative, but at least partial data were obtained from the other 26 thermocouples. The outlet temperature profile for 600-kw operations is shown in Figures 28 and 29 for beginning and end of the core operational life. These temperatures are averaged over several days since, at beginning of life, some coolant channel temperatures would vary by as much as 50° F. This variation was eventually attributed to fuel element bowing from thermal and hydrogen redistribution forces.⁽⁶⁾ As expected, the outlet temperature dips down toward the core center. It is evident that the core edge was considerably over-cooled at the expense of the mid-radius elements. The outlet temperature measurements during the 1000-kw operation are shown in Figure 30.

Figure 27. S8DR In-Core Thermocouple Map

9-JA22-006-34A

6531-6111

Figure 28. Outlet Temperature Profile - P=600 kw BOL

6531-6112

Figure 29. Outlet Temperature Profile - P=600 kw EOL

6531-6113

Figure 30. Outlet Temperature Profile - P=1000 kw

Figure 31. S8DR Operating and Sampling History - 1969

The large perturbations in the outlet coolant temperature thermocouple measurements gradually disappeared over the first few months of operation. A high coolant outlet temperature can be expected where the three fuel elements surrounding a channel bow together and, thus, reduce the channel area. However, the hydrogen in the fuel slug tends to migrate toward the cold side of the slug. This would expand the material on the cold side of the slug and shrink it on the hot side, opposing the thermal bow force. It has been proposed that after sufficient time for the hydrogen to move, the rehydriding force would bend the fuel element away from the cluster channel. This would increase the channel flow and lower its outlet temperature. This action would be stopped by fracture of the fuel slug from continued bending, or by fuel swelling to decrease the fuel-cladding gap and, thus, the temperature gradients inside the fuel element. The outlet temperature perturbations significantly decreased after April 1969.

F. FISSION PRODUCT MONITORING

A program to periodically sample the primary NaK coolant and the primary helium cover gas was carried on during the reactor operation. The NaK samples were analyzed for traces of soluble fission products and activation products. The cover gas samples were analyzed for gaseous fission products and activation products. In addition, an on-line gamma spectrometer was used to measure the relative activity of Xe^{135} and Ar^{41} in the primary cover gas.

The NaK sample (N) and cover gas sample (CG) dates are shown in Figure 31 along with the reactor power and temperature history. Results of the NaK sample analyses are shown in Tables 6 and 7. The more important fission products are shown in Table 6. Of the three shown, I^{131} and BaLa^{140} are less soluble in the NaK coolant than Cs^{137} and are also more likely to adsorb on the system piping or NaK sample bomb walls. Thus, Cs^{137} is the only reliable source in the NaK for a quantitative measure of fission product release from the fuel elements. Its long half life, 30.2 years, makes it unnecessary to correct for decay during the 1-year reactor operation. The Cs^{137} concentrations in the coolant, converted to fraction of total core inventory, are plotted in Figure 32.

TABLE 6

S8DR NaK SAMPLE ANALYSIS – MAJOR FISSION PRODUCTS

						Major Fission Products												
						¹³¹ I				BaLa ¹⁴⁰				Cs ¹³⁷				
Sample No	Date	Time (hr)	Cumulative Days	Power Level (kw/t)	Total Energy (Mwh)	Sample (μCi/gm)	Total Coolant (Ci)	Calculated Core (Ci)	Fraction of Core Inventory	Sample (μCi/gm)	Total Coolant (Ci)	Calculated Core (Ci)	Fraction of Core Inventory	Chem Sep Sample (μCi/gm)	γ Spect Sample (μCi/gm)	Aver Total Coolant (Ci)	Calculated Core (Ci)	Fraction of Core Inventory
4 N	1 19 69	13 30	19 56	0	24	—	—	—	—	8 × 10 ⁻⁴	2 59 × 10 ⁴	2 02 × 10 ³	1 28 × 10 ⁻⁷	6 × 10 ³	—	1 94 × 10 ⁵	3 42	5 67 × 10 ⁶
5 N	2 14 69	11 30	45 48	600	330	2 5 × 10 ⁻²	8 10 × 10 ⁻³	1 16 × 10 ⁴	6 98 × 10 ⁻⁷	4 0 × 10 ⁻³	1 30 × 10 ³	2 05 × 10 ⁴	6 34 × 10 ⁻⁸	3 7 × 10 ⁴	—	1 20 × 10 ⁻⁴	4 70 × 10 ¹	2 55 × 10 ⁶
6 N	4 2 69	22 00	92 92	600	617	6 4 × 10 ⁻¹	2 07 × 10 ¹	1 13 × 10 ⁴	1 83 × 10 ⁻⁵	8 6 × 10 ⁻¹	2 79 × 10 ¹	2 04 × 10 ⁴	1 37 × 10 ⁻⁵	2 1 × 10 ³	1 9 × 10 ³	6 48 × 10 ⁴	8 80 × 10 ¹	7 36 × 10 ⁶
7 N	4 21 69	13 00	111 54	990	1015	6 4 × 10 ⁻¹	2 07 × 10 ¹	2 20 × 10 ⁴	9 41 × 10 ⁻⁶	1 4 × 10 ⁻¹	4 54 × 10 ²	4 03 × 10 ⁴	1 13 × 10 ⁻⁶	3 6 × 10 ³	3 7 × 10 ³	1 18 × 10 ³	1 45 × 10 ²	8 14 × 10 ⁶
8 N	5 7 69	11 50	127 49	600	1306	1 1 × 10 ⁰	3 56 × 10 ¹	1 95 × 10 ⁴	1 83 × 10 ⁻⁵	3 8 × 10 ⁻¹	1 23 × 10 ¹	4 07 × 10 ⁴	3 02 × 10 ⁻⁶	6 2 × 10 ³	5 9 × 10 ³	1 96 × 10 ³	1 86 × 10 ²	1 05 × 10 ⁵
9 N	5 13 69	9 00	133 38	600	1392	1 2 × 10 ⁻²	3 89 × 10 ⁻³	1 80 × 10 ⁴	2 16 × 10 ⁻⁷	5 3 × 10 ⁻²	1 72 × 10 ²	3 88 × 10 ⁴	4 43 × 10 ⁻⁷	9 1 × 10 ³	6 6 × 10 ³	2 54 × 10 ³	1 99 × 10 ²	1 28 × 10 ⁵
10 N	5 21 69	17 10	141 71	600	1505	2 39 × 10 ⁻¹	7 75 × 10 ⁻²	1 69 × 10 ⁴	4 58 × 10 ⁻⁶	2 64 × 10 ⁻¹	8 55 × 10 ²	3 70 × 10 ⁴	2 31 × 10 ⁻⁶	5 5 × 10 ³	7 4 × 10 ³	2 09 × 10 ³	2 15 × 10 ²	9 73 × 10 ⁶
11 N	5 28 69	20 00	148 83	600	1630	2 0 × 10 ⁻¹	6 48 × 10 ⁻²	1 64 × 10 ⁴	3 95 × 10 ⁻⁶	5 0 × 10 ⁻²	1 62 × 10 ²	3 60 × 10 ⁴	4 50 × 10 ⁻⁷	1 1 × 10 ²	1 68 × 10 ²	4 51 × 10 ³	2 32 × 10 ²	1 95 × 10 ⁵
12 N	6 4 69	17 40	155 74	0	1674	7 3 × 10 ⁻²	2 36 × 10 ⁻²	1 27 × 10 ⁴	1 86 × 10 ⁻⁶	1 1 × 10 ⁻²	3 56 × 10 ³	3 04 × 10 ⁴	1 17 × 10 ⁻⁷	1 92 × 10 ²	1 85 × 10 ⁻²	6 09 × 10 ³	2 39 × 10 ²	2 54 × 10 ⁵
13 N	6 12 69	18 00	163 75	600	1781	8 3 × 10 ⁻¹	2 69 × 10 ¹	1 39 × 10 ⁴	1 93 × 10 ⁻⁵	1 6 × 10 ⁰	5 18 × 10 ¹	3 10 × 10 ⁴	1 67 × 10 ⁻⁵	1 8 × 10 ²	2 8 × 10 ²	7 45 × 10 ³	2 54 × 10 ²	2 93 × 10 ⁵
14 N	6 18 69	14 30	169 60	600	1863	6 6 × 10 ⁻³	2 14 × 10 ⁻³	1 46 × 10 ⁴	1 47 × 10 ⁻⁷	1 6 × 10 ⁻³	5 18 × 10 ⁻⁴	3 17 × 10 ⁴	1 63 × 10 ⁻⁸	—	3 6 × 10 ²	1 17 × 10 ²	2 66 × 10 ²	4 40 × 10 ⁵
15 N	6 25 69	10 15	176 43	600	1958	1 3 × 10 ⁻³	4 21 × 10 ⁻⁴	1 46 × 10 ⁴	2 88 × 10 ⁻⁸	6 1 × 10 ⁻⁴	1 98 × 10 ⁴	3 16 × 10 ⁴	6 27 × 10 ⁻⁹	—	7 8 × 10 ²	2 53 × 10 ²	2 79 × 10 ²	9 07 × 10 ⁵
16 N	7 4 69	14 50	185 62	600	2088	6 6 × 10 ⁻²	2 14 × 10 ⁻²	1 53 × 10 ⁴	1 40 × 10 ⁻⁶	2 3 × 10 ⁻²	7 45 × 10 ³	3 25 × 10 ⁴	2 29 × 10 ⁻⁷	—	1 46 × 10 ¹	4 73 × 10 ²	2 98 × 10 ²	1 59 × 10 ⁴
17 N	7 16 69	12 22	197 51	600	2242	1 6 × 10 ⁻¹	5 19 × 10 ⁻²	1 48 × 10 ⁴	3 50 × 10 ⁻⁶	6 8 × 10 ⁻³	2 20 × 10 ³	3 18 × 10 ⁴	6 93 × 10 ⁻⁸	—	2 2 × 10 ¹	7 13 × 10 ²	3 20 × 10 ²	2 23 × 10 ⁴
18 N	7 24 69	17 45	205 74	600	2354	—	—	—	—	—	—	—	—	1 7 × 10 ¹	1 6 × 10 ¹	5 35 × 10 ²	3 36 × 10 ²	1 59 × 10 ⁴
19 N	7 31 69	14 48	212 62	600	2460	1 54 × 10 ⁻¹	5 00 × 10 ⁻²	1 56 × 10 ⁴	3 20 × 10 ⁻⁶	5 4 × 10 ⁻²	1 75 × 10 ²	3 29 × 10 ⁴	5 32 × 10 ⁻⁷	2 2 × 10 ¹	1 32 × 10 ¹	5 70 × 10 ²	3 51 × 10 ²	1 62 × 10 ⁴
20 N	8 7 69	15 13	219 64	600	2563	8 8 × 10 ⁻¹	2 85 × 10 ⁻¹	1 57 × 10 ⁴	1 82 × 10 ⁻⁵	1 23 × 10 ⁻¹	3 98 × 10 ²	3 32 × 10 ⁴	1 20 × 10 ⁻⁶	2 0 × 10 ¹	1 68 × 10 ¹	5 96 × 10 ²	3 66 × 10 ²	1 63 × 10 ⁴
21 N	8 21 69	11 03	233 46	600	2758	2 7 × 10 ⁻²	8 75 × 10 ⁻³	1 58 × 10 ⁴	5 54 × 10 ⁻⁷	5 7 × 10 ⁻³	1 85 × 10 ³	3 35 × 10 ⁴	5 52 × 10 ⁻⁸	2 0 × 10 ¹	2 04 × 10 ¹	6 54 × 10 ²	3 93 × 10 ²	1 66 × 10 ⁴
22 N	9 4 69	11 09	247 46	600	2965	2 16 × 10 ⁻¹	7 00 × 10 ⁻²	1 58 × 10 ⁴	4 42 × 10 ⁻⁶	1 63 × 10 ⁻²	5 28 × 10 ³	3 36 × 10 ⁴	1 57 × 10 ⁻⁷	2 13 × 10 ¹	2 16 × 10 ¹	6 95 × 10 ²	4 22 × 10 ²	1 65 × 10 ⁴
23 N	9 11 69	10 35	254 44	600	3066	1 8 × 10 ⁻²	5 83 × 10 ⁻³	1 58 × 10 ⁴	3 69 × 10 ⁻⁷	8 5 × 10 ⁻⁴	2 75 × 10 ⁴	3 37 × 10 ⁴	8 16 × 10 ⁻⁹	2 54 × 10 ¹	2 65 × 10 ¹	8 41 × 10 ²	4 37 × 10 ²	1 92 × 10 ⁴
24 N	9 25 69	10 53	268 45	600	3267	3 5 × 10 ⁻²	1 13 × 10 ⁻²	1 58 × 10 ⁴	7 18 × 10 ⁻⁷	2 9 × 10 ⁻²	9 40 × 10 ³	3 37 × 10 ⁴	2 79 × 10 ⁻⁷	2 48 × 10 ¹	2 58 × 10 ¹	8 20 × 10 ²	4 66 × 10 ²	1 76 × 10 ⁴
25 N	10 9 69	10 40	282 44	600	3461	1 5 × 10 ⁻²	4 86 × 10 ⁻³	1 55 × 10 ⁴	3 14 × 10 ⁻⁷	9 7 × 10 ⁻³	3 14 × 10 ³	3 31 × 10 ⁴	9 50 × 10 ⁻⁸	2 47 × 10 ¹	2 88 × 10 ¹	8 67 × 10 ²	4 94 × 10 ²	1 76 × 10 ⁴
26 N	10 23 69	9 00	296 37	600	3664	3 7 × 10 ⁻²	1 20 × 10 ⁻²	1 57 × 10 ⁴	7 64 × 10 ⁻⁷	5 4 × 10 ⁻³	1 75 × 10 ³	3 35 × 10 ⁴	5 23 × 10 ⁻⁸	2 57 × 10 ¹	3 32 × 10 ¹	9 54 × 10 ²	5 23 × 10 ²	1 82 × 10 ⁴
27 N	11 7 69	9 45	311 41	600	3881	6 6 × 10 ⁻¹	2 14 × 10 ⁻¹	1 58 × 10 ⁴	1 35 × 10 ⁻⁵	1 5 × 10 ⁻¹	4 86 × 10 ²	3 36 × 10 ⁴	1 45 × 10 ⁻⁶	3 09 × 10 ¹	3 56 × 10 ¹	1 08 × 10 ¹	5 54 × 10 ²	1 95 × 10 ⁴
28 N	11 25 69	10 37	329 44	300	4097	4 21 × 10 ⁰	1 36 × 10 ⁰	1 29 × 10 ⁴	1 06 × 10 ⁻⁴	7 6 × 10 ⁻¹	2 46 × 10 ¹	2 94 × 10 ⁴	8 37 × 10 ⁻⁶	5 05 × 10 ¹	4 74 × 10 ¹	1 58 × 10 ¹	5 84 × 10 ²	2 71 × 10 ⁴
29 N	12 5 69	10 25	339 43	600	4199	2 09 × 10 ⁰	6 76 × 10 ⁻¹	1 23 × 10 ⁴	5 50 × 10 ⁻⁵	5 8 × 10 ⁻¹	1 88 × 10 ¹	2 74 × 10 ⁴	6 85 × 10 ⁻⁶	3 57 × 10 ¹	4 19 × 10 ¹	1 26 × 10 ¹	5 98 × 10 ²	2 11 × 10 ⁴
30 N	12 11 69	10 54	345 45	600	4284	1 53 × 10 ⁻¹	4 96 × 10 ⁻²	1 37 × 10 ⁴	3 62 × 10 ⁻⁶	2 94 × 10 ⁻²	9 52 × 10 ²	2 92 × 10 ⁴	3 26 × 10 ⁻⁶	4 76 × 10 ¹	4 91 × 10 ¹	1 57 × 10 ¹	6 11 × 10 ²	2 57 × 10 ⁴
31 N	12 16 69	12 54	*	0	4345	8 92 × 10 ⁻¹	2 89 × 10 ⁻¹	1 44 × 10 ⁴	2 00 × 10 ⁻⁶	2 22 × 10 ⁻²	7 20 × 10 ²	3 01 × 10 ⁴	2 39 × 10 ⁻⁶	4 56 × 10 ⁻¹	5 14 × 10 ⁻¹	1 57 × 10 ⁻¹	6 19 × 10 ²	2 54 × 10 ⁴
32 N	12 18 69	13 02	*	0	4345	6 44 × 10 ⁻¹	2 08 × 10 ⁻¹	1 44 × 10 ⁴	1 45 × 10 ⁻⁶	1 40 × 10 ⁻¹	4 54 × 10 ⁻²	3 01 × 10 ⁴	1 51 × 10 ⁻⁶	5 5 × 10 ⁻¹	4 03 × 10 ⁻¹	1 54 × 10 ⁻¹	6 19 × 10 ²	2 50 × 10 ⁴
Coolant drained and system refilled																		
33 N	1 13 70	10 30	*	0	4345	3 42 × 10 ⁻¹	1 11 × 10 ⁻¹	1 44 × 10 ⁴	7 70 × 10 ⁻⁷	4 4 × 10 ⁻²	1 43 × 10 ²	3 01 × 10 ⁴	4 7 × 10 ⁻⁷	—	1 45 × 10 ⁻¹	4 70 × 10 ²	6 19 × 10 ²	7 60 × 10 ⁵

*Used date of shutdown

12 15 69 | 15 00 | 349 63

TABLE 7
S8DR NaK SAMPLE ANALYSIS – FISSION AND ACTIVATION PRODUCTS

Sample No	Date	Time (hr)	Cumulative Days	Power Level (kw/t)	Total Energy (Mwh)	Fission Products ($\mu\text{Ci/gm}$)				Corrosion Products ($\mu\text{Ci/gm}$)					NaK and Impurities ($\mu\text{Ci/gm}$)				
						Sb ¹²⁵	Te ¹³²	Zr ⁹⁵	Nb ⁹⁵	Co ⁶⁰	Cr ⁵¹	Fe ⁵⁹	Mn ⁵⁴	Sb ¹²⁴	Cs ¹³⁴	K ⁴²	Na ²² (n,2n)	Na ²⁴	Rb ⁸⁶
4 N	1 19 69	13 30	19 56	0	24	—	**	†	†	3.0×10^4	3.0×10^3	†	8.0×10^4	—	†	**	9.0×10^5	**	9.0×10^2
5 N	2 14 69	11 30	45 48	600	330	—	9.0×10^2	†	†	1.6×10^4	6.5×10^3	†	4.1×10^4	**	†	**	8.4×10^4	**	5.0×10^1
6 N	4 2 69	22 00	92 92	600	617	†	**	†	†	9.7×10^5	8.4×10^3	†	4.5×10^3	4.0×10^4	†	—	1.7×10^3	$2.1 \times 10^{+3}$	**
7 N	4 21 69	13 00	111 54	990	1015	—	—	—	—	—	—	—	—	—	—	—	—	—	—
8 N	5 7 69	11 50	127 49	600	1306	†	**	†	†	2.3×10^5	1.1×10^2	†	1.8×10^3	1.2×10^3	†	**	3.6×10^3	**	**
9 N	5 13 69	9 00	133 38	600	1392	—	—	—	—	—	—	—	—	—	—	—	—	—	—
10 N	5 21 69	17 10	141 71	600	1505	—	—	—	—	—	—	—	—	—	—	—	—	—	—
11 N	5 28 69	20 00	148 83	600	1630	—	—	—	—	—	—	—	—	—	—	—	—	—	—
12 N	6 4 69	17 40	155 74	0	1674	†	**	†	†	2.6×10^4	$<1.0 \times 10^2$	†	9.0×10^4	3.5×10^4	9.0×10^4	**	5.8×10^3	**	7.8×10^1
13 N	6 12 69	18 00	163 75	600	1781	—	—	—	—	—	—	—	—	—	—	—	—	—	—
14 N	6 18 69	14 30	169 60	600	1863	—	—	—	—	—	—	—	—	—	—	—	—	—	—
15 N	6 25 69	10 15	176 43	600	1958	—	—	—	—	—	—	—	—	—	—	—	—	—	—
16 N	7 4 69	14 50	185 62	600	2088	†	**	†	5.2×10^2	5.0×10^5	†	†	6.0×10^4	1.0×10^4	4.7×10^3	**	7.6×10^3	**	1.06
17 N	7 16 69	12 22	197 51	600	2242	—	—	—	—	—	—	—	—	—	—	—	—	—	—
18 N	7 24 69	17 45	205 74	600	2354	—	—	—	—	—	—	—	—	—	—	—	—	—	—
19 N	7 31 69	14 48	212 62	600	2460	—	—	—	—	—	—	—	—	—	—	—	—	—	—
20 N	8 7 69	15 13	219 64	600	2563	†	**	†	2.2×10^1	$<3.5 \times 10^4$	†	†	$<4.0 \times 10^4$	$<3.3 \times 10^4$	7.5×10^3	**	9.0×10^3	**	1.17
21 N	8 21 69	11 03	233 46	600	2758	—	—	—	—	—	—	—	—	—	—	—	—	—	—
22 N	9 4 69	11 09	247 46	600	2965	†	**	†	8.5×10^2	5.0×10^5	†	†	2.0×10^4	2.0×10^4	9.1×10^3	**	9.5×10^3	**	1.24
23 N	9 11 69	10 35	254 44	600	3066	—	—	—	—	—	—	—	—	—	—	—	—	—	—
24 N	9 25 69	10 53	268 45	600	3267	—	—	—	—	—	—	—	—	—	—	—	—	—	—
25 N	10 9 69	10 40	282 44	600	3461	†	**	†	1.72×10^1	$<1.0 \times 10^4$	†	†	$<2.0 \times 10^4$	$<3.0 \times 10^4$	1.17×10^2	**	1.26×10^2	**	1.17
26 N	10 23 69	9 00	296 37	600	3664	—	—	—	—	—	—	—	—	—	—	—	—	—	—
27 N	11 7 69	9 45	311 41	600	3881	$<3.1 \times 10^3$	**	†	7.54×10^1	$<5.0 \times 10^5$	†	†	$<5.0 \times 10^4$	$<6.0 \times 10^4$	1.71×10^2	**	1.51×10^2	$1.55 \times 10^{+3}$	1.26
28 N	11 25 69	10 37	329 44	300	4097	—	—	—	—	—	—	—	—	—	—	—	—	—	—
29 N	12 5 69	10 25	339 43	600	4199	—	—	—	—	—	—	—	—	—	—	—	—	—	—
30 N	12 11 69	10 54	345 45	600	4284	$<4.0 \times 10^4$	**	†	1.00×10^1	$<4.0 \times 10^5$	†	†	$<4.0 \times 10^5$	$<4.0 \times 10^5$	2.62×10^2	**	1.47^2	**	1.31
31 N	12 16 69	12 54	*	0	4345	—	—	—	—	—	—	—	—	—	—	—	—	—	—
32 N	12 18 69	13 02	*	0	4345	$<3.0 \times 10^3$	**	†	7.68×10^1	$<1.0 \times 10^4$	†	†	$<2.8 \times 10^4$	$<2.5 \times 10^4$	2.16×10^2	**	1.43×10^2	**	1.20
Coolant Drained and System Refilled																			
33 N	1 13 70	10 30	*	0	4345	$<1.0 \times 10^3$	**	†	2.01×10^1	$<1.0 \times 10^4$	†	†	$<1.7 \times 10^4$	$<1.5 \times 10^4$	8.7×10^3	**	3.9×10^3	**	3.5×10^1

*USED DATE OF SHUTDOWN

12 15 69 15 00 349 63

†NOT DETECTED

**DETECTED BUT NOT MEASURED

6531-6115

Figure 32. Fraction of Cs^{137} Core Inventory in Primary Coolant

The release fraction of fission products into the NaK coolant from a fuel element with cracked cladding is calculated to be 0.07 to 0.08%, using a fission fragment recoil model at the fuel surface. The Cs^{137} measurements in NaK samples from S8ER showed a release fraction of about 0.1% from the elements with cracked cladding.

During the post-test examination⁽¹⁾ of the S8DR core, it was found that 72 of the 211 fuel elements had cracked cladding. For the ending Cs^{137} concentration in the coolant of 2.5×10^{-4} of the total core inventory (Table 6), this amounts to a release fraction of 0.073% from the cracked fuel elements.

Results of the primary helium cover gas sample analyses are given in Table 8. The Ar^{39} and Ar^{41} isotopes are high energy (n, p) reactions on K^{39} and K^{41} , respectively.⁽⁷⁾ The Xe^{133} , Xe^{135} , and Kr^{85} isotopes are fission products (shown in Figure 33). The cover gas sample results were affected by the sampling frequency. Note particularly the depletion in concentration during the frequent sampling in April and May (Figure 33). It would be expected that the longer half-life isotopes, Kr^{85} (10.7 years) and Xe^{133} (5.3 days), would be more affected by sample depletion than Xe^{135} (9.2 hr). It appears that the Xe^{133} required about 2 weeks to approach equilibrium after a system perturbation such as a cover gas sample or a reactor power change.

The Xe^{135} , on the other hand, would become depleted if there were a significant holdup between the fuel and the surge tank because of its relatively short half-life. The time for the xenon to escape from the fuel element to the coolant through the cracked cladding is expected to be short (less than 1 hr), particularly since hydrogen is also being lost and should sweep the xenon along. The NaK loop time is about 1 min, and xenon solubility in NaK would allow less than 1% of the xenon in NaK solution.^(8,9) Post-operation hydraulic tests showed that surge tank gas was entrained in the NaK, but the entrainment mixing at the surge tank would also give the NaK and surge tank gas approximately the same composition. Thus, the expected holdup time is of the order of 1 hr. This would not significantly deplete the Xe^{135} activity in the cover gas. The fact that reported Xe^{135} concentrations are always less than the Xe^{133} concentrations may be attributed either to a longer than expected holdup time (12 to 24 hrs) before the gas reaches the surge tank, or to uncertainties in the instrument calibrations and the yields of the γ decay branches measured in the analyses. The

TABLE 8
S8DR PRIMARY COVER GAS ANALYSIS

Gas Sample No	Date 1969	Time (hr)	Cumulative Days	Power Level (kw/t)	Total Energy (Mwh)	Gas Volume (10 ⁶ cc)	Fission Products									Activation Products			
							Kr ⁸⁵			Xe ¹³³			Xe ¹³⁵			Ar ³⁹		Ar ⁴¹	
							Sample (μCi/cc)	Calculated Core (Ci)	Fraction of Core Inventory	Sample (μCi/cc)	Calculated Core (Ci)	Fraction of Core Inventory	Sample (μCi/cc)	Calculated Core (Ci)	Fraction of Core Inventory	Sample (μCi/cc)	Total Gas (Ci/Mwh)	Sample (μCi/cc)	Total Gas (Ci/kwt)
4 CG	2 21	10 00	52 42	630	410	0 071	9.2 x 10 ⁻⁵	8.3 x 10 ⁰	7.9 x 10 ⁻⁷	1.7 x 10 ⁻³	3.05 x 10 ⁻⁴	3.96 x 10 ⁻⁹	—	—	—	7.3 x 10 ⁻³	1.27 x 10 ⁻⁶	—	—
5 CG	4 8	10 30	98 44	730	715	0 071	5.9 x 10 ⁻⁵	1.45 x 10 ⁻¹	2.9 x 10 ⁻⁷	—	—	—	—	—	—	—	—	—	—
6 CG	4 15	9 42	105 40	964	870	0 071	4.6 x 10 ⁻³	1.76 x 10 ⁻¹	1.86 x 10 ⁻⁵	1.0	4.75 x 10 ⁻⁴	1.50 x 10 ⁻⁶	2.1	5.26 x 10 ⁻⁴	2.8 x 10 ⁻⁶	7.8 x 10 ⁻¹	6.4 x 10 ⁻⁵	3.7 x 10 ⁻¹	2.7 x 10 ⁻³
7 CG	4 24	4 10	114 17	980	1077	0 071	1.7 x 10 ⁻⁴	2.18 x 10 ⁻¹	5.5 x 10 ⁻⁷	2.6	5.43 x 10 ⁻⁴	3.39 x 10 ⁻⁶	3.8	5.36 x 10 ⁻⁴	5.0 x 10 ⁻⁶	8.5 x 10 ⁻¹	5.6 x 10 ⁻⁵	2.47 x 10 ⁻¹	1.8 x 10 ⁻³
8 CG	4 28	3 00	118 12	963	1170	0 071	1.8 x 10 ⁻³	2.37 x 10 ⁻¹	5.4 x 10 ⁻⁶	6.0	5.56 x 10 ⁻⁴	7.64 x 10 ⁻⁶	4.2	5.25 x 10 ⁻⁴	5.7 x 10 ⁻⁶	5.8 x 10 ⁻¹	3.52 x 10 ⁻⁵	1.67 x 10 ⁻¹	1.23 x 10 ⁻³
9 CG	5 5	4 00	125 17	600	1275	0 071	1.0 x 10 ⁻³	2.58 x 10 ⁻¹	2.75 x 10 ⁻⁶	4.7	4.35 x 10 ⁻⁴	7.68 x 10 ⁻⁶	2.2	3.28 x 10 ⁻⁴	4.8 x 10 ⁻⁶	5.8 x 10 ⁻¹	3.23 x 10 ⁻⁵	8.5	1.00 x 10 ⁻³
10 CG	5 8	4 00	128 17	600	1317	0 071	8.9 x 10 ⁻⁴	2.68 x 10 ⁻¹	2.36 x 10 ⁻⁶	3.8	4.07 x 10 ⁻⁴	6.63 x 10 ⁻⁶	2.1	3.28 x 10 ⁻⁴	4.6 x 10 ⁻⁶	6.9 x 10 ⁻¹	3.72 x 10 ⁻⁵	8.7	1.03 x 10 ⁻³
11 CG	5 12	4 00	132 17	600	1376	0 071	7.0 x 10 ⁻⁴	2.78 x 10 ⁻¹	1.79 x 10 ⁻⁶	3.4	3.83 x 10 ⁻⁴	6.58 x 10 ⁻⁶	2.4	3.28 x 10 ⁻⁴	5.2 x 10 ⁻⁶	3.9 x 10 ⁻¹	2.01 x 10 ⁻⁵	9.6	1.14 x 10 ⁻³
12 CG	5 15	4 00	135 17	600	1419	0 071	9.3 x 10 ⁻⁴	2.87 x 10 ⁻¹	2.30 x 10 ⁻⁶	3.6	3.72 x 10 ⁻⁴	6.88 x 10 ⁻⁶	3.0	3.28 x 10 ⁻⁴	6.5 x 10 ⁻⁶	4.0 x 10 ⁻¹	2.10 x 10 ⁻⁵	2.41 x 10 ⁻¹	2.83 x 10 ⁻³
13 CG	5 19	4 00	139 17	600	1477	0 071	8.8 x 10 ⁻⁴	2.98 x 10 ⁻¹	2.10 x 10 ⁻⁶	4.0	3.62 x 10 ⁻⁴	7.85 x 10 ⁻⁶	2.9	3.28 x 10 ⁻⁴	6.27 x 10 ⁻⁶	3.7 x 10 ⁻¹	1.78 x 10 ⁻⁵	1.54 x 10 ⁻¹	1.85 x 10 ⁻³
14 CG	5 26	4 10	146 17	606	1577	0 071	1.8 x 10 ⁻³	3.20 x 10 ⁻¹	3.96 x 10 ⁻⁶	6.8	3.54 x 10 ⁻⁴	1.36 x 10 ⁻⁵	5.0	3.28 x 10 ⁻⁴	1.07 x 10 ⁻⁵	6.0 x 10 ⁻¹	2.7 x 10 ⁻⁵	1.60 x 10 ⁻¹	1.87 x 10 ⁻³
15 CG	6 9	4 00	160 17	606	1730	0 099	5.4 x 10 ⁻³	3.51 x 10 ⁻¹	1.51 x 10 ⁻⁵	1.21 x 10 ⁻¹	2.74 x 10 ⁻⁴	4.38 x 10 ⁻⁵	9.6	3.28 x 10 ⁻⁴	2.9 x 10 ⁻⁵	5.5 x 10 ⁻¹	3.1 x 10 ⁻⁵	1.44 x 10 ⁻¹	2.36 x 10 ⁻³
16 CG	7 9	11 53	190 49	0	2157	0 106	2.41 x 10 ⁻²	4.38 x 10 ⁻¹	5.85 x 10 ⁻⁵	3.99 x 10 ⁻¹	3.43 x 10 ⁻⁴	1.23 x 10 ⁻⁴	1.84 x 10 ⁻¹	3.28 x 10 ⁻⁴	6.0 x 10 ⁻⁵	1.66	8.2 x 10 ⁻⁴	1.85 x 10 ⁻¹	4.2 x 10 ⁻³
17 CG	7 17	4 00	198 17	589	2250	0 11	2.21 x 10 ⁻²	4.57 x 10 ⁻¹	5.31 x 10 ⁻⁵	3.99 x 10 ⁻¹	3.28 x 10 ⁻⁴	1.33 x 10 ⁻⁴	1.83 x 10 ⁻¹	3.28 x 10 ⁻⁴	6.15 x 10 ⁻⁵	1.44	7.1 x 10 ⁻⁴	2.01 x 10 ⁻¹	3.73 x 10 ⁻³
18 CG	8 14	4 00	226 17	600	2653	0 12	5.5 x 10 ⁻²	5.38 x 10 ⁻¹	1.23 x 10 ⁻⁴	5.3 x 10 ⁻¹	3.48 x 10 ⁻⁴	1.82 x 10 ⁻⁴	2.2 x 10 ⁻¹	3.28 x 10 ⁻⁴	8.0 x 10 ⁻⁵	4.0	1.81 x 10 ⁻⁴	2.5 x 10 ⁻¹	5.0 x 10 ⁻³
19 CG	9 17	2 00	260 08	600	3147	0 12	7.48 x 10 ⁻²	6.38 x 10 ⁻¹	1.41 x 10 ⁻⁴	6.14 x 10 ⁻¹	3.48 x 10 ⁻⁴	2.12 x 10 ⁻⁴	3.02 x 10 ⁻¹	3.28 x 10 ⁻⁴	1.10 x 10 ⁻⁴	5.0	1.91 x 10 ⁻⁴	1.70 x 10 ⁻¹	3.40 x 10 ⁻³
20 CG	10 15	2 00	288 08	600	3544	0 123	7.75 x 10 ⁻²	7.20 x 10 ⁻¹	1.32 x 10 ⁻⁴	7.60 x 10 ⁻¹	3.44 x 10 ⁻⁴	2.72 x 10 ⁻⁴	2.68 x 10 ⁻¹	3.28 x 10 ⁻⁴	1.01 x 10 ⁻⁴	5.0	1.74 x 10 ⁻⁴	2.23 x 10 ⁻¹	4.57 x 10 ⁻³
21 CG	10 29	4 02	302 17	600	3749	0 123	8.41 x 10 ⁻²	7.60 x 10 ⁻¹	1.36 x 10 ⁻⁴	7.88 x 10 ⁻¹	3.48 x 10 ⁻⁴	2.78 x 10 ⁻⁴	2.69 x 10 ⁻¹	3.28 x 10 ⁻⁴	1.01 x 10 ⁻⁴	5.2	1.71 x 10 ⁻⁴	2.04 x 10 ⁻¹	4.18 x 10 ⁻³
22 CG	11 17	4 00	321 17	600	4011	0 129	7.11 x 10 ⁻²	8.15 x 10 ⁻¹	1.13 x 10 ⁻⁴	6.6 x 10 ⁻¹	3.33 x 10 ⁻⁴	2.56 x 10 ⁻⁴	3.0 x 10 ⁻¹	3.28 x 10 ⁻⁴	1.18 x 10 ⁻⁴	5.8	1.86 x 10 ⁻⁴	—	—
23 CG	11 24	4 00	328 17	300	4088	0 134	5.16 x 10 ⁻²	8.28 x 10 ⁻¹	8.35 x 10 ⁻⁵	6.0 x 10 ⁻¹	2.80 x 10 ⁻⁴	2.88 x 10 ⁻⁴	1.19 x 10 ⁻¹	1.64 x 10 ⁻⁴	9.7 x 10 ⁻⁵	4.4	1.44 x 10 ⁻⁴	7.4	3.3 x 10 ⁻³
24 CG	12 8	4 00	342 17	600	4238	0 130	5.11 x 10 ⁻²	8.58 x 10 ⁻¹	7.76 x 10 ⁻⁵	5.25 x 10 ⁻¹	2.93 x 10 ⁻⁴	2.67 x 10 ⁻⁴	2.52 x 10 ⁻¹	3.28 x 10 ⁻⁴	9.97 x 10 ⁻⁵	6.1	1.9 x 10 ⁻⁴	1.98 x 10 ⁻¹	4.29 x 10 ⁻³
25 CG	12 11	4 00	345 17	600	4281	0 128	5.24 x 10 ⁻²	8.69 x 10 ⁻¹	7.72 x 10 ⁻⁵	6.24 x 10 ⁻¹	3.11 x 10 ⁻⁴	2.56 x 10 ⁻⁴	2.62 x 10 ⁻¹	3.28 x 10 ⁻⁴	1.02 x 10 ⁻⁴	6.0	1.8 x 10 ⁻⁴	1.76 x 10 ⁻¹	3.76 x 10 ⁻³

Figure 33. Fraction of Core Inventory in Cover Gas

6531-6117

Figure 34. On-Line Cover Gas Monitoring Results
(Gamma Spectrometer)

release fraction for the xenons, as observed in the cover gas, was 0.076% for Xe^{133} and 0.034% for Xe^{135} .

Results of Xe^{135} from the on-line γ spectrometer are shown in Figure 34. Although the results are relative, they showed the same rise in fission product activity during the May, June, and July time period as the other sample results did.

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APPENDIX A SCRAM SUMMARY

Date and Time	Cause	Remarks
11-27-68 1058	Spurious period scram, Channel 1	Zero-power testing
11-30-68 1930	Spurious period scram, Channel 4	Zero-power testing
12-03-68 1123	Spurious period scram, Channel 3	Zero-power testing
12-03-68 2130	Spurious period scram, Channel 3	Zero-power testing
12-05-68 0635	Manual scram, loss of power supply for console indicators	Zero-power testing
12-16-68 1850	Spurious period scram, Channel 3	Zero-power testing
12-17-68 0512	Spurious period scram, Channel 3	Zero-power testing
12-21-68 1301	Manual scram to permit power transfer from UPS to Edison	Reactor critical, zero power
1-10-69 1554	Spurious flow scram; flow switch setpoint being adjusted	Reactor critical, zero power
1-11-69 2212	Spurious period scram; Channel 4 fuse blew	Startup Run B-2 aborted, zero power
1-14-69 1015	Spurious period scram; Channel 4 noise	Terminated Startup B-4, zero power
1-17-69 2150	Spurious flow scram; noise feedback from special recording instrumentation	Reactor at 300 kwt, T_{in} decreased $\sim 50^{\circ}F$ in 15 min, T_{avg} decreased $\sim 100^{\circ}F$ immediately due to ΔT collapse, then decreased $50^{\circ}F$ with T_{in} .

Scram Summary (Continued)

Date and Time	Cause	Remarks
2-9-69 1203	Power scram; power perturbation from flow spike caused by high voltage spike on UPS	Maximum power, 720 kw; Minimum period, 35 sec; Maximum system temperature, 1322° F Total down time, 15 hr
2-11-69 1002	Spurious core outlet high temperature scram; suspect trip was caused by noise from flow oscillation relay	Reactor at 600 kwt, 1300° F outlet temperature; total downtime, 22 hr
3-18-69 1343	Loss of flow scram; pen drove down scale during work on flow recorder circuits, tripping pump	Reactor at 32 kwt, 480° F outlet temperature; total downtime, 3 hr
3-20-69 2100	Loss of air blast fan; speed control motor failed and blew fuse in control circuit	Reactor at 600 kwt, 1300° F outlet temperature; total downtime, 3 hr
4-29-69 1523	Spurious core outlet high temperature scram; temperature switches tripped by failed thermocouples	Reactor at 730 kwt, 1250° F outlet temperature; recovery action prevented system temperatures from falling below 990° F; total downtime, 10 hr
6-1-69 2041	Low flow scram caused by blown fuse in primary pump power supply	Reactor at 600 kwt, 1300° F outlet temperature; outlet temperature increased to 1330° F maximum; total downtime, 4-1/2 days
6-24-69 1336	Spurious low flow scram; flow scram switch was accidentally bumped while replacing a temperature switch	Reactor at 600 kwt, 1200° F outlet temperature; total downtime, 10 hr
7-9-69 1113	Low flow scram caused by failure of primary pump powerstat	Reactor at 600 kwt, 1200° F outlet temperature; total downtime, 24 hr

Scram Summary (Continued)

Date and Time	Cause	Remarks
10-2-69 0631	Spurious period scram caused by instrument noise, following failure of power supply to period scram bypass circuitry	Reactor at 600 kwt, 1200° F outlet temperature; total downtime, 12 hr
11-11-69 1332	Low flow scram caused by failure of primary pump powerstat	Reactor at 600 kwt, 1200° F; total downtime, 15 hr

APPENDIX B MAINTENANCE SUMMARY

S8DR maintenance and modifications were performed in accordance with a Facility Availability Plan. The plan identified critical components, gave specific procedures for repair and recovery operations, included detailed preventive maintenance procedures, gave spare parts requirements, identified instrumentation calibration requirements, delineated responsibilities, and presented the operation and maintenance philosophy.

Following is a short history of the maintenance, repair and modification performed on S8DR systems and facility systems, and an operating history of systems which did not perform as expected.

A. REACTOR HEAT TRANSFER SYSTEMS

- 1) In April 1968, following the initial fill, NaK flow could not be achieved in the primary loop. Flow blockage was traced to a plug in the inlet line to the dummy core. A 5-in. section of the line containing the plug was removed, the lines cleaned, a replacement spool installed, and testing continued normally. The material constituting the plug was not identified; it was described by those who saw it as "fibrous" but the origin and identity of the material were never determined with certainty.
- 2) On May 4, 1968, during the preoperations testing of heat transfer systems, the primary NaK pump throat developed a leak which then caused arcing in the stator. The cause of the leak was traced to intergranular stress corrosion brought about by high residual stress in the rectangular tubing, and probably aggravated by a corrosive media from wash water salts leaching contaminants from the insulating material between throat and stator. The discovery of this problem led to replacement of the primary and secondary pump throats with redesigned throats prior to nuclear power operation. As a temporary measure, spare throats left over from S8ER operation were installed for temporary use.

- 3) On May 14, 1968, during pressurization of the secondary surge tank for calibration of pressure instrumentation, the rupture disk ruptured. Examination of strip charts indicated that the design pressure was not exceeded; the cause is believed due to a defect in the material or to the existence of a wrinkle in the installation, coupled with fatigue.
- 4) On November 15, 1968, valve V-153B in the primary plugging meter loop was discovered to be leaking through a failed bellows. The valve was replaced.
- 5) On November 25, 1968, a corroded area was discovered on the primary pump throat in an area on the edge of the copper bar section. The cause was traced to a braze-contaminated weld. Metallurgical samples were taken following 200 hr additional operation at 1300° F, and analysis indicated the throat was safe to operate.
- 6) On February 9, 1969, the backup ABHX X-204 tubes developed a leak. Inspection indicated that the cause was insufficient allowance for material expansion.
- 7) On March 20, 1969, the reactor scrammed on loss of power to the ABHX fan motor. Repair was effected by replacing the failed speed control motor.
- 8) On March 30, 1969, the secondary surge tank level was observed to be dropping, accompanied by a gradual rise in secondary fill tank level, due to a slow leak in secondary fill and drain valve V-463. The leak rate was reduced by increasing fill tank pressure and decreasing surge tank pressure to remove the driving pressure for the leak. The pressure differential across the valve was controlled to approximately zero during subsequent operations.
- 9) On June 1, July 9, and November 11, 1969, reactor scrams occurred on loss of flow when the primary pump powerstat blew fuses in one phase. The problem was traced to poor contact between the carbon brush and coil winding of the powerstat, resulting from long-term operation in the same position. The UPS voltage was so steady that

routine "trimming" of flows was unnecessary. Following the November 11 scram, the UPS voltage was varied on a weekly basis, which required adjustment of flows, and resulted in a wiping and cleaning action of the brushes. High running currents, resulting from line voltage phase misbalance may have contributed to this problem.

- 10) During the period May through August 1969, a primary NaK drain valve leaked, resulting in a very gradual leak of primary NaK to the drain tank. In late August, the drain tank pressure was increased from a near vacuum to 15.5 psia. This lowered the driving pressure for the leak and reduced the leak rate to a negligible value.
- 11) The secondary drain line from the IHX, line 482, was found to be plugged when an attempt was made to drain the loop on January 15, 1970. The plug was cleared by heating the line and heating the NaK to 1050° F (instead of 300° F) for the drain.
- 12) The Primary Surge Tank Cover Gas Sampling System was plagued with problems throughout reactor operations. The major problem was plugging, believed due to deposition of NaK vapor in the loop. This problem was never solved, but by controlling flow, raising vault ambient temperature, "blowing down" etc., sufficient flow was maintained to supply the required data.

B. FACILITY SYSTEMS

- 1) The refrigeration system for cooling the vault atmosphere was over-designed and led to numerous failures of the heater in the compressor oil separator.
- 2) The Uninterruptible Power Supply (UPS) suffered control problems in the early part of the test program, which led to numerous unexplained diesel engine takeovers and one reactor scram (see Appendix A).
 - a) On February 2, 1969 the UPS was put back in service following replacement of certain electrical components in the motor field application package and power supply.

- b) On August 14, 1969, the UPS was returned to service following replacement of high-leakage electrolytic capacitors in the power supply which supplies eddy-current coupling clutch control voltage.
- c) After the above problems were corrected, the UPS performed as expected; so well, in fact, that NaK pump powerstat failure developed due to infrequency of adjustments to compensate for diurnal line voltage fluctuations.

C. VACUUM SYSTEM

- 1) Indicator light holders on control board BM-3 were apparently of poor quality. On several occasions shorting occurred during bulb replacement, resulting in vacuum pump shutdowns, closed valves, etc. Repairs were never made; instead, pilot bulbs were left burned out so as not to risk recurrence of the problem, and parallel indicator lights on BM-4 were monitored.
- 2) The two diffusion pump cryogenic baffles interacted through the common exhaust line. This led to poor liquid level control and freezing of chilled water in one of the pumps when shut down. The problem was corrected by re-routing the exhaust lines. Flowmeters were added to chilled water lines to monitor water flow.

D. INSTRUMENTATION

- 1) One specific model of strip chart recorder used to monitor process systems suffered a 100% failure rate. Problems were traced to poor design of the zero-adjustment potentiometer and failed capacitors in the dc amplifiers, possibly aggravated by storage of these recorders for more than 2 years before being put into use.
- 2) Reactor in-core thermocouples suffered a 50% failure rate during reactor power operation. These failures were related to stretching and flexing of thermocouples due to motion of the thermocouple "spreader" in response to temperature gradients. Most of the failed thermocouple "healed" when reactor power, and hence radial ΔT , were reduced.

- 3) Considerable difficulty was encountered in calibrating the NaK pressure indicator-transmitter units in the test vault. Most of the difficulty was traced to incorrect calibrating procedures. Records indicate that calibrations held within specifications between August 21, 1968, and December 30, 1969.