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DISASSEMBLY
AND
POSTOPERATION EXAMINATION
OF THE
SNAP 8 EXPERIMENTAL REACTOR

AEC Research and Development Report

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By
R. E. DYER

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ABSTRACT

A discussion of the shutdown and disassembly of the SNAP 8 Experimental Reactor and of the postoperation evaluation of the heat transfer system, the reflectors, and portions of the core. Information regarding material compatibility with system operation conditions was compiled. The reactor was shut down after 12,000 hr of operation, which included 100 days at 600 kwt, and over 1 year at 400 kwt or greater. The core coolant outlet temperature was 1300°F in both cases. The fuel examination results are included in another report.

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I. INTRODUCTION AND SUMMARY

The SNAP 8 reactor is designed to operate at a thermal power of 600 kw with a coolant outlet temperature of 1300°F. The SNAP 8 Experimental Reactor (S8ER) project, a test of the reactor core only, was the first of a series of nuclear ground tests to be conducted by Atomics International (AI) under contract to the United States Atomic Energy Commission (USAEC). In the evaluation of the SNAP 8 nuclear system, the reactor parameters determined experimentally in the S8ER test program and data obtained from a postoperational examination of the core and associated components were needed to evaluate the calculational techniques used in the design of the flight reactor. The postoperation examination objectives were to determine any changes that had taken place in the components during operation, and to evaluate the significance of these changes.

The S8ER was shut down on April 15, 1965, following approximately 2 years of operation.⁽¹⁾ During this time the reactor operated over 12,000 hr and the NaK coolant systems operated more than 14,000 hr. The reactor released a total of 5.1×10^6 kwh of energy. The reactor was operated at a power of 600 kwt with a core outlet temperature of 1300°F for 100 days and at or above 400 kwt with a 1300°F core outlet temperature for 365 days. Prior to the final shutdown, the reactor had operated continuously for over 5,000 hr at or above 450 kwt with a core outlet temperature of 1200°F or greater.

The postoperation evaluation of the S8ER was divided into three phases with specific types of examination required of each phase.

A visual inspection was conducted to detect physical abnormalities of the primary and secondary coolant systems piping and components (e. g. , corrosion, mass transfer, erosion, discoloration and deformation).

Metallographic characteristics of selected sections of piping and components were determined.

All of the fuel elements were subjected to an initial screening examination. Each element was inspected visually, photographed, weighed, dimensioned and gamma scanned. On the basis of the results of the tests, and considerations of core location and as-built data, several elements were selected for various

additional examinations such as: permeation testing, fission gas analysis, fuel density, hydrogen content and distribution, metallographic analysis of fuel and cladding material, tensile tests of cladding, and fuel burnup. Detailed results of the nondestructive and destructive fuel examination are reported separately.⁽²⁾

The core vessel, core inlet and outlet piping, tie rods, internal reflectors and grid plates were visually inspected and photographed. Dimensional measurements were taken of the vessel, tie rods, internal reflectors and the grid plates. Samples of the vessel, internal reflectors and inlet and outlet piping were metallographically examined and segments of the core vessel were scanned for induced gamma activity.

The postoperation examination indicated that, in general, the materials selection and design of the heat transfer system, core components and reflectors were good, and that the system could have given continued service at the design conditions. The need to redesign the core internal reflectors was indicated by the cracking of the cladding material and by the failure of the end-pin welds.

II. S8ER SHUTDOWN AND DISASSEMBLY

A. SHUTDOWN AND PREDISASSEMBLY OPERATIONS

To provide less ambiguous data for the postoperation fuel examination (hydrogen redistribution, phase change swelling, irradiation damage, fuel physical properties and fission gas release), the final reactor shutdown was accomplished by decreasing the reactor core temperature and power rapidly as shown in Figure 1.

During the 6-week decay heat removal period following reactor shutdown, the core temperature was maintained at $< 600^{\circ}\text{F}$ by regulation of the primary system coolant flow rate. The primary coolant system was drained and flushed twice during the decay heat removal period. A NaK sample was obtained from the primary system cold trap during the first drain. The system was drained a second time to repair a leak in the cold trap drain valve. NaK circulation was stopped for ~ 5 min on May 2, 1965 as the result of a scheduled electrical power

Figure 1. Final Reactor Shutdown Temperature and Power Decay

outage. Figure 2 presents the core temperature history for the duration of the decay heat removal period plus the period that the core was drained until core removal.

B. DISASSEMBLY

Disassembly of the S8ER facility began with the disassembly of the secondary NaK system on April 18, 1965. Fuel removal was accomplished during July. The dismantling of the reactor assembly and remaining systems, and decontamination of the facility was completed by January 13, 1966. The bar chart (Figure 3) presents a dated sequence of the different phases of the S8ER disassembly.

1. Heat Transfer System Disassembly

The NaK coolant systems piping and components were drained, thermal insulation was removed, and the low spots in the system were x-rayed for evidence of residual NaK pockets. A four-wheeled pipe cutter was used to cut the piping while still preserving the pipe's physical characteristics for postoperation

Figure 2. S8ER Core Temperature History Following Final Shutdown

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Figure 3. S8ER Disassembly Schedule

examination. An inert gas purge was maintained on the system during disassembly. Disassembled piping was marked for identification as to its exact location in the system. A metallurgist visually inspected each section of piping and each component for any abnormalities that would be of interest for further metallurgical examination. All abnormalities observed in the visual inspection were photographed. Special pipe samples, which were marked prior to installation in the primary coolant system, were retrieved for examination and comparison with pipe sections retained as archive samples. Piping and components selected for metallographic tests were removed from the facility for the examination. The remainder of the piping and components were stored for ultimate disposal upon completion of the examination.

2. Reactor Core Removal

The S8ER fuel was removed as a unit package to allow a detailed visual inspection of the entire core in its operating configuration during the disassembly in the AI Hot Lab (AIHL). Handling of the core as a package minimized the possibility of fuel damage and excessive personnel radiation exposure by reducing the number of fuel handling operations.

A lifting tool was designed to remove the fuel and core internals as a unit. This tool (Figure 4) had 30 toggle rods positioned by lower and upper plates so that the rod locations matched flow holes in the upper and lower grid plates of the core. Lifting toggles at the bottom of each rod were retained in a closed position by a lower plate which rested on shoulders near the lower ends of the rods. As the lifting tool was lowered onto the core, the toggle rods passed through the flow holes in the upper grid plate. The lower plate came to rest on the upper grid plate permitting the toggle rods to slide first through the flow channels between the fuel elements and then through the flow holes in the lower grid plate. The toggles, attached to the bottom of the rods, bottomed against the flow baffle plate beneath the lower grid plate forcing the toggles from a closed to an open position as the lifting tool was fully lowered. The toggles were then brought up snug against the lower grid plate by the tightening of distance bolts in the upper plate of the tool.

A special core transport cask (Figure 5) was designed and fabricated to transfer the unitized core to the AIHL for examination. A 52 in. square lead

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Figure 4. Core Lifting Tool

Figure 5. Core Transport Cask

biological shield (Figure 6) was installed over the reactor containment vessel during core removal. Suitable access plugs for a lead viewing window and long handled tools were provided. An inert argon atmosphere was maintained in the containment vessel during core removal to minimize oxidation of residual NaK in the core.

Figure 7 illustrates the arrangement of equipment during core removal. The core transport cask rested on the lead shield. It provided a transfer lock, with the 40-in. diameter gate valve, for the insertion and withdrawal of tools while maintaining an inert atmosphere in the core region.

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Figure 6. Lead Shield

The center plug was lifted from the shutdown shield and stowed on a specially installed bracket against the inside wall of the containment vessel. This step eliminated the manufacture of a shielded container for removal of the plug from the containment vessel.

Long handled tools were inserted through the gas lock to remove the core hold-down bolts from the tie rods and install tie rod extension pieces to serve as guides for insertion of the core lifting tool. The lifting tool was then inserted and engaged. The core was raised into the cask, a pin was inserted through the lifting tool handle to support the core while in the cask, and the lifting tool handle extension was removed. A seal cap was bolted over the protruding handle and the cask lifted from the lead shield and set on the cask lower plate located adjacent to the containment vessel. The lower plate was bolted to the cask. The core was then transported to the AIHL for disassembly and examination.

The recorded radiation levels were approximately a factor of 10 lower than predicted (Table 1). The shielding proved effective in limiting the maximum

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Figure 7. Core Removal Equipment Arrangement

TABLE 1
RADIATION LEVEL COMPARISON

Location	Calculated	Actual
Reactor Core	61,250 r/hr at 1 ft after 90 days	37,400 r/hr at 9 in. 6/2/65 Two readings taken at different ports through shutdown shield (1) 20,400 r/hr at 9 in., 7/14/65 (2) 14,400 r/hr at 9 in., 7/14/65
Surface of Core Transport Cask	900 mr/hr 90 days after	7/28/65 (1) 16.0 mr/hr gamma (2) 4.4 mrem/hr thermal neutrons (3) 27 mrem/hr fast neutrons

dose received by the operating personnel to 50 mr. Contamination control consisted of cleaning the tools and equipment used inside the containment vessel. The facility was free of contamination at the conclusion of the operation.

3. Disassembly of Reactor Core Vessel and Reflector Assembly

The reactor core vessel and the beryllium reflector assembly, mounted on the bottom of the shutdown shield (Figure 8), were disassembled in the S8ER primary equipment vault. Temporary shielding was arranged around the vault and the containment vessel to permit transfer of the shutdown shield assembly. The disassembly was accomplished using standard remote handling and viewing techniques.

An enclosure was constructed over the manipulator and primary equipment vault to control contamination. The enclosure was constructed of a wood framework covered with plastic. Exhaust from the enclosure was routed to the building stack through an absolute filter. The components were removed with an abrasive saw and shipped to AIHL for examination. These components consisted of the core vessel coolant piping, control drum drive shafts, beryllium control drums and supporting framework, and the reactor core vessel.

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Figure 8. Reactor Core Vessel and
Reflector Assembly Mounted
Beneath Shutdown Shield

Following the disassembly operations, the shutdown shield was returned to the containment vessel for storage. The disassembly equipment and tools were decontaminated and removed from the reactor room. The reactor room was decontaminated, and the containment vessel and primary equipment vault lids replaced.

Table 2 is a list of radiation levels measured during disassembly of the core vessel and reflector assembly.

TABLE 2
RADIATION LEVELS DURING DISASSEMBLY OF
CORE VESSEL AND REFLECTORS

Location of Radiation Measurement	Dosage (mr/hr)
Dose rates to occupable spaces during disassembly	
At the remote handling station (outside SW corner of building) during transfer of the complete assembly from the containment vessel to the primary equipment vault	50
At the remote handling control station during disassembly operations	1.3
Floor area in reactor room outside the temporary shielding during disassembly	10
Dose rates of sources of radiation during disassembly	
At the bottom edge of the shutdown shield (shutdown shield in primary vault) prior to disassembly	900,000
West wall of the containment vessel (at ~3 in.) at elevation of core vessel (containment vessel empty)	120,000

III. POSTOPERATION EXAMINATION RESULTS AND CONCLUSIONS

A. HEAT TRANSFER SYSTEM

The S8ER heat transfer system used Type 316 SS for all components exposed to the NaK (78 wt% K) coolant with the exception of some components in the core which were Hastelloy. The system was designed to operate at temperatures up to 1300°F and power levels up to 600 kwt. Refer to Tables 3 through 6 for S8ER temperature and power history. The majority of operation was conducted with a reactor ΔT of 200°F. A minor portion of operating time was with a reactor ΔT of 300°F. Coolant flow rate was varied up to a maximum of 1000 lb/min. The flow rate history is shown in Table 7.

Prior to reactor shutdown, portions of the heat transfer system were designated for specific types of detailed postoperation examination. In addition, all components were visually inspected during disassembly to select for comprehensive examination those which appeared affected by operation.

TABLE 3
S8ER OPERATING TIMES AT VARIOUS
TEMPERATURES AND POWERS

Core Outlet Temperature (°F)	Reactor Power (kwt)					Line
	Shutdown	0 to 100	100 to 400	400 to 500	> 500	Total
0 to 200	2227	14				2,241
200 to 400	831	48				879
400 to 600	617	77				694
600 to 800	481	477	29			987
800 to 1000	255	227	127	936	7	1,552
1000 to 1200	94	73	124	547	27	865
> 1200	167	407	59	5845	2959	9,437
TOTALS	4672	1323	339	7328	2993	16,655

TABLE 4
S8ER POWER HISTORY

Month	Power (kw)				
	0 to 100	101 to 400	401 to 500	501 to 599	600 or Greater
1963					
May	53*	0	0	0	0
June	168†	0	0	0	0
July	616†	0	0	0	0
August	82†	0	0	0	0
September	0	0	0	0	0
October	14§	0	0	0	0
November	91	24	0	7	22††
December	15	40	188	75	75††
1964					
January	146	2	314	119	0
February	7	19	241	190	0
March	3	1	681	52	0
April	72	5	400	0	0
May	77	238	379	3	25
June	30	3	242	0	432
July	0	1	0	0	740
August	0	0	0	0	744
September	2	3	504	25	152
October	0	0	745	0	0
November	0	0	720	0	0
December	0	0	744	0	0
1965					
January	0	0	493	0	251
February	0	0	591	0	81
March	0	0	744	0	0
April	0	0	342	0	0
Totals (hr)	443**	336	7328	471	2522

*Dry Critical

†Wet Critical

§Critical: NaK in Core only

**Items marked with *, †, or § are not included in total

††Runs of 620 kw, all other runs in this column were 600 Kw.

TABLE 5
S8ER HOT-LEG TEMPERATURE HISTORY

Month	Temperature (°F)								
	0 to 200	201 to 400	401 to 600	601 to 800	801 to 1000	1001 to 1200	1201 to 1299	at 1300	1301 to 1370
	Time (hr)								
1963									
May	0	0	0	0	0	0	0	0	0
June	0	186	280	76	92	26	36	0	0
July	0	0	0	293	53	55	343	0	0
August	252	7	135	245	7	6	92	0	0
September	720	0	0	0	0	0	0	0	0
October	745	0	0	0	0	0	0	0	0
November	426	16	84	100	5	35	54	0	0
December	94	121	48	29	151	61	5	193	37
1964									
January	0	136	18	75	262	18	52	183	0
February	0	206	6	17	43	11	6	105	302
March	0	0	0	7	253	20	5	407	52
April	0	201	78	31	25	6	0	378	0
May	0	0	0	33	497	129	32	53	0
June	0	0	0	0	43	4	0	673	0
July	0	0	0	0	3	1	0	740	0
August	0	0	0	0	0	0	0	744	0
September	0	0	3	20	14	8	32	643	0
October	0	0	0	0	0	0	0	745	0
November	0	0	0	0	0	0	0	720	0
December	0	0	0	0	0	0	0	744	0
1965									
January	0	0	0	0	0	0	0	744	0
February	0	0	0	0	0	0	0	672	0
March	0	0	0	0	0	131	0	613	0
April	0	0	376	1	2	341	0	0	0
May	48	205	491	0	0	0	0	0	0
Totals (hr)	2285	1078	1519	927	1448	852	657	8362	391

TABLE 6
S8ER COLD-LEG TEMPERATURE HISTORY

Month	Temperature (°F)						
	0 to 200	201 to 400	401 to 600	601 to 800	801 to 1099	At 1100	1100 to 1270
	Time (hr)						
1963							
May	0	0	0	0	0	0	0
June	0	186	280	76	118	0	36
July	0	0	0	293	108	0	343
August	252	7	135	245	13	0	92
September	720	0	0	0	0	0	0
October	745	0	0	0	0	0	0
November	426	16	84	100	40	0	54
December	94	121	48	108	134	235	4
1964							
January	0	136	72	254	15	190	77
February	0	206	6	17	60	407	0
March	0	0	0	260	25	459	0
April	0	201	78	31	31	378	0
May	0	0	0	530	161	53	0
June	0	0	0	0	47	673	0
July	0	0	0	0	4	740	0
August	0	0	0	0	0	744	0
September	0	0	3	20	54	643	0
October	0	0	0	0	0	745	0
November	0	0	0	0	250*	470	0
December	0	0	0	0	0	744	0
1965							
January	0	0	0	0	0	744	0
February	0	0	0	0	275*	397	0
March	0	0	0	0	577*	167	0
April	0	0	376	1	2	341	0
May	48	205	491	0	0	0	0
Totals (hr)	2285	1078	1573	1935	1912	8130	606

*Hours at 1000°F Steady State

TABLE 7
PRIMARY EM PUMP HOURS AT
VARIOUS FLOWRATES*

Months	Flowrate (lb/min)				
	0 to 200	201 to 400	401 to 600	601 to 800	801 to 1000
	Time (hr)				
1963					
June	26	75	328	177	110
July	0	0	0	0	740
August	1	215	24	31	183
September	0	0	0	0	0
October	0	0	0	0	0
November	33	20	47	91	74
December	24	23	187	102	218
1964					
January	0	72	295	126	94
February	5	6	21	428	32
March	0	1	275	459	8
April	1	0	81	433	14
May	18	36	365	121	204
June	0	0	0	449	271
July	0	0	0	0	741
August	0	0	0	0	744
September	0	0	0	480	210
October	0	0	224	521	0
November	0	0	744	0	0
December	0	0	744	0	0
1965					
January	0	0	493	0	251
February	0	0	591	0	81
March	0	0	613	131	0
April	9	0	0	710	1
May	0	0	27	337	152
Totals (hr)	117	448	5059	4596	4128

*In addition to the above time, ~256 hr of pump running time were accumulated during July and November of 1962; however, accurate flow-rate data for these times, were not recorded.

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Figure 9. ID Surface of
As-Received Pipe

Figure 10. Section of Primary
Pump Outlet Header with
Flow Deflector

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To provide a reference for the postoperation analysis, a section of pipe from the 2-in. -diameter, schedule 5, core outlet line had been retained as an archive sample. The ID surface of a section of this pipe is shown in a photomicrograph (Figure 9).

The metallographic examination indicated that many of the surfaces contacted by NaK developed a slight surface reaction that was 0.0005 to 0.002 in. thick, such as grain boundary leaching. X-ray diffraction did not identify any oxides, carbides, or nitrides. A layer on the inner surface of the end cap for the primary system coolant pump (Figure 10), investigated by electron microprobe analyses, revealed a depletion of Cr and Fe and an enrichment of Ni.

In general, hardness changes persisted to 0.005 to 0.006 in. The OD surfaces of most components and piping displayed a thin oxide layer. The thickest layer measured was about 0.002 in.

This physical and metallurgical examination of the heat transfer system shows that it was capable of many thousands of hours more of dependable service. There was no evidence of significant mass transfer or major corrosion which would have led to system failure. The postoperation examination revealed only one failure, which was in the bellows of a bellows seal valve. Due to the special design of this valve for liquid metal service, the failure did not cause a leak of the system itself.

One system piping failure was reported⁽¹⁾ and is repeated here for completeness. The originally installed electromagnetic (EM) pump header end caps failed in December, 1963 and January, 1964. From an evaluation of these end cap failures using a water hydraulic mockup, it was concluded that the metal was being eroded by coolant vortexing (Figure 11). A redesigned end cap with flow deflectors was installed. Postoperation examination of these new end caps showed no erosion or metal loss on the flow deflectors.

Table 8 is a summary of the examination results of major areas of interest in the primary heat transfer system. The locations of the metallography samples are indicated in the system schematic diagram (Figure 12).

2-17-64 UNC

7568-54138

Figure 11. Primary EM Pump Discharge Header Cap, Showing Erosion Failure

6-9-67 UNC

7570-8099C

Figure 12. Schematic of S8ER System, Metallographic Sample Locations Circled

TABLE 8

SUMMARY OF POSTOPERATION EXAMINATION OF PRIMARY HEAT TRANSFER SYSTEM

Component	Sample Location	Examination Results
Cold Trap	Near top of vertical leg of cold trap	A surface layer, 0.002-in. maximum depth, which was less sensitive to attack by etchant. The lower hardness and etching characteristics indicate a possible leaching condition. Degree of reaction is slight considering exposure time. (Refer to Figure 13)
Primary Surge Tank	Vertical strip of tank wall including gas interface	Gas region showed dark coloring. Slight surface layer and case hardening in both the gas and NaK regions due to contamination and work hardening during fabrication. Less pronounced in NaK region due to leaching of contaminants and stress relieving by NaK (Figure 14 and 15).
Pipe Elbow	Pipe elbow on outlet of surge tank (a pre-installed measured sample)	ID surface had a thin (<0.0005 in.) intermittent surface layer. OD surface had layer which often exceeded 0.002 in., believed to be oxide. Reaction layers destroyed orientation etch marks for dimensional comparison with pre-installation measurements (Figure 16).
Intermediate Heat Exchanger	Sections of tubes near inlet and outlet headers	ID surface of tube near inlet showed slight attack along grain boundaries near the surface due to leaching. ID surface of tube near outlet showed thin reaction layer or deposit and chemical analyses revealed no conclusive evidence of mass transfer (Figure 17.)
Primary Coolant Pump	1. Throat 2. Outlet Header End Cap	1. Visual inspection showed that the pump throat was unaffected. Copper did not migrate into stainless steel pump throat through nickel alloy overlay on copper bar (Figure 18.) 2. No erosion; slight surface reaction on surface exposed to flowing NaK; electron microprobe analysis confirmed a depletion of Cr and Fe and an enrichment of Ni in the reaction zone near the surface; no significant change in microhardness detected between reaction zone and base material (Figures 10 and 19).
Primary Coolant Flowmeter	2-in. throat section of flowmeter plus transition pieces to 2-1/2 in.	Inlet and outlet sections exhibited extensive precipitation along grain boundaries (Figure 20).
Valve	Primary system block valve	Visual inspection showed bright, shiny, sharp edges; bellows was unaffected (Figure 21.)

6-9-67 UNC

7695-4708

a. Microstructure (Grain Boundaries Near Surface Appear Less Sensitive to Attack by Etch)

6-9-67 UNC

7695-4731

b. Hardness (175 KHN Near Surface, 280 KHN 0.004 in. Below Surface, 210 KHN 0.010 in. Below Surface)

Figure 13. Section of Primary Cold Trap Line

6-9-67 UNC

7695-4709

Figure 14. Section From Primary Surge Tank Wall

a. Section From Cover Gas Region (290 KHN surface hardness, decreasing to average of 185 KHN 0.012 in. below surface)

6-9-67 UNC

7695-4710

b. Section From NaK Region (235 KHN surface hardness, decreasing to average of 180 KHN 0.004 in. below surface)

6-9-67 UNC

7695-4732

Figure 15. Inside Surface of Surge Tank

a. ID Surface Exposed to NaK (Sensitized structure evident from chromium carbide precipitation at grain boundaries, corrosion minimal)

6-9-67 UNC

7695-4711

b. OD Surface Exposed To Primary Vault Nitrogen Atmosphere ($< 0.1\% \text{ O}_2$) (Reaction layer considerably greater than on surface exposed to NaK, but neither is significant)

6-9-67 UNC

7695-4733

Figure 16. Reactor (Hot Leg) Coolant Pipe

6-9-67 UNC

7695-4712

a. Near Inlet To Tube Coil (Some leaching evident along grain boundaries at surface)

b. Near Outlet of Tube Coil (Thin deposit or reaction layer present along surface exposed to NaK)

6-9-67 UNC

7695-4734

Figure 17. Inside Surface of Intermediate Heat Exchanger Tube

Figure 18. Section of Inlet Header

6-9-67 UNC

7695-4713

Figure 19. Section of Primary Pump Outlet Header Insert

6-9-67 UNC

7695-4714

a. Microstructure
Typical of Inlet and
Outlet Sections

b. Section of Throat
(Microhardness
Traverses indi-
cate no surface
hardening)

Figure 20. Flowmeter

Figure 21. Section of Body of 2-in. Valve V-104
After Removal From System (All surfaces
sharp, bright, and shiny -- appear
unaffected by service)

B. REACTOR CORE VESSEL AND REFLECTOR ASSEMBLIES

The components mounted beneath the shutdown shield were examined at AIHL; they included the beryllium control reflectors, internal beryllium reflectors, core inlet and outlet coolant piping, core vessel, core tie rods, and grid plates.

Figure 22 shows the components mounted beneath the shutdown shield and the approximate location of the various examination samples.

1. Beryllium Reflector Assembly

Visual and photographic examination showed no visible oxides, discoloration, deformation, or other defects. Figure 23 is a photograph of one of the beryllium control drums taken during the postoperation examination. The markings on the surface are the result of handling with a remote manipulator. Figure 24 is a photograph of one of the control drums taken prior to the system test.

A history of the reflector scram times as measured from an oscilloscope trace of the reflector position versus time is presented in Table 9. This table indicates a slight lengthening in the scram time toward the last of operation, but all times measured were within specification.

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Figure 22. S8ER Postirradiation Examination with Approximate Location of Various Test Samples

Figure 23. Beryllium Control
Drum During Postoperation
Examination

Figure 24. Beryllium Control
Drum Prior to System
Test

6-9-67 UNC

7695-4718

TABLE 9
REFLECTOR SCRAM TIME HISTORY

Date	Control Drum No.					
	1	2	3	4	5	6
3-1-63	410	430	400	420	410	400
6-20-63	390	450	400	430	400	390
10-22-63	400	400	370	420	370	350
11-22-63	400	420	360	400	400	390
12-3-63	380	410	360	440	390	390
2-10-64	390	420	410	410	400	390
9-9-64	430	470	420	420	420	460
4-16-65	470	430	460	450	410	460

The overall dimensional measurements of the beryllium reflectors showed that no significant dimensional change had occurred.

Postoperation tests of the beryllium control reflectors indicated that no change in density occurred. Two samples, one each from the central area of the curved face of No. 2 and No. 4 control reflectors had minimum exposure. The average density of three measurements of each sample was 1.849 gm/cc. Subsequent to this initial density determination, samples were annealed at 1300, 1500, and 1600°F. No significant density change occurred due to the annealing.

Postoperation metallography samples of the beryllium control reflectors revealed no abnormalities. Some porosity was evident at the grain boundaries but this is characteristic of beryllium. Three samples from each of No. 2 and No. 4 control drums were taken for metallography tests. The samples selected represented the maximum and minimum integrated neutron flux exposure and maximum and minimum temperature exposure. See Figure 25 for a representative beryllium metallography test sample.

6-9-67 UNC

7569-4719

Figure 25. Typical Microstructure of Irradiated Beryllium, Porosity Along Grain Boundaries is Also Characteristic of Unirradiated Beryllium (250x)

These postoperational examinations of the beryllium reflectors indicate that no significant changes to the beryllium occurred during operation. This, plus the fact that no beryllium contamination was found inside the containment vessel, indicates that anodizing the reflectors prevented chemical reaction with the environmental gases.

2. Core Inlet and Outlet Pipe

The core inlet and outlet coolant piping located beneath the shutdown shield was examined by AIHL. Visual inspection revealed no abnormalities on the exterior surfaces except for a thin black oxide layer that was easily scratched through by normal handling with the manipulator. The interior surfaces showed varying degrees of NaK oxide deposits and a residue which was the result of

reaction with "Dowanol" used as a reagent to consume the residual NaK. See Figure 26 for view of inside of core outlet coolant pipe adjacent to the vessel. No erosion or corrosion was evident.

Figure 22 shows the general location of pipe samples used in the metallographic examination. This examination of the Type 316 SS showed no evidence of excessive NaK corrosion. Heavy, dark intergranular precipitates were visible. In addition, there was some intragranular precipitation visible. Though not clearly identified, this could be sigma-phase material. The intergranular precipitate is probably agglomerated carbide; however, no specific determination was made. The general appearance is of aged, austenitic stainless steel with slightly heavier precipitation than normal.

6-9-67 UNC

7695-4720

Figure 26. Section of Reactor Outlet
Coolant Pipe Adjacent to Reactor
Outlet Nozzle

NAA-SR-11520

3. Core Vessel

Visual and photographic examination of the outside surface of the core vessel showed an irregular pattern of black oxide. The layer was very thin and was readily scratched through where contacted by the manipulator fingers (Figure 27). No other defects were noted on the exterior surface. The interior surface could not be observed in detail because of a thin, residual NaK oxide layer left after cleanup, but there were no abnormalities apparent to visual inspection.

The reactor core vessel diameters (inside and outside) were taken at three different locations. The inside diameter measurements were suspect due to the oxide layer mentioned previously. The outside diameter measurements increased an average of 0.007 in., or approximately 0.08%. This indicated that the creep of the vessel materials under the operating conditions was not excessive.

6-9-67 UNC

7695-4721

Figure 27. S8ER Core Vessel

Metallographic examination of the core vessel wall samples indicated that the degree of attack was less than expected and that the material (Type 316 SS) performed satisfactorily (Figures 28 and 29). The outside surfaces appeared normal. The inside surfaces showed some leaching between grain boundaries near the surface and a layer that appeared to be a corrosion product, probably an oxide. Some intragranular and intergranular precipitates were evident. The grain boundary phase was undetermined but may be a carbide. The phase within grain boundaries responded to etching and appeared to be hard and spherical. This may be sigma phase appearing after long exposure to high temperatures. The samples appeared to be in a worked condition as evidenced by turns within grains and the work hardening noted near the surfaces. This was probably imparted during fabrication.

Six samples were removed from the core vessel wall and subjected to tensile testing. Prepared test specimens were 0.250 in. wide, 0.084 in. thick, with a gage length of 1.250 in. The tests were conducted at 1200°F (in air) at a strain rate of 0.02 in. /min. The results of these tests are presented in Table 10 with data from irradiated samples tested under similar conditions presented for comparison.^(3, 4)

TABLE 10
RESULTS OF TENSILE TESTING OF CORE VESSEL WALL SAMPLES

Test Specimens	Yield Stress (psi)	Ultimate Stress (psi)	Elongation (%)
S8ER Samples (6)			
Average	19,200	39,200	26.9
Range	17,000 to 22,000	34,600 to 44,000	17.2 to 39.6
Reference Data from Irradiated Samples	19,000	50,000	25 to 46

The comparison of measured data with previously reported data indicates a lower ultimate yield. Typically, radiation of this type of material causes a reduction in the ultimate stress above 1100°F. The temperature at which this reduction is initiated varies from one heat to another. It is felt that the

Figure 28. Typical
Microstructure of
Core Vessel
Inside Wall

6-9-67 UNC

7695-4722

Figure 29. Microstruc-
ture of Core Vessel
Wall With Heavy
Precipitates at
Grain Bounda-
ries

6-9-67 UNC

7695-4723

difference in ultimate stress, as shown in the table, reflects the fact that the materials in question are derived from different heats.

Gamma scans of core vessel wall samples indicated that Co^{60} and Mn^{54} are uniformly distributed axially about the core vessel midpoint. The Co^{60} activity is asymmetrical peripherally whereas the Mn^{54} activity is symmetrical. The distribution of these activities corresponds to the thermal and fast neutron dose to which the core vessel was exposed. Attempts to correlate the asymmetric distribution with control drum position or areas of depleted hydrogen were unsuccessful.

4. Tie Rods

Visual and photographic examination revealed that four of the six tie rods were slightly bowed. No other abnormalities were noted except slight scoring due to the core removal tool and some NaK oxide deposits. Refer to Figure 30. The bowing is probably due to stresses imposed on the grid plates, as discussed later.

The tie rod length measurements showed an average increase of 0.002 in. over pre-irradiation measurements. The accuracy of all measurements was ± 0.001 in. This change in length is considered negligible.

5. Internal Reflectors

Visual and photographic examination of the internal reflectors showed several visible cracks in the Hastelloy N cladding, discolorations due to NaK oxides, some axial dark streaks probably due to contact with fuel elements, and missing locating pins from the bottom end caps (Figure 31). The missing locating pins may be the result of the method of removal from the bottom grid plate which was coated with NaK oxides. Since all reflectors (cracked and uncracked) contained NaK as determined by weight test, it is believed that the pin welds in the end caps were the most vulnerable area for failure and possible leakage. Table 11 is a summary of the visual inspection. Figures 32 through 35 show the different types of defects noted. The locator pin and cladding failures indicate a necessity for a redesign of the internal reflectors.

Figure 30. Core Tie Rods

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NAA-SR-11520
47

6-9-67 UNC

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6-9-67 UNC

7695-4736

a. View of Reactor Side of Reflectors

b. View of Core Vessel Side of Reflectors

Figure 31. Internal Beryllium Reflectors, Two D-Shaped, Four Airfoil-Shaped

TABLE 11
INTERNAL REFLECTOR EXAMINATION

TYPE	AIRFOIL SHAPED	D-SHAPED	BeO INSERTS
NUMBER OF REFLECTORS	12	6	2
VISUALLY DISCERNIBLE DEFECTS	6 HAD BOTH BOTTOM PINS MISSING 3 HAD ONE BOTTOM PIN MISSING 6 WERE CRACKED 2 WERE BOWED 10 WERE DISCOLORED AND STREAKED	1 HAD A BOTTOM PIN MISSING	NO VISIBLE DEFECTS
AVERAGE % ΔL	+0.1	+0.4	NOT A CRITICAL DIMENSION OF INDIVIDUAL INSERTS
AVERAGE % Δt	-3.4	-3.1	WITHIN AS-BUILT TOLERANCE
AVERAGE % Δw	-1.6	-1.1	AIRFOIL SHAPED INCREASED <1% OF AS-BUILT TOLERANCE D-SHAPED DECREASED <1% OF AS-BUILT TOLERANCE

NOTE: WEIGHT TESTS DETERMINED THAT ALL INTERNAL REFLECTORS CONTAINED NaK DUE TO IN-LEAKAGE. ONLY TWO INSERTS FROM EACH TYPE OF REFLECTOR WERE DIMENSIONED

Figure 32. Airfoil-Shaped Internal Beryllium Reflector With Large Crack in Narrow Edge at 5-1/2 in. to 8-1/2 in. From the Top

6-9-67 UNC

7695-4726

Figure 33. Magnification of View Shown in Figure 32

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6-9-67 UNC

7695-4728

Figure 34. View of Reactor Side of Airfoil-Shaped Internal Beryllium Reflector with Discoloration and Fuel Contact Marks

12-9-65

7568-02796

Figure 35. View of Bottom End of an Internal Reflector with Missing Locator Pin

Dimensional changes of the Hastelloy-N cladding and the BeO inserts are shown in Table 11. The BeO in each internal reflector consisted of several stacked segments. The length dimension of each piece was not critical but the length of the overall stack was finished to 14.354 ± 0.002 in. The width and thickness dimensions of the inserts were not individually recorded as manufactured but were made with a tolerance of ± 0.002 in.

Metallographic examination of the internal beryllium reflector cladding showed that the material (Hastelloy-N) was compatible with the NaK and BeO environment as the degree of interaction observed was not excessive. The sample sections (see Figure 22 for location) are coarse grained and show no

evidence of excessive work. There are two precipitates: one intergranular that was preferentially etched; and a clear precipitate within grains and across grain boundaries that was in relief. A narrow reaction zone (<0.001 in.) was observed on both the inside and outside surfaces.

Six samples of the internal reflector cladding material were removed from the flat sides of selected reflectors, and subjected to tensile testing. These samples were 1.10 in. thick, 0.250 in. wide, with a gage length of 1.250 in. Tensile tests were conducted at 1200°F (in air) at a strain rate of 0.02 in. /min. These test results are presented in Table 12. Data from irradiated specimens tested under similar conditions are included for comparison.⁽⁵⁾ The good comparison shows that the liquid-metal environment had little effect on the mechanical properties of the material.

TABLE 12
RESULTS OF TENSILE TESTING OF INTERNAL REFLECTOR
CLADDING MATERIAL

Test Specimens	Ultimate Stress (psi)	Elongation (%)
S8ER Samples (6)		
Average	53,500	6.3
Range	51,500 to 57,000	4.0 to 8.0
Reference Data From Irradiated samples	53,400	8.0

6. Grid Plates

Both grid plates were heavily coated with NaK oxides (Figure 36) which were formed during core removal. The only abnormality noted was slight warping of the plates.

It was initially thought that the apparent warp in the upper grid plate was due to the excessive strain required to remove the plate from the core package. However, subsequent measurements did not bear this out. Contour maps showed that the upper grid plate was dished upward approximately 0.023 in. and the

Figure 36. Bottom View of Top Grid
Plate with Heavy Oxide Deposits
Formed During Period of Core
Removal

bottom grid plate was dished downward 0.010 in. This maximum depth was offset from the center of the grid plates to the side of the core, which had the higher frequency of fuel elements with cladding cracks. The postoperation length measurements of fuel elements showed that some of the elements exceeded the as-built clearance between the grid plates by 0.020 in. It is concluded that the warpage of the grid plates, and the bowing of the tie rods previously noted, was due to stress imposed by fuel growth.

REFERENCES

1. L. M. Fead, L. D. Felton, and V. L. Rooney "SNAP 8 Experimental Reactor Operation and Test Results," NAA-SR-10903 (Confidential RD), June 28, 1965
2. P. S. Olson, K. J. Miller "Postoperation Evaluation of Fuel Elements from the SNAP 8 Experimental Reactor," NAA-SR-12029, (to be published)
3. M. J. McNelly, "Liquid Metal Fast Breeder Reactor Design Study (1000 Mwe $\text{UO}_2\text{-PuO}_2$)," GEAP-4418, Vol. I and II, January, 1963
4. T. T. Claudson, et al, "Evaluation of Iron and Nickel-Base Alloys for Medium and High Temperature Reactor Application: Part II," BNWL 154 (11-65)
5. W. R. Martin and J. R. Weir, "Effect of Elevated Temperature Irradiation on Hastelloy N," Nuclear Applications, Vol. 1, No. 2, 1965, p 161.