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

AEC Research and Development Report

ATOMICS INTERNATIONAL

A DIVISION OF NORTH AMERICAN AVIATION, INC.

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By

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ABSTRACT

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

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I. INTRODUCTION AND SUMMARY OF RESULTS

The SNAP 10 Flight System 3 (S10FS-3) was operated at the Santa Susana site of Atomics International (AI) for the Atomic Energy Commission (AEC). The FS-3 test was a full-scale acceptance test of a complete flight system, modified only to the extent necessary to adapt it for ground testing. The test was conducted in a space simulation chamber located in one of the power test vaults of the SNAP Environmental Test Facility (SETF). Prior to installation in this test location, the system had been subjected to acceptance and qualification level shock and vibration testing simulating launch environment.

The nuclear acceptance testing consisted of an automatic startup to design conditions, and extended operation at design conditions for over 10,000 hr. This operation was continuous and undisturbed by system shutdown, and control action was taken only once to return the system to initial operating conditions a month prior to shutdown.

During this operation, the reactor produced 382.9 Mwh of thermal energy. It operated for 10,028 hr continuously, and accumulated 10,005.5 hr of system operation at temperatures above 900°F. For more details on the description of the system test and operational results see Reference 1.

The FS-3 program was terminated and system shutdown was initiated on March 15, 1966. The month immediately following system shutdown was spent in setting up equipment and training in the remote handling techniques to be used in the disassembly of the system. On April 26, 1966, the power test vault was opened and the placement of television equipment by remote means was initiated. On May 21, 1966, the reactor core was removed from the SNAP unit and placed in a shipping cask for transfer to the Atomics International Hot Laboratory (AIHL). By June 3, 1966, the remote disassembly of the system had progressed to the point that physical access of personnel to the power test vault was allowable. Subsequent cleanup and decontamination was accomplished manually. All radioactive components were removed and decontamination of the facility was completed by September 1, 1966.

Postoperation examination of critical components of the system was performed to evaluate the materials selection and component design after exposure

to system conditions. The evaluation of the fuel was accomplished by the AIHL and is reported separately.⁽²⁾ The evaluation of the reactor vessel, core internals, reflector assembly, and NaK pump and outlet piping assembly was also conducted at the AIHL due to the high radiation levels of these components. Those components whose activity did not preclude manual handling (the radiation shield, the expansion compensator units, the thermoelectric modules, and the NaK coolant) were sectioned and disassembled in the transfer cell of the SETF. Samples of interest were then analyzed in various laboratories throughout AI.

The postoperation examination revealed that the component design, material selections, and fabrication techniques were more than adequate to meet the operating environment. There was no evidence of attack by the liquid metal coolant, no unexpected component wear, and no component degradation attributable to the radiation environment. Two component failures were noted. One of these, a failure of the inner containment of an expansion compensator, had been observed during system operation. This failure did not interrupt operation, due to double containment in the compensator design. One of the springs supporting the lithium hydride block within the shield assembly was found to be broken upon decanning the shield. Both of these failures were evaluated by metallographic examination. In each case, the mode of failure was typical of material fatigue, possibly induced by the mechanical and thermal cycling to which these units were subjected during testing.

II. TEST DESCRIPTION

A. S10FS-3

The S10FS-3 nuclear power unit was a standard SNAP 10A flight system which had been modified for nuclear ground testing. The SNAP 10A system includes the nuclear reactor, the thermoelectric converter, a complete heat-transfer system, a support structure, and the instrumentation necessary to perform nuclear startup and control and diagnostic functions (Figure 1).

The reactor for the SNAP 10A system is beryllium reflected, and has a core containing 37 cylindrical fuel elements made of a hydrided uranium-zirconium alloy. The reactor is depicted in Figure 2. Hydrogen moderation is provided by the elements themselves, which are hydrided to a hydrogen concentration slightly less than that of cold water. The reactor is cooled by a liquid sodium-potassium alloy (NaK). The NaK enters the reactor at about 900°F, extracts ~420 kw, and leaves the reactor at about 1020°F. Active control of the reactor is provided by four rotating beryllium control drums which regulate the neutron leakage from the core.

The heat-transfer system contains a liquid-metal (NaK 78) coolant which is pumped through the reactor and power converter by a thermoelectric, dc conduction pump. The piping which interconnects the system components hydraulically and thermally contains two volume expansion compensator units (ECU) which maintain a void-free coolant transfer system at all times.

The support structure is a truncated cone made of titanium. Its cross section is a corrugated structure designed to increase its load-carrying capability and to protect and support the tubes composing the thermoelectric converter assembly.

Diagnostic and special ground test instrumentation are installed throughout the unit and terminate in an instrument compartment contained in the base of the support structure and separated from the converter section by a thermal barrier.

A lithium hydride shadow shield is positioned below the reactor to provide radiation protection to the instrument compartment. The shield was designed to fit within the conical support structure immediately below the reactor.

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Figure 1. SNAP 10 System

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Figure 2. Core Assembly

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The thermoelectric converter-radiator assembly consists of 40-tube sub-assemblies all mounted on the support structure. The tubes carry the coolant which provides the heat to be converted by the thermoelectric elements into electrical energy. The temperature differential developed between the hot and cold junctions of the elements results from the cold-junction radiator which rejects waste heat to the vacuum chamber wall. The thermoelectric elements are connected electrically to provide the load power at the requisite voltage.

The ground test modifications to the FS-3 included:

- 1) The springs which drive the coarse drums in all the flight system were removed. Coarse drum drives were installed which permit motorized operation of the two coarse drums in either direction.
- 2) The reflector assembly band device was modified to reduce the band tension and to incorporate a pin which will release the band when actuated by a pneumatic pin puller.
- 3) The temperature actuated band release device on the reflector assembly was not installed.
- 4) A NaK drain line was provided so that the system could be drained at the conclusion of the test.
- 5) A 1-curie neutron source, which approximates the startup source level in space, was installed.
- 6) An expansion compensator adapter was installed to compensate for gravity effects during ground testing.
- 7) The reflector assembly was modified to allow for the remote removal of the reflectors.
- 8) The 28-volt power for the instrument compartment was supplied from an external source. The electrical output of the FS-3 system was dissipated through electrical resistors.
- 9) The flight system reactor controller was located outside the test vault and an identical dummy controller was installed in the instrument compartment.
- 10) The demodulators for the drum position sensors and the expansion compensator position sensors were located outside the test vault.

B. FACILITY

The FS-3 test was conducted in the SETF. This facility was designed to support the nuclear testing of SNAP systems operating at their design condition, and to facilitate the systematic removal of such systems after testing. An artist's concept of the facility is shown in Figure 3.

Those areas of the facility in which the disassembly of the system took place are the high bay, access area, and vault complex. The vault complex is a subsurface, parallel assembly of three shielded cells. An access area is adjacent to a common face of all cells, and provides access to these cells. This area also acts as the operating gallery for remote handling operations. The high bay area is that section of the building built above the subsurface cell complex and operating gallery. The building is equipped with a 5-ton bridge crane; this crane has sufficient reach not only to move material at ground level, but also to service the operating gallery 35 ft below. To handle the cell roof plugs and heavy items such as shielded casks, a 20-ton gantry crane is available. A door in the east wall of the high bay area enables the gantry to move heavy and bulky plugs and casks outside the building onto a concrete storage apron.

The cell complex consists of the three parallel cells (two power test cells and the center transfer cell), a partial rear corridor which interconnects the cells, and an adjacent front area known as the operating gallery, or lower access area. The facility floor plan is shown in Figure 4.

The cell walls and roof are of borated concrete construction of varying thicknesses. The top of the 8-ft-thick roof is at grade level, and serves as the floor of the high bay area. Walls which separate the three cells are 4-1/2 ft thick, and the front walls of the two test cells have a thickness of 9 ft. A 3-ft-thick partial wall, referred to as a labyrinth wall, is built across the rear of each test cell. The corridor formed by the labyrinth walls is called the transfer corridor, since the device for moving the rectilinear manipulator from cell to cell travels through this corridor. Slots are provided in the labyrinth walls to allow clearance for the bridge of the manipulator as it passes from the corridor into the test cells. The primary purpose of the labyrinth walls is to afford shadow shielding to the area where the corridor opens into the center cell. Sliding doors are provided to completely seal off the cells from one another by closing off the corridor where it intersects the cell. The sealing is accomplished

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Figure 3. Cutaway of SETF Building

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Figure 4. SETF First-Floor Plan

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by use of a double set of inflatable seals mounted in the jambs. A single screw shaft on each door, driven by motor or manually from outside the cell, is used to open and close the door.

To ensure gas tightness, the cells are lined on the inside with aluminum plate. The integrity of the cell lining is such that the maximum atmospheric leakage from any cell is less than 1% of the cell volume in 24 hr at a pressure differential of +3 in. of H_2O with reference to the access area.

The test cells and the center cell are of different size; the test cells have a floor area of 15 ft by 15 ft, the center cell is 15 ft wide and approximately 26 ft long (including the corridor in the rear which is not separated in this cell). All three vaults are 20 ft high.

The outside face of the cells is serviced from an operating gallery and access area. Access into each of the test cells from the operating gallery is possible by means of plug doors. The center cell front face is penetrated by a plug door and also by a shielded window. The remote-handling operating console is located in the vicinity of the shielded window. Over the window two master slave manipulators penetrate the cell face through shielded and sealed tubes.

The front access doors are stepped plugs of the same thickness as the shielded walls which they penetrate. The doors fill 4-ft by 7-ft openings which are provided for personnel or equipment access from the operating gallery into each cell. Tracks are provided in the floor for the self-powered plugs to roll on. In the event of a failure in the electrically powered moving mechanism, provision is made for cranking the plugs open. Sealing of the doors is accomplished by a gasketed seal plate which is bolted over the door opening after the plug has been inserted.

The roof of each cell is penetrated by a 9-ft circular access port. The plugs filling this port are stepped for shielding purposes. In order not to exceed the capacity of the 20-ton gantry provided to handle the roof access plugs, the shielding in each port consists of three layers of plug which make up a total thickness of 8 ft. The roof access port minimum diameter, 9 ft, was sized to allow for lowering a shake table, vacuum chamber, or other environmental equipment into the cells. The center of the plugs which fill the roof access ports

are penetrated by a 24-in. minimum-diameter port which is available for insertion of special remote handling tools for removing radioactive material into shielded casks.

The roof, front face, and walls separating the cells from one another are all pierced by 8-in. and 10-in. minimum-diameter stepped ports. These ports are provided for a variety of uses: the roof ports are used for penetration of the periscope or "through-roof" television cameras; the front face ports are used as transfer drawers, process piping penetrations, etc. The 8-in. and 10-in. access ports are sealed by O-ring-sealed cover plates which are installed after shielding plugs have been inserted in the ports.

C. TEST INSTALLATION

The FS-3 was installed in a vacuum chamber in power test vault No. 2 of the SETF. This vacuum chamber provided a simulated space environment and its walls provided the cold surface for the radiation of heat from the system. The walls of the vacuum chamber were cooled by a chilled kerosene system. Figures 5 and 6 present views of the FS-3 installed in the test vault.

The presence of the vacuum chamber presented specific problems to system disassembly. The vessel was made of aluminum so that residual activity would be minimized, allowing the sections to be handled with no shielding. The bolts mating the vacuum vessel sections were designed to be loosened by remote means, and all connections to the vacuum vessel were located so that they could be decoupled or severed by remote means.

D. REMOTE HANDLING

The SETF was designed so that radioactive systems could be remotely disassembled and removed from the facility. A rectilinear manipulator is installed within the vault complex and can be transferred from vault to vault through the interconnecting doors and transfer corridor. This transfer capability allows the manipulator to operate in any cell, and also to transfer radioactive materials from one vault to another.

The two power test vaults have no direct viewing provision. A system of remotely controlled television cameras allows viewing of operation in these vaults. The vaults have wall-mounted rails to receive television cameras, which

Figure 5. FS-3 in Test Vault Prior to
Vacuum Chamber Section
Installation

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Figure 6. FS-3 in Test Vault with
Vacuum Chamber Partially
Installed

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may be placed on these rails by the rectilinear manipulator. Television cameras left in the power test cells during operation would be rendered inoperable by radiation, so cameras are placed in the cell by remote means after termination of operation. Figure 7 is a view of power test vault No. 2 showing the section of the television camera mounting which remained in the vault during reactor operation. Shown in this view is a portion of the wall mounted rack which permits horizontal traverse of the mounting. The screw jack permits vertical traverses of the pad upon which the camera is placed. The drive motors and cabling associated with the control of these traversing devices are transferred as an integral unit separate from the camera. Figure 8 is a view of the camera, control drive, and mounting assembly installed in a full-scale mockup of the test vault.

To cope with specific problems encountered in the disassembly of the FS-3, a second remotely controlled manipulator was procured. This manipulator had polar motion, and was sized to rest on one of the lips of the stepped opening in the ceiling of the vault. When positioned in this opening, this manipulator, together with the rectilinear manipulator, allows the performance of operations requiring simultaneous, independent manipulation.

The removal of the vacuum chamber sections from the vault required removal of the shield plugs from the roof of the test vault. Operations had to be performed over this opening while the radioactive core remained in the vault. To permit the remote removal of the vacuum chamber sections, a vacuum vessel handling assembly was fabricated. This assembly could be lifted and positioned by the gantry crane in the high bay, and contained within it a separate, remotely controllable crane to allow a vacuum vessel section to be lifted into the body of the assembly. The gantry crane was modified so that it could be operated from a remote station. Figure 9 is a sketch of the vacuum vessel handling assembly positioned in the roof opening of the power test vault. With the gantry crane modified for remote operation, it was possible to position the polar manipulator in the vault ceiling opening, and to place the auxiliary shielding above this manipulator. Figure 10 shows the vault configuration with the polar manipulator and auxiliary shielding in place.

The control station for the manipulators and television equipment is normally in the lower access area immediately in front of the center cell. A view

Figure 7. Power Test Vault No. 2
With Camera Mounting

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Figure 8. Television Camera,
Drive, and Rack Assembly

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Figure 9. Vacuum Chamber Head Removal

Figure 10. FS-3 After Vacuum Chamber Removal

of this station is shown in Figure 11. For special circumstances, the control of a manipulator or television camera may be placed in other portions of the facility by proper cabling.

Figure 11. Control Station for Manipulators and Television Equipment

III. FS-3 DISASSEMBLY

A. DISASSEMBLY PLAN

The plan for the disassembly of the FS-3 system centered around the desire to remove the reactor core as a unit. This plan would allow the reactor vessel to be opened, and the core internals examined in the AIHL with a minimum of prior handling of the internal components. The initial portion of the disassembly effort was to be devoted to placing remotely controlled television cameras in the power test cell, and to removing the three-sectioned vacuum vessel which contained the SNAP unit. After exposing the system, the reflector sections were to be removed, and then the core and thermoelectric pump assembly cut off of the system. This core and pump assembly was then to be placed in a shipping cask and shipped to the AIHL. The remainder of the system would be remotely disassembled and packaged in shielded casks. Remote disassembly would be continued until the radiation level was reduced to a point where manual entry into the test vault could be made.

B. INITIAL PREPARATION

The month immediately after test termination was devoted to preparation for the disassembly effort. The polar and rectilinear manipulators were checked out. The facility cranes and vacuum vessel handling assembly were proof tested. The vacuum vessel coolant system was drained and the manipulator boots for the master-slave manipulators were replaced. The equipment necessary for the disassembly operation was staged in the center cell, which would act as the support cell for the operation. Figure 12 shows the layout of some of the special tooling in the center cell.

Personnel access to the center cell during disassembly was to be through vault No. 1, so a contamination control and stepoff area was established in the rear corridor of this vault. Figure 13 is a plan view of the vault complex, showing the access to the center cell from vault No. 1 through the corridor door.

C. TEST VAULT PENETRATION AND TELEVISION INSTALLATION

The initial remote operation associated with system disassembly was the placement of television cameras within the vault by the rectilinear manipulator

Figure 12. Layout of Equipment in Transfer Cell Prior to Disassembly

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Figure 13. SETF Basement Plan

Figure 14. Pedestal-Mounted Television Camera

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so that subsequent operations could be viewed. Figure 14 is a picture of the first camera transferred into the vault. This camera was designed to be placed on the floor at the end of the vault No. 2 corridor, giving a view of the west wall of the vault proper. During this operation, a transfer bridge interlock malfunction occurred which prevented driving the manipulator back into the center cell. The interlock circuits were bypassed in the control console and the manipulator then was driven into the center cell for maintenance. All travel interlock switches were removed, cleaned, lubricated, adjusted, and reinstalled to prevent a recurrence of such a malfunction.

During all such maintenance operations, personnel entry into the transfer cell was made with due regard to contamination control and radiation safety. All entry of personnel into possibly contaminated or radiation areas, and removal of equipment from such areas was done under strict contamination control and radiation safety practices. For the purpose of brevity, such control and monitoring activities will not be specifically discussed except when they substantially affected the operation.

Upon completion of the manipulator maintenance, transfer of the television equipment into the test vault was resumed. A television camera and drive equipment were mounted on the west wall television rack and the cabling plugged into the in-cell electrical connectors. After installation, it was found that all camera and drive functions were inoperable. The camera assembly was removed from the test vault and installed in a cell mockup for detailed trouble shooting. The malfunction was corrected and the camera unit reinstalled in the test vault.

Further functional testing of the west wall television installation revealed limited vertical drive and no traverse drive capability. Investigation showed that the vertical drive rack seemed to be sitting askew. The limited viewing possible with the one pedestal-mounted television camera, located at the end of the corridor, did not allow a complete definition of the specific problem.

One of the through-roof plugs, located in the southeast corner of the test cell, was removed and a periscope installed. The view of the top mounting of the television rack showed that the top guide rollers were off the track, effectively binding the rack into immobility. A through-roof plug was removed from the northwest corner of the cell, and a through-roof television camera installed

to provide additional viewing. The attitude of the rack was reproduced in the full-scale mockup of the test cell to aid in arriving at a solution to the problem.

It was decided that a grappling tool could be lowered through the southeast through-roof port and used to return the rack to its normal position. A grappling tool was fabricated, engaged with the untracked rack, and a Ballymore lift was used to provide sufficient lift to free the rack from its jammed position. The rack was then repositioned, and subsequently satisfactory operation was verified. The utilization of these cell penetrations to successfully solve an in-vault problem illustrates the flexibility of operation possible in a properly designed facility. These penetrations had been provided for just such situations as this.

A contributing cause of the rack binding problem was that the television drive unit fit too snugly on the traversing rack. Remote placement and removal of the drive unit applied sufficient force to the rack to cause it to lift and tip on its tracks. To prevent a recurrence of this problem, all drive units were modified by smoothing and rounding all locator pins and possible interfering corners and edges.

D. VACUUM SYSTEM REMOVAL

The next step in the disassembly of the FS-3 was to remove the vacuum chamber sections to expose the system proper. An impact wrench was engaged by the manipulator and transferred from the center cell to the test cell. This wrench was used to initiate the removal of the vacuum vessel bolts from the dome of the vessel. Figure 15 shows an array of impact wrenches modified for use with the manipulator. Figure 16 is a view of the interior of the test cell, showing the bolted flanges of the vacuum vessel and the cooling lines which had to be decoupled or severed for vessel removal. At this time, flange bolt removal was initiated. Five of the dome section flange bolts were retained to keep the vessel in place during subsequent preparation activities prior to vacuum vessel removal. The impact wrench was then replaced with an abrasive saw so that external lines could be severed from the vacuum vessel.

During the bolt removal operation, several transfers of the manipulator between the center cell and the test vault took place. These transfers were necessitated by malfunctions of the impact wrench adapter and by the placement of the

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Figure 15. Remote Control Impact Wrenches

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Figure 16. Interior of Test Vault

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east wall television camera, which was required to improve viewing conditions on that side of the vacuum vessel.

After severing all external lines to the vacuum vessel, the remaining flange bolts were removed. This operation completed the in-vault preparation for the vacuum chamber removal; it was now possible to lift individual sections without interference from external connections.

A shielded cask was installed in the transfer cell to receive the plutonium beryllium startup source, which was mounted adjacent to the reactor. The shield blocks were then removed from the roof of the test vault. The removal of these blocks exposed the radioactive cell environ and raised the radiation level in that area of the high bay in the vicinity of the open vault so that personnel access was precluded. The vacuum vessel handling assembly was positioned over the vault by the remote-control gantry crane, using previously established index marks and positioning tapes. The rectilinear manipulator performed the in-cell hookup of the vacuum vessel handling assembly crane to the dome of the vessel. This dome was then hoisted into the handling assembly which was then moved by the gantry crane to the accessible end of the high bay where the dome section was lowered into a storage box. It was possible to handle the vessel sections without shielding, since the vessel was made of low activation material (aluminum) and was a source of only low direct radiation. The thermon used to thermally bond the cooling coils to the vessel did contain activated constituents, and was the source of possible contamination, so the vessel sections were handled only over plastic laydown pads, and were sealed in plastic-lined boxes for temporary storage.

Removal of the vacuum vessel dome exposed the system to view and to access by the manipulator. The source and source holder were removed and placed in a transfer cask. This cask was then removed from the cell complex and placed on the outside gantry pad for shipment to the Radioactive Materials Disposal Facility (RMDF).

The Electrically Actuated Band Release Device (EABRD) was fired, allowing the reflector halves to separate and tilt away from the core. This placed the reactor in a far subcritical condition, and also allowed the removal of interconnecting linkages between the reflector assembly and the second vacuum section. This action removed the last restraints on vacuum section removal, and

the two lower sections were sequentially transferred to storage boxes, using the handling assembly and the procedures described previously.

E. SYSTEM DISASSEMBLY

The core transfer cask, secondary core container and core handling equipment were placed in the center cell. The polar manipulator was moved into position in the roof opening of the test vault, and the supplemental shield blocks placed over the opening. This action gave additional handling capability within the cell, permitted access to all areas of the high bay again, and reestablished the atmospheric integrity of the test vault. Until this point in the disassembly operation, the control of the test vault atmosphere had been aimed at precluding the spread of contamination. This had been accomplished by exhausting the cells at the rates, and from the exhaust points, required to maintain air flow from the less to more contaminated areas. The next operations would breach the liquid metal system, with the possibility of gross NaK spillage, and the assurance of residual NaK atmospheric exposure. Therefore, an atmospheric exchange of the two cell complex was made to reduce the oxygen concentration to less than 3%. As an additional precaution against NaK fires, a supply of calcium carbonate had been placed in the center cell, and drip pans were placed at points of possible NaK leakage.

The liquid metal was drained from the system without difficulty. The draining was accomplished by using a drill motor operated by the rectilinear manipulator to activate the drill-through drain valve. This valve was one of the ground test modifications to the system, and is shown in the sketch of Figure 17. Figure 18 is a photograph of this valve as installed in the system. The system was drained into a tank preplumbed for this purpose, with successful draining being verified by the activation of conduction probes located in the drain tank.

After the successful drain of the system, the reflector assembly was removed. Wiring to the reflector was removed by the use of an abrasive saw on the rectilinear manipulator, or by the use of a cutting hook on the polar manipulator. Each half of the reflector assembly was lifted in turn from the system, and transferred to a shield cask in the center cell. Figure 19 is a photograph of a reflector half suspended from the manipulator.

Figure 17. NaK Drill-Through Drain Valve

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Figure 18. Drill-Through Drain Valve Installation

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Figure 19. Reflector Half Suspended
from Rectilinear Manipulator

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Figure 20. Core-Pump Unit in AIHL

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The next operation performed was the removal of the reactor core-pump assembly which was held by the polar manipulator while the core support legs were cut by the abrasive saw operated by the rectilinear manipulator. The core-pump assembly was then lowered into the secondary core container, which in turn was transferred to the center cell. Here it was placed in the core shipping container and readied for removal from the facility.

The afterglow heating of this reactor was quite small, so there was no requirement for heat removal during shipment. A double O-ring seal arrangement trapped the oxygen-depleted cell atmosphere within the shipping container, and also prevented the release of contamination. The primary function of the shipping container was to provide shielding and to protect the core from mechanical shock. The core was removed from the cell complex and transported to the AIHL on May 23, 1966. Figure 20 is a photograph of the core-pump unit as received by the AIHL.

The next section of the system to be removed was the shield with its surrounding support structure. The removal was accomplished by cutting through the supporting cone with the abrasive saw operated by the rectilinear manipulator while the shield assembly was supported by the polar manipulator. After the assembly was severed it was transferred by the rectilinear manipulator into a shielded cask in the center cell.

A remote radiation survey of the test vault indicated that the ground test heaters were the primary remaining source of radiation. They were removed remotely, placed in a cask, and sent to the RMDF. A subsequent radiation survey indicated that personnel could enter with proper suiting and breathing apparatus, and under controlled exposure conditions. The remainder of the disassembly effort and subsequent facility decontamination was accomplished using standard controlled practices.

F. EVALUATION

The radiation levels of components handled during the disassembly effort were well predicted by previous calculation (Table 1). This enabled the planning for the examination or disposal of the system portions to be accomplished effectively. Table 2 presents the radiation level of some components as shielded for shipment or storage.

TABLE 1
RADIATION LEVEL COMPARISON

Component	Calculated, After 60 Days, (r/hr)	Actual (r/hr)
Source		25 at 5 in. after 52 days
Reflectors (Be only)	0.5 at 3 ft	16 at 6 in. after 66 days
Actuators	1.7 at 3 ft	22 at 4 in. after 66 days
Shield	0.36 at 3 ft from top surface	7 at 2 in. from top surface after 77 days
Electric Heaters		25 at 2 in. after 77 days
Electrically Heated Thermoelectric Modules	0.02 at 3 ft from upper end	0.19 at top of upper module 0.075 at top of lower module after 60 days
Converter Modules	0.02 at 3 ft from upper end	0.05 at surface after 77 days
'+Z' Expansion Compensator	0.007 at 3 ft	0.45 to 0.6
'-Z' Expansion Compensator		0.5 to 0.7 after 107 days

Note: The distance from the radiation source is not the same in the calculated and measured cases. A $1/x^2$ factor should be applied for more direct comparison.

TABLE 2
RADIATION LEVELS

Sealed Component Cask	Radiation Level (mr/hr)
Sealed Component Cask	
Source	47 maximum
Core	~200 general 400 at 1 in. from bottom maximum 50 at 1 meter maximum
Reflector	250 at surface 470 maximum general
Shield	1.02 r/hr maximum 700 at surface general
Vacuum Chamber Sections	
Head	80 maximum at brass valve 25 at surface of storage box
Center section	80 maximum at 2 in. from brass valve 20 (general) at 2 in. from surface
Lower section	15 (general) at 2 in. from surface

The disassembly effort was successful in that it retrieved components of the system intact so that subsequent detailed examination would be meaningful. The effort was completed within the scheduled interval (Figure 21). Contamination control and cleanup was successful so that the facility is now reusable with minimum ingress control. The problems encountered in the initial placement of viewing equipment, and the solutions to these problems, point out the advisability of having secondary access to hot cells, and the desirability of using full-scale mockups of in-cell conditions.

Figure 21. System Disassembly History and Schedule

IV. COMPONENT EXAMINATION

A. SELECTION OF COMPONENTS FOR EXAMINATION

The operation of the FS-3 exposed the system components to an operating environment of radiation, elevated temperature, and vacuum. Although the operation was lengthy, the low power density at which the reactor operated kept the total radiation exposure of components well within the range of material experience for most components. The temperature at which the system operated was well below that at which similar liquid metal systems have been operated. Previous testing of individual SNAP components gave a background of data on the behavior of materials under conditions of temperature and vacuum. Therefore, the examination of components was primarily limited to those components where background data was lacking or where particular problems had been evidenced. In some cases, an examination was made to confirm that no component degradation had occurred in areas where none was anticipated.

Table 3 presents the list of components selected for evaluation, together with the total fast neutron dose to the component, the type of examination given the component, and the information desired from the examination. The neutron dose listed for the components is that calculated from reactor leakage considerations.

The examination of the fuel was performed in the AIHL, and is reported separately.⁽¹⁾

B. CORE VESSEL

Visual inspection of the core vessel interior revealed no evidence of corrosion or NaK reaction. Aside from a layer of residual NaK oxide on the bottom, the vessel was clean and bright inside. There was some darkening of the outside surface, probably due to oxidation from atmospheric residuals within the vacuum chamber.

Two samples were cut from the Type 316 SS core vessel for metallographic examination. Typical postoperation microstructures are presented in Figure 22. No leaching by the NaK or oxide products was noted. Sensitization had started, but had not progressed to a significant level. Additional inter- and intragranular precipitation was observed but there was no evidence of any sigma phase material.

TABLE 3
SUMMARY OF COMPONENTS EXAMINED

Component	Type of Examination	Technical Information to be Obtained	Calculated Fast Neutron Exposure
Reactor Vessel	Visual examination. Metallographic examination of vessel wall.	Examination of vessel material will yield information on sensitivity of Type 316 SS to pitting corrosion and stress corrosion cracking during reactor operation. Tensile tests will indicate any changes in strength and ductility which the cold-worked, sensitized material had experienced.	2.0×10^{19}
Internal Reflectors	Examine for evidence and/or extent of diffusion. Determine extent of Be swelling.	Data on beryllium swelling due to irradiation is sparse. Additional information is required correlating swelling with neutron flux and temperature. Beryllium diffusion and material deposition was noted in S2DR and no adequate explanation of the phenomena was made.	2×10^{19}
Core Hold-Down Springs	Dimensional measurements. Force-deflection measurements. Visual inspection for corrosion or mass transfer. Metallographic examination.	Force-deflection measurements will yield data for predicting stress relaxation behavior of spring material in a SNAP environment.	1.2×10^{19}
Reflector and Reflector Components	Photographic and visual inspection for sticking condition of mating surfaces, condition of actuator connectors, and condition of gears. Measure backlash in fine control drum drive system. Measure drum rotational torque.	These tests will provide long-term information on material self-welding, oxidation, and sublimation effects.	
1. EABRD	Fire EABRD and measure power and time to separate.		2.1×10^{18}
2. Control drum bearing assembly	Photograph bearings, bearing components, and shaft surfaces.		3.6×10^{18}
3. Actuators	Measure pin-to-pin resistances. Torque measurements.		1.7×10^{18}
4. Position sensor	Measure torque.		1.7×10^{18}
Shield	Observe for distortions. Dimensional check. Remove LiH and visually check for cracks, surface condition, etc. Measure density of LiH samples from top and bottom.	The data will provide information on the affect of radiation on the growth of LiH.	3.5×10^{18}
Handlebar Assembly	Visual and photographic examination of the assembly. Thermal switch wells, examine for corrosion, erosion, and weld integrity. Metallographic examine welds of lowest quality. Pinch off and seal welds, metallographic examination.	Examination is primarily concerned with the evaluation of containment integrity of the components following prolonged exposure to high temperature flowing NaK. The evaluation of erosion, mechanical fatigue of materials, and containment integrity of joints following the system test will be used to verify previous simulated tests and design evaluation. The weld joints on the temperature switch are the same as the welds that failed on FSM-4 and will contribute information towards understanding what occurred on FSM-4.	
ECU	Visually inspect and photograph. Weigh bellows to determine which is heavier. Section and examine the heavier unit. Locate area of leak. Metallographic examination.	The information is required to determine the mode and cause of the leak between the primary and secondary bellows.	2.9×10^{16}
Thermoelectric Converter	Electrical resistance measurements. Tensile tests. Metallographic examination. Visual examination of individual modules.	The resistance measurements will yield information on the change in resistivity of the SiGe element at design temperature and in a radiation environment. The tensile tests will determine the effect of irradiation on the strength of the bonds of individual stacks. The metallographic examination will yield information on the effect of irradiation (if any) on void formation in the shoe bond and on the consumption rate of copper at the radiator bonds. The visual examination will yield information on the effect of radiation on the coating at the Al_2O_3 insulators.

Figure 22. Postoperation Microstructure of Type 316 SS Samples from the FS-3 Core Vessel

65-1-1

This examination confirmed that the Type 316 SS was relatively unaffected by exposure to the system operating conditions.

C. INTERNAL BERYLLIUM REFLECTORS

The six beryllium internal reflectors were removed from the core, cleaned of NaK oxide, and visually inspected. Typical reflector appearance was as shown in Figures 23 and 24. The reflector surfaces were free of defects, but did exhibit a number of small black stains which appear to have occurred during cell handling and residence. Some of the spots were removed or altered by wiping.

The density of each of the six reflectors was determined by the liquid displacement method using distilled water as the fluid. Each density measurement was made three times and averaged. The data are summarized in Table 4.

Two samples were removed from internal reflectors No. 3 and No. 6 for metallographic processing. Figure 25 presents typical postoperation microstructures for the beryllium reflector samples. No cracks or major structural damage was noted, but porosity at the grain boundaries was observed in all four samples examined. This porosity cannot be ascribed to radiation-induced

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Figure 23. Typical Internal Reflector

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Figure 24. FS-3 Typical Internal Reflector Surface

TABLE 4
INTERNAL REFLECTOR DENSITIES

Reflector	Density (gm/cc)			Average
	Measurement No. 1	Measurement No. 2	Measurement No. 3	
1	1.8578	1.8578	1.8584	1.8580
2	1.8584	1.8585	1.8584	1.8584
3	1.8585	1.8584	1.8585	1.8585
4	1.8581	1.8581	1.8580	1.8581
5	1.8572	1.8572	1.8572	1.8572
6	1.8574	1.8575	1.8575	1.8575

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Figure 25. Postoperation Microstructure
of Beryllium Samples from the FS-3
Internal Reflectors (Bright
Field Illumination)

swelling without direct comparison between unirradiated and irradiated samples, since unirradiated beryllium typically shows grain boundary porosity.

D. CORE HOLD-DOWN SPRINGS

Spring deflection tests were performed on 10 unirradiated and 12 irradiated core hold-down springs. Each Rene 41 spring was placed between the jaws of a remote-operated Instron TT-C testing machine which was used to apply equal increments of compressive load. The deflection produced at each load increment was subtracted from the original spring free length to obtain successive spring absolute lengths. Figure 26 shows a plot of absolute length vs load, in which each point is the average of all spring values at a particular load. Both the unirradiated and irradiated curves show a linear decrease of spring length with increasing load until a point is reached at which smaller deflections for the same increment of load were measured. At this point, the springs were partially nested. Where no increase in deflection was measured with an increase in load, the springs were fully closed. The slope of the linear portion of each curve represents the spring constant. This was determined to be 344.8 lb/in. for both unirradiated and irradiated springs. The average nested, or solid length of the spring was measured to be 0.424 in. for unirradiated springs, and 0.436 in. for irradiated springs. Variations in nested length are probably due mainly to differences in initial lengths, and are dependent on the number of coils in each spring. Springs could vary considerably in initial free length, since the fabrication tolerance was ± 0.032 in.

Measurements of spring free lengths were taken before and after testing. Unirradiated springs averaged 0.618 in. (range 0.616 to 0.620 in.) before testing, 0.620 in. (range 0.618 to 0.622 in.) after testing. Irradiated springs averaged 0.593 in. (range 0.577 to 0.603 in.) before testing, 0.594 in. (range 0.573 to 0.607 in.) after testing. In no case did the posttest measurement vary by more than ± 0.002 in. from the pretest measurement.

The difference in free lengths between the radiated and unirradiated springs can be attributed to the fact that the irradiated springs were operated under a compressive load at $\sim 1000^\circ\text{F}$ throughout the test, resulting in a permanent set. The unirradiated springs had no comparable environment.

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Figure 26. Core Hold-down Springs — Length vs Load

Following these tests, samples were cut from three irradiated springs and two unirradiated springs for metallographic processing. Figure 27 presents typical posttest microstructures for both unirradiated and irradiated Rene 41 samples. The unirradiated material (Figure 27a), exhibited inter- and intra-granular precipitation, especially in large stringers oriented parallel to the centerline of the coil. Grain size varied greatly within each sample, as can be seen by the layers of coarse-grained material surrounded by fine-grained material. The irradiated material shown in Figure 27b was very similar in appearance to the unirradiated. The only difference noted between the irradiated and unirradiated material was an apparent increase in the amount of work present in the irradiated samples, as shown in Figure 27c.

It is concluded, therefore, that the Rene 41 core hold-down springs were not damaged by reactor operations. No difference in spring constant was noted in unirradiated and irradiated springs, and the springs returned to the original free lengths after compression, indicating elastic behavior. No significant difference was detected from metallographic examination.

E. RADIATION SHIELD ASSEMBLY

The radiation shield was decanned under a nitrogen atmosphere in the transfer cell. Prior to decanning, the internal pressure in the shield vessel was measured to be about 430 torr. This was slightly lower than the original 1 atm pressure of helium (760 torr). A similar reduction in internal pressure had been observed in an earlier, similar SNAP 10A cold-pressed lithium hydride shield which had been subjected only to thermal cycling tests. Therefore, the change in pressure is not believed to be radiation induced, but rather is caused by slow leakage of the helium into the test chamber vacuum.

The shield vessel was cut with an abrasive wheel below the closure weld. The domed bottom was then removed revealing the base of the shield. The springs were severed releasing the block and the upper portion of the vessel removed. Dimensional measurements were made of the block, and samples for chemical analysis were removed with a hammer and chisel at the top, middle, and bottom. It was noted during sample removal that the block was hard and solid similar to an unirradiated block. Sampling was done by creating a fracture in the block with the chisel edge.

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a. Unirradiated Sample

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b. Irradiated Sample

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c. Irradiated Sample Showing Work

Figure 27. Microstructure of Rene 41 Samples from the Core Hold-down Springs

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Analysis of the dimensional measurements revealed that the block had not changed in dimensions within the accuracy of the measurements which was about $\pm 2\%$.

1. Shield Spring

The neutron shield was supported by eight tension springs. The spring support was provided to give freedom of movement during thermal cycles. A static tension load of 245 lb was imposed on each of the springs, which operated between 400 to 600°F. During shield decanning one of the neutron shield springs was found to have failed. The failed spring is shown in Figure 28. It fractured in a 45-deg helix. The fracture surface, shown in Figure 29, consists of two smooth areas along the outside periphery of the spring and coarse-textured area for the remaining section. Figure 30 shows that the fracture occurred transgranularly. The surface of the spring, although tarnished, was free from oxide scale. Some tool marks were found on the surface as shown in Figure 31.

The characteristic and mode of fracture indicates that the neutron shield spring suffered from fatigue failure. The smooth areas were the source of the fatigue cracks, and the 45-deg helix indicates subsequent cracking from alternating torsion loads.

A review of the spring fabrication procedure indicated that the heat treatment was an aging of 16 hr at 1350°F. The current recommended stress equalization procedure for Hastelloy X consists of 1 hr at 1750°F and air cooling. Therefore, it appears likely that the residual stresses resulting from coiling of the spring were not effectively removed during aging. This raised the mean operating stress above the approximately 50,000 psi due to the 245-lb load imposed and was responsible for shortening the spring fatigue life.

The tool marks found on the spring had a notch effect which was detrimental to the fatigue life of the spring. Figure 32 shows the endurance limit for a Hastelloy X bar with a smooth surface to be in excess of 70,000 psi, but lower than 30,000 psi for notched bars. The tool marks on the spring lowered its fatigue resistance and were responsible for initiating the fatigue cracks.

Based on the evidence revealed in the examination, it is concluded that the neutron shield spring failed because of fatigue. The cyclic loads which were imposed on this spring, and which could have attributed to fatigue failure, were

Figure 28. Failed SNAP 10A Lower Shield Spring (Full Size)

Figure 29. Fracture Surface of Spring
(A. Origin of Fatigue Cracks,
B. Fracture Area)

Figure 30. Microstructure of Failed Spring Showing Transcrystalline Failure Due to Fatigue

Figure 31. Tool Marks on Surface of Springs

Figure 32. Effect of Heat Treatment on Fatigue Life of 3/4-in. Hot-Rolled Bar (Samples were tested in completely reversed bending)

those associated with acceptance and qualification level shock and vibration testing, and those imposed by thermal cycling during thermal acceptance tests.

2. Activation Analysis of Shield Vessel

An activation analysis was performed on samples of the shield vessel to verify calculated radiation exposures. A spectrographic determination of the Mn^{54} gamma activity, from the $\text{Fe}^{54} (n,p) \text{Mn}^{54}$ reaction, yielded a fast (>1 Mev) exposure estimate of $2.6 \times 10^{18} \text{ n/cm}^2$ for a sample taken from the top of the shield casing. This is to be compared with a calculated value of $3.5 \times 10^{18} \text{ n/cm}^2$. This comparison indicates that the dose estimates given in Table 3 are reasonable values for component exposure.

A spectroscopic determination of the Co^{60} activity, from the $\text{Co}^{59} (n,\gamma) \text{Co}^{60}$ reaction, yielded a thermal neutron exposure estimate for the shield case samples. Samples from the top and bottom of the shield case were analyzed. Results are presented in Table 5. It is to be pointed out that the exposure of the sample from the bottom of the shield case is not that which would be expected from system operation in space. Scatter from the test vault walls exposed the volume under the shield to doses approximately 10^3 greater than would be experienced in space.

TABLE 5
ACTIVATION ANALYSIS OF THE FS3 SHIELD VESSEL

Position	Fast Neutron ($E > 1.0$ Mev)		Thermal Neutron	
	Flux (n/cm ² /sec)	Dose* (n/cm ²)	Flux (n/cm ² /sec)	Dose (n/cm ²)
Top	7.16×10^{10}	2.58×10^{18}	1.36×10^{10}	4.9×10^{17}
Bottom	2.47×10^8	8.9×10^{15}	1.04×10^9	3.75×10^{16}

*10,000 hr or 3.6×10^7 sec

3. Lithium Hydride Block

When the lithium hydride sample container was reopened for chemical analysis it was observed that the samples had picked up a fairly heavy layer of whitish hydride contamination which was not present during the original sampling. The results of the analysis for gross sample constituents are shown in Table 6. An attempt was made to remove the contamination by scraping the surface and another analysis was made. The scraping of the surface was partially successful, as noted in Table 6 by the difference in the hydroxide values. No further attempts were made to remove the surface contamination, but it is believed that the uncontaminated lithium hydride contained no more than 2 to 3% hydroxide. This is based on the physical appearance of the material when first removed from the shield container.

The lattice spacing of the lithium hydride was determined by x-ray diffraction to be 4.085 Å. Unirradiated lithium hydride samples have been determined to have a lattice spacing of 4.084 Å. Samples removed from both the top and bottom sections of the shield were determined to have the same density (0.78 ± 0.01 gm/ml), which agrees with the established 0.775 gm/ml for unirradiated lithium hydride.⁽³⁾ This confirms that no mechanical damage or material swelling was induced by the radiation environment. Based on the results of the examination of the shield assembly components, it can be concluded that exposure to the operating environment of the system had no adverse effects on the performance of the shield. This is excepting the broken support spring.

TABLE 6
CHEMICAL ANALYSIS OF THE IRRADIATED LITHIUM
HYDRIDE FROM THE FS3 SHIELD

Constituent	Location of Sample	
	Top	Bottom
Hydrogen (wt %)	11.65 [*]	11.90 [†]
Lithium (wt %)	82.10	82.55
Hydroxide (wt %)	4.65 (4.99) [§]	3.38 (4.08) [§]
Carbonate (wt %)	0.04	0.03

*Equivalent to 92.0 wt % LiH

†Equivalent to 93.8 wt % LiH

§Original analysis values

F. EABRD

During the disassembly of the FS-3 the EABRD was fired to release the reflector halves. Voltage and current traces were recorded during the firing. The firing required 2300 msec of electrical energy. The energy required for firing, and the current vs time traces were consistent with firing requirements of new, unirradiated EABRD's, indicating that the device was not significantly affected by exposure to the system environment.

G. ECU

After approximately 2500 hr of system operation, a decrease in ECU deflection was observed. The deflection indication stabilized at a position corresponding to a change in system volume of 50 in.³, which is the void volume between the primary bellows and the containment can of one ECU. Gamma scanning of gaseous effluent cold traps and vacuum vessel pressure decay tests gave no indication that NaK was leaking from the system, so it was concluded that the change in compensator deflection was caused by a leak of one ECU into its secondary containment. This secondary containment had been incorporated into the design for just this eventuality. Postoperation examination of the compensator confirmed this conclusion.

After removal from the system, the failed compensator was cleaned of NaK, and the bellows assembly was separated from the container. Visual examination revealed a rupture slit along the root of the outside weld joint between the

first and second convolutions, adjacent to the bottom plate. The slit extended almost 180 deg around the bellows, as shown by the arrows in Figure 33.

When examined with a stereomicroscope (50X), the fracture surfaces appeared to be rough and irregular. However, no subcracking or corrosion was evident. The outside welds of the sector of the primary bellows above the rupture were subjected to Penestrip dye check to locate incipient failure. The result was negative.

Based on the stereomicroscopic examination, seven metallographic samples were selected to determine the modes of the bellows failure. The results are:

1) The bellows fracture occurred and propagated along the weld heat-affected zone.

2) While variations in bead size and geometry were observed, the welds appeared to be consistent with the specified requirements. No cracks or significant porosity was found in the welds examined.

3) No sign of corrosion or reaction was found around the fracture area.

4) At one location (Figure 34), the crack propagation appears to have resumed in the heat-affected zone after it was arrested by the weld metal.

5) Microhardness tests performed at the fractured area indicate that the hardness of the material in the heat-affected zone was slightly softer than both the weld and base metals. The measurements in KHN are:

<u>Weld Metal</u>	<u>Heat Affected Zone</u>	<u>Base Metal</u>
470	423	438
409	409	454
454	423	470
470	409	454
438	409	
<u>423</u>	<u>409</u>	
444	421	454 Average

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Figure 33. Bellows Assembly of ECU SN-028,
Arrow Shows Ends of Fracture

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Figure 34. Fracture Section. Arrow Shows
Crack Arrested at Weld Interface

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During the acceptance tests, the compensator was subjected to pressure, thermal cycles, and vibration loads at 750°F. Stress analysis data show that at full deflection, a total stress of 85,000 psi would be developed at the bellows weld joint. Any deviation in local joint geometry or metal thickness could increase the stress significantly. Based on the results of the examination, it appears likely that incipient cracking may have developed in the heat-affected zone of the compensator bellows during the acceptance tests. The crack or cracks propagated during the FS-3 system test and resulted in the failure. The acceptance tests involved cyclic loading. The fracture characteristics of the bellows resemble fatigue cracking, as described in (4) of the preceding paragraph. Consequently, the cause of compensator bellows failure may be attributed to fatigue.

H. OUTLET PIPING

The NaK outlet piping was visually examined and evidenced no defects. Samples for metallographic examination were cut from fill-tube pinch-off areas and at the junction of instrumentation wells with the main line to show weld condition and to look for evidence of flow impingement. Typical microstructures are presented in Figure 35. All closure welds in the pinch-off areas of the NaK fill-tubes were sound, as shown in Figure 35a. All welds surrounding the junction of instrumentation wells with the main NaK line were sound, although all exhibited some degree of the porosity shown in Figure 35b. No evidence of NaK attack on the outlet piping was observed, as shown in Figure 35c.

I. THERMOELECTRIC MODULES

Included as a supplement to the FS-3 test were two electrically heated, specially instrumented thermoelectric modules. These modules were identified by the numbers X1459-1 and X1505-3; they were mounted within the vacuum chamber and were exposed to the same operating environment as the test system. Figure 5 shows the mounting of these modules, X1459-1 being the one on top of the mounting, nearest the reactor. The fast flux dose to these modules ranged from 1.14×10^{18} nvt to 6.2×10^{17} nvt for the X1459-1 and from 2.4×10^{17} nvt to 1.37×10^{17} nvt for the X1505-3. These modules operated with the center stacks at a hot-shoe temperature of $950^\circ \pm 20^\circ\text{F}$ with ΔT 's across the elements from 310° to 350°F . Elements near the ends of the modules operated about 20°F cooler.

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a. Closure Weld on NaK Fill-Tube Pinch-Off Area

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b. Weld at Junction of an Instrumentation Well with the Main NaK Line

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88-3-2

c. Inside Surface of Main NaK Line

Figure 35. Postoperation Microstructure of Type 316 SS Samples from the FS-3 Handlebars (NaK Outlet Piping)

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After system disassembly it was decided that these two modules would be examined in detail, since their operating history was better defined than that of individual modules incorporated into the system. Since a large background of information was present on modules operated in similar environments, with the exception of radiation exposure, the evaluation was directed at comparing these irradiated modules with unirradiated modules of similar operating history. This approach would define damage incurred as a result of radiation.

1. Resistance Changes

Shoe-to-shoe element resistances were measured before and after the system test to determine the effect of radiation on the resistance properties of SiGe elements. Measurement of the resistance of P-type elements in the irradiated modules showed that there was no resistance change within the $\pm 5\%$ accuracy obtainable. This compares with unirradiated P-type element measurements which showed a 3% increase in resistance under testing for 20,000 hr at 1030°F maximum. The favorable comparison indicates that no significant resistance changes were induced by radiation.

The average increase in resistance for N-type elements for X1459-1 was 67% and for X1505-3 was 55%. This resistance change is to be compared with an average resistance change of 66% for unirradiated elements which operated for approximately 10,000 hr at a hot-shoe temperature of 970°F. These deviations are of the same order as the deviations of individual unirradiated elements, indicating that no systematic resistance change in N-type elements can be attributed to radiation effects.

2. Tensile Test

Tensile testing of the thermoelectric stacks was accomplished by applying an axial strain to the stack. Failure of the stacks occurred in only three areas: the molybdenum alumina bond, the N-type SiGe, and the P-type SiGe. These were the frequent sources of failure in testing performed on unirradiated specimens that had been exposed to similar time and temperature conditions (5,000 to 11,000 hr at 970°F average hot strap temperature). A comparison of tensile test data for the irradiated and unirradiated modules is presented in Figure 36. In general, the specimens from the FS-3 test compare well with those that had not received a radiation dose. The unirradiated specimens should have a higher

Figure 36. Comparison of Tensile Failures

average value and upper limit of load since many of the specimens were operated for significantly less time. Since all of the data from the FS-3 elements fall within the limits of the data from unirradiated elements, it may be concluded that the radiation involved had a negligible effect on the mechanical properties of the elements.

3. Metallography

The metallographic examination of unirradiated modules is presented in Reference 4. The analysis of these modules showed that the areas of bonding which were most susceptible to failure due to time at temperature were the radiator bonds, the shoe bonds, the shoe-to-gold ring bond, and the molybdenum-alumina bond.

Failure at the radiator bond occurs when the copper plate has been consumed (diffusion into the aluminum radiator) and the aluminum, copper, and nickel (plating between the copper and the tungsten shoe or upper compensator) form a brittle compound which causes failure under a shear stress such as that

applied during a thermal cycle. Figure 37 is a typical view of the radiator-shoe bond area of the FS-3 specimens examined. The copper consumption is less than observed on unirradiated specimens, which is to be expected due to the lower temperature of operation of the FS-3 modules. In some of the specimens, sufficient aluminum had penetrated the copper to begin the formation of the brittle phase in the nickel plate. The formation of this phase has not been a primary problem in the past, however, until the copper has been consumed. At this stage there is about 1.5 mils of the original 2.5 mils remaining. This is somewhat more than expected. From Reference 4 one might expect 1.3 to 1.4 mils of copper to remain.

Figure 38 is typical of the shoe bands of the FS-3 modules. No void formation above normal is noted. The microscope did not have a light source sufficient to take polarized light pictures which would show the structure of the WSi_2 .

Figure 39 shows the hot shoe-gold ring band. This region is typical and shows that the band has seemingly suffered no deterioration due to irradiation.

Figure 40 shows the alumina-molybdenum band which is typical. It also is typical of SNAP 10A specimens.

The neutron irradiation did not affect the tensile or resistance properties of N- or P-type stacks within the accuracy of the measurements. No differences were noted in the examination of pre- and postirradiation metallography. Thus, irradiation to 10^{18} n/cm^2 had no measurable effect on the performance of SiGe thermoelectric stacks.

J. REFLECTOR ASSEMBLY

The external reflector assemblies were delivered to the AIHL for post-test evaluation. Both assemblies were visually inspected and photographed through a wall periscope. No defects were discovered. The gears appeared to be in excellent condition. Typical views are shown in Figures 41 and 42. One of the reflector halves had scratches on the inside surface, as shown in Figure 43. The scratches suggest tool marks from fabrication, or marks made during the handling of the units during disassembly.

8-2-67 UNC

7696-4707

Figure 37. Radiator-Shoe Bond of Element No. 32 Module X1505-3

8-2-67 UNC

7696-4708

Figure 38. Hot-Shoe Bond of Element No. 32 Module X1505-3

8-10-67 UNC

7696-4709

Figure 39. Gold Ring-Hot Shoe Bond of Element No. 32, Module X1505-3

8-10-67 UNC

7696-4710

Figure 40. Molybdenum-Alumina Bond of Element No. 32, Module X1505-3

8-2-67 UNC

10FS3-219

Figure 41. Gear Sector of Fine Control
Drum Drive

8-2-67 UNC

10FS3-211

Figure 42. Reflector Block
Assembly

8-2-67 UNC

10FS3-218

Figure 43. Reflector Block Inside Surface

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1. Drum Drive Gear Backlash Measurements

Backlash measurements were taken with a dial indicator held in a specially designed fixture which was keyed to the upper drum support bracket. Nine measurements were taken at approximately equal increments of fine control drum travel from full out to full in for each reflector block assembly. The data are listed in Table 7 for each fine control drum. Included at the end of the table are backlash measurements made on these units prior to system operation. It is to be noted that all measurements but one met the acceptance criteria (0.002 to 0.006).

TABLE 7
FINE CONTROL DRUM BACKLASH DATA
(in.)

Drum Position	+Z Drum	-Z Drum
1 (full out)	0.0020	0.0040
2	0.0030	0.0050
3	0.0030	0.0045
4	0.0020	0.0040
5	0.0030	0.0065
6	0.0025	0.0050
7	0.0020	0.0060
8	0.0020	0.0055
9 (full in)	0.0020	0.0045
Full in (preoperation)	0.0020	0.0025

2. Fine Control Drum Rotation Torque

The actuators and position sensors were removed from both of the fine control drum assemblies allowing the drum rotation to be restrained only by the bearing drag. The torque required to overcome this drag was measured at several drum positions with a small torque meter. Care was exercised to assure that the drum shafts were positioned vertically. The test data are presented in Table 8. Presented in parentheses in this table are similar measurements taken prior to system operation. All measurements fall within the acceptance criteria of 40 oz-in. and variations are deemed insignificant.

TABLE 8
FINE CONTROL DRUM ROTATION TORQUE
(oz-in.)

Drum Position	+Z Drum	-Z Drum
Rotating in		
1 (Out)	4.0 (9.0)	7.5 (6.0)
2 (Intermediate)	8.0 (9.0)	12.0 (9.0)
3 (In)	11.0 (10.0)	11.0 (11.0)
Rotating out		
4 (In)	6.0 (10.0)	4.0 (6.0)
5 (Intermediate)	8.0 (10.0)	7.5 (6.0)
6 (Out)	9.5 (10.0)	11.0 (11.0)

3. Bearing, Ball and Socket and Gear Inspection

The drum bearings were disassembled and visually examined. The bearings turned freely during all phases of in-cell handling. Ball and socket surfaces showed evidence of contact and pressure in the form of small apparently burnished areas but there was little indication of wear or other damage to the surfaces. Figures 44 and 45 illustrate typical ball and socket conditions.

The shafts were removed from the balls and examined. Again, although the surfaces showed some evidence of use and were somewhat darkened, there was no indication of significant wear or damage. Figure 46 shows a typical shaft surface compared to an unused shaft. Drum drive gear teeth were examined through the Kollmorgen periscope. No evidence of damage or severe wear could be detected. Tooth faces on the drum sector gear were slightly burnished but only to a barely detectable extent.

4. Actuator Winding Resistance

Fine drum actuator winding resistances were measured by connecting a cable of known resistance to each actuator, measuring cable plus winding resistance out-of-cell, and then subtracting the cable resistance from the gross value. The cell temperature was 102°F. At the time the resistances were measured the actuators had been in this cell environment for several days with no

Figure 44. Ball – Fine Control Drum
Bearing

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10FS3-235

8-2-67 UNC

10FS3-252A

8-2-67 UNC

10FS3-253

Figure 45. Sectioned Ball –
Fine Control Drum
Bearing

Figure 46. Comparison of Used and
Unused Shafts, the Latter is
at the Bottom

electrical power applied to the windings. It was, therefore, assumed that the winding temperature was also 102°F when the measurements were made. The resistance measurements are presented in Table 9. Presented in the same table are resistance measurements taken prior to system operation. These values have been corrected for the temperature difference which existed between sets of measurements. The differences shown are judged to be insignificant.

5. Actuator Torques

Stalling torques for the fine control drum actuators were measured with currently calibrated torque meters which covered the range from 4 to 108 oz-in. torque. Actuator brakes energized (released) for each stalling torque determination. Test data are presented in Table 10. In parentheses are measurements taken prior to system operation.

Torque required to override the deenergized brake for each actuator was as follows:

<u>+Z (oz-in.)</u>	<u>-Z (oz-in.)</u>
36 clockwise (15.0)	35 clockwise (17.1)
37 counterclockwise (15.0)	34 counterclockwise (18.6)

Actuator bearing torques measured with the brakes released and no power to the motors were as follows:

<u>+Z (oz-in.)</u>	<u>-Z (oz-in.)</u>
16 clockwise (6.5)	8 clockwise (11.25)
17 counterclockwise (6.0)	10 counterclockwise (12.7)

6. Fine Drum Position Sensor Torques

Varying mechanical torques in the 4 to 7 oz-in. range were measured for both sensors with no electrical power applied. With a 26-volt, 400-cps power supply, the torque increased slightly, but results were not conclusive. Apparently the varying mechanical drag of the sensors masked the small electrically induced torque to the extent that reliable measurements could not be made. It

should be noted that removal of the sensors from the reflector assemblies was accomplished with difficulty. It is possible that mechanical damage such as some distortion to the sensor shafts may have occurred, which precluded the accurate measurement of electrically induced torques. Prior to system operation, torques measured on the sensors were less than 4 oz-in.

TABLE 9
ACTUATOR MOTOR WINDING RESISTANCES
(ohm)

Winding	+Z Actuator Postoperation	-Z Actuator Postoperation	+Z Actuator Preoperation	-Z Actuator Preoperation
A to E	8.304	8.102	8.109	7.967
B to E	8.219	7.234	8.210	8.703
C to E	8.007	7.607	7.897	7.992
D to E	8.627	7.602	8.077	8.045

TABLE 10
ACTUATOR STALLING TORQUES
(oz-in.)

Voltage	Stalling Torque			
	+Z		-Z	
1.5		6.8		(1.5)
3.0	8	(17.8)	5	(12.2)
4.5	10	(27.0)	9.5	(22.3)
6.0	33	(37.0)	22	(33.1)
8.0	42	(44.0)	35	(41.8)
12.0	79	(62.8)	49	(63.5)
16.0	104	(76.5)	61	(80.7)
23.0	105	(80.0)	87	(86.0)
25.0	108		93	
26.0	108+		-	
27.0			100	
30.0		(92.5)	107	(98.2)
33.0		(102.0)	108	(98.2)

REFERENCES

1. S. Miner et al. , "Preliminary Test Results — SNAP 10A FS-3," NAA-SR-1106, September 1965
2. T. A. Golding, "Postoperation Evaluation of Fuel Elements from the SNAP 10 Flight System 3 Reactor," NAA-SR-12031 (To be published)
3. F. H. Welch, "Lithium Hydride Technology: Properties of Lithium Hydride for SNAP Shielding Applications," NAA-SR-9400, Vol III, May 1967
4. R. M. Willard, "SNAP 10A Thermoelectric Module Post-Mortem Examination and Material Bond Characteristics," NAA-SR-11205, August 1965