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CRBRP DESIGN AND TEST RESULTS FOR FUEL HANDLING SYSTEMS, PLUGS, AND SEALS*

by

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Abstract

The fuel handling system and reactor rotating plugs for the Clinch River Breeder Reactor Plant (CRBRP) are based primarily on existing technology and, in many respects, follow the concept developed for the Fast Flux Test Facility (FFTF). However, several requirements are considerably more stringent and demand more advanced or larger equipment. This paper first briefly describes the equipment, and then concentrates on the development programs initiated to verify its performance. Test results obtained from the development program, and the extent to which these results verified original design selections, or suggested potential improvements, are discussed.

Among the designs (and tests) included are the In-Vessel Transfer Machine (component and full-size prototype tests), Reactor Rotating Plugs (dynamic inflatable seal, sodium dip seal, and functional test), Ex-Vessel Transfer Machine (heat transfer and component tests), Ancillary Components (core component pot siphon and rotating guide tube tests), and Fuel Handling Cell (component operability and operator training).

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INTRODUCTION

The reactor refueling concept for the Clinch River Breeder Reactor Plant (CRBRP) incorporates major features and/or experience from the refueling systems of earlier sodium-cooled reactors in the United States: Fast Flux Test Facility (FFTF), Hallam Nuclear Power Facility (HNPF), Sodium Reactor Experiment (SRE), Fermi-1, EBR-II, and Southwest Experimental Fast Oxide Reactor (SEFOR). In addition, refueling system designs and experience from the German, French, and British LMFBR programs were factored into CRBRP. Maximum use of FFTF and U.S.-LMFBR development program technologies was a key factor in selection of the reference design concepts, and minimizes the required development for CRBRP.

The refueling system provides for replacement of the reactor core assemblies, including fuel, blanket, control, and radial shield assemblies. The system consists of the facilities and equipment necessary to accomplish the normal scheduled refueling operations and all other functions incident to handling of core assemblies. These functions include: (1) receiving and unloading new core assemblies; (2) inspection; (3) preheating and storage; (4) transfer between storage facilities and the reactor; (5) transfer of core assemblies within the reactor; (6) removal and examination of spent core assemblies; and (7) preparation and loading spent fuel for shipment.

Onsite fuel handling is accomplished in two buildings. The majority of the fuel handling equipment and facilities are located in a Reactor Service Building (RSB) which is adjacent to the Reactor Containment Building (RCB). A perspective of the fuel handling equipment and facilities as arranged in these two buildings is shown in Figure 1. Important fuel handling facilities and equipment located in the service building include a sodium-filled Ex-Vessel Storage Tank (EVST) for interim decay of spent fuel, a Fuel Handling Cell (FHC) for examination and loading of spent fuel into the shipping cask, and facilities for receiving new fuel. Equipment and facilities located in the Reactor Containment Building are limited to those involved with fuel handling in the reactor. The important items include the In-Vessel Transfer Machine (IVTM) supported by triple rotating plugs for moving fuel within the reactor vessel; and the Auxiliary Handling Machine (AHM) for installing and removing the IVTM. Fuel movement between buildings is accomplished during reactor shutdown using an ex-vessel transfer machine (EVTM) mounted on a gantry trolley. The EVTM and gantry travel between the reactor and the EVST through a large diameter equipment hatch which is removed when the reactor is shut down for reactor refueling. The EVTM and gantry remain in the RSB during reactor operation and are used for receiving and shipping of fuel during this time.

A detailed description of the refueling system and associated components has been presented in a previous paper.* The objective of this paper is to briefly describe the design of equipment on which development tests are underway, the tests themselves, and their results. These development test programs center around the IVTM, the reactor rotating plugs, the EVTM, the FHC, and several small ancillary pieces of equipment. The Advanced Reactor Division (ARD) of Westinghouse Electric has been assigned design and procurement responsibilities for the reactor rotating plugs. Atomics International (AI), a division of Rockwell International, is responsible for the design and procurement of the remainder of the equipment. AI is also responsible for the majority of development testing.

IVTM DEVELOPMENT PROGRAM

The IVTM is a simple, lightweight machine that is readily installed in the reactor after shutdown for refueling operations and removed prior to reactor startup. The machine, as shown in Figure 2, is mounted on a tripple rotating plug, and consists of an in-vessel section, which operates in reactor sodium at 400^{o}F, and a drive section which is open to the building environment. The IVTM is a vertical, straight pull, rising stem concept which grapples a core assembly and exerts a push-pull force on the assembly to insert it into or raise it out of a core position. Because it is a first-of-a-kind concept, although many of the components and materials use FFTF technology, a development program has been planned for the grapple stem dynamic seals, for the grapple and for some drive components. In addition, a full-scale prototype IVTM will be fabricated and tested in sodium.

One of the key features of this machine is the dynamic elastomer seals which are located between the grapple rising stem and the seal housing and prevent reactor cover gas from escaping to the containment building. The seal concept utilizes three ethylene propylene T-ring seals with nylon backup rings. The space between each seal is buffered with argon gas and is continuously monitored for leakage. Wiper seals clean the stem before entering the seal housing, and nylon bushings provide guidance. Photographs of the test apparatus, seals, and housing are shown in Figure 3.

The 3.25 in. diameter grapple stem travels at a maximum speed of 15 ft/min and varies in temperature between 70^{o}F and 300^{o}F. Although the stem that passes through the seal is never wetted by sodium, the reactor pool sodium does produce vapor that condenses on parts of the

*Foster, K. W., "Clinch River Breeder Reactor Project Fuel Handling System," Proceedings of 23rd Conference on Remote Systems Technology 1975, pp. 415-423.

stem that reach the seal. Because of this severe environment, prototypical seals were tested under simulated service conditions of temperature, sodium environment, and dynamic operation. Two different vendor seals were qualified for a 5-year service life. Leakage did not exceed the permeation rate of argon and xenon through the seal material, and seal wear was negligible after 200,000 ft of stem travel. No deleterious effects were noticed from sodium-sodium frost deposits on the stem and housing.

A prototype IVTM grapple and a core assembly handling socket have been tested in room temperature air under misalignment conditions. This test was conducted to verify that the three-finger grapple design would function under push-pull loads up to 5,000 lb and under misaligned conditions. Side loads that would result from centerline misalignment of 1.75 in. between the grapple and handling socket were imposed. After 10,000 engagement cycles, which are equivalent to 20 average refuelings, only slight rubbing marks were observed on the grapple nose; no wear patterns were seen at the grapple fingers or cam surfaces. The full-scale IVTM prototype test will provide confirmation that sodium and temperature will not affect the grapple service life.

The assembly of IVTM components used for the grapple stem vertical and rotational drive systems, including such components as motors, controllers, gear reducers, brakes, chains, and sprockets have been tested under the full range of speed and loading conditions. Both drive systems operate at slower speeds than normal for these types of components. In this low speed range, efficiencies are lower than normal and difficult to predict because of the rapid change in friction with respect to low velocities of the dynamic components. Tests were, therefore, necessary to verify that the drives operated smoothly and efficiently. Performance characteristics, including gear train efficiency, brake stopping power, dynamic response and actuator drive power, were determined and components met all specified operational requirements.

Fabrication of the IVTM prototype machine has begun. Testing will begin in mid-1979 and will be completed early in 1980. This test will be a complete checkout of the integrated IVTM system, including mechanical, electrical, instrumentation, and computer control (including software) components. Testing will be done in both air and sodium at 400°F, under the full range of operating conditions including horizontal misalignment of up to 1.75 inches between the IVTM grapple and the core assembly handling socket. The objective of the test will be to demonstrate satisfactory performance for a minimum of five refuelings (equivalent to a 5-year life) and, by inspection, to infer that all components (with the exception of electrical components and elastomer seals) could operate without failure for 10 years.

REACTOR ROTATING PLUG DEVELOPMENT PROGRAM

The reactor rotating plugs are part of the closure head assembly which forms the upper closure of the reactor vessel, supports associated reactor equipment, and contains various penetrations for control rods, in-vessel instrumentation, upper internals, fuel handling equipment, and dip seal maintenance and filling equipment. (See Figures 4 and 5.) There are three independently rotatable plugs interconnected by plug risers; a bull gear and bearing atop each riser permits plug rotation by a plug drive and control system.

The structural portion of the closure head consists of three rotating plugs that are made from SA-508, Class-2, low-alloy steel forgings, 22-inches thick. Suspended beneath each rotating plug are three radiological shield plates, twenty thermally insulating reflector plates, and a gas entrainment suppressor plate assembly which is immersed in several feet of sodium.

The three closure head rotating plugs are supported by the plug risers, which are arranged in pairs and bolted or welded to the periphery of each plug and the reactor vessel flange as shown in Figure 5. The risers provide the structural support for the plugs, contain the bearings for rotation, and also contain the reactor head seals. Attached to the tops of the risers are the bull gears which engage the plug drive system. The riser flanges are forged from the same material as the closure head plugs, but the webs are Inconel-600 which provides a high thermal impedance. This keeps the elastomer inflatable seals and the plug drives at the top of the risers below 125^{o}F, compared with the head temperature of 400^{o}F. The annulus between each rotating plug and between the closure head and reactor vessel is isolated from reactor cover gas during reactor operation and refueling by a sodium dip seal located in each plug. This reduces gaseous radioactivity and sodium frost deposition in the annulus and permits free rotation of the plugs. Metallic o-rings are required to seal the bolted face of the riser base flanges to the head because they can withstand the 400^{o}F head temperature for the required 30-year life of the plant. The larger outer riser is provided with a welded hermetic "C" seal. The elastomer inflatable seals, made from Buna-N, permit relative rotation of the plugs and must be replaced at planned service internals. The elastomer o-ring seals are made of EPR compound (ethylene-propylene rubber) and have a life cycle of about 10 years. The buffer spaces between double seals are pressurized and purged by a buffer gas service system. (All refueling system seals are double). Dynamic seals are provided with a pressurized buffer gas and continuous leak monitoring, while static seals are provided with a capability for periodic leak monitoring.)

In order to maintain gaseous radioactivity releases as low as reasonably achievable (ALARA), elastomeric seals must be essentially leak-tight. In practical terms this means mechanical leakage around or through the seal must be eliminated, and elastomeric seal materials must be selected for their resistance to permeation of radioactive krypton and xenon. An LMFBR cover gas seal development program has been underway for many years to select materials with the best combination of low permeability and long life at operating conditions. As a result of this program, a Buna-N polymer has been selected for use in all CRBRP refueling applications where ambient temperatures are below $\sim 140^{o}F$ (based on a five-year life requirement), because of its combination of lower permeability and good mechanical properties. For higher temperature applications (up to $\sim 200^{o}F$), an ethylene propylene polymer has been selected.

The rotating plug dynamic inflatable seal's design and material were also selected as a result of this program. The inflatable seals are the same cross-section design and material as those used on the FFTF in-vessel handling machine (IVHM), but the maximum CRBRP diameter, misalignment offsets (total indicated), plug rotation speed, and amount of usage are all higher than for FFTF (23 vs 6 ft, 60 vs 30 mils, 15 vs 2 fpm, and 210,000 vs 75,000 ft, respectively). Additional tests were performed to demonstrate adequate seal life and suitable operating parameters under the CRBRP conditions. The effects of misalignment, seal inflation and buffer space pressures, temperature (70 to $150^{o}F$), plug rotation speed and direction, and total travel on parameters such as starting and running friction, static and dynamic leakage, and krypton and xenon permeation were determined. Over 500,000 ft of travel were accumulated at prototypic conditions; seal friction and leakage were acceptable, and wear was negligible. The lubricant and adhesive and installation and removal procedures to be used for CRBRP were also selected by these tests.

Qualification tests using prototypic CRBRP dynamic inflatable seals from two vendors are in progress. These tests will be used to select the seal vendor for the initial plant items and to qualify two vendors for future purchases. Additional development tests are also underway to improve seal misalignment capability and to reduce starting torque for advanced applications.

The sodium dip seal concept is also a product of many years of LMFBR development testing which demonstrated there efficacy and practicality. The specific geometry of the CRBRP dip seal was developed in water tests. These were followed by sodium tests of a dip seal with prototypic cross-section and reduced diameter (~ 3 ft). The effects of pressure differential across the seal, plug rotational speed, substantial oxide and hydride contamination, and operating time on parameters such as radioactivity gas attenuation, startup and running torque, and sodium frost deposition characteristics were determined. The tests demonstrated satisfactory torques, frost formation, and resistance to pressure fluctuations and corrosion at the gas-sodium interface in the seal. However, cover gas transmission was ~ 100 times the claculated values. Investigations showed that this resulted from gas bypassing along the seal blade which had not wetted at the $400^{o}F$ seal operating temperature. Further tests showed that both sacraficial coatings (for example, gold or tin) on the SA-508 steel blade and exposure of the blade in sodium to ultrasonic vibrations induced wetting. A combination of both techniques was incorporated in the CRBRP design, and in subsequent sodium testing the revised dip seal design met the leakage requirements. A final series of tests are now in progress to develop effective maintenance techniques for removing accumulated sodium/oxide/hydride deposits from the dip seal trough and plug annulus.

Prior to delivery to the plant site, a functional test of the plug drive and control system and the assembled closure head will be conducted to verify proper functioning, including the positioning accuracy of the rotating plugs at refueling temperature (400°F). Fabrication of the head plugs is currently underway and testing is scheduled to begin late in 1979.

EVTM DEVELOPMENT PROGRAM

The EVTM development program has been quite extensive in two areas: (1) verification of the decay heat removal system and (2) demonstration of the grapple-chain system to perform with the chain and sprockets wetted by sodium. Ancillary articles of equipment associated with the performance of this machine are also undergoing testing, namely, the core component pot sodium siphon and the rotating guide tube.

The original design of the EVTM was based on the FFTF-CLEM (Closed Loop Ex-Vessel Machine). However, the CLEM concept needed improvement in several areas in order to satify CRBRP requirements. First, the decay heat removal system had to be upgraded to remove 20 kW of heat while assuring that peak fuel cladding temperatures do not exceed 1,250°F. Second, the machine overall size had to be considerably reduced in order to pass through a reasonable size access hatch between the containment and service buildings.

The EVTM is a heavily shielded cask body transported on a gantry-trolley system. The machine has a central 8-in. diameter cavity which contains an inert atmosphere (argon) for carrying a core assembly in a sodium-filled core component pot (CCP). The sodium, because of its good heat transfer properties, provides the medium for transfer of heat from the fuel pins to the CCP wall, which in turn radiates to the machine "cold wall." Heat is dissipated from the machine by forced-air cooling on the outside of the cold wall.

The two principal uncertainties in the heat transfer mechanism were the mode of transfer in the sodium and the effect of sodium-sodium oxide deposits on the CCP and cold wall emissivity properties. A one-third scale test was conducted, employing an electrically heated fuel assembly in a sodium-filled CCP which was assembled in a cold wall environment. The objective of the test was to characterize the heat removal parameters, particularly the sodium convection currents, with an analytical model. Parameters that were varied included heater power, annulus width between the simualted fuel assembly duct and the CCP wall, and sodium column height above and below the heater section. Test results showed that a weak convection loop existed through the fuel assembly, but the predominant mode of heat removal was through local convection loops in the fuel duct and between the duct and CCP wall.

By means of the small scale parametric heat transfer test results, an analytical model was calibrated and used to design the EVTM heat removal system. In order to assure that scaling laws were appropriate and to evaluate steady state and transient operating conditions, a full-scale heat transfer test was conducted. A 217 pin, electrically heated fuel assembly was fabricated, and much of the actual hardware fabricated for the CLEM was used, such as the cold wall, blowers, and control system. Figure 6 shows a test schematic and a photograph of the simulated fuel assembly being lowered into the test apparatus.

Tests were conducted for steady state vs power, transient conditions, loss of blowers, low sodium level, heatup rate and with various atmospheres, including vacuum, argon and helium. Test results provided the following information:

(1) The analytical model was confirmed after some adjustment to several parameters;

(2) Steady-state operation at 20 kW resulted in peak cladding temperatures below 1250°F;

(3) Cladding temperatures remained below 1500°F with the blowers shut off and cooled only by natural convection of air over the cold wall (20 kW decay heat);

(4) Radial power distribution is unimportant to peak cladding temperature because of sodium convection;

(5) Substitution of helium for argon significantly improved the heat transfer; and

(6) Below power levels of 4 kW, sodium in the bottom of the CCP froze.

The remaining question of sodium/sodium oxide effects on emissivity was resolved by a one-third scale test of an electrically heated CCP in a cold-wall environment. The CCP was repeatedly immersed in a pool of sodium and withdrawn into the argon atmosphere which was contaminated with oxygen and water moisture. After the equivalent of one average refueling, no significant change in heat transfer characteristics was observed. A small and gradual decrease in cold wall emissivity was attributed to sodium deposits and was easily corrected by turning on the cold wall heaters.

Based on the test results and utilizing the correlated analytical model, the EVTM heat transfer characteristics were predicted as shown in Figure 7. These results verify the adequacy of the EVTM to maintain peak cladding temperatures below 1250°F for a 20-kW fuel assembly.

The EVTM was originally designed to follow the principle of previous machines of not allowing the grapple suspension members (chain or tape) that pass over the drive sprockets to become wetted by sodium. To prevent sprocket wetting, a long grapple was required, which resulted in a 53 ft high cask body to house the grapple and CCP. In order to shorten the EVTM body height to only 32 ft, a drive and grapple system was designed which immersed the suspension members into sodium and permitted the wetted portion to enter the drive sprocket system. The reference drive concept utilizes a 1.036 in. pitch, 3/8 in. diameter carbon steel crane chain. In the original design, the chain passed over 3-8 pocket idler sprockets and then through an 8-pocket drive sprocket. The slack chain is collected in a chain fall tube. An alternate tape drive system, which coils a 0.035 in. thick by 4.5 in. wide stainless steel tape on a 29 in. diameter drum, has been designed. The chain concept is preferred from a design standpoint since the drive system is more compact, and the tape concept is not compatible with the rotating guide tube.

A test was initiated on both the chain and the tap drive systems to determine the amount of sodium carryover and any tendency for binding. The test setup is shown in Figure 8. Testing consisted of repeated lowering and raising of the suspension member, which supports a 1,200 lb weight, into a pool of 400°F sodium. A prototypical length of wetted and nonwetted member passed through the drive system.

The first series of tests accumulated five refueling cycles, defined as transfer of 150 core assemblies, between the reactor and EVST.

Test results indicated both concepts were feasible for the EVTM, but the chain system was retained as the reference design because of reasons discussed before. However, several operational anomalies were observed in these initial tests which would limit the chain drive system life. Hence, testing was continued to further optimize the system. Sodium build-up between the chain sprocket and guide cap caused a frictional increase which would require cleaning on an annual basis. The guide cap and other components were redesigned so that the frozen sodium will be free to extrude from the sprocket/guide cap assembly, thereby reducing frictional buildup.

Chain wear between links was observed to be greater than expected. After two refueling cycles, the average length of five links increased by 0.250 in. due to wear between links. This wear is a factor of two higher than allowable, which would limit the chain life to one year. The chain wear was proportional to the number of sprockets it passed over, and directly proportional to the number of times the chain links moved against each other as they pass around the sprocket wheel. In order to reduce the wear, the drive sprocket was changed from the last to the first position. Hence, the chain will pass over the first sprocket under load and the last 3 idler sprockets under minor take-up tension. This approach was expected to reduce the wear by a factor of 7.

Additional tests were performed and showed the need for additional detail design changes of the pocket wheel and chain guide. The last series of tests showed higher than anticipated wear when operating in argon. However, immersion of the chain in sodium, followed by either continued wet or dry testing, resulted in satisfactory performance for the equivalent of 5 refuelings, with a projected life of 10 reactor operating years. A series of supplemental tests confirmed the apparent sodium lubricity implied by these results. Further tests are now underway to investigate the effect of cover gas and/or sodium oxygen content on sodium carryover. Excessive sodium buildup of sodium requires periodic mechanical or thermal cleaning of the chain to avoid high friction and kinking of the chain. Although mechanical cleaning is practical, these tests are expected to lead to better understanding of the phenomenon and a more desirable solution.

Fabrication and testing of a rotating guide tube are planned in the near future. The rotating guide tube is incorporated at the reactor head port such that when the EVTM couples to the port, the grapple can be directed to one of two transfer positions in the vessel. By means of this guide tube feature, the EVTM can install a new fuel assembly and remove a spent assembly with one coupling operation, which reduces the refueling time significantly. Since the guide tube design involves rotating components, such as seals and bearings, and involves metal-to-metal contact of the grapple against the guide tube, a full-scale prototype assembly will be tested in air. The guide tube is presently in the design phase; prototype testing is expected to be initiated in the spring of 1980. Another test involving the rotating guide tube was dictated by the higher than expected wear observed in the EVTM chain test. A test was performed to determine wear at the point of contact between the rotating guide tube and EVTM chain, EVTM grapple nose, and CCP nose. The test showed insignificant wear and acceptable performance.

Another area of testing associated with the EVTM is the development of a siphon for the CCP to prevent overflow of sodium when in the EVTM. An internal flow passage in the form of an inverted U has been added to the walls of the CCP in the upper 14 in. One end of the passage is open to the internal volume of the CCP and the other end opens to the outside of the pot. As soon as the CCP leaves the reactor or EVST sodium pool, the siphon drains out the upper 8 in. of sodium from the pot. This drain action occurrs before the CCP enters the EVTM cask body, thereby eliminating sodium spillage and drippage on machine internals which has created maintenance problems in previous machines. A water test utilizing a transparent plastic mockup of the CCP handling head and siphons to permit visual observation was performed first. This test provided design information on the proper passage configuration for initiation of the siphoning action and for determining the time required to complete drainage. A prototypic siphon was then fabricated from stainless steel for testing in sodium. The purpose of this test was to demonstrate the effect of design changes made as a result of the water tests and the effects, if any, of substituting sodium for water. 160 cycles of filling and draining were conducted at grapple speeds ranging from 6 to 12 ft/min. In all cases the siphon performed its draining function reliably and completely. Maximum draining time was only 20 seconds compared to the requirement of less than 37.5 seconds.

FUEL HANDLING CELL DEVELOPMENT PROGRAM

The fuel handling cell is a small hot cell with three stations: a maintenance and service station for the EVTM, a fuel examination station for gross dimensional measurements, and a spent-fuel loading station for loading fuel through a port in the cell floor into a spent-fuel shipping cask. A cell mockup has been constructed (Figure 9) for assembly and checkout of cell components prior to shipment to the site. Several mockup components will be built and tested to provide design information before fabrication of plant hardware. These mockups include the in-cell cooling grapple and the shipping cask-to-FHC floor seal assembly. After equipment has been checked out, the mockup will be disassembled and shipped to the site as an operator training facility.

The initial test conducted in the FHC mockup used a physical mockup of the long gas cooling grapple to load a dummy core assembly into a mockup of the spent fuel shipping cask. Successful loading of the cask was demonstrated under conditions of maximum misalignment (1-inch) between the FHC port and the cask. A second test to confirm the capability for remote removal and reinstallation of the shipping cask port plug without lubrication resulted in cocking and binding of the plug. The problem was eliminated by reducing diametral clearance between the pilot diameter on the nose of the plug and the mating diameter of the cask port.

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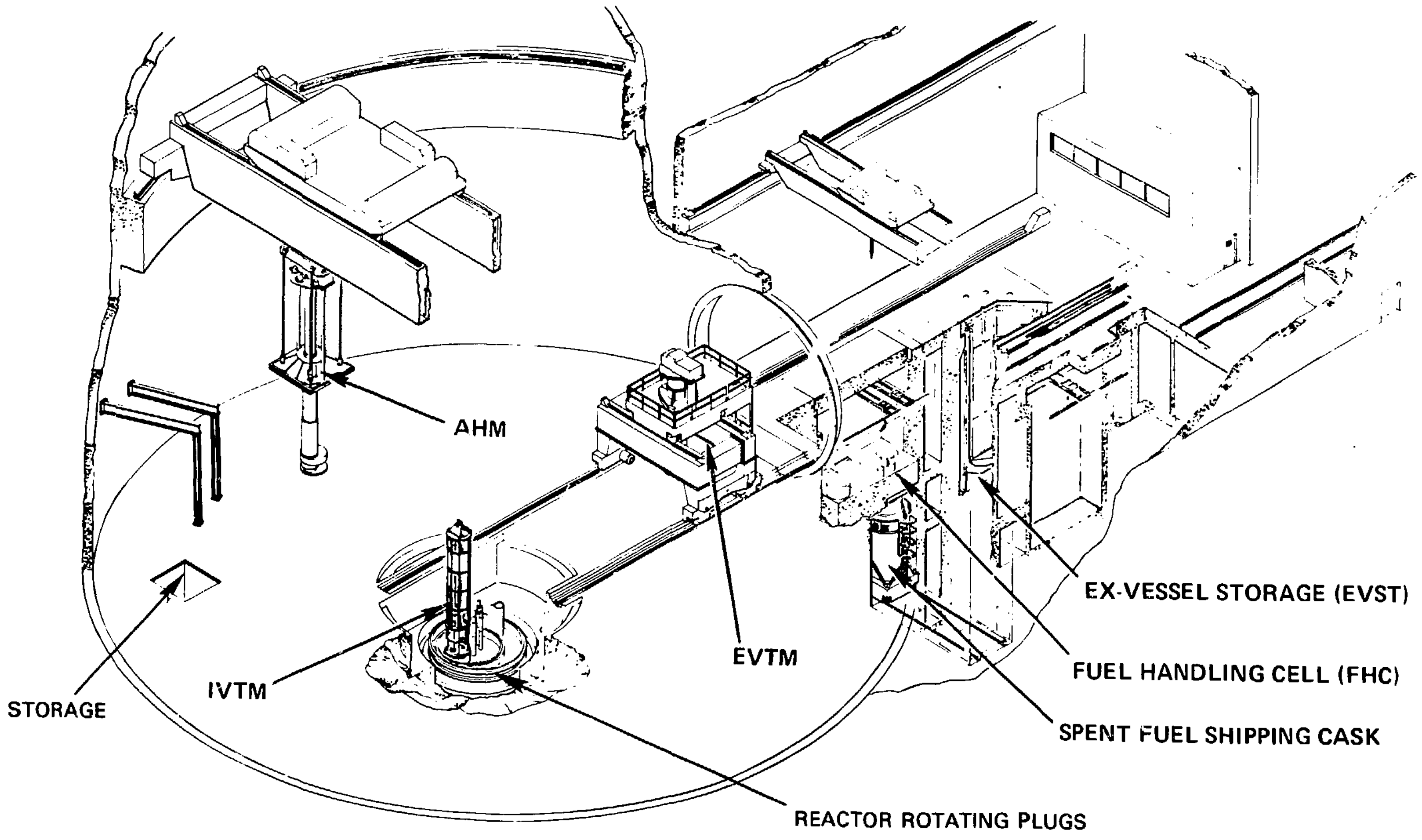
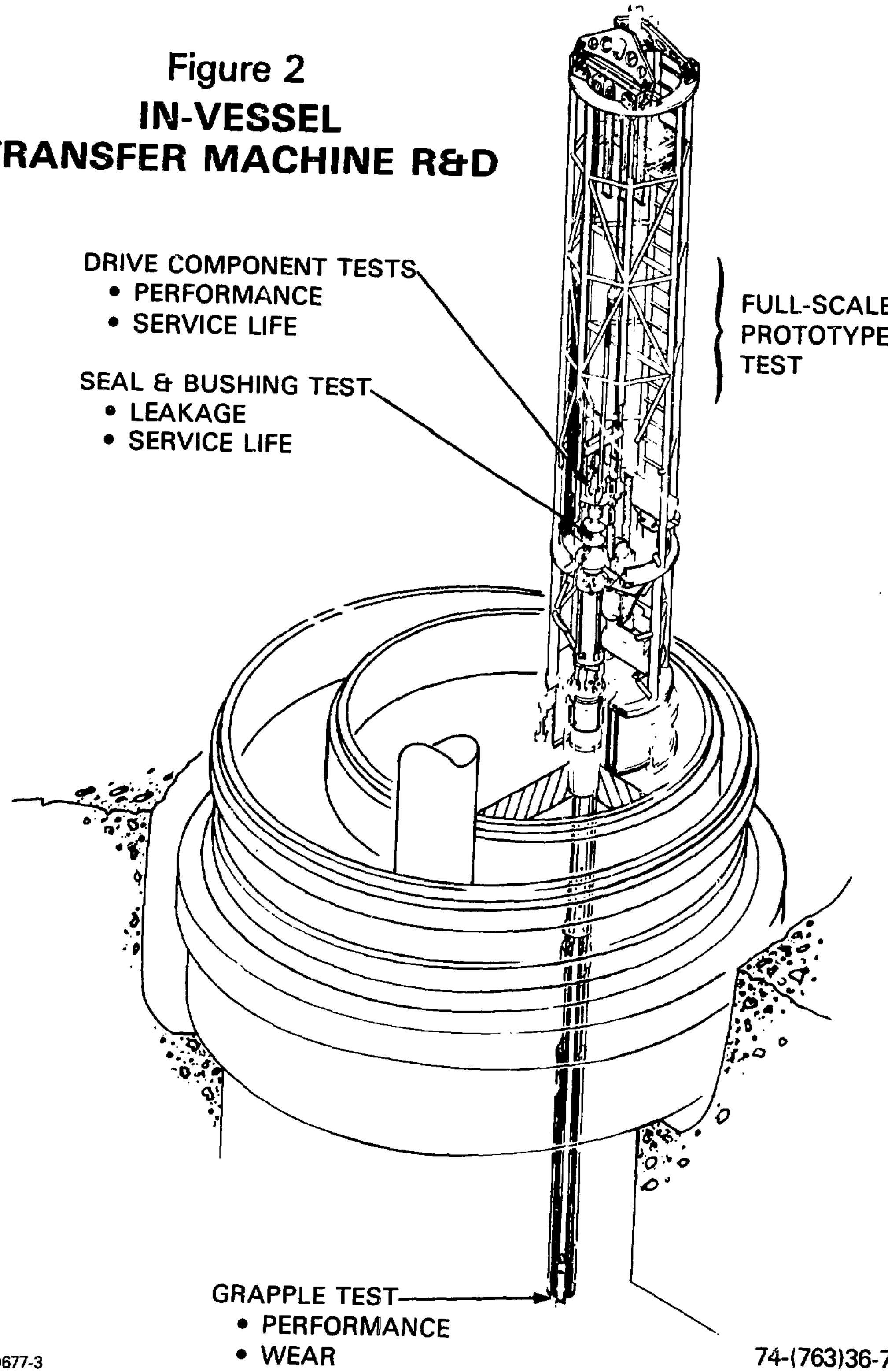
Figure 1
CRBRP REACTOR REFUELING



76-PO421-1

74(763)36-B

Figure 2
IN-VESSEL TRANSFER MACHINE R&D



76-P0677-3

74-(763)36-73A

FIGURE 3

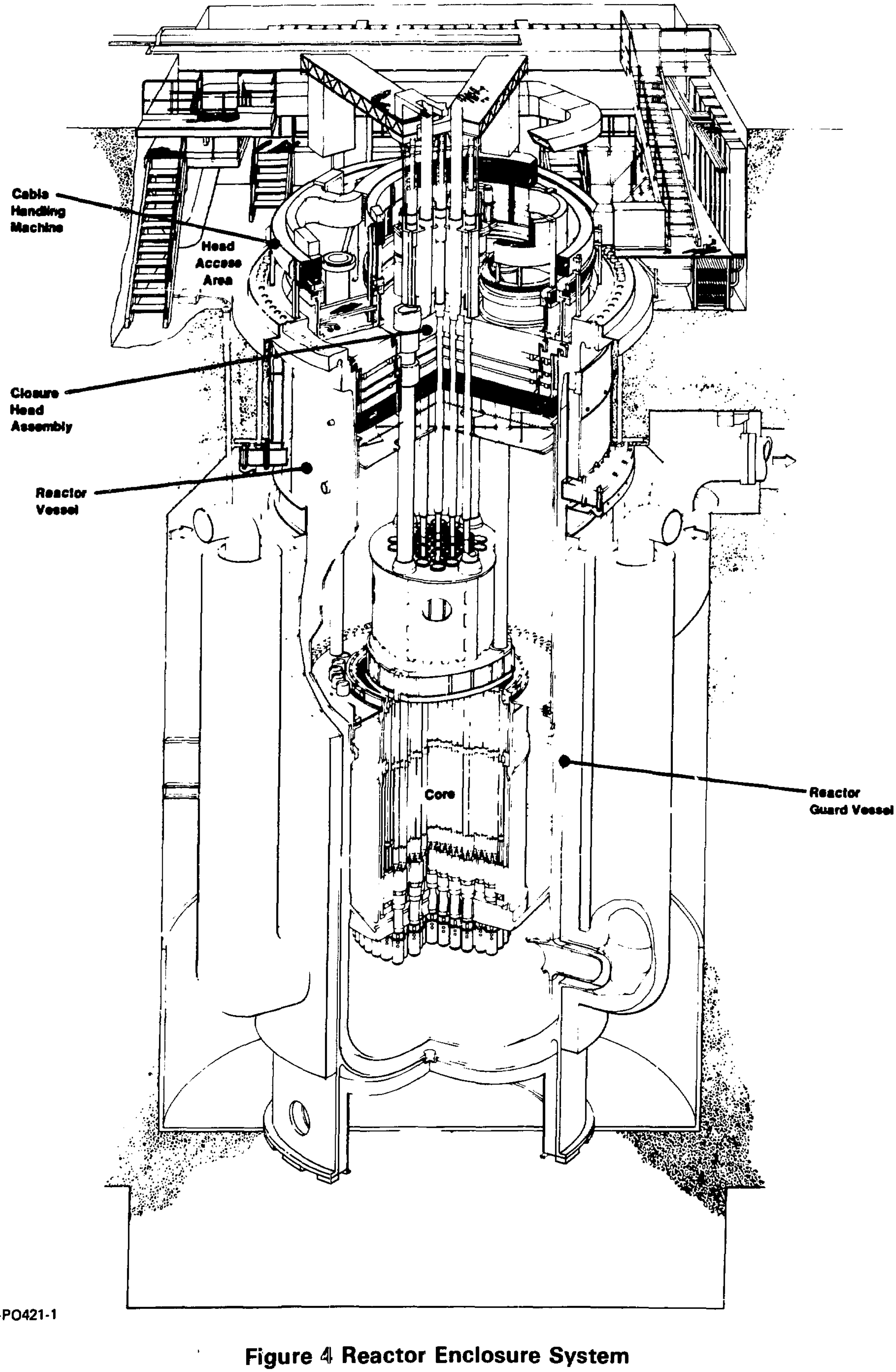
IVTM SEAL TEST

TEST APPARATUS

SEALS & HOUSING

ETHYLENE
PROPYLENE
SEALS

NYLON
BACK UP
RINGS



76-PO421-1

Figure 4 Reactor Enclosure System

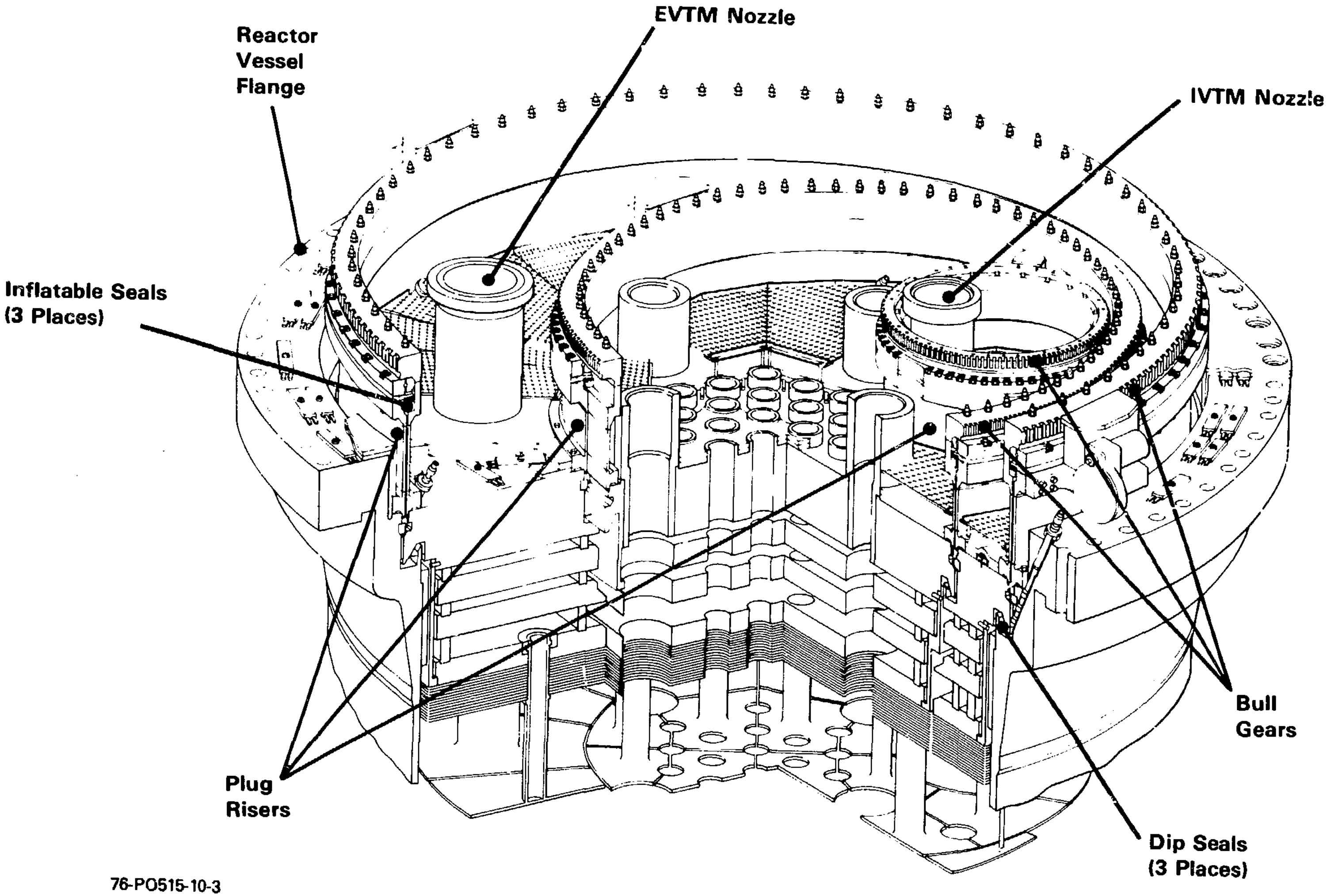
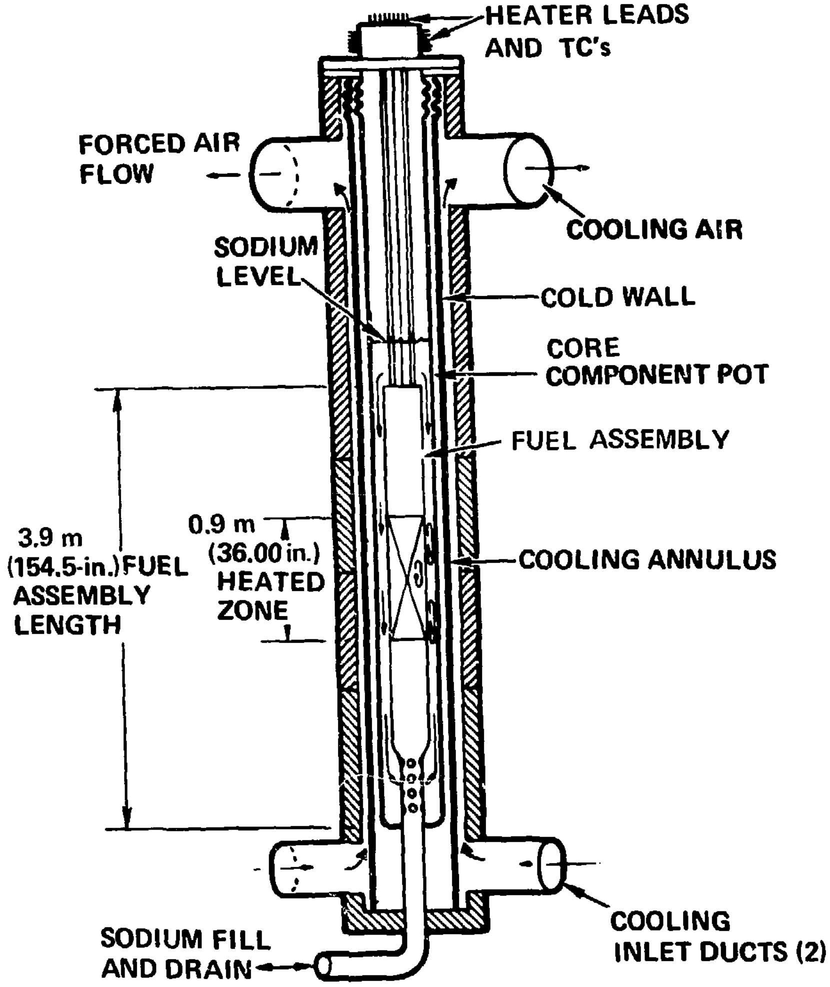
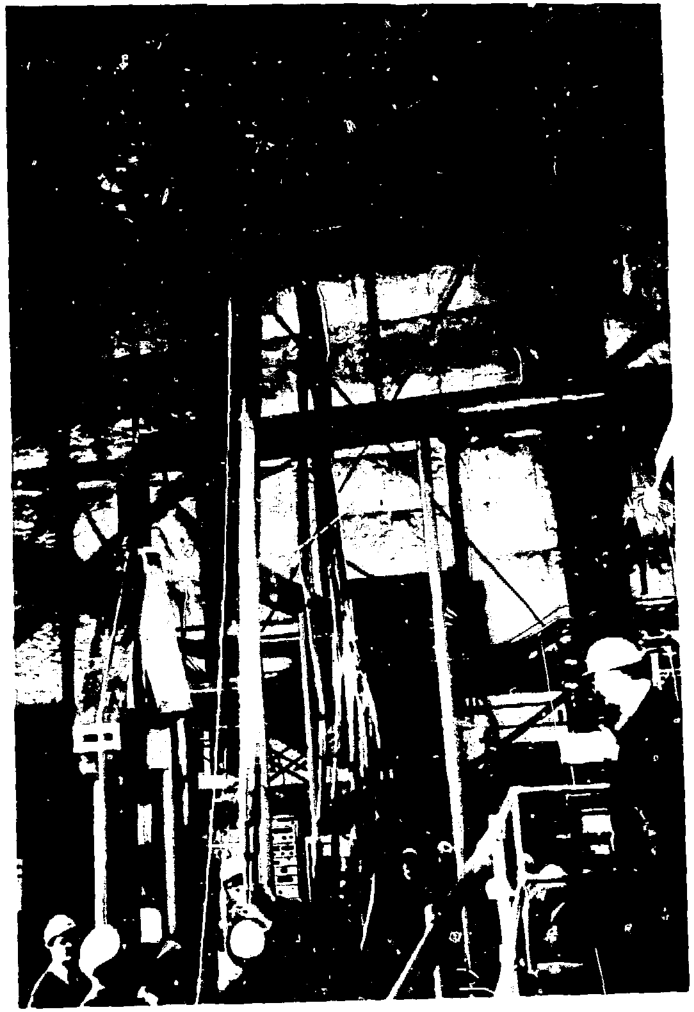


Figure 5 Reactor Closure Head Assembly

Figure 6

EVTM FULL-SCALE HEAT TRANSFER TEST





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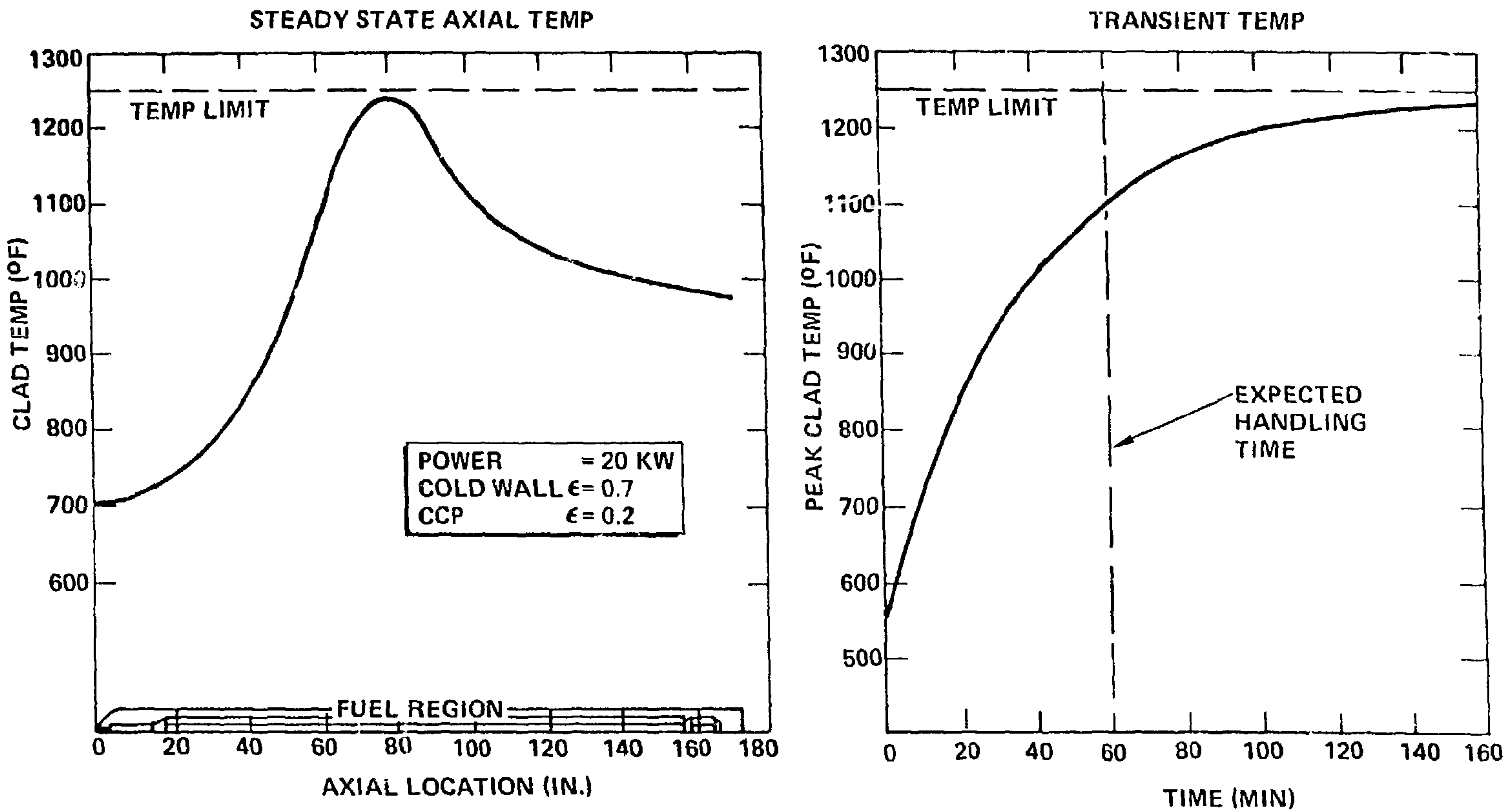
75-A10-30-12B

FIGURE 7

EVTM HEAT TRANSFER PREDICTIONS



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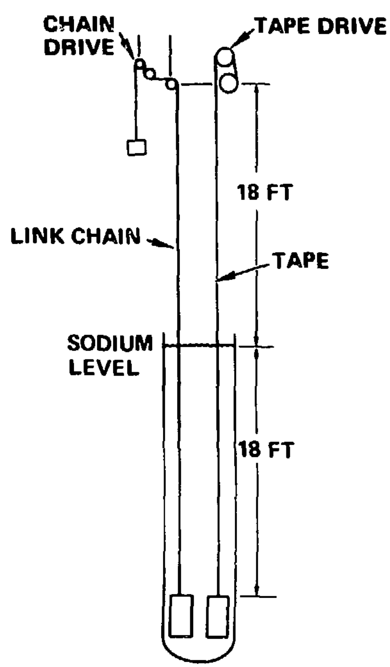
76-M3-23-8

FIGURE 8

GRAPPLE CHAIN Na IMMERSION TEST

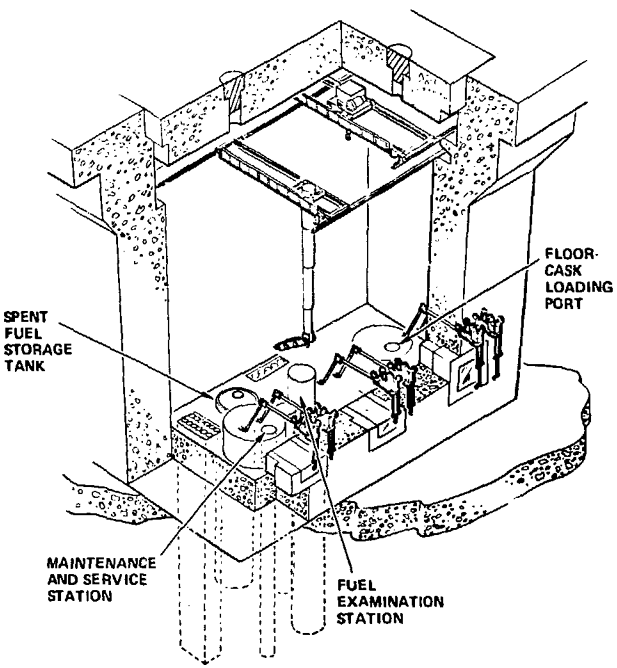


75-A10-30-13A

Figure 9

FUEL HANDLING CELL MOCKUP

FUEL HANDLING CELL



78-M3-23-9