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Initial Operation of the Sodium Graphite Reactor at the Hallam Nuclear Power Facility

MASTER

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I. INTRODUCTION

The Hallam Nuclear Power Facility (HNPF), operated by the Consumers Public Power District of Nebraska (CPPD) for the United States Atomic Energy Commission (AEC), has successfully completed the planned testing program up to full power operation at approximately 250 megawatts thermal (MWT) and a net electric output of about 77 megawatts (MWe). Since November, 1963, the plant has been supplying power in accordance with the power system demands.

CPPD formally accepted responsibility for operation of the HNPF on February 6, 1964, coincident with the issuance of an Operating Authorization to CPPD by the AEC.

Plant operation continues to demonstrate the advantages of a sodium cooled reactor by producing high quality steam at conditions comparable to those in conventional power plants with the safety factors inherent in the sodium-graphite reactor concept.

Reference 1 describes the HNPF, its construction, and the planned testing program. References 2 and 3 contain further plant description and detail of early operating and testing experience.

Table I lists the dates of significant steps in the HNPF program.

TABLE I

SCHEDULE OF MAJOR EVENTS

Contract signed by CPPD and AEC	September	1957
Ground breaking ceremonies	April	1958
Full time construction initiated	March	1959
Electrical generation from conventional plant	April	1961
Commercial operation of conventional plant	July	1961
Completion of Nuclear Plant construction	October	1961
Dry critical (no sodium in the reactor)	January	1962
Wet critical	August	1962
15% full power attained	January	1963
First electrical generation with steam from the Nuclear Plant	May	1963
100% full power attained	July	1963
Transfer of operational responsibility to CPPD	February	1964

II. PLANT DESCRIPTION

CPPD's Sheldon Station includes a conventional coal or gas fired boiler, the HNPF with its three sodium "fired" steam generators, a turbine, administrative and service facilities. The turbine may be supplied from either steam source, or from both in parallel. Figure 1 is a photograph of the station.

The HNPF reactor is a graphite-moderated, uranium fueled, thermal reactor. Present fuel loading is 140 Uranium-Molybdenum elements, and 10 Uranium-Carbide elements.

Heat transfer is by means of three identical sodium circuits (Figure 2).

Table II lists some of the important plant parameters.

The sodium pumps are standard centrifugal pumps rated at 7200 gpm at 160 ft discharge head. Being of free surface construction, with helium in the pump barrel above the sodium, no sodium seals are required. Oil filled gas seals at the floor level prevent loss of the helium cover gas.

The intermediate, sodium-to-sodium, heat exchangers are of conventional tube-and-shell design with a bellows in the shell to accommodate differential thermal expansion.

The steam generators are three-shell units with separate kettle type evaporator, steam separator and superheater. Sodium flows through double-walled bayonet tubes to heat water or steam in the shells of the evaporator and superheater. The narrow annulus between the double-walled tubes is filled with helium. Helium pressure is monitored to provide leak detection.

TABLE II

PLANT PARAMETERS

Licensed Reactor Thermal Power	256 MW
Electrical Output (net)	77 MW
Sodium Temperatures	
Reactor Inlet	610 F
Reactor Outlet	945 F
Steam Conditions	
Superheater Outlet	875 F
Evaporator Pressure	900 psig
Sodium Flow Rate thru Reactor	8.4×10^6 #/hr
Steam Flow	753,000 #/hr

III. PERSONNEL

The plant complement at Sheldon Station now numbers 82. The plant organization chart as depicted in Figure 3 has been developed to utilize the skills and talents of all personnel in both conventional and nuclear portions of the station.

A. STAFF

The nucleus of the plant staff was organized in 1958. Approximately one-half of the plant complement was added in 1960 when the formal reactor plant training began. The Engineering Staff includes an Assistant Plant Superintendent for Nuclear Engineering; a Reactor Engineer; a Chemical Engineer; 2 Performance Engineers; an Office Engineer and 2 technicians. The Health Physics Group is composed of a Health Physicist and 2 technicians. The Clerical Group includes the Plant Accountant; 3 clerical personnel and the 2 tour guides. The Operations Group includes a supervisor; 5 shift supervisors, and 29 operators in the 3 classifications. There are 2 operators of each classification on each shift. Operators are trained to work in either the conventional or nuclear portions of the plant. Every 6 or 7 working days they rotate from conventional to nuclear plant and vice versa in order to keep familiar with current operations.

The 29-man maintenance group is sized for normal plant operation. Outage time is held to a minimum by supplementing the normal maintenance crew with outside help.

B. TRAINING

The 1960 reactor plant formal training program was attended by 29 of Consumers personnel. This was a full time program of six months' duration. Extended training up to one year in duration was provided for individuals on the technical staff in the areas of their responsibilities.

A 300-hour formal classroom and in-plant training program was conducted for several groups during 1961, 1962 and 1963 prior to the AEC operator licensing examinations. A similar 120-hour classroom program is currently in session for AEC Senior Operator license applicants.

Formal classroom training for the conventional plant was also conducted during plant startup. All plant personnel have also profited from on-the-job training during the conventional and nuclear plant startups and testing.

IV. TESTING

The plant testing program was divided for convenience into four major sections:

1) Preoperational, up to and including initial critical loading

2) Post critical, phase I, reactor testing at "zero" power ($<$ 100 Kwt)

3) Post critical, phase II, nuclear plant testing (no electrical generation) up to 15% design power (38 Mwt)

4) Post critical, phase III, power plant testing from 15% to 100% design power.

The preoperational testing proved the adequacy of plant equipment and systems to meet design requirements. Fuel loading to initial wet (reactor-filled with sodium) criticality culminated the preoperational test program.

The phase I post critical tests carried the reactor to design fuel loading, verified calculated values of nuclear parameters and assured nuclear safety of the plant.

Phase II post critical tests demonstrated the operability of the entire plant, up to design temperatures. During phase II testing reactor power was "dumped" to an atmospheric steam dump condenser, which allowed the reactor testing to proceed at any desired level without interfering with station output.

Phase III post critical tests demonstrated the performance of the plant "on line". During this period steam produced by the reactor plant was used to generate electric power for the electrical power grid. The reactor and the conventional boiler were operated alternately or in parallel to best meet the requirements of the testing program while simultaneously fulfilling CPPD electrical generation commitments. A detailed test and evaluation report was prepared on each of the 56 separate tests.

V. OPERATIONAL EXPERIENCE

As discussed later, there have been several nuclear plant problems which resulted in operational delays. Use of the conventional boiler to meet system load demands during these periods has provided the opportunity to make permanent fixes. Operation of the two facilities in parallel has also permitted a flexible operation which would not otherwise have been possible. Figure 4 depicts the gross electrical generation by the nuclear plant from July 1963 through January 1964.

A. PLANT CAPABILITY

Steam conditions of 825 psig 860°F as delivered to the turbine at 100% load were better than the base design figure of 800 psig 825°F. The reactor plant also developed greater than the 75 net electrical megawatt rating. These tests were conducted with 140 fuel elements. Space is provided for a maximum of 182 fuel elements. Response of the plant to load changes has been tested to the design of 5 electrical megawatts per minute with no transient problems.

B. PLANT AVAILABILITY

A very short base line must be used to discuss plant availability. Considering the months of October 1963 through January 1964 results in a plant availability of 80.8%. Considering the months of November 1963 through January 1964 results in a plant availability of 97.4%. Once the initial problems have been resolved an annual availability factor of 90% or higher (including outages for fuel changes) seems possible.

C. MAINTENANCE

At this stage of plant operation, the primary objective of the maintenance group is to effect repairs in a manner to reduce the frequency of failure. A great deal of this type of work has been accomplished. A number of such modifications are currently being planned for incorporation into the plant this year. Routine calibration, inspection, and maintenance schedules are being revised as experience develops. The two-week decay period required before access can be gained to primary sodium cells has not been a problem. Scheduled shutdowns are planned to accomplish other work prior to entry of the sealed cells. No unanticipated problems have developed in maintaining the sodium system.

VI. SIGNIFICANT PROBLEMS

In the course of the extensive test program many minor equipment and design discrepancies were discovered and corrected. Also, modifications were made to improve plant safety and operability. These were the usual problems encountered in the startup of any large plant. There were also a few problems of a more serious nature which caused delays to the program.

A. INTERMEDIATE HEAT EXCHANGERS

In November 1962, a leak occurred in one of the Circuit No. 1 heat exchanger units. Reference 5 describes the heat exchangers, the failure and repair. The loop was shut down and drained in accordance with the standard operating procedures. Plant testing was continued with the number one circuit isolated. In December, the reactor was shut down and the damaged unit was removed. Plant testing was resumed with the number one circuit limited to one-half design sodium flow. The broken tube (only one) was identified and the heat exchanger was washed with anhydrous ammonia (NH_3) to remove the residual sodium. Reference 4 describes the ammonia cleaning process and the reasons for selecting the process rather than such alternates as alcohol or low pressure saturated steam.

The damaged tube was removed by grinding out the weld and pulling the tube through the tube sheet. To fully analyze the problem, seventeen tubes were removed.

The failure was caused by flow induced tube vibration. The inlet baffle arrangement allows part of the sodium to bypass the baffle and impinge directly on the periferal tubes. This local high velocity sodium stream caused the tubes to oscillate in a rotary manner so that the tubes were worn and overstressed where they passed through the support baffles. Tests conducted by Mr. R. C. Baird, consultant, on the failed heat exchanger showed that stiffeners installed between the tubes at the point of maximum displacement would prevent the vibration. Other testing demonstrated that the affected tubes were not structurally damaged by the vibration prior to actual failure. Tensile and burst tests on tubes removed from the heat exchanger showed no tendency of the tubes to fail preferentially in the baffle areas. Therefore, it was concluded that, if the vibratory oscillations could be prevented, there was no reason to expect additional failures.

The failed unit was modified by the insertion of stainless steel shims between the tubes as shown in Figure 5. Similar modifications were performed on the other five units with the units in place. Vibration measurements conducted by the International Research and Development Company engineers detected no vibrations at the frequencies of interest. These tests demonstrated the efficacy of the modification and all heat exchangers were returned to unrestricted service in May 1963.

B. MAIN SODIUM PUMPS

Sodium is circulated through the main sodium loops by six vertically mounted, centrifugal, overhung, free-surface pumps. Figure 6 shows the essential components of the secondary pumps, the primary pumps are identical except for longer shafts and the radiation shielding. Reference 6 describes the HNPF pumps and their operating history.

1. <u>Problems Created During IHX Repair</u>

While draining the intermediate heat exchanger, helium from the secondary loop was allowed to pass through the failed tube into the primary loop. The secondary cover gas was at a higher pressure than the primary. This higher pressure forced sodium up into the annulus around the pump barrel and also around the pump shaft. When the sodium reached the lower temperature areas of the pump case it froze. The pump was removed, washed and inspected.

The sodium was removed from the pump barrel by steaming in the pump wash cell. It was necessary to use a steam lance to remove the sodium around the pump shaft. Inspection of the pump showed that it had not been damaged in any way by the excessive sodium level, so it was returned to the system.

2. <u>Helium Entrainment</u>

A small stream of sodium is channeled from the pump discharge to the hydrodynamic bearing which centers the pump impeller. This sodium, and that which leaks by the seal rings, is taken out of the pump case by leakage through weep holes through the pump impeller and by the overflow line. The weep holes were sized to approximately balance the inleakage so that flow through the overflow line is minimized. The secondary pumps operated as expected, but early operation of the primary pumps indicated that sodium was drawn through the weep holes at a rate faster than it was supplied by the bearing and by leakage through the seal rings. At high flow rates this led to the entrainment of helium from the pump case into the sodium stream. Investigation of operation and design parameters disclosed that the pumps were designed for 160 ft discharge head, but early testing with a partially loaded reactor core was conducted with only about 80 ft of system pressure drop. This lower discharge pressure resulted in less than normal sodium flow into the pump case and, therefore, a net flow of sodium from the pump case. Each of the primary pumps was removed from the system and four of the eight weep holes plugged. This was effective in eliminating the entrainment of helium in the sodium stream under all operating conditions.

3. Temperature and Clearance Problems

In February 1963, rubbing sounds were detected in the number two secondary pump. The pump was removed and inspected. It was determined that the thrust bearing had not been properly shimmed to provide the pump impeller with proper clearance to the lower pump case. Differential thermal expansion at the higher temperatures caused the pump impeller to contact the diffuser. Proper shims were installed and the pump was returned to service.

In June 1963, the secondary pumps became difficult to rotate by hand, and could not be started with normal motor power. They were removed for inspection. Quantities of foreign material were found in the close clearances of the wear rings and the hydraulic bearings. Some of this material was identical to the weld material in the steam generators and is assumed to have been debris left in the steam generators during fabrication. Concurrently, excessive circumferential temperature gradients existed around the secondary pump cases. These were caused by uneven cooling of the pump cases due to interruptions of the convective cooling paths by structural members. The combination of reduced clearances due to the deposition of foreign material and pump case warpage due to uneven distribution led to binding of the pumps. The pumps were cleaned and reinstalled. A forced, ducted cooling system was installed with provision for adjusting the circumferential air distribution and pump case deflection monitors (plumb bobs) were installed.

4. Helium Leakage from Seal Oil

In October 1963, the number two secondary pump bound up. Disassembly of the gas seal disclosed sodium. Operating records indicated that the loop had been operated for several days with an excessively high sodium level. Sodium at this level isolated the pump shaft from the vent so that the sodium was forced up the pump shaft annulus into the oil seal. The pump was removed, steam cleaned, inspected, and reinstalled. The vent system has been modified to provide a positive helium supply to the oil seal.

C. CONTROL ROD THIMBLES

The HNPF control rods move inside closed-end thimbles which isolate the rods from the reactor coolant. The original thimbles were constructed of Zircalloy II. The thimbles are filled with helium at about two atmospheres pressure to minimize corrosion and improve heat transfer from the neutron absorbing material. This also insures that a leak from the thimble will not admit oxygen to the sodium. In June 1963, a leak occurred in one of the control rod thimbles. The thimble was replaced with a spare and, since the failure represented no hazard, testing was resumed. In August 1963, another control rod thimble failed. The symptoms were identical to those of the first failure. The failures were caused by massive hydriding of the zirconium, leading to embrittlement of the material. The hydrogen was released from a titanium-hydride shield installed to minimize activation of the control rod pull tubes.

The thimbles were replaced with new thimbles made of type 304 stainless steel. The loss in reactivity (about one dollar) does not seriously affect the fuel management schedule. A gas chromatograph was added to the control rod atmosphere system to allow evaluation of hydrogen release. A purification system was added to remove hydrogen. Operation has demonstrated that hydrogen is indeed being added to the control rod thimbles, but at a decreasing rate, and that the purification system can maintain the hydrogen concentration below 200 ppm.

D. STEAM GENERATOR CARRYOVER

The steam generators are horizontal, kettle type units designed for 251,000 pounds per hour steam flow at 850 psig and 833 degrees F. Operation of these units at near design steaming rates indicated water carryover in excess of the 0.25% guarantee. Through testing, and evaluation of operating conditions, water level in the steam generators was determined to be five inches above the recommended level. Design steaming rates were attained, at less than 0.25% moisture carryover, by lowering the water level. The level controllers have been modified to automatically program water level as steaming rate is varied. Testing is being continued to insure that operation under varying loads will not result in either excessive carryover or in operation with the top sodium tubes uncovered.

VII. PLANS FOR FUTURE OPERATION, PLANT EXPANSION

Station load factor has considerably increased since 1961. Contractual arrangements for purchase and interchange of power are such that system load growth is essentially added to this station.

A second turbine generator is on order and will be ready for service in 1966. The present conventional boiler will be converted to its original design rating to supply steam to the second turbine at 1800 psig 1000°F/1000°F reheat. At that time, the present turbine-generator will be supplied solely from the reactor plant. Both units will be operated from the common control room. It is anticipated that no additional personnel will be required to operate both units.

Capability of the present turbine generator is greater than the present authorized power of the reactor plant. Original plant design gave consideration to increasing the reactor power. A future goal is to demonstrate that the reactor plant can be uprated to match the turbine capability.

Core II fuel will be Uranium Carbide. Fabrication of this fuel is in progress. A conservative fuel element design is being used to achieve a peak burnup in excess of 20,000 MWD/MTU.

VIII. SUMMARY

Testing and early operation of the HNPF has demonstrated the safe, simple operation of the sodium cooled-graphite moderated reactor. Equipment failures encountered were of the type foreseeable in the startup of any large, complex industrial plant. The inherent safety features of the low pressure liquid metal coolant with its high retention of fission products; the high thermal efficiency attainable with high pressure, high temperature superheated steam; the ease of control and the simplicity of operation and maintenance; justify the continuing interest in the sodium graphite reactor concept as a vital part of the nuclear power program.

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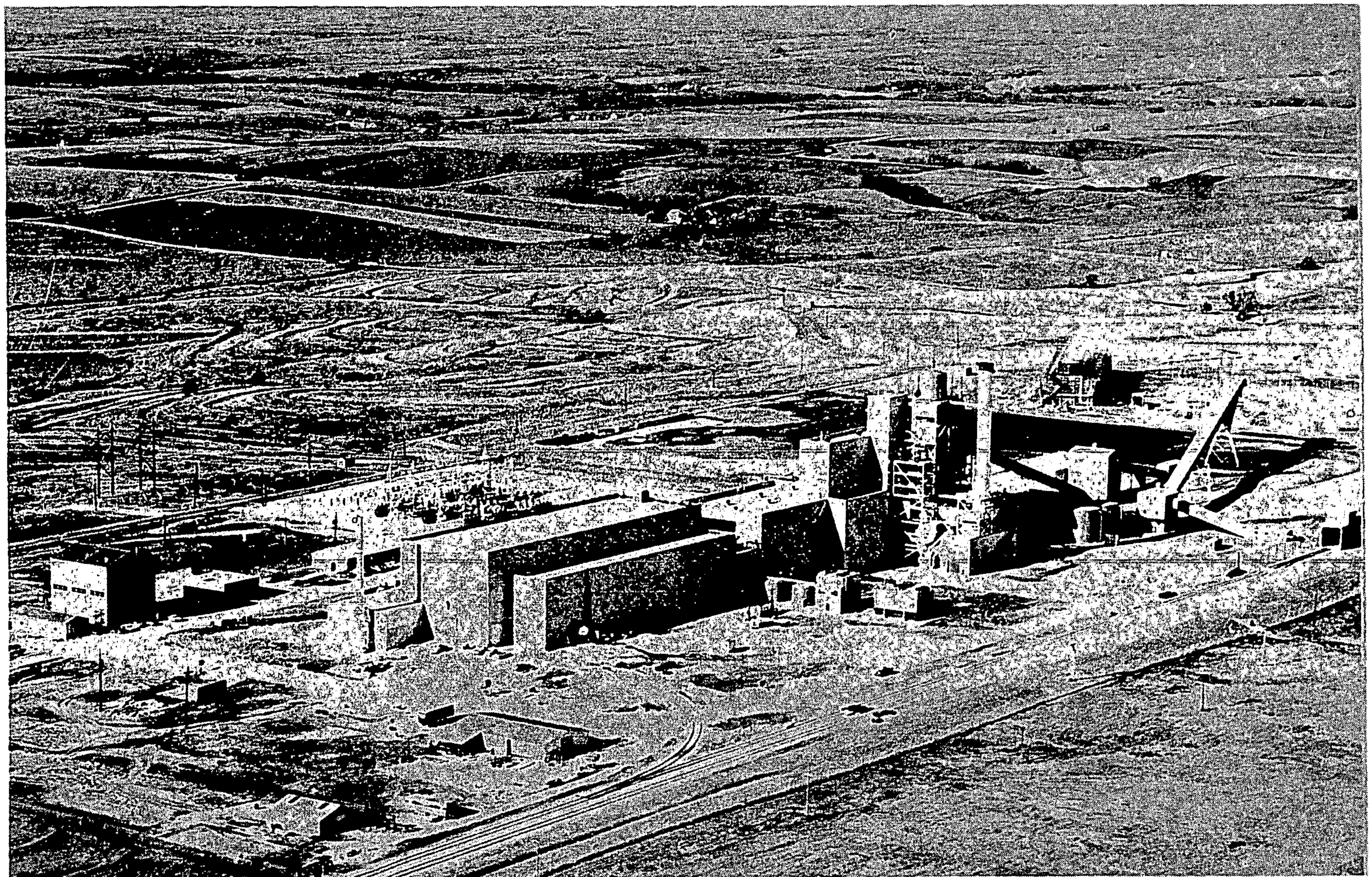

Figure 1. Aerial View of Sheldon Station

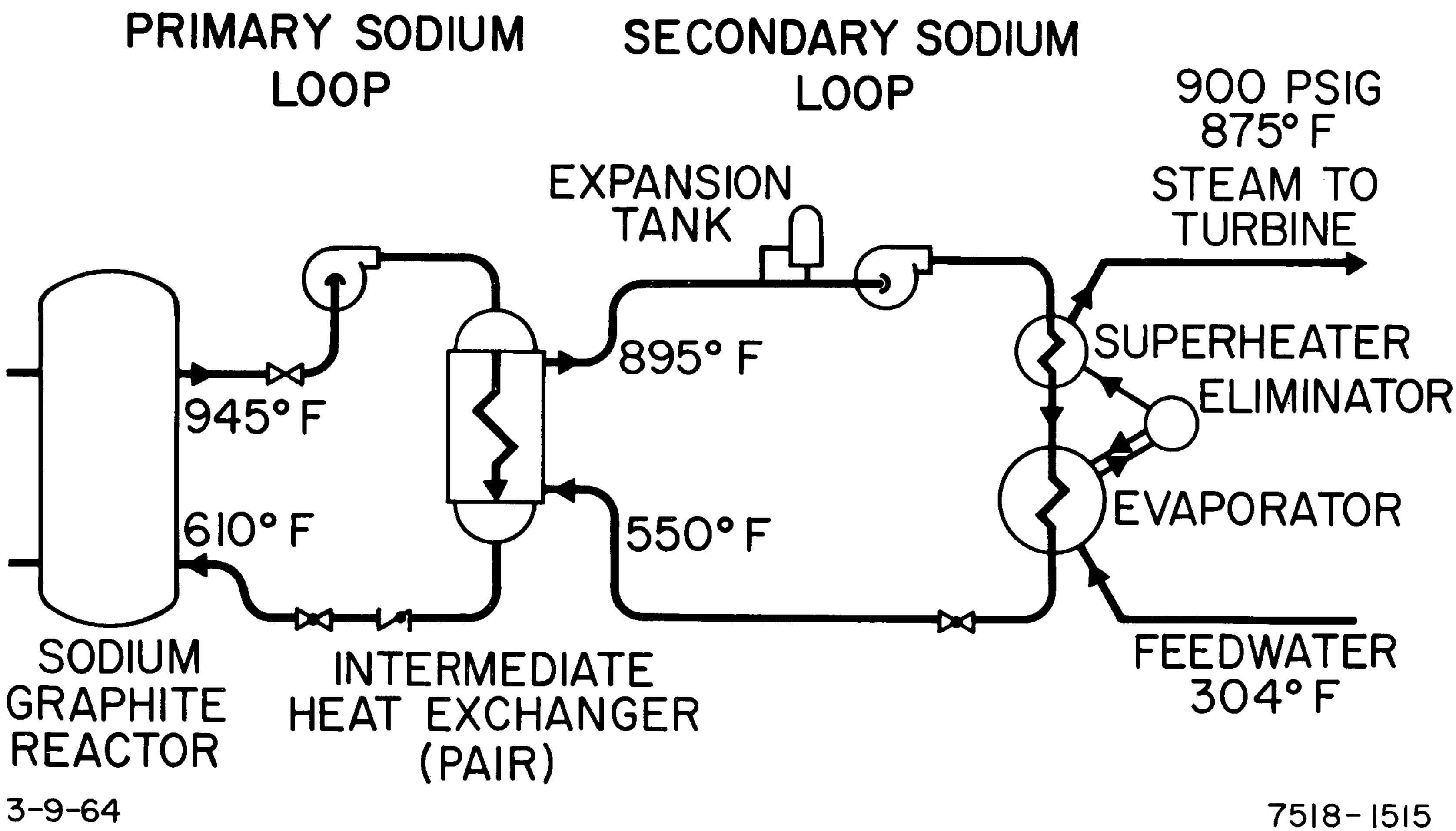

Figure 2. HNPF Sodium Heat Transfer System

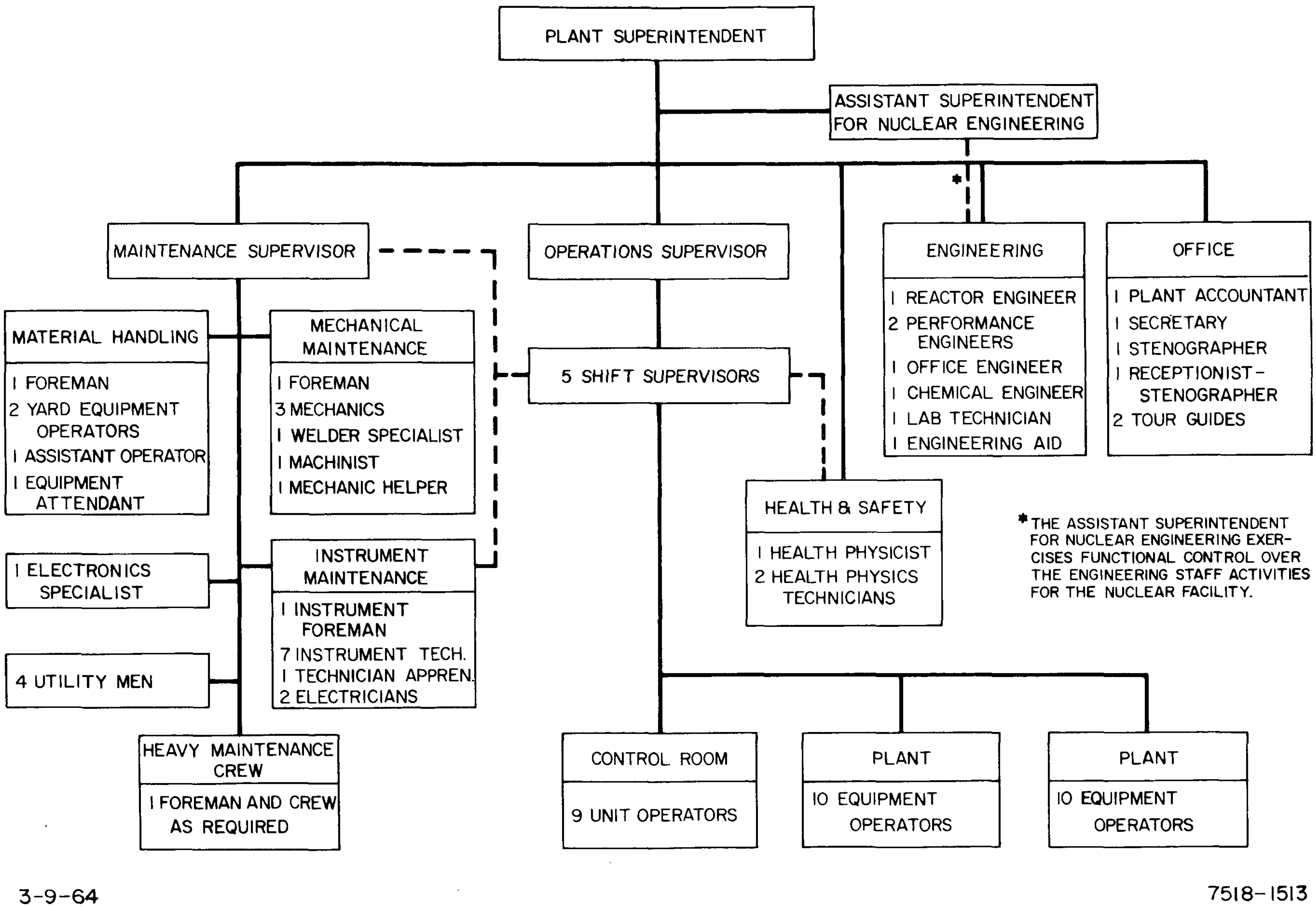

Figure 3. Consumers Public Power District — Sheldon Station Organization

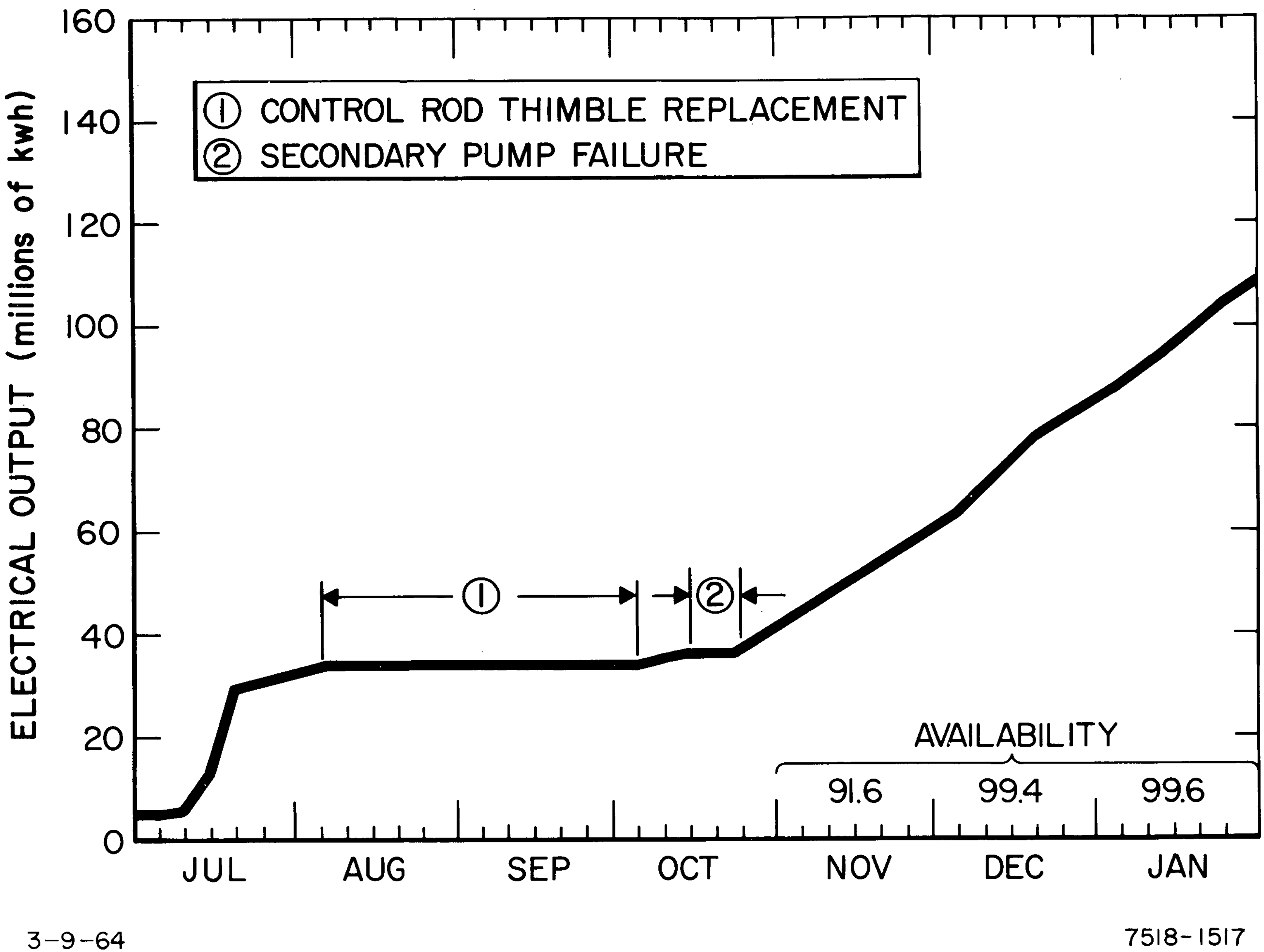

Figure 4. HNPF Gross Electrical Generation, July 1963 Through January 1964

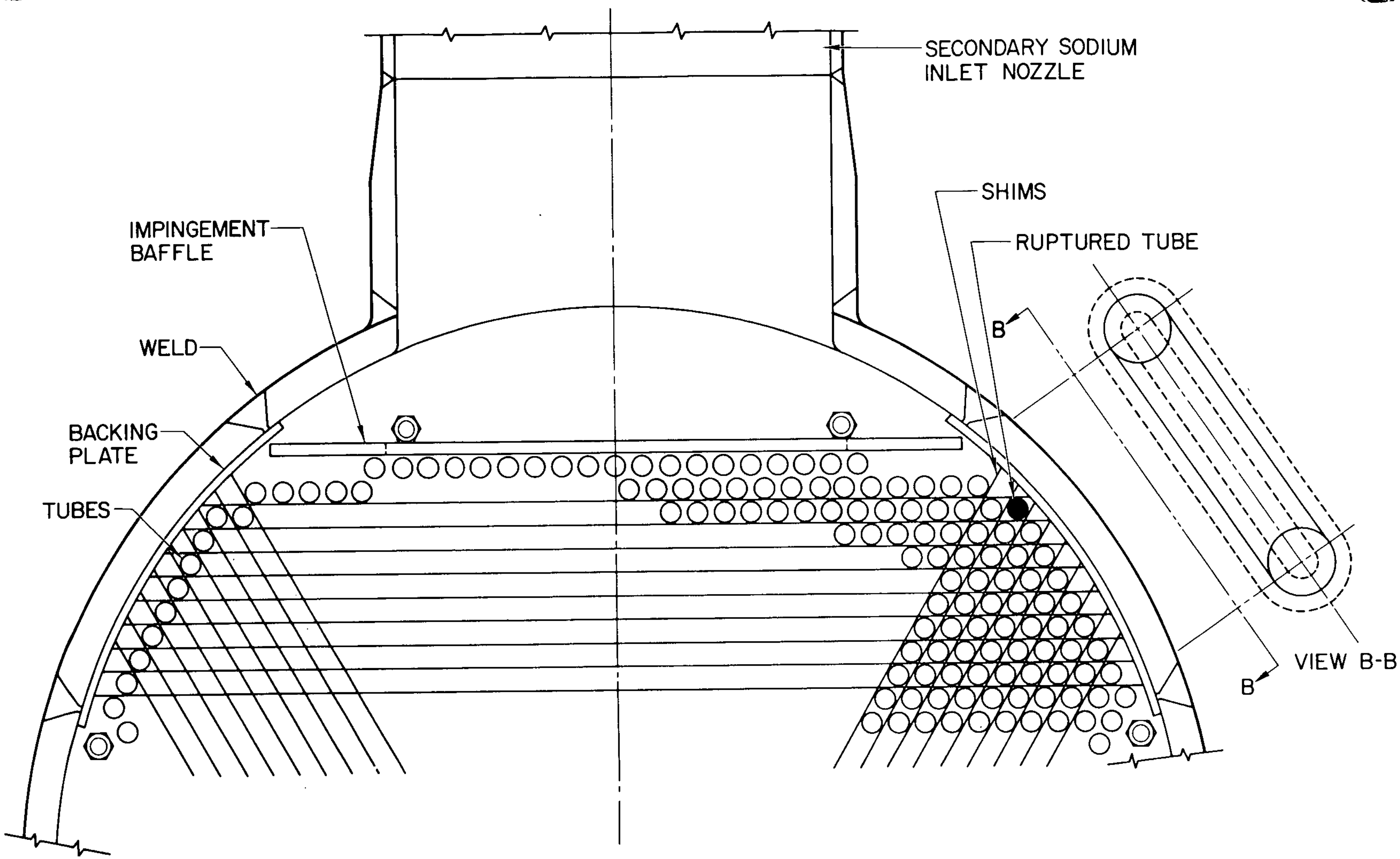

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Figure 5. Heat Exchanger Tube Bundle Cross-Section Showing Shim Installation

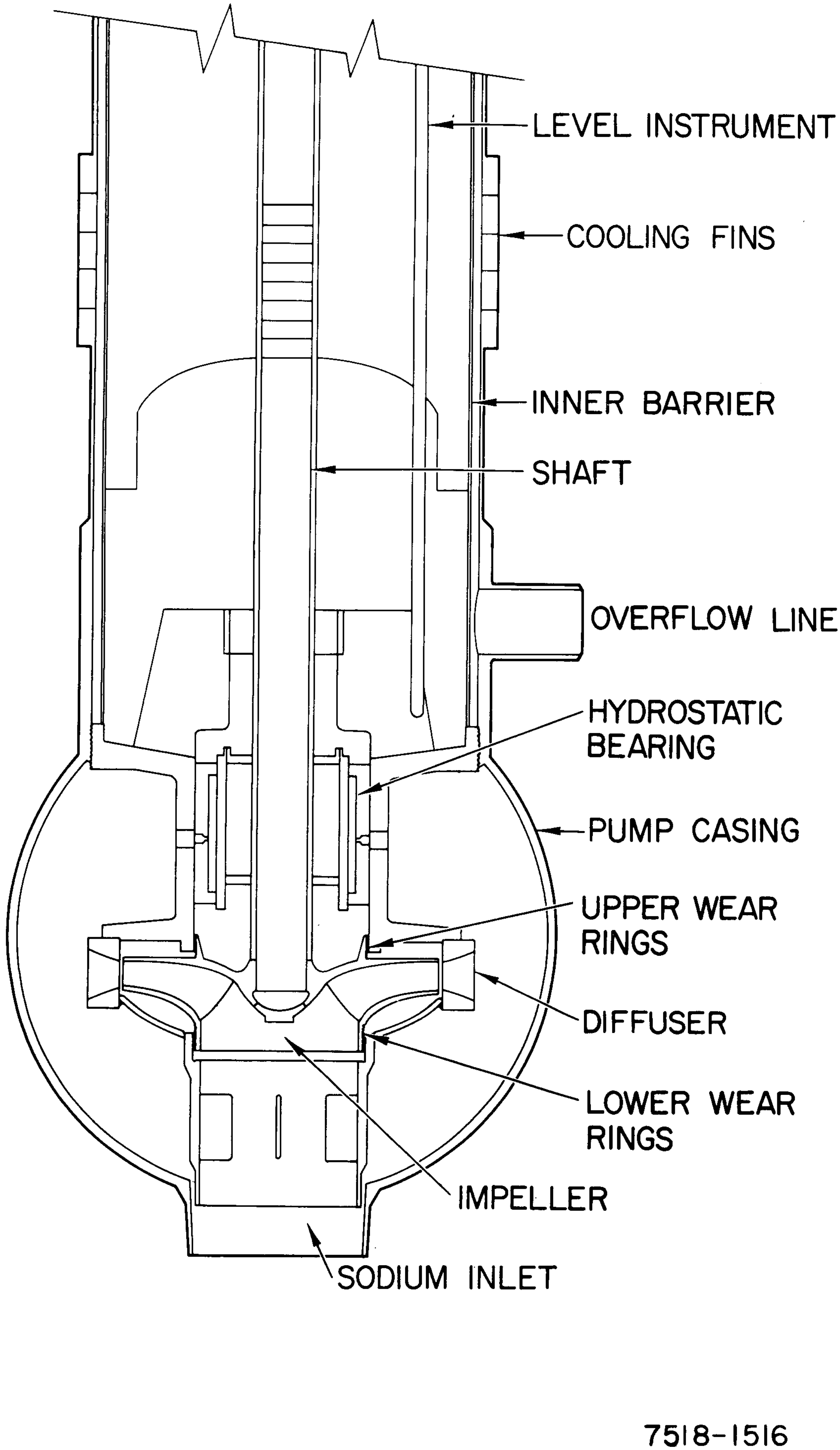

Figure 6. HNPF Sodium Pump