FINAL EVALUATION REPORT
HALLAM NUCLEAR POWER FACILITY
VOLUME I

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MASTER CONTENTS

Volume I

Summary Evaluation

AI-P-1103 Main Heat Transfer System

AI-P-1104 Sodium Service System

AI-P-1106 Cooling Water System

AI-P-1107 Reactor Loading Face Shield Cooling System

AI-P-1108 Helium System

Volume II

AI-P-1109 Nitrogen System

AI-P-1110 Radioactive Vent System

AI-P-1111 Radioactive Liquid Waste System

AI-P-1113 Auxiliary Steam System

AI-P-1116 Preheating System

AI-P-1117 Instrument Air System

AI-P-1118 Emergency Power System

AI-P-1119 Nuclear Instrumentation System

AI-P-1120 Plant Control System

Volume III

AI-P-1121 Plant Protective System

AI-P-1122 Radiation Detection and Monitoring System

AI-P-1130 Lighting, Power and Communication System

AI-P-1133 Wash Cells

AI-P-1137 Portable Purge Unit

AI-P-1138 Decontamination Room

AI-P-1139 Maintenance Cell

AI-P-1145 Control Rod Drive and Actuator Assemblies

AI-P-1146 Control Rod Actuator Support Structure

AI-P-1148 Control Rod Operability

AI-P-1149 Neutron Chamber Cart Drives, and Cables

AI-P-1150 Purge of the Secondary Sodium System

MASTER CONTENTS

Volume III (Continued)

- AI-P-1151 Purge of the Primary Sodium System and Reactor Vessel With Helium
- AI-P-1152 Sodium Leak Detection System
- AI-P-1154 Temperature Instrumentation Using Thermocouple Elements
- AI-P-1155 Dry Criticality
- AI-P-1156 Initial Fill of Secondary System With Sodium
- AI-P-1157 Initial Fill of Primary System With Sodium

Volume IV

- AI-P-1158 Wet Critical Test
- AI-P-1159 Sodium Level Instrumentation
- AI-P-1160 Sodium Flow Instrumentation
- AI-P-1161 Sodium Pressure Instrumentation
- AI-P-1163 Dry Excess Loading
- AI-P-1164 Reactor Heaters
- AI-P-1166 Protective System Modification for Fuel Loading
- AI-P-1167 Hot Sodium Circulation

Volume V

- AI-PC-510 Hydraulic Test of Main Heat Transfer System
- AI-PC-530 Heat Transfer Tests
- AI-PC-552 Steam Generator – Steam Purity
- AI-PC-560 Cooling System
- AI-PC-620 Plant Control System
- AI-PC-630 Plant Protective System
- AI-PC-640 Emergency Feedwater System
- AI-PC-660 Scram and Loss-of-Load
- AI-PC-690 Radiation Shielding

Volume VI

- AI-PC-710 Wet Excess Fuel Loading
- AI-PC-740 Axial Power Distributions

MASTER CONTENTS

Volume VI (Continued)

AI-PC-750 Control Rod Calibration
AI-PC-760 Zero Power Coefficients of Reactivity
AI-PC-770 Power Coefficient
AI-PC-800 Radial Power Distribution and Fuel Channel
Orifice Adjustment
AI-PC-830 Master Schedule Including Rise-to-Power
AI-PC-840 Xenon Poisoning

CONTENTS

	Page
Master Table of Contents	i
Abstract	vii
I. PLANT DESCRIPTION	1
A. General	1
B. Reactor Core	2
C. Fuel Elements	6
II. TEST AND ANALYSIS	7
A. Heat Transfer System.	7
1. Intermediate Heat Exchangers	7
2. Sodium Free-Surface Pumps	9
3. Secondary System Modification	12
4. Throttle Valves	13
5. Bellows Seal Valves	17
6. Fuel Channel Orifices	17
7. Steam Generators	18
8. Preheat, Fill and Testing	29
B. Reactor Critical Operation	34
1. Reactivity Coefficients	36
2. Xenon and Samarium	37
3. Power Distribution	38
C. Reactor and Heat Transfer System General	40
1. Control Rod Thimbles	41
2. Control Rods	49
D. Sodium Service System	51
1. Initial Charge and Operation	51
2. Sodium Circulation and Component Performance	51
E. Plant Control System	56
F. Plant Protective System	59
1. Noise	59
2. Temperature and Drift	60

CONTENTS

	Page
3. Operational Amplifiers	62
4. System Interaction	62
5. Operating Limits	63
6. System Adequacy	64
G. Nuclear Instrumentation	64
1. Startup Channels	64
2. Intermediate Range Instruments	67
3. Power Range Instruments	68
4. Instrument Cooling and Reliability	69
5. Nuclear Instruments Summary	69
H. Radioactivity — Detection and Control	69
1. Radiation Detecting and Monitoring System	69
2. Radiation Shielding	71
3. Radioactive Vent	72
4. Radioactive Liquid Waste	72
I. Core Component Handling	72
J. Other Systems	73
III. GENERAL CONCLUSIONS	75

TABLE

1. HNPF Design Features	3
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FIGURES

1. Cutaway View of the Reactor Structure	1
2. Horizontal Cross-Section through the Reactor	5
3. Intermediate Heat Exchanger	7
4. Heat Exchanger Tube Bundle Cross-Section Showing Shim Installation	9
5. Sodium Free-Surface Pump	10

FIGURES

	Page
6. Primary Sodium Throttling Valve	13
7. Steam Generator	18
8. HNPF Steam Generator Water Level Indicators — South Level Column	21
9. HNPF Steam Generator Water Level Indicators — Center Level Column	22
10. HNPF Steam Generator Water Level Indicators — North Level Column	23
11. Steam Generator No. 1 Water Level Profile	24
12. Steam Generator No. 1 Water Level Profile	25
13. Steam Generator No. 1 Performance	26
14. Steam Generator No. 3 Performance	26
15. Steam Generator Performance With Normal Water Level Conditions	27
16. HNPF Steam Generator Velocities	28
17. Zircaloy-2 Control Rod Thimble TZ-12 Damage	41
18. Zircaloy-2 Control Rod Thimble TZ-3 Damage	42
19. Core Map Showing Location of Failed Thimbles	43
20. Zircaloy-2 Control Rod Thimble TZ-6 Damage	47
21. HNPF Packed and Packless Cold Traps	53
22. HNPF Freeze Trap	55
23. Large Area Inductive Loop Caused by System Grounding Pattern	60
24. Cooling Installation	61
25. Neutron Chamber (BF_3) Pulse Transformer and Preamplifier . . .	65
26. Differentiating Modifier with Time Constant Changed	68
27. HNPF Gross Electrical Generation, July, 1963 through January 1964	75
28. HNPF Sodium Heat Transfer System	76

ABSTRACT

This document is intended to provide for the power industry pertinent testing, start-up and operating data on the Hallam Nuclear Power Facility (HNPF), the prototype sodium-cooled, graphite-moderated central station nuclear plant. Designed and placed in operation by Atomics International, this 80 Mwe plant was turned over to Consumers Public Power District of Nebraska on February 6, 1964, following successful completion of required testing. At the time of turnover, approximately 15,531 Mwd of thermal energy generation had been logged, yielding 110,978 Mwhe total generation.

Plant construction was initiated in March 1959. Beneficial occupancy began in August 1961, with start of preoperational testing. January 1962, marked initial dry criticality at the HNPF; critical loading was 31 fuel elements (U-Mo). Following dry excess loading to 42 elements, the reactor was unloaded. After six months of further system checkout and acceptance testing, permission was received to load the reactor for wet criticality. Loading of the first fuel increment was accomplished on August 14, 1962. At 1306 hr, August 25, 1962 the sodium filled reactor was brought to criticality with a 53-element loading. Wet excess loading and rise-to-power testing followed, capped by the attainment of 100% full power in July 1963. The formal post critical testing program was finished in August 1963, with completion of activities being conducted under the Master Schedule, AI-PC-830.

Problems encountered and corrective actions or modifications performed prior to culmination of the formal testing program are included in the test completion reports comprising the greater part of this document. Each report is divided into five sections, delineating purpose, method, results, compliance with criteria, and an independent evaluation of system adequacy. Key activities not included within the scope of the formal test program, are summarized in the opening sections of this report. Where the activity has not been formally documented by Atomics International, a more detailed report is presented.

Data showing the successful testing and operation of this sodium-graphite-reactor-powered steam generating plant are supplemented with recommendations for improving plant reliability and performance. Areas discussed include support equipment as well as major plant systems.

I. PLANT DESCRIPTION

A. GENERAL

The Hallam Nuclear Power Facility (HNPF) consists of a thermal neutron reactor and associated process and auxiliary equipment for producing steam to operate the turbine-generator at Consumers Public Power District fossil-fired plant, Sheldon Station. Both plants are located in a single structure (see cover) at Hallam, Nebraska, 19 miles southeast of Lincoln.

The reactor (Figure 1) is fueled with slightly enriched uranium-molybdenum alloy. The graphite moderator is contained in stainless-steel elements, and liquid sodium is used as the reactor coolant. Reactivity control is obtained by moving neutron-absorbing shim-safety-regulating rods into or out of the core.

Figure 1. Cutaway View of the Reactor Structure

When necessary, these rods can be dropped by gravity into the core to shut down (scram) the reactor. The core and supporting structures are contained in a stainless-steel reactor vessel which is surrounded by a thermal shield, a reactor outer vessel, thermal insulation, a gas-tight, water-cooled cavity liner, and a concrete biological shield.

Three identical sodium heat-transfer circuits carry the heat generated in the reactor to the steam generators. Each circuit consists of a radioactive primary loop and a non-radioactive secondary loop coupled through a sodium-to-sodium intermediate heat exchanger (IHX). Steam from all three steam generators is fed into a common header which serves a single turbine-generator. The required steam rate at the turbine throttle is 710,000 lb/hr at 800 psig and 825°F.

Design features of the plant are listed in Table 1.

B. REACTOR CORE

The HNPF reactor core consists of a moderator element matrix (Figure 2) into which are suspended fuel elements, control rods, and miscellaneous elements. The moderator assembly is approximately in the form of a right circular cylinder 17 ft in diameter by 17 ft high. The initial fuel loading approximately forms a 13-ft-diameter cylindrical core 13.5 ft high surrounded by a graphite reflector from 1.75 to 2 ft thick.

The initial fuel is U-10 wt % Mo alloy enriched to 3.6 wt % U^{235} . This provides 8% excess reactivity in the reference 137-fuel-element core, hot and clean. Reactivity requirements are 2.5% for equilibrium xenon and samarium poisoning and 0.5% for control, leaving 5.0% for long term fuel depletion and fission product buildup. The initial fuel is designed to achieve a peak exposure of ~5,000 Mwd/MTU. Approximately 14 additional elements have been fabricated using 3.5% enriched fuel from the exponential and critical facilities, and are included in the current core loading.

Control rod poison columns move vertically inside stainless steel thimbles which are supported and sealed at the loading face and extend to the grid plate beneath the core. The control rod worth of 12.5% provides a shutdown margin of 6% for the most reactive core under normal operating conditions.

The primary sodium inlet pipes enter the reactor cavity just above the tops of the moderator assembly and run downward to the lower part of the reactor

TABLE 1
HNPF DESIGN FEATURES

Reactor Nominal Full Power Rating	240 Mwt
Reactor Design Base Rating	256 Mwt
Reactor Core	
Cavity depth	46 ft
Reactor vessel dimensions	19 ft dia by 33 ft high
Active core dimensions	13 ft dia by 13.5 ft high
Average thermal neutron flux in fuel	1×10^{13} neutrons/cm ² -sec
Axial peak to average flux	1.5
Radial peak to average flux	1.4
Initial conversion ratio	0.4
Reference core positions, initial loading fuel*	137
control rods	19
neutron source	1
sodium temperature elements	3
sodium level instruments	2 (not in active core)
dummy elements	45
total core positions	205
Fuel element	
type	18-rod cluster
fuel material	U-10 wt % Mo alloy
fuel enrichment*	3.6 wt % U ²³⁵
cladding material	Type 304 stainless steel
overall length	18 ft
active length	13.5 ft
fuel area	4.92 in. ²
weight of fuel	224 kg (U-Mo)
average specific power	256 kw/kg U ²³⁵
peak exposure (objective)	~5,000 Mwd/MTU
Fuel inventory (137 elements)	
U-Mo alloy	30,700 kg
uranium	27,600 kg
Moderator-reflector element	
material	graphite
sheath material	Type 304 stainless steel
sheath thickness	0.016 in.
number of elements	141

TABLE 1
HNPf DESIGN FEATURES (Continued)

Reactor Core (continued)	
Moderator-reflector element (continued)	
configuration	hexagonal prism
length	17 ft
width across flats	16 in.
sodium gap between elements	0.160 in. (average)
weight	2400 lb
Control rods	
poison material	$Gd_2O_3 - Sm_2O_3$
number of shim-safety-regulating rods	19
maximum worth of one rod	1.5 to 2% Δk
total reactivity worth of rods	12.5% Δk
rod speed	12.4 in./min
maximum rate of reactivity insertion (19 rods)	0.03% Δk /sec
rod drop time (full-up to snubber)	0.9 sec
Sodium systems (base rating, 256 Mwt)	
flow rate per circuit	2.8×10^6 lb/hr
primary hot leg temperature	945°F
primary cold leg temperature	610°F
secondary hot leg temperature	895°F
secondary cold leg temperature	557°F
Steam systems (base rating, 256 Mwt)	
Superheater	
flow rate	753,000 lb/hr
pressure	851 psig
temperature	851°F
Throttle	
flow rate	752,000 lb/hr
pressure	800 psig
temperature	843°F
Feedwater	
flow rate	768,000 lb/hr
temperature	305°F
<hr/> * Fourteen additional U-Mo elements enriched to 3.5 wt % U^{235} are available from the development program for use as needed.

Figure 2. Horizontal Cross-Section through the Reactor

vessel. The major portion of the sodium enters the lower plenum between the grid plate and the bottom of the reactor vessel and flows upward through the fuel element process tubes to the sodium pool above the core. This fluid removes the heat from the fuel elements. The remaining sodium (~5%) is routed by a parallel piping system to the plenum between the grid plate and the bottom of the moderator elements. From here it flows upward through the gaps between the moderator elements into the upper sodium pool. This sodium removes the heat generated in the moderator. The gap between moderator elements is nominally 0.160 in. wide and is maintained by spacer fittings which are fastened to the top heads of the cans.

Within the reactor cavity, the sodium piping is double-walled. The outer guard pipes extend from the reactor outer vessel to include the blocking valves just outside of the reactor cavity. At the lower ends, the guard pipes open into the reactor outer vessel. Any leaks of liquid sodium interior to the blocking valves will be drained back to this outer vessel. Leak detectors and pumpout

lines are provided. The outer vessel is sized so that a major sodium leak will not allow the sodium level to drop below the outlet nozzles of the primary pumps.

There are no openings into the reactor cavity below the upper surface of the core. The cavity is sized such that, in the event of a major disaster which dumps the primary system sodium (including standby inventory) into the cavity, the active core would remain flooded. Elevations of the primary sodium system were selected to prevent siphoning in case of a leak. The only low point is the primary pump dip leg, which is double-walled to guard against leakage.

C. FUEL ELEMENTS

The fuel for the first core loading is contained in 18-rod fuel bundles. Each bundle consists of an outer ring of 12 rods and an inner ring of 6 rods, enclosed in a Zircaloy-2 process tube. A hollow central tube carries spacers, located at 1-ft intervals, which maintain the spacing between rods. A fuel rod consists of solid cylinders of U-10 wt % Mo, 0.590 in. in diameter, and 3 to 12 in. long in a Type 304 stainless-steel tube. The total active fuel length is 13.5 ft. The cladding tube has 0.660-in. OD x 0.010-in. wall thickness and is welded at each end to an end cap. The gap between the fuel slugs and the cladding (nominally 0.025 in.) is filled with sodium to assure good thermal contact. Each rod has an 18-in.-long space at the upper end for potential radiation growth of the fuel and buildup of fission product gases. The tube wall along this gas space is supported internally by a coil spring. The design requirements for the gas space are to allow 6 vol % fuel growth and 2 vol % fission gas release. A peak fuel burnup of ~ 5,000 Mwd/MTU is expected for spent elements, which will have an average burnup of ~ 3000 Mwd/MTU.

The maximum design temperature for the U-Mo fuel is 1250°F, based on expected radiation growth of the fuel. This allows a maximum fuel rod power of 19.4 kw/ft, and a peak power of 2.5 Mw in any fuel channel. The maximum fuel temperature expected in the reactor is 1184°F. The 137 elements for the initial loading has an enrichment of 3.6 wt % U²³⁵.

II. TEST AND ANALYSIS

A. HEAT TRANSFER SYSTEM

Jointly, the preoperational (AI-P-1103 and -1167) and post critical (AI-PC-510 and -530) tests performed on the HNPF heat transfer system served to demonstrate the adequacy of system performance over the operating range from zero to full power. Several modifications generated on the basis of this testing helped improve overall soundness of the Hallam concept.

1. INTERMEDIATE HEAT EXCHANGERS

Resonant vibration caused a tube failure in one Intermediate Heat Exchanger (IHX) (Figure 3) in November of 1962.* The failure was discovered by Operations personnel during investigation of an unexplained sodium level decrease in secondary expansion tank No. 1. Follow-up checks verified the situation, and the investigation for cause was initiated. Lower operating pressure on the primary side allowed the sodium to drain in the safe direction, as anticipated in the plant design.

Following leak checking to determine which IHX unit was leaking, procedures were developed for removal, internal cleaning and inspection of the unit. The entire unit, including supporting framework, was lifted from the IHX cell and placed on the highbay floor, for further examination. By gas pressurization and use of a sonic detector, the ruptured tube was isolated and the leak pinpointed.

Figure 3. Intermediate Heat Exchanger

*R. E. Durand, "Operating Experience with Intermediate Heat Exchangers at the Hallam Nuclear Power Facility", (NAA-SR-9718) Atomics International, April 1964.

Next, the unit was cleaned internally with liquid ammonia,^{*} and the damaged tube was removed. In all, 17 tubes were pulled from the IHX to check for markings similar to those found on the failed tube, and to allow for testing and determination of cause.

All pieces of the failed tube were subjected to metallurgical and mechanical examination. This examination indicated that the failure was due to vibration-induced fatigue in the baffle plate area. Tests of the failed tube material showed no defects. Sections of tubing of the same diameter, wall thickness, and material were force-vibrated until failure occurred. The failure of test sections in all cases was strikingly like that of the failed IHX tube. Tensile tests of the tubing showed that the tubing material remained ductile.

Vibration measurements were then initiated on the operating IHX's. It was determined that vibration amplitude did not vary with primary sodium flow, but did vary with secondary sodium flow, indicating that the vibration was a function of secondary sodium flow only. Through frequency measurements made in the secondary sodium system piping, the vibration was shown to be caused by internal flow characteristics, rather than external energy.

Analysis of the pertinent factors involved pointed up that if in-phase vibration of the tubes occurred, the amplitude of vibration could be greater than the tube sheet ligament thickness, thus enabling the tube material design endurance limit to be exceeded. Failure of the tube in the baffle area, rather than in the tube center where a fatigue failure of this nature might be expected, is explained by tube bowing. At a critical flow velocity, vibration of the first span was induced transverse to the plane of bowing by fluid drag forces. At high velocity flow, the bowing caused alternating deflections in subsequent spans, and the edges of each baffle aperture provided stress-raising points. Alternating rotational flow is considered the cause of vibration. The combination of static and dynamic deflection causes rolling point contact, with violent impacting.

Damping shims were selected as the easiest means for eliminating the velocity-induced resonant vibration. These shims were installed between tube rows, in the high velocity area, for additional support to counteract deflection tendencies, while simultaneously providing vibration inhibiting friction.

^{*}R. L. Robb, "Cleaning of a Sodium Heat Exchanger at the Hallam Nuclear Power Facility", (NAA-SR-8314) Atomics International, May 1963.

Shims were installed in all IHX units (Figure 4) after an experimental setup on unit 1A proved successful. The shims were installed between the tubes by insertion through a slot cut in the side of the unit. After insertion, the shims were tack welded to a backing plate with a slot cut down the center. The slot cut in the shell was then plugged with the original material, and rewelded. Further vibration tests and subsequent operation have proved the modification successful. It is important to note that, although an IHX tube failed through vibration, there was no hazard to the plant or personnel, as provided by plant design characteristics.

2. SODIUM FREE-SURFACE PUMPS

At Hallam, six vertically mounted, centrifugal, overhung, free-surface pumps are used to circulate the system coolant. There are three primary and three secondary pumps, identical except for shaft length and shielding. The

Figure 4. Heat Exchanger Tube Bundle Cross-Section
Showing Shim Installation

three primary pumps have longer shafts to accommodate the biological shielding which is not required in the secondary system pumps (Figure 5).*

All six of the Hallam pumps have been removed from the system at one time or another, for examination or modification.

Primary pump No. 1 was the first pump to be removed (December 1962). Following the IHX-1A tube failure in November 1962, secondary loop No. 1 was drained to prevent further transfer of sodium through the tube crack. Draining was accomplished by pressurizing the loop with helium to ≈ 15 psig. When the sodium level dropped below the leak, some of the gas was forced into the primary system, and in turn forced sodium into the casing of primary pump No. 1. Since the upper part of the casing is not heated, the sodium solidified around the shaft, thus preventing the shaft from turning.

The pump was removed to allow determination of the cause of binding. At that time it was decided to take advantage of the removal to correct a sodium level control problem within the casing. It had been discovered that with the pump operating at 60 to

Figure 5. Sodium Free-Surface Pump

*R. E. Durand, "Operating Experience with Heat Transfer Pumps at the Hallam Nuclear Power Facility", (NAA-SR-9717) Atomics International, April 1964.

100% of design flow, but at a head lower than the 160 ft @7200 gpm design value, a low sodium level existed within the pump casing. This low level was attributable to a total system pressure drop well below the 38.2 psig required to maintain pump casing sodium level at full flow.

Hydraulic analysis of the internal circulation paths within the pumps verified that, at higher speeds, it was possible for more sodium to be pumped from the pump case through the impeller weep holes to the suction side of the impeller, than from the pump discharge through the various flow paths into the pump case. Following the pump manufacturer's recommendations, four of the eight impeller weep holes were plugged to restore the hydraulic characteristics necessary for a net flow of sodium into, rather than out of, the pump case. This also served to eliminate the potential of entraining gas in the system through the suction side of the pump. Subsequent testing verified the effectiveness of this modification; primary pumps No. 2 and No. 3 were removed later, at a convenient point in the start-up program, and modified in the same manner as pump No. 1.

Problems encountered with operation of the secondary pumps involved reduced running clearances due to thermal deformation of the pump cases, and binding as a result of foreign metallic particulates.

During the initial rise to power (February 5, 1963), when system temperatures were increased to design levels, secondary pump No. 2 began emitting rubbing sounds. Subsequent removal and inspection revealed that the thrust bearing was not properly shimmed to establish the required impeller axial clearance. At elevated operating temperatures, there was sufficient differential expansion between the shaft and inner barrel to cause the impeller to rub against the diffuser.

In June 1963, during reactor restart from a flow ratio scram, secondary pump No. 2 could not be restarted. Some time later, secondary pump No. 1 could not be turned by hand (standard check for free movement), and secondary pump No. 3 would not increase in speed after being restarted. All three secondary pumps were removed and steam cleaned in the pump wash cell.

Deposits of metallic material were found in each of the three pumps, and internal scoring and burnishing were noted on the bearing and wear ring surfaces. The damage and binding were directly attributed to the foreign material, which

was later analyzed and found to be of the same composition of materials used during initial fabrication of the system, e.g., the material in secondary pump No. 1 matched the chemical composition of weld rod used in fabrication of the steam generators. A contributing factor was deflection or warpage of the outer pump cases, and resultant misalignment of the pump internals. Although the misalignment was too slight to cause seizure, the running clearances were reduced sufficiently to cause seizure upon accumulation of the foreign material.

Diametral running clearances were increased for the upper and lower wear rings. Correction of the temperature distribution problems was effected by installation of a 500 scfm blower, ducting and a distribution manifold to blow air upward past the cooling fins.

As a precaution, the 10-micron sintered metallic filters used at the drum melt stations for initial fill of the system were installed to permit bypass filtration of secondary system sodium. No further foreign particulate problems have been encountered. Overall performance of the six heat transfer pumps has been satisfactory. No problems inherent to the pump design have been encountered. The pressure drop difficulties which necessitated plugging of the impeller weep holes was simply because the system head curve was not as steep as had been expected. All of the difficulties actually requiring plant down time to allow corrective action were due to extraneous problems not associated with pump design.

3. SECONDARY SYSTEM MODIFICATION

Early in the preoperational testing it was found that loop cover gas (helium) was being entrained into the sodium and distributed through the system during operation at elevated pump speeds. Testing indicated the expansion tanks to be the origin of entrainment.

This problem was eliminated by effecting a secondary system loop design modification whereby ~96% of the flowing sodium bypassed the expansion tank. The bypass action was accomplished by cutting and capping the inlet line to the expansion tank, and diverting the flow directly to the pump suction line. A three-inch-diameter bypass line was installed to permit a small flow through the expansion tanks, to provide for continuous degasification. (Figure 5 in AI-P-1103, Main Heat Transfer System, shows the modification.) Subsequent system operation has been satisfactory, indicating the efficacy of the modification.

4. THROTTLE VALVES

Malfunctions associated with the heat transfer system main throttle valves proved quite bothersome. During initial operation, three throttle valve and valve operator failures occurred as the valves underwent extensive testing at temperatures from 350°F to 540°F. Figure 6 is a detailed drawing of a primary throttle valve.

Two of the early failures were due to misalignment of the actuating mechanism, and one was the result of internal binding. Subsequent to completion of corrective maintenance, five of the six valves were cycled over the full travel 30 – 40 times. Primary throttle valve No. 1 showed evidence of excessive stem actuating force requirements.

The valve actuating difficulties resulted partly from the accumulation of sodium oxide in the stem seal, which increased the torque required to activate the valve. There was also a 25% natural starting friction (resistance) to overcome. Further valve operator failures were encountered, leading to several modifications.

Reinforcing bars mounted on the yoke underside constituted the earliest modification to the primary throttle valves. This same modification was performed on the yokes of the three secondary throttle valve operators. A later failure in which the stem of primary throttle valve No. 3 was bent indicated the need for more extensive modifications. Strengthening of the yokes had lessened the center bending which was causing binding of the spindle nuts; the improvement, however, was at the expense of relocating the probable bend point from the center to the ends of the yoke.

Figure 6. Primary Sodium Throttling Valve

In March 1963, all six throttle valves were modified. The work was designed to preclude valve shaft binding experienced as a result of rapid wear and subsequent grooving of the brass thrust-washer upper surface. Included in the modifications were:

- a) Tightening of all throttle valve handwheel clutch mechanism friction adjustment screws per manufacturer's instructions.
- b) Increasing the area of the thrust washer-to-shaft bearing contact surfaces for better loading and longer thrust-washer life.
- c) Modification of the existing keyway, to eliminate friction with the brass thrust-washer.
- d) Cutting of two keyways, 180 degrees apart and 90 degrees from the existing keyway, to stop key shearing by rapid down travel of the shaft.
- e) Resetting of all torque limiters between 165 and 210 ft-lb to prevent damage to the valve actuator mechanism in the event of excessive binding in the mechanism.
- f) Installation of shims to prevent vertical movement when changing direction of travel. The thrust washers at the top of the jack screws were shimmed, so that the thrust bearings on the input shaft, instead of the bushings, would carry the load.

On April 16, 1963, during operator training on the convection flow control system, secondary throttle valve No. 1 was operated cold. As a result, the yoke was bent and, due to failure of one of the yoke drive nuts, the valve stem and jack screws were bent. Following repair of this valve operator, a torque limit check was performed. Torque limit repeatability ranged from 210 to 105 ft-lb for the "open" setting and from 144 to 126 ft-lb for the "close" setting.

Another throttle valve operator failure occurred at this same time. While primary throttle valve No. 1 was being stroked, the valve was opened under power beyond normal operating limits. One of the yoke drive nuts failed, allowing the yoke to "cock". The jack screws and valve stem were bent, and the bushings which support the worm gear and shaft were found to be out of place. The drive assembly for this valve was disassembled to determine the extent of damage. Four of the six preload springs and bushings for the torque limiter were removed for rework, in order to adjust to a lower torque setting.

Tension was reduced on the torque limiter preload springs for each of the throttle valves, to permit lower torque limit settings. A Prony brake was constructed for use in checking and setting torque limits on the throttle valves. A check of primary and secondary throttle valves No. 2 and No. 3 to determine the amount of torque required for operation, yielded readings ranging from 30 ft-lb to 114 ft-lb, indicating a wide spread in operational torque value among the valves.

The drive mechanism on primary throttle valve No. 3 was operated to check clutch tension and obtain torque measurements. It was determined that the torque readings obtained were a function of actuator motor speed vs speed with which the Prony brake was applied. These factors produced a variation of 45 to 60 ft-lb in measurements obtained for a given torque limit setting. Following completion of the checkout, the torque limiter for this valve was set at 200 ft-lb.

At this time, the valve stems of primary and secondary throttle valves No. 1 were straightened using a special jig fabricated at the site. Manual and power operation of the reassembled valves proved the repair to be satisfactory.

Next, the actuator mechanisms on secondary throttle valve No. 1 and No. 2 were disassembled, to permit fabrication and installation of new yoke drive nuts for the drive screws.

In early May 1963, following completion of maintenance and adjustments, all three primary throttle valves were checked under simulated scram conditions. A manual scram was initiated with the primary throttle valves on automatic control along with convection controller TRC-4. Sodium flow at the time was ~88% in all loops; throttle valve torque limit setting was 150 ft lb. Primary throttle valves No. 2 and No. 3 were almost immediately tripped off by the torque limiter and lock-out switches (specially installed a few days prior to this test) after the valves moved away from the upper limits. Primary throttle valve No. 1, not being equipped with a lock-out switch, closed and then re-opened.

A 1/2" x 4" steel plate was welded on the yoke of secondary throttle valve No. 3 after it was determined that the yoke deflected ~0.084" on upward movement and ~0.02" on downward movement. During the modification effort one of the jack screws was found to have excessive end play, causing binding and subsequent overheating. Testing performed after correcting the end play

condition and completing the modification showed marked improvement in valve operation. Secondary throttle valves No. 1 and No. 2 were modified in the same manner.

After completion of the above described modifications, a simulated scram test was performed. Secondary loop flows and throttle valve positions were recorded on Sanborn recorders and valve operation observed by personnel stationed at each valve. All six throttle valves closed and then re-opened as intended. Some of the torque limit switches actuated during the "opening" portion of the cycle.

In late May 1963, following a series of "unreal"* scrams, the yokes of secondary throttle valves No. 2 and No. 3 were found slightly cocked. The bronze bushings on the yokes were readjusted and a cap screw was installed through each bushing into the yoke, to prevent loosening of the bushings.

It was decided to redesign the valve topworks, to eliminate the difficulties encountered during initial operation of the throttle valves. The original design had no provisions to accommodate normal and expected misalignment, tolerances, wear, and deflection.

Over-design of the valve operator, relative to the topworks, resulted in failure of the topworks rather than the operator or shafting. Actuation of the operator under sufficient force to provide necessary starting torque and required speed resulted in damage to the topworks. The only protection provided (the torque limiter) could not be set to both protect the valve topworks and guarantee valve operation.

The above, coupled with a low-efficiency (~25%) gear train, and a static or starting torque 25% over the normal operating torque, required imposing a torque on the valve input shaft (at the operator) in excess of that allowable for the topworks.

The valve topworks were subsequently redesigned to:

- a) Close the gap between required thrust at the valve stem shaft and input torque required at the valve operator.
- b) Provide a uniform and predictable efficiency in the drive train.

*"Unreal" = caused by an instrument anomaly rather than by a hazardous condition in the systems.

- c) Make the design relatively independent of clearance, adjustment, wear, misalignment, and poor lubrication.
- d) Provide a slip clutch at the valve operator to limit the torque in event of stoppage and failure of the torque limiter to protect the unit.

These design requirements were met by replacing the existing valve topworks with a heavy cast frame which supports two ball-screw drives. These ball screws drive a massive cast yoke which is coupled to the valve stem. A slip-clutch was added to the valve operator to limit the torque which can be applied to the valve topworks.

5. BELLOWS SEAL VALVES

Approximately 12% of the bellows seal valves installed in the sodium circulating portion of the HNPF heat transfer system have suffered bellows failures. The failures were generally associated with throttling of the valve for extended periods of time. Under these conditions the bellows vibrates to the extent that it can be detected audibly in the vicinity of the valve when ΔP across the valve is high. These oscillations eventually lead to fatigue failure of the bellows.

Certain critical bellows seal valves will be modified to replace the bellows with ambient-cooled-stem freeze seals, and thus ensure reliability. Six valves (1 on the vent line from the top of each IHX to the expansion tank and one from each loop high point to the expansion tanks) were removed and replaced with orifices. With the remaining bellows seal valves, personnel have exercised care in operation and have limited use of the valves for throttling.

In future plants, use of bellows seal valves should be kept to an absolute minimum. When required, care must be exercised to select a valve which will operate in any position and with high pressure drops without bellows oscillation.

6. FUEL CHANNEL ORIFICES

Binding of the fuel channel orifice plug drives hampered operations. In some cases, two full turns of the drive failed to change the plug position. In others, allowing the drive cable to remain in tension after turning became difficult would eventually cause the plug to move; follow-up adjustment would then proceed as anticipated.

In view of the above problems, the orifice plug drive mechanism was redesigned for the second core loading. The flexible shaft linkage in the actuator assembly has been replaced with a rod in the Core II design. The new rod has a larger top diameter to prevent radiation streaming. In addition to providing greater strength and dependability, the cost of fabrication is also less than the earlier design. Twenty-eight of the new design plugs will be installed during the Core II loadings. The Core I design plugs will remain in service. Introduction of the new plugs should provide refined values of orifice sensitivity, since movement up and down will be smoother. Orifice sensitivity is discussed elsewhere in this report under "Power Distributions," and in AI-PC-800, Axial Power Distribution and Fuel Channel Orifice Adjustment (Atomsics International, February 1964).

7. STEAM GENERATORS

Initial preheat of the steam generators (Figure 7) was hampered by poor preheat temperature distribution. It appeared that the steam heating on the water side was concentrating heat at the steam inlet nozzle rather than

Figure 7. Steam Generator

distributing evenly across the steam generator. This condition was rectified by maintaining a partial vacuum on the steam generators while admitting preheat steam.

During initial rise-to-power testing, difficulties were encountered with the steam generators due to excessive moisture carryover.* Full power operation was achieved for the three generators, by adjusting evaporator water level.

It was determined that the evaporator level recorder-controllers (LRC) were in error and causing excessive evaporator water levels. The error was a function of steam generating rate.

Sample probes (steam generators No. 1, 2, and 3) and "T" bar splash baffles (above the sodium tube bundle in steam generator No. 3) were installed to obtain a better fix on actual water level and water level variation with steaming rate. The sample lines (probes) are used to define evaporator optimum water level, or the elevation of the steam-water interface. Modification of steam generator No. 3 by the introduction of splash baffles was an unsuccessful attempt to decrease carryover. Limited success in overcoming carryover above design steaming rates led to extensive steam generator testing beyond the cognizance of the formal post critical test (AI-PC-552).

The test results presented herein are from the Hallam Nuclear Power Facility Operating Report for December 1963 (No. 17), published by Consumers Public Power District.

a. Presentation of Test Results

To conform with HNPFF Technical Specifications governing other portions of the plant, steam generator testing was conducted at steam flow rates of 2.4×10^5 lb/hr and below.

"Normal evaporator water level" as used in this discussion is defined as "that water level which causes the bottom of the steam-water transition zone to be two inches above the top of the tubes at a distance of 34 inches from the helium-water tubesheet." This definition is required since there is no single, smooth interface between the liquid water and water vapor. The point 34 inches from the helium-water tubesheet corresponds to the location of the south probes,

*Carryover = the amount of liquid water entrained in steam leaving the separator. Maximum acceptable level = 0.25%.

near which very little steam is generated and where the transition zone has a depth of only two to three inches at full power.

During steam generation, water vapor (steam bubbles) is formed at the tube wall and rises upward through the water, due to the density differential. Some liquid water is carried along by bubbles breaking through the surface; this condition is called "priming". Some of the bubbles do not surface, but instead are carried within the water, traveling downward between the tube bundle and evaporator shell, then flowing upward again through the tube bundle. This condition, termed "vapor recirculation" is responsible for another condition called "false water level" (as indicated on the normal level gauges).

The vapor recirculation anomaly, coupled with the wave action which exists on the surface of the water in the absence of steaming, causes a "transition zone" between the pure liquid and pure vapor phases. The depth of this transition zone at a particular point longitudinally in the evaporator is primarily a function of the steam generation rate at that point.

(1) Indicated Evaporator Water Level vs Steam Flow Rate

"Normal" evaporator water level was established at several flow rates. All level indicators were read at each test point. Information gained from the gauge glass readings was nominal, as the readings fluctuated by three inches or more and often the level was between ports.

The south gauge glass consistently indicated higher water level than the north gauge glass, and the difference increased with increasing steam flow. The three density compensated level columns indicated that water level at the south end decreased as steam flow increased (Figure 8). This error is explained by the previously noted "false" water level. Two key points should be kept in mind:

- a) The false water level is a function of the steaming rate and becomes significant at this point in the evaporator after steam flow reaches 8.0×10^4 lb/hr.
- b) No appreciable difference exists between the three steam generators.

The center level column presents a different correlation (Figure 9) with the "false" water level appearing at 4.0×10^4 lb/hr. In the No. 1 and No. 2 evaporators the resultant water level is slightly greater than that

Figure 8. HNPF Steam Generator Water Level Indicators – South Level Column

found in the south level columns at the same flow rate. This would indicate that more steam is generated and more vapor recirculation exists near the center columns than near the south columns of evaporators No. 1 and No. 2. The error in the indicated water level column of evaporator No. 3, however, is greater than in No. 1 and No. 2. This discrepancy is attributed to the "T" bar splash baffles added in an effort to correct carryover in steam generator No. 3.

Figure 10 shows that the north level columns also indicate a false water level; the resultant error is greater than in the center level columns, and is greater in steam generator No. 3 than in the other two generators. As previously stated, this area of discrepancy between steam generator No. 3 and generators No. 1 and No. 2 is directly related to the splash baffles in the No. 3 unit. Since these baffles have cut the cross-sectional area for vertical steam

Figure 9. HNPF Steam Generator Water Level Indicators - Center Level Column

flow in the north half of evaporator No. 3 by ~50%, much of the steam will travel south, under the "T" bars, instead of upward through the bars. This condition causes more vapor recirculation, consequently causing more steam generation further south in the evaporator, all of which increases the magnitude of the indicated water level error.

(2) Evaporator Water Level Profile

Steam generator No. 1 was used to study the water level profile. The conductivity of water samples taken from the internal probes indicated which phase (vapor, transition zone, or liquid) was being sampled.

By changing evaporator water level it was possible to define the elevation of the top of the transition zone within one inch (Figure 11). The exact

Figure 10. HNPF Steam Generator Water Level Indicators – North Level Column

profile, however, can only be estimated, since only three positions (longitudinally) can be tested in this manner. The sharp bend in the Figure 11 plots would undoubtedly be a smooth curve if more positions could be tested. Figure 12 presents the same data, with the longitudinal probe location as the third parameter.

It appears that the top of the transition zone reaches a maximum elevation near 2.4×10^5 lb/hr steam flow. Although steam generation occurs further toward the cold (south) end of the evaporator as steam flow increases, estimates are that 90% of the steam is generated in the north half at 2.5×10^5 lb/hr steam flow (design flow).

Figure 11. Steam Generator No. 1 Water Level Profile

(3) Evaporator Water Dissolved Solids Concentrations

Dissolved solids concentration in the evaporator water was found to decrease following an increase in steam flow, much as in conventional, fossil-fired boilers, although the amount of dissolved solids in the evaporator remains constant, and liquid water volume decreases with increasing steam flow. A decrease in flow, conversely, was followed by an increase in solids concentration.

This phenomenon is attributed to the fact that a thin film of concentrated solution is formed at the phase boundary between the hot surface and the liquid.* The concentration is directly proportional to heat flux, and therefore to steam flow. Varying plant conditions made exact data taking impossible, since the concentration of dissolved solids in the evaporator water was being forced to seek a new equilibrium throughout the testing. Several hours (dependent on magnitude of most recent power change) of steady-state operation is required to reach an equilibrium value at which more precise data can be logged.

* Samuel, T., "The Influence of Dissolved Substances and the State of the Heated Surface on the Mechanism of Boiling," THE BROWN BOVERI REVIEW (Switzerland), Vol. 49, Mo. 11/12: 519-531 (1962).

Figure 12. Steam Generator No. 1 Water Level Profile

(4) Moisture Carryover Determinations

Subsequent to installation of the steam generator level probes, it became possible to accurately define the water level with respect to the top of the tubes, without having to rely on the calibration of the level indicators.

A comparison of the performance curves for steam generators No. 1 and No. 3 (Figures 13 and 14) shows less carryover in generator No. 3 at steam flows below 2.0×10^5 lb/hr. Above that level, however, the No. 3 unit has greater carryover, indicating that the "T" bar splash baffles compound carryover conditions at high steaming rates. Figure 15 depicts gradual increase in carryover with increasing steam flow for steam generator No. 1, while generator No. 3 increases disproportionately with flow.

Figure 13. Steam Generator No. 1 Performance

Figure 14. Steam Generator No. 3 Performance

(5) Calculation of Steam Velocities

Horizontal steam velocity past the north baffle box is shown as a function of water level in Figure 16. Steam flowing past this point was assumed to be saturated at 900 psig and flowing at 2.0×10^5 lb/hr. From a comparison of Figures 11, 13, 14, and 16, it is concluded that moisture carryover becomes excessive when the horizontal velocity past the north baffle box reaches 15 – 20 ft/sec.

(6) Level Profile Near the Tubesheet

One of the probes in evaporator No. 2 was installed ~14" nearer the helium-water tubesheet (further south), to determine if the water surface "tilts" downward toward the tubesheet. Test data indicate the bottom of the transition zone remains at essentially a constant elevation from the helium-water tubesheet to a point about 35" north of the tubesheet. From the data,

Figure 15. Steam Generator Performance With Normal Water Level Conditions

logged at high and low steam flow rates, it can be inferred that there is no possibility that a portion of the top row of tubes near the tubesheet might be uncovered even though the south probes are covered.

(7) Steam Pressure Effects

Moisture carryover tests were made on steam generator No. 1 with the evaporator pressure at 920 psig and 885 psig. There was no measurable difference in the results.

b. Conclusions and Recommendations

As a result of testing conducted to date, certain basic conclusions were reached. First, it was determined that corrections must be made for the false water level indications resulting from vapor recirculation, in order to maintain desired evaporator water level. Being a function of steam generating rate, this condition becomes more serious at higher power levels. Programmed evaporator water level equipment has been installed to maintain the desired level.

Figure 16. HNPf Steam Generator Velocities

The "T" bar splash baffles in steam generator No. 3 aggravate the false water level and cause increased carryover above 2.0×10^5 lb/hr steam flow. These baffles are detrimental to steam generator performance and will be removed.

It appears that the basic cause of moisture carryover (under normal water level conditions) is insufficient freeboard. Since the primary separation of vapor and liquid must occur inside the evaporator (by gravity), the steam velocities must be maintained low enough to allow this separation to take place. Further testing will be conducted at 100% steam flow to assist in developing the most expedient corrective action for the carryover problem which occurs at high steaming rates.

8. PREHEAT, FILL AND TESTING

a. Preheat

During preoperational preheat of the sodium systems, the preheating system was determined adequate for the intended use. Normal "first operation" difficulties such as faulty relays were corrected without need for modifications or severe time losses.

Heating rates exceeded recommended values, twice during the preheat testing: (a) IHX-3B and (b) Steam Generator No. 1. Stress analyses showed the transients to be insufficient to cause damage to the components. The only system change suggested as a result of this test is the replacement of the oven-type thermocouples with shell-type thermocouples. The oven-type preheat thermocouples, each consisting of a 3/8"-diameter sheathed probe, penetrate into the heater ovens to within $\sim 3/8"$ of the component or piping. Temperatures sensed by these thermocouples generally ran below the component and piping temperature; as much as 50 – 100F° in some cases. These discrepancies were caused by items such as insufficient insulation around the thermocouple probe insertion, heat shorts from support structures and seals, and proximity to the heaters. Many thermocouples had to be relocated closer to the component shell, while others required more insulation around the probes to reduce heat loss through the sheath.

b. Initial Fill

Initial fill of the primary (AI-P-1157) and secondary (AI-P-1156) systems with sodium was easily accomplished. Certain refinements were introduced to the standard operating procedures developed for the fill operations, but the basic procedures were adequate.

The reactor was filled to elevation 1428'6", to ensure adequate sodium in the pump cases for lubrication of the fluid bearing during initial operation. A total of 7663 ft³ of sodium @338°F was introduced to the system from the primary fill tanks during initial fill. Earlier the sodium had been pumped into the fill tanks from tank cars brought into the building on a special spur line, and had been cold-trapped to obtain a plugging temperature below 260°F. An approximate total of 3900 ft³ of sodium @300°F was transferred into the secondary loops from the secondary fill tanks. The loops were filled to elevation 1468'-9".

During the filling operations, care was taken to purge the system with helium until the oxygen content had been lowered to not more than 0.5%. This number is perhaps unnecessarily low, in light of the rate with which sodium is added to the system. The flow will have a tendency to force the air up and out of the high points before much oxidation can occur. Since $\sim 4.0\%$ oxygen content is required to sustain burning, time and expensive volumes of helium could be saved by setting the maximum permissible oxygen content at 2 – 3%.

c. Hot Sodium Circulation

Prior to initiation of fuel loading for wet criticality, the heat transfer systems were filled with sodium, in order to verify the adequacy of the plant components and systems. Piping displacement, system hydraulics, heat transfer system to plant control system interrelationship, temperature effect on control rod operation, core alignment, and component stability were all closely scrutinized during the hot circulation testing.

The systems were operated at temperatures ranging from 340°F to 540°F. Testing was performed with temperatures rising, then again during the cool-down. Figures depicting system hydraulic characteristics for preoperational and post critical test periods are found in AI-P-1167 and AI-PC-510. Piping interferences, inoperable hangers and valve difficulties were encountered, as well as deviations between predicted and measured displacements. The valve, hanger and interference difficulties were corrected, and the test was extended to cover the displacement discrepancies.

A remote pipe displacement transmitter-recorder system^{*} was installed during the time the system was being maintained at 540°F, for accurate displacement measurements in those areas which are inaccessible during reactor critical operation. Six locations were instrumented for remote determination of displacement. East-west and north-south displacement was checked by instruments located at each of the primary pumps. Other instrumentation was installed at each of the primary reactor outlet lines on the pump side of the outer bellows on the guard pipe, to determine east-west and vertical displacement. This system proved effective in verifying stress calculations during high temperature operation after initial wet criticality. The piping displacement was

*P. F. Shaw and J. D. Berg, "Remote Pipe Displacement Transmitter-Recorder System Hallam Nuclear Power Facility," (NAA-SR-8258) April 1963.

monitored periodically until January 1964, by which time the reactor had logged several months of high temperature, high flow operation. Throughout this time the maximum noted displacements were within allowable limits. In the post critical testing master schedule completion report, AI-PC-830 (Atoms International, November 1963), pipe displacement vs sodium temperature are plotted (Figure 830-2).

Post critical test AI-PC-510, Hydraulic Test of Main Heat Transfer System, was a logical follow-up to the hot sodium circulation test. System operating parameters were compared with design or anticipated values to check for satisfactory operation, and to establish base data for future comparison. Testing was initiated under isothermal conditions, and completed during power operation, with the plant functioning as a fully automatic load-follower.

System parameters reported in AI-PC-510 are:

- 1) Sodium pressures
- 2) Pump speed
- 3) Pump motor current
- 4) Coupling current
- 5) Sodium flow rates
- 6) Moderator valve position
- 7) Sodium temperatures.

Pump sodium level problems are discussed elsewhere in this evaluation (Section II. A. 2). Subsequent to sealing of four weep holes in each primary pump, two-loop operation was satisfactorily accomplished. Moderator coolant flow ratio was essentially the same with any two loops operating, or with all three loops operating at any flow rate.

System hydraulic characteristics at isothermal conditions were in excellent agreement with predicted values from AI-P-1167, based on the original recommended flow meter calibrations. The head curves determined during power operation were also near those predicted, based on the 1167 data. However, new calibration curves developed during performance of AI-PC-530, Heat Transfer Tests, diverge from the earlier predictions at higher flow rates. Actual flows were $\sim 23\%$ greater than expected for the primary system, and 17.7%

greater for the secondary system. System pressure drops required at a given flow rate are from 1/2 to 2/3 of those predicted; pump capacity is hence adequate to meet system requirements. Various reasons for the discrepancy are discussed in the evaluation section of AI-PC-510.

Pressure instrumentation performance during this test was less than desirable at low (below 40%) flow. Low differential pressures and inherent process pressure instrumentation inaccuracies caused data scatter at the lower flow rates. Zero point drift contributed to the inaccuracies noted in the pressure instrumentation.

System and pump efficiencies listed in the test report show improvement at higher power, but the data must be tempered with acceptance of decreasing instrument accuracy at the lower end of the operating range.

Full power temperature differences between core components are apparently related to moderator coolant flow. Test operations indicate that a moderator coolant valve position of 85% closed yields the smallest temperature gradient across the core. Moderator temperatures, however, are a function of reactor power, and are not appreciably affected by valve position.

Data plotted on July 19, 1963, with ~1.4% moderator coolant flow showed the moderator elements to be slightly overcooled (core centerline data).

Regarding system hydraulic characteristics, the systems performed satisfactorily over the range from zero to full power. Deviations noted between expected and measured results were not of sufficient magnitude to place the deviating parameter outside the design capabilities of the plant.

Results recorded during scram and loss-of-load tests also showed excellent performance in substantiation of plant design. Only three minor areas of concern were encountered during the scram tests:

- 1) Rapid decay rate of reactor outlet temperature after low power scrams.
- 2) Short duration temperature undershoot caused by slow initial closing of the throttle valves immediately after scram.
- 3) Damage to the throttle valve topworks during rapid post-scram operation.

The rapid cooling is ascribed to the fact that very little afterglow heat is generated at low power operating conditions. Restroking of the throttle valves to close to the "99% closed" position immediately after scram proved a satisfactory solution to the problem. It was also determined that the throttle valves can be manually controlled to maintain acceptable temperature transients after scram with no appreciable afterglow heat.

It is anticipated that the temperature undershoot and valve topworks damage problems have been rectified by redesign of the topworks (see "Throttle Valves", Section II. A. 4 of this evaluation), and by setting the new mechanisms to close in 2/3 of the time previously required (10 vs 15 sec).

A loss-of-load (turbine) test was performed at 25 Mwe to check system reliability and performance under conditions requiring dissipation of sudden excess energy in a controlled manner. Plant performance was as expected, with only minimal operator action required. Several months later, while the nuclear and conventional plants were operating in parallel at low power, the plant control and heat transfer systems underwent an inadvertent test. With the two plants jointly responding to grid demand, the conventional boiler tripped off the line; the reactor and associated systems responded immediately, successfully withstanding a short duration 20 Mwe/min demand change while picking up the plant load.

The heat transfer coefficients for the steam generators and the intermediate heat exchangers were calculated during the time of the heat transfer tests. In each case, low power results deviated greatly from predicted, due to normal instrument inaccuracies at lower powers, and to the deviation in log-mean temperature difference resulting from the regenerative effect of sodium-to-sodium heat transfer in the steam generator bayonet tubes. Also, at lower loads most of the heat transfer is accomplished in a short portion of the tube, rather than along the entire length. The log-mean difference is an integrated value taken along the length of the tube, and was necessarily low at low power because of the unknown variable involved (active tube length at a given power). At design conditions, with heat balances completed, the measured coefficients are either within the anticipated range, or show a higher than predicted value (superheaters). Pressure drops in the evaporators were as expected; slightly higher than expected pressure drops in the superheaters are attributed to pipe friction and are not serious. It was determined in the performance of AI-PC-530,

Heat Transfer Tests, that the Hallam steam generating equipment is capable of producing sufficient quantities of steam under adequate pressure and temperature conditions to generate the specified net electrical plant output.

Calculation of overall heat transfer coefficients for the IHX's yielded results within $\pm 12\%$ of predicted. Errors in temperature measurements, and unequal flow distribution between the two parallel units comprising each IHX are the probable causes of the minor deviations. Heat exchanger pressure drops in excess of predicted values are attributed to pipe friction; as in the case of the superheaters, the differences are too minor to be of concern.

Heat losses from the heat transfer system to the cooling water system, and from the steam generator amounted to ~ 1.5 Mwt at full power (Generator Gross = 76.16) during the heat transfer tests. Average loop steam flow at full power was $\sim 238,000$ lb/hr with a net heat rate of $\sim 11,000$ Btu/kw.

B. REACTOR CRITICAL OPERATION

Initial reactor criticality (dry) was achieved in January 1962 with a 31 element fuel loading. Dry excess loading brought the core to 42 elements, yielding excess reactivity of $\sim 35\%$ of the reactivity worth of the control rods. The isothermal temperature coefficient of the dry core was positive from 71°F to 369°F , as anticipated. Control rod reactivity worth was found to change only slightly with temperature. E.g., at 365°F a segment of gang 5 had a reactivity worth of 20.1 cents ($\Delta\rho$) as compared with 23 cents at 71°F . Another segment had a reactivity worth of 88.6 cents at 365°F and 90 cents at 71°F .

Plans to load the reactor to the approximate wet critical loading were abandoned, to comply with a test procedure restriction prohibiting a loading which would allow achieving criticality with $< 50\%$ withdrawal of the control rods. Experimental determination of the sodium worth at any fuel loading was hence not performed. Extrapolation of the experimental dry excess loading data yields $K_{\text{eff}} = 1.064$ @ 70°F with 52.4 fuel elements (wet critical loading). Corrected to 350°F (using fuel and moderator temperature coefficients of reactivity presented in NAA-SR-5700*, Suppl 4), K_{eff} becomes 1.065, or $6.1\% \Delta k/k$. The calculated value of sodium worth at 350°F in the 53 element core is $5.5\% \Delta k/k$, showing excellent agreement between experimental and calculated sodium worth.

*"Final Summary Safeguards Report for the Hallam Nuclear Power Facility," (NAA-SR-5700) Atomics International, April 1961.

During loading for wet criticality 16 plots of fuel loading status and nearness to criticality were maintained. After each loading increment, the incore nuclear instrumentation count rates were recorded at steady-state conditions with the rods in four configurations:

- 1) All rods inserted
- 2) Gangs 1 and 2 fully withdrawn, with gangs 3, 4, and 5 fully inserted
- 3) Gangs 1, 2, and 3 fully withdrawn, with gangs 4 and 5 withdrawn 50%
- 4) All rods fully withdrawn (156").

In addition, the inverse average count rate $1/N$ was plotted vs:

- 1) The square root of the total number of fuel elements loaded
- 2) The log of the total number of elements loaded
- 3) A number proportional to the radial buckling of the partially loaded core configuration.

In future fuel loadings of the HNPF and similar reactors, valuable operating time can be saved by utilizing only the "all rods withdrawn" inverse multiplication curve and the "all rods inserted" curve. Joint use of these curves will assure adequate shutdown margin at all times during the loading, and provide a basis for choosing the number of fuel elements to be loaded per increment. Core loading can thus proceed to the operational configuration more rapidly, with no decrease in safety. Overall accuracy in obtaining expected results in the initial dry and wet critical experiments lends great confidence to the method and assumption adopted for calculating this core.

Wet excess loading, as with the previous fuel loadings, was conducted with due regard for safety, but with greater confidence in achieving the expected results. The operational loading was accomplished by loading fuel elements into the wet core in increments of $<1\% \Delta k/k$, taking care to maintain a calculated shutdown margin of sodium worth plus $2\% \Delta k/k$. Half of the 2% figure was for safety, and half to allow for the increase in reactivity expected as a result of the 500°F increase in core temperature during power operation.

Five values were obtained in the determination of excess reactivity; as a result of an initial discrepancy between measured and calculated excess, three additional methods were employed to calculate the excess. At the operational

loading (140 elements), the value of the measured K_{eff} was 1.109 (9.8% $\Delta k/k$) as compared with an "analytical" value of 1.069 (6.5% $\Delta k/k$). Two possible causes of the discrepancy were:

- 1) The change in reactor "size" with fuel loading
- 2) Distortion of flux due to control rod insertion.

Three methods were used to obtain a "true" estimate of the excess reactivity actually loaded. The methods and calculated estimates are:

- 1) Perturbation correction of measured excess
($K_{\text{eff}} = 1.074\rho = 6.9\% \Delta k/k$).
- 2) Two-region calculation of excess based on measured banked rod position ($K_{\text{eff}} = 1.072\rho = 6.0\% \Delta k/k$).
- 3) Extrapolation of excess loaded in the first two increments
($K_{\text{eff}} = 1.063\rho = 5.9\% \Delta k/k$).

The 6.5% $\Delta k/k$ value calculated prior to loading lies mid-range in the 5.9 to 6.9% $\Delta k/k$ spread developed by the three methods described above, indicating this number to be more accurate than that value obtained by summing incremental excesses.

The discrepancies, postulations and calculations are discussed at length in the evaluation of AI-PC-710, "Hallam Nuclear Power Facility Preoperational Test Completion Report Wet Excess Fuel Loading".

1. REACTIVITY COEFFICIENTS

The isothermal temperature coefficient of reactivity was determined in the range of 370°F to 540°F. This coefficient is the sum of one negative (fuel) and two positive (moderator and coolant) partial coefficients having a combined positive value. The value at 400°F is +0.123¢/F°, and 0.088¢/F° at 500°F. As listed in the power coefficient test report, AI-PC-770, the "prompt" power coefficient is negative, e.g., -0.4¢/Mwt at 2.5 Mwt with 20% primary sodium flow, and -0.05¢/Mwt at 240 Mwt with 96% primary sodium flow. In the evaluation of the test it is reported that correcting the data, to account for control rod worth increases corresponding to power and temperature rises, yields a more negative coefficient.

Values for the "slow" power coefficient are positive with a slight downward trend at higher power ($+0.4\text{¢}/\text{Mwt}$ to $+0.26\text{¢}/\text{Mwt}$ from 2.5 to 240 Mwt). The steady-state power coefficient at constant flow is positive ($+0.05\text{¢}/\text{Mwt}$ measured at 15% flow and 38 Mwt power, and $+0.21\text{¢}/\text{Mwt}$ calculated at 96% flow and 240 Mwt power). A negative flow coefficient was determined ($+0.16$), with the combined (power and flow) steady-state coefficient having a small positive-negative range about zero.

A comprehensive review of the zero power coefficients of reactivity and power coefficient testing performed at Hallam is included in the related test reports, AI-PC-760 and AI-PC-770, respectively. The results obtained are in reasonable agreement with the results expected.

Power operation was suspended from August 3 to October 15, 1963, to allow replacement of the Zircaloy-2 control rod thimbles. During that time 10 uranium carbide (UC) fuel elements were added to the core, bringing the total core loading to 150 fuel elements (140 U-Moly and 10 UC). Addition of the UC elements and replacement of the Zircaloy-2 control rod thimbles* with stainless steel thimbles affected core excess reactivity (23¢ decrease) and control rod worth (decrease by factor of 0.9).

2. XENON AND SAMARIUM

Data were gathered for two important fission products during the post critical test program. Build-up, equilibrium, peak and decay were plotted for Xenon¹³⁵ and calculated for Samarium¹⁴⁹ (based on the xenon data).

Xenon, which increases with flux, was plotted at several power levels, to determine build-up time and reactivity worth. Peak xenon after a scram from full power is $\sim 263\text{¢}$ ($\sim 4\text{-}1/2$ hr after scram), or an increase of 24¢ (10% rise) over full power equilibrium. The increase following a low power (15%) scram is almost negligible ($\sim 1\text{¢}$ after 1-1/2 hr).

Samarium, the other poison plotted for build-up and peak, differs from xenon in two key characteristics: (1) the equilibrium value is independent of flux, and (2) there is no decay after shutdown. This poison will build up to a post-shutdown peak, and remain at that level until the next startup, at which

* Discussed under "Control Rod Thimbles" elsewhere in this report (See Table of Contents).

time the level decreases to approximately the previous equilibrium level. Build-up time as depicted in Figure 840-7 (AI-PC-840) is 160 full power days. Equilibrium reactivity worth is $\sim 122\text{¢}$, increasing to $\sim 139\text{¢}$ (14% rise) by 10 hr after scram.

There is excellent agreement between measured and calculated reactivity values, especially in view of the innumerable partial temperature coefficients encountered with the system in a non-isothermal transient temperature condition. The accuracy demonstrated in predicting these values lends confidence to the reactivity history calculations to be performed in checking core normalcy with these numbers as a base.

3. POWER DISTRIBUTION

As pointed out in the evaluation of the axial power distribution test, AI-PC-840, quantitative interpretation of the results is complicated by two "outside" disturbances:

- a) The test was carried out in a perturbed core as a result of having the fission chamber thimbles located in core positions normally containing fuel.
- b) The dimensional resolution of the chambers was too poor to show local flux peaking effects, which might be expected near the control rod tips.

Only a limited number of control rod patterns were investigated during the traverses associated with axial power determination. Care was taken, however, to perform those traverses which would yield the most valuable data for the intended mode of operation. A total of 24 traverses was accomplished, utilizing five fuel channels. The results, as reported in Table II of AI-PC-840, range from a peak-to-average ratio of 1.400 (C-71) to 2.625 (C-46).

Whereas, the lowest axial peak-to-average ratio is the most desirable in terms of fuel life, rod motion effect on radial power distribution becomes a limiting factor in a load-following plant. Load following, which may require frequent large percentage power changes, is best accomplished with short movement of many rods, thus producing no significant effect on the radial power distribution. In such a configuration there may be only minimum control over the axial peak-to-average power. Therefore, several ratios were determined

to establish the range of vertical control rod movement which permits the minimum peak-to-average axial power of distribution.

The poorest axial peak-to-average ratio reported (2.625) was obtained with the "all rods banked" configuration. This configuration causes flux depression in the upper reactor area, with a resultant axial flux peak in the lower portion of the core, yielding the poorer values. For the cases examined, gang 1 at five different axial positions produced peak-to-average axial power ratios ranging from 1.400 to 1.684. The upper end is slightly above the 1.65 called out by the Final Summary Safeguards Report, NAA-SR-5700.

Checks performed to verify the acceptability of controlling with the central control rods (if necessary) were satisfactory. A comparison of calculated and measured axial thermal neutron distribution, both above the active core and in the sodium, yielded favorable results.

For the Hallam reactor, two conclusions were drawn as a result of the axial power checks.

- a) Any operation of all the rods as a bank should be discouraged.
- b) The use of adjacent gangs in a banked configuration at the same axial position should be discouraged for positions between 39.0" and 117.0". If operation in these configurations is necessary, the axial temperature distribution should be observed closely.

With identical flow and/or fuel channel exit temperature conditions established through each fuel channel, the radial peak-to-average ratios were determined by dividing the peak core ΔT by the average core ΔT . Results of the testing from 0 – 80 Mwe indicate that the radial power peak-to-average ratios are maintainable within the 1.5 limit called out in NAA-SR-5700. The calculated peak fuel temperature (1238°F w/error allowance) is below the 1250°F fuel design limitation.

A later change in control rod configuration, to allow plant operation at 15 – 100% power with essentially one control rod configuration for at least three months, produced a shift in radial power distribution. The change caused a flatter distribution in the core center, resulting in a higher radial peak-to-average ratio (1.46). This new ratio, however, was below the 1.5 limit. Operation over the range from 12 – 80 Mwe produced ratios of 1.45 ± 0.03 , gradually

reducing with banked gangs (1 and 4) withdrawn to compensate for burnup. Post test data indicate radial power shifts relative to daily load changes are quite small.

In obtaining the desired flow and outlet temperature conditions for radial power determinations, the fuel channel orifices underwent several adjustments. An apparent marked change in sensitivity was noted in the 2-1/2" to 3" range (full range = 0" to 4"). Exit temperature discrepancies uncovered during these adjustments is attributable to temperature stratification and is relative to the orifice plug position in the channel. At the fully closed position, the thermocouples "see" only the central (hottest) portion of the flow (Figure 800-2). As the plug is moved upward in opening, the thermocouples are moved into a locality where the "colder" sodium from the peripheral region on the channel is starting to mix with sodium from the center of the cluster.

Channeling of fuel cluster center sodium to the exit thermocouples has likewise been given as the reason for non-uniform temperatures from top-to-bottom and side-to-side in the upper plenum. Because of temperature distribution changes in the reactor upper plenum, reactor outlet temperature varies as a function of power. These variations in reactor outlet temperature occur even though the associated temperature control attempts to maintain a constant plenum temperature over the full power range.

To improve the accuracy of reactor outlet temperature control, thermocouple leads from the process thermocouples in the sodium piping have been brought to the patch panel for use as inputs for the reactor outlet temperature controller.

C. REACTOR AND HEAT TRANSFER SYSTEM GENERAL

Overall reactor and heat transfer system performance is equal to or better than design ratings. High temperature, high flow, low pressure operation has been effected for prolonged periods with satisfactory results. Operation of the plant as a fully automatic load-following station has proven successful on every occasion.

Some areas of concern, however, have attended reactor operation. Those items range from an improperly installed moderator valve to leaking control rod thimbles. The reversed moderator valve, as with other construction errors

such as reversed polarity on the sodium pumps, is a relatively minor nuisance. Difficulties of this nature might well be eliminated by better defined specifications and closer inspection during the construction phase.

1. CONTROL ROD THIMBLES

Failure of the Zircaloy-2 control rod thimbles was of prime concern. In all, three thimbles failed before sufficient data were gathered to indicate the necessity of replacement with stainless steel thimbles.

A leak was detected in Zircaloy thimble TZ-12, after 297 hr of high temperature (945°F) operation. Once the leak had been verified, the thimble was removed for inspection in the maintenance cell. A series of bulges and cracks (Figure 17) was noted in an area immediately below the transition zone. The incore elevation of this area corresponds to the top of the moderator elements. Following site examination, the failed section was cut out of the thimble and shipped to the Component Development Hot Cell (CDHC) at Santa Susana, California.

Figure 17. Zircaloy-2 Control Rod Thimble TZ-12 Damage

About one month later, the other two failures (TZ-3 and TZ-6) were noted. Maintenance cell inspection revealed bulges above and below the transition weld on each thimble. In addition, a 1/2" dia. hole (Figure 18) was found above the transition joint of TZ-3. Thimble TZ-6 bore a raised, porous appearing area, 1" dia., in a similar location. Sections were removed from above and below the transition joint on TZ-6 and shipped to the CDHC for failure analysis.*

a. General Operating Conditions

The thermal history for TZ-12 in the area of failure (reactor upper plenum) is:

<u>Temperature Range (°F)</u>	<u>Exposure Period (hr)</u>
550 - 620	486
620 - 720	113
720 - 820	93
820 - 900	95
900 - 945	297

Thimbles TZ-3 and TZ-6 were additionally exposed for ~550 hr at 930°F to 950°F, during which time the thimbles were flooded with reactor grade helium. Prior to reactor startup, the inside surfaces of the Zircaloy-2 thimbles had been exposed to open air for 30 - 40 days, while at temperatures ranging to 500 - 600°F. During operation, in-thimble pressure is 11 - 15 psig; it is estimated that a pressure differential of 10 psi existed at the points of failure.

The three failed thimbles were all located in channels at a distance of ~27.5" from core center (Figure 19). Locations and operating positions were:

Figure 18. Zircaloy-2 Control Rod Thimble TZ-3 Damage

*R. K. Wagner and S. L. Englebert, "Results of HNPF Control Rod Thimble Examination," NAA-SR-TDR-9142, October 1963.

<u>Thimble No.</u>	<u>Location</u>	<u>Operating Position</u>
TZ-3	C-28	Full In
TZ-6	C-16	On Control
TZ-12	C-32	On Control (50" - 80" Out)

With TZ-3 "full in", the control rod section opposite the TZ-3 transition joint contained the titanium hydride shielding. For TZ-6 and TZ-12, the poison columns (gadolinium and samarium oxides) would have been opposite the transition joints during operation. Two of the three major failures noted occurred above the transition joint, and were, hence, in an area where the thimble surface is shielded from the poison column by means of a boron carbide shroud.

Figure 19. Core Map Showing Location of Failed Thimbles

Many of the contaminant bearing materials in the poison columns were clad in individual containers to partially compensate for the high "gettering" characteristics of the thimble material. Since a degree of decomposition was anticipated, provisions were incorporated into the design of the containers, to vent gaseous decomposition products into the "protective" thimble environment.

Four major sources of contamination, principally oxygen and hydrogen, on the poison column include:

- 1) The powdered poison materials — oxides of gadolinium and samarium.
- 2) A neutron streaming moderator material — titanium hydride.
- 3) A neutron streaming capture material — boron carbide.
- 4) Impurities contained in each of the above.

Exact composition of the above items could not be established, since, in many cases, certification of material composition was not required for acceptance. None of the materials were available for analysis to establish contamination levels.

Additional sources of contamination which could have been unintentionally introduced include:

- 1) Petroleum base products — oil and grease.
- 2) A polystyrene material.
- 3) The atmosphere (open air) contained in the thimbles during preliminary elevated temperature operation prior to reactor startup.
- 4) The impurities contained in the protective operating environment — reactor grade helium.

Post-failure examination of the hydraulic shock absorber (snubber) assembly indicated that oil could have been released during operation.

Grease was found on the inside surfaces of the TZ-6 and TZ-12 sections sent to the CDHC. The grease location corresponded to the mechanical joint elevation at the top of the thimbles.

b. Evaluation of Original Thimble Material

Trimmed ends remaining from original fabrication of some of the thimbles were examined. Testing included tensile, hardness, bend, metallographic and chemical analyses. The results show nothing unusual regarding the strength, ductility or composition of the material. Likewise, the material proved to be of good quality.

A total of 80 specimens, longitudinal and transverse, of the tubing were formed through 180° around T mandrels. Only one showed evidence of cracking.

Twenty-five metallographic specimens were taken from the Zircaloy-2 tubes and transition joint. Typical Zircaloy-2 microstructures were found in each. Except for a crack in one weld, no significant defects were revealed. The crack had been found during radiographic examination prior to trimming.

A section of 3.74" dia. tube was used as a control in emission spectrograph checks of the material. The sample and the Zircaloy-2 standard have the same spectrum.

In checking for X-ray fluorescence, no significant difference was found between the sample and Zircaloy-2 standard.

c. Examination and Analyses

All of the sections of failed thimble sent to the CDHC were cut off at the site by use of the fuel handling machine and a pipe cutter. The material in the failed area was brittle in nature. An area of the large diameter tube, 15" above the TZ-6 transition joint, shattered during an attempted cut. Greater ductility was noted in the material at a distance of 34" - 38" below the transition joint.

Several of the thimbles were inspected in the HNPF Maintenance Cell. The results ranged from no evidence of failure (TZ-13) to a 6-1/2"-long x 0.050" wide crack below the TZ-12 transition joint.

Visual examination of various poison column containers (poison cans, titanium hydride cans, and boron carbide cans) yielded no evidence of surface defects, but many localized areas of discoloration were noted. These areas appeared to be covered principally by black, carbon-like deposits, the location

of which generally corresponded to an operating position opposite the failure zones in the thimbles. No surface indications were found during dye penetrant inspection of a titanium hydride can.

A black amorphous appearing mass was found in the bottom portion of TZ-12. Gamma spectrum analysis indicated the presence of Cr^{51} , Zr^{95} , Nb^{95} , Co^{60} , Mo^{99} and other unidentified sources of 0.004 mev and 0.1 to 0.102 mev. A quantitative analysis for these elements could not be performed.

Discoloration of the TZ-12 outer surfaces was noted during inspection at the CDHC. The degree of discoloration was similar to that usually found on Zircaloy-2 exposed in sodium at 800°F – 1000°F. The longitudinal weld was identified and found to be located 180° from the failure. Material in the failed area had broken away from the edges of the crack, leaving larger gaps than had been observed at Hallam.

Measurements were made to establish the approximate bulge configurations on TZ-12. Next, the section was split into half-shells to allow inspection of the inside surfaces.

The surface in the failed area showed extensive interaction, particularly in that portion directly adjacent to the crack. In addition, a 1/2"-wide burnish extended through the failed area. Outside the failed area the surface appeared to be uniformly oxidized. Chemical analysis and metallographic examination of the material showed severe hydriding in the failed area. Hydrogen contents to 1.7 wt % were measured, with hydrogen levels decreasing outside the failed area to 150 – 300 ppm.

X-ray fluorescence was used to confirm the Zircaloy-2 composition. Metallographic sections showed that zirconium hydride had formed on the inner surfaces and was progressing outward. The hydride phase is readily apparent in Figure 20. A residual deposit in the failed area was found to contain 3 wt % oxygen, 2 – 4 wt % carbon, 1 wt % hydrogen, and 0.3 wt % nitrogen.

Pickling of several samples prior to analysis had little effect on the hydrogen, carbon and nitrogen levels, but the oxygen was decreased to the levels usually found in Zircaloy-2 (800 – 1400 ppm).

Hydrogen analysis of material from the burnished area showed that area was not particularly susceptible to hydriding. Similar burnishing was noted on TZ-6.

Figure 20. Zircaloy-2 Control Rod Thimble TZ-6 Damage

Bulges, large amounts of contaminants (indicative of highly localized surface reaction products), and brittle fractures were encountered with TZ-6, as in the TZ-12 sections. Evidence of sodium was found on the inside surface of one TZ-6 section. A series of hydrogen, carbon, oxygen and nitrogen determinations performed on TZ-6 and on the collected residue yielded results similar to those obtained on sections from TZ-12. Areas with significant hydrogen gradients were found; these corresponded with the failed or bulged zones. In each case where a bulge appeared in the tube wall, the corresponding inside surface and microstructure showed severe hydriding. However, all of the reaction spots on the inside surface did not produce bulges.

The results developed in the examination and analyses of TZ-6 closely match the results of the TZ-12 checks. The inside surface appearance indicates that chemical reactions have occurred in localized areas. The failures originate on the inside surfaces. A residue high in carbon, hydrogen and oxygen blankets

the area containing the failures. Specific reaction zones or failed areas (bulges) show high hydrogen concentrations. Microstructures in the areas of high hydrogen content show that the hydride is not uniform through the tube wall.

Irregularities noted on the cleanliness of inside surfaces of TZ-6 and TZ-12 are discussed in detail in NAA-SR-TDR-9142 (R. K. Wagner and S. L. Englebert, Atomics International, October 21, 1963), as are all details of the examinations and analyses of the failed thimbles.

It appears that the hydrogen distribution in the upper thimble sections is closely associated with the temperature profile. The most significant increase occurs at a position corresponding to the sodium level.

From the relative densities of Zircaloy-2 and Zirconium Hydride, the volume change which occurs in Zircaloy-2 during hydriding was calculated to be ~15%. With a volume change of this magnitude, bulges of the type encountered could be expected.

The coefficient of expansion for Zirconium Hydride is 1-1/2 times greater than that of Zircaloy-2, indicating that significant stresses would be imposed on Zirconium Hydride during reactor operation. These stresses would be magnified by temperature changes which occur during reactor shutdown. The initial leak in TZ-12 was detected after a shutdown for pre-full power control rod inspection.

Maximum hydrogen solubility in zirconium is 700 ppm at 1020°F. For higher levels the hydrogen exists as the compound (Zirconium Hydride) at temperatures to 1020°F. Above that temperature, hydrogen is dissolved in the zirconium, producing a phase transformation. A slightly lower transformation temperature value, as a result of the combined effect of the alloying elements in Zircaloy-2, brings this temperature within the HNPF operating range. The occurrence of such transformations, and the attendant changes in physical mechanical properties, would contribute to the rapid diffusion of hydrogen and to the resultant thimble deformation.

It is concluded that the failure-causing reactions which occurred in the HNPF control rod thimbles were probably the result of highly localized reactants which allowed direct diffusion of hydrogen into the Zircaloy-2. The deformation was not the result of localized hot spots and internal pressures;

rather, the bulging was the direct result of a rapid diffusion reaction where significant volume changes produce high, localized pressures.

As an aftermath of the hydriding encountered with the Zircaloy-2 thimbles, a system was installed for monitoring thimble atmosphere for the presence of hydrogen, and removing hydrogen if detected. All of the Zircaloy-2 thimbles were replaced with Type 304 stainless steel thimbles. The snubbers have been modified to eliminate one possible source of hydrogen (hydraulic oil) into the thimbles. Rod drop times were remeasured after the thimble exchange, and operation since that time has been satisfactory.

2. CONTROL RODS

During preoperational testing, performance of the control rod drive and actuator assemblies, and control rod operability were checked (AI-P-1145 and AI-P-1148). Rod speeds were determined to be within the limitations imposed by the Technical Specifications for the HNPF dry, zero power experiments. Operability of all related mechanisms was ascertained at the time of the early tests.

The report on the rise-to-power, AI-PC-830, compares the early results with those obtained during post-critical testing. The measured parameters included motor currents and voltages, withdrawal and insertion drive times, magnet latch release times, and partial and full drop times.

Generally, the motor currents were smaller than in the preoperational tests. There is, however, no indicated trend toward drive malfunction, and no effect on measured drive times.

Only minor difficulties were encountered in checking of the indicators associated with control rod operation. The difficulties were of the "normal" operating type, and were corrected by replacement of the faulty components.

Magnet latch release times were consistently longer than those reported during preoperational testing. Rod drop times to full insertion were, however, faster. The discrepancies are not of concern, since such variables as the precision in reading of the recording chart and the precision of observation must be allowed for.

Further drop tests have been performed as routine checks, to verify operation within technical specifications after replacement of the thimble, and

to ensure proper operation after detection of the moderator can flooding. Throughout these checks, magnet latch release, nine-foot free fall and full drop (to snubber closure) times remained within the specified limits. Variances in drop time for a given rod ranged both positive and negative. Control Rod No. 12 showed a 31 msec decrease in drop time during tests conducted in September -- October 1963. At that same time, CR-13 had a 24 msec increase in drop time.

As mentioned in the "Master Schedule" report (AI-PC-830), experimental error associated with the precision of reading of the Sanborn chart, used for recording "nine-ft-in" and "full-in" times, can cause a substantial deviation in reported results. The reported deviations over five drop tests from November 1962 to February 1964 fail to indicate any trend toward excessive increase or decrease of rod drop times.

Control rod measurements performed after 38 Mwt testing, and after ~50 drops (expected normal = 3 -- 4 drops/year) showed no indications of deformation or abnormality. The rods have demonstrated an excellent ability to withstand operation at accelerated frequency without failure.

An area of concern was that of individual control rods dropping off the holding magnet, without the associated scram relay actuating. Immediate corrective action was to transfer the scram magnet power from the non-noise sensitive to the noise sensitive supply, on the assumption the drops could have been due to line transients. Subsequent investigations, however, showed the spurious drops to be caused by problems related with initial startup, not with design.

The three prime causes were:

- a) Faulty magnets
- b) Shorted coils
- c) Improper limit switch settings.

The first two conditions obviously provided only partial holding power, thus making the rod susceptible to such drops. Improper limit switch settings allowed the rods to be pulled up to the "stops", literally pushing the rods off the magnets. All three problems have been rectified, and no further spurious drops have been encountered.

Although these problems are basically of a system check-out nature, i.e., predictable startup "bugs", periodic preventive maintenance checks utilizing the test cart will detect any recurrence.

It is concluded that the control rod system is operable per design. Drive and drop times have been verified to comply with technical specifications, and the indicators perform properly. Operation from preoperational to post critical testing of the plant shows no negative or limiting trends.

D. SODIUM SERVICE SYSTEM

1. INITIAL CHARGE AND OPERATION

Initial fill of the sodium service system was accomplished by means of a temporary system involving use of railroad tank cars, electric and hot oil heating, and the electromagnetic, sodium-transfer pumps. A total of 741,500 lb of sodium was transferred in filling the primary and secondary fill tanks. As with the later "first fill" of the heat transfer system, the service system had been pre-purged to reduce the oxygen concentration to $\sim 1\%$. System relative humidity was reduced to $\leq 10\%$, by heating. The melt-and-fill operations are discussed in AI-P-1104, the sodium service system preoperational test completion report.

Following completion of the fill operation, flow was established in both the primary and secondary sodium service systems, and cold trapping was initiated to lower the plugging temperature to $\sim 255^\circ\text{F}$. Cold trap performance was excellent, with only 24 hr of cold trapping being required to lower the sodium in each secondary tank to the desired plugging temperature. Cold trap runs on the primary fill tanks were ~ 36 hr each, because of a higher initial plugging temperature and greater volume of sodium.

Problems encountered during filling and preoperational testing of the service system were minor, and of the type normally associated with initial operation of a process system. Specific design-related component or system difficulties are discussed below.

2. SODIUM CIRCULATION AND COMPONENT PERFORMANCE

a. Purification Equipment

Low pressure differential at power levels below 30% of full power has created problems in cold trapping. The ΔP across the heat transfer pumps is

not sufficient to supply adequate flow to the service system. Throttling of the flow in the secondary system, to provide the required ΔP , at low power, is so difficult an operation as to have caused several primary-secondary flow ratio scrams.

Plans have been made for future modification of the service system which will allow use of the EM pumps as boosters to increase flow to the desired levels without throttling the convection control valves.

b. Sodium Valve Leakage

Leakage of radioactive sodium across the reactor fill and drain line isolation valves into the primary service system effectively limits access to the service cells during reactor operation. Safety precautions require that the reactor fill and drain line be maintained in a heated condition. Maintenance is, thus, frequently limited to periods when the reactor is shut down. This situation will be corrected by addition of a bypass line during the above mentioned service system modification. The bypass line, with three valves in the primary sodium service system will provide connection between the main coolant loop and the fill tanks, bypassing the sodium service cell. This arrangement, utilizing valves and frozen pipe sections will completely prevent the leakage of R/A sodium to the service cells.

c. Cold Traps

Cold trap efficiency and reliability, as demonstrated during the pre-operational and post critical testing, was excellent. The only difficulties associated with the traps concern remote removal of the primary cold traps. In the past, binding of the trap disconnect mechanism has necessitated personnel entry into the cell during each removal operation. Modifications made to improve reliability of the remote disconnect to the trap flange did not completely correct the problem. Remote removal is not necessary at this time, but may be required later, since a cold trap might contain long-lived fission products which would restrict access for removal.

An experimental "packless" cold trap (Figure 21) was installed in the secondary service system to test the operability of this cheaper design. Previously, successful operation of the traps was dependent on limiting ΔT between body and plugging temperatures to 50°F , with a 250°F minimum operating temperature. With the new trap, the ΔT was spread as far as 130°F without

Figure 21. HNPF Packed and Packless Cold Traps

interruption of flow due to plugging of the cold trap. Relative efficiency of the new trap was apparently slightly less than for the original design. However, it cannot be flatly stated that such was the case since operator inexperience with this unit caused certain "first time" problems which undoubtedly affected the length of the plugging runs.

Overall performance was very good, under the circumstances, but the lower capacity for total contained oxide does not warrant changing from "packed" to "packless" units at Hallam. Consideration should be given, however, to possible design modifications which will provide a high efficiency "packless" trap for future plants of the potential cost reduction.

d. Plugging Meters

Plugging meter operation has been excellent throughout the testing and post test operation. The evaluation section of AI-P-1104, Sodium Service System (Atomics International, October 1963) indicates the desirability of a

minor modification to eliminate poor flushing when the oxide saturation temperature approaches the bulk sodium temperature. This change is not considered necessary, but might be of help for initial operation of future systems.

e. Carbon Trap

The HNPF carbon trap is operated at intervals of 6 months or less at 1200°F and for periods of 100 hours. At the completion of the carbon trapping run, an inlet sodium sampler containing Type 304 stainless steel tabs (three thicknesses) is removed for chemical and metallurgical analysis of the steel tabs and sodium. Thus far, tabs exposed at the carbon trip inlet have shown no evidence of carburization, based on chemical analysis of the tabs for carbon and on metallurgical examinations which include visual etched case depth and micro-hardness traverses.

Fresh, Type 304SS tabs are exposed to the sodium sample at 1200°F for 100 hours in a laboratory test. The tabs are invariably carburized to varying degrees in this test and the estimated carbon contents of the sodium are relatively high. Chemical analysis of the sodium for elemental carbon are erratic, ranging from 4 to 147 ppm. This large variation in carbon content is due to inadequate techniques for measuring carbon in sodium.

At present, the results obtained for the steel tabs exposed at the carbon trap inlet are given the greatest credence.

f. Valve Operators

Early binding and failure of the service system valve operators was easily rectified. Difficulties arose when oxide deposits in the stem seals increased valve operating torque beyond operator design.

Installation of heavy duty universal joints and shear pins corrected the binding and alignment problems. A change in the gear ratio of each freeze seal valve remote operator ensured satisfactory operation of the valves. No further detrimental effects have been noticed in operation of the freeze seal valves.

g. Freeze Traps

Freeze traps (Figure 22) installed as part of the sodium service system high points to provide for venting or purging have caused much operational delay during fill and drain operations. A freeze trap is essentially a float valve

Figure 22. HNPF Freeze Trap

connecting the sodium system to a vent tap. Sodium provides the lift required to seat the float; a cam is provided for manually breaking the resulting seal. Secondary sealing, in the event the float fails to seal, is provided when sodium flowing past the float valve freezes in the cooled shaft housing.

The shaft packing provided for helium pressurization during back-filling with helium is not helium-leaktight. As a result, helium will leak away from the seal during heating operations. When the seal melts, sodium leaks up into the shaft annulus and into the vent, causing the vent to plug. An additional problem is found in operation of the cam to break the seal; the linkage for manually operating the shaft is such that there is poor "sensitivity" at the operator's end. Excessive pressure can inadvertently be applied, causing the shaft stop to fail.

Neither problem, however, is severe enough to warrant redesign or modification of the freeze traps. Aside from the operational inconvenience associated with opening the freeze traps, no other adverse affects result. A better system of venting and back-filling should be provided in future plants, based on HNPF experience.

h. Vapor Traps

Operation of the vapor traps, installed in the system to prevent sodium vapor from entering and clogging the helium lines, has been satisfactory except at high reactor venting rates, such as when control rod thimble leaks developed. Trap design and efficiency should provide many years of satisfactory maintenance-free service, but consideration should be given to the installation of more heaters.

i. Drum Melt Stations

The drum melt stations have been utilized on several occasions, to assist in the addition of sodium to the heat transfer system as required. Station operation was as expected, with only minor difficulties, such as leaking gaskets in one of the transfer lines. These stations should offer no tangible problems in future use.

E. PLANT CONTROL SYSTEM

Many minor problems were encountered during preoperational check-out of the plant control system. These discrepancies, discussed in the related test completion report, AI-P-1120, Plant Control System (Atomics International, October 1962), are indicative of poor quality control on the part of the manufacturer. Of the 19 failures and malfunctions discussed in the appendix to AI-P-1120, more than half are related to fabrication errors or faulty components. The remainder were basically design shortcomings of a rather insignificant nature, and were readily corrected under the guidance of the responsible test engineer. This test was effectively an operability check of the system, in many instances utilizing simulated signals to check system response.

Post critical test AI-PC-630, Plant Control System (Atomics International, January 1964), was a valuable follow-up to AI-P-1120, in that it served as an operational check under design conditions. System control adequacy and operational stability were demonstrated over the operating range, at conditions which were probably more severe than will be encountered at normal load-following conditions.

Of the 13 interrelated subsystems comprising the plant control system, seven demonstrated satisfactory operation, without a need for corrective action, during post critical testing. These were:

- 1) Plant Power Control
- 2) Nuclear Power Computer
- 3) Convection Flow Control
- 4) Attenuator Control
- 5) Power Relief Control
- 6) Primary Block Valve Control
- 7) Secondary Sodium Flow Control.

One subsystem, Primary Sodium Flow Control, was not completely checked out as part of the test program; the primary cold leg temperature recorder-controllers were not placed in service until a later date. Subsequent operation proved satisfactory, but the actual need for recorder-controllers is not clear. Evaluation of the need for this circuit will continue during extended operation under actual load-following conditions.

The Division-of-Load Computer, Moderator Coolant Control, and Steam Dump Control subsystems required only minor adjustments to produce the desired operating characteristics.

Those subsystems which posed slightly more serious difficulties were:

- 1) Neutron Flux Control — Below average component reliability prompted the decision to replace the original neutron flux controller. The new unit has been ordered and is scheduled for installation circa mid-1964. This replacement is functionally the same as the original, but is comprised of high reliability plug-in components which will provide longer periods of uninterrupted operation and greater ease of maintenance. The existing unit will be utilized as a spare.
- 2) Feedwater Level Control — Minor system changes were required to ensure satisfactory operation of this controller. The system was modified to provide low flow control through the bypass feedwater valve, thus creating a two-stage, or split-range, control system. As discussed elsewhere in this report (under Steam Generators) level programming equipment has been installed in the system as a result of a study concerning evaporator moisture carryover problems. The

level programmer should effectively help to lessen the possibility of carryover at higher powers. This condition is, however, being investigated further.

The efficacy of the Hallam plant control system is perhaps best pointed up by examples of attempted manual override, and an unexpected anomaly on the conventional side. On a few occasions, before the operators had gained sufficient operating experience, manual correction of apparent improper compensation by the control system (on automatic operation) resulted in a reactor scram. Subsequent investigation revealed the automatic action (e.g., rod withdrawal) was legitimate.

At a later date, with the nuclear and conventional plants operating in parallel, the conventional boiler suddenly dropped off the line. Nuclear plant response was immediate and completely as expected, with the reactor picking up full turbine load at a rate of ~ 20 Mwe/min.

Control system response to intentional transients has likewise been gratifying. As reported in the NAA-SR-9584^{*}, load-following by the nuclear plant meets all expectations, and assumptions based on post critical test program observations have been substantiated. Specific recommendations set forth as a result of the "response to transients" study include removal of the control rod position feedback mechanism, reduction of the setpoint for flux deviation automatic transfer to manual control, and a study to be made regarding possible removal of the primary cold leg temperature recorder-controllers because of the apparent self-regulating temperature effect discussed previously. A further study should also be made into the current excessive duty cycle on control rod equipment when the reactor is on automatic control.

The suggestion to remove the feedback mechanism is based on two factors: (a) the value of the mechanism is not sufficient to warrant the probable maintenance, and (b) the mechanism is so designed that it is possible to cause an automatic transfer of reactor control to manual, during the transient conditions which occur at the time of a large power swing. A transfer to manual control at this time will generally cause a scram.

^{*}P. F. Shaw, "HNPF Control System Response During Transient Conditions," NAA-SR-9584 (1964).

Reduction of the "flux deviation transfer" setpoint is likewise desirable for two reasons. Principally, by the time the present 8% deviation is reached, it is likely too late for operator corrective action to prevent a scram. Since the maximum deviation observed during the ramp response test was 1%, a 3% limit would provide a realistic limit which would not cause a transfer if the control system is operating properly, but would allow adequate time for corrective action by the operator in the event of a control system malfunction.

System reliability has been enhanced by such items as shrouds around the pumps to provide for decreased heat load to the speed control cabinets and elimination of dual (noise-free and critical) power supplies to individual components. A complete study of system and component reliability, however, will not be possible until such time as the reactor is brought to full power and carries the turbine load alone (with the conventional boiler off the line). Based on demonstrated design adequacy and excellent performance to date, it is concluded that the control system will successfully continue to meet all demands presented during future operation.

F. PLANT PROTECTIVE SYSTEM

Testing and operation of the HNPF Plant Protective System has been very extensive. Beginning with temporary modifications required for initial dry critical fuel loading (reported in AI-P-1166), protective system characteristics have been under close scrutiny.

1. NOISE

From the start, internal noise has been a cause for serious concern. An extensive investigation in late 1963 provided the most conclusive results to date. Based on the findings of this study, several system modifications have been initiated.

It was determined that a basic cause of noise was a poorly regulated AC power supply. Oscilloscope observation of line voltage produced a greatly distorted waveform with evidence of an excessive amount of "hash" or sporadic interference.

This noise was passed on and amplified by system design, which permits any disturbances on the noise-free bus to be directed to the most sensitive low-signal-level part of the system, the magnetic amplifiers. Since these units

provide a balanced differential input for the system computers, inputs from all the computers are necessarily fed into the amplifiers. The natural sensitivity of the units was further compounded by operation of the amplifiers at less than saturated excitation.

A lesser noise contributor is the grounding pattern. The neutral leg of the power line is grounded at the generator, and the rack and other components have separate grounds. A separate path or large-area inductive loop is formed as shown in Figure 23. The system is thus left open to inductive pickup. Unintentional or high-resistance grounds in the presumably full-floating thermocouple and flowmeter leads made a considerable portion of this "inter-ground" signal appear at the system input, where effective line voltage regulation is poor.

Figure 23. Large Area Inductive Loop Caused by System Grounding Pattern

2. TEMPERATURE AND DRIFT

Drifts in gain and trip points were experienced. Each computer output is read by one or more Schmidt triggers, the setpoints of which represent the operating limits "safe", "alarm", "setback", and "scram". It is important that the trigger does not load the circuit being measured; even more important is the unit hysteresis, or the difference in trip points as the input is first raised above, then lowered below the nominal setpoint. Loading of the measured circuit, and

inability to pass specification on hysteresis was found in over half of the Schmidt triggers, thus necessitating hand-selection of the units.

Temperature-related drift is of far more import. Actually, the temperature is more to be considered than the drift, because this parameter also affects component reliability. Design cooling provided for the protective system proved inadequate because of the normal heat load generated during system operation, and because of a high heat transfer from the steam generator area into the control room. Modifications are underway to correct this situation. Forced-air cooling will be directed across the components via special ducting, and flow will be assisted by suction fans installed at the cabinet tops (Figure 24). This is

Figure 24. Cooling Installation

designed to keep component degradation to a minimum, reduce calibration drift between test fixture and actual operation, and to prevent thermal runaway in the event of a building air conditioning system failure.

3. OPERATIONAL AMPLIFIERS

Chronic trouble has been experienced with lifetime of the mechanical choppers used in all of the 60 or so amplifiers in the protective system input circuits and in the computers. Investigation of the system ascertained that the apparently short mean chopper lifetime was actually not unreasonable for this type chopper under the HNPF operating conditions (continuous operation at low contact currents).

Chopper failures at Hallam are of two types:

- a) Burning, pitting and corrosion of the contacts
- b) Dwell angle discrepancies

The only means for extending the lifetime of the existing choppers appears to be a well planned preventive maintenance program. Procurement of a plug-in, direct replacement type that is entirely solid state and electronic is currently being investigated. Since completion of the formal test program, state-of-the-art advancements have made available a "better" chopper. Until such time as this area of failure is resolved, chopper maintenance and replacement will pose an operating nuisance, but involves no hazard.

4. SYSTEM INTERACTION

Interaction between protective system components has served to increase the incidence of "unreal" scrams. The two-out-of-three coincidence back-up is of little real value since a failure in one computer can send one or both of the related computers in the circuit into a scram condition.

The previously described circuit grounding pattern (see "II. F. 1 Noise") is considered one of the prime causes of computer interaction. Of the several solutions considered, the most effective is to divide the power feed at the protective panel (rather than at the generator) into five separate circuits to be fed from individual, fast-response line-voltage regulators. The regulators selected have a 100 db effective isolation between input and output power and 100 μ sec recovery time. So long as adequate power is being supplied, spurious anomalies

occurring between the generator and the protective panel will have no adverse effects. The isolation will allow the ground connection to act as a true reference, rather than an alternate return as has been the case in the past. Also, the provision for five voltage regulators continues the plant design philosophy of allowing any one part of the system to be worked on without causing the plant to be shut down.

A further aid in eliminating component interaction is planned in the form of constant voltage transformers to provide component isolation. This type of isolation will eliminate interactions between magnetic amplifiers, especially when a unit is reacting to a line transient such as experienced during removal of a computer. Likewise, noise pick-up from associated servo-motors will not affect the amplifiers at normal operating levels. Original system design allows a disturbance on any one computer to be transmitted directly to the most sensitive part of another.

Installation of DC power supply decoupling within the individual computer units, and the installation of a single-point grounding system to serve as the reference of computation for each computer will be implemented. This will enhance the value of a two-out-of-three coincidence by lessening interaction. Chassis and racks should serve as containment and Faraday shields, not as conductors.

Completion of the above listed system changes to correct for noise, ground pattern, temperature and drift, and system interaction will ensure reactor operation without "false" interruptions from the plant protective system. In future designs, it must be considered that it is all but impossible to provide a single, noise-free power supply for all "clean" power needs in a large plant having many sources of electrical noise. The most practical approach is to provide a source of continuous power to be regulated at the point of use. This, then, requires adequate consideration be given to fast response regulators, isolation and decoupling.

5. OPERATING LIMITS

With plant design altered to provide a reliable and credible protective system, the system setpoints must be scrutinized. At Hallam, it has been found that in some instances the system may be safe to the point of being detrimental. Specifically, it has been proposed that the dT/dt computer setpoints be changed

to allow a 50F° step-change instead of the present 25F° limits. This would not only lessen scram incidence, it would actually mean that the system would be subjected to less strain, i.e., the shock transients after a scram are equal to or greater than any introduced by a 50F° step-change.

As with other proposed system changes, either to equipment or operating limits, future design should be closely studied to yield a better definition of function and purpose. The bases for any particular design or limitation should be even more carefully enumerated than the later reasons for change; then perhaps there will be fewer occasions for later modification.

6. SYSTEM ADEQUACY

System performance to date has proven adequate from the safety aspect. All automatic actions occur per design, and no hazardous conditions have been allowed to prevail. With the one previous exception (over-protection), the system concerned is an excellent prototype.

G. NUCLEAR INSTRUMENTATION

Electrical noise interference and high instrument temperatures were encountered during check-out and operation of the Hallam nuclear instrumentation. Very short time constants also proved to be a major problem.

Following completion of the formal test program, a detailed investigation of the nuclear instrumentation was conducted. Findings from this investigation form the basis for the statements and conclusions carried in this segment of the final evaluation report.

1. STARTUP CHANNELS

a. Noise

Excessive fluctuation in readings, excessive sensitivity to electrical noise, and an annoying frequency of period alarms (due to statistical fluctuations) led to an examination of the pulse (startup) channels. These channels provided readings with fluctuations of far greater magnitude than normally associated with nuclear statistics. Contributing causes were: electrical noise, high input impedance and cable capacity between the preamplifier and counter.

The average signal-to-noise ratio in the nuclear instrumentation was ascertained to be 5:1, with at least two components making up the noise.

- 1) A low, continuous level of RF hash which is picked up between the chamber and preamplifier.
- 2) A higher amplitude, less frequent rate of impulse noise comparable to the signal amplitude.

Noise entry into the system apparently came at two points. The higher voltage impulse noise was traced to the log count rate meter chassis, with entry through the output leads; i.e., remote period, level, and recorder leads, and through the relay leads from the annunciator system. The remainder, lower in magnitude, but potentially more serious, entered between the preamplifier and log count rate meter input.

Corrective measures were undertaken to alleviate these problems (ref. Fig. 25). Since the signal cable between the pulse preamplifier could not be altered, it was made less sensitive to noise pickup by using pulse transformer coupling. This change in the system improved the signal-to-noise ratio in a threefold manner. The lumped capacity of the signal cable as seen by the BF_3

Figure 25. Neutron Chamber (BF_3) Pulse Transformer and Preamplifier

neutron detection chamber was reduced by a factor of 64 ($1/(\text{turns ratio})^2$). This produced a higher signal voltage

$$V_{\text{out}} = \frac{q}{c} \frac{(\text{chamber discharge})}{(\text{chamber and lumped cable capacity})}$$

The second way in which the circuit was improved was to match the pulse transformers to the 75 ohm characteristic impedance of the signal cable. This eliminated reflections in the cable and made it more difficult for external noise fields to induce spurious pulses into the system. The third method in which the signal-to-noise ratio was improved was the discrimination against gamma pulses. These pulses did not have enough energy to magnetize the pulse transformer core.

The original preamplifier was replaced with one designed to operate in the high temperature environment of the preamplifier pit. It has a fixed gain of 40, low noise, high stability, and an output impedance which matched the 300 ft cable between the pit and the control room at the sending end. This impedance match enabled the input impedance of the log countrate meters to be increased from 75Ω to $10K\Omega$. This increased the signal into the countrate meter by a factor of two, but eliminated signal reflections in the long signal cable.

Several tests were made to determine the results of these changes. An oscilloscope was used to observe the neutron pulses at the remote preamplifier. With the high voltage to the chamber turned off the noise level referred to the input of the preamplifier was less than 10 μv . The high voltage was then applied to the preamplifier. The input signal to the preamplifier was 30 mv, an increase by a factor of eight over the cable without the transformers. The output of the preamplifier was observed and no reflections were noted.

After channel II was modified a comparison was made with the unmodified low level channel I. A printer and scaler were connected to each channel while the reactor was shut down. The neutron pulses were counted for a 24 hour period. A statistical analysis and χ^2 * test were made on these data. The statistical analysis showed a ± 3 standard deviation spread on the modified channel

*The χ^2 test is used to compare expected and measured statistical distribution, and to estimate the probability that a system obeying the assumed statistics would produce the measured distributions. (A "perfect" distribution would have a 50% probability.)

II and a ± 6 standard deviation spread on channel I. Repeated tests of the modified channel yielded a range of probabilities of from 8% – 50%, with a most commonly occurring value of 40%. Tests made concurrently on the unmodified channel consistently produced probability estimates of less than 0.1%, showing that there was almost no chance that it was detecting nuclear events alone. Further tests of a different nature were performed at the same time, and indicated that 20% – 40% of the registered events were noise.

An operational interpretation of the test results is that the modified channel gave a reliable indication of the relative magnitude and sign of a change in flux level, and that the unmodified channel was unreliable in both sign and magnitude, particularly over the short term. It is the short term response that guides the reactor operator in the manipulation of the controls. From these results it was determined that the noise discrimination in the low-level channels has been markedly improved, and there is visual and statistical evidence that these channels are counting neutrons in the core.

b. Short Time Constant

Initial design (and specification), and a discrepancy during checkout of the short period circuit were the key to the extremely short time constant associated with this circuit. As installed, the circuit contained a one-sec RC time constant (1000 megohm resistor in parallel with a 0.001 mfd capacitor), which was entirely too short and resulted in numerous false period alarms due to statistical variations in count-rate.

It is important to smooth the data before presentation to a differentiating operational amplifier. This was subsequently effected at Hallam by the addition of another resistor and capacitor, giving the desired 5-sec time constant in advance of the differentiating loop (Figure 26). System performance since the change has been satisfactory. The trip circuit and level meter time constants were also increased. There is no safety hazard involved in the increases, since the new time constants are actually the initially intended values.

2. INTERMEDIATE RANGE INSTRUMENTS

Basic difficulties with the Log N (intermediate range) channels stemmed from poor design and poor quality control on the part of the manufacturer. Crude changes made to previous designs are evident in the "tack-on" type components

Figure 26. Differentiating Modifier with Time Constant Changed

delivered. Field changes by the manufacturer failed to rectify design deficiencies. E.g. A resistor added in parallel with the series pass tube increased the available current, but destroyed the regulation necessary for the desired DC stability of the instrument. Eventually, the tube had to be replaced with another of higher power handling ability.

Future designs should provide non-series-string electrometer filament circuits and independent positive and negative power supplies. The five second time constant as shown in Figure 26 was also added to the period circuit of the log N amplifier.

3. POWER RANGE INSTRUMENTS

Operation of the power range instruments has not been hampered by the problems encountered with the startup and intermediate range channels. Installation of high quality, stable, low-temperature-coefficient resistors has enhanced power range operation, as have later provisions for better filtering of "trouble" circuit pulses. This circuit is now filtered in such a manner as to allow reactivation in the flux control channels (previously disabled there because of incessant actuation of the control rods), and to keep the "trouble" pulse out of the pulse channels (I and II).

4. INSTRUMENT COOLING AND RELIABILITY

Design arrangement of system components was such that inadequate cooling was provided, thus leading to greater incidence of component failure and malfunction. Several corrective actions were undertaken:

- a) JAN tube shields were replaced with black, heat dissipating tube shields physically connected to specified heat sinks.
- b) A "beefed-up" forced air, ducted, suction-vented cooling system was installed (similar to that described in the protective system segment of this report.)
- c) Replacement parts of only the highest quality (high grade MIL-Spec or Minuteman) have been stocked and are being installed on a planned preventive maintenance schedule.

5. NUCLEAR INSTRUMENTS SUMMARY

Extensive testing, repair and modification of the HNPF nuclear instrumentation system was necessitated by inadequate quality control and apparent cost vs reliability considerations by the manufacturer. More stringent contract terms, placing responsibility squarely on the manufacturer, would undoubtedly reap valuable rewards in future plants. This means tighter basic specifications regarding component reliability guarantees, extensive environmental and electrical noise interference tests, as-installed noise rejection and cross-talk tests, and better in-service checks and calibration features.

The modified system should now yield effective monitoring of reactor nuclear operations for a great many years. As with the many other initial check-out problems encountered and overcome in this effort an invaluable amount of experience in the design and operation of nuclear components has been gained and documented. Future designs reflecting this added depth will be of key importance in advanced sodium cooled reactors.

H. RADIOACTIVITY – DETECTION AND CONTROL

1. RADIATION DETECTING AND MONITORING SYSTEM

Results of preoperational testing of the radiation detecting and monitoring system are reported in the associated preoperational test completion report,

AI-P-1122. As described therein, each of the nine interrelated subsystems was checked for operation according to the manufacturers' instructions. Minor modifications and system adjustments satisfactorily prepared the major portion of this system for long term operation.

RD-1 is a single-channel gamma spectrometer which is used to continuously monitor reactor cover gas, specifically for the 81 kev Xe^{133} activity. However, measurements obtained during initial operation were not consistent with the anticipated Xe^{133} level. A detailed study was conducted to determine the cause of the discrepancies. The results of this search comprise the bulk of the ensuing text.

It was proposed that the inconsistency might be related to one of three possible causes:

- a) Interference from sodium particles accumulated in the filter in the vicinity of the RD-1 detector.
- b) Interference from other cover gas constituents.
- c) Instrument malfunction.

RD-1 was tested with the reactor in operation, by means of the fuel cell purge unit.

The fuel cell purge unit spectrometer (which also serves as a back-up to RD-1) was calibrated over the 0 – 1.0 mev range, using a $0.5 \mu\text{c Cs}^{137}$ source. This calibration showed excellent instrument resolution (7.8%), and a very good peak-to-valley ratio (32.5:1). Further calibration, over the 0 – 1.0 mev range, was conducted with standard $0.5 \mu\text{c Ba}^{133}$, Cs^{137} and Co^{60} sources in cluster.

Next, a series of spectra was taken of the RD-1 filter (covered with sodium particles) onto which a standard Ba^{133} source had been mounted. This was to ascertain the possible interfering effect of sodium particulate on the monitoring of Xe^{133} activity. (Ba^{133} has the same gamma energy as Xe^{133} .)

Following subsequent back-up calibration of the spectrometer, using standard barium and cesium sources, core cover gas activity was measured. Spectra were obtained using both wide and narrow window width openings on the detector. These measurements pointed up three factors capable of impairing detection of Xe^{133}

- a) The presence of argon in the cover gas in addition to the anticipated xenon activities
- b) Particulate sodium accumulated in the filter to RD-1
- c) The lack of appropriate spectrometer settings.

Hence, as a result of 1) and 2), recorded Xe^{133} activity is actually a composite reading of Xe^{133} , A^{41} and Na^{24} , the extraneous items being capable of obscuring low level xenon activity.

Calibration of RD-1 was attempted after conclusion of the above tests, but could not be completed due to the RD-1 gamma spectrometer malfunctioning. Eventually, the preamplifier, amplifier and pulse height analyzer had to be replaced. Subsequent to the replacements, RD-1 was calibrated over the 0 – 1.0 mev range. Final recalibration, however, was in the 0 – 0.5 mev range, since this yielded optimum measurability of Xe^{133} .

RD-1 was returned to service, a series of cover gas measurements was recorded, final adjustments were made, and RD-1 was considered calibrated and operational. Based on the study, new operational settings were developed for RD-1. Since these new settings have been in use, the reactor cover gas monitoring system has operated satisfactorily.

The HNPF radiation detection and monitoring system is considered fully functional, and is operating properly in performing the design tasks, i.e., monitoring in-plant areas to ensure personnel protection by indicating off-normal conditions, and by preventing release of high-level R/A material to the surrounding area.

2. RADIATION SHIELDING

The results of a detailed evaluation of plant radiation shielding are presented in Section V of AI-PC-690, Radiation Shielding. The design is more than adequate, but how much more is not known. More than just adequate protection was, of course, the prime consideration. The degree of over-design, however, must be known in order to allow refinement of calculational techniques, as well as to increase confidence in these techniques. Construction costs are likewise dependent upon refinement of good techniques. By the lessening of excessive and unnecessary shielding, future estimates can be lowered, thus helping to

make nuclear plants more economical. For now, it is known that the present techniques will keep calculated values well on the safe side.

3. RADIOACTIVE VENT

Startup and operation of the R/A vent system has been normal, with the system satisfactorily collecting, monitoring and controlling the disposal of gas effluent from all potentially radioactive equipment and systems. Only minor startup problems were encountered; these have been corrected by simple modifications.

Most of the required changes were in the dry gas subsystem, where water was encountered. All water-containing equipment is now isolated from the dry gas headers and vented directly to the compressor suction tank. As further assurance against the entry of water into the subsystem, a moisture alarm was installed in the vent compressor suction tank, and moisture traps were installed in the system, to provide absolute stoppage of water. These and minor work in the wash cell vent system have the effect of assuring long term, satisfactory operation of the system.

4. RADIOACTIVE LIQUID WASTE

This system has been evaluated as being adequate and in proper operating condition. The only area of concern after completion of the formal test was RD-1100, which operates a control valve to stop flow to the leach field when the activity exceeds $2 - 4 \times 10^{-6} \mu\text{c/cc}$. This detector suffered two failures brought on by over-pressure. Temperature sensitivity is also a problem. A bypass was installed upstream of the detector, and a pressure regulator was added to the line, providing protection against further collapse of the detector tubes because of high line pressure. A temperature indicator installed on the unit provides information for compensation of the temperature sensitivity. Upstream and downstream nipples and T-joints were provided to allow flushing the unit in the event radionuclides are deposited on the sensing elements and interfere with proper indication of the liquid waste activity.

I. CORE COMPONENT HANDLING

Core component handling at Hallam has been concerned with fuel elements and control rod thimbles. Minor changes have been incorporated to facilitate

machine operation and/or to ensure greater component reliability. The basic design is considered adequate.

A detailed list of difficulties encountered during wet critical fuel loading shows only two specific areas of difficulty in 15 separate instances. Chamfering of the inside edges of the piston ring ends proved to be a good general solution to the problems encountered. This action served to allow proper positioning of the elements, without binding.

Other specific difficulties involved lock ring design failure to engage the component properly, grapple failure, and improper sequence of operation. In no instance was personnel safety compromised. The lock rings had to be chamfered to lessen the "land" and eliminate a potential hang-up area.

Based on experience gained at Hallam, advanced design fuel handling machine components have been fabricated for testing at the Atomics International Field Test Laboratories, Santa Susana, California. The new machine incorporates such refinements as in-core fuel pick-up (thus eliminating the need for pick-up cups and adapters presently in use at Hallam), roll-up steel tapes, instead of chains for raising and lowering operations, and high-volume helium circulation to allow immediate initiation of core component cooling as a component is being withdrawn from the core. These advances will provide faster, more efficient fuel loading and core component handling for future plants.

Facilities for cleaning and inspection of core components are discussed in the related test reports, AI-P-1133, Wash Cells, and AI-P-1139, Maintenance Cell. Both areas are considered adequate, and have been shown to operate within design requirements.

J. OTHER SYSTEMS

Results of preoperational and post critical testing of other systems not covered in this summary evaluation are included in the pertinent test completion reports which comprise the major portion of the overall final evaluation. "Backup" reporting is provided on a continual basis by monthly and semi-annual documents prepared by Atomics International's Operations Analysis Program. These reports are the result of extensive data taking and comprehensive analysis of all HNPF systems and subsystems, performed during daily operation. Raw data collected and processed by site OAP personnel are air mailed to Canoga

Park for coding and computer analysis. Trends are watched, and operating patterns plotted are utilized to develop a detailed analysis of component performance, reliability and lifetime. An example of this form of reporting is found in NAA-SR-9799, Hallam Nuclear Power Facility Reactor Operations Analysis Program Semiannual Progress Report No. 3, September 1, 1963 – February 29, 1964. These reports are available through the Office of Technical Services, Dept. of Commerce, Washington 25, D. C.

III. GENERAL CONCLUSIONS

Startup, testing, and subsequent operation of the Nuclear Power Facility at Hallam, Nebraska served a multitude of purposes beyond the basic scope of proving plant readiness for central station power generation. New concepts put into service will have far reaching effects on the design of future power plants and on the profitable development of nuclear systems in the area of commercial power generation. As a prototype, the HNPF will undergo testing and refinement modifications for a period beyond turnover to the customer (Consumers Public Power District of Nebraska). Efficacy of the basic design concept is best pointed up by the statistics logged during the last three months (November – January) prior to plant turnover. Figure 27 graphically depicts plant availability and power generation during that time.

Figure 27. HNPF Gross Electrical Generation, July, 1963 through January 1964

Further proving of the "plus" aspects of a sodium cooled reactor plant was of prime concern. High temperature, (945°F reactor outlet), low pressure (15 psia reactor outlet) operation, for instance, was demonstrated. Low operating pressure, of course, presents a relatively simple containment consideration (Hallam is not encased in the bubble dome of other concepts). This is amplified by the routine, unhurried actions which accompanied the IHX tube failure discussed elsewhere in this report.

Figure 28 shows the steam conditions associated with operation of the Nebraska facility^{*}. Excellent heat transfer, coupled with high operating temperatures, means modern, compact, more efficient equipment for conversion of heat-to-steam-to-electrical power. Compact equipment equates to a more compact plant, leading to lower cost per installed kilowatt.

Since the demonstration of safe and economical operation, as a reliable central station power producer is a prime HNPF objective, Atomics International's Operations Analysis Program is monitoring plant operation to provide

Figure 28. HNPF Sodium Heat Transfer System

^{*}Data presented are actual, determined during the post critical test program.

recommendations leading to higher plant availability, lower generating costs, and safe operation. Operating data accumulated and reduced to meaningful engineering parameters are interpreted and issued through periodic reports to provide industry, plant operators, and the scientific community with information covering the results of the performance analysis.

HALLAM NUCLEAR POWER FACILITY
PREOPERATIONAL TEST COMPLETION REPORT
MAIN HEAT TRANSFER SYSTEM

AI-P-1103

Test and Analysis

by

P. F. SHAW

Evaluation

by

L. L. JOHNSON

A T O M I C S I N T E R N A T I O N A L
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CONTENTS

	Page
I. Purpose of Test	1
II. Discussion of Test Performance	1
A. Visual Inspection of the System	1
B. Helium Entrainment Determination	2
C. Hydraulic Testing	3
III. Analysis and Presentation of Results	4
A. Visual Inspection of the System	4
B. Helium Entrainment, Determination and Resolution	4
C. Hydraulic Testing of the Main Heat Transfer System	20
IV. Compliance With Test Completion Criteria	21
V. Evaluation	23
A. Prefill Testing	26
B. Piping	26
C. Valves	27
D. Pumps	28
E. Secondary System Modification	29

APPENDICES

A. Piping Interferences	App. A-1
B. HNPF Main Heat Transfer System Throttle Valves Reliability Test	App. B-1

TABLES

I. Measured Secondary System Pipe Motion	6
--	---

FIGURES

		Page
1103-1	Secondary Loop No. 3 - Pump Speed <u>vs</u> Expansion Tank Pressure	7
1103-2	Secondary Loop No. 3 - Pump Speed <u>vs</u> Expansion Tank Level	8
1103-3	Secondary Loop No. 2 - Expansion Tank Level and Pressure <u>vs</u> Time	10
1103-4	Secondary Loop No. 3 - Expansion Tank Level and Pressure <u>vs</u> Time	11
1103-5	Isometric Drawing of Secondary Loop Modification (typical)	13
1103-6	Points of Maximum Secondary Pipe Motion	14
1103-7	Hydraulic Characteristics Secondary Loop No. 3	16
1103-8	Hydraulic Characteristics Secondary Loop No. 1	18
1103-9	Hydraulic Characteristics Secondary Loop No. 2	19
1103-10	Primary System Pump Speed <u>vs</u> Flow (obtained from Byron Jackson Pump Co. Water Tests and Primary System Performance with core pressure drop simulated by partial throttling of loop throttle valves). . .	24
1103-11	Secondary System Pump Speed <u>vs</u> Flow (obtained from Byron Jackson Pump Co. Water Tests & Secondary System Performance).	25

HALLAM NUCLEAR POWER FACILITY
PREOPERATIONAL TEST COMPLETION REPORT
MAIN HEAT TRANSFER SYSTEM, AI-P-1103

I. PURPOSE OF TEST

The purposes of the main heat transfer system test were:

- a) To determine that the main sodium heat transfer system was prepared for sodium filling and circulating operations.
- b) To verify that the main sodium heat transfer system, together with the pertinent portions of the plant control system and the plant protective system, functioned correctly under actual conditions of sodium flow. System hydraulic characteristics were also to be determined.

II. DISCUSSION OF TEST PERFORMANCE

A. VISUAL INSPECTION OF THE SYSTEM

This part of the test was performed at preheat temperature (350 F) prior to filling the system with sodium. Careful independent visual inspections were made by different individuals to determine the extent and effect of interferences between piping and equipment in the main heat transfer system. Numerous interferences were determined to exist; all were subsequently cleared by relocation of hanger attachment points, removing small quantities of insulation, modification of piping supports and clearing away structural material.

All main heat transfer system valves were operated through full stroke from the normal control points. Each remote valve operator was verified to operate the correct valve. Maintenance was performed on valve operators as required to ensure that each was properly operable.

B. HELIUM ENTRAINMENT DETERMINATION

The main heat transfer system was filled with sodium in accordance with AI-P-1157, Initial Fill of the Primary System With Sodium, and AI-P-1156, Initial Fill of the Secondary System With Sodium. Initial operation of the loops at low flow conditions was performed under the cognizance of these test procedures.

Testing of the six main free surface sodium pumps was initiated under a "Temporary Operating Procedure", for the purpose of verification of design, fabrication and installation. It was quickly determined that some malfunction of the secondary loops was occurring. Activity on the temporary operating procedure was suspended to permit an orderly analysis of the situation. An initial evaluation indicated that some method existed in each of the secondary loops for entrainment of helium into the flowing sodium. Subsections were added to this part of the test to define the conditions under which entrainment occurred and the mechanism which permitted it to occur.

The helium entrainment was determined to originate in the secondary expansion tanks. These tanks were designed to be flow-through devices with low-flow velocities where degasification of the sodium could take place. A helium supply and vent system communicates directly with each expansion tank to provide expansion space and an inert cover. The helium systems provide the method whereby the secondary loops are maintained at a higher pressure than the primary loops to ensure that any leakage between systems is into, rather than from, the normally radioactive primary system.

The conditions of gas entrainment in terms of pump speed, sodium flow and time were defined by the aforementioned additions to the test. The location of the entrainment process was determined by the processes of elimination and comparison of secondary to primary loop characteristics. It was deduced that the entrainment mechanism was high sodium turbulence in the expansion tank. The turbulence was created by high sodium flow through the expansion tanks. Sodium entered the tank at approximately the midpoint of the side and passed up over a baffle. Pump suction was taken from the bottom of the tank. The tank has antivortex vanes across the outlet to prevent vortexing.

Having defined the conditions under which the continuing entrainment process became objectionable, this test was continued with secondary pump speeds and times limited to those conditions that were considered safe for equipment, and that would not unduly affect test data.

In the interim, a fix was devised for the three secondary loops. This consisted of a modification whereby approximately 96 percent of the flowing sodium would bypass the expansion tank. The expansion tank thus became a head tank, with a bypass bleed from the main sodium stream passing through the tank to permit continuous degasification.

C. HYDRAULIC TESTING

Hydraulic testing was performed as a part of this test procedure. The testing accomplished served the primary purpose of promoting confidence in, and familiarity with, the system. Similar testing was accomplished during the performance of AI-P-1167, Hot Sodium Circulation Test. The hydraulic characteristics and other findings were presented in the test completion report for that test, and are therefore not duplicated here.

Two portions of this test, namely the determination of moderator flow characteristics and loop flow decay characteristics, could not be adequately investigated with the core unloaded. Since the normal core drop was not available, the primary throttle valves were partially closed to simulate core pressure drop and thus permit testing to continue; this imposed unrealistic conditions on the moderator loop and throttle valve operation. Hence, these two characteristics will be investigated during the performance of AI-PC-510, Hydraulic Test-Main Heat Transfer System.

Following modification of the secondary loops, each was subjected to a re-test to accomplish the following:

- 1) Ensure that the continuous process of gas entrainment in the flowing sodium stream was no longer present.
- 2) Determine what effects the modification produced on the hydraulic characteristics.
- 3) Accomplish a 48-hour full speed run.

Results of the post-modification test indicated that the condition of progressive gas entrainment was eliminated. Hydraulic characteristics of the loops were unchanged within the accuracy of the installed monitoring instrumentation. The 48-hour run was uneventful; all equipment performed as expected.

III. ANALYSIS AND PRESENTATION OF RESULTS

A. VISUAL INSPECTION OF THE SYSTEM

Independent visual inspections of the various areas containing piping and equipment were performed by several individuals. Lists of interferences, and observations compiled during the inspections, were compared. Areas where doubt or concern existed were discussed, and appropriate corrective actions were taken. A list of the piping interferences observed is presented in Appendix A of this report. All interferences were subsequently cleared.

Results of the operation of the air-motor-operated sodium system throttling and blocking valves indicated that no malfunctions existed at the time of the test. Subsequent to this test, primary throttle valve No. 3 failed mechanically. After repair of this valve, a reliability test was performed on the valves. Results of the test were documented by letter; this letter is presented as Appendix B of this report for reference purposes.

B. HELIUM ENTRAINMENT, DETERMINATION AND RESOLUTION

1. Investigation of Helium Entrainment

During the performance of pump tests, certain anomalies were noted that indicated that the secondary loops were not performing properly. These indications consisted of pump case high sodium level alarms, pipe motion, changing sodium levels in the expansion tanks, and inbreathing of helium gas. The pump test was discontinued, and a series of tests were performed to define the problem. This series of tests consisted of the following:

- a) The three loops were operated at pump speeds from zero up to 660 rpm, and the lines were observed for motion and sounds. Both became appreciable in each loop by the time a 660 rpm pump speed was reached. Points were selected on lines which appeared to have the highest amplitudes of motion, and measurements of the movement were obtained.
- b) At the point where maximum pipe motion was observed, 0.200 inch deflection was arbitrarily chosen as a limiting criterion. Each loop was subjected to another test run, and the following data were recorded for analysis:
 - 1) Pump speed in rpm
 - 2) Pump motor phase current
 - 3) Flowmeter signal in millivolts
 - 4) Pressures:
 - a) Pump suction
 - b) Pump discharge
 - c) Expansion tank (helium supply blocked)
 - d) Pump case
 - e) Evaporator inlet
 - f) Evaporator outlet
 - 5) Pipe motion at two points on each of two lines (IHX secondary inlet and outlet) (data are presented in Table I.)

In the above test, loop No. 3 had the smallest maximum amplitude and was therefore chosen for further definitive testing.

- c) The helium cover gas pressure on loop No. 3 was established at 20 psi; the helium supply was blocked, and the pump was operated at speeds from 170 rpm to 700 rpm while recording all of the foregoing data, except pump case pressure and evaporator pressures. The data obtained are presented as "Run A" in Figures 1103-1 and 1103-2.
- d) The helium cover gas pressure in loop No. 3 was decreased to 3 psig and the pump speed was varied from 170 to 700 rpm, taking data as before. The data obtained are presented as "Run B" in Figure 1103-2.
- e) The sodium level in loop No. 3 expansion tank was increased 13 inches (from 1466' - 5" to 1467' - 6" elevation) and the pump speeds again varied from 170 to 700 rpm.

TABLE I
MEASURED SECONDARY SYSTEM PIPE MOTION

Loop No.	Before Modification					After Modification				
	Pump Speed rpm	Movement - Inches At Point				Pump Speed rpm	Movement - Inches At Point			
		A	B	C	D		A	B	C	D
1	350	0.040	0.035	----	----	300		0.012		0.006
	420	0.050	0.065	0.030	0.050	400		0.040		0.010
	490	0.070	0.100	0.030	0.070	500		0.045		0.020
	560	0.120	0.140	0.040	0.105	600		0.065		0.040
	630	0.115	0.190	0.050	0.125	660		0.100		0.050
2	170	0.035	0.055	0.010	0.015	300		0.012		0.003
	450	0.040	0.080	0.020	0.025	400		0.020		0.004
	570	0.080	0.150	0.030	0.045	500		0.040		0.007
	630	0.110	0.185	0.035	0.065	600		0.075		0.020
						625		0.110		0.025
3	195	0.005	----	----	----	300		0.020		
	380	0.015	0.020	----	----	400		0.030		0.025
	450	0.020	0.030	0.013	0.020	500		0.050		0.050
	520	0.035	0.040	0.020	0.030	600		0.060		0.055
	590	0.055	0.080	0.025	0.040	700		0.065		0.070
	660	0.055	0.105	0.040	0.070					

NOTE: Movements listed are peak to peak amplitudes at the points indicated in Figure 1103-6. Frequency of movements were approximately 120 cycles/min.

Figure 1103-1. Secondary Loop No. 3 - Pump Speed vs Expansion Tank Pressure

Figure 1103-2. Secondary Loop No. 3 - Pump Speed vs Expansion Tank Level,

The data were recorded as in c) (preceding) at a helium pressure of 20 psig, with the helium supply block valve closed. The data obtained are presented as "Run C" in Figure 1103-1 and 1103-2.

- f) Loops No. 2 and No. 3 were operated at pump speeds of 400, 500, 550 and 600 rpm for 30 minutes or more at each speed to determine the speeds at which entrainment became appreciable and to determine if a time dependency of gas entrainment mechanism existed. Data obtained are presented in Figures 1103-3 and 1103-4.
- g) The final test was performed on loop No. 3, and consisted of using an argon cover gas in place of helium to investigate the relationship between cover gas density and entrainment. The results of this test established that the indicated entrainment process was clearly worsened by the use of argon.

2. Analysis of Data Obtained

Analysis of data from the previously mentioned series of tests revealed the following:

- a) Cover gas was being entrained into the sodium and was transported along by the flowing sodium and distributed through the system during operation at elevated pump speeds under all test conditions.
- b) Entrainment was speed (actually, flow) and time dependent; essentially no indication of entrainment was observed below 400 rpm, nor before an hour's operation at higher speeds.
- c) Pipe motion was related to pump speed (sodium flow), and to the amount of entrainment that had occurred.
- d) Reduced helium cover gas pressure, use of argon in place of helium for cover, and loop operation with reduced sodium level each resulted in a higher rate of indicated entrainment.
- e) Bubbling and gurgling sounds in certain areas of the system increased proportionately to the other indications of entrainment.
- f) The pump case vent lines were clear.
- g) Gas entrainment was not occurring in the pump case, since pump case sodium level increased with continuing entrainment. The pump case overflow line communicates directly with the expansion tank, not the pump suction.

Figure 1103-3. Secondary Loop No. 2 - Expansion Tank Level and Pressure vs Time.

Figure 1103-4. Secondary Loop No. 3 - Expansion Tank Level and Pressure vs Time

Vortexing in the pump case is prevented by antivortex baffles. It was therefore concluded that no mechanism existed in the pumps to support the gas entrainment process.

A review of the secondary system design revealed no location where entrainment could occur, except in the pumps and expansion tanks. Since entrainment in the pumps had been eliminated as a possibility, it was concluded that the fault must lie in the expansion tanks.

3. Modification of the Secondary Main Heat Transfer Loops

A modification to the secondary loop design was developed, whereby approximately 96 percent of the flowing sodium bypassed the expansion tank. This was accomplished, as indicated in Figure 1103-5, by cutting and capping the inlet line to the expansion tank and diverting the flow directly to the pump suction line. A three inch diameter bypass line was installed as indicated in Figure 1103-5 to permit a small flow through the expansion tank to provide for continuous degasification. The modification was first completed on loop No. 3. After satisfactory retesting of this loop was accomplished, loops No. 1 and No. 2 were similarly modified and tested.

4. Post-Modification Hydraulic Testing of Secondary Loop No. 3

After completion of the modification, secondary loop No. 3 was preheated and filled with sodium to an expansion tank sodium surface elevation of 1467' 4". The following tests were performed to verify that the entrainment no longer existed.

- a) Pump speeds were increased in 100 rpm steps from zero to 700 rpm. At each speed, pump suction and discharge pressures, pump speed, sodium flowmeter signal in millivolts, and pipe motion at certain locations, were recorded. This portion of the test was repeated when it was discovered that the pump case vent valve was not tightly closed, and the helium cover gas pressure had decreased. Maximum pipe motion occurred at points B and D (Figure 1103-6) on lines 275 and 266 in the IHX cell. Maximum motion during the above test was determined to be 0.070 inch.

Figure 1103-5. Isometric Drawing of Secondary Loop Modification (typical)

Figure 1103-6. Points of Maximum Secondary Pipe Motion

- b) A 48-hour continuous run was made at this time at speeds from 620 to 645 rpm. Helium usage during the run was below instrument sensitivity. The expansion tank sodium level remained essentially unchanged.
- c) At the end of the 48-hour run, the pump speed was decreased in 100 rpm steps while data were recorded as before. These data were essentially unchanged from that obtained before the 48-hour run.

NOTE

Pertinent data obtained in a) and c) above are presented in Figure 1103-7 and Table I. Hydraulic characteristics obtained at 350 F before the modification (during the performance of AI-P-1167, Hot Sodium Circulation Test) are included on the figure for reference.

- d) A 16 hour run at 675 rpm pump speed was performed with the helium supply blocked. The helium pressure decreased 1.9 psi, from an initial pressure of 19.4 psig, while the sodium level remained constant. It was subsequently determined that the relief valve on the vent line was leaking sufficiently to account for the pressure decrease.
- e) An 8-hour run at 675 rpm pump speed was made with a helium cover gas pressure of 7.7 psig and the helium supply blocked. As was the case in the premodification tests, pipe motion was slightly higher with reduced pressure. The helium pressure decreased very slightly, while the sodium level remained constant.
- f) With the pump stopped, the helium cover gas pressure was bled off in 5 psi steps from 20 psig to 0 psig., while recording expansion tank sodium level. During this operation, the sodium level instrumentation indicated an increase of .02 feet. This verified the belief that some gas remained in the system at this time.

Throughout the above tests, flow noises existed in the IHX at pump speeds above 585 rpm. The noises were similar to those observed during loop operation prior to the modification, but were of lower amplitude.

By observation of the system performance and analysis of the data accumulated during the above series of tests, the following conclusions are drawn:

Figure 1103-7. Hydraulic Characteristics Secondary Loop No. 3

- a) The problem of progressive helium gas entrainment was resolved by the modification.
- b) Gas was present in the loop in an unknown amount after the loop was filled with sodium.
- c) The amount of entrained gas decreased with time and operation of the loop.

5. Post Modification Hydraulic Testing of Secondary Loop No. 1

Upon completion of the modification of secondary loop No. 1, the loop and associated equipment were preheated and refilled with sodium at a temperature of approximately 350 F. After approximately two days of operation at 200 rpm, the verification test on this loop was performed, as described in the following paragraphs.

The pump speed was increased in 100 rpm increments from 0 to 660 rpm, while recording pump speed, suction and discharge pressures, loop sodium flow signal (in millivolts), and piping motion at points B and D indicated in Figure 1103-6. Upon reaching a pump speed of 660 rpm, an extended run of 48 hours at this speed was performed. At the end of the 48 hours, pump speeds were decreased in 100 rpm steps to obtain additional hydraulic data. Data obtained during the speed decrease agreed favorably with that obtained during the speed increase prior to the 48 hour extended run.

Performance of loop No. 1 was similar to that observed on loop No. 3. Greater pipe motion existed in modified loop No. 1 than in No. 3, however this was also the case prior to the modification of each loop. Hydraulic characteristics of the two modified secondary loops are essentially the same. The results of the testing of loop No. 1 are presented in Figure 1103-8.

6. Post Modification Hydraulic Testing of Secondary Loop No. 2

The post modification of secondary loop No. 2 was essentially the same as that for loop No. 1. The data obtained are presented in Figure 1103-9.

Figure 1103-8. Hydraulic Characteristics Secondary Loop No. 1

Figure 1103-9. Hydraulic Characteristics Secondary Loop No. 2

C. HYDRAULIC TESTING OF THE MAIN HEAT TRANSFER SYSTEM

The core pressure drop was simulated by positioning the primary throttle valves to the point where a pump differential pressure of 38.4 psig was developed at the pump speed calculated to be necessary for full design loop flow (approximately 675 rpm).

Control of each loop was transferred from manual pump speed control to manual circuit flow control to verify that a bumpless transfer of control could be made. Circuit flow was varied to determine if any interrelationships existed between loops, and to verify that the loops would respond properly to flow change demands. No interaction was observed to exist under the conditions of the test.

Flow trimming was accomplished as necessary to equalize the measured flows to demand flow by adjustment of FCCp-501 through FCCp-506. Control of the six loop flows was then transferred to HCS-504 to verify that the six loops would respond to demand signals. The setpoint of HCS-504 was varied, and the loop flows responded as expected. Data were recorded at each setting to yield system performance characteristics.

After establishing the above base conditions, the six main heat transfer system loops were operated at several pump speeds. At each speed, data were recorded to permit the determination and evaluation of hydraulic characteristics. Each pump was ramped from zero to full speed and back to zero speed to determine if any deleterious effect on the system existed.

By using the loop pump speed hand control station to unbalance the loop flows, each circuit was placed first in alarm and then in scram condition. The alarm and scram conditions were generated in the flow ratio computers of the plant protective system.

Each circuit was removed from and returned to service, using the one-circuit shutdown method to verify that one-circuit shutdown could be accomplished.

The moderator throttle and block valves were operated through one full-open to full-shut to full-open cycle to verify operability. With the flow in each primary loop at 50 percent of rated value, moderator flow indication

was recorded for the moderator throttle valve position controller (HCS-9) at 50 percent and 100 percent closed. The data obtained are not applicable to operating conditions, since the core was not loaded at this time.

An attempt was made to determine loop flow-decay characteristics. The data obtained at this time were meaningless, since the secondary loops were not yet modified and the primary throttle valves were more than 50 percent closed to simulate operating system pressure drop.

The three primary loops were operated at full speed for a period exceeding 48 hours. No problems were encountered during the run.

IV. COMPLIANCE WITH TEST COMPLETION CRITERIA

Following are the test completion criteria associated with preoperational test AI-P-1103, Main Heat Transfer System, As listed in NAA-SR-5700, Supplement 4, Page XI(a)-2, -3, and -6, with remarks regarding the test to explain compliance with the criteria.

- a) Criterion: No binding or interference of pipes, pipe hangers and valve operators exists at preheat temperatures.
Remarks: All observed interferences were cleared.
- b) Criterion: No gaps in piping insulation exist at preheat temperatures.
Remarks: This was verified by the several inspections performed by different individuals.
- c) Criterion: All valves operate freely in accordance with plant design at preheat temperatures.
Remarks: All main heat transfer throttling and blocking valves were operated by hand and by use of the airmotor operators with no indications of binding or malfunction.
- d) Criterion: Each pump motor starts and comes up to rated speed (approx. 870 - 890 rpm) with no load.
Remarks: This was verified prior to and following the filling of the systems with sodium.
- e) Criterion: A bumpless transfer from demand pump speed hand control station to the circuit demand steam flow hand control station can be made for each circuit.
Remarks: This was verified and reported under preoperational test AI-P-1167, Hot Sodium Circulation Test.

- f) Criterion: Each pump can develop a sodium flow of about 2.8×10^6 lbs/hr in each circuit with a pressure drop of about 38.4 psig in the primary loops.
- Remarks: Actual operating characteristics will be determined in post critical test AI-P-510, Hydraulic Test - Main Heat Transfer System. The present indications are that the pump capacities are adequate to meet the above criterion.
- g) Criterion: Each circuit can be bumplessly removed from and placed back into service with no adverse effect on the other two operating circuits.
- Remarks: This was verified.
- h) Criterion: The sodium circulation systems can be operated continuously at simulated full power flow conditions and pre-heat temperatures for 48 hours minimum continuous run.
- Remarks: This was verified for each of the six loops.
- i) Criterion: A flow decay-time relationship has been obtained for each circuit following a "scram" from simulated full power flow conditions.
- Remarks: Data presently obtainable from this test are meaningless, since the primary throttle valves were utilized to simulate loaded core pressure drops. This relationship will be determined in the post critical testing program.
- j) Criterion: All flow-ratio computer and controller alarms and annunciators actuate and operate as intended. Setpoints have been verified.
- Remarks: This was verified.
- k) Criterion: All sodium scram setpoints have been verified to be set as given in the test procedure by actually "scramming" the system.
- Remarks: Primary to secondary flow ratio alarm and scram, setpoints were verified for each of the nine flow ratio computers.
- l) Criterion: Each pump operates without excessive vibration, noise, or other indication of malfunction for extended runs, and can be "scrammed" instantaneously to no-load without distress.
- Remarks: This was investigated and reported in the test completion report for AI-P-1167, Hot Sodium Circulation Test.
- m) Criterion: The sodium flow, pump speed and pressure drop data requested in the test procedure have been obtained for each loop at different flows.
- Remarks: This was accomplished. Refer to the test completion report for AI-P-1167, Hot Sodium Circulation Test for analysis and evaluation of the data obtained.

V. EVALUATION

Preoperational test AI-P-1103 was conducted to verify the preparedness of the system for elevated temperature low power operation. Specifically, the system was tested to determine that no detrimental conditions exist as a result of preheating or filling, that valves and pumps perform satisfactorily, and that no potential problems exist that would impair system integrity or operation at design conditions.

The test purposes were accomplished in accordance with the approved test procedure. The test completion criteria as specified in the HNPF Safeguards Report (NAA-SR-5700, Supplement 4) were accomplished, with the following exceptions:

- 1) System flow decay-time relationship was not determined. As mentioned in the test completion report, it was not possible to obtain meaningful data, since the primary throttle valves were in a partially throttled position to simulate the loaded core pressure drop. It is intended that this test be conducted during the initial part of the post critical testing of the heat transfer system (Hydraulic Test - Main Heat Transfer System, AI-PC-510).
- 2) System hydraulic characteristics were not accurately determined due to the inability to obtain precise flow measurements.* However, comparison of pump performances with test data obtained by the pump manufacturer (Figures 1103-10 and 1103-11) provides assurance that flows in excess of the design full power flow (2.8×10^6 lb/hr) have been achieved. Recalibration of flow meters will be based on heat balance calculations which will be obtained during the performance of post critical test AI-PC-530, Heat Transfer Tests.
- 3) The moderator coolant circuit flow characteristics were not established. Although the ability to flow sodium in this circuit was determined during the test, the core must contain an operational fuel loading to provide meaningful data. Actual hydraulic characteristics will be determined during performance of AI-PC-510.

Many system deficiencies were pointed up by both the heat transfer system test (AI-P-1103) and the hot circulation test (AI-P-1167). Problems

* Reference evaluation of preoperational test of the sodium flow instrumentation (AI-P-1160).

Figure 1103-10. Primary System Pump Speed vs Flow (obtained from Byron Jackson Pump Co. Water Tests and Primary System Performance with core pressure drop simulated by partial throttling of loop throttle valves).

Figure 1103-11. Secondary System Pump Speed vs Flow (obtained from Byron Jackson Pump Co. Water Tests & Secondary System Performance).

such as gas entrainment in the secondary loop, pipe support binding, valve malfunctions and pump level anomalies were experienced during the initial operation of the system. From all indications, these problems have been adequately corrected and the system can be expected to provide satisfactory and dependable service at design operating conditions.

The portion of the test pertinent to operation after introducing sodium into the system was conducted as a part of the hot sodium circulation test, AI-P-1167. Additional testing was performed after completion of AI-P-1167, but only for the purpose of checking portions of the system which were modified subsequent to that test. To provide a complete evaluation of the heat transfer systems in this document, portions of the hot circulation test evaluation are repeated herein.

A. PREFILL TESTING

The test adequately verified that the system was prepared both to be filled and to circulate sodium. It was determined that satisfactory preheat conditions existed, and that all anomalies * which occurred during preheating had been resolved.

Visual inspection of the system and subsequent elimination of all interferences assured freedom of piping and equipment movement during initial sodium circulation. Pipe hangers were adjusted as required to assure adequate support of the piping when filled with sodium.

B. PIPING

The hot circulation test confirmed that the heat transfer system piping will not be subjected to detrimental stress and strain resulting from thermal expansion over the full operating range. Actual measured piping movements which resulted from increasing the system temperature from 350 F to 540 F compared well with the calculated predicted movements, thus confirming the

* Reference HNPF preoperational test completion report - Preheating System, AI-P-1116.

validity of the piping stress analysis performed during the system design phase. In addition, free movement of the piping was assured by elimination of all potential interferences which would provide restraint at the maximum design operating temperatures.

All variable spring hangers were readjusted to the proper setting, which will result in design support load at operating temperature. Some difficulties were experienced with binding of the supports for the core cavity piping, but these were reworked to eliminate all interferences. New, more accurate, position indicators were installed to permit comprehensive determination of proper piping movement in the cavity during future high temperature operation.

C. VALVES

The main throttle and block valves were tested extensively at temperatures from 350 F to 540 F. During initial operation, three throttle valve failures occurred. Two were due to misalignment of the actuating mechanism, and one was a result of internal binding. Subsequent to the repair of these malfunctions, five of the six valves were subjected to a cycling test consisting of operating the valves over the full travel 30 - 40 times. All valves performed satisfactorily except for primary throttle valve No. 1, which showed evidence of excessive stem actuating force requirements. This is attributed to an inherent characteristic of freeze seal valves; i. e., the accumulation of sodium oxide in the stem seal which occurs frequently in systems operating with a high oxide content for prolonged periods of time.

Continuing problems of this nature can be expected unless some corrective action is taken. Freeze seal valve actuating requirements have been maintained within tolerable levels at other installations by periodically melting, blowing down and reestablishing the seal. In the future, if this is considered a preventive maintenance operation and performed during each regularly scheduled plant shutdown, the valves should perform satisfactorily.

D. PUMPS

All heat transfer system pumps have demonstrated the ability to produce design mass flow (2.8×10^6 lb/hr) at temperatures from 350 F to 540 F. With the exception of a few apparently minor problems (discussed below), the pumps can be expected to perform adequately at design operating conditions.

Low sodium levels in the primary pump cases have resulted when the pumps are operated at the higher flow rates (60 to 100 percent of design flow) with less than design head conditions. Under design head conditions, the sodium flow from the pump discharge across the impeller wear rings, through the fluid bearing and across the barrel peripheral seal into the case, is greater than the flow from the case through the impeller weep holes to the pump suction. This provides a positive flow into the pump case and a constant operating level is maintained by the case overflow.

When the pumps are operated at low head conditions this flow relationship is upset and the flow from the case to the pump suction becomes greater than the flow supplied to the case. This results in lowering of the case sodium level; ultimately gas could be pulled into the suction side of the pump, become entrained in the flowing stream, and passed through the core.

Discussions with representatives of the pump manufacturer indicated that this condition could be rectified by plugging a portion of the impeller weep holes. In view of the gas entrainment potential and the undesirable effects on core stability, it was thought that a modification of all primary pumps was essential. Therefore, four of the eight impeller weep holes in each primary pump impeller were plugged. Tests performed on pump No. 1 after modification revealed that the above undesirable condition no longer existed.

The pumps have exhibited squealing sounds at speeds from 350 rpm to 700 rpm. The noise is speed dependent and does appear to be effected by changes in flow or head conditions. On one occasion the rotation of secondary pump No. 3 was reversed and the noise was still present in essentially the same speed range. A reduction in noise level has been effected on the secondary pumps by reducing the expansion tank pressure below the normal

operating pressure. An increase of expansion tank level also produced similar results.

Extensive investigation of the noise indicated it was frequency related to the impeller-diffuser vane interaction. No obviously detrimental effects have resulted, nor are expected to result from this condition.

E. SECONDARY SYSTEM MODIFICATION

Modification of the secondary circuits has eliminated progressive gas entrainment. This is substantiated by the fact that there is no system level change nor helium usage associated with operation at the high flow rates.

Hydraulic noises (gurgling) are audible in the vicinity of the evaporator outlet, which indicates the possibility of trapped gas. The evaporator inlet and outlet plenums are equipped with vents to prevent gas pockets from building up in these regions. It is quite conceivable that the inlet plenum vent, which is only a 1/4 inch radius semicircular hole, could become plugged. In this event there could be a small quantity of gas trapped in the inlet plenum above the top row of tubes.

If the outlet plenum vent became plugged it would be somewhat more serious as the evaporator outlet is on the bottom and large quantities of gas could accumulate. The outlet plenum vent lines are one inch in diameter and are less likely to plug than the smaller inlet plenum vent. An investigation into the possibility of plugged vents yielded indications that they were all clear.

APPENDIX A
PIPING INTERFERENCES

The following piping interferences were determined by the investigation of system piping:

- 1) IHX Cell No. 1
 - a) Hanger No. 148 on line No. 273 bearing against steel brace.
 - b) Line No. 237 - bolt on pipe hanger is too close to line 273.
 - c) Line No. 231 - pipe hanger rod between east-west run to cell ceiling interfering with line; hanger No. 864A on line No. 231 bearing against line No. 241.
- 2) IHX Cell No. 2
 - a) Two insulation boxes on line No. 186 bearing against west wall.
 - b) Line No. 238 - discharge from IHX - Clearance to cable tray stiff leg insufficient.
 - c) Line No. 235 - interference with a conduit (E side of cell, approximately 15 feet above floor).
- 3) IHX Cell No. 3
 - a) Hanger No. 421 on line No. 239 bearing against line No. 233.
 - b) Line No. 239 - W side of cell - Sharp corner of angle iron bearing against line.
 - c) Cable tray leg in NW corner is unrestrained.
 - d) Hanger No. 452A on line No. 233 bearing against line No. 249.
- 4) Nitrogen Cooling Area No. 1
 - a) Line No. 192 bearing against nitrogen cooling housing.
 - b) Air duct tearing insulation box from line No. 273 at hanger No. 182.
 - c) Line No. 273 bearing against cable tray.
- 5) Nitrogen Cooling Area No. 2 - Line No. 198 bearing against nitrogen cooling housing.

- 6) Nitrogen Cooling Area No. 3
 - a) Line No. 204 bearing against nitrogen cooling housing.
 - b) Air duct bearing against secondary throttle valve No. 3.
- 7) Steam Generator Room No. 1
 - a) Line No. 209 bearing against service lines.
 - b) Line No. 267 one-fourth inch from kick plate.
 - c) Vertical support on personnel platform penetrates insulation on line No. 303.
 - d) Shield on insulation off at hanger No. 154B on line No. 264.
 - e) Hanger No. 156B bearing against line No. 270.
 - f) Thermocouple tubing on line No. 270 bearing against room divider.
 - g) Line No. 312 has a bent hanger eyebolt - Replace.
 - h) Line No. 201 touching lines No. 1020 and 1828.
 - i) Building wall penetrates insulation on line No. 217.
- 8) Steam Generator Room No. 2
 - a) Hanger No. 260B on line No. 265 bearing against air duct.
 - b) Line No. 325 bearing against building joist.
 - c) Line No. 268 one-fourth inch from kick plate.
 - d) Line No. 210 bearing against service lines.
 - e) Vertical support on personnel platform penetrates insulation of line No. 307.
 - f) Line No. 210 touching lines No. 1020 and 1828.
- 9) Steam Generator Room No. 3
 - a) Line No. 211 bearing against service lines and west wall.
 - b) Line No. 326 bearing against building joist.
 - c) Line No. 269 bearing against kick plate.
 - d) Vertical support on personnel platform penetrates insulation on line No. 269.
 - e) Insulation on line No. 266, at expansion tank, sagging.
 - f) Hanger No. 262D bearing against line No. 272.
 - g) Hanger No. 261A bearing against air duct.

- h) Thermocouple terminal box and tubing on line No. 272 bearing against room divider.
- i) Loose thermocouple tubing on line No. 272.
- j) Line No. 278 bearing against west wall.
- k) Line No. 295 bearing against west wall.
- l) Conduit on line No. 269 loose.
- m) Line No. 211 touching line No. 1828.

10) Primary Pipe Tunnel (West)

- a) Hanger No. 671A on line No. 168 bearing against thermocouple terminal on line No. 239.
- b) Hanger No. 208A on line No. 141 bent loose and bearing against line No. 232.
- c) Line No. 145 bearing against south wall.

11) Primary Pipe Tunnel (North)

- a) Two insulation boxes on line No. 159 bearing against south wall. One torn from pipe.
- b) Hanger No. 506 on line No. 224 bearing against line No. 123.
- c) Hanger rod on trapeze hanger on line No. 231 bearing against line No. 169.

12) Primary Pipe Tunnel (East)

- a) Valve V-463 bearing against line No. 237.
- b) Valve V-461 bearing against building column; insulation is cut away.
- c) Line No. 413 bearing against steel beam.
- d) Spare piece of scrap angle iron on ledge at east reactor cavity seal plate welded to liner plate.
- e) Insufficient clearance between line No. 116 and trapeze pipe hanger on line No. 231, 3 places.

APPENDIX B

HNPF MAIN HEAT TRANSFER SYSTEM THROTTLE VALVES RELIABILITY TEST

Following is the text of an IOL from J. D. Berg to S. F. Burton, regarding the reliability test of the HNPF main heat transfer system throttle valves:

"Objective:

"To determine the reliability of the throttle valves in the main heat transfer system.

"Method:

"Prior to this test a visual inspection of each throttle valve was made to determine the condition of 1) the shaft and 2) the packing gland of each throttling valve. The shafts were cleaned with butyl-alcohol and water to remove any sodium oxides and/or contaminants. Photographs were taken at this time to show any scoring marks on the shaft and the condition of the graphite-asbestos packing gland.

"The throttle valves in the main heat transfer system were cycled from full open to the 98% closed position, with a time interval of approximately 15 minutes between cycles. The three primary throttle valves were operated through 41 scram cycles (open-close-open) while the two secondary throttle valves were subjected to 33 scram cycles. (Secondary throttle valve No. 3 was omitted from this test, as loop No. 3 was drained for system modifications.) Scram action was initiated by resetting the convection flow control subsystem and de-energizing the scram circuit with keyswitch 1/cs. The position of the throttle valve stems was measured in both the full open (Figure 1) and 98% closed position, (Figure 2), for each cycle. Periodically, the travel of each valve stem was carefully observed to ensure that the throttle valve operated freely (without excessive noise or binding) and that the valve did not strike the backseat. During these observations, measurements were

FIGURE 1

FIGURE 2

taken of the time required for the throttle valves to close to the 96% position (upon initiation of a scram signal, the throttle valves drive to the 96% closed position where they remain for ~30 seconds and then drive to the 98% closed position). Following this test, another visual inspection was made of the valve stems and the packing glands.

"Results:

1. None of the valves failed to open or close.
2. A comparison of the throttling valves, stems and packing glands showed no visible damage occurred during this test. However, some sodium was observed on the valve stems when the valves were in the full open position.
3. The throttle valves drive from full open to the 96% closed position in 15 seconds.
4. Primary throttle valve No. 1 had slowed down noticeably by the end of the test. An investigation revealed that the cross-arm, attached to the valve stem, had bent slightly (~0.040") and was causing binding in the jack screws. The cross-arm was straightened and stiffened with a steel channel. A like modification had previously been accomplished on primary throttle valve No. 3. As a result of this test, modification will be performed on the four remaining throttle valves.
5. Repeatability of the throttle valve positions (open-close) is shown in Tables I and II.

"Conclusion:

"The throttle valves can be expected to operate properly for an extended period of time. "

(Signed)
J. D. Berg
Test Engineer

TABLE I

Run No.	Primary Throttle Valves					
	No. 1		No. 2		No. 3	
	open	closed	open	closed	open	closed
1	9 - 1/8	21 - 3/8	5	17 - 1/4	4 - 1/2	16 - 5/8
2	9 - 1/8	21 - 1/4	5	17	4 - 1/2	16 - 7/8
3	9 - 1/8	21 - 1/2	5	17 - 1/8	4 - 1/2	16 - 3/4
4	9 - 1/8	21 - 3/8	5	17	4 - 1/2	16 - 7/8
5	9 - 1/8	21 - 1/2	5	17	4 - 1/2	16 - 1/2
6	9 - 1/8	21 - 1/2	5	17	4 - 1/2	16 - 5/8
7	9 - 1/8	21 - 3/8	5	17	4 - 1/2	16 - 7/8
8	9 - 1/4	21 - 3/8	5	17	4 - 1/2	16 - 7/8
9	9 - 1/8	21 - 3/8	5	17 - 1/4	4 - 1/2	16 - 3/4
10	9 - 1/8	21 - 3/8	5	17	4 - 1/2	16 - 5/8
11	9 - 1/8	21 - 3/8	5	17	4 - 1/2	16 - 3/4
12	9 - 1/8	21 - 1/2	5	17	4 - 1/2	16 - 7/8
13	9 - 1/8	21 - 1/2	5	17	4 - 1/2	16 - 7/8
14	9 - 1/8	21 - 1/4	5	17	4 - 1/2	16 - 3/4
15	9 - 1/8	21 - 1/4	5	17	4 - 1/2	16 - 3/4
16	9 - 1/8	21 - 3/8	5	17	4 - 1/2	16 - 7/8
17	9 - 1/8	21 - 3/8	5	17	4 - 1/2	16 - 7/8
18	9 - 1/8	21 - 3/8	5	17	4 - 1/2	16 - 3/4
19	9 - 1/8	21 - 3/8	5	17	4 - 1/2	16 - 3/4
20	9 - 1/8	21 - 3/8	5	17	4 - 1/2	16 - 5/8
21	9 - 1/8	21 - 3/8	5	16 - 7/8	4 - 1/2	16 - 7/8
22	9 - 1/8	21 - 1/4	4 - 7/8	16 - 7/8	4 - 1/2	16 - 7/8
23	9 - 1/8	21 - 1/4	5	16 - 7/8	4 - 1/2	16 - 1/2
24	9 - 1/8	21 - 1/4	5	17 - 1/8	4 - 1/2	16 - 7/8
25	9 - 1/8	21 - 3/8	4 - 7/8	16 - 7/8	4 - 1/2	16 - 5/8
26	9 - 1/4	21 - 3/8	5	16 - 7/8	4 - 1/2	16 - 7/8
27	9 - 1/8	21 - 3/8	5	16 - 7/8	4 - 1/2	16 - 7/8
28	9 - 1/8	21 - 1/4	4 - 7/8	16 - 3/4	4 - 1/2	16 - 3/4
29	9 - 1/8	21 - 1/4	5	16 - 7/8	4 - 1/2	16 - 3/4
30	9 - 1/8	21 - 3/8	5	16 - 7/8	4 - 1/2	16 - 3/4
31	9 - 1/8	21 - 3/8	5	16 - 7/8	4 - 1/2	16 - 3/4
32	9 - 1/8	21 - 3/8	5	16 - 7/8	4 - 1/2	16 - 3/8
33	9 - 1/8	21 - 1/4	5	16 - 3/4	4 - 1/2	16 - 3/4
34	9 - 1/8	21 - 1/4	5	16 - 3/4	4 - 1/2	16 - 3/4
35	9 - 1/8	21 - 1/4	5	16 - 7/8	4 - 1/2	16 - 3/4
36	9 - 1/8	21 - 3/8	5	16 - 3/4	4 - 1/2	16 - 3/4
37	9 - 1/8	21 - 1/4	5	16 - 7/8	4 - 1/2	16 - 3/4
38	9 - 1/8	21 - 1/4	5	16 - 7/8	4 - 1/2	16 - 5/8
39	9 - 1/8	21 - 1/4	5	16 - 7/8	4 - 1/2	16 - 3/4
40	9 - 1/8	21 - 3/8	5	16 - 3/4	4 - 1/2	16 - 3/4
41	9 - 1/8	21 - 3/8	5	16 - 3/4	4 - 1/2	16 - 3/4

TABLE II

Run No.	Secondary Throttle Valves			
	No. 1		No. 2	
	open	closed	open	closed
1	4 - 5/8	17 - 1/4	4 - 5/8	17
2	4 - 5/8	17 - 1/4	4 - 5/8	17 - 1/8
3	4 - 5/8	17 - 1/4	4 - 5/8	17
4	4 - 5/8	17 - 1/4	4 - 5/8	17
5	4 - 1/2	17 - 1/8	4 - 7/8	16 - 7/8
6	4 - 1/2	17 - 1/8	4 - 1/2	17
7	4 - 1/2	17 - 1/8	4 - 1/2	17
8	4 - 1/2	17 - 1/8	4 - 1/2	16 - 7/8
9	4 - 1/2	17 - 1/8	4 - 1/2	17
10	4 - 1/2	17 - 1/8	4 - 1/2	16 - 7/8
11	4 - 1/2	17 - 1/8	4 - 1/2	17
12	4 - 1/2	17 - 1/8	4 - 1/2	17
13	4 - 1/2	17 - 1/8	4 - 1/2	17
14	4 - 1/2	17 - 1/8	4 - 1/2	16 - 7/8
15	4 - 1/2	17 - 1/8	4 - 1/2	16 - 7/8
16	4 - 1/2	17 - 1/8	4 - 1/2	17
17	4 - 1/2	17 - 1/8	4 - 1/2	17
18	4 - 3/8	17 - 1/8	4 - 5/8	17
19	4 - 3/8	17	4 - 5/8	17 - 1/8
20	4 - 1/2	17	4 - 1/2	16 - 7/8
21	4 - 1/2	17	4 - 1/2	16 - 7/8
22	4 - 1/2	17	4 - 1/2	16 - 7/8
23	4 - 1/2	17	4 - 1/2	16 - 7/8
24	4 - 1/2	17	4 - 1/2	16 - 3/4
25	4 - 1/2	17	4 - 1/2	16 - 5/8
26	4 - 1/2	17	4 - 1/2	16 - 7/8
27	4 - 1/2	17	4 - 1/2	16 - 3/4
28	4 - 1/2	17	4 - 1/2	16 - 3/4
29	4 - 1/2	17	4 - 1/2	16 - 3/4
30	4 - 1/2	17	4 - 1/2	16 - 3/4
31	4 - 1/2	17	4 - 1/2	16 - 7/8
32	4 - 1/2	17	4 - 1/2	16 - 7/8
33	4 - 1/2	17 - 1/8	4 - 1/2	16 - 3/4

HALLAM NUCLEAR POWER FACILITY
PREOPERATIONAL TEST COMPLETION REPORT
SODIUM SERVICE SYSTEM

AI-P-1104

Test and Analysis

by

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A T O M I C S I N T E R N A T I O N A L

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CONTENTS

	Page
I. Purpose of Test.	1
II. Discussion of Test Performance	3
A. Visual Examination of the System	3
B. Purging of the System	3
C. Filling of System	3
D. System Operational Testing	4
III. Analysis and Presentation of Test Results	7
A. Visual Examination of the System	7
B. Purging of System	7
C. Filling of the System	8
D. System Operational Testing	10
IV. Compliance With Test Completion Criteria	12
V. Evaluation	15
A. Prefill Testing	15
B. Filling	16
C. Sodium Circulation and Component Performance	16

TABLES

I. Results of Plugging Meter Runs Prior to Cold Trapping	11
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FIGURES

1104-1.Sodium Service System	2
1104-2.Solubility of Oxygen and Sodium Oxide in Sodium	6

HALLAM NUCLEAR POWER FACILITY
PREOPERATIONAL TEST COMPLETION REPORT
SODIUM SERVICE SYSTEM, AI-P-1104

I. PURPOSE OF TEST

The purpose of this test was to:

- a) Determine that the sodium service system is adequately prepared for purging, filling, and subsequent circulating operations, and that no distress or binding was caused by preheating.
- b) Purge the sodium service system with helium and verify that the sodium fill tanks and associated purification equipment contain helium atmosphere with less than 1% oxygen concentration.
- c) Fill the sodium fill tanks and verify the operability of the equipment which is used to perform this operation.
- d) Verify that the sodium service system piping can be filled with sodium from the fill tanks, and that sodium can be circulated through the system plugging meters, cold traps, and carbon trap. Also, to determine the operating characteristics of the plugging meters, cold traps, and carbon trap, by putting these components into service, using the service pumps.

The sodium service system (Figure 1104-1) is designed to fill, drain, and flush the sodium heat transfer loops and to purify the sodium. Comprising the system are the sodium melt stations, the primary and secondary fill tanks, the electromagnetic service pumps, the plugging meter and cold trap equipment for sodium purification, and the carbon (hot) trap with sodium sampler. All piping and equipment which come in contact with sodium can be preheated to 350°F by electric resistance heaters on the outside of the piping and equipment. Helium and vent connections are provided to purge the system before admitting sodium.

Figure 1104-1. Sodium Service System

II. DISCUSSION OF TEST PERFORMANCE

A. VISUAL EXAMINATION OF THE SYSTEM

Preheating of the sodium service system (primary and secondary) to 350°F was effected in preparation for an overall visual examination of the system. When this temperature was reached, the necessary access plugs were removed to permit entry into the primary sodium service system cells. Visual inspections were made to check for any piping interferences and/or binding of piping, pipe hangers, and valve operators. All sodium lines were examined, and the system valves were operated to determine whether the correct valve was actuated and that the remote operator was correctly identified.

Preheating of this system was accomplished in accordance with preoperational test procedure AI-P-1116, Preheating System.

B. PURGING OF THE SYSTEM

In preparation for filling the sodium service system with sodium, all system piping and components were purged with helium to reduce the oxygen concentration to approximately 1.0%. In addition, system relative humidity was reduced to 10% or less.

Purging of sodium service system components and piping was accomplished by flowing helium through the system and exhausting all purge gases through the carbon trap sample outlet. Using the normal helium supply to each component as a purge gas source, samples of the exhaust gas were obtained from the carbon trap sample outlet and analyzed with an Orsat analyzer. Moisture content was measured with a portable relative humidity gage. Upon completing the purge of a particular system component, that component was isolated from the system.

C. FILLING OF SYSTEM

Ten tank cars, each containing about 80,000 lb of sodium, were received at the HNPF. A total of 741,500 lb of sodium was used to fill the primary and secondary sodium fill tanks.

These tank cars were individually spotted on the siding next to the sodium melt station. The portable oil heating system was then connected to each tank

car, and hot oil was circulated through the heating pipes to melt the sodium in the car. Melting time of the sodium in any individual tank car ranged from 14 to 30 hr, depending on climatic conditions and malfunctions encountered in the operation of the oil heating unit or the electrical heaters.

The fill tanks and associated piping and valves were electrically preheated to $\sim 350^{\circ}\text{F}$. The transfer line was connected to the tank car, and the transfer of liquid sodium to the fill tanks was effected by using the electromagnetic sodium transfer pumps. The pumps were assisted by a head of nitrogen pressure in the tank car. A period of from 2 to 4 hr was required to empty each tank car.

The primary fill tanks were filled first, then the secondary fill tanks. After the fill tanks were filled, sodium was back-filled into the transfer tank to verify that sodium could be so transferred.

Testing of the drum melt station was effected by simulating the melting of drum loads of sodium. Drums of oil were substituted in the drum melt stations; these drums were heated above the melting point of sodium, and heat transfer data were extrapolated to the melting of sodium.

D. SYSTEM OPERATIONAL TESTING

Testing of the primary and secondary sodium service systems consisted of three basic steps: (1) establishing flow in both sodium service systems; (2) determining the plugging temperature of the sodium prior to cold trapping; and (3) operating the cold traps until the plugging temperature was 255°F or less.

Circulation had been established in the secondary sodium service system, and one plugging run had been made when failure of the secondary sodium service pump interrupted the test. The pump was replaced with an identical pump from the sodium melt station, since that pump was only required during initial fill of the system. At the same time, the power supply for the electromagnetic pumps was modified so that a variable voltage could be supplied. Operation of the pumps was such that the design flow rate could be achieved with considerably less than the design 480 v input. Further, operation at 480 v resulted in high temperatures and low reliability. The test was resumed when these modifications were complete; no further delays were encountered during the remainder of the secondary sodium service system tests.

Initial plugging temperatures of the secondary fill tanks sodium ranged from about 320 to 360°F (see Figure 1104-2 for plugging temperature-to-ppm conversion), with the first tank filled having the highest oxide content, as anticipated. Plugging meter performance was as expected.

The secondary cold trap was placed in service and the plugging temperature of the sodium in the secondary fill tanks was reduced in each tank to approximately 255°F. The cold trap operated properly at design operating conditions and required only 24 hr per tank to reduce the plugging temperature to 255°F.

Testing of the primary sodium service system followed much the same pattern. One of the primary sodium service pumps worked properly, but the other drew excessive current. This problem was corrected and the pump was subsequently tested.

Because of a faulty solenoid, only one of the plugging meters could be used initially. In addition, the brushes for the plugging meter fan motors continually failed. Nitrogen cooling lines were placed over the fan motors to blow a cooling nitrogen atmosphere across the motors. This resulted in elimination of the failure of the motor brushes due to overheating. The solenoid was subsequently repaired and the testing of both plugging meters was completed.

Plugging temperature of the primary fill tanks prior to cold trapping ranged from ~330 to 380°F; somewhat higher than that of the secondary fill tanks. The primary cold traps are identical to the secondary cold trap and operate in a similar manner. The greater volume of the primary fill tanks necessitated about 36 hr of cold trapping in order to reduce the plugging temperature to ~255°F. All cold trapping was done with No. 1 primary cold trap, in order to maintain No. 2 primary cold trap as an unused spare. However, at a later date it became necessary to use primary cold trap No. 2; the test of primary cold trap No. 2 was thus effected.

Flow was established through the carbon trap in order to obtain a sodium sample, but carbon trapping was not attempted at this time, since the sodium as purchased was very low in carbon content and had not been exposed to any carbon source in the sodium service system.

Figure 1104-2. Solubility of Oxygen and Sodium Oxide in Sodium.

III. ANALYSIS AND PRESENTATION OF TEST RESULTS

A. VISUAL EXAMINATION OF THE SYSTEM

During the visual inspection of the sodium service system piping, some irregularities were observed. Discrepancies such as the following were noted:

- 1) Piping insulation interferences were found. In particular, some valve operators were found to be rubbing on piping insulation, and sections of insulation on adjacent piping were found to be in direct contact.
- 2) One valve operator was found rubbing against an electrical junction box.
- 3) The normal aluminum covering was not installed over a short section of pipe insulation.
- 4) One flowmeter lead was in contact with the aluminum cover on a sodium line.
- 5) Several system valves were binding and were extremely difficult to operate.
- 6) The universal connectors between the auxiliary bay floor and valve board were sheared on a few valve operators.
- 7) A number of valves were not in a closed position when the valve board indicators showed a closed position.

Most of the preceding irregularities were of a mechanical nature and subsequent corrections and repairs were made. For example, pot metal universal connectors were replaced with steel connectors, and valve indicators were adjusted to indicate a true closed position.

B. PURGING OF SYSTEM

A purge of all sodium service system piping and components was performed with good results and only a few minor problems. By purging with helium and venting through the carbon trap sample outlet, the oxygen concentration of all system piping and components was lowered to $<1.0\%$. During the purging operation, some R/A vent valve operators were found to be binding and corrective maintenance was necessary.

In the purge of primary cold trap No. 2, it was discovered that the metal gasket for the cold trap flange leaked helium purge gas to the cell atmosphere. Subsequent maintenance work remedied this situation.

During a check of system relative humidity, significant moisture concentration was found in the primary fill tank No.3 and the primary sodium service system drain tank atmosphere. The initial relative humidity for primary fill tank No. 3 was 80% at 80°F. In addition, condensate was discovered in the primary sodium service system drain tank, which is the low point of the sodium service system. In both cases, system preheat and purging lowered the relative humidity to 10% or less.

C. FILLING OF THE SYSTEM

1. Electromagnetic Pump Operation

Performance of the sodium transfer pump used in the unloading operation was deemed satisfactory, in spite of minor trouble encountered at the beginning of the test. At that point, the pump tripped off on high temperature (trip point = 400°F on the 0-400°F temperature controller, TS-14). An investigation was made to determine the temperature limitations of the transfer pump. It was found that the pump could withstand up to 1000°F without damage to the coils or insulation. Therefore, the 0-400°F range controller was exchanged for a 0-1200°F range controller. The high temperature trip point was set for 950°F, and no further trips occurred. Operating temperature of the pump varied between 700 and 800°F during the entire unloading operation. Amperage for the three phases was recorded each time the pump was started, and the pump was "meggered" after each use.

When the first tank car was being unloaded, the transfer pump showed evidence of cavitation. This was rectified by throttling the valves downstream of the pump. The transfer pump used for the sodium unloading operation was later used as a replacement for the secondary sodium service pump and has performed satisfactorily since that time.

2. Valves and Lines

During the unloading operation, some troubles were encountered with stem packing leakage from the valves in the melt station room. As each leak was discovered, leakage was stopped (in most cases), or at least greatly reduced, either by tightening the flange nuts to effect a better packing seal, or by opening the valves all the way to take advantage of back-seating action.

Bread pans containing calcium carbonate were placed under each leak as it occurred and left there after the leak was corrected, in case of recurrence. Less than a gallon of sodium was lost due to leakage during the entire unloading operation.

3. Sodium Melting and Line Preheating

Performance of the oil heating station was slightly below expectations. It was necessary to first bring the tank car up to temperature, then throttle the tank car discharge valves to divert a larger portion of the oil into the oil tracing lines in order to preheat the lines to temperatures necessary for unloading.

Several malfunctions occurred in the oil heating unit during the unloading operation. Most bothersome of the malfunctions was that of the solenoid-operated fuel shutoff valve jamming in a partially open position, allowing fuel to trickle into the burner unit. An explosive mixture of oil fumes would thus accumulate before the igniter plug was energized. This condition was rectified by disassembling the shutoff valve and lapping the valve seat with crocus cloth to effect proper seating. Another malfunction was "coking" of the igniter plug due to nonstoichiometric mixtures of air and fuel. This condition was rectified by trial and error adjustments of the air supply dampers to the burner unit.

Water occasionally accumulated in the heating-oil system, necessitating use of the vent valves in the oil lines to eliminate steam bubbles and prevent flow stoppages.

4. Transfer Tank Operation

The final phase of the sodium unloading operation was a demonstration of the use of the transfer tank in effecting a movement of sodium from the melt station to the secondary fill tanks. This demonstration was accomplished by back-filling the sodium into the transfer tank and then using helium pressure to move the sodium to the secondary fill tank.

Although the demonstration was successful, some difficulty was encountered when the transfer tank was over-filled and sodium spilled over into the vent and helium supply line. The sodium solidified and blocked the helium supply to the transfer tank. This plug was melted out by wrapping heater cable around the vent and helium supply piping, and heating the piping to 250°F.

5. Sodium Drum Melt Station Test

After unloading all the sodium, a test was conducted to determine the effectiveness of the drum melt station by heating drums of Mobiltherm 600 oil. No drums of sodium were available at the site, since drum sodium was not needed at that time.

Heating the first drum from ambient to 300°F took 10 hr and 13 min. The second drum was heated in 6 hr and 18 min, the shorter period being brought about by paralleling some of the heaters that had been in series on drum No. 1. By intercomparison of the various heating properties and characteristics of sodium and Mobiltherm 600, it was calculated that melting of a drum-load of sodium in the melt stations would take about four hr.

D. SYSTEM OPERATIONAL TESTING

1. Electromagnetic Pumps

Difficulties encountered with the sodium service pumps, when operating continuously at a maximum voltage of 480 v, showed the desirability of a variable voltage supply for these pumps. A variable voltage supply was installed; no further difficulties have been encountered with these pumps since this installation.

During the transfer of sodium through the sodium service system, it was shown that the pump controls could reverse the flow in a quick and satisfactory manner.

2. Cold Traps

The cold traps operated according to design conditions and succeeded in reducing the oxide content of the sodium to very low levels in a relatively short period of time. Only a few pounds of sodium oxide were removed from the relatively oxide-free sodium.

3. Plugging Meters

The plugging meters and the associated automatic control system operated satisfactorily. Repeatability of the results was good, especially in view of the fact that many people interpreted the recorder data. Proof of this is shown by the results of the plugging runs made prior to cold trapping, as shown in Table I.

TABLE I
RESULTS OF PLUGGING METER RUNS
PRIOR TO COLD TRAPPING

Tank	Plugging Meter Flow (gpm)	Indicated Plugging Temperature (°F)
No. 1 Secondary	1.20	343
No. 1 Secondary	1.24	345
No. 2 Secondary	0.93	325
No. 2 Secondary	0.96	314
No. 3 Secondary	0.96	354
No. 3 Secondary	0.96	356
No. 1 Primary	0.70	373
No. 1 Primary	0.75	370
No. 1 Primary	0.61	378
No. 2 Primary	0.65	336
No. 2 Primary	0.50	337
No. 3 Primary	0.55	340
No. 3 Primary	0.55	340
No. 4 Primary	0.65	334
No. 4 Primary	0.65	324
No. 4 Primary	0.66	328
No. 5 Primary	0.58	327
No. 5 Primary	0.59	326
No. 5 Primary	0.61	325
No. 5 Primary	0.61	325
No. 5 Primary	0.68	328

4. Carbon Trap

As sodium flow was introduced to the carbon trap, satisfactory freeze trap seals were obtained. No problems were encountered with the system during the establishment of flow through the carbon trap.

In addition to establishing flow through the carbon trap, sodium samples have been removed from the carton trap sampler with no major difficulties.

IV. COMPLIANCE WITH TEST COMPLETION CRITERIA

Following are the test completion criteria associated with preoperational test AI-P-1104, Sodium Service System, as listed in NAA-SR-5700, Supplement 4, pages XI (a)-8 through XI (a)-14, with remarks regarding the test to explain compliance with the criteria.

- a) Criterion: A visual inspection has revealed the absence of breaks in insulation, no binding in piping supports, no interferences exist with valve operators nor reach rods.

Remarks: Inspection of the system piping and valve operators revealed some irregularities which were subsequently corrected. The system was in satisfactory condition for preheating, purging, and admission of sodium upon completion of corrections.

- b) Criterion: All valves can be opened and closed and the proper valve is actuated by the proper valve operator.

Remarks: After adjusting some valve indicators to indicate true closure, all valves were operable and properly designated.

- c) Criterion: Each sodium line and vessel has been purged with helium until analysis of the effluent gases from each zone indicates an oxygen content of <1%.

Remarks: A complete system purge was effected. The purge not only reduced the oxygen content to <1%, but also reduced the system relative humidity to 10% or less.

- d) Criterion: Primary and secondary sodium fill and drain tanks are filled with sodium to prespecified levels.

Remarks: The following tabulation lists the final levels of the fill tanks:

- 1) Primary fill tank No. 1 - 122 in.
- 2) Primary fill tank No. 2 - 122 in.
- 3) Primary fill tank No. 3 - 104 in.
- 4) Primary fill tank No. 4 - 122 in.
- 5) Primary fill tank No. 5 - 122 in.

- 6) Secondary fill tank No. 1 - 99 in.
- 7) Secondary fill tank No. 2 - 108 in.
- 8) Secondary fill tank No. 3 - 99 in.

All these levels were between the high and low alarm setpoints.

- e) Criterion: Sodium service system piping components necessary for the fill function properly.

Remarks: Although some sodium valves leaked slightly until corrected, piping components necessary for the fill functioned satisfactorily.

- f) Criterion: There are no sodium leaks in the affected portions of the service system.

Remarks: Sodium leaks did occur through gate valves between the tank car and sodium filters; however, these leaks were corrected by either back seating the valves or tightening the flange nuts. No sodium leaks existed in the balance of the system.

- g) Criterion: Simulation of melting drums of sodium in the sodium drum melt station has been accomplished, verifying proper operability of that unit.

Remarks: Simulation of melting drums of sodium in the melt station was performed by heating drums of oil. This operation verified proper operability of the units and indicated a drum of sodium could be melted in about four hours.

- h) Criterion: Sodium can be transferred to the service system from the sodium transfer tank.

Remarks: Sodium was successfully transferred from the sodium transfer tank to the secondary fill tanks, using helium pressure.

- i) Criterion: There are no interferences due to expansion on loading the piping and equipment, and no discontinuities in insulation have developed.

Remarks: No interferences due to expansion on loading the piping and equipment nor discontinuities in the insulation were observed.

- j) Criterion: The primary and secondary sodium service systems have been filled with sodium and the service systems pumps used to circulate sodium through the plugging meters, cold traps, and the primary carbon trap without malfunction.

Remarks: Sodium was circulated throughout the primary and secondary sodium service system without incident.

- k) Criterion: Sodium service pumps operate at design conditions for extended periods of time.

Remarks: Although these pumps would not operate reliably at all time when operating with maximum voltage, succeeding operation with a variable voltage supply was reliable and satisfactory over extended operating periods.

- l) Criterion: Cold traps have been operated for extended periods of time to verify operability of the controls, and to purify sodium to a plugging temperature of less than 300°F, prior to filling the heat transfer systems.

Remarks: Operation over extended periods of time has exhibited that the cold traps will operate satisfactorily. Cold traps performed according to design conditions and purified the sodium to a plugging temperature of approximately 255 F prior to filling the heat transfer system.

- m) Criterion: The carbon trap system has been demonstrated operable.

Remarks: Sodium has been flowed through the carbon trap system and the system components have been checked for satisfactory performance. Since the sodium, as purchased, was very low in carbon content, carbon trapping was not attempted.

- n) Criterion: Plugging meters have operated for extended periods of time. Their capability for measuring a plugging temperature, and for repeatability over extended runs has been demonstrated.

Remarks: Plugging meters have operated for extended periods of time in a satisfactory manner. Although the solenoid unit and the fan motors malfunctioned and were subsequently repaired, the plugging meters have exhibited capability to measure a plugging temperature with repeatability.

V. EVALUATION

The sodium service system is designed to melt, store, and purify the heat transfer system sodium. This includes facilities for the individual filling and draining of the reactor and heat transfer loops, plus continuous purification of the sodium during plant shutdown and normal operation. Preoperational testing and subsequent operation of the system provides assurance that these design criteria were accomplished. The test purpose was accomplished and the test completion criteria as specified in NAA-SR-5700, Final Summary Safeguards Report for the Hallam Nuclear Power Facility, were met.

The problems encountered during the test were minor and of the type normally associated with the initial operation of any process system. These included such items as: (a) piping interferences; (b) improperly installed and/or adjusted valve operators; (c) electrical malfunctions of the service pumps; and (d) miscellaneous component failures on the tank car heating system. In addition, some difficulties of an operational nature were experienced. These occurred during the initial system operation before the individual operating characteristics of the equipment were known. For example, premature plugging of the cold traps resulted before operating techniques were perfected. Once the operating peculiarities of the equipment were recognized, the performance of the cold traps was excellent. In general, this condition existed throughout the entire system and, as in the case of the cold traps, the system performance was considered adequate after familiarity with the system was fully established.

There are some inherent characteristics of the system which present some limitations (discussed in following sections) in operational flexibility. Correction of these conditions would be very desirable and would improve the operating efficiency. However, the system is presently operable and will perform the intended function.

A. PREFILL TESTING

The test verified that the system was adequately prepared for filling. Satisfactory preheat conditions were established and the oxygen content of the system was reduced, by means of a helium purge, to less than 1.0%.

All system discrepancies found during the visual inspection were corrected. Piping interferences were eliminated, thus assuring freedom of piping and equipment movement during elevated operating temperature conditions. Valve operators were adjusted to eliminate all binding and satisfactory valve operation was confirmed by manipulation of the valves over the full operating range. Valve position indicators were adjusted and proper remote operator designations were verified.

B. FILLING

The sodium tank car heating system was a temporary installation and was only used to introduce the initial charge of sodium to the system. Although some problems were encountered, the system did prove adequate to accomplish the necessary operation.

Proper performance of the drum melt station was demonstrated during the test by heating drums of oil. After the preoperational test was completed, drum loads of sodium were actually melted and introduced to the systems. The sodium melting time of 2 to 4 hr was greater than expected (designed to melt a drum in 1 hr) but, the additional time is not considered excessive due to the infrequent service. The drum melt station will only be used occasionally to introduce a few drums of sodium to compensate for the small quantity of sodium lost as a result of cold trap and carbon trap replacements.

C. SODIUM CIRCULATION AND COMPONENT PERFORMANCE

Sodium circulation was established in the service system with essentially no trouble. The operability of the purification equipment and associated components was determined during the test and subsequent operation. Certain conditions were encountered, however, which pose limitations in operational flexibility as discussed below.

1) Low Pressure Differential at Reduced Plant Load Conditions

Efficient operation of the purification equipment is limited to the higher plant load conditions. At power levels below approximately 30% there is not sufficient pressure differential across the heat transfer system pumps to supply adequate flow to the service system. In the secondary systems it is possible to throttle the convection control valve to provide the required pressure differential. However,

this is a delicate operation which can result in an inadvertent reactor scram due to an upset in flow ratio between primary and secondary loops.

2) Leakage of Radioactive Sodium Across the Reactor Fill and Drain Line Isolation Valves

Access to the primary service cells will be prevented when the primary sodium is activated. This is due to constant leakage of radioactive sodium across the reactor fill and drain line isolation valves into the service system.

Operating experience with sodium valves seems to indicate that seat leakage is the rule rather than the exception. Normally, if absolute isolation of a system or component is required, the isolation valves are closed and the sodium allowed to solidify, forming a frozen sodium seal in the associated piping. In the case of the primary sodium service system, this is not possible since the reactor fill and drain line must be maintained in a heated condition to insure that sodium can be transferred to the reactor from primary fill tank No. 3 in the event of a core tank failure.

It is apparent that future access to the primary service areas will always be restricted during reactor operation. Maintenance of the service pumps and bellows sealed valves in the primary sodium service system will be limited to periods when the reactor is shut down.

3) Susceptibility of Bellows Seal Valves to Failure

Extensive throttling of bellows seal valves has resulted in bellows failures. From all indications the seal bellows oscillates under this condition and eventually fails due to fatigue. The oscillation is of such magnitude that the vibration is audible in the vicinity of the valve when the pressure differential across the valve is high.

Since the cause of failure is now recognized, operating personnel have exercised extreme care in the operation of the bellows seal valves. The use of the valves for throttling has been limited when possible, and has resulted in less frequent failures.

1. Cold Traps

The cold traps successfully reduced the oxide content of the sodium in the fill tanks from 60-80 ppm to less than 20 ppm. This operation required an average of only 24 hr for each secondary tank and 36 hr for each primary, which is indicative of the high cold trap efficiency.

The quantity of oxide removed by the trap during this operation is estimated to be approximately 8 lb for the secondary and 28 lb for the primary system. After the heat transfer system was filled, the traps were utilized for initial system cleanup. During this period the oxide content of the system was reduced from a saturated condition at 540°F to a concentration corresponding to a plugging temperature of less than 250°F. In the process, primary cold trap No. 1 was filled to capacity which was estimated to be at least 232 lb of sodium oxide (Na_2O). This estimate is considered conservative since it is based on periodically measured plugging temperatures and doesn't take into account the quantity of oxide added to the system during the cold trapping period. Therefore, it is reasonable to assume that the actual capacity of the traps approaches, if not exceeds, the design capacity of 250 lb.

Initially some difficulty was experienced with localized plugging in the traps when the oxide concentration of the system was high and the trap was operated with a large temperature gradient. It was found that by limiting the minimum internal temperature of the trap to a maximum of 50°F below the plugging temperature no problems with plugging was experienced. This mode of operation has been adapted as standard practice and from all indications will eliminate the problem in the future.

Although the remote removal of the primary cold traps was not included as a part of the preoperational test it is considered sufficiently important to be considered in this evaluation. During removal of the primary cold traps to date, it has been necessary to enter the cell in order to disconnect the trap. The only difficulties encountered were related to binding of the disconnect mechanism. Minor adjustments or slight modifications should be sufficient to correct this problem.

It is considered extremely important that these difficulties be resolved and the ability to remotely remove the trap be demonstrated at an early date.

In the future, it is conceivable that the cold trap will contain long lived fission and/or activated corrosion products that will restrict access for removal.

2. Plugging Meter

Performance of the plugging meters is considered excellent; the results appear to be consistent and of sufficient sensitivity. Automatic operation has been satisfactory except that, when the oxide saturation temperature approaches the bulk sodium temperature, the plugging meter does not flush properly. Among corrective actions considered was the installation of a heater to increase the temperature of the sodium flowing through the plugging meter above the bulk temperature. This would provide better sensitivity at the high saturation conditions and eliminate the problem with flushing. Such a modification would be desirable, but it is not essential, particularly since during future operation it is expected that the oxide concentration will be low.

3. Valve Operators

Much difficulty has been experienced with valve operator binding and failure. For the most part the operators associated with the freeze stem valves were under-designed. This was largely due to a greater than anticipated valve operating torque resulting from build up of oxide deposits in the stem seal. Improved performance of the operator was effected by installing heavier duty universal joints, shear pins, and relieving binding and alignment problems. This corrective action provided more reliable operation, but excessive force was still required to operate the valves.

Subsequently, torque multipliers were installed at each freeze seal valve equipped with a remote operator. Since this modification was incorporated, no difficulty with the remote operation of the sodium service valves has been experienced.

4. Freeze Traps

Freeze traps have been a continuing source of operational delay. The accumulation of sodium oxide in the freeze seal annulus hampers and in some cases prevents opening the trap for purging or venting. Even when it is possible to "melt out" the seal the operation usually requires 4 to 6 hr time. During this period it is necessary to supply helium to force the molten sodium back into the system. In some cases the packed-stem seal leakage has been such

that sufficient helium pressure can not be maintained, resulting in sodium entering and solidifying in the helium and vent lines.

Even though performance of the freeze traps to date has been inadequate, it is expected that future operation will be more satisfactory. The oxide content of the heat transfer systems was extremely high during initial operation, providing the potential for greater and more rapid diffusion of oxide to the freeze traps. During future operation the oxide content will be maintained at very low concentrations. In addition, the necessity for using the freeze traps will be limited to the very rare occasions when the plant is shut down. Additional time to open a freeze trap will be an operational inconvenience, but will probably not have a significant effect on the overall operation.

AI-P-1106

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CONTENTS

	Page
I. Purpose of Test	1
II. Discussion of Test Performance	1
A. System Description.	1
B. Closed Loop Cooling Water Subsystem	4
C. Open Loop Cooling Water Subsystem	5
III. Analysis and Presentation of Test Results	9
A. Closed Loop Cooling Water Subsystem	9
B. Open Loop Cooling Water Subsystem	17
IV. Compliance with Test Completion Criteria	18
A. Closed Loop Cooling Water Subsystem	18
B. Open Loop Cooling Water Subsystem	21
V. Evaluation.	22
A. System Filling	23
B. Core Cavity Cooling Coils	24

TABLES

I. Condensate Supply Level Setpoints	10
II. Flows and Equipment	12
III. System Operation Under Normal Service Conditions.	15
IV. System Operation Requirements	16

FIGURES

1106-1. Closed Loop Cooling Water System	2
1106-2. Open Loop Cooling Water System for Nitrogen Coolers	3
1106-3. Cooling Water Diagram for Typical N ₂ Cooler with Standby Cooler	6
1106-4. Condensate Supply Head Tank (Level vs Quantity of Water)	11
1106-5. Pump Performance Data	14
1106-6. Core Cavity Cooling Circuits with Leak Regions Spotted	27

HALLAM NUCLEAR POWER FACILITY
PREOPERATIONAL TEST COMPLETION REPORT
COOLING WATER SYSTEM, AI-P-1106

I. PURPOSE OF TEST

The purpose of the cooling water test was to:

- a) Fill spent fuel storage pit No. 3 with demineralized water and to determine whether the portion of the system from the condensate supply head tank to the storage pit will operate satisfactorily.
- b) Establish proper system water pH, and to determine whether the closed loop system serving the reactor pipe chase and pipe gallery cooling coils, and the coolers associated with the R/A liquid waste and vent systems will operate satisfactorily.
- c) Determine whether the portion of the cooling water system serving the reactor building heating and ventilating coolers, nitrogen coolers and closed system heat exchangers will operate satisfactorily (open loop system).

II. DISCUSSION OF TEST PERFORMANCE

A. SYSTEM DESCRIPTION

The cooling water system (Figures 1106-1 and -2) provides auxiliary cooling for those components and areas in which there is no potential danger of contact between the water and sodium. This system is comprised of two separate subsystems, the closed loop and the open loop. The closed loop services equipment areas where there is a radioactive contamination potential, whereas, the open loop services the cell atmospheres and the closed loop heat exchangers, where there is essentially no possibility of contamination.

The closed loop includes two pumps of 300 gpm capacity (each), and two shell-and-tube, water-to-water heat exchangers. The two pumps are connected in parallel, with either pump having sufficient capacity to handle the normal system operational requirements. The closed loop pump motors are connected

Figure 1106-1. Closed Loop Cooling Water System

Figure 1106-2. Open Loop Cooling Water System for Nitrogen Coolers

to the emergency power system to ensure continuity of cooling in event of loss of electrical power supply. The heat exchangers are also connected in parallel; each has a heat dissipation capacity of 4.2×10^6 Btu/hr at design cooling tower conditions. In normal operation, water is circulated from fuel storage pit No.3, cooled in the heat exchangers, circulated through the various system components, and returned to the fuel storage cell pit.

The open loop includes two 1600 gpm pumps; one pump is for normal operation, and one is a spare. This loop supplies coolant water for the nitrogen coolers which cool the cell atmospheres, the closed loop heat exchangers, and the loading face shield coolant heat exchangers. Heat is dissipated to the atmosphere in the auxiliary cooling tower. An auxiliary supply line from a head tank is provided to service the auxiliary cooling tower water system in event of pump failure, or complete loss of power. The auxiliary cooling tower pump motors are connected to the emergency power source to ensure continuity of operation in the event of a loss of normal power.

B. CLOSED LOOP COOLING WATER SUBSYSTEM

1. Filling Spent Fuel Storage Pit No. 3

Condensate was admitted to the condensate supply pump from the CPPD condensate storage tanks. The pump was started manually and the condensate head tank filled. The water level was varied to test head tank level switches and alarms for proper operation. Pump capacity was determined by timing the fill of the condensate supply head tank.

The storage pit was filled to the high alarm level and the pit level instrumentation was verified to be functioning properly. Water samples were taken to determine pH and conductivity.

2. Initiation of Flow

Water was admitted to the closed loop cooling water pumps from storage pit No. 3. Pump No. 1 (P-603) was started manually, and water was circulated through one of the system heat exchangers and back into the storage pit.

Flow was established to the cavity cooling coils and each sight-glass flow indicator was observed for flow. System instrumentation and temporarily installed instruments were observed and data recorded to determine system flows, pressures, and pressure drops. Each pump was stopped with the second

pump on standby, to test the automatic start-on-loss-of-discharge header pressure feature. The annunciators associated with these operations were observed for proper functioning. The demineralizers and filters provided as a part of the system were put in service and flows adjusted to the proper values.

Valves in lines to the radioactive liquid waste system coolers were opened and flow established. Equipment and piping high points associated with the coolers were vented. Flows were established through the coolers and the entire closed loop cooling water system was tested.

Tests were conducted on the two pumps to determine the head-capacity performance. The high temperature alarm associated with the system heat exchangers effluent was tested. The wash cells cooling loops were isolated and blown dry of water with nitrogen. The system was left in service at test termination, supplying cooling water to the reactor cavity cooling coils and to the components in the radioactive waste facility. The closed loop water was sampled and analyzed to determine if corrosion inhibitor was required to obtain proper pH.

C. OPEN LOOP COOLING WATER SUBSYSTEM

1. System Fill

That portion of the test requiring filling of the system and placing the pumps in service (one operating, one standby) was not required as this had been previously accomplished.

2. System Testing

Flow was established through each piece of equipment and adjusted according to design requirements. For purposes of this test, instrument air supply pressure to all of the N₂ cooler units temperature controllers was reduced to zero, thus diverting flow through the cooler coils rather than through the bypass. The throttle valves were then adjusted to achieve the required flow balance between the individual N₂ cooler units. Water was next routed through the standby cooler from the normally operating cooler water supply line by making use of a crossover piping and valve arrangement (see Figure 1106-3).

In order to route the cooling water through the normally operating cooler, it was necessary to temporarily reduce the instrument air supply pressure to all nitrogen cooler unit temperature controllers to zero, thus diverting flow through the cooler coils rather than through the bypass. The throttle valves

Figure 1106-3. Cooling Water Diagram For Typical N_2 Cooler with Standby Cooler

were then adjusted to balance cooling water flow through the cooler coils. Water was routed through the standby cooler from the normally operating cooler water supply line by making use of a crossover piping and valve arrangement. The system is arranged so the two coolers can be switched in function and still make use of the same temperature control system. This makes it possible to equalize total duty of the two coolers and recirculation fans. With the crossover system in operation, the throttle valve in the standby cooler could be adjusted to provide the required maximum cooling water flow through the cooler coils.

When a loading face shield coolant compressor or H & V exhaust fan is not in operation, cooling water supply to the units is blocked by solenoid valves in the supply line. When the units are started, a signal is transmitted to the solenoid which opens the valve. Hence, cooling water flow through the compressor water jacket and the H & V exhaust fan coupling cooler was adjusted while the units were in operation.

3. Double Cooler Units

Instrument air pressure was set at 20 psig on the temperature indicator-controller. It was observed that the control valve closed and the indicated flow on the normally operating cooler coil was zero. The sensing element for the temperature indicator-controller was removed from the nitrogen inlet duct and placed in a controlled temperature water bath. Temperature of the water bath was slowly raised until the control valve was fully open. Necessary adjustments were made to the temperature indicator-controller setpoint and proportional band so that indicated temperature on the temperature indicator-controller agreed with measured temperature of the water bath. It was also noted that, at 130 F, flow through the cooler coil matched previously adjusted flow. Heat was removed from the sensing element and it was noted that the control valve closed and that flow through the cooler coil returned to zero.

The temperature switch sensing element was removed from the nitrogen inlet duct and placed in a water bath. Temperature of the bath was raised until the temperature switch energized the solenoid and thus opened the valve to the standby cooler coil. The temperature switch was adjusted so the valve would be opened when measured temperature of the water bath reached 150 F. It was noted that flow through the standby cooler matched previously adjusted flow set by the throttle valve. Heat was removed from the sensing element and it was

noted that the solenoid valve closed and that flow through the standby cooler coil returned to zero. When temperature was raised to 150 F, the temperature switch also provided a signal to the standby cooler fan controls (which were set on "Standby") which automatically turned on the standby cooler recirculation fan.

4. Single Cooler Unit

The single cooler unit was tested in exactly the same manner as the normally operating cooler on the two-cooler unit; i.e., the temperature indicator-controller sensing element was removed from the nitrogen duct and heated in a water bath until the control valve opened. The temperature indicator-controller was adjusted as necessary so the control valve was full open when the sensing element saw 130 F. It was verified that the control valve closed properly and that flow through the cooler was as required while the system was on automatic control.

5. Cooler Room Moisture Switches

The moisture switches in the intermediate heat exchanger (IHX) cell nitrogen unit cooler rooms and in the room housing H & V cooling unit No. 10 (IHX cell N₂ cooler area) were actuated by submerging the sensing element in water. The appropriate control valves in the water supply line to the coolers closed and the associated visual and audible alarms in the main control room were energized. When the sensing element was wiped dry, the solenoid valves reset and the control valve opened.

6. Testing Pressure Switch on Pump Discharge

Cooling tower water pump discharge pressure switch (PS-600), and low discharge pressure alarm (PAL-600) were set at 25 psig. Cooling tower water pump P-601 was running and pump P-602 was on standby. Pump P-601 discharge valve was slowly closed until pump discharge pressure indicator PI-601 indicated 25 psig. The "Low Pressure Tower Pump" annunciator on the main board was energized by PAL-600. PS-600 subsequently transmitted a signal to P-602 pump controls which started the pump automatically. When P-602 started, the "Auto-Start Tower Pump" on the main board was also energized. Discharge valve V-604 was then manually opened (full) and pump P-601 was stopped. The annunciator board was cleared and pump P-601 was placed on standby. Pump P-602 discharge valve was then slowly closed until PI-601 indicated 25 psig. It was noted that the annunciators and windows on the main board were energized

as before and that P-601 started automatically. The discharge valve was then opened full; pump P-601 was stopped and placed on standby with pump P-602 still running.

Following the pressure switch test, the 440 V breaker to pump P-602 was manually tripped and it was verified that:

- a) The "Trip Tower Pump" annunciator alarmed and the associated window light on the main board was energized.
- b) The "Low Pressure Tower Pump" annunciator window light on the main board was energized.
- c) The "Auto Start Tower Pump" annunciator window light on the main board was energized.
- d) Pump P-601 started.

The breaker to P-602 was closed and the pump was placed on standby. This completed the test on the open loop cooling water system.

III. ANALYSIS AND PRESENTATION OF TEST RESULTS

A. CLOSED LOOP COOLING WATER SUBSYSTEM

1. Condensate Supply

The condensate supply pump manual start control operated properly when employed for initial starting of the pump. Tank level instrumentation setpoints are given in Table I. The pump auto start and stop controls functioned as required.

With the normal operating water level in the head tank established between 43.5 inches and 45.5 inches, the corresponding pressure as indicated on the HNPF supply header pressure gage, PI-1114, was 22.7 psig. During future operation, deviation from this value will be indicative of an off-normal level condition in the head tank.

Condensate supply pump capacity was established to be 41 gpm, by measuring the time (2.93 minutes) required to fill the condensate head tank from 33.5 inches to 45.5 inches. At this flow rate, the pump head was determined

to be 26 psig (33 psig discharge, 7 psig suction). Condensate tank and pump operating data are illustrated in Figure 1106-4.

TABLE I
CONDENSATE SUPPLY LEVEL SETPOINTS

Instrument	Function	Setpoint	Reset
LS-1126-B	Low level alarm	33.5 inches	35.5 inches
LS-1127-A	Pump auto stop	45.5 inches	-
LS-1126-A	High level alarm	49.5 inches	47.0 inches
LS-1127-B	Pump auto start	43.5 inches	-

2. Fill of Storage Pit No. 3

Storage pit No. 3 was filled at a rate of approximately 47 gpm. Filling was not a continuous operation due to the condensate supply being limited to 8,000 to 10,000 gallons of condensate maximum (at any one time) by the CPPD conventional plant demineralizer train capacity. The 47 gpm was the maximum gravity flow rate obtainable from the head tank to pit No. 3 (all valves open to pit No. 3). The condensate supply pump could maintain a level of only ~1 to 2 inches in the head tank under these conditions.

The pit was filled until the high level alarm actuated (36 feet 6 inches). In the process of filling the pit the low level alarm reset at 35 feet 6 inches, as indicated by LI-602. The alarm points, as indicated on LI-602, were 2 inches higher than anticipated in each case. This can be attributed to the pit level transmitter calibration and slight discrepancy in the instrumentation installation.

Water in the pit was sampled following the filling operations and analyzed for pH and conductivity. It was planned to add corrosion inhibitor (lithium hydroxide) to the water to bring the pH to 9.0 to 10.0 as a part of this test. However, the analysis indicated that the pH of the water was already 9.5, therefore establishing that no corrosion inhibitor was required.

3. Heat Exchanger Piping System Fill

The heat exchanger portion of the system was filled with water and temperature indicator TI-619 element placed in a constant temperature bath. The associated high temperature alarm was determined to actuate at 105 F and reset

Figure 1106-4. Condensate Supply Head Tank
(Level vs Quantity of Water)

at 104 F. Proper functioning of the main board annunciator, "High Temp Treated Water Pit HX Outlet", was also determined.

4. Operational Testing

Flow was established through the cavity liner cooling coils with one heat exchanger and one pump in service per normal operation. Water was observed to be flowing through all of the cooling coil sight-glasses.

Data obtained during initial testing of the system disclosed that the two closed loop cooling-water pumps were incapable of meeting design flow requirements. The pumps were operated in parallel with flow through both heat exchangers. Minor modifications effected in an attempt to reduce system pressure drop to a minimum, thereby increasing pump capacity, were unsuccessful. An evaluation of maximum total system flow requirements was conducted. The existence of adequate flow to the various serviced equipment was also determined. As a result of this study, it was resolved that maximum flow requirements exist during a fuel wash cycle utilizing one wash cell and accessory equipment, with design flow established to the reactor cavity. Table II lists the equipment and related flows which exist under the above described conditions.

TABLE II
FLOWS AND EQUIPMENT

Unit	Flow (gpm)
Cavity liner	191.0
Cooling one wash cell	10.0
Vent condenser	2.0
R/A waste transfer tank	54.8
R/A vent compressor No. 1 or No. 2	3.5
R/A vent compressor No. 3 or No. 4	2.0
Wash cell compressor seal water cooler	10.0
Wash cell compressor seal oil cooler	1.0
Total Flow Required	274.3

Each of the two pumps, direct-coupled to a 5 hp motor, were rated at 200 gpm with a 50 ft total dynamic head. After a thorough review of total system requirements, with due consideration being given to economical replacement, these pumps were replaced with pumps rated at 300 gpm at 110 ft head, direct-coupled to a 15 hp, 3600 rpm motor. Pump performance data were obtained for the replacement pumps by throttling the pump discharge valve. Data recorded are shown in Figure 1106-5.

Design flow was established to all normally operating equipment, exclusive of additional equipment required during a fuel wash cycle. This includes the reactor cavity liner cooling coils, demineralizers, filters, R/A vent compressors and the evaporator condenser (other than reactor cavity usage will normally be on an intermittent basis). The data shown in Table III were obtained with one pump and one heat exchanger in service.

The total system flow requirement was established to the reactor cavity cooling loops, equipment associated with the fuel wash cell cycle (utilizing one fuel wash cell), the R/A waste facility, and the demineralizers. The data obtained are tabulated in Table IV. The data were obtained with one pump in operation and circulation through one heat exchanger. It was determined that the present pumps have sufficient capacity and discharge head to exceed revised minimum system requirements.

Manual stopping of either pump, by means of local station controls, resulted in the following:

- a) "Low Pressure Treated Water Pump Discharge," main board alarm annunciates when header pressure drops to 35 psig.
- b) Auto start of the second pump (if it is on standby) when the discharge pressure of the pump in service drops to 30 psig.
- c) "Auto Start Treated Water Pump," main board alarm annunciates.
- d) The low pressure alarm clears when the standby pump comes on.

Figure 1106-5. Pump Performance Data

TABLE III
SYSTEM OPERATION UNDER NORMAL SERVICE CONDITIONS

Item (Service)	Instrument	Pump 603	Pump 604
Pump 603 discharge pressure	PP-603	53 psig	0 psig
Pump 604 discharge pressure	PP-604	0 psig	52.8 psig
No. 1 heat exchanger inlet*	PP-609	51 psig	51 psig
No. 1 heat exchanger outlet*	PP-611	42 psig	42 psig
No. 2 heat exchanger inlet*	PP-610	52.5 psig	52 psig
No. 2 heat exchanger outlet*	PP-612	45 psig	45 psig
Cavity liner flow	FI-626	191 gpm	191 gpm
Cavity liner supply pressure	PI-614	39.8 psig	39.7 psig
R/A vent compressors flow	†	9.0 gpm	9.0 gpm
Evaporator-condenser flow	†	4.4 gpm	4.4 gpm
Demineralizer No. 1 flow	FI-647	8.0 gpm	8.0 gpm
Demineralizer No. 2 flow	FI-648	2.0 gpm	2.0 gpm
Fuel storage filter No. 1 pressure in	PI-617	40 psig	40 psig
Fuel storage filter No. 1 pressure out	PI-618	37.8 psig	37.8 psig
Fuel storage filter No. 2 pressure in	PI-619	27 psig	27 psig
Fuel storage filter No. 2 pressure out	PI-620	13.5 psig	13.3 psig
Motor current		12.4 amp	13.7 amp
Total flow		214.4 gpm	214.4 gpm

*System operated and data obtained with one heat exchanger on line at a time.

†Design flow — no flow meter associated with equipment.

TABLE IV
SYSTEM OPERATION REQUIREMENTS

Item (Service)	Instrument	Pump 603	Pump 604
Pump 603 discharge pressure	PP-603	45.5 psig	0 psig
Pump 604 discharge pressure	PP-604	0 psig	46.0 psig
No. 1 heat exchanger inlet*	PP-609	45.0 psig	45.0 psig
No. 1 heat exchanger outlet*	PP-611	27.0 psig	27.0 psig
No. 2 heat exchanger inlet*	PP-610	46.0 psig	46.0 psig
No. 2 heat exchanger outlet*	PP-612	27.0 psig	27.0 psig
Cavity liner flow	FI-626	191.0 gpm	191.0 gpm
Cavity liner supply pressure	PI-614	22.0 psig	22.3 psig
Fuel wash cell No. 3 flow	FI-622	11.0 gpm	11.0 gpm
Vent condenser flow	FI-634	2.0 gpm	2.0 gpm
R/A waste transfer tank flow	FI-633	54.8 gpm	54.8 gpm
R/A vent compressors 1,3, and 4	†	5.5 gpm	5.5 gpm
Wash cell vent compr. seal H ₂ O cooler	†	12.0 gpm	12.0 gpm
Wash cell vent compr. seal oil cooler	†	1.0 gpm	1.0 gpm
Evaporator-condenser	†	4.0 gpm	4.0 gpm
Total flow — fuel wash cycle and R/A facility	FI-619	90.5 gpm	90.5 gpm
R/A header pressure	Test gage	21.0 psig	21.0 psig
Demineralizer No. 1 flow	FI-647	8.0 gpm	8.0 gpm
Demineralizer No. 2 flow	FI-648	2.0 gpm	2.0 gpm
Motor current		14.0 amp	15.0 amp
Total system flow (sum of FI-626, -619, -647, and -648)		291.5 gpm	291.5 gpm

*System operated and data obtained with one heat exchanger on line at a time.

†Design flow — no flow meter associated with equipment. Flow determined by valving in equipment and observing flow increase on FI-619.

B. OPEN LOOP COOLING WATER SUBSYSTEM

1. System Fill

As stated in II-C. 1., the system was already filled and pump P-601 was in operation at the time this test was started. With all throttle valves wide open, indicated pump discharge pressure and total flow was approximately 45 psig and 1825 gpm, respectively. This total flow exceeds design requirements by approximately 6%.

2. System Testing

Throttle valve adjustments were made to establish design flow conditions to each of the serviced components. At the completion of this operation, the total computed flow was 1725 gpm. The total indicated flow was 1765 gpm on the main flow indicator, FI-627. Indicated pump discharge pressure, after throttle valve adjustments were made, was 48 psig. The pump appeared to be running easily with no indication of motor overload or overheating.

During the test it was discovered that the supply piping and valves to one of the two-cooler units were piped wrong and that the standby cooler water supply line was routed to the normally operating cooler and vice versa. The piping was rerouted according to the drawing and tested satisfactorily.

3. Double Cooler Unit Test

The temperature indicator-controllers and temperature switches were set and the double units tested. Under various test conditions, these units and related components function satisfactorily. The associated control valve ran full open at 130 F and the standby unit was automatically energized at 150 F.

4. Single Cooler Unit Test

The temperature indicator-controllers and control valves operated satisfactorily as designed for the system. The associated control valve was full open at 130 F.

5. Cooler Room Moisture Switch Test

Performance of the moisture switches was as per design, with proper closing of the cooling water supply header upon actuation of the header control valve through the associated solenoid valve. The manual reset on the solenoid valve operated smoothly with no failures.

6. Testing Pressure Switch on Pump Discharge

The pump controls, low pressure switch, low pressure alarm, main board annunciators and windows all worked perfectly according to drawings and specifications.

IV. COMPLIANCE WITH TEST COMPLETION CRITERIA

Following are the test completion criteria associated with preoperational test AI-P-1106, Cooling Water System, as listed in NAA-SR-5700 (Supplement 4), pages XI(a)-15 through XI(a)-21, with remarks regarding the test to explain compliance with the criteria.

A. CLOSED LOOP COOLING WATER SUBSYSTEM

1) Criterion: The condensate supply pump will:

- a) Provide about 25 gpm flow under a 70 ft head
- b) Start or stop automatically upon receipt of a signal from the condensate supply head tank low or high level switches.

Remarks: Proper operation of the condensate supply pump was verified by the test as described within this report.

2) Criterion: Low level switch on condensate supply head tank will actuate pump starter when low level set point is reached and low level alarm will annunciate.

Remarks: These operations were verified as described within this report.

3) Criterion: High level switch on condensate supply head tank will stop pump when high level set point is reached and high level alarm will annunciate.

Remarks: These operations were verified as described within this report.

4) Criterion: Water will flow freely from storage tank to fuel storage pit No. 3 at rate of approximately 25 gpm.

Remarks: Flow rate of water to storage pit No. 3 during fill operations was determined to be approximately 47 gpm.

- 5) Criterion: High and low level alarms in fuel storage pit No. 3 annunciate at respective set points (el 1431 ft and 1430 ft).

Remarks: The high level alarm was verified to annunciate at elevation 1431 ft 2 in. and the low level alarm was verified to reset at elevation 1430 ft 2 in.

- 6) Criterion: The storage pit coolant pumps function properly and pump approximately 200 gpm each under full system head. Shutoff head has been determined for each pump.

Remarks: This criterion has been met as described in the report.

- 7) Criterion: Subsystem No. 1 was vented and filled and the flow through each coil in the pipe chase and gallery will be adjusted so that the flow through all the coils is approximately the same in each coil.

Remarks: The operations were accomplished as described within this report.

- 8) Criterion: With coolant pump P-604 on standby and P-603 circulating, it has been verified that P-604 will start automatically when the discharge pressure on P-603 falls below the set point (10 psig) on the low pressure alarm downstream of the pumps. It has been verified that the annunciator windows on the main board light when this condition occurs.

Remarks: These operations were accomplished as described within this report. The automatic startup of the pump on standby was set at 30 psig instead of 10 psig, due to the higher discharge head of the replacement pumps.

- 9) Criterion: With wash cell and storage pit coolant pump P-603 on standby and P-604 circulating, it has been verified that P-603 will start automatically when discharge pressure falls below the low pressure alarm set point and that the annunciators on the main board light when this condition occurs.

Remarks: These actions have been verified as described within this report.

- 10) Criterion: Flow through the demineralizers and filters has been adjusted to 10 gpm.

Remarks: Flow adjustment was accomplished as described within this report.

- 11) Criterion: Subsystem No. 2 will be vented and filled and the flow adjusted through the service cells and equipment in the R/A Waste Bldg to achieve the required specified flow through each. A balanced flow in Subsystem No. 1 and Subsystem No. 2 will be achieved to maintain a 190 gpm flow through Subsystem No. 1 and a total flow of 320 gpm.

Remarks: The criterion flow of 320 gpm is considered to be unrealistic in that it assumes simultaneous usage of all three fuel wash cells, the disconnect cell, pump wash cell, R/A waste transfer tank and all equipment requiring cooling water in the R/A waste facility building. Figure 1106-5, preceding, shows flows and pressures obtained with the replacement pumps. These values exceed the revised minimum requirements. A total flow was established with one pump and one heat exchanger in operation. If on any future occasion additional flow exceeding 291.5 gpm is needed, it can be obtained either by opening the other heat exchanger or by operating the two pumps in parallel.

- 12) Criterion: The set point on the low pressure alarm will be adjusted so that the pump on standby will come on the line when the system total flow reaches 220 gpm (with flow through the pipe chase and gallery maintained at 191 gpm).

Remarks: The low pressure alarm is adjusted as per criteria 8) and 9) above. The criterion that the standby pump starts when total flow reaches 220 gpm was based upon the design output of the original pumps which were rated at 200 gpm, and was to place these pumps in parallel operation to maintain flow above 220 gpm. The replacement pumps are rated at 300 gpm, and the standby pump will come on the line when the operating pump discharge pressure drops below 30 psig.

- 13) Criterion: It has been verified that the solenoid valves (upstream of the vent compressors in the R/A Waste Bldg) open automatically when the compressor drives start, thus furnishing cooling to the compressors.

Remarks: Proper automatic operation of the solenoid valves has been verified.

14) Criterion: The service cell cooling coils have been blocked out of the system and blown out with N₂.

Remarks: This operation has been accomplished as described within this report.

15) Criterion: The pH of the filled and circulating system is established as 9.5 to 10 after 2 or 3 days circulation. Lithium hydroxide has been added in sufficient amounts to water in fuel storage pit No. 3 to achieve approximate pH of 9 to 10.

Remarks: Proper system water pH has been verified (between 9.5 and 10).

16) Criterion: The level of water in the fuel storage pit No. 3 has been adjusted to normal with the service cells blocked out of the system and the system shutdown.

Remarks: The normal water level in fuel storage pit No. 3 (35 ft 10 in.) has been verified.

B. OPEN LOOP COOLING WATER SUBSYSTEM

1) Criterion: The cooling tower water pumps, P-601 and P-602, will both function properly and will each pump at least 1725 gpm at 10 psig differential pressure.

Remarks: These pumps will each pump more than 1765 gpm (measured flow), with discharge head of more than 45 psig and differential pressure greater than 10 psig.

2) Criterion: The system has been vented and filled and flow adjusted through all coolers to obtain the required flow through each cooler.

Remarks: This criterion has been met. Refer to Sections II C-1 and -2.

3) Criterion: The control valves on the inlet of each nitrogen cooler open when the nitrogen temperature reaches 130# and the solenoid valves on the standby cooler inlets open when the nitrogen reaches 150#.

Remarks: The criterion has been met. Refer to Section II C-3 and -5. NOTE: Symbol # in criterion should be F.

- 4) Criterion: With pump P-601 circulating and P-602 on standby, it has been verified that P-602 starts automatically when the discharge pressure falls below the setpoint (10 psig) on the low pressure alarm. It has also been verified that the "Low Pressure Tower Pump" annunciators window and the "Auto-Start" annunciator window on the main board light when the above condition exists.

Remarks: The above criterion has been met. Refer to Section II C-6.
NOTE: Setpoint has been changed to 25 psig.

- 5) Criterion: With pump P-602 circulating and P-601 on standby, it has been verified that P-601 starts automatically when the discharge pressure falls below the setpoint on the low pressure alarm and that the proper annunciator windows light on the main board.

Remarks: The criterion has been met. Refer to Section II C-6. It should be noted here and in (4) above that the standby pump receives the signal from a pressure switch, not the low pressure alarm.

- 6) Criterion: With either pump circulating and the other pump on standby, it has been verified that if the circulating pump is "tripped out" the pump on standby will start automatically and that the "trip" and "Auto-Start" annunciator windows on the main board will light when this condition exists.

Remarks: The criterion has been met. Refer to Section II C-6.

V. EVALUATION

The purpose of the cooling water system preoperational test was to demonstrate that the system could be filled and the required coolant flows established to the various serviced components. Verification of proper performance of the associated system components (pumps, coolers, controls, instrumentation) was also included as a part of the test.

The test purpose and the intent of the test completion criteria were accomplished as specified in the HNPF Final Summary Safeguards Report, NAA-SR-5700. Some minor system discrepancies which were uncovered during the test were corrected as a matter of routine. Of a more serious nature was the leakage

of water from fuel storage pit No. 3, inability of the closed loop pumps to supply adequate flow, and corrosion of the piping and equipment internals.

With the exception of pit No. 3 leakage problems, all of the system deficiencies have been corrected and the system can be expected to perform satisfactorily during normal plant operation.

A. SYSTEM FILLING

No difficulties were encountered during the system filling operation. Adequate performance of the condensate supply system was verified. Level-actuated controls for the supply pump functioned as required to maintain the condensate supply head tank level within the prescribed limits. The capacity of the supply pump at the specified head of 70 ft was not determined during the test. However, at the actual operating head conditions of 60 ft, the pump delivered 41 gpm, which is approximately 60% greater than the required 25 gpm.

A maximum gravity flow rate from the condensate supply tank to pit No. 3 was measured to 47 gpm, or approximately 22 gpm greater than the minimum required rate. The high and low level alarms for pit No. 3 were checked, adjusted, and proper operation confirmed during the filling operation. During initial fill of pit No. 3, it was intended that lithium hydroxide be added to the water for pH control. However, this was not considered necessary since an analysis of the water indicated that the pH was already within the desired limits of 9.0 to 9.5. This was due to the presence of dissolved ammonia used in the CPPD boiler water treatment.

After the system was filled, considerable difficulty was experienced with plugging of the fuel storage cleanup filters. Analysis of the particulate material removed from the filter indicated that it was primarily iron oxide. During this period the pH of the water in the system gradually decreased to approximately 8.3 due to the evaporation of dissolved ammonia.

The system was then drained and flushed with demineralized water. Inspection of accessible internal piping and equipment surfaces revealed that corrosion had taken place. Corrosion pits up to 1/32 in. deep were observed. Although the corrosion was relatively severe there was no evidence that it was of sufficient magnitude to affect the structural integrity of the system. The system was re-filled with water directly from the demineralizer train rather than with condensate from the condensate storage tank.

Lithium hydroxide was then added to the system to increase the pH to the desired level (9.0 to 9.5) and a magnetic separator was installed to remove the iron oxide remaining in the system following the flushing operation.

During subsequent operation, the system was scrutinized to determine whether or not the corrosion control was adequate. Metallurgical specimens installed in the system have shown no detectable signs of corrosion and no further filter plugging has been experienced. From all indications, the present corrosion control is satisfactory and, if maintained, should provide the required protection during future operation.

B. CORE CAVITY COOLING COILS

Since the closed loop cooling system has been placed in operation, 3 of the 164 cooling coils have developed leaks. Due to the inaccessible location of the coils it has not been possible to determine the nature of the leaks. However, all indications point to mechanical damage which occurred during construction and/or material or fabrication defects.

As previously mentioned, there was some corrosion of the piping and equipment internals during the initial system operation. Corrosion was originally considered as a possible cause but all indications tend to rule this out, except as a minor contributing factor.

Maximum depth of corrosion detected in the accessible portion of the system was approximately 0.032 in., whereas the wall thickness of the coils is 0.175 in. This would only result in a reduction in wall thickness of approximately 18%, which is considered insignificant for this low operating pressure application. In addition, coil No. 1651 failed before the system was placed in operation and, therefore, the failure could not possibly have been a result of corrosion. The leaks in the other two coils (1559 and 1560), as determined by measuring the hydrostatic head above the leak, are in very close proximity (see Figure 1106-6) to one another. The odds that two adjacent coils would fail at essentially the same elevation due to random corrosion are astronomical, which again would tend to point to some other type of failure, such as mechanical damage.

Although the three leaking cooling coils are located in the same general region, there is still adequate cooling of the concrete. The original heat transfer analysis pertaining to coolant pipe spacing and maximum concrete temperatures specified that the pipe tunnels from the reactor cavity to the gallery seal

shall have coolant pipes 1 in. or larger in diameter and spaced 10 in. on center. The pipe spacing is based on a heat load of 109 Btu/hr - ft² and gives a maximum concrete temperature of 135 F. With one cooling pipe out, the maximum concrete temperature rises to 159 F. Extrapolation shows that loss of two adjacent cooling coils will result in concrete temperatures of about 180 F. Thus, concrete cooled by lines No. 1559 and 1560 will reach about 180 F while that cooled by line No. 1651 will reach about 160 F. Since the actual coil spacing is 8 in., instead of 10 in. as used in the above analysis, the concrete temperatures will be somewhat less than that given above.

The predicted concrete temperatures will have no detrimental effect on the strength or shielding properties of the concrete, or on the cavity liner. Concrete temperatures can approach the 500 to 600 F range before structural damage results. In terms of shielding, the only adverse effect of concrete temperatures near or above the boiling point of water is the degradation of fast neutron shielding capabilities. Since the temperature of the concrete in question should not approach the boiling point of water and since the region is one of low fast neutron flux, no detrimental effect on shielding should occur. The cavity liner proper is one-inch-thick steel plate and should certainly not be adversely effected. Recent hydrostatic tests of the adjacent coils indicate that the coils are sound and, therefore, it is doubtful that future failure in the same area will occur.

Leakage of water from fuel storage pit No. 3 has been a continuing problem. After the closed loop system was filled, the level in pit No. 3 gradually decreased corresponding to a leak rate of approximately 2 gph. An investigation determined that the bulk of the leakage was across the pit No. 3 drain valve. Subsequently, the leakage was reduced to an insignificant amount by installation of a valve in line No. 1547-2-H, which serves as a backup for the pit drain valves. No further problems were experienced for a period of several months. Then a leak developed equivalent to approximately 1.5 gph. The leak continued until the pit was drained to flush the system, at which time the rate had decreased to approximately 0.37 gph.

When pit No. 3 was refilled following the system flushing operation, the leak rate out of the system was approximately 168 gph. The majority of the leakage was subsequently determined to be from one of the core cavity cooling coils. However, after the coil was isolated the leak rate was still approximately 35.4 gph. The pit was drained and a test conducted to determine location of the

leaks. A dye penetrant test of all the pit liner welds revealed an 8-in. crack in a seam weld near the return line penetration and a 3-in. crack in the weld which seals the return line to the liner. In addition, several pinholes were found in the welds at various locations. The above mentioned leaks were repaired by welding. In an attempt to seal other undetected leaks, all weld seams, except for three floor welds, were covered with lead adhesive tape.

The pit was refilled and another leak test conducted. It was found that the pit still leaked at a rate of 14 gph. Storage pit No. 3 was again drained and inspected. The entire liner surface was carefully scrutinized, but the location of the leak could not be found.

It is evident that there will always be leakage problems with the presently installed liner. The design of the present liner necessitates making extremely difficult welds with resulting poor quality. This is further aggravated by warpage of the liner, which tends to crack the welds each time the water level is changed significantly. Serious consideration should be given to some other method of sealing pit No. 3, so that continued repair will not be required.

Figure 1106-6. Core Cavity Cooling Circuits With Leak Regions Spotted

HALLAM NUCLEAR POWER FACILITY
PREOPERATIONAL TEST COMPLETION REPORT
REACTOR LOADING FACE SHIELD COOLING SYSTEM

AI-P-1107

Test and Analysis

by

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by

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CONTENTS

	Page
I. Purpose of Test	1
II. Discussion of Test Performance	1
III. Analysis and Presentation of Test Results	2
IV. Compliance With Test Completion Criteria	8
V. Evaluation.	10
A. Piping Vibrations	10
B. Compressors	11

TABLES

I. Reactor Loading Face Shield Cooling Coil Pressure Drop Parameters.	4
II. Reactor Loading Face Shield Moderator Plug Cooling Coil Pressure Drop Comparison Data.	4
III. Alarm and Relief Valve Setpoints	5
IV. Reactor Loading Face Shield Cooling System Compressor Performance	5

FIGURES

1107-1. Compressor No. 1 Operating Parameters	6
1107-2. Compressor No. 2 Operating Parameters	7
1107-3. LFS Coolant Recompression System	12

HALLAM NUCLEAR POWER FACILITY
PREOPERATIONAL TEST COMPLETION REPORT
REACTOR LOADING FACE SHIELD COOLING SYSTEM, AI-P-1107

I. PURPOSE OF TEST

The purpose of this test was the purge and fill of the loading face shield (LFS) cooling system with nitrogen, and verification of satisfactory system operation.

The reactor loading face shield cooling system is designed to cool the shield by means of circulating nitrogen. This system consists of two compressors with snubbers on both the inlet and exhaust of each compressor, two heat exchangers, eight cooling coils in the reactor loading face shield, and the necessary piping, valves, and instrumentation.

The compressors are single cylinder, double acting, reciprocating machines. The compressor piston rods are sealed with both packing and solid graphite seals with seal chambers separating the two types. The graphite seals retain the nitrogen in the compressors. Air from the plant service air system is used to pressurize the seal chambers to reduce the pressure drop across the graphite seals. The packing seals retain the service air and nitrogen which leaks through the graphite seals into the seal chambers.

Six of the eight cooling coils are used to cool the reactor loading face shield proper. The two remaining coils are used to cool the three moderator plugs in the loading face shield. There are alarms for high system pressure, low system pressure, low flow, low compressor differential pressure, low nitrogen supply pressure, and high nitrogen return temperature, with alarm annunciators located in the control room. Relief valves are provided to relieve high system pressure and high compressor differential pressure.

II. DISCUSSION OF TEST PERFORMANCE

The loading face shield cooling coils were individually tested for abnormal flow restrictions. Service air was used to produce a pressure across the cooling coils. The air flow was measured through each coil except one moderator plug

coil. This coil had a leak during the time the rest of the coils were tested and was, consequently, not tested until after the system was filled with nitrogen and the reactor was filled with sodium. This coil was then tested by closing the supply valve and observing the effect on system flow, system-pressure drop, and motor current. These data were compared with those obtained by closing off other previously tested coils.

To purge the system, nitrogen was admitted to the compressor discharge piping, allowed to flow through the cooling coils, and vented from the compressor inlet piping. The compressors were started momentarily during the purging operation to allow purge of the compressors.

System pressure was increased to the operating pressure using each compressor in combination with each heat exchanger. System pressures, temperatures, flow, and motor current were recorded at different operating pressures.

System leak rate was determined by closing the nitrogen supply valve and recording the system pressure decay. System volume, which was needed to calculate leak rate, was determined by measuring the pressure in the nitrogen bottles used to fill the system, the pressure in the empty bottles, and the system pressure. The graphite seal leak-rate was measured by attaching a flow meter to the seal chamber while the service air was shut off.

System high and low-pressure alarms were tested with the system operating. Testing of the low-supply-pressure alarm was accomplished by throttling with the nitrogen bottle block valve until the alarm annunciated. The high-temperature alarm was tested by removing the sensing element from the nitrogen piping and placing the element in a temperature bath. Low flow and low differential pressure alarms were checked for operation by stopping each compressor. The mainboard annunciator for each alarm was checked for proper operation.

The system high-pressure relief valve was tested with the nitrogen supply control center closed off from the remainder of the system. It was, therefore, not necessary to pressurize the whole system.

III. ANALYSIS AND PRESENTATION OF TEST RESULTS

All of the cooling coils, except the two moderator plug cooling coils, had about the same flow and the same pressure drop. Both moderator plug cooling coils

had approximately the same flow, but less than that of the other coils, due to the smaller diameter of these two coils. These flows are shown in Table I and Table II. Since the area cooled by each moderator plug coil is less than that cooled by the other coils, the flow requirements are less. No cooling-coil throttle valve adjustments were made during this test. These adjustments will be made as required, on the basis of loading face shield temperature distribution, when the reactor power and system temperatures are raised during the rise-to-power testing.

The compressors were operated only a short time during the purging of the system because the check valves were very noisy. The check valves were hammering at about the same frequency as the compressor discharge flow pulse frequency. Since the compressors are equipped with integral check valves, the check valves in the piping were removed. No other problems were encountered with the purging operation.

The compressor motors were found to be 50 % overloaded with the system operating at design pressure. Subsequently, the 10 in. bore compressor cylinders were replaced with 10.5 in. cylinders. These larger cylinders have larger valves which were expected to have less pressure drop. Motor overload remained a problem, however, and smaller motor sheaves were required to reduce the compressor speed and motor load. An 8 in. pitch-diameter sheave was installed on compressor No. 2 motor, and a 7.5 in. sheave was installed on compressor No. 1, replacing the original 9 in. diameter sheaves. Neither compressor motor overloaded after the sheave changes.

Excessive vibration of the piping was observed upon initial operation of the system. The vibration was believed to be induced by the pulsing nitrogen flow. Snubbers were installed on the inlet and the outlet of each compressor, to obtain uniform nitrogen flow. Additional pipe hangers were also installed in the system. ~~These two modifications eliminated the excessive vibration in the~~ system.

The system and low-supply-pressure alarms were found inoperative and required adjustment and repair before they would operate. Subsequently, all of the alarms operated and annunciated properly in the control room. The pressure span between alarm and reset on the high and low-pressure alarms was greater than was desirable. The system high-pressure alarm did not reset until a pressure below the operating pressure was reached; the low-pressure alarm reset

TABLE I
REACTOR LOADING FACE SHIELD COOLING COIL
PRESSURE DROP PARAMETERS

Cooling Coil Between Pipe No.	Upstream Test Pressure (psig)	Downstream Test Pressure (psig)	Test Flow Meter (cfm)
2312 and 2313-2"-P	10	8.5	140
2314 and 2315-2"-P	10	8.7	138
2316 and 2317-2"-P	10	8.5	146
2318 and 2319-2"-P	10	8.5	146
2320 and 2321-2"-P	10	8.5	147
2322 and 2323-2"-P	10	8.5	147
2326 and 2325-2"-P (moderator plug coil)	10	8.5	94
2327 and 2324-2"-P (moderator plug coil)	This coil leaked when the others were tested. After the leak was repaired the coil was tested by comparison with two other coils (see Table II).		

TABLE II
REACTOR LOADING FACE SHIELD MODERATOR PLUG
COOLING COIL PRESSURE DROP COMPARISON DATA

Closed Cooling Coil Between Pipe No.	Compressor Discharge Pressure (psig)	Compressor Inlet Pressure (psig)	System Flow Meter (cfm)	Motor Current (amp)
None	125	115	2770	22
2327 and 2324-2"-P (previously untested moderator plug coil)	125	115	2730	23
2326 and 2325-2"-P (moderator plug coil)	125	115	2740	23
2322 and 2323-2"-P	125	114	2720	23.8

just before reaching operating pressure. Since this results in only an operational inconvenience, no modification of the alarm units, to reduce the span, was deemed necessary. Alarm setpoints are given in Table III.

TABLE III
ALARM AND RELIEF VALVE SETPOINTS

Alarm	Operating Point	Reset Point
System high pressure	256 psi	229 psi
System low pressure	212 psi	<235 psi
Low supply pressure	275 to 300 psi	
Low flow	2500 cfm	
High temperature	150°F	
Relief valve (high-system-pressure)	270 psig	

The system pressure relief valve was found defective and was replaced. Relief and reset pressures are listed in Table III.

The system was successfully operated at the design operating pressure without overloading the motors. Table IV and Figures 1107-1 and -2 show the parameters of the operating system for compressors No. 1 and No. 2. Installation of the snubbers had no apparent effect on motor load.

TABLE IV
REACTOR LOADING FACE SHIELD COOLING
SYSTEM COMPRESSOR PERFORMANCE

Compressor No.	1	2	2	Design Value
Heat exchanger No.	1	1	2	
Compressor discharge pressure (psig)	236	236	235	235
Heat exchanger outlet pressure (psig)	229	228	229	-
Heat exchanger outlet temperature (°F)	100	-	98	95
Compressor inlet (psig)	218	216	216	-
Nitrogen flow rate (scfm)	3610	3780	3820	3700
Compressor motor current (amps)	37	46	46	46 max
Motor sheave diameter (in.)	7.5	8.0	8.0	9.0

Figure 1107-1. Compressor No. 1 Operating Parameters

Figure 1107-2. Compressor No. 2 Operating Parameters

The system leak rate was nearly 4 scfm. Most of the leakage was from the compressor piston rod graphite seals, although some valve leakage was also noted. The graphite seal leak rate with design operating pressure was independent of the seal chamber pressures which did not exceed 120 psig. With the compressors not operating and the nitrogen supply closed, the system pressure stabilized at 2 psi below the service air pressure. The system volume used to determine the system leak rate with the snubbers installed was 122 cubic ft. A study is being made of possible ways to reduce the compressor seal leak rate.

IV. COMPLIANCE WITH TEST COMPLETION CRITERIA

Following are the test completion criteria associated with preoperational test AI-P-1107, Reactor Loading Face Shield Cooling System, as listed in NAA-SR-5700 (Supplement 4), pages XI (a)-23 and -24, with remarks regarding the test to explain compliance with the criteria.

- a) Criterion: Pressure drop through each cooling coil in the loading face shield has been determined and the necessary adjustments made in individual coil throttle valve to assure a balanced flow through the entire system of coils.

Remarks: The pressure drop was measured through each of the cooling coils, except for one moderator plug coil. A qualitative indication of flow in the moderator plug coil was obtained. No individual coil throttle valve adjustments were found necessary.

- b) Criterion: The system has been purged and nitrogen successfully circulated by each compressor (K-901 and K-902) for a period of 5 min each.

Remarks: The system was purged and nitrogen was successfully circulated with each compressor.

- c) Criterion: Pressure will have been raised in the system until the high pressure alarm (located at the snubber tank) annunciates at 255 psig. It will have been verified that the proper annunciator window lights (on the main board) when the high pressure alarm setpoint is reached.

Remarks: The high pressure alarm did annunciate on the main board at 256 psig and reset at 229 psig.

- d) Criterion: After the high pressure alarm test the system relief valves will have opened and seated properly and the system checked for leaks.

Remarks: This criterion was met as described in the text. Relief pressure was 270 psig.

- e) Criterion: The system will have been bled down until the low pressure alarm (located at the snubber tank) annunciates at 215 psig. It will have been verified that the proper annunciator window on the main board lights when the low pressure setpoint is reached.

Remarks: The low pressure alarm on the main board annunciated at 212 psig and reset before reaching operating pressure.

- f) Criterion: The system nitrogen supply will have been blocked off behind the pressure control valve and bled down until the low pressure alarm (located upstream of the PCV) annunciates when the pressure drops to 275 psig. It will have been verified that the proper annunciator window on the main board lights when the low pressure setpoint is reached.

Remarks: The low nitrogen supply alarm on the main board annunciated between 275 and 300 psig. There is no indicator on this alarm in the compressor room. The gage for the supply pressure was marked in 100 psig intervals, but had only 4 marks for the first 500 psig. For these reasons the alarm point is approximate.

- g) Criterion: Both nitrogen compressors (K-901 and K-902) will have been operated individually and will have satisfactorily produced 235 psig at the snubber tank. Each compressor will have been checked for leakage and proper cooling of the compressor jacket in conjunction with the cooling water system.

Remarks: Both compressors were operated individually with each heat exchanger and found satisfactory. Neither motor overloaded at 235 psig operating pressure. Each compressor was checked for leakage and proper cooling.

- h) Criterion: The nitrogen supply will have been blocked off and the system checked for leakage with the compressor running.

Remarks: The nitrogen supply was blocked off and the system was tested for leakage.

- i) Criterion: It will have been verified that low differential pressure alarm and low flow alarm will not trip when a compressor is running.

Remarks: The low differential pressure alarm and the low flow alarm did not trip when a compressor was running.

- j) Criterion: It will have been verified that the low differential pressure alarm and low flow alarm trips when the compressor is stopped and that proper window (for each alarm) lights on the main board when this condition occurs.

Remarks: The low differential pressure and the low flow main board alarms annunciated when the compressors were stopped.

V. EVALUATION

The purpose of the reactor loading face cooling system preoperational test was to confirm that the system could be purged and filled with nitrogen, and to verify proper performance of the system and components at normal operating pressure.

Initial testing of the system confirmed the existence of three problems:

- a) Piping vibration
- b) Overloading of the compressor drive motors
- c) Excessive leakage

With the exception of system leakage, all of these problems have been resolved. Subsequent testing determined that the system is adequately prepared for power operation and can be expected to provide dependable service.

The test completion criteria as specified in the HNPF Safeguards Report, NAA-SR-5700, were met and the test purposes were accomplished.

A. PIPING VIBRATIONS

Violent piping vibrations were experienced during initial operation of the system. This was due to the large pressure pulsation induced by the reciprocating

compressors. Attenuation of these vibrations was effected by the installation of pulsation snubbers in the suction and discharge side of the compressors. Additional pipe supports were installed at strategic locations, further reducing the magnitude of the vibrations. Operation of the system following the completion of these modifications verified that the vibrations were reduced to tolerable levels.

B. COMPRESSORS

As originally installed, the compressor motors were approximately 50% overloaded when operated at design pressure. Compressor load reduction was effected by the installation of larger cylinders with larger valves and the replacement of the motor sheaves with sheaves of smaller diameter. The larger valves provided decreased friction losses and the smaller sheaves reduced compressor output. Checkout of the modified compressors verified proper performance. However, as specified in the test report, the flow capacity of the No. 1 unit was slightly less than the design value. Following completion of the test the 7.5 in. sheave on unit No. 1 was replaced with an 8.0 in. sheave. This modification brought compressor No. 1 output in line with design requirements and with compressor No. 2 output. A check of pressure drop through the LFS cooling coils established that adjustment of the individual circuits was not required in order to achieve the desired flow balance. At a constant differential pressure, flow through the moderator plugs was less than in the other circuits but the flow-to-surface cooled ratio was essentially the same.

Excessive system leakage has been a continuing problem with this system. The leak rate was reduced considerably by tightening and soldering of flanges, valves, and compressor attachments. However, the gland seals on the compressors are still leaking at ~ 4 scfm.

Attempts to correct this condition by the replacement of seals have been unsuccessful. It is presently planned to install a recompression system (Figure 1107-3) which will collect and return the leakage to the system.

Although installation of the recompression system will increase the system complexity, it appears to be the most expedient method of correcting the leakage problem. After this equipment is installed and placed in operation, the loading face cooling system should perform adequately during normal plant operation.

Figure 1107-3. LFS Coolant Recompression System

HALLAM NUCLEAR POWER FACILITY
PREOPERATIONAL TEST COMPLETION REPORT
HELIUM SYSTEM

AI-P-1108

Test and Analysis

by

E. C. TURNER

Evaluation

by

L. L. JOHNSON

A T O M I C S I N T E R N A T I O N A L

A DIVISION OF NORTH AMERICAN AVIATION, INC.

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CONTENTS

	Page
I. Purpose of Test	1
II. Discussion of Test Performance	1
A. Helium System	1
B. Helium Third Fluid System.	15
III. Analysis and Presentation of Test Results	18
A. Helium System	18
B. Helium Third Fluid System.	25
IV. Compliance With Test Completion Criteria	27
A. Helium System	27
B. Helium Third Fluid System.	28
V. Evaluation	29
A. Controls and Instrumentation.	29
B. Dessicant Dryers	30
C. Reactor Leak Rate Monitoring System	32
D. Steam Generator Third Fluid System.	32

FIGURES

1108-1	Helium Control Manifold	2
1108-2	Typical Helium Pressure Control Station.	4
1108-3	Reactor Helium Supply and Vent System	6
1108-4	Typical Pressure Control System Using Separate Supply and Vent Control Valves.	9
1108-5	Helium Third Fluid System	16
1108-6	Helium System Control Manifold Flow Characteristics.	19
1108-7	Helium Third Fluid System Pressure Decay Test	26

HALLAM NUCLEAR POWER FACILITY
PREOPERATIONAL TEST COMPLETION REPORT
HELIUM SYSTEM, AI-P-1108

I. PURPOSE OF TEST

The purpose of the helium system test was to:

- a) Pressurize the main helium system header and each individual helium control valve station.
- b) Demonstrate that the helium system controls and instrumentation will function properly.
- c) Evacuate and purge the helium third fluid monitoring system of each of the three steam generators to prepare the systems for operation.
- d) Verify correct operability of the controls and instruments in the helium third fluid system.

II. DISCUSSION OF TEST PERFORMANCE

A. HELIUM SYSTEM

This test was started by connecting 10 helium cylinders to each bank of the helium supply manifold, closing all valves in the manifold and opening the supply valve on one cylinder. The system was tested and placed in operation, one subsystem at a time.

1. Helium Control Manifold

Figure 1108-1 is a schematic diagram showing the helium control manifold set-up. Pertinent components of the system are included by number designation, whereas "typical" line set-ups are utilized in other figures incorporated in this report.

Figure 1108-1. Helium Control Manifold

The normal (50 psig) operating side of the manifold was pressurized and pressure control valve (PCV)-800 (Figure 1108-1) was adjusted to provide a static downstream pressure of 50 psig. Low pressure tank T-801 was allowed to come up to 50 psig, thus clearing tank low pressure alarm (PAL)-801. The tank was isolated from the supply and the pressure observed to determine if relief valve (RV)-800 was leaking. The helium supply valve at the cylinder was closed, and the entire system was checked with a helium leak detector to determine if any leaks existed in the complete manifold. After the leak test was completed, all cylinders in bank 1 were opened to the manifold. Pressure in T-801 was reduced through the drain to 40 psig, to verify that PAL-801 would initiate the alarm circuit and energize the main board "Low Pressure Helium Tank" annunciator. The low pressure (45 psig) side of the manifold was pressurized and PCV-801 was adjusted to provide a downstream static pressure of 45 psig. T-801 was pressurized to 45 psig and the pressure point where PAL-801 cleared was noted. The manifold was isolated from the helium supply at V-811, and T-801 was bled down until PAL-800 alarmed and the alarm circuit energized the main board "Low Pressure Helium Supply" annunciator. PAL-800 was adjusted to alarm when the helium supply pressure dropped to 100 psig. The system was bled down below 45 psig, and the helium bottles in bank 2 were opened to the pressure side of the manifold, thus verifying that pressure would be maintained at 45 psig in T-801 from bank 2 when the supply in bank 1 was depleted. The manifold valving was then adjusted so that bank 2 would be in service and bank 1 on standby with V-814 closed and V-812 and V-813 open. A spare helium bottle was temporarily connected to the drain on T-801 and pressure increased until RV-800 lifted. RV-800 was adjusted to open at 55 psig. The spare bottle was disconnected from T-801 and the pressure bled off to 50 psig. The pressure point where RV-800 reseated was noted. At this time, testing of the helium manifold system was complete.

2. Drum Melt Station

Helium was supplied to the drum melt station pressure control station (see Figure 1108-2 for typical helium pressure control station), and

Figure 1108-2. Typical Helium Pressure Control Station

PCV-830 was adjusted to provide a downstream pressure of one psig. Each drum melt station was independently pressurized (through the control station bypass valve) to lift the relief valve at each station. Both relief valves were adjusted to relieve (lift) at three psig. Pressure was vented to less than three psig and the system observed (after the relief valves reseated) to assure that the pressure did not decay in a 2 minute period, thus verifying that the relief valves were not leaking after reseating. Reseating pressure was noted to be safely above operating pressure.

3. Sodium Transfer Tank

Helium was supplied to the sodium transfer tank pressure control station and PCV-831 was adjusted to provide a downstream pressure of 30 psig. The vent valve was adjusted to provide a flow of 25 scfm and system pressure was noted at this flow rate. The vent valve was closed and the system pressurized (through the bypass valve) to lift the relief valve. The relief valve was allowed to reseat, and the reseating pressure noted. System pressure (after relief valve reseating) was observed for decay over a 2 minute period, to verify that the relief valve did not leak. When the test was completed, the system was valved for normal operation.

4. Reactor Helium Supply

This test involved simulating the reactor helium pressure by controlling instrument air pressure to PIC-803T, pressure indicator-controller transmitter. Although CV-1004, vent gas control valve, is actually a part of the radioactive vent system, it is included in this test.

Pressure indicator-controller PIC-803 (Figure 1108-3) was tested as a system. Both control mechanisms in PIC-803 were set with a 5% proportional band. The lower control was set at one inch of water pressure and the upper control was set at six inches of water pressure. PIC-803T was isolated from the reactor gas atmosphere and a pneumatic loading instrument, Wallace & Tiernan Series A-755, was connected to PIC-803T. Pressure to PIC-803T was reduced to atmospheric to verify that PIC-803T would provide

the proper instrument air loading pressures to CV-803 (pressure to open valve) and CV-1004 (pressure to close valve) and would maintain CV-1004 fully closed and CV-803 fully open. It was also verified that PAL-808 was loaded properly, energized the alarm circuit and actuated the main board "Hi-Low Pressure Reactor Atmosphere" annunciator when pressure to PIC-803T dropped to + 1/2 inch of water. Adjustments were made to PIC-803 so that CV-803 would be fully open when PIC-803T was indicating one inch of water pressure. Pressure to PIC-803T was increased with the loading instrument, and it was noted that the alarm cleared and that the loading pressure to CV-803 decreased, allowing CV-803 to close at approximately 1-1/2 inches of water, thus shutting off the helium supply. The pressure increase to PIC-803T was continued and proper adjustments were made to PIC-803 so that the loading pressure to CV-1004 decreased, thus allowing CV-1004 to start opening at approximately 5-1/2 inches of water pressure and to be fully open at six inches of water pressure. When pressure to PIC-803T reached seven inches of water, it was noted that PAL-808 energized the high alarm circuit, thus lighting the main board "Hi-Low Pressure Reactor Atmosphere" annunciator. Pressure to PIC-803T was reduced to four inches of water, and it was noted that the alarm cleared when pressure dropped below seven inches of water. The loading pressure to CV-1004 increased, to fully close CV-1004 at approximately 5-1/2 inches of water pressure.

The reactor cover gas integrating flow indicators FI-848 and FI-849 were tested by establishing flow through both meters in series, in a bypass loop around the reactor. Such a loop was included to provide a common flow through the meters for comparison of the two instruments. The data taken during this test were used to determine the error coefficient between the meters. This coefficient will be used during normal operation when computing the reactor leak rate. It was verified that the control system was capable of supplying the necessary rate of flow to maintain pressure in the reactor during normal operations.

Pressure relief valves RV-841 and RV-1004 were isolated from the system and pressurized to verify that they would lift at the proper pressure

and reseal when pressure was reduced. The isolated systems were observed for pressure decay after the relief valves reseated, to check for relief valve leakage. Necessary adjustments were made to the relief valves to provide relief at the proper pressure and eliminate leakage. When the test was completed, the system was valved for normal operation except for bypassing the flow integrators FI-848 and FI-849.

5. Primary Sodium Fill Tanks Helium Supply

This test involved simulating the fill tank pressure by controlling instrument air pressure to PIC-805. Although the vent gas control valve (CV-1004) is actually a part of the radioactive vent system, it is included in this test.

Pressure indicator-controller PIC-805 was tested as a system (see Figure 1108-4 for typical pressure control system using separate supply and vent control valves). Both control mechanisms in PIC-805 were set with a 5% to 10% proportional band. The lower control was set at 1/2 inch of water pressure and the upper control was set at seven inches of water pressure. Under normal operation, the primary fill tank cover gas and the reactor cover gas are made common by a vent line joining the two atmospheres. Hence, the fill tank atmosphere operating pressure range is set outside that of the reactor atmosphere operating pressure range on both upper and lower limits, to prevent PIC-803 and PIC-805 from fighting each other. Both controllers are in operation at all times, but either provides backup for the other in case of failure. PIC-805, was isolated from the fill tank atmosphere and a pneumatic loading instrument, Wallace & Tiernan Series A-755, was connected to PIC-805. Proper operation of PIC-805, pressure indicator-controller, for helium to the primary fill tanks; CV-805, helium supply control valve, CV-1005, fill tank vent control valve, PAL-810, helium to primary fill tank low pressure, and PAH-810, helium to primary fill tanks high pressure alarm, were tested in the same manner as PIC-803 discussed in 4. Helium supply flow rate through CV-805 was adjusted to provide a maximum of 15 scfm to the fill tanks.

Figure 1108-4. Typical Pressure Control System Using Separate Supply and Vent Control Valves.

RV-1005 was isolated from the fill tanks atmosphere and pressurized to verify that it would lift and reseat properly. After reseating, the isolated system was observed for pressure decay, to determine if the relief valve was leaking. Necessary adjustments were made so RV-1005 would relieve at the proper pressure and would not leak after reseating. When the test was completed, the system was valved for normal operation.

6. Reactor Cavity Helium Supply

This test involved simulating the reactor cavity atmosphere pressure by controlling instrument air pressure to PIC-804. Although CV-1003, vent gas control valve, is actually part of the radioactive vent system, it is included in this test.

PIC-804 was tested as a system (see Figure 1108-4 for typical pressure control system using separate supply and vent control valves). Both control mechanisms in PIC-804 were set with a 10% proportional band. The lower control was set at one inch of water pressure. PIC-804 was isolated from the reactor cavity atmosphere and a pneumatic loading instrument, Wallace & Tiernan Series A-755, was connected to PIC-804. Proper operation of PIC-804, helium to reactor cavity controller; CV-804, helium supply control valve; CV-1003, cavity vent control valve; PAL-809 low pressure alarm; and PAH-809, high pressure alarm, were tested in the same manner as PIC-803, as discussed in 4. Helium supply flow rate through CV-804 was adjusted to provide a maximum of 100 scfm to the cavity.

RV-827 was isolated from the reactor cavity atmosphere and pressurized to verify that it would lift and reseat properly. After reseating, the isolated system was observed for pressure decay, to determine if RV-827 was leaking. Necessary adjustments were made and the system was valved for normal operation.

7. Reactor Service Connection

Helium was supplied to the service connection control station (see Figure 1108-2 for typical helium pressure control station) and PCV-824

was adjusted to provide a downstream pressure of 15 psig. The service connection was opened and the throttle valve adjusted to provide a flow of 5 scfm. The service connection was closed and the system observed, to verify that pressure would remain constant. The control station block valve was closed and system pressure bled down. When the test was completed, the system was valved for normal operation.

8. Control Rods Helium Supply

Helium was supplied to the control rod helium control station and PCV-823 was adjusted to provide a downstream pressure of 15 psig. Header valve V-8119 was opened and the throttle valve adjusted to provide a flow of 5 scfm. The header valve was closed and the system observed, to verify that the pressure remained constant. RV-826 was isolated from the system and pressurized through the control station bypass valve, for verification that the relief valve would lift and reseal properly. After reseating, the isolated system was observed for pressure decay, to determine if RV-826 was leaking. Necessary adjustments were made and the system was valved for normal operation up to V-8119.

9. Wash Cells, Disconnect Cell, and Moderator Can Pickup Cell Helium Supply

Helium was supplied to the control station and PCV-832 was adjusted to provide a downstream pressure of one psig. The throttle valve was adjusted to provide a flow of seven scfm and the lines to each cell were blown out and the valves closed. The downstream header valve was closed and the system observed, to determine whether the pressure remained constant. The disconnect cell trench line was blown out through V-8151, thus ensuring that the system was ready for operation.

10. Fuel Cask Service Connection

The fuel cask helium service connection pressure control valve, PCV-833, and throttle valve V-8169, were adjusted to provide a downstream

pressure of approximately five psig and a flow of five cfm. The downstream header valve was closed and system pressure observed for verification of constant pressure.

11. Primary Sodium Service Drain Tank

The drain tank helium pressure control valve, PCV-828, and throttle valve V-865, were adjusted to provide a downstream pressure of 10 psig and a flow of 10 cfm. The downstream header valve was closed and system pressure observed, to determine that the pressure remained constant.

RV-830 was isolated from the system and pressurized through the control station bypass valve, to verify that it would lift and reseal properly. After resealing, the isolated system was observed for pressure decay to determine if RV-830 was leaking. RV-830 was adjusted as necessary to provide proper relief pressure and positive resealing. Subsequently, the line to the moderator coolant piping freeze trap was blown out and the system was valved for normal operation.

12. Primary Sodium Service System Freeze Traps Helium Supply

The system helium pressure control valve (PCV-826) supplying the primary cold traps, carbon traps and freeze traps was tested and adjusted for a flow of one cfm and a pressure of 10 psig downstream of PCV-826, in the same manner as discussed in 11.

RV-828 was tested in the same manner as were the other relief valves previously discussed.

After the lines to all freeze traps were blown out, the block valve was closed and the system was bled down and closed.

13. Intermediate Heat Exchanger Freeze Trap Helium Supplies

PCV's -817, -818 and -819, (IHX No. 1, No. 2 and No. 3, respectively) were each tested and adjusted for a flow of 15 cfm and a pressure

of 1.75 psig downstream of the respective pressure control valves. When each control station had been pressure adjusted, the downstream header valve (to the freeze traps) on each unit was closed and the system observed for constant pressure.

RV's -816, -817 and -818 were each tested in the same manner as previously discussed for the other relief valves.

After the test was completed on each control station and relief valve, the block valves were closed and the systems bled down through the downstream header valves and the header valves were closed.

14. Secondary Sodium Expansion Tanks Helium Supplies

The tanks helium pressure control valves -814, -815 and -816 (secondary expansion tanks No. 1, No. 2 and No. 3, respectively) were each tested and adjusted for a flow of 35 cfm and a pressure of 20 psig static downstream of the respective pressure control valves. When each control station had been pressure adjusted, the downstream header valve to each vapor trap was closed (the vent valves being already closed) and the system observed to verify that the pressure in each remained constant.

Each system throttle valve was closed and the pressure increased through the bypass valve until the high pressure alarm, PAH-817, -818 and -819, (on tanks No. 1, No. 2, and No. 3, respectively) energized the alarm circuit and actuated the associated main board "Hi-Low Pressure Expan Tank" annunciator for each tank. Necessary adjustments were made to all high pressure alarms to ensure tripping at, or just below, 22 psig. The pressure was further increased until relief valves -813, -814 and -815 lifted. The bypass valve was closed and the relief valves allowed to reseal. After resealing, the system was observed for pressure decay to determine whether the relief valves were leaking. Necessary adjustments were made to the relief valves. The vent valve on each tank was then cracked open and the system allowed to bleed down until the low pressure alarms, PAL-817, -818, and -819 energized the alarm circuits and actuated the main board "Hi-Low Pressure Expan Tank" annunciator. Necessary adjustments were made to all low pres-

sure alarms to ensure tripping at, or just above, 15 psig. Each system was bled down and the vent valve closed.

15. Secondary Sodium Fill Tanks Helium Supplies

The helium pressure control valves for secondary fill tanks No. 1, No. 2 and No. 3 (PCV's -802, -803 and -804, respectively) were tested and adjusted in the same manner as the pressure control valves discussed in 13., to provide a downstream pressure of 20 psig and a flow of 35 cfm. After each control valve was adjusted, the system was closed and observed for pressure constancy.

RV's -801, -802 and -803 were each tested in the same manner as those discussed in 13. Necessary adjustments were made to the relief valves to ensure relief at 23 psig and that the valves would not leak after re-seating.

Following testing of each relief valve, the system was bled down through the respective vent valve and the vent valve closed.

16. Secondary Sodium Service Drain Tank Helium Supply

The secondary sodium service drain tank helium pressure control valve, PCV-807, and the associated throttle valve were tested and adjusted to provide a downstream pressure of 20 psig and a flow of 15 cfm. The downstream header valve was closed and the system observed, to verify that the pressure remained constant.

RV-806 was tested in the same manner as discussed in 13. Necessary adjustments were made to RV-806 so that the valve would relieve at 23 psig and would not leak after reseating. The system was bled down through the vent valve and the vent valve closed.

17. Secondary Sodium Cold Trap Helium Supply

PCV-805 was tested in the same manner as PCV-807 discussed above in 16. PCV-805 and related throttle valve were adjusted to provide

a downstream pressure of 20 psig and a flow of one cfm. The downstream header valve was closed and the system observed for constant pressure.

RV-804 was tested in the same manner as discussed in 13. and adjusted to relieve at 23 psig and prevent leakage after reseating. The system was then bled down through the vent valves and the vent valves closed.

18. Helium Main Header Flow Indicators

Flow indicators FI-846 and FI-847 (in the primary and secondary system main helium headers, respectively) were valved into the system. With no flow to the system control valve stations, only main header leakage was indicated. These data were recorded and the flow indicators valved out of the systems.

B. HELIUM THIRD FLUID SYSTEM

Helium bottles were connected to the helium third fluid system main header (Figure 1108-5). PCV-838, and block valve V-5151, were closed and PCV-838 was set to provide a downstream pressure of 5 psig.

The main header vent valve V-8237, the helium dessicant dryer block and throttle valves V-5151, V-5154 and V-5155 were opened and the dryer was purged for one minute. After purging, the dryer was isolated from the system and V-5151 closed.

A vacuum pump was connected to the system at the vent valve and the entire third fluid system was evacuated to 25 in. Hg. PCV-838 was set for 15 psig and the system pressurized with helium. After reaching 15 psig, the helium supply was again shut off and the system evacuated to 26 in. Hg. PCV-838 was set for 300 psig downstream pressure and the system was pressurized after the vacuum pump had been isolated (by the main header vent valve).

All block valves were closed tight, thus isolating each component from the main header. Each system was observed for five minutes for pressure decay.

Figure 1108-5. Helium Third Fluid System

Evaporator No. 1 vent valve, V-543, was cracked open and helium pressure allowed to bleed down until PAL-820 energized the alarm circuit and actuated the main board "Hi-Low Pressure Helium Steam Generator No. 1" annunciator. The vent valve was closed and block valve V-5156 opened slowly, thus verifying that the alarm cleared as the pressure increased above the setpoint of PAL-820. Necessary adjustments were made to PAL-820 so that the alarm circuit would be energized when pressure dropped to 250 psig. The low pressure alarms on the other five components were tested and adjusted in the same manner.

All helium system block valves were closed and relief valve RV-842 was "gagged". PCV-838 was adjusted to provide a downstream pressure of 375 psig. Evaporator No. 1 block valve, V-5156, was cracked and the pressure allowed to increase in the evaporator helium system until PAH-820 energized the alarm circuit and actuated the main board "Hi-Low Pressure Helium Steam Generator No. 1" annunciator. When the alarm circuit was energized, the block valve was immediately closed. Vent valve V-543 was opened, and the pressure allowed to decay to 300 psig. It was noted that PAH-820 cleared as the pressure decreased. Necessary adjustments were made so that PAH-820 would initiate the alarm circuit when pressure reached 340 psig. The high pressure alarms on all other components were tested and adjusted in the same manner.

The main header pressure was allowed to decay, and PCV-838 was adjusted to provide a downstream pressure of 300 psig. The gag was removed from RV-842 and the pressure was increased in the main header until RV-842 lifted, at which time the header block valve, V-5151, was closed. RV-842 then reseated and the pressure in the header was observed for decay, to determine if the valve was leaking. Necessary adjustments were made to RV-842 so that it would relieve at 350 psig and would not leak after reseating.

Main header pressure was bled off and PCV-839 adjusted to provide a downstream pressure of 300 psig. All block valves were opened and the entire system pressurized to 300 psig. The system was observed to verify that PCV-839 would hold the pressure constant.

All helium system block valves and pressure control station block valves V-5151 and V-5152 were closed and the main header vented to approximately 300 psig. The pressure in each component was then noted and recorded for a period of eight hours. Necessary repairs and adjustments were performed on system piping, valves and rupture discs, until pressure in all components held constant at 300 psig for a period of at least eight hours. When the pressure decay test was completed, the third fluid system preoperational test was completed.

III. ANALYSIS AND PRESENTATION OF TEST RESULTS

A. HELIUM SYSTEM

1. Helium Control Manifold

The manifold was operational and had no leaks. However, the pressure control valves originally installed in the system were single stage regulators and would not maintain a constant downstream pressure. A multistage helium regulator (Harris Calorific Co. Series X-92-2SC) was purchased and a flow test performed. Upon satisfactory completion of the regulator test (see Figure 1108-6 for test results), two Harris regulators, Series X-92-3V2, were purchased and installed in place of the original regulators. The Harris regulators maintain a constant, static, header pressure and will maintain system pressure above the low alarm setpoint up to 50 cfm, with only one bank in operation:

- a) PAL-801 alarms at 43 psig and clears at 43.8 psig
- b) PAL-800 alarms at 100 psig
- c) RV-800 lifts at 60 psig and reseats at 55 psig

Figure 1108-6. Helium System Control Manifold Flow Characteristics.

2. Drum Melt Station

PCV-830 maintains a constant static downstream pressure of 1.25 psig. Results of the relief valve test were as follows:

<u>Valve No.</u>	<u>Lifting Pressure</u>	<u>Reseating Pressure</u>
RV-823	3.25 psig	2.25 psig
RV-833	3.75 psig	2.00 psig

3. Sodium Transfer Tank

PCV-831 maintains constant static downstream pressure of 30.25 psig. The control station was adjusted to provide a helium flow of 25 cfm. Results of the relief valve test were as follows:

<u>Valve No.</u>	<u>Lifting Pressure</u>	<u>Reseating Pressure</u>
RV-834	40.5 psig	30 - 25 psig

4. Reactor Helium Supply

With the reactor atmospheric pressures listed below, PIC-803 will provide air loading pressure to position CV-803 and CV-1004 as follows:

Reactor Pressure (Inch of H ₂ O)	CV-803		CV-1004	
	Loading Pressure	Position	Loading Pressure	Position
0	20 psig	Full Open	20 psig	Full Closed
1	15 psig	Full Open	20 psig	Full Closed
1-1/4	9 psig	Half Open	20 psig	Full Closed
1-1/2	3 psig	Full Closed	20 psig	Full Closed
2 - 5	0 psig	Full Closed	15 psig	Full Closed
5-1/2	0 psig	Full Closed	15 psig	Full Closed
5-3/4	0 psig	Full Closed	9 psig	Half Open
6	0 psig	Full Closed	3 psig	Full Open

Low pressure alarm -808 is set for, and trips at, 0.5 inches of water. PAH-808, is set at 7.0 inches of water and trips at that pressure. The diaphragms in the instrument sensing elements are good for only 3-4 psig; capillary tubes were inserted in the instrument sensing lines to prevent pressure surges from the helium supply control station from rupturing the diaphragms. When testing started on the American Meter Company positive displacement type flow integrators, FI-848 and FI-849, the same trouble was encountered. A surge tank (5 cu. ft. capacity) filled with steel wool, was installed upstream of the flow integrators in an unsuccessful attempt to prevent damage to FI-848 which is immediately downstream of the pressure control station. A pressure regulator was then installed upstream of CV-803, to reduce the helium supply pressure from 50 psig to 15 inches of water pressure and provide at least 15 cfm flow. The flow integrators were tested for comparative volume measurement; it was found that for the same volume flowing through each, the FI-848 average measurement was 1.7% greater than that of FI-849. FI-848 had failed twice before the pressure regulator was installed upstream of CV-803, but had been repaired and recalibrated to within 1% accuracy. With the installation of the pressure regulator, it was found that the maximum flow attainable through CV-803 was approximately 3-1/4 cfm. It was decided that this was insufficient flow to maintain the system; so the valve seal and plug in CV-803 was exchanged for one of larger size. A final test on the flow rate showed a capacity of more than 10 cfm which was considered satisfactory. When the meters were being calibrated and repaired, it was learned that this meter should never be subjected to pressures greater than 17 inches of water or pressure surges greater than 5 inches of water if accuracy is to be maintained and damage to the meter avoided.

Results of the relief valve tests were as follows:

<u>Valve No.</u>	<u>Lifting Pressure</u>	<u>Reseating Pressure</u>
RV-840	7.2 psig	4.8 psig
RV-1004	5.2 psig	3.6 psig

5. Primary Sodium Fill Tanks Helium Supply

When simulated fill tank pressures were supplied to PIC-805, pressure indicator controller, the controller loaded CV-805 and CV-1005 so that CV-805 was full open at 0.5 inches of water pressure, half open at 0.75 inches of water pressure and closed at 1.0 inches of water pressure. CV-1005 was closed, up to 6.5 inches of water pressure, half open at 6.75 inches of water pressure and full open at 7.0 inches of water pressure, verifying proper operation of the helium supply control and fill tank vent control stations from the dual pilot controller, PIC-805. RV-1005 lifted at 5.3 psig and reseated at 3.2 psig when tested.

PAL-810 energized the alarm circuit at 0.5 inches of water pressure, and PAH-810 energized the alarm circuit at 7.0 inches of water pressure. These results were considered satisfactory.

6. Reactor Cavity Helium Supply

The setpoints on PIC-804, pressure indicator controller, were the same as those on PIC-803, controller for reactor atmosphere. The results of the test on PIC-804 and CV's -804 and -1003 were exactly the same as those discussed in 4., for PIC-803, CV-803 and CV-1004.

PAL-809 energized the alarm circuit at 0.5 inches of water pressure, and PAH-809 energized the alarm circuit at 7.2 inches of water pressure. RV-827 lifted at 1.6 psig and reseated at 1.3 psig.

7. Reactor Service Connection

Pressure control valve -824 was adjusted to provide a downstream static pressure of 15 psig. When rated flow was achieved, the downstream pressure was 12 psig.

8. Control Rod Helium Supply

PCV-823 was adjusted to provide a downstream static pressure of 15 psig. When rated flow of 5 scfm was achieved, downstream pressure was 14.5 psig. Relief valve RV-826 lifted at 21.5 psig and reseated at 19.2 psig.

9. Wash Cells, Disconnect Cell and Moderator Can Pickup Cell Helium Supply

Pressure control valve -832 was adjusted to provide a downstream static pressure of one psig. Seven scfm flow could be achieved with this setting; the test results were considered satisfactory.

10. Fuel Cask Service Connection

Pressure control valve -833 was adjusted to provide a downstream static pressure of five psig. The throttle valve could be adjusted to provide a flow of five scfm with this pressure control valve setting; the test results were considered satisfactory.

11. Primary Sodium Service Drain Tank

Pressure control valve -828 was adjusted to provide a downstream static pressure of 10 psig. The throttle valve could be adjusted to provide a flow of 9.5 scfm with this pressure control valve setting. RV-830 lifted at 11.75 psig and reseated at 10 psig.

12. Primary Sodium Service System Freeze Traps Helium Supply

Pressure control valve -826 was adjusted to provide a downstream dynamic pressure of 10.8 psig with a flow of one scfm. Static pressure would rise to 11.2 psig with the throttle valve closed. RV-828 lifted at 14.0 psig and reseated at 12.5 psig.

13. IHX Freeze Trap Helium Supply

Pressure control valves -817, -818 and -819 were adjusted to provide a downstream dynamic pressure of 1.75 psig with the following flows:

<u>Valve No.</u>	<u>Flow</u>
PCV-817	14.0 scfm
PCV-818	14.0 scfm
PCV-819	15.0 scfm

Results of the relief valve tests were as follows:

<u>Valve No.</u>	<u>Lifting Pressure</u>	<u>Reseating Pressure</u>
RV-816	7 psig	6.5 psig
RV-817	7 psig	6.0 psig
RV-818	7 psig	5.5 psig

14. Secondary Sodium Expansion Tank Helium Supplies

PCV's -814, -815 and -816 were adjusted to provide a static pressure (to expansion tanks No. 1, No. 2 and No. 3, respectively) of 20 psig. With required flow, pressure conditions were as follows:

<u>Expansion Tank No.</u>	<u>Valve No.</u>	<u>Flow</u>	<u>Pressure</u>
1	PCV-814	35 scfm	17.5 psig
2	PCV-815	35 scfm	17.5 psig
3	PCV-816	35 scfm	18.5 psig

High pressure alarms PAH-817, -818 and -819 each energized the respective alarm system at 22 psig, and low pressure alarms PAL-817, -818 and -819 each energized the respective alarm system at 15 psig.

Results of the relief valve tests were as follows:

<u>Expansion Tank No.</u>	<u>Valve No.</u>	<u>Lifting Pressure</u>	<u>Reseating Pressure</u>
1	RV-813	23 psig	22 psig
2	RV-814	23 psig	22 psig
3	RV-815	24 psig	21.5 psig

15. Secondary Sodium Fill Tank Helium Supplies

Pressure control valves -802, -803 and -804 were adjusted to provide a static pressure (to fill tanks No. 1, No. 2 and No. 3, respectively) of 20 psig. With required flow, pressure conditions were as follows:

<u>Fill Tank No.</u>	<u>Valve No.</u>	<u>Flow</u>	<u>Pressure</u>
1	PCV-802	35 scfm	17.5 psig
2	PCV-803	35 scfm	16.0 psig
3	PCV-804	35 scfm	16.5 psig

16. Secondary Sodium Service Drain Tank Helium Supply

Pressure control valve -807 was adjusted to provide a downstream pressure of 22 psig with a flow of 10 scfm. RV-806 lifted at 23 psig and reseated at 22 psig.

17. Secondary Sodium Cold Trap Helium Supply

Pressure control valve -805 was adjusted to provide a downstream pressure of 19 psig with a flow of one scfm. RV-804 lifted at 26 psig and reseated at 20.5 psig.

18. Helium Main Header Flow Indicator

With helium supply to all stations shut off, only FI-846 (on secondary sodium service system helium supply header) showed flow. Indicated flow on FI-846 was 0.2 scfm.

B. HELIUM THIRD FLUID SYSTEM

Low pressure alarms -820, -821, -822, -823, -824 and -825 were all adjusted and energized the proper alarm circuit when pressure was decreased to 250 psig. All alarms cleared at approximately 270 psig when pressure was increased.

High pressure alarms -820, -821, -822, -823, -824 and -825, were all adjusted and energized the proper alarm circuit when pressure was increased to 340 psig. All alarms cleared at approximately 320 psig when pressure was allowed to bleed down. RV-842 lifted at 350 psig and reseated at 340 psig.

The eight hour pressure decay test on all six units showed leaks in evaporator No. 1, superheater No. 1 and superheater No. 3 (Figure 1108-7). Leaks found in rupture discs, vent valves and headers were repaired and the test repeated until all units (isolated) would each hold 300 psig for at least eight hours.

The above test results verified the integrity of all six subsystems and system alarm circuits in the helium third fluid system.

Figure 1108-7. Helium Third Fluid System Pressure Decay Test

IV. COMPLIANCE WITH TEST COMPLETION CRITERIA

Following are the test completion criteria associated with preoperational test AI-P-1108, Helium System, as listed in NAA-SR-5700, Supplement 4, pages XI(a)-25 through XI(a)-28, with remarks regarding the test to explain compliance with the criteria.

A. HELIUM SYSTEM

1. Criterion: Helium pressure control valves will control pressure and pass the required flow within the limits prescribed in the test procedure.

Remarks: These conditions have been met in all cases, with the exception of the reactor atmosphere helium supply flow. The test calls for 15 scfm flow. With the addition of a pressure regulator upstream of CV-803, and a drop in pressure to approximately 15 inches of water inlet to CV-803, maximum flow attainable was approximately 10 scfm with CV-803 wide open. This was reviewed by the Test Engineer and Atomics International management and considered adequate under standard plant conditions. Should purging of the reactor be required, the bypass valve (CV-803) control station can be adjusted for the necessary flow, to keep pressure in the reactor constant at any discharge flow attainable through the bypass valve around the reactor exhaust control station (CV-1004) to the radioactive vent system.

2. Criterion: Helium relief valves will relieve overpressure conditions at the pressures specified in the test procedure.

Remarks: All relief valves lift at specified pressure, or within acceptable limits above or below the specified pressure.

3. Criterion: Reactor cavity, reactor atmosphere and primary sodium fill tanks pressure controllers will control pressure by admitting helium or exhausting gases to the R/A Vent System under overpressure conditions within the limits called for in the test procedure.

Remarks: All three dual-pilot controllers function properly to maintain pressure in the reactor, reactor cavity and primary fill tanks within the specified upper and lower setpoints on the controllers.

4. Criterion: Helium pressure alarms will be actuated at the pressures specified.

Remarks: All pressure alarms, low or high, energized the respective alarm circuits at specified pressures and all circuits functioned properly to light the proper annunciator windows and sound the audible alarms.

B. HELIUM THIRD FLUID SYSTEM

1. Criterion: The third fluid chamber of each component is filled with helium to the pressure specified (300 psi).

Remarks: This condition was met.

2. Criterion: The third fluid system high and low pressure alarms are set and function properly.

Remarks: All high pressure alarm circuits are energized when pressure rises to 340 psig. All low pressure alarm circuits are energized when pressure drops to 250 psig. All alarm circuits function properly to actuate audible and visual annunciators.

3. Criterion: System pressure holds constant with supply valves shut-off, indicating no leakage into or out of the system.

Remarks: All component systems were pressurized, isolated and held pressure constant for at least eight hours, without any change in temperature in the system.

V. EVALUATION

Preoperational test AI-P-1108 was performed to verify that the helium system, and related system components, operated satisfactorily and in accordance with design requirements. Specifically, the helium system was tested to determine that the system can supply the flow and pressure required to introduce and maintain a helium atmosphere in the various sodium-containing systems and equipment. Verification of helium system pressure controls, instrumentation, and operability of alarms were thus also a part of the test.

The helium system was designed to provide and maintain an atmosphere for the sodium systems, compatible with sodium and the associated structural materials, and capable of preventing infiltration of oxygen and other contaminants. Preoperational testing of the helium system demonstrated conformance of the system to these design criteria. The test purpose was accomplished, and the test completion criteria as specified in the Final Summary Safeguards Report for the Hallam Nuclear Power Facility, NAA-SR-5700, were met.

The system has been in operation for almost one year and has functioned well. A few component malfunctions were encountered during the system testing and subsequent operation, but these were corrected. In addition, some minor system simplifications and modifications were accomplished to ensure satisfactory future operation.

A. CONTROLS AND INSTRUMENTATION

Each pressure control station was individually leak tested and the related pressure regulating valves and relief valves were adjusted to the design settings. Some leakage of relief valves and regulating valve pressure creep was experienced; in each case the valves were replaced or repaired to function properly. Flow capacity and the ability to maintain the specified pressure were determined and are considered adequate in all cases.

Pressure controls for the reactor, reactor cavity, and the primary fill tanks were functionally checked and the ability to control flows and pressure within the prescribed limits was demonstrated. Some undesirable control characteristics are inherent in the reactor and primary fill tank controls. The controller sensing points are located in the helium supply lines, causing a false increase in indicated pressure when helium is flowing in the lines. This is due to friction losses in the supply line, from the sensing point to the equipment. This condition is not serious enough to necessitate a modification; in future plants, however, an effort should be made to eliminate this problem.

The various pressure alarms actuated at the specified pressures during testing, the associated annunciating devices were also found to operate satisfactorily.

B. DESSICANT DRYERS

The dessicant regenerating system contains a condenser which requires circulating cooling water. Prior to placing the helium system in service, a hazard analysis of the plant pointed up the potential hazard associated with a failure in cooling water piping within the reactor building and auxiliary bay. As a result, all cooling water lines, including the lines to the helium system dessicant dryer, were removed from these areas.

Since it was not possible to regenerate the dryer units, these units were removed from service and bypassed. At present, the helium is being supplied to the system directly from the high pressure supply bottles, with no provisions for moisture removal. A study of this condition has shown that there are no detrimental effects associated with this mode of operation.

Helium specifications, as supplied by the U. S. Bureau of Mines, call for a maximum moisture content of 0.002 percent water by volume. If it is assumed that every helium bottle contains the maximum allowable moisture content, the total water contribution to the system would be less than 0.4 pounds per year at the present use-rate of 1000 scf/day.

Purity of the bulk helium is very rigidly controlled by the U. S. Bureau of Mines. The only means by which the helium can be accidentally contaminated with excessive moisture is from a cylinder which contains moisture prior to filling. Normally, a cylinder containing less than 50 psig is evacuated and purged before being recharged. Cylinders containing more than 50 psig are assumed to be free of moisture, and are filled directly. In such a case there is a remote possibility that the cylinder could be back-filled with moisture-laden gas to a pressure greater than 50 psig, returned to the supplier and then recharged. Assuming that the cylinder was filled to 50 psig with moisture-saturated air at atmospheric conditions, it would contain only about 0.008 pounds of water. Should as many as five percent of the cylinders become contaminated in this manner, only 0.6 pounds of water per year would be added to the sodium systems.

Even in the most extreme case, assuming that the cylinder actually contained a substantial amount of water, the quantity of moisture introduced to the sodium systems would be small. Due to the pressure effect on the water vapor content of the compressed gas, only 0.05 pounds of water would flow from each cylinder. Considering the almost inconceivable condition whereby one percent of the cylinders contained water, the condition would result in a total of only 0.81 pounds of water per year being introduced into the systems.

Under the most pessimistic conditions, it is difficult to imagine that the helium system would contribute more than 2 to 3 pounds of water per year to the sodium systems.

Moisture in the helium system will be absorbed by the sodium to form sodium oxide, which will be removed by the sodium cold traps. One pound of water will form approximately 3.4 pounds of sodium oxide. Since the sodium system cold traps can easily remove 200 pounds of sodium oxide, it is evident that the additional oxide contributed by moisture in the helium is negligible, and that the dessicant dryers are, therefore, not needed.

C. REACTOR LEAK RATE MONITORING SYSTEM

The integrating flow indicators (FI-848 and FI-849) originally installed in the reactor leak rate monitoring system were rated for a maximum working pressure of 14 inches Wg. Friction loss in the reactor helium supply between flow indicator FI-848 and the reactor is sufficiently high that the pressure at the flow meter exceeded this value when helium was admitted to the reactor. On several occasions the pressure was high enough to rupture the case on the flow meter. To correct this condition, a pressure regulating valve was installed upstream of the control valve, and a small surge tank was installed between the control valve and the flow meter. After the above described modifications were incorporated, no problem with over-pressurization occurred. However, the FI-848 failures pointed up a serious problem with the companion flow meter FI-849 (identical in construction to FI-848) which is installed in the reactor vent line. The reactor relief valve is set at 5.0 psig, and the reactor rupture disc has a bursting pressure of 10 psig. Due to the very low pressure rating of FI-849, it would undoubtedly fail before the reactor pressure relief devices function.

Subsequent to the preoperational test, the flow meters were replaced with a high pressure (100 psig rating) type. This modification should completely eliminate any future problems associated with case failures.

A series flow comparison indicates that the new flow instruments agree within approximately 0.2%. However, when actually placed in operation, the indicated helium flow out of the reactor is consistently greater than the supply flow. During a 23 day period of operation, the indicated volume of helium flowing from the reactor was approximately 50% greater than the volume supplied. Additional tests will be performed during the post critical testing program to determine the cause of this anomaly.

D. STEAM GENERATOR THIRD FLUID SYSTEM

The steam generator third fluid system was evacuated to approximately 27 in/Hg vacuum and then pressurized with helium to 300 psig. Initially, the

system pressures decayed rapidly due to leaks in piping, valves, and other external piping components. After these leaks were repaired, the system was determined to be leak tight. An independent checkout of the system, after the test was completed, indicated no pressure decay over a one week period.

All pressure alarms were checked by varying the system pressures. The alarms and associated annunciating devices functioned properly.

The third fluid dessicant dryer unit is not a required piece of equipment (for the reasons described in section B., of this evaluation), but there is no necessity for removing it from the system. In future designs of similar nature, consideration should be given to the elimination of dryer units.